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C.1 AUTHOR LIST

M. Abdou, UCLA
D. Anderson, U. of Wisconsin
H. Baldis, LLNL
R. Bangerter, LBNL
J. Barnard, LLNL
D. Barnes, LANL
I. Bernstein, Yale U.
A. Bhattacharjee, PPPL
E. Bloom, ORNL
S. Bodner, NRL
A. Boozer, Columbia U.
L. Bromberg, MIT
B. Carreras, ORNL
V. Chan, GA
B. Chapman, U. of Wisconsin
R. Cohen, LLNL
S. Cohen, PPPL
G. Collins, LLNL
J. Conrad, U. of Wisconsin
D. Craig, U. of Wisconsin
A. DaSilva, U. of Maryland
L. Da Silva, LLNL
R. Davidson, PPPL
J. Dawson, UCLA
J. Degnan, AFRL
D. Den Hartog, U. of Wisconsin
J. Drake, U. of Maryland
G. Dunn, U. of Colorado
A. Faltens, LBNL
G. Fiksel, U. of Wisconsin
N. Fisch, PPPL
A. Friedman, LLNL
D. Garnier, Columbia U.
K. Gentle, U. of Texas
G. Glendinning, LLNL
R. Goldston, PPPL
D. Goodin, GA
M. Gouge, ORNL
H. Griem, U. of Maryland
R. Hazeltine, U. of Texas
C. Hegna, U. of Wisconsin
D. Hill, LLNL
A. Hoffman, U. of Washington
B. Hooper, LLNL
J. Hosea, PPPL
W. Houlberg, ORNL
I. Hutchinson, MIT
T. James, UCSD
H. Ji, PPPL
J. Johnson, Florida A&M
J. Johnson, PPPL
J. Kesner, MIT
H. Khater, U. of Wisconsin
R. Kirkpatrick, LANL
S. Krasheninnikov, MIT
W. Kruer, LLNL
J. Kwan, LBNL
B. LaBombard, MIT
N. Lanier, U. of Wisconsin
J. Latkpwski, LLNL
E. Lee, LBNL
M. Lieberman, U. of Berkeley
I. Lindemuth, LANL
B. Lipschultz, MIT
G. Logan, LLNL
J. Lyon, ORNL
E. Marmar, MIT
K. Matzen, SNL
M. Mauel, Columbia U.
D. Meade, PPPL
W. Meier, LLNL
M. Menon, ORNL
F. Meyer, ORNL
W. Miller, GA
J. Minnervini, MIT
P. Mioduszewski, ORNL
R. Moir, LLNL
A. Molvik, LLNL
H. Monkhorst, U. of Florida
F. Najmabadi, UCSD
G. Neilson, PPPL
C. Olson, SNL
S. Payne, LLNL
M. Peng, ORNL
F. Perkins, PPPL
J. Perkins, LLNL
J. Peterson, LLNL
P. Peterson, U. of Cal., Berkeley
R. Petrasso, MIT
D. Petti, INEEL
R. Petzoldt, U. of Cal., Berkeley
C. Phillips, PPPL
R. Post, LLNL
H. Powell, LLNL
S. Prager, U. of Wisconsin
D. Rasmussen, ORNL
B. Remington, LLNL
M. Rosen, LLNL
N. Rostoker, U. of Cal., Irvine
P. Rutherford, PPPL
D. Ryutov, LLNL
M. Saltmarsh, ORNL
J. Santarius, U. of Wisconsin
J. Sarff, U. of Wisconsin
K. Schoenberg, LANL
D. Schultz, ORNL
K. Schultz, GA
W. Seka, U. of Rochester
J. Sethian, NRL
T. Shannon, U. of Tennessee
J. Sheffield, ORNL
R. Siemon, LANL
T. Simonen, GA
C. Skinner, PPPL
D. Smith, ANL
W. Stacey, Georgia Tech.
R. Stambaugh, GA
C. Surko, UCSD
I. Sviatoslavsky, U. of Wisconsin
D. Swain, ORNL
M. Tabak, LLNL
J. Talmadge, U. of Wisconsin
W. Tang, PPPL
R. Taylor, UCLA
R. Temkin, MIT
K. Thomassen, LLNL
M. Tillack, UCSD
N. Uckan, ORNL
E. Uchimoto, U. of Montana
M. Ulrickson, SNL
C. Verdon, U. of Rochester
S. Willms, LANL
P. Woskov, MIT
K. Young, PPPL
M. Zarnstorff, PPPL
S. Zweben, PPPL

Table C.1. Topical areas in fusion energy sciences[a,b]

No.	MFE M-1 to M-20	IFE I-1 to I-12	Technologies T-1 to T-20	Plasma Science S-1 to S-17	Near-Term Applications N-1 to N-5
1	Stellarator	National Ignition Facility	Superconductivity	Hamiltonian Dynamics	Semiconductors
2	Compact Stellarator	Indirect-Drive Inertial Fusion Energy	Electromagnetic Heating and Current Drive	Long Mean-Free Path Physics	Advanced Materials Processing and Manufacturing
3	Tokamak	Direct-Drive Inertial Fusion Energy	Neutral Beams	Wave-Particle Interactions	Environment
4	Advanced Tokamak	Fast Ignition Approach to Inertial Fusion Energy	Fueling and Vacuum	Turbulence	Medical Applications
5	Electric Tokamak	Heavy Ion Accelerators for Fusion	Divertor	Hydrodynamics and Turbulence	Plasma Propulsion
6	Spherical Torus	Repetition-Rate Krypton Fluoride Laser	High Heat Flux Components and Plasma Materials Interactions	Dynamo and Relaxation	
7	Reversed-Field-Pinch Concept	Solid-State Laser Drivers	MFE Liquid Walls	Magnetic Reconnection	
8	Spheromak	Laser and Plasma Interactions	Shield/Blanket	Dense Matter Physics	
9	Field-Reversed Configuration	Pulsed Power	Radiation-Resistant Materials Development	Nonneutral Plasmas	
10	Levitated Dipole Fusion Concept	Target Design and Simulations	International Fusion Materials Irradiation Facility	Electrostatic Traps	
11	Open-Ended Magnetic Fusion Systems	Final Optics—Laser IFE	Tritium Systems	Atomic Physics	
12	Gas Dynamic Trap	Laser-Driven Neutron Sources	Remote Maintenance	Opacity in ICE/IFE	
13	Plasmas with Strong External Drive		MFE Safety and Environment	MFE Plasma Diagnostics	
14	Magnetized Target Fusion		IFE Safety and Environment	IFE Diagnostics	
15	Boundary Plasma/ Wall Interactions		IFE Liquid-Wall Chambers	Advanced Computation	
16	Burning Plasma Science		Dry Wall Chambers	Computer Modeling of Plasma Systems	
17	Burning Plasma Experimental Options		IFE Target Fabrication	Astrophysics Using Fusion Facilities	
18	Integrated Fusion Science and Engineering Technology Research		IFE Target Injection and Tracking		
19	Volumetric Neutron Source		IFE Power Plant Technologies		
20	Advanced Fuels		Advanced Design Studies		

[a]Note that the upper seven technologies are for MFE, the lower ones for IFE, and the middle seven and the last one can apply to both.

[b]Note that single-pulse laser driver development for IFE has traditionally been supported primarily by the ICF program within the DP element of DOE. A key issue for IFE is the development of repetitively pulsed drivers.

C.2 MAGNETIC FUSION ENERGY (MFE)

The diagram of a reference tokamak power plant in Fig. C.1 shows the key components of a typical magnetic fusion power plant.

- The characteristics of potential magnetic configurations are given in M-1 through M-14. As discussed in Sect. 2.2 of the main document, these configurations fall into two main categories: externally controlled and self-ordered plasmas.
- Plasma wall interactions and divertors to handle particle and heat removal and impurity control are discussed in M-15 and T-5, respectively.
- Burning plasma physics and opportunities for burning plasma experiments are discussed in M-16 and M-17, respectively. International Thermonuclear Experimental Reactor (ITER) scale facilities are discussed in M-18.

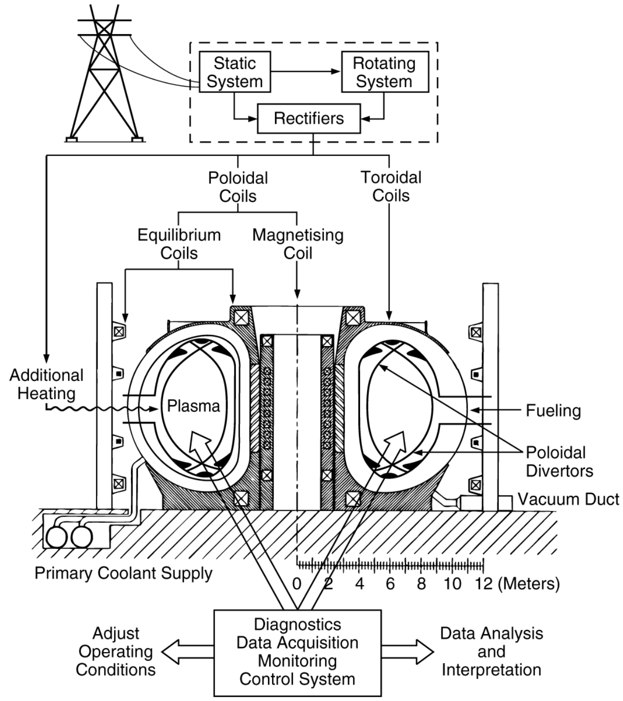


Fig. C.1. Representative tokamak.

The plasma, nuclear, and safety technologies are shown, respectively, in T-1 to T-6, T-7 to T-12, and T-13 and T-14. Power plants are discussed in T-20.

A power plant that burns plasma will need to achieve a value of the product (density) × (ion temperature) × (energy confinement time) of $\langle n_{DT}T_i\rangle\,\tau \approx 2 \times 10^{24}$ ($m^{-3}\cdot eV\cdot s$) for a deuterium-tritium (D-T) plasma, at an ion temperature of ~10 keV, and $\langle n_{DT}T_i\rangle\,\tau > 2 \times 10^{25}$ ($m^{-3}\cdot eV\cdot s$) for a D-D or D-^{3}He plasma of 30 keV. A temperature of 1 eV ≈ 10,000 K.

The energy confinement time (τ) is defined as the energy in the plasma divided by the power required to keep it hot. Another commonly used parameter is beta (β), the ratio of the plasma pressure to the magnetic pressure $\beta \propto nT/B^2$, where B is the magnetic field.

The magnetic field of most of the configurations being studied has two main components: a toroidal field (B_t) (going the long way around the torus) and a poloidal field (B_p) (going the short way around the torus), as shown in Fig. C.2. The safety factor, q, is the amount of twist of the helical magnetic field lines (the number of toroidal field-line transits per poloidal field-line transit), and $q = r\,B_t/R\,B_p$, if $\beta << 1$, $a/R << 1$, where r and R are the minor and major radii of the torus, respectively [the outer plasma minor radius is denoted by (a)]. For the case of stellarators, it is convention to use the parameter iota bar, $\iota/2\pi = 1/q$ to describe the twist in the total field. The parameters (κ) and (δ) are used, respectively, to describe the ellipticity and triangularity of noncircular plasma cross sections.

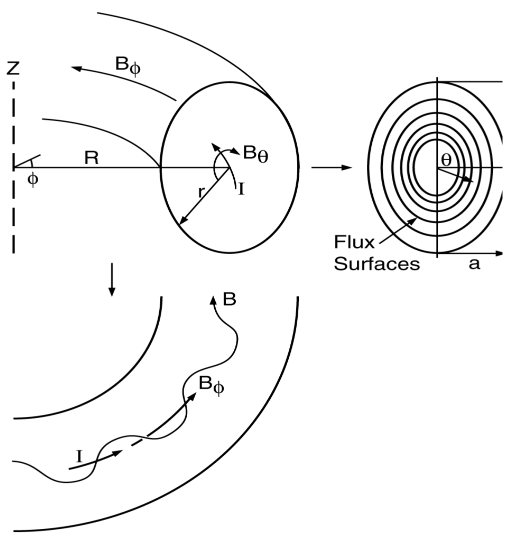


Fig. C.2. Toroidal and poloidal fields.

Table C.2. MFE experiments, operational or construction (major international)

Experiment	R (m)	a (m) [κ]	Pulse length (s)	B_t (T) [B_p (T)]	I (MA)	P_{heat} (MW)	Div-ertor?	$T_i(0)$ (keV) [T_e (0)]	$\langle n \rangle$ 10^{20} m^{-3}	τ (s)	β (%)
External control					**Stellarator(1)**						
HSX (Wisconsin)	1.2	0.15	0.2	1.37	N.A.	0.2	No	Operate 1999			
W7-AS (Germany)	2.0	0.18	3	2.5	N.A.	3	Yes	1.5	3	0.05	2
TJ-II (Spain)	1.5	0.22	0.5	1.2	N.A.	2	No	Start of operation			
LHD (Japan)	3.9	0.6	St. St.	4	N.A.	40	Yes	Start of operation			
CHS (Japan)	1.0	0.2	0.74	2	N.A.	1.7	Yes	1	0.8	0.01	2.1
Heliotron J (Japan)	1.2	0.17	0.2	1.5	N.A.	2	No	Construction			
W7-X (Germany)	5.5	0.52	St. St.	3	N.A.	30	Yes	Construction			
External control					**Advanced tokamak (AT)**						
Alcator C-Mod	0.67	0.22 [1.8]	2 (7)	9	1.5 (2.5)	8	Yes	6 (10)	10	0.09 (0.2)	1.5 (5)
D-IIID	1.67	0.67 [2.5]	10	2.2	3	27	Yes	27	3	0.5	13
HBT-EP	0.92	0.15 [1.0]	0.02	0.4	0.02	0.2	No	0.2	0.2	0.001	1.5–3
ET	5	1 [1.5]	0.1	0.5	0.2	2	No	2	0.1	0.1	3
ASDEX-U (Germany)	1.65	0.5 [1.6]		3	1.6	27	Yes				
FT-U (Italy)	0.93	0.3 [1.0]		8	1.3	7	No				
JET (ECC)	3	1.25 [1.8]	60	40	7	42	Yes	40	0.8	1.8	4
Textor (Germany)	1.75	0.46 [1.0]		3	0.8	8	No				
Tore Supra (France)	2.47	0.8 [1.0]		4.5	1.7	9	No				
JRT2-M (Japan)	1.31	0.35 [1.7]		2.2	0.25		Yes				
JT-60 (Japan)	3.3	0.8 [1.8]	10	4.4	5	55	Yes	45	1	1	1.7
Triam-1M (Japan)	0.8	0.12 [1.5]	7200	8	0.4	0.2	Yes	5	0.1	0.01	
K-Star (Korea)	1.8	0.5 [2]	20–300	3.5	2	15–40	Yes	Design/Construction			
Intermediate					**Spherical torus (ST)**						
HIT-I (Shutdown)	0.3	0.2 [1.85]	0.01	0.5	0.25		Yes		0.6		
START (UK) (Shutdown)	0.32	0.25 [1.8–3]	0.05	0.3	0.31	1	Yes	0.35	1	0.005	40
TS-3 (Japan)	0.25	0.15 [1.5]	0.001	0–0.2	0.1	OH	No	0.1	0.5		5–70
HIT-II	0.3	0.2 [1.85]	0.035	0.5	0.2	OH	No				
HIST (Japan)	0.3	0.24 [2]	0.005	0.2	0.15	OH	No	0.03	0.5		10
CDX-U	0.34	0.22 [1.6]	0.05	0.2	0.2	0.3	Yes	0.1	0.5	0.001	5
TST-M (Japan)	0.35	0.25 [1.5]	0.01	0.3	0.1	OH	No		0.1		
ETE (Brazil)	0.3	0.2 [1.8]	0.02	0.6	0.4	OH	No	Construction			
TST-2 (Japan)	0.37	0.23 [1.5]	0.1	0.4	0.2	OH	No	Construction			
TS-4 (Japan)	0.5	0.4 [1.5]	0.01	0–0.4	0.4	OH	No	Construction			

Table C.2. (continued)

Experiment	R (m)	a (m) [κ]	Pulse length (s)	B_t (T) [B_p (T)]	I (MA)	P_{heat} (MW)	Div-ertor?	$T_i(0)$ (keV) [T_e (0)]	<n> 10^{20} m^{-3}	τ (s)	β (%)
Intermediate	**Spherical torus (ST)**										
Pegasus	0.45	0.4 [3.7]	0.08	0.15	0.4	2	No	First plasma 1999			
Globus-M (Russia)	0.5	0.35 [1.6]	0.2	0.6	0.4	1	No	First plasma 1999			
MAST (UK)	0.75	0.55 [2.3]	5	0.5	2	5	Yes	Commissioning, first plasma 1999			
NSTX	0.86	0.68 [2]	5	0.3	1	11	Yes	First plasma 1999			
Intermediate	**Reversed-field pinch (RFP)**										
MST	1.5	0.50	0.08	0.1	0.5	OH + (2 RF)	No	0.2 (0.8)	0.3	0.006	10
RFX (Italy)	2.0	0.46	0.25	0.4	1 (2)	OH	No	0.3	0.5	0.002	10
TPERX (Japan)	1.7	0.45	0.10	0.2	1	OH	No	Commissioning in 1999			
TPE-RX											
Self-ordered	**Field-reversed configuration (FRC)**										
U. Washington											
Magnetic dipole											
LDX (MIT)	1	1	>10	[0.25]	0		Yes	Construction			

Notes: (1) For stellarators, a is average plasma radius <a>. Plasma parameters are maximum achieved values, not simultaneous sets of parameters. (2) Numbers in parentheses are projected for Alcator C-Mod.

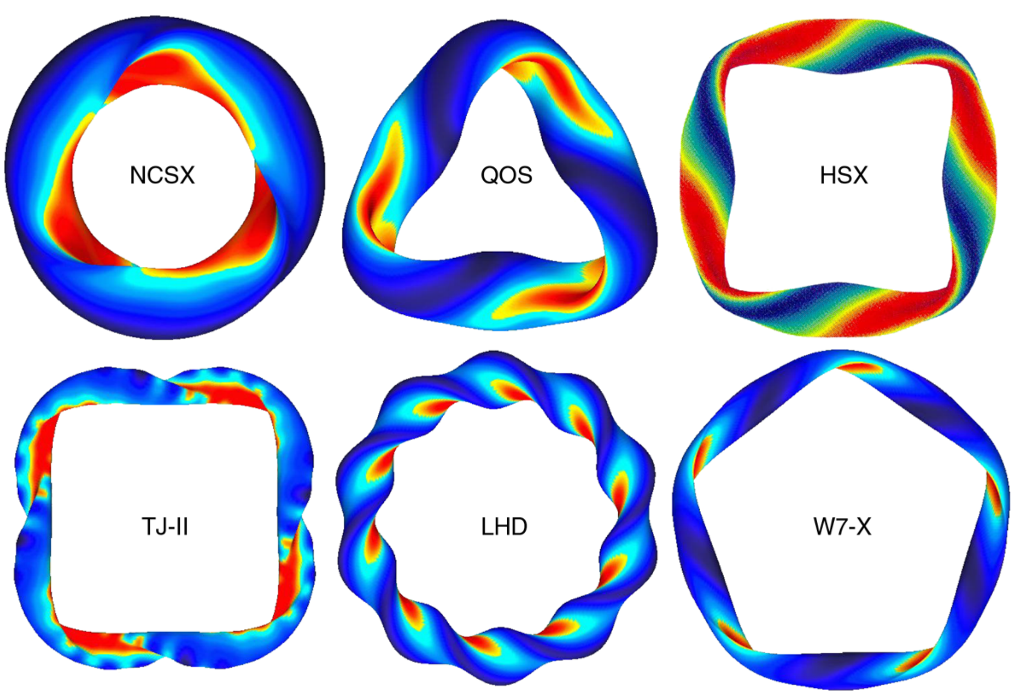
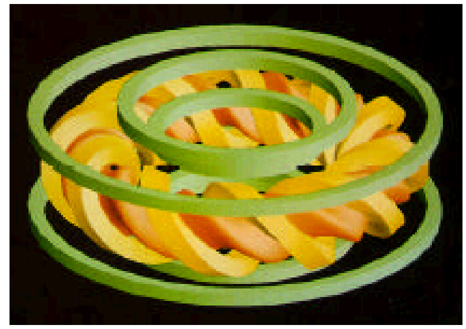


Fig. C.3. 3-D shaping of stellarator magnetic surfaces that provides flexibility to address a wide range of toroidal physics issues.

Description

Stellarators are toroidal confinement devices with helical magnetic field lines similar to those in a tokamak, but the confining poloidal magnetic field is created by currents in non-axisymmetric coils outside the plasma, rather than by a toroidal current within the plasma. The three-dimensional (3-D) plasma shaping in stellarators allows inherent steady-state operation with low recirculating power and avoids damaging disruptions. The flexibility of the externally generated poloidal field allows exploration of a wide range of magnetic configurations with different degrees of helical excursions of the toroidal magnetic axis, rotation and shear in the field lines, magnetic well depth, and plasma cross section shaping and aspect ratio, the essential building blocks of toroidal confinement systems. In the large stellarator program outside the United States, three lines of currentless stellarators with large-to-medium aspect ratios are being pursued. Stellarators are second only to the related tokamaks in investment, performance, and degree of physics understanding.



The LHD plasma and coil configuration.

Status

The technical basis for the design, fabrication, and projected performance of stellarators is well advanced. A confinement scaling (ISS95) based on data sets from all the world's stellarators also fits the tokamak L-mode database.

- Experiments. The main existing experiments are the $1B-class superconducting-coil Large Helical Device (LHD) with R = 3.9 m, $\langle a \rangle$ = 0.65 m, $B \le 4$ T, and $P \le 40$ MW in Japan; the Wendelstein 7-X (W7-X) with R = 5.5 m, $\langle a \rangle$ = 0.52 m, B = 3 T, P = 30 MW under construction in Germany; the Japanese Compact Helical System (CHS) with R = 1 m, $\langle a \rangle$ = 0.2 m, $B \le 2$ T, and $P \le 2$ MW; the German W7-AS with R = 2 m, $\langle a \rangle$ = 0.18 m, $B \le 2.5$ T, and $P \le 3$ MW; and the smaller helical-axis stellarators (heliacs) in Australia (H-1), Spain (TJ-II), and Japan (Heliotron J).
- Theory and computational tools. The 3-D codes for calculations of magnetohydrodynamic (MHD) equilibrium and stability, magnetic configuration optimization, coil optimization, and divertor topology are well developed. The 3-D neoclassical transport, including the bootstrap current, energetic orbit confinement, and ambipolar electric fields, is well understood. Stellarators are now designed to meet a set of physics criteria by optimizing the shape of the plasma boundary, given the plasma pressure and current profiles, and a set of coils can then be generated to produce the desired boundary shape.
- Engineering capabilities. Computer-aided design and fabrication of complex vacuum vessels and coils is now routine as demonstrated by the successful fabrication of W7-AS, TJ-II, LHD, and the W7-X test coil. Manufacturing and assembly accuracies of <1 part in 1000 are routinely obtained. Large superconducting coils are in use on LHD and are under construction for W7-X.

Current Research and Development (R&D)

R&D Goals and Challenges

The key issues are (1) demonstrating reduced neoclassical transport, (2) further improvement in confinement over the ISS95 scaling, (3) understanding what limits the beta in a stellarator and obtaining $\langle\beta\rangle > 5\%$, (4) obtaining parameters (T_e, T_i, $\langle\beta\rangle$, τ_E) comparable to those in the mainline tokamak, and (5) developing practical particle and power handling techniques. The non-U.S. stellarator program is addressing these issues in large-aspect-ratio systems with very low bootstrap currents. Theoretical studies have recently identified promising concepts for "compact stellarators" that have relatively low aspect ratios and use bootstrap currents to advantage, but which are not currently being studied experimentally. A U.S. proof-of-principle (PoP) program is proposed to attack the stellarator R&D issues using these new configurations as a complementary part of the world stellarator program.

Related R&D Activities

Stellarators share many physics features with tokamaks, so many of the theory and modeling tools, plasma heating systems, and results from reactor studies are useful for both. The inherently 3-D nature of stellarators allows fundamental studies relevant to a variety of 3-D plasma applications: the magnetosphere, free electron lasers, accelerator transport lattices, and nonaxisymmetric perturbations to tokamaks and other toroidal confinement systems.

Recent Successes

- W7-AS: τ_E up to 2.5 times ISS95 scaling, demonstration of access to the electron root of the electric field ambipolarity condition, ion cyclotron range of frequency (ICRF) heating and plasma sustainment with no increase in density or impurities, and significant plasma parameters of T_e = 5.7 keV, T_i = 1.5 keV, $n_e = 3 \times 10^{20}$ m^{-3}, $\langle\beta\rangle$ = 1.8%, and τ_E = 55 ms (not simultaneously).
- LHD: τ_E up to 0.17 s, 1.5 times ISS95 scaling, and quasi-steady-state operation up to 22 s with neutral beam injection (NBI) in initial experiments.
- CHS: $\langle\beta\rangle$ = 2.1% and demonstrated large parallel viscous damping in a nonsymmetric configuration, radial electric field control, ICRF heating of the bulk plasma, and demonstration of enhanced pumping with a local magnetic island divertor.

Budget

The world stellarator program is funded at $100M–200M/year. DOE–OFES funded activities (≈$0.8M/year) on conventional stellarators principally involve collaborations on LHD, CHS, and W7-AS.

Anticipated Contributions Relative to Metrics

Metrics

- The W7-X-based Helias Stellarator Reactor (HSR, with R = 22–24 m and B = 5 T) study indicates that a stellarator reactor can be built with present superconductors (NbTi) and present physics understanding. The key measures of necessary performance follow:
  - — Neoclassical transport much less than ISS95 scaling and losses of energetic particles $\lesssim 5\%$.
  - — Thermal plasma confinement better than two times ISS95 scaling.
  - — Plasma parameters competitive with tokamaks ($T_i > 10$ keV, $\langle\beta\rangle > 5\%$, $\tau_E > 0.3$ s, and $n\tau_E T > 10^{20}$ keV·s·m^{-3}).
  - — Bootstrap current <10% of that in a comparable tokamak at high β and low collisionality.
  - — True steady-state operation (~1 h) without current drive and without disruptions at $\langle\beta\rangle > 5\%$.
  - — Superconducting coils with B = 5 T and fabrication and assembly accuracies <1 part in 1000.
  - — Practical power and particle handling schemes that are extrapolatable to a reactor-relevant configuration.

Near Term ≤5 years

- LHD will extend stellarator plasma parameters to more relevant regimes ($T_i \sim 10$ keV, $\langle\beta\rangle \geq 5\%$, $\tau_E > 0.2$ s, $n\tau_E T > 10^{20}$ keV·s·m^{-3} with $P \sim 40$ MW) and study improved confinement modes, steady-state (30-min) operation with $P \sim 3$ MW, simulated alpha-particle confinement, and particle control with a local island divertor.
- W7-AS will operate through 2001 and test a W7-X-relevant magnetic island divertor, operate at full power for 3-s pulses, use particle control (with the divertor and pellet injection) to test the effect on H-mode and confinement improvement and verify the density scaling of confinement, and study control of the electric field with perpendicular neutral beam injection.
- Helical-axis stellarators in Spain, Australia, and Japan will test the advantages of higher rotational transform, larger helical axis excursions, and beam cross-sectional shaping of the plasma with multimegawatt plasma heating for a range of heliac configurations.
- Reactor studies in Germany and Japan will assess the reactor potential of the W7-X and LHD approaches.

Midterm ~20 years

- The superconducting-coil LHD will extend operation to B = 4 T and operate with a full helical divertor in Phase II of the LHD program starting in 2001.
- The superconducting modular coil stellarator W7-X will begin operation in 2005–2006 and demonstrate low neoclassical transport, low equilibrium and bootstrap currents, a practical magnetic-island-based divertor, and more reactor-relevant plasma parameters: $T > 10$ keV, $\langle\beta\rangle \geq 5\%$, and $n\tau_E T > 10^{20}$ keV·s·m^{-3}.
- If results from the world stellarator programs meet expectations, a deuterium-tritium (D-T) stellarator in which a large part of the power is produced by fusion reactions would be constructed at the end of this period to study the key reactor issues of thermal confinement, energetic and alpha-particle losses, MHD stability and beta limits, and steady-state particle and power handling where a significant fraction of the heating power is from fusion-generated alpha particles.

Long Term >20 years

Depending on results from D-T stellarator operation and the status of the world fusion program, a superconducting-coil stellarator in the demonstration reactor (DEMO) class would be constructed. If successful, this would open up an inherently steady-state disruption-free route to a fusion power plant with good confinement and beta and low recycled power.

Proponents' and Critics' Claims

Proponents claim that (1) stellarators can have good performance (confinement, beta) with no disruptions in true steady-state operation without the need for current drive or a potentially unstable bootstrap-current-dominated equilibrium; (2) decades of experience demonstrate that stellarators can be built with the desired accuracy; (3) the key issues for the viability of the stellarator approach will be addressed in the large superconducting-coil stellarators LHD and W7-X; (4) the wide range of stellarator configurations extends our scientific understanding of toroidal confinement; and (5) stellarators could lead to a reactor that is more reliable than an advanced tokamak reactor and would have low recycled power.

Critics claim that (1) conventional stellarators lead to very large reactors; (2) helical coils are risky because they cannot be tested before the whole device is completed; (3) nonplanar coils are difficult to manufacture, are costly, and lead to too high a ratio of field on the coils to that in the plasma; and (4) stellarators have not yet demonstrated the improved confinement regimes and the particle and power handling of tokamaks.

M-2. COMPACT STELLARATOR

Description

Compact stellarators, with one-half to one-third plasma aspect ratios of conventional stellarators, are nonaxisymmetric toroidal confinement devices that have helical magnetic field lines similar to those in tokamaks and conventional stellarators, but the confining poloidal magnetic field is created by both the plasma-generated internal "bootstrap" current and currents in external coils. This additional flexibility allows exploration of magnetic configurations that could combine the low aspect ratio and good performance of advanced tokamaks (ATs) with the disruption immunity and low recycled power of stellarators. Two new approaches are proposed: quasi-axisymmetry (QA), which uses the bootstrap current to produce about half of the confining poloidal field and has tokamaklike symmetry properties, and quasi-omnigeneity (QO), which approximately aligns bounce-averaged drift orbits with magnetic surfaces and aims at a smaller bootstrap current. The edge magnetic shear can be opposite to that of the AT, stabilizing neoclassical magnetic islands and permitting higher external kink stability limits without a nearby conducting wall. Complementing this is the quasi-helically symmetric (QH) approach, which produces configurations with high effective rotational transform, small deviations from a magnetic surface, and little bootstrap current. The main element of the proposed U.S. compact stellarator proof-of-principle (PoP) program is the QA National Compact Stellarator Experiment (NCSX).

Status

The technical basis for design, fabrication, and projected performance of compact stellarators is well advanced. The stellarator confinement scaling ISS95 also fits the tokamak L-mode data base. Experiments on low-beta high-aspect-ratio stellarators with plasma current showed that disruptions were suppressed when the fraction of the rotational transform generated externally exceeded 20%. Control (and even reversal) of the bootstrap current and its agreement with theory have been demonstrated.

- The Helically Symmetric Experiment (HSX, with R = 1.2 m, $\langle a \rangle$ = 0.15 m, $B \leq 1.3$ T, and P = 0.2 MW) at the University of Wisconsin will begin operation in May 1999. It will be the world's first quasi-symmetric (QH) stellarator.
- The Compact Auburn Torsatron (CAT, with R = 0.5 m, and $\langle a \rangle$ = 0.1 m) at Auburn University studies field errors, plasma flow, and ion cyclotron range of frequency (ICRF) heating. It is being upgraded to B = 0.5 T, P = 0.2 MW, and 25-kA ohmic current in order to study kink stability.
- Theory and computational tools. The shape of the last closed flux surface determines the magnetic configuration properties. It is used to design the coils that create optimized configurations. Three-dimensional (3-D) codes for calculations of magnetohydrodynamic (MHD) equilibrium and stability, configuration and coil optimization, and divertor geometry are well developed. Neoclassical transport, the bootstrap current, energetic orbit confinement, and ambipolar electric fields are well understood.
- Engineering capabilities. Computer-aided design and fabrication of accurate, complex vacuum vessels and coils are now routine.
- Concept design. The NCSX PoP facility is being designed based on a QA plasma configuration with an outer boundary shaped to satisfy physics goals: stability to ballooning and kink modes at $\langle \beta \rangle$ = 4% without a conducting wall, >50% of the poloidal field from external coils, and profiles consistent with the bootstrap current. Scoping studies for a QO concept exploration experiment with higher rotational transform are focusing on optimizing energetic particle confinement and β.

Current Research and Development (R&D)

R&D Goals and Challenges

The key issues for compact stellarator configurations are (1) demonstrating improved neoclassical transport, (2) improving confinement over the ISS95 scaling, (3) understanding what determines the limiting beta and obtaining $\langle \beta \rangle \geq 5\%$, (4) demonstrating disruption-free operation at high beta, and (5) developing practical particle and power handling approaches.

Related R&D Activities

Compact stellarators combine stellarator and AT features in the same device and thus share many physics features with stellarators and tokamaks. Many of the physics results, theory and modeling tools, plasma heating systems, and reactor studies are useful to all three approaches. The U.S. compact stellarator program complements the large world stellarator program in which large-to-medium aspect ratio and low bootstrap current are emphasized. The inherently 3-D nature of compact stellarators allows fundamental studies relevant to a variety of 3-D plasma applications.

Recent Successes

- Configuration optimization codes now generate plasma configurations with specified physics properties such as drift surface alignment with magnetic surfaces, magnetic symmetry, plasma current profile, total current, rotational transform profile, magnetic well, plasma aspect ratio, magnetic field ripple, outer surface curvature, and ballooning and kink stability limits.
- Coil optimization codes now include desired engineering properties such as coil type (nonplanar, saddle, or helical); distance between the last closed flux surface and the coil winding surface; and degree of harmonic content in the coils.
- QA and QO configurations have been found that should be free of disruptions with good neoclassical confinement, attractive beta limits, and practical modular coils that can take advantage of existing toroidal facilities to minimize cost.

Budget

DOE–OFES: FY 1998 = $5.9M; and FY 1999 = $8.3M. The total PoP program budget [covering HSX, CAT, theory, international collaboration, and the proposed NCSX and quasi-omnigeneity symmetry (QOS) experiments] for FY 2000 is $15.7M, increasing to $30M/year in later years. The largest part ($20M/year) of this is for the NCSX program. The proposed Compact Stellarator PoP program was reviewed favorably by a DOE review panel for PoP programs, and FESAC recommended funding to maintain the program momentum.

Anticipated Contributions Relative to Metrics

Metrics

- The 1994 Stellarator Power Plant Study indicated that a modular stellarator with R = 14 m and B = 5 T would be competitive with the second-stability Advanced Reactor Innovation and Evaluation Studies (ARIES)-IV tokamak reactor for the same costing and materials assumptions if $\langle\beta\rangle$ = 5% and $\tau_E \geq 2$ times τ_E (ISS95). Compact stellarators offer the possibility of reducing R by a factor ~2 and higher (more economical) wall loading. Measures of the required performance follow:
  - Neoclassical transport much less than ISS95 scaling and losses of energetic particles $\lesssim$ 10%.
  - Thermal plasma confinement better than two times the ISS95 scaling.
  - Plasma parameters competitive with tokamaks (T_i > 10 keV, $\langle\beta\rangle$ > 5%, τ_E > 0.3 s, and $n\tau_E T > 10^{20}$ keV·s·m^{-3}).
  - Compatibility of the bootstrap current (and its control) with operation at high β and low collisionality.
  - Immunity to disruptions with a large bootstrap current contribution to the rotational transform in true steady-state operation.
  - Superconducting coils with B = 5 T and fabrication and assembly accuracies <1 part in 1000.
  - Practical steady-state power and particle handling schemes that can be extrapolated to a reactor-relevant configuration.
  - Reactor designs with good plasma-coil spacing $\Delta(R/\Delta < 4)$ and coil utilization $(B_{max}/B_0) < 3$.

Near Term ≤5 years

- A coordinated U.S. PoP program is proposed to attack key issues in combination with the world program.
- Two complementary compact stellarators will be built and start operation in 2003–2005: NCSX, a QA PoP experiment, and QOS, a concept-exploration-level experiment—the new elements in the U.S. compact stellarator PoP program.
- HSX will explore the QH approach with coils that allow the degree of symmetry, neoclassical transport, magnetic well depth, stability, rotational transform, and parallel viscosity to be varied. HSX will (1) demonstrate the reduction in particle drifts from flux surface due to large effective rotational transform; (2) test reduction of neoclassical electron thermal conductivity and direct orbit losses; (3) demonstrate whether reduction of parallel viscosity in symmetry direction decreases the momentum damping rate; and (4) explore whether large E × B shear can be obtained due to quasi-symmetry or the ambipolarity constraint.
- CAT-upgrade will investigate the disruptivity of current-carrying helical plasmas over a wide range of rotational transform profiles and test different ICRF heating scenarios for application to other stellarators.
- Scoping and ARIES studies will assess the reactor potential of compact stellarators and help define the critical issues.
- The Large Helical Device (LHD) will study improved confinement modes, steady-state operation, and particle control with a local island divertor.
- The German Wendelstein 7-AS (W7-AS) will test a compact-stellarator-relevant island divertor, study H-mode and confinement improvement, study operation with a net plasma current and control of the electric field with perpendicular neutral beam injection.

Midterm ~20 years

- The NCSX PoP facility (R = 1.5 m, $\langle a\rangle$ = 0.45 m, B = 1–2 T, and P = 6–12 MW) will explore the QA optimization approach and address key compact stellarator issues: (1) operation at high β with bootstrap currents and external transform without disruptions; (2) understanding of β limits and the limiting mechanisms; (3) reduction of neoclassical transport to a low level by proper configuration design; (4) control of turbulent transport (e.g., by flow shear), leading to enhanced global confinement; and (5) suppression of neoclassical islands and tearing modes by bootstrap current and stellarator magnetic shear.
- The QOS concept-exploration device ($R \leq 1$ m, $\langle a\rangle$ < 0.28 m, B = 1–2 T, and $P \leq 4$ MW) would test reduction of (1) neoclassical transport via nonsymmetric QO and the effect of electric fields on confinement; (2) energetic orbit losses in nonsymmetric low-aspect-ratio stellarators; (3) the bootstrap current, its control, and the configuration dependence on β; and (4) anomalous transport by methods such as sheared E ×B flow, and understand flow damping in nonsymmetric configurations.
- W7-X will extend our understanding of reduction of neoclassical and anomalous transport, reduction of equilibrium and bootstrap currents, scaling of beta limits, and optimization of island-based divertors for steady-state particle and power handling.

Long Term >20 years

- If results from the total U.S. stellarator PoP, LHD, and W7-X programs meet expectations, a proof-of-performance superconducting-coil and/or D-T Compact Stellarator could be used to study key issues of confinement, MHD stability, particle and power handling, and possibly D-T physics at T > 10 keV, $\langle\beta\rangle$ > 5%, and $n\tau_E T > 10^{20}$ keV·s·m^{-3} with P > 30 MW and B > 3 T.
- If results from the above experiments prove favorable, then the next step would be a Compact Stellarator Experimental Test Reactor.

Proponents' and Critics' Claims

Proponents claim that compact stellarators could combine the low aspect ratio and good performance of tokamaks (confinement, beta) with the disruption immunity and low recycled power of stellarators. Compact stellarator configurations, stellarator-tokamak hybrids, that use the bootstrap current and quasi-symmetry or QO, would extend our scientific understanding of toroidal confinement and could lead to a reactor that is economically competitive with, but more reliable than, the AT.

Critics claim that (1) nonplanar stellarator coils are difficult to manufacture, are costly, and lead to a higher ratio of field on the coils to that in the plasma, so the beta limit needs to be higher to be more attractive than in tokamaks; and (2) stellarators have not yet demonstrated the improved confinement regimes and the particle and power handling of tokamaks.

M-3. TOKAMAK

Description

The tokamak is an axisymmetric toroidal magnetic configuration that combines a strong toroidal magnetic field with toroidal plasma current to produce a helical field, having a safety factor >1 at the edge of the plasma. Within this definition, there is considerable freedom in further specifying the configuration, its properties, and ultimately, its potential as a fusion power system. Because of its demonstrated superiority in confining plasmas at high pressure and temperature, the tokamak has played the dominant role in the remarkable progress over the last three decades in the quest to understand and harness fusion energy.

Status

Recent progress in worldwide tokamak fusion research includes groundbreaking physics experiments with deuterium-tritium (D-T) plasmas, which have produced substantial fusion power, to study confinement of energetic fusion alpha particles and energy transfer from them to the thermal plasma; developing fundamental understanding and demonstrating the suppression of turbulence-driven energy transport; producing plasmas with significant self-generated bootstrap plasma currents, a key step on the route to energy efficient steady-state operation; and demonstrating practical methods to disperse plasma power exhaust and control particle exhaust, a synergy between plasma and atomic physics. These advances have validated tokamak power plant concepts, generated a new research strategy (the advanced tokamak program), stimulated further innovation, and broadened the fusion science base.

The tokamak is ready to progress to the next crucial step in the development of fusion, the study of the nuclear self-heating regime ($Q > 10$). Tokamaks capable of achieving this goal have been designed and proposed. They span a range of sizes and magnetic fields, from the compact high-field copper coil of the Compact Ignition Tokamak (CIT), Ignitor, and Burning Plasma Experimental Reactor (BPX) class, up to the full-size International Thermonuclear Experimental Reactor (ITER) as embodied in the engineering design activity (EDA), which seeks to combine the goal of studying ignited plasmas with a comprehensive technology development mission. Recent international activity is focusing on a reduced-scale (and cost) superconducting reactor, which might incorporate some of the advanced features being developed in the research and development (R&D) programs; domestically, refinements of the high-field burning plasma tokamak options are being pursued.

Power plant system studies show the possibility of a tokamak power plant with competitive cost of electricity, steady-state operation, maintainability, low-level waste, and public and worker safety. Examples include the American Advanced Reactor Innovation and Evaluation Studies (ARIES)-RS and Japanese SSTR studies.

Current Research and Development

Crucial tokamak issues are pursued in a vigorous, worldwide tokamak program of experiments, fundamental theory, and theory-based simulations. Cross-field plasma transport in toroidal devices has been dominated by the almost universal presence of turbulence; indeed, the electromagnetic interaction has made the plasma medium ideal for studies of turbulence, allowing detailed diagnosis and even control. The discovery of high confinement operating regimes in the tokamak has revolutionized expectations for plasma performance. Transport barriers (influenced by magnetic shear and momentum input), regions of reduced low turbulence, and therefore slow diffusion, have been discovered, with H-Mode edge and pellet-enhanced performance (PEP) mode and supershot core barriers being long-standing examples. A unifying interpretation, based on the paradigm that sheared plasma flow decorrelates and stabilizes the important turbulent modes, has shown a remarkable and robust ability to explain the observations. Many aspects of tokamak transport physics are generic to all magnetic confinement; tokamaks have the simplicity of configuration, long pulse length, low collisionality plasmas, and the detailed diagnostics needed to study them and obtain improved confinement for fusion.

Maximizing the fusion power density by increasing the plasma pressure is crucial for a reactor. A major program is devoted to understanding beta limits and increasing plasma stability, through current, pressure, and rotation profile control. Many aspects of tokamak magnetohydrodynamic (MHD) stability are well understood theoretically, and a good capability to predict global limits in standard confinement regimes has been developed. A critical area of intensive research is that of the sudden disruption of plasma current due to MHD instabilities. There are engineering consequences of disruptions for the design of burning plasma tokamak experiments and ultimately power plants. Experiments on mitigation and avoidance, coupled with the development of a fundamental understanding of the physics mechanisms, have significant priority. The important effects of nonideal MHD behavior on stability is another area of active research.

To minimize recirculating power in a reactor, maximum use must be made of the bootstrap-driven current, a major thrust of the advanced tokamak program. To increase stability, localized plasma current can be driven by ion cyclotron range of frequencies (ICRF), lower hybrid, electron cyclotron, and mode conversion techniques. Wave–plasma interactions may also prove to be key in controlling transport in a burning plasma. The radio frequency (rf) approaches have been proposed to control plasma rotation and thus transport barrier evolution.

The temperature and density just inside the edge transport barrier are governed by plasma stability and transport and have a profound influence over core confinement and therefore over accessibility of self-heating and ignition. Determining the dependence of these pedestal characteristics on other plasma parameters will allow more reliable understanding of fusion ignition requirements, more detailed comparisons to stability predictions and better general understanding of high-confinement physics. In most current tokamaks, heat and particle outflux are handled with a poloidal divertor configuration, in which the plasma edge is diverted magnetically into a chamber for helium ash removal and control of plasma-wall interactions by atomic processes. Divertor experiments need to establish a practical and preferably fundamental basis for predicting cross-field energy and particle transport in the scrape-off layer.

Budget

For FY 1999, the Office of Fusion Energy Sciences (OFES) support in the tokamak science area, including international collaborations and Small Business Innovation Research (SBIR)/STTR grants, is \$49.1M; in addition, \$38.9M supports major facility operations.

Anticipated Contributions Relative to Metrics

Metrics

Parameters	**nτ**	**nτT**	**<pressure>**	**Q**	**Pulse length**	**P_{fusion}**	**P_{in}/S_{plasma}**
Achieved	1.4×10^{20} s/m^3	1.5×10^{21} s-keV/m^3	1.6×10^5 Pa	~1	10^2 s	2×10^7 W	0.5 MW/m^2
Reactor	2×10^{20} s/m^3	2×10^{21} s-keV/m^3	5×10^5 Pa	>20	>10^4 s	10^9 W	~1 MW/m^2

Note: not all parameters achieved simultaneously.

Near Term ~5 years

The world program is technically ready to move to the next step in fusion development, the construction of a tokamak to produce and study burning plasma with dominant nuclear self-heating. This step can start within the next 5 years. Europe, Japan, and the Russian Federation are continuing the ITER EDA, now focused on a reduced-cost, reduced technical objectives version of ITER. When and if a firm commitment is made by the ITER parties for construction and a site is selected, the United States could request the opportunity to rejoin as a research partner. If, on the other hand, ITER does not go forward, other approaches to addressing burning plasma physics issues, most likely on an international basis, could be pursued. In parallel, the tokamak program is moving aggressively to develop solutions to the challenges that must be met for subsequent steps, including disruption control, long-pulse/steady-state operation, current, particle and momentum profile control, active control of MHD instabilities, and advanced divertor dissipation of ultra-high heat flux.

Midterm ~20 years

The construction of ITER would occur over the next 10 years, and its operation period could be 10 to 20 years. If ITER is not constructed, then burning plasma science could be studied on a next-generation tokamak, and an advanced integrated experimental reactor could be designed and construction begun by about 2015. Assuming continued success of the tokamak, which is likely with advanced control features and enhanced performance, this device will incorporate reactor prototypical features (superconducting magnets, including the possibility of high-field/high-temperature technology, steady-state operation), which will be close to a demonstration reactor. Even if some alternate to the tokamak is chosen for this device, the knowledge gained from the burning plasma tokamak experiment is a required prerequisite for moving forward in the development of magnetic fusion energy on this time scale.

Long Term >20 years

We can anticipate that the operation of the Advanced Integrated Experimental Reactor could commence as early as 2025, to be followed by the demonstration of a commercial reactor prototype by the middle of the century.

Proponents' and Critics' Claims

Proponents think that the world MFE program is ready to move forward to a burning plasma experiment; the tokamak is the vehicle with which to do so. Compared to its nearest fusion competitor (the stellarator), the tokamak has demonstrated 100 times larger ntE Ti, 30 times higher plasma pressure, >40 times higher first-wall neutron load, and >10 times higher duty cycle. With advanced confinement, current drive and MHD features, the tokamak has the promise to be an economically viable, environmentally attractive fusion power producer.

Critics note that major disruptions are unacceptable for a power plant. Low recirculating power solutions must be demonstrated for steady-state current maintenance.

M-4. ADVANCED TOKAMAK

Description

Advanced tokamak (AT) research seeks to determine the ultimate potential of the tokamak as a magnetic confinement system by pursuing innovation and optimization and by raising the upper limits to performance. The technical gains have the potential to reduce the cost of fusion development steps and power plants. The physics base built using the tokamak is sufficiently well developed to define theoretically the expected ultimate potential of the tokamak.

- Stability. Theory experiments have established a beta limit given by β_N (= β_T/(I/aB) ~ 3 for free boundary equilibria with standard tokamak profiles and no wall stabilization. With optimized current and pressure profiles and assuming wall stabilization of low-order kink modes, stability calculations show a capability of $\beta_N < 6$ in the conventional aspect ratio range.
- Confinement. The theoretical minimum in cross-field transport is the rate set by charged particle collisions, neoclassical transport. Turbulence in the plasma raises the transport above the neoclassical level. It now appears possible to attain the theoretical minimum transport level in the ion transport by suppression of turbulence by sheared $E \times B$ flows in the plasma.
- Steady state. The theoretically predicted and experimentally confirmed bootstrap effect is the basis for steady-state operation of a tokamak. In principle, all the plasma current can be self-generated by the bootstrap effect. Methods of noninductive current drive are well developed. With research on noninductive startup, the AT thrust seeks to define fully transformerless operation.
- Power and particle control. The high beta states sought in the AT thrust lead to high power density systems that call for advanced methods to exhaust power, fuel, and ash without impairing the advanced core plasma performance. The leading candidate approach is the divertor in which the edge plasma is guided into a separate (divertor) chamber for power and particle exhaust.

Status

The advanced state of scientific maturity of plasma measurements and available theory for comparison on tokamaks has enabled the AT thrust. Current effort is on improved plasma control methods and hardware.

- Stability. Target stability levels have been produced transiently. Wall stabilization has a physics basis in the reversed-field-pinch (RFP) and spheromak; preliminary investigations in the tokamak have begun.
- Confinement. The discovery and physics principle of turbulence suppression by sheared $E \times B$ flow (generating a transport barrier) has been shown to be active in a variety of magnetic confinement devices: tokamaks, stellarators, mirrors, and perhaps even the RFP. Plasmas have been made with a transport barrier across most of the discharge, resulting in neoclassical ion transport rates and almost no detected turbulence in the ion gyroradius wavelength range across the plasma. Transport barriers in the electron channel have been seen, but reductions in transport have been less than for the ions. Improved confinement, gauged by a multiplier over nominal confinement, has been produced transiently.
- Steady state. Current and pressure profiles consistent with such operation and stability and transport barriers have been calculated. Methods of noninductive current drive are well developed. Bootstrap fractions up to 80% have been measured.
- Power and particle control. From experimental work and modeling, it appears possible to push toward states in which the plasma nearly completely recombines before the hot plasma can touch the material wall.

Current Research and Development (R&D)

R&D Goals and Challenges

- Stability. Two paths appear possible to realize the optimized current and pressure profiles and wall stabilization. Current profiles that have a magnetic shear reversal [negative central shear (NCS)] put a current peak closer to the wall for wall stabilization. The high internal inductance (l_i) path seeks to move the current as far away from the edge as possible to allow operation at perhaps β_N ~ 4 without wall stabilization. Wall stabilization is sought by maintaining plasma rotation and by feedback control of low-order kink modes. Neoclassical tearing modes must be dealt with in positive shear regions.
- Confinement. R&D seeks to expand the radius of transport barriers to raise the overall confinement and to sustain them in steady state. Progress in the electron transport channel is sought.
- Steady state. Off-axis current drive is needed for the NCS scenario. Sawtooth stabilization or axial safety factor control is needed for the high l_i scenario.
- Power and particle control. Further work is needed on recombining plasmas, the use of impurity ions on plasma edge to radiate power, and the use of the parallel flow of impurities to store ions in the divertor region and radiate power.

Related R&D Activities

AT issues are being pursued in all the major tokamaks in the world. Worldwide, the DIII-D tokamak pioneered and defined the AT research thrust. DIII-D is seeking to develop the reversed shear (RS) scenario at high beta and at low magnetic field using 110-GHz electron cyclotron current drive and a divertor for highly triangular plasmas. The Alcator C-mod tokamak, which operates at high magnetic field and density, is pursuing AT scenarios using lower hybrid wave current drive. The required AT performance levels have been attained transiently (see Fig. 1) and need to be extended to steady state.

Recent Successes

Recent advances in the AT performance gauge, the product of β_N and the H-factor, from the DIII-D tokamak are shown in Fig. 1. The JT-60U tokamak has reported the world record QDT equivalent = 1.25 using an AT mode.

Budget

See M-3.

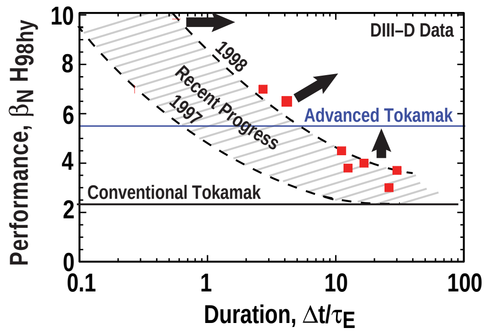


Fig. 1. Tokamak performance levels.

Anticipated Contributions Relative to Metrics

Metrics

Metrics have been quantitatively and qualitatively derived from predictive theory. Overall, the main metric will be the degree to which theoretical expectations can be achieved. The stability targets are calculated by accurate codes; a numerical target is to push β_N up toward 6, which requires wall stabilization. Attainment of ion transport near, at, or even below the neoclassical minimum has been shown in many tokamaks. Transport barriers in the electron channel have been seen, but it is probably not possible to reduce electron transport to its very small neoclassical minimum. A reasonable metric would be electron transport rates as low as ion neoclassical, which will suffice for future tokamak power systems. Achieving full self-generated current is the correct goal for steady state; but for control purposes, it will probably be desirable to drive some of the plasma current with external systems. Power and particle control progress can be gaged by the fraction of recombination and the fraction of radiated power that can be produced in the plasma mantle and divertor.

Near Term ≤5 years

In 5 years, it should be possible to sufficiently flesh out the ultimate potential of the tokamak to enable future tokamak designs to be based on AT approaches. The two tokamaks in the United States, DIII-D and Alcator C-mod, will carry out the bulk of this research if they are able to complete their necessary plasma control upgrades [ECCD and divertor on DIII-D, lower hybrid current drive (LHCD) and core diagnostics on Alcator C-mod]. Programs are in place to implement the required current profile control and divertor systems. A feedback system for wall stabilization is being implemented on DIII-D. While great progress is being made in transport barrier physics, transport barrier control methods are only just emerging. The JT-60U tokamak is working to extend AT modes to very long duration. The Joint European Torus (JET) tokamak expects to produce a record fusion power output using the “optimized shear” mode in its next D-T campaign.

Midterm ~20 years

A number of future devices that can take advantage of the AT physics can be demonstrated in the near term. The superconducting tokamaks KSTAR in Korea and HT-7U in China will take AT physics into very long pulse demonstrations. The JT-60SU superconducting tokamak in Japan predicates its D-T phase on AT operating modes. The reduced-cost (RC) ITER design has moved to the kinds of plasma shapes optimal for AT performance. RC-ITER will be a good development facility for the AT approach and may recover full ITER performance levels through AT operation. Compact, pulsed ignition experiments might be able to achieve high thermonuclear performance at lower machine parameters (7 MA and 7 T instead of 11 MA and 11 T) and for longer pulse lengths. We also note that the basic AT physics approaches (high β_N through hollow current profiles and wall stabilization, transport barriers through radial E-fields, high bootstrap fractions, and radiative mantles) are important elements of the spherical torus approach. The wall stabilization techniques developed in the AT program will be necessary in future long pulse spheromaks, and RFP and possibly in the field-reversed configuration (FRC). Suppression of turbulence by sheared ExB flow can be expected to appear in any device dominated by electrostatic turbulence.

Long Term >20 years

From the AT work done to date, the physics potential of the tokamak appears to exceed what is necessary to define a minimum-sized power plant based on technology limitations alone (stress levels on magnets, neutron fluxes at the blanket). For the superconducting tokamak, the minimum size is set by inboard radial build constraints to be over 5-m major radius. Using elements of AT physics at low aspect ratio, spherical tokamak (ST) pilot plants in the 1.5-m major radius range and power plants in the 3- to 4-m major radius range might be possible.

Proponents’ and Critics’ Claims

Proponents stress that “What is the ultimate potential of the tokamak configuration?” is the kind of basic scientific question the fusion science program should address (and for all configurations to be brought forward). Critics cite the uncertainty in achieving the AT objectives and the undesirability of delays in committing to construction of new devices while waiting for AT results. Proponents argue that higher performance and less costly devices may be built once the ultimate potential of the tokamak is known.

M-5. ELECTRIC TOKAMAK

Description

The electric tokamak (ET) differs from the advanced tokamak (AT) because it proposes to use the electric shear-flow stability not for improving confinement but for reaching deep second stability as its principal regime of operation. The goal is to approach unity beta in the plasma core at a high aspect ratio (A ~ 5). Strong radio frequency (rf) driven fast ion loss will be used for stability control. Because of the resulting rapid poloidal rotation, the usual thermal ion bananas do not have a chance to develop. In fact, the thermal ion drift surfaces become "omnigenous" for the ions, that is, locked to the magnetic surfaces. This is referred to as "electric omnigeneity," which is in effect when $V_{poloidal} > \varepsilon v_{ti}$. The resulting orbit level ion control should result in a low level of turbulence and in "classical"—not just neoclassical—ion confinement. Present-day tokamaks can only achieve neoclassical ion confinement. In addition to removing the ion bananas electrically, researchers will try to remove the electron bananas through developing magnetic omnigeneity in a deep magnetic well, where the mod-B surfaces tend to align themselves with the magnetic surfaces. In this high-pressure limit, expectations are that CLASSICAL electron thermal confinement can be approached as well. ET is now producing discharge cleaning type plasmas and 100-kA reversed-field-pinch (RFP) currents at only 70-gauss toroidal fields. Tokamak operation is expected to start in the summer of 1999. The initial physics research needed will be conducted in an R = 5 m, a = 1 m, b = 1.5 m torus at 0.25-T magnetic fields. If the physics exploration phase bears fruit the next 2 years, then a proof-of-principle experiment could be conducted in the same device, at 1 T, with minor upgrades. At 1-T fields, ignition-grade hydrogen plasma could be confined with the required Lawson parameters. This strictly requires near unity beta and near classical ion confinement. The electron channel can still remain below neoclassical.

Status

- The chamber is completed, and toroidal field coils will be completed by May 1999 at University of California–Los Angeles (UCLA).
- Conventional tokamak operation is scheduled to begin in the summer of 1999.

Current Research & Development (R&D)

R&D Goals and Challenges

Ion cyclotron range of frequency (ICRF) control for current drive, heat, and rotation control involves the development of codes and hardware. The code development is approached through a connection to numeric tokamak programs in the United States. The challenge will be to provide high-grade rf heating to the ET plasma.

Related R&D Activities

A low-cost chamber, magnet, and rf system development has been undertaken, so university-size resources can be used for fabrication of an International Thermonuclear Experimental Reactor (ITER) sized tokamak at low magnetic fields.

Recent Successes

Fabrication and checkout of the chamber and magnet systems has been more successful than anticipated. This can be attributed to the wide use of in situ chamber fabrication using computer-guided plasma torches and to the low magnetic field stress environment of this high-beta tokamak. The rf input is tested in one port, and 100-kA type RFPs have been produced for low-cost pulsed discharge cleaning.

Budget

$1.5 M/year or more is needed for the physics test.

Anticipated Contributions Relative to Metrics

Metrics

- High beta stability and control code development using low-cost parallel computing.
- Removal of the density limit in the tokamak using ET's high field heat input (MARFE control).
- Electron physics understanding using omnigenous magnetic surfaces (this is the only option).
- Development of techniques for reducing the cost of the tokamak reactor.

Near Term ≤5 years

Demonstration of reaching classical confinement and unity beta in a large torus with a minimum of 1-s energy confinement at $T(0) = 5$ kV, $n(0) = 5 \times 10^{13}/\text{cm}^3$.

Midterm ~20 years

Ignition and burn could be demonstrated using long pulses in an ITER-like chamber at near unity beta and at 2-T magnetic fields, using low-cost magnets. This requires a new facility with deuterium-tritium.

Long Term >20 years

With higher cost magnets, operated at 5 T, advanced fuels could be burned, if near unity beta and near classical confinement could be maintained. This would reduce "nuclear" costs.

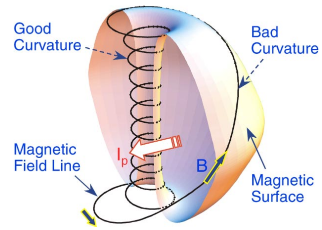
Proponents' and Critics' Claims

Proponents note that classical confinement for ions and near unity beta in the core can be reached through active plasma control, requiring rapid poloidal rotation of the ions. Present-day tokamaks tend to be limited to slow poloidal rotation. Then we will be able to study the stability properties of magnetic wells at high aspect ratio.

Critics claim that there is not solid theoretical evidence that strong poloidal rotation will stabilize ideal MHD modes at near-unity beta. Critics also say that the power requirements for the proposed level of plasma rotation may result in a very high (likely greater than unity) power recirculation fraction in a reactor. But here the critics refer to their own reactor that they know so well.

M-6. SPHERICAL TORUS

Description



ST plasma magnetic configuration with A ~ 1.25, elongation κ = 2, and edge safety factor q = 12.

- The aspect ratio (A = R_0/a) of the spherical torus (ST) plasma approaches unity (1.1–1.6 typically) compared to A = 2.5–5.0 so far for the tokamak and advanced tokamak (AT). As a result, the ST uses a modest applied toroidal field (TF) and has large I_p/aB and I_p/I_{tfc} values.
- Its magnetic surfaces combine a short field line of bad curvature and high pitch angle (relative to the horizontal plane) toward the outboard plasma edge with a long field line of good curvature and low pitch angle toward the inboard plasma edge. The dominance of good field line curvature leads to magnetohydrodynamic (MHD) stability at high plasma pressure, giving the potential for order-unity average toroidal and central plasma betas ($\beta = 2\mu_0 p/B^2$). High beta and the magnetic configuration combine to widen the parameter domain for magnetic fusion plasmas, the investigation of which will strengthen the scientific basis for attractive magnetic fusion energy (MFE) and other applications.
- The wider domain promises high-performance fusion plasmas possessing large trapped particle fraction (up to 90% near edge), Pfirsch-Schlüter current (~I_p), and toward the outboard, a magnetic well (~30%) with nearly omnigenous particle trajectories, dielectric constant $\left(\sim \omega_{pe}^2 / \omega_{ce}^2 >> 1\right)$, normalized gyroradius ($\rho^* = \rho/a \sim 0.03$–0.01), supra-Alfvén fast ions ($v_{fast} > v_A$), gradient-driven flow shearing rate (~$10^6\ s^{-1}$), magnetic mirror ratio (~4), and flux tube expansion (≥10) in the naturally diverted (ND) outboard scrape-off layer (SOL) of inboard limited plasmas.
- The ST configuration also requires certain features in engineering and technology to maximize the potential ST benefits. These include single-turn demountable, normal-conducting toroidal field coil (TFC) center leg and vertical replacement and assembly of fusion core components in toroidally symmetric sections, anticipated to simplify remote maintenance.

Status

- The small concept exploration experiments ($R_0 \leq 35$ cm), Small Tight Aspect Ratio Tokamak (START) in the United Kingdom, the Helicity Injected Tokamak (HIT) at the University of Washington, and the Current Drive Experiment-Upgrade (CDX-U) at Princeton Plasma Physics Laboratory (PPPL) have recently produced very encouraging results. These include average toroidal betas up to 40%, central local betas ~100%, H-mode confinement, a large range in operating density ($\leq 2 \times 10^{20}\ m^{-3}$) and q (1–30), a small and nearly symmetric halo current (~5% of I_p) during plasma disruptions, and helicity injection startup of I_p (up to 200 kA).
- New or upgraded ST experiments include the National Spherical Torus Experiment (NSTX) at PPPL with $R_0 \sim 0.8$ m, $I_p \sim$ 1–2 MA; and smaller experiments Pegasus (University of Wisconsin), HIT-II (University of Washington), CDX-U (PPPL), to explore ST physics boundaries in extreme low A, noninductive startup, and radio frequency (rf).
- NSTX is a national proof-of-principle (PoP) research facility at PPPL. Construction is expected to be completed in April 1999.

Current Research and Development (R&D)

R&D Goals and Challenges

- The NSTX R&D goals are to investigate noninductive startup and maintenance of plasma current to eliminate the central induction solenoid; efficient plasma heating and current drive; turbulence suppression to improve confinement; stability at high toroidal average beta (up to 45%) with large well-aligned pressure-gradient-driven (thermoelectric) current (up to 80–90% of I_p); and dispersion of plasma exhausts over large wall and tile areas.
- The NSTX with other ST experiments aim to establish a database in 4–5 years for a performance extension ST device, which in turn aims to produce a database for an energy development ST device (see Metrics).

Related R&D Activities

- Foreign ST experiments, such as the Mega-Amp Spherical Tokamak (MAST) in the United Kingdom; Globus-M (Russian Federation); TS-3&4, TST-M, and HIST (Japan); and ETE (Brazil) are complementary to the domestic ST experiments in goals and capabilities (e.g., MAST has poloidal field coils inside a large chamber and no stabilizing shell near the plasma).
- Advanced computational simulation is needed to account for the strongly toroidal ST geometry and physics features.
- Enabling technologies are of particular importance to ST due to the anticipated high power densities in its small size.
- Power plant conceptual studies identify innovative approaches for near-term R&D to maximize the ST advantage.

Recent Successes

- ST experimental results are very encouraging; several new ST experiments are approaching completion (see Status).
- Under recent Small Business Innovation Research (SBIR) funding, feasible design concepts were developed for ST-based volumetric neutron source (VNS) and ST-driven transmutation power plants to burn fission actinides, requiring only ST plasmas of modest-Q in the first-stability regime.
- Feasible ST reactor plasma operation scenarios, assuming the advanced physics regime, and simplified maintenance concepts were developed by the Advanced Reactor Innovation and Evaluation Studies (ARIES) group, the United States, and the United Kingdom Atomic Energy Authority (UKAEA) Culham Fusion, United Kingdom.

Budget

For NSTX in FY 1999, PPPL receives \$5.5M to complete the project, \$3.5M to prepare neutral beam injection (NBI), \$0.9M for a laser scattering system, and \$7.9M to operate the national facility; scientists from PPPL and 13 collaborating institutions receive \$5.4M and \$2.8M, respectively, to begin research. Collaborators' research funding for FY 2000 will hopefully grow toward \$5M.

Anticipated Contributions Relative to Metrics

Metrics

R&D targets and opportunities estimated for the ST-based VNS (R_0 ~ 1.1 m) and power plant (R_0 ~ 3 m):

Energy metrics	VNS	Power	Physics metrics	VNS	Power
TF center leg material	DS-Cu[a]	DS-Cu[a]	Full noninductive I_p ramp up	Yes	Yes
Q-engineering	~0	~4–6	Thermoelectric current fraction	~0.5	~0.9
Accessibility	Ample[b]	Ample[b]	Maximum β limit (%)/required β (%)	50/25	75/60
Maintenance approach	Eased[c]	Eased[c]	Required H_f (ITER98-H)	~1	~2
Environment and safety	*d*	*d*	Plasma power/wall area (MW/m^2)	~1	~1.6
Development path			Power and particle handling	ND	DN[h]
Modularity	*e*	*e*	Plasma/applied magnetic energy	~0.1	~0.1
Commonality of physics	*f*	*f*	B at R_0/B at TF coil	~0.25	~0.25
Unit capital cost ($B)	~1	~3	Uncontrolled shutdown frequency	$\rightarrow 0$	$\rightarrow 0$
Cost of electricity (mills/kW-h)		~70	Neutron wall loading (MW/m^2)	~1	~5
Near-term nonelectric applications	*g*				

[a]Dispersion-strengthened copper, which is estimated to be adequate for the VNS and could be improved for the reactor.

[b]The modest TF permits ample space between the TF return legs for access to the fusion plasma and core.

[c]Demountable center leg for the TF coil and compact fusion core enable vertical access to axisymmetric core components.

[d]Environment and safety issues, such as due to the juxtaposition of actively cooled normal, current-carrying conductors and high-performance power handling and conversion components, need to be investigated and resolved in the ST VNS.

[e]ST offers a modular low-cost path beyond NSTX: the deuterium-tritium spherical torus (DTST) for performance extension (R_0 ~ 1.1 m), VNS for energy development (R_0 ~ 1.1 m, Q_{eng} = 0), pilot plant (R_0 ~ 1.4 m, Q_{eng} ~ 1), and power demonstration plant (R_0 ~ 3 m, Q_{eng} ~ 4–6).

[f]The electron cyclotron wave effects on the ST plasma outboard resemble rf wave experimentation of the earth ionosphere.

[g]Success of ST VNS could enable viable nearer-term nonelectricity applications such as source for neutron science, transmutation of nuclear waste, and production of tritium and isotopes for fusion, industrial, and medical uses. The fusion physics and technology metrics for such applications are significantly less than those required for fusion power.

[h]Double-null (DN) divertors may be needed to ensure advanced physics regime operation assumed in an ST power plant.

Near Term ~5 years

- NSTX and MAST research programs plan to verify the physics metrics for the first-stability regime for DTST and VNS, which do not require active profile and mode control. The programs further plan to clarify the physics metrics for the advanced physics regime for a power plant, which requires detailed control and optimization of ST plasma properties.
- Design studies for the DTST and the ST VNS can be carried out, incorporating technologies already available or developed for the International Thermonuclear Experimental Reactor (ITER) Engineering Design Activity (EDA). Existing U.S. R&D facilities can be used to reduce costs for DTST and VNS.
- To improve Q-engineering for ST reactors, compact toroid (CT)-like approaches should be explored to minimize the applied TF via edge current drive (e.g., CHI and Rotamak current drive) while maintaining the desirable ST plasma properties.

Midterm ~20 years

- A DTST would verify the physics metrics for VNS and the reactor in fusion-relevant regimes at very high average plasma pressures ($\propto \beta B^2 \geq 100\%\ T^2$). Early verification of the advanced physics regime would enable integrated testing of ignited burn and energy technologies, which is the mission of an engineering test reactor (ETR), in a single VNS-size device.
- An ST VNS would be built to extend DTST results to steady state and begin R&D toward establishing the energy metrics, including TFC conductors to resist neutron damage and activation, and fusion blanket and core components.
- Equally important would be progress in theory and computation, fusion technologies and materials, environmental and safety techniques, and advanced systems design to ensure readiness to embark on an ETR.

Long Term >20 years

- An ST ETR (or pilot plant) would be built to demonstrate all metrics for fusion power, including scientific and technological feasibility, energy technology qualification, and safety and environmental soundness.
- The ST configuration would offer for a demonstration reactor (DEMO) a range of unit powers, from the small pilot plant (P_{DT} ~ 300 MW, Q_{eng} ~ 1) for zero net electricity, to the full-size power plant (P_{DT} ~ 3000 MW, Q_{eng} ~ 4–6) for 1000 MW in net electricity.

Proponents' and Critics' Claims

Proponents claim that the ST will provide an innovative, low-cost development path to an attractive fusion power source, with relatively simple technology and device maintenance. Disruptions may be less of a problem in the ST relative to the tokamak due to (1) lower magnetic field, (2) the absence of a delicate zero-magnetic-shear point in the plasma, and (3) high q operation. Neoclassical tearing modes may be stabilized by Pfirsch-Schlüter effects, and only very modest rotation is required for shear-flow stabilization of turbulence and rotational stabilization of MHD wall mode.

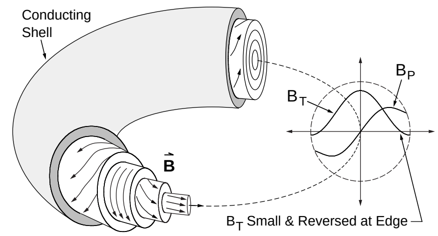
Critics claim that the modest engineering Q of a 1-GW(e) ST, largely due to the need to drive current in the copper center column, will push it to larger unit size. This may not be attractive in the energy market of the future. The ST could also suffer from disruptions and may have nearly the same size and complexity as a tokamak.

Description

The reversed-field-pinch (RFP) is a toroidal, axisymmetric, magnetically confined plasma. At first glance, the RFP looks much like a tokamak, but its substantial differences could lead to lower cost fusion. As in a tokamak, the RFP fusion plasma is confined by a helical magnetic field composed of both toroidal and poloidal components. Unlike a tokamak, plasma current generates most of the field, and the toroidal component is small and points in opposite directions in the edge and center of the plasma. This reversal of the toroidal field within the plasma leads to unique behavior and gives the RFP its name. The RFP's advantages, which could lower the cost of a fusion power plant, stem from the smaller toroidal magnetic field: (1) nonsuperconducting magnet construction, (2) lower field and forces at the magnet coils, (3) naturally large beta value, (4) high power density, and (5) possibility for Ohmic heating to ignition, and (6) simple assembly. Studies that project RFP plasmas to fusion conditions confirm that these features could reduce the cost of a fusion power plant, but such projections are relatively uncertain because the RFP is not as developed as the tokamak. RFP contributions to plasma science are especially important in topics such as magnetic relaxation, reconnection, dynamo, nonlinear coupling of magnetohydrodynamic (MHD) modes, inertial range MHD turbulence, and transport from magnetic stochasticity.

Status

Recent advancements in the scientific understanding of the RFP have led to an identification of possible solutions to many of the challenges confronting the RFP as a reactor concept. In the laboratory, multifold improvement in the heating and confinement performance of RFP plasmas is being realized in new experiments motivated by this understanding. In particular, the RFP's smaller magnetic field has greater susceptibility to magnetic turbulence. RFP research is providing a wealth of information on the behavior of plasmas with strong magnetic turbulence, from which new strategies for controlling plasma turbulence are being formulated and tested. Fundamental turbulence-related phenomena, such as magnetic relaxation and dynamo field generation, connect the RFP's unique terrestrial laboratory to similar processes occurring in stellar and planetary magnetic fields.



Magnetic field structure of the RFP.

The RFP program is one of three under consideration for proof-of-principle (PoP) status in the United States. Experimental research is undertaken at the Madison Symmetric Torus (MST) facility (1.5-m major radius, 0.52-m minor radius) at the University of Wisconsin–Madison. A small, low-aspect-ratio experiment is proposed to be constructed at the Princeton Plasma Physics Laboratory (PPPL). A small theoretical research effort (~3 person-years per year) is distributed in three to four laboratories.

Current Research and Development (R&D)

R&D Goals and Challenges

The key outstanding scientific and technical issues in RFP research are (1) understanding and improving confinement using advanced techniques such as current, pressure, and flow profile control; (2) determining the beta limit; (3) developing efficient current sustainment; (4) controlling resistive shell MHD instabilities; (5) power and particle handling; and (6) maximizing performance through configurational modifications, such as optimizing shape and aspect ratio. The proposed, favorably peer-reviewed, PoP program addresses most of these issues.

Related R&D Activities

In addition to MST in the United States, similar sized RFP experiments operate in Italy (RFX facility) and in Japan (TPE-RX facility). A smaller, resistive-shell RFP experiment operates in Sweden (Extrap-T2 facility). This worldwide RFP program is reasonably well coordinated through an International Energy Agency (IEA) working agreement.

Recent Successes

- Well-developed resistive MHD theory of the RFP, including the MHD dynamo (self-generation of current).
- Experimental confirmation of the MHD dynamo active in the RFP core and collisionless edge.
- Experimental determination that magnetic turbulence drives energy and particle flux in the core.
- Fivefold energy confinement improvement by halving magnetic turbulence using current profile control.
- Confinement improvement associated with induced and spontaneous changes in plasma flow and flow shear.

Budget

DOE–OFES: FY 1998 = $2.83M, FY 1999 = $3.53M; proposed PoP budget = ~$l0M/year.

Anticipated Contributions Relative to Metrics

Metrics

- All RFP experiments operate routinely with beta ~10%, and in best cases up to ~20%. The RFP is one of a few configurations that support the plasma pressure gradient by large magnetic shear, yielding theoretical ideal MHD interchange stability for beta <50%. Attractive reactor scenarios require beta ~20%.
- The minimum global heat conductivity achieved to date is $\chi \sim 10$ m^2/s (energy confinement time $\tau_E \sim 5$ ms) in plasmas with reduced magnetic turbulence through transient current profile control, a roughly fivefold improvement relative to a conventional RFP. Cases with up to threefold improved confinement associated with plasma flow changes occur spontaneously and with plasma biasing. The most recent RFP reactor study (TITAN) assumed an energy confinement time of 200 ms, smaller than typical tokamak reactor studies because a large plasma density is believed possible in the RFP.
- The magnetic field utilization efficiency in the RFP is high because most of the confining magnetic field is produced by plasma current. The field is maximum at the plasma center, with a value about 2.5 times larger than at the plasma boundary. The external magnet force and field requirements are therefore relatively small. The TITAN reactor study assumed a current of 18 MA ($B_{coil} \sim$ 5 T), and ≤1-MA experiments are under way ($B_{coil} \leq 0.5$ T).
- To take full advantage of the RFP's compact reactor potential, first walls capable of withstanding large heat flux (~5 MW/m^2) and deuterium-tritium (D-T) neutron flux (~20 MW/m^2) must be developed. Pioneering ideas for highly radiating plasma boundaries to symmetrize the heat flux evolved from RFP reactor studies.
- Empirically, RFP plasmas do not suffer current disruptions as observed in tokamaks. This could result from the larger magnetic turbulence in the RFP. With progress in turbulence suppression, disruptions could appear.

Near Term ≤5–10 years

- The observed beta in experiments is not believed to be limited by pressure-driven instability; rather, it reflects large transport in Ohmic-only heated plasmas. Auxiliary neutral beam and radio frequency (rf) heating are proposed to control directly the plasma pressure and to determine the beta-limiting physics.
- To minimize magnetic turbulent transport, precise, nontransient current profile control using electrostatic and rf current drive schemes are being developed and tested. The influence of plasma flow on turbulence and transport in the RFP is a new research topic that will be developed theoretically and exploited experimentally.
- Oscillating field current drive, a potentially highly efficient steady-state current sustainment technique, will be tested. This technique may not be consistent with requirements to reduce magnetic turbulence, so additional ideas are being sought for testing.
- Almost all RFP experiments have been circular in cross section with an aspect ratio R/a ≥ 3. Optimization of the cross-sectional shape and aspect ratio will be explored theoretically and tested experimentally, most likely with new, smaller devices.
- MHD kink modes are expected to grow on the flux diffusion time scale of the conducting shell surrounding the plasma. If such modes arise, control techniques will be devised using plasma rotation, active feedback, or smart shell techniques. This effort should be cooperative with R&D for the control of similar instabilities in advanced tokamaks, spherical tokamaks, and other high-beta configurations.

Midterm ≤10–20 years

- A successful PoP program warrants a next step, high-current, long-pulse, proof-of-performance facility with the necessary systems integrated to assess fully the scientific and technological readiness of the RFP reactor concept. Key technologies in power and particle handling, which do not limit existing device operation or the proposed PoP program, would be developed at this stage, as has occurred in tokamak research.
- As the optimization of the RFP proceeds, reactor studies will be updated to assess the impact of added system complexity on reactor cost. This will help guide the future of RFP research. The reduced Lorentz force stresses implied by the RFP's smaller field requirements could make a pulsed reactor scenario economical and reduce the demands on current sustainment. Advanced liquid-metal blanket concepts are well-suited to the RFP.

Proponents' and Critics' Claims

Proponents claim that the RFP's smaller toroidal magnetic field requirement could lead to an easily assembled, compact, high-power-density reactor. These cost-saving features have been verified in reactor system studies based on to-be-verified extrapolations of RFP results.

Critics claim that the RFP's small toroidal field permits inherently larger turbulence and transport, which could prevent achieving high power density RFP plasmas. Steady-state current sustainment is more difficult in an RFP because the required current is large and the plasma pressure-driven (bootstrap) current is small when the dominant magnetic field component is poloidal.

M-8. SPHEROMAK

Description

- The spheromak is a compact magnetofluid configuration of simple geometry with attractive reactor attributes, including no material center post, high engineering beta, and sustained steady-state operation through helicity injection. It is a candidate for liquid walls in a high-power-density reactor.
- It has a toroidally symmetric equilibrium with toroidal and poloidal fields of comparable strengths. The simplicity and compact size provide good diagnostic access, and spheromaks are relatively inexpensive to build.
- Magnetic helicity (linked magnetic fluxes) plays an important role in forming and sustaining spheromaks. An initial configuration with sufficient helicity and energy will spontaneously relax to a spheromak given appropriate boundary conditions.
- Electrostatic helicity injection has been demonstrated to sustain the spheromak current efficiently via a magnetic dynamo involving flux conversion and has been implemented in several experiments.

Status

- Previous experiments have demonstrated the basic tenets of the spheromak: self-organization at constant helicity; control of the tilt and shift modes by shaped flux conservers; and sustainment by helicity injection, including the role of magnetic fluctuations and reconnection in current drive via the magnetic dynamo.
- Spheromaks were inductively formed without close-fitting walls in S-1 and their global stability maintained by passive figure-eight walls. The relationship of relaxation phenomena and confinement was studied on S-1 and CTX.
- Several groups attained electron temperatures above 100 eV in decaying plasmas, with CTX reaching 400 eV. This experiment had a high magnetic field (>1 T on the edge and ~3 T near the symmetry axis).
- Analysis of CTX found the energy confinement in the plasma core to be consistent with Rechester-Rosenbluth transport in a fluctuating magnetic field, potentially scaling to good confinement at higher electron temperatures.
- The SPHEX group (Manchester, England) studied the dynamo in sustained spheromaks in a cold plasma.
- The Sustained Spheromak Physics Experiment (SSPX) at Lawrence Livermore National Laboratory (LLNL) is addressing the physics of a midscale sustained spheromak with tokamak-quality vacuum conditions and no diagnostics internal to the plasma.
- Supporting spheromak experiments are the Swarthmore Spheromak Experiment (SSX) (reconnection) and the Caltech Helicity Experiment (spheromak formation issues). The fundamental physics of reconnection is being studied in MRX at Princeton Plasma Physics Laboratory (PPPL). The Helicity Injected Torus (HIT) at University of Washington studies coaxial helicity injection in a tokamak.

Current Research and Development (R&D)

R&D Goals and Challenges

- Improve understanding of the coupling of the helicity injector to the spheromak and how to sustain it with minimum perturbation to the axisymmetric configuration, optimizing helicity current-drive efficiency.
- Achieve T_e of several hundred electron volts in a sustained spheromak, and understand the physics of energy confinement in the presence of the magnetic dynamo during sustainment; use this understanding to improve confinement.
- Determine the beta-limiting processes in the spheromak plasma, and maximize the beta.
- Understand and control the processes that determine the properties of the edge/boundary plasma, including the role of edge current density in helicity injection into the core plasma, atomic and molecular processes, instabilities, impurity generation, and other effects due to wall interactions.

Related R&D Activities

- The reversed-field-pinch (RFP) depends on much of the same physics as the spheromak (e.g., magnetic dynamo), but with an applied toroidal magnetic field and different unstable magnetic modes due to the different safety factor profile.
- Compact tori used in fueling experiments for tokamaks are small spheromaks accelerated to high velocity.
- Magnetic reconnection, studied in spheromak experiments and modeling, is a fundamental physics issue applicable to astrophysics as well as to helicity current drive in fusion confinement configurations.
- The spheromak is a possible target plasma for magnetized target fusion.

Recent Successes

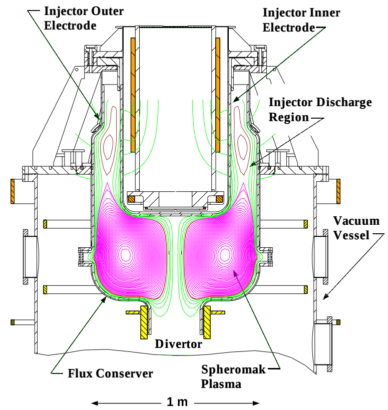
- MRX and SSX merged spheromaks and studied the magnetic reconnection and generation of energetic plasma flows. On MRX, the three-dimensional (3-D) structure of the reconnection layer has been extensively studied. SSX is examining the 3-D structure of the reconnection layer and the generation of energetic ions.
- The 3-D movies of the Caltech Helicity Experiment show very clearly the time evolution of the twisted magnetic flux tubes emanating from the muzzle of the coaxial spheromak gun.
- Initial spheromak modeling using the NIMROD resistive magnetohydrodynamic (MHD) code has demonstrated buildup of closed time-averaged magnetic surfaces from a current column in a cylindrical flux conserver.

Budget

DOE–OFES: FY 1998, 1999 = $1.5M/year for SSPX. Los Alamos National Laboratory (LANL) is funded ($0.1M) to collaborate on SSPX. An LLNL project on spheromak physics is funded at $1.05M this year with internal R&D funds; this money needs to be replaced by additional Office of Fusion Energy Sciences (OFES) funds for the project to progress at a reasonable rate. There is presently no funding of theoretical support for spheromak research.

Anticipated Contributions Relative to Metrics

Metrics



SSPX spheromak, showing MHD equilibrium

- Confinement parameter/required level (power plant): Values similar to other magnetic confinement configurations are required for a power plant. In previous experiments, confinement has been poor due to magnetic fluctuations. Theory indicates that the dynamo can be sustained at very low magnetic fluctuation levels for reactor conditions. If achieved, confinement may be determined by electrostatic fluctuations, not yet studied in spheromaks.
- Reactors require beta ~10%. Peak beta >20% was demonstrated in CTX, and spheromak configurations stable to the Mercier criterion have been modeled with $\beta_p = 1$. However, the lowest energy configurations (Taylor state) generally have low beta (few percent), and the consistency of high-beta configurations with low magnetic fluctuation levels remains to be demonstrated.
- B(plasma)/B(max coil) ≈ 1.
- P(heat-wall)/area: In spheromaks with conventional walls, this condition is similar to other magnetic confinement devices. The simple geometry may allow liquid walls and a much more compact reactor.
- Beta(%) $\times$ $[B_{plasma}(T)]^2$ is potentially high, but difficult to evaluate given physics uncertainties.
- Uncontrolled shutdown frequency: No plasma-driven shutdown mechanisms have been identified.
- Power and particle handling: A divertor is naturally formed in spheromaks, allowing handling of power and particles. The close-fitting wall may limit power handling at a level not presently quantified.
- Fundamental physics: Extend understanding of helicity, reconnection, and magnetic dynamo current drive.

Near Term ≤5 years

- The SSPX experiment is in the final stages of construction. SSPX is designed to study the confinement in a short-pulsed, sustained device; to relate this confinement to magnetic fluctuation physics; and to extrapolate confinement to larger, long-pulse spheromaks and to possible reactors.
- Understanding of reconnection physics explored in smaller devices (e.g., SSX) will be applied to the confinement experiment, and results from the confinement experiment will be applied to space plasmas. Close collaborations with RFPs and spherical tokamaks (STs) should yield synergistic advances in common physics.
- Beta limits will be studied as a function of the current profile and other parameters in SSPX.
- A divertor will be used in SSPX to evaluate power and particle handling requirements and techniques.
- The need for feedback control of the tilt and shift instabilities in long-pulse experiments and reactors will be evaluated and the development of any needed feedback system started.
- A long-pulse experiment will be designed and proposed before the end of the 5-year period to build on successful results. It will study instability control, beta limits, and use of external poloidal-field coils to evaluate B(plasma)/B(max coil).

Midterm ~20 years

- A long-pulse experiment will extend performance and address spheromak physics in the kiloelectron-volt temperature range. Issues will include confinement scaling, beta limits.
- Advanced helicity drive techniques will be developed. Other current drive techniques will be evaluated.
- Technology needs for innovative spheromak reactors will be developed. Possible options include liquid walls of lithium and fluorine-lithium-beryllium (Flibe). $\beta_p \sim 1$ may be possible in a repetitively pulsed reactor.

Long Term >20 years

- A proof-of-performance experiment will be conducted.
- Innovative reactor concepts (e.g., pulsed, high-beta reactors with liquid walls) could lead to early deployment of a power reactor based on the spheromak concept.

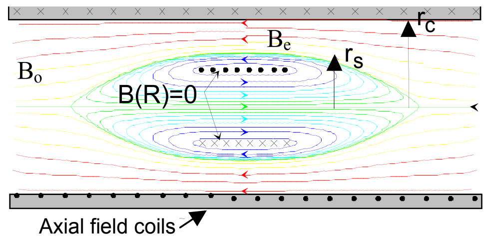
Proponents' and Critics' Claims

Proponents claim that the spheromak leads to a potentially compact and simple reactor, with a potentially efficient current drive using coaxial (or other) helicity injection. The physics studied in the spheromak may also contribute to other concepts such as the RFP and is pertinent to studies of interstellar plasmas.

Critics claim that the spheromak may not have adequate confinement for a power plant and that there is no fully demonstrated way to sustain the plasma current for long pulses.

Description

A field-reversed configuration (FRC) is a compact toroidal plasma with negligible toroidal magnetic field. It is usually fairly elongated, contained in a solenoidal magnetic field, and possesses a simple, unobstructed divertor. The plasma beta is close to unity, and an FRC is thus both extremely compact and geometrically simple. The essentially diamagnetic current can be primarily sustained by central fueling, but some augmentation will be required at the field null. The observed stability in present experiments is thought to be due to kinetic effects, which have been characterized by a parameter s, equal to the number of ion gyro-radii between the field null R and the separatrix r_s. The utility of the concept depends on demonstrating stability as s is increased from present values of about 4 to the 20–30 levels thought needed to provide reactor level confinement. The enhancement of kinetic stabilization, either through addition of energetic particles (e.g., ion ring merging or neutral beam current drive) or naturally occurring fusion reaction particles, may be an essential component of the concept, although there is some theoretical and experimental evidence that FRCs may be naturally occurring minimum energy states stabilized by sheared rotation akin to spheromaks and reversed-field-pinch (RFP) devices when total helicity (including angular momentum) is conserved.



FRC Geometry

Status

The FRCs have been formed with high plasma pressures in theta-pinch devices. Without an external current drive, these current rings decay on sub-millisecond L/R times. Due to typical densities of 5×10^{21} m^{-3}, nτ products of close to 10^{18} m^{-3}s have been achieved in FRCs with major radii of 15 cm at several 100-eV temperatures. Lifetime has been observed to increase with density: shorter lived FRCs are easily produced at n ~ 10^{21} m^{-3} with kiloelectron volt temperatures. Stable FRCs with s values of up to 4 and poloidal fluxes of 10 mWb were produced in a Large s Experiment (LSX), which was built from 1986–1990, but only operated for 1 year due to reductions in funding for alternate concepts at that time. Rotating magnetic fields (RMFs) have formed and sustained FRCs in small rotamak experiments in Australia and will be applied to a modified version of LSX [Translation, Confinement, and Sustainment (TCS) experiment] to attempt to increase the flux and sustain the configuration in quasi steady state.

Current Research and Development (R&D)

R&D Goals and Challenges

The principal R&D goal is to develop an understanding of the highly unique physics of this extremely attractive reactor configuration. The principal technological challenge is to produce hot, higher s FRCs and sustain them for sufficient time to study their properties. A list of specific goals and challenges follows:

- Form large, low-density FRCs by translating and expanding theta-pinch-formed FRCs.
- Increase the flux and produce higher s FRCs by applying high-power RMF.
- Develop fueling and heating methods to go along with RMF current drive.
- Sustain hot FRCs with moderate s values for millisecond timescales.
- Develop specialized diagnostics for internal profile measurements.
- Develop an efficient technology for both forming and sustaining hot, high flux FRCs.
- Develop a theoretical understanding of FRC stability in its unique kinetic regime, and develop sufficient understanding of FRC confinement to allow confident extrapolation to larger devices.

Related R&D Activities

- RMF theory in Japan and possible RMF current drive on spherical torus.
- U.S.–Japan FRC research coordination and low-density confinement studies at Osaka University.
- U.S.–Japan studies on D-^{3}He reactor—ARTEMIS beam-driven design with direct conversion.
- NASA-sponsored studies of simpler formation techniques for space propulsion.
- Ion ring generation studies at Cornell University.
- Merging spheromak experiments at Princeton Plasma Physics Laboratory (PPPL).

Recent Successes

- Production of stable s = 4 FRCs in LSX.
- Production of low-density FRCs with enhanced confinement in Osaka FIX experiment.
- Translation and acceleration of FRCs in Tokamak Refueling by Accelerated Plasmoids (TRAP).
- Steady state (40-ms) RMF formation and sustainment of FRCs and spherical torus in Australia.
- Formation of hot FRCs through merging of opposite helicity cold spheromaks at PPPL and Tokyo University.

Budget

- DOE–OFES: FY 1998 = $1.5M, and FY 1999 = $1.5M.
- NASA: FY 1998 = $100K, and FY 1999 = $75K.

Anticipated Contributions Relative to Metrics

Metrics

The FRC is extremely compact, has near unity beta, plus a simple solenoidal magnet system. To realize its high reactor promise, stability must be demonstrated at high s, and the present empirical confinement scaling must be improved at low densities. The technology must also be developed to produce higher flux FRCs and to sustain them. This includes heating and fueling in addition to current drive.

Near Term ~5 years

The theta-pinch-formation technique is capable of forming somewhat higher flux FRCs than the 10 mWb presently demonstrated in LSX, but the method is technologically limited to the tens of milliweber level. Several weber will be required for a reactor. Some method of both flux buildup and sustainment is needed for the continued development of the concept. The RMF approach is not technologically limited for either flux buildup or sustainment, and high-power RMF will be applied to hot, theta-pinch-formed FRCs in a collaboration (TCS experiment) between the University of Washington and Los Alamos National Laboratory.

The FRC stability theory has advanced to the point where combinations of profile shaping and gyro-viscous effects can account for the presently observed stability. Progress is also being made on understanding the tendency of counter-helicity spheromaks to merge into a higher temperature FRC. Sheared flow may perform the same stabilizing function as magnetic shear in more conventional magnetic confinement schemes. Ponderomotive forces due to the RMF are also thought to provide stabilizing properties, although the theoretical understanding of this exciting technique, as applied to hot FRCs, is only in its infancy. A coordinated approach to the development of stability theory for the unique FRC confinement scheme is now underway in a multi-institutional effort led by PPPL.

Empirical scaling laws have been developed to reflect experimental FRC confinement times in high-density experiments, but no theory exists to adequately explain this scaling. The empirical diffusivity, $D \sim 5\ m^2/s$, is adequate for high-density pulsed reactors but must be reduced by about an order of magnitude to realize more desirable, low-density, small compact reactors. There is some evidence of better low-density confinement in Japanese experiments (FIX), and this will be explored in the TCS program where lower densities are reached through translation and expansion.

As currently funded, most of the experimental burden for FRC development will fall on the newly constructed TCS facility. It is particularly important that facilities other than TCS alone should undertake FRC research to fully explore the many facets of its unique physics. PPPL has proposed the SPIRIT facility to form FRCs by merging opposite helicity spheromaks. This facility would allow modification of the FRC shape (from prolate to oblate) and eventually provide for neutral beam fueling that could both sustain the configuration and increase the kinetic ion component. It would also be useful if the technology for producing field-reversing ion rings at Cornell was further developed, in case it was needed for stabilization.

Midterm ~20 years

If RMF current drive were successful, very rapid progress could be made because reactor level magnetic fields are only on the order of a few tesla in simple solenoidal magnets, and the required RMF power is not expected to increase significantly with device size. The 80-cm-diam TCS facility could be extended somewhat to the s ~10 levels by extending the pulse length of the RMF power supplies and developing fueling methods. A larger ~$50M facility should be built at a national laboratory to produce steady-state FRCs with thermonuclear temperatures. With success, capabilities could be rapidly increased to the level of those proposed for the International Thermonuclear Experimental Reactor.

Long Term >20 years

The remaining technologies relevant to any deuterium-tritium fusion device, mainly a first wall and tritium-producing blanket, would need to be developed. To realize the ultimate potential of the FRC concept, the use of more aneutronic fuels should be investigated. The principal technology here would be an efficient direct converter.

Proponents' and Critics' Claims

Proponents claim that FRCs are the most attractive of all magnetic confinement schemes and that their unique physics is largely unexplored. Many nonideal magnetohydrodynamic (MHD) effects are present that could lead to maintenance of stability even at large s. It is hoped that if the RMF current drive scheme is successful, FRCs will no longer be regarded as mere pulsed curiosities by other fusion researchers. Critics claim that that FRCs are mere pulsed curiosities lacking basic MHD stability and that they are only stable due to their present small size and low s number.

M-10. LEVITATED DIPOLE FUSION CONCEPT

Description

The dipole magnetic field is the magnetic field far from a single, circular current loop. The use of a dipole magnetic field to confine a hot plasma for fusion power generation was first considered by Hasegawa (*Comm. Plasma Physics,* 1987). In this configuration, a relatively small superconducting ring floats within a large vacuum chamber. The dipole confinement concept is based on the idea of generating pressure profiles having gentle gradients and being stable to all low-frequency magnetic and electrostatic fluctuations. From ideal magnetohydrodynamic (MHD) conditions, marginal stability results when the pressure profile satisfies the adiabaticity condition, $\delta(pV^{\gamma}) \geq 0$, where V is the flux tube volume and $\gamma = 5/3$. For usual magnetic confinement devices, this constraint can not be satisfied, and MHD stability must be provided by field-line averaging and magnetic shear. In contrast, because of the large dimensions of the dipole plasma on the outer radius of the levitated coil, the pressure profile can scale with radius as rapidly as $R^{-20/3}$ and still remain absolutely MHD stable to beta of order 100%. Because the coil set for a possible dipole fusion power source is very low weight and axisymmetric, operation is inherently steady state, easy to maintain, and potentially of low unit cost. Because the dipole concept permits high beta without magnetic shear, the concept may allow removal of fusion products without degrading energy confinement and thus be appropriate to advanced fusion fuels, such as D-^{3}He. Conceptual reactor studies have supported the possibility of an attractive fusion power source using a levitated dipole.

Status

- The scientific literature for dipole confinement includes space-based observations and theory, laboratory experiments, and fusion theory. This understanding establishes a bridge from the high-beta confinement observed in planetary magnetospheres to the confinement expected in levitated dipoles—where hot plasma will be confined for many collision times. The understanding gained from decades of space plasma research supports the levitated dipoles as a fusion confinement device. Approximately a dozen studies have been published specifically addressing the use of dipole-confined plasmas for energy production or space propulsion. Finally, the CTX device at Columbia University has illustrated the stability properties of collisionless energetic electrons confined by a dipole magnetic field.
- The first experiment to investigate the levitated dipole concept is presently under construction at the Massachusetts Institute of Technology (MIT) as a joint project between Columbia University and MIT. First plasma is expected in the year 2000.

Current Research and Development (R&D)

R&D Goals and Challenges

An essential first step for the understanding of the scientific feasibility of a levitated dipole is a laboratory test of confinement properties of such a device. The Levitated Dipole Experiment (or LDX) has been designed to test the scientific feasibility of levitated dipole confinement at high beta. Construction of the experiment began in July 1998. A major technical challenge has been met through the collaboration between superconducting magnet technology experts and innovative experiment design: a large current (1.3 MA) must be sustained in a ring having low mass. Another challenge is to complete more detailed modeling of dipole power sources. Limited reactor engineering studies are now under way.

Related Research Activities

The conceptual development of the dipole fusion power source has been inspired by space plasma studies and planetary exploration. Continued exploration of magnetospheric plasmas (e.g., the exploration of the Io plasma torus that surrounds Jupiter and the development of physics-based models of space weather) will add to the cross-fertilization of this area of plasma science. Although the dipole plasma confinement concept is a radical departure from the better known toroidal-based magnetic confinement concepts, a scientific investigation of magnetized plasma confinement with high compressibility will add insight to these other more traditional confinement concepts. In the technology area, the development of advanced and high-temperature superconductors will aid significantly in the reactor conceptualization of a dipole power source. Experiments to investigate low-density, nonneutral plasma confinement with a levitated dipole device are planned for the University of Tokyo.

Recent Successes

The engineering design of the LDX facility is almost complete. The major fabrication items, including the vacuum chamber and the floating coil, are either under construction or awaiting final selection of the vendor. Theoretical research supporting the possibility of classical confinement in a dipole-confined plasma was recently published.

Budget

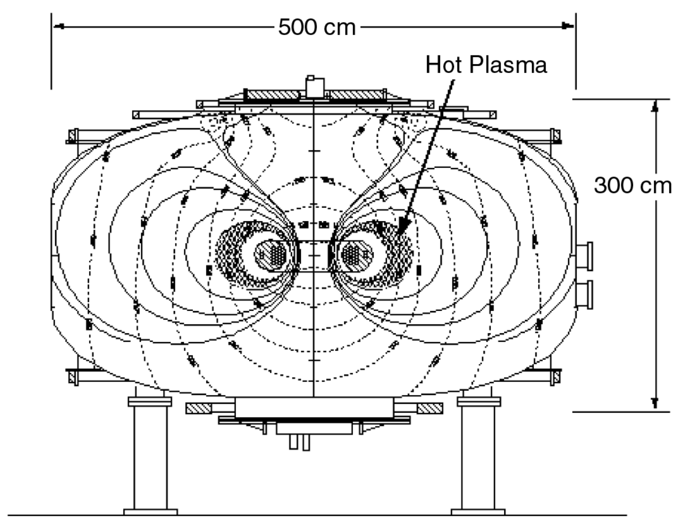
The LDX project budget in FY 1999 is $1.2M, which is divided between Columbia University and MIT.

Anticipated Contributions Relative to Metrics

Metrics

The levitated dipole confinement concept has the potential to be an attractive fusion power source while at the same time contributing to the understanding of magnetospheric physics. The relative simplicity of the magnetic field structure and the good understanding of plasma confined in planetary magnetospheres indicate that plasmas confined by levitated dipole magnets will have (1) high peak beta (≥50%) and volume-averaged engineering beta (≥10%), (2) excellent field utilization (≥90%), and (3) possibly sufficient confinement to ignite D-^{3}He fuel. Because the plasma equilibrium and magnetic geometry are essentially determined by sustainable current in superconducting magnets, the dipole fusion concept is disruption free and promises to be highly reliable. Because (1) the particle flux to the floating ring is 100% recycled, (2) the shear-free magnetic field allows convection of fusion ash without energy confinement degradation, and (3) the outer plasma surface is open, accessible, and naturally diverted, power and particle handling should be superior to other magnetic confinement concepts. The dipole is best suited for D-^{3}He fusion fuel, and this may lead to a reduction in fusion's adverse impact on the environment.

Near Term ≤5 years



For the first time in the laboratory, high-beta plasma having near classical energy confinement scaling for time scales long compared with particle and energy confinement times will be investigated. Experimental observations will be compared with theory. For this purpose, the LDX has been conceived and designed as the lowest cost approach for investigating the key physics issues while simultaneously maintaining high confidence of its technical success. The experimental approach takes two stages. First, multiple-frequency electron cyclotron resonance heating (ECRH) (with frequencies between 6 and 28 GHz) will be used to produce a population of energetic electrons at high $\beta \sim 1$. This technique has been proven effective in magnetic mirror experiments (e.g., Constance and Tara). Based on experience generating hot electrons within mirrors and within CTX, the creation and maintenance of high-beta plasmas using a few tens of kilowatts of ECRH power is expected. Secondly, after formation of the high-beta hot electron plasma, fast deuterium gas puff techniques or the injection of lithium pellets will be used to thermalize the energy stored in the hot electrons and to raise the plasma density. The resulting thermal plasma will provide a test of the MHD limits and of the confinement of a thermal plasma in a levitated dipole.

Midterm ~20 years

Proof-of-principle dipole confinement experiments that operate with plasma parameters resembling those which might be found in fusion power sources need to be designed and constructed. Better understanding of the possibility of practical sources of ^{3}He fuel need to be achieved.

Long Term >20 years

Successful operation of a fusion power source and an assessment of the applicability of D-^{3}He dipole fusion for commercial energy and for spacecraft power and propulsion are long-term objectives.

Proponents' and Critics' Claims

A dipole power source has the possibility of being steady state with classical confinement and high beta. Compared with a tokamak it would not require current drive; it is disruption free; and it has a natural divertor. The dipole has a relatively simple magnetic configuration that does not have interlocking coils. Critics challenge the practicality of a floating ring within a fusion plasma. In a reactor embodiment, the large diameter of the outer wall may result in an undesirable small power flux (although a number of design options could ameliorate this problem). Convective cells could lead to enhanced transport; however, at marginal profiles they may provide a means to fuel and to remove ash. The dipole concept is most compatible with the burning of advanced fuels, such as D-^{3}He, and this fuel simplifies some of the technologies required for dipole fusion power sources. Critics argue that the need for lunar mining or ^{3}He breeders to provide adequate supplies of fusion fuel will add significantly to the cost of commercial dipole fusion power systems.

M-11. OPEN-ENDED MAGNETIC FUSION SYSTEMS

Description

- Open-ended magnetic fusion systems are a generalization of the classic magnetic mirror. In the "classic" system, plasma is confined radially by the magnetic field and axially by the reflection from regions of high magnetic field due to the constancy of the magnetic moment. Magnetohydrodynamic (MHD) stability to interchange modes is provided by quadrupole (or higher) magnetic fields, which generate a minimum in the magnetic field.
- To enhance axial confinement, the tandem mirror generates electric potentials, which traps ions and/or electrons. The potentials are essentially ambipolar in nature, with one or both particle species having a non-Maxwellian component.
- An approach that has been investigated is the multimirror system, in which end losses are limited by coupling a series of short mirror cells at each end of a longer central cell. To escape from confinement, the particles must diffuse from cell to cell so that if certain conditions are satisfied, the confinement time increases at least as the square of the number of end cells.
- Another approach is the gas dynamic trap (GDT), in which the bulk plasma is collisional. This plasma is confined in an axisymmetric magnetic field, with MHD stability generated by the flow of plasma in the good-curvature region where the field expands outside the mirror. One possible application of the GDT is to a neutron source (see M-19. Volumetric Neutron Source and M-12. Gas Dynamic Trap for details).
- New concepts could also provide enhanced axial confinement, for example, the kinetic tandem concept, which uses energetic ion beams injected into the magnetic field from the ends with sufficient perpendicular energy to provide high-density, high-potential regions where the ions reflect from regions of strong magnetic field. In a sufficiently long linear system, net power could be produced.

Status

- Tandem mirror research is being conducted on Gamma-10 in Tsukuba, Japan, using both ion cyclotron resonance heating (ICRH) and electron cyclotron resonance heating (ECRH) to heat the plasma and provide the non-Maxwellian distributions necessary to enhance confinement. The plasma density is typically about 2×10^{19} m^{-3}, central-cell T_i ~ 3 keV, and T_e ~ 100 eV.
- An experiment operating at the Budker Institute of Nuclear Physics, Novosibirsk, Russia, has demonstrated much of the GDT neutron source physics with a pulse length of 3–5 ms: stability against MHD modes, stability against ion velocity–space modes so that the ions decay and scatter classically, and T_e = 130 eV (equal to classical predictions for the available neutral beam power—3.5 MW at 15 keV). The device is routinely operated at plasma beta of 30%, without any signs of gross MHD instabilities. Electron energy balance has been shown to be classical within the parameter regime explored to date.
- Several experiments have reported similar confinement parameters. The tandem mirror, TMX-U, had a global energy confinement of $n\tau = 2 \times 10^{16}$ s/m^3. The ion perpendicular energy ("temperature") was 2 keV (parallel was about 0.4 keV). Thus, $n\tau T_i$ was 4×10^{16} keV-s/m^3. Drag on the electrons limited energy confinement in several devices: Gamma-10 has a central cell density of 10^{18} m^{-3} and T_i up to 7 keV (also perpendicular). The energy confinement is limited by drag on electrons, so τ ~ 2 ms. Thus, $n\tau T_i = 1.4 \times 10^{16}$ s/m^3. The GDT at Novosibirsk has n = 10^{19} m^{-3}, T_i = 7 keV, and is also limited by drag on relatively cold electrons so that τ = 0.4 ms. Thus, $n\tau T_i = 2.8 \times 10^{16}$ s/m^3.

Current Research and Development (R&D)

R&D Goals and Challenges

- The U.S. has no operating open confinement experiments.
- Research on the GDT is being carried out on the GDT physics and its application to a neutron source at the Budker Institute.
- Research on tandem mirror physics is being conducted at Tsukuba in the Gamma-10 experiment and in the AMBAL experiment at the Budker Institute.
- The Hanbit tandem mirror experiment is being put into operation in Korea.

Budget

There are presently no U.S. funds being spent on open systems.

Anticipated Contributions Relative to Metrics

Metrics

Open-ended systems have demonstrated the possibility of confining plasmas in a near-quiescent state, one where the confinement time approaches the “classical” value, that is, value calculated from collisional processes alone. Until larger facilities are constructed, the metric for subscale open-ended experiments should be the degree to which the confinement time is found to scale in a classical manner. If this scaling is convincingly demonstrated, the extrapolation of small-scale experiments to fusion-relevant scale can be done with a high degree of confidence.

Near Term ≤5 years

The most important short-term experiments are those in confinement physics and scaling. The GDT experiments have shown classical confinement in axially symmetric confinement fields. The GDT results could, therefore, be a starting point for the design of a small-scale U.S. experiment. In addition, university-scale experiments could investigate the new approaches and the plasma physics issues that they involve. Because there has been a hiatus in open-ended research in the U.S. program, it will be necessary to rebuild aspects of the intellectual infrastructure that was in place before that hiatus. In parallel with the confinement physics investigations should be work on critical technological issues for open-ended fusion systems, such as direct conversion.

Midterm ~20 years

If the near-term experiments verify the predictability of confinement of the new breeds of open-ended systems, scale-up to achieve fusion-relevant conditions would be justified.

Long Term >20 years

Success at fusion scale in the achievement of classically confined plasmas implies fusion power systems that could be much smaller and less complex than present closed systems.

Proponents’ and Critics’ Claims

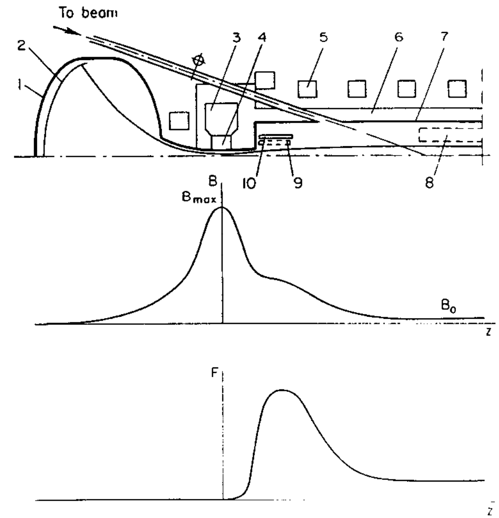
Proponents believe that open-ended systems, with their flexibility for innovation, their theoretical tractability, and their demonstrated ability to achieve near-classical (turbulence-free) confinement time, offer the best possibility for overcoming the widely recognized limitations of main-line closed-geometry systems, such as the tokamak.

Critics claim that the electron physics concepts have not been demonstrated at the required temperature, that the Q is small and results in low-power efficiency, and that the mirror physics and technology are deadend developments for fusion power.

M-12. GAS DYNAMIC TRAP

Description

- The Gas Dynamic Trap (GDT) Neutron Source (GDTNS) is a volumetric plasma (14-MeV) source with a neutron spectrum and intensity very close to that predicted for International Thermonuclear Experimental Reactor (ITER) and fusion reactor designs.
- Neutrons are produced by a neutral beam injected at 30°–45° to the magnetic field in an axisymmetric mirror machine, with most of the neutrons produced at the sloshing-ion turning points near the mirrors (see figure).
- Magnetohydrodynamic (MHD) stability is produced by the outflow of dense plasma through one of the mirrors into a magnetic cusp, with the flow momentum in the good-curvature part of the field dominating the plasma instability drive in the bad-curvature part of the confined volume.
- Calculations and extrapolation of experiments indicate that 1–2 MW/m^2 of uncollided 14-MeV neutrons can be produced in a 100-L volume with small tritium consumption (<150 g/year at 100% availability) and intrinsically steady-state operation.
- Positive-ion-based neutral beams (<65 keV, 60 MW) can be used at mirror fields of 13 T (mirror ratio 10); upgrades to 20 T and 250-keV negative-ion-based neutral beams could produce as much as 4 MW/m^2 of 14-MeV neutrons.



Schematic of GDT based on neutron source: (1) expander vacuum chamber; (2) plasma absorber; (3) superconducting part of the mirror coil; (4) water-cooled part of the mirror coil; (5) one of the coils of superconducting solenoid; (6) shield; (7) vacuum chamber of a mirror coil; (8) zone of a moderate neutron flux; (9) zone of a high neutron flux; (10) neutron reflector. The plots above show the relative magnetic field β and the relative dependence of neutron flux F_N, on the position. *Source*: D. D. Ryutov, *Plasma Phys. Controlled Fusion*.

Status

- An experiment operating at the Budker Institute of Nuclear Physics (BINP), Novosibirsk, Russia, has demonstrated much of the GDTNS physics with a pulse length of 3–5 ms: stability against MHD modes, stability against ion velocity-space modes so that the ions decay and scatter classically, and T_e =130 eV (equal to classical predictions for the available neutral beam power; 3.5 MW at 15 keV).
- The device is routinely operated at plasma beta of 30%, without any signs of gross MHD instabilities. The cross-field transport is nonclassical, but the diffusion coefficient is sufficiently small (for the neutron source applications), less than $5 \times 10^{-3}(cT_e/eB)$.
- A hydrogen prototype of the neutron source has been designed, the experimental hall has been prepared at the BINP (Novosibirsk), and approximately 50% of hardware has been manufactured in the years 1993–1997. A decline in funding has stopped this activity. With $2M annual funding, this device could be brought into operation by the year 2001. A hydrogen prototype is a full-scale copy of a real neutron source. Its low cost is determined by two factors: the pulsed (0.2–0.5 s) mode of operation and the absence of radioactivity.
- A preconceptual design of the neutron source has been carried out by BINP in collaboration with the Efremov Institute (St. Petersburg, Russia), All-Russian Institute of Technical Physics (Snezhinsk, Russia), and Forshungszentrum Rossendorf (Germany). Issues of tritium handling and neutron shielding have been studied in great detail. The systems for installing and removing the samples from the test zone have been designed.

Current Research & Development (R&D)

R&D Goals and Challenges

- Research on the GDTNS is being carried out on GDT physics and its application to a neutron source at BINP. This work builds on previous results from there and elsewhere in the world, including on past mirror machines in the United States. The experiment has a mirror-to-mirror length of 7 m, a plasma radius at the midplane of 0.1 m, a magnetic field at the coils of 15 T, and a plasma density up to 10^{20} m^{-3}.
- The increase of the magnetic field in the midplane from the present level of 0.22 to 0.5 T has been planned. This would allow considerably improved performance of the device and would broaden the domain for deriving necessary scaling laws. However, the lack of funding (~$300K for the upgrade) has stopped this activity.

Budget

There are presently no U.S. funds being spent on the GDTNS. The funding to the BINP is low, allowing slow progress on the experiment.

Anticipated Contributions Relative to Metrics

Metrics

Plasma experiments need to demonstrate the full operating regime; magnetic field strength in the mirrors at 13 T, magnetic field ratio of 10, plasma density at $>10^{20}$ m^{-3}, electron temperature at >0.5 keV, and neutral beam injection at 65 keV at 30°–45° to the magnetic field at powers extrapolatable to 60 MW. The sloshing-ion distribution must remain stable in the high power system, as predicted by theory.

Near Term ≤5 years

Complete the construction of the "Budker hydrogen prototype." Assistance from the United States in equipment, personnel, or construction funds would greatly speed construction and give the United States a relatively inexpensive opening to explore the development of this neutron source. An international collaboration could draw on the extensive experience at the BINP; together with the U.S. construction capabilities, this would provide a strong basis to evaluate this option for fusion testing.

Midterm ~20 years

Build a GDTNS and apply it to the neutron testing and development of materials, components, and reactor systems.

Long Term >20 years

Use GDTNS to test and develop advanced reactor materials and reactor systems, for example, for a DEMO or power reactor.

Proponents' and Critics' Claims

Proponents claim that most of the physics has been demonstrated for the GDTNS, although experiments at higher power are required to verify the extrapolation of the electron temperature to the neutron source regime. The proposed neutron source can test materials, reactor components, and many of the blanket systems and subsystems needed for a reactor. Tritium usage is low enough to be supported from supplies elsewhere in the world, without local production. Construction is relatively straightforward, and the cost is reasonable (~$500M) for a large 14-MeV neutron source. This appears to be one (perhaps the only) path to a true plasma-based 14-MeV neutron source that can be developed in the near future. The resulting neutron flux could also be used in spin-off applications, which would benefit from a 14-MeV primary energy.

Critics claim that the electron physics has not been demonstrated at the required temperature, that the Q is small resulting in low power efficiency, and that the mirror physics and technology are deadend developments for fusion power.

M-13. PLASMAS WITH STRONG EXTERNAL DRIVE

Description

Particle distributions driven, for example, by beams and including the effects of nuclear polarization can provide certain benefits in magnetic fusion devices. Beams of ions, colliding at energies near the peak in their fusion cross section, lead to a higher Q than a thermal distribution of the same mean energy (see Fig. 1 opposite). One important gain is an increase in fusion reactivity at higher energies. This increase may be in the form of a nuclear resonance; hence, such distributions are far better than simply hotter plasmas. The dramatic success of the Tokamak Fusion Test Reactor's (TFTR's) two-component supershot program is one concrete experimental example, displaying not only enhanced reactivity but also improved confinement. Other benefits have been predicted, such as the conversion of fusion products' energy into plasma current, the "alpha channeling process." However, it has also been long recognized that such strongly driven systems may be prone to instabilities or that there may be practical or fundamental difficulties in producing or maintaining the desired distribution. Determining what precise departures from simple thermal distributions are desirable and maintainable merits an extensive research program because of the large magnitude of the potential benefits. The benefits and needs are greatest for high-beta magnetic fusion energy (MFE) devices, for example, the field-reversed configuration (FRC), spheromak, and spherical tokamak (ST). Because of its demonstrated very high beta and potential for direct electrical conversion of the exhaust, the FRC is particularly interesting as a candidate to burn aneutronic fuels (see Fig. 2 and M-20. Alternate Fuels).

Status

In beam-heated tokamaks, the fuel ions are Maxwellian, the electrons are a drifted Maxwellian, and the beam particles have a slowing down distribution. It is an experimental fact that operational modes can be found where the beam particles slow down and diffuse classically while the thermal particles experience anomalous transport. Other energetic particles, such as runaway electrons and radio-frequency (rf)-heated ions, typically strongly non-Maxwellian, have significantly better confinement than the Maxwellian particles in the thermal range of energies. The experimental data base on improved confinement for non-Maxwellian particles is solid and attractive, but the fundamental understanding needs improving for optimization in other configurations.

No similar experiments have been performed on FRCs or spheromaks because of the short-duration pulsed nature of their operation, previous heating methods, relatively high collisionality, and the more limited research programs. First theoretical analyses of colliding beam distributions in FRCs have been carried out, including stability studies. Beam- (and rf-) heating experiments on STs are only now commencing. The theoretical underpinnings of alpha channeling theory are strong. Preliminary alpha channeling experiments were performed on TFTR with mixed results. Definitive comparisons between experiments and theory are still lacking.

Current Research and Development (R&D)

R&D Goals and Challenges

In the United States, there is only a vestigial theoretical research program on alpha channeling and no experimental program whatsoever. There is no Department of Energy (DOE)-supported experimental or theoretical research program on enhancing reactivity by maintaining strongly driven ion distributions.

If an R&D program were to be funded, the major goals would include understanding of the four following issues:

- the fundamental limitations to producing and sustaining non-Maxwellian distribution,
- the stability limits of allowed distributions in the high-beta devices,
- the enhancements to reactivity possible with the attainable distributions, and
- the efficiency for converting non-Maxwellian energy distributions into driven currents.

In Russia, there is research on a beam-driven mirror system for use as a 14-MeV neutron source for materials irradiation testing (see M-12. Gas Dynamic Trap).

Anticipated Contributions Relative to Metrics

Metrics

The principal goals are to develop and test the basic concepts that will allow the most benefit to be derived from non-Maxwellian distributions.

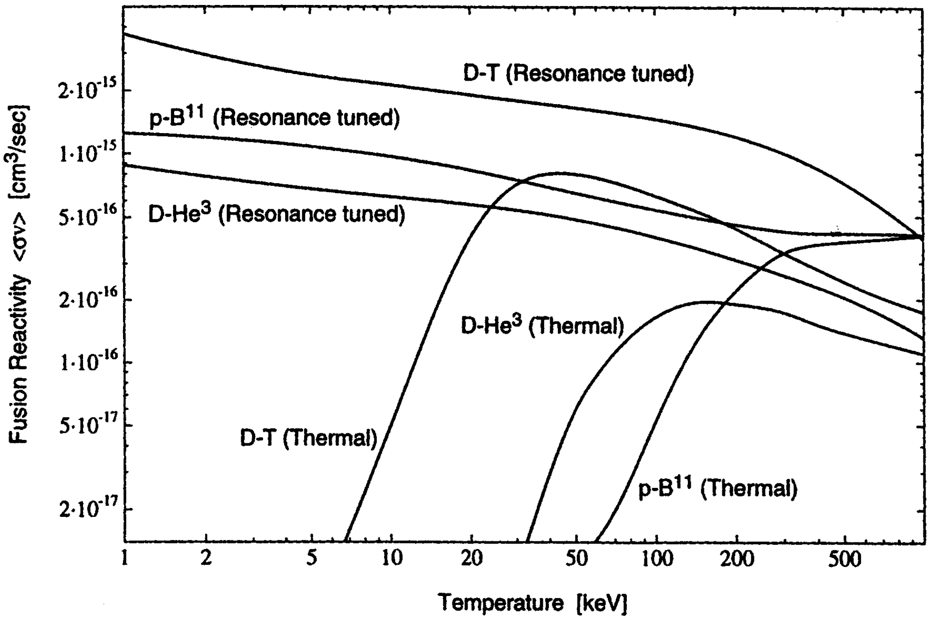


Fig. 1. Various fusion reactivities.

Near Term ~5 years

- Develop and test techniques to divert alpha power of an ST reactor into current drive.
- Develop and test concepts and methods to maintain non-Maxwellian distributions, including beam and rf approaches.
- Perform pulsed experiments in an FRC to test colliding-beam concepts.
- Perform pulsed experiments in an FRC to test rf concepts.

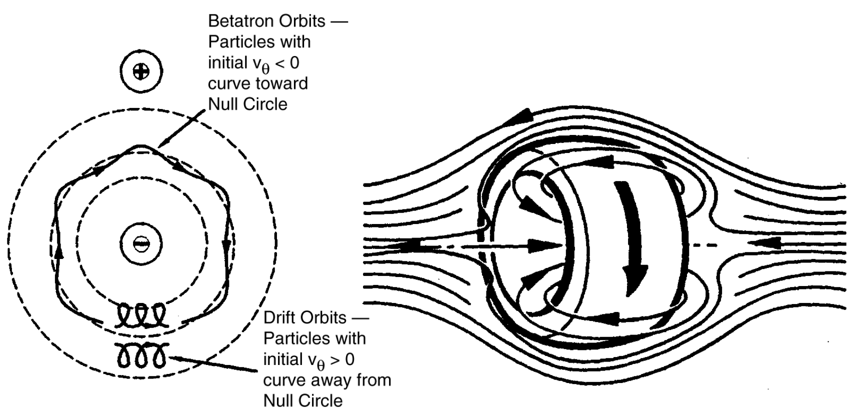


Fig. 2. FRC with typical particle orbits.

Midterm ~5–20 years

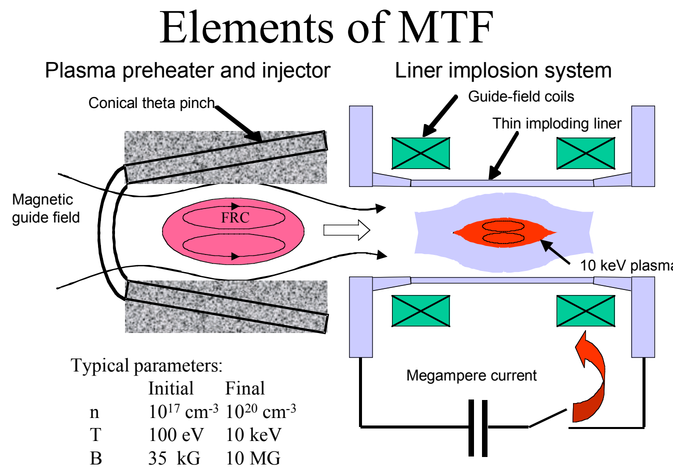
Research with periodic state experiments at low power levels and development of a 100-MW prototype.

Proponents' and Critics' Claims

Proponents claim that the benefits of strongly driven plasmas are essential to clean and economic fusion and that the basic aspects of maintaining strongly driven distributions have not been properly examined. Critics claim that a rapid Coulomb scattering rate will not allow such plasma systems to be maintained long enough for benefits to accrue and that technologies to provide the drive are too large, expensive, unreliable, and cumbersome to be practical.

Description

In the magnetized target fusion (MTF) approach to fusion, deuterium-tritium (D-T) fuel in a preheated magnetized target plasma is rapidly compressed to thermonuclear temperatures by pdV heating. For example, the figure shows compression using a metal liner imploded with high velocity (typically a few kilometers per second), resulting from the self-pinching of megampere currents. Operating in a fuel density and time scale regime intermediate between magnetic confinement fusion and inertial confinement fusion has the advantage of allowing orders of magnitude smaller system size compared with conventional magnetic fusion. Magnetic insulation has the potential for orders of magnitude reductions in power requirements compared with conventional inertial confinement fusion. If this avenue to low-cost energy-producing plasmas is successful, MTF permits fusion development without billion-dollar facilities, thus circumventing one of the most serious obstacles to fusion development.



Status

A number of promising target plasma configurations have been identified. Current emphasis on the field-reversed configuration (FRC) derives from 20 years of compact torus research, which provides a good understanding of FRC equilibrium, compressional heating, and confinement scaling. Present understanding projects to near break-even parameters with available pulsed-power facilities. In recent years liner implosion technology, with the required ~10-MJ energy and ~10-km/s velocity needed for MTF, has been developed by Defense Programs (DP) in the Department of Energy (DOE) and the Department of Defense (DOD), and facilities are available to allow very cost-effective tests of the MTF concept. In May 1998 a national team of six institutions led by Los Alamos National Laboratory (LANL) proposed a 3-year, $6.6M/year proof-of-principle (PoP) test of MTF to DOE. The proposal received favorable peer review, and the Fusion Energy Sciences Advisory Council (FESAC) recommended the team be kept in place while FESAC conducted a review of overall program balance. Since the proposal, the team has grown and now includes LANL, Air Force Research Laboratory, Lawrence Livermore National Laboratory, General Atomics, University of Washington, Westinghouse, Massachusetts Institute of Technology, University of California—San Diego, and University of California—Berkeley.

Current Research and Development (R&D)

R&D Goals and Challenges

The main PoP goal is to establish whether heating to thermonuclear temperature is possible with liner compression. Scientific issues include (1) achieving the relatively high initial density (~10^{17} cm^{-3}) and sufficient temperature (~300 eV) needed for a target plasma, (2) stability of liner and plasma during compression, and (3) wall-plasma interactions throughout the process. For practical generation of electricity, the major issue is cost of material that must be processed for each pulse (the “kopeck” problem). Low-cost refabrication of electrical leads or methods of stand-off power delivery are being studied.

Related R&D Activities

Research complements and depends upon ongoing compact torus research, pulsed-power liner implosion work, and development of inertial fusion energy (IFE) power-plant energy technology. Both inertial confinement fusion (ICF) and MTF involve containment of pulsed fusion energy (multi-megajoule to gigajoule) with plasma-facing liquid walls of fluorine-lithium-beryllium molten salts (Flibe) or lithium.

Recent Successes

In recent years liner implosion experiments have demonstrated the symmetry, kinetic energy, and velocity needed for MTF. Integrated plasma formation and liner-on-plasma experiments have not been done since the 1970s. Compact torus research in the 1980s and 1990s has developed improved methods for plasma formation and preheating.

Budget

FY 1998: $115K; FY 1999: $1M; and FY 2000 (needed): $6.6M.

Anticipated Contributions Relative to Metrics

Metrics

- **Energy concepts**—The long-term application of MTF to energy production has not been examined as extensively as for conventional magnetic or inertial fusion, and metrics are less well defined. For pulsed systems like MTF and ICF, the product of gain and efficiency enters strongly into economics. MTF is expected to operate with smaller gain but higher efficiency than ICF, and the product will be an important metric for the research program. With MTF, yields in the gigajoule range would allow advantages at a lower repetition rate than conventional ICF. Several energy approaches are being studied. Pulsed compression with circulating liquid metal similar to the early LINUS concept is one approach. Low-cost refabrication of electrical leads that deliver power through a liquid blanket as proposed in the 1978 Conceptual Fast Liner Reactor Study is another. Stand-off delivery of power by efficient lasers, ion beams, or electron beams is a third. A completely different approach to power conversion might be possible if neutron energy were used to flash vaporize the blanket and the 1- to 2-eV vapor were then used for magnetohydrodynamic (MHD) generation of electricity.
- **Science**—The intermediate density regime, which differs by 5 to 6 orders of magnitude from both MTF and ICF, allows many tests of scientific understanding. New insights into the physics of FRCs should result from MTF compression experiments that seek kiloelectron-volt temperatures and much higher density but still work with similar values of dimensionless parameters, such as size relative to ion gyroradius compared with present and most previous FRC experiments. Bohmlike turbulence is observed in solar wind studies of an FRC-like field-reversed current sheath, and MTF is expected to work with a comparable level of turbulence. Generation of magnetic fields and magnetic reconnection at high temperature and high beta is expected in MTF experiments and is also seen in astrophysics. Improved understanding of wall-plasma interactions through theory, computational modeling, and experiment is a major goal for MTF. This understanding may have application to tokamak divertors, plasma processing, and radiative-condensation instabilities in beta »1 astrophysical plasma.

Near Term ≤5 years

A PoP test appears possible in less than 5 years if funding is made available. Assuming success with the PoP and an increased level of funding, a near break-even test could be done in about 2003–2005 using the Los Alamos ATLAS pulsed-power facility being constructed by DOE–DP. An interesting aspect to MTF is that university-scale experiments can fully test MTF targets, and the community-based MTF research program assumes a multi-institutional campaign of testing targets developed on small-scale experiments in the large-scale defense program facilities. Success in the laboratory would give strong incentive for expanded work on technologies needed for economic energy production.

Midterm ~20 years

From a development perspective, MTF can be viewed as a broad class of possibilities that are characterized by low cost and pulsed operation. Possible MTF embodiments range from FRC or spheromak target plasmas to a class of Z-pinch-like wall-confined plasmas as represented by the Russian MAGO configuration. Heating is possible with liner-driven implosions or stand-off laser-beam or particle-beam drivers with reduced power and intensity requirements compared with ICF. Development can proceed rapidly because the necessary scientific studies (including burning plasma physics) require no billion-dollar-class facilities.

Long Term >20 years

If MTF is successful, the development phase for fusion energy should be significantly accelerated, and the ultimate cost of electricity should be reduced in accord with reduced development costs.

Proponents' and Critics' Claims

Proponents are excited because MTF offers an affordable path to burning plasma experiments and an intriguing and generally unexplored possibility for practical fusion energy. In the restructured fusion program that emphasizes finding low-cost development paths for fusion, MTF is a logical new element. In addition, MTF strengthens the fusion portfolio because it represents a qualitatively different approach compared with conventional magnetic and ICF approaches. So far no physical limitation has been identified that precludes developing MTF as a practical fusion energy system, and several promising development paths have been identified.

Critics argue that pulsed systems like ICF and MTF are unlikely to meet the practical requirements for pulse repetition rate and cost per target, especially in the case of MTF if it involves replacement of liner hardware on every pulse. There are also technical concerns that high-Z liner material will mix rapidly with the relatively low-density fusion fuel, leading to unacceptably large radiation losses. Some have expressed concern that pulsed fusion approaches like MTF might lead to new types of nuclear weapons. However, scientists who analyzed this possibility say, "We see no immediate danger of a militarily attractive new type of weapon being developed from the current unclassified research programs on pure-fusion explosions." [S. Jones, R. Kidder, and F. Von Hippel, *Physics Today*, September 1998, p. 57.]

Description

The boundary plasma is the interface between the hot core plasma and the material walls of the surrounding vacuum vessel. It constitutes a buffer zone that protects the walls from the hot plasma and shields the plasma core from impurities originating at the walls. Access to most of the improved plasma confinement modes has been achieved through the application of wall conditioning techniques as depicted in Fig. 1, which shows the effect of lithium wall conditioning on plasma performance in the Tokamak Fusion Test Reactor (TFTR). The so-called Supershot regime, depicted in this figure, is attained through reduction in hydrogen and carbon recycling from the first wall in TFTR. Inside the separatrix, the plasma processes involve transport phenomena, magnetohydrodynamic (MHD) effects, transition physics, and atomic physics. Between the separatrix and the wall, the scrape-off layer (SOL) is dominated by atomic physics and by perpendicular and parallel transport processes. The wall surface plays an important role in the recycling of hydrogen isotopes, determining plasma fueling and wall inventory (of tritium). Plasma-surface interactions lead to impurity generation and to wall erosion. Transport of particles and energy from the plasma core to the divertors or limiters is discussed separately.

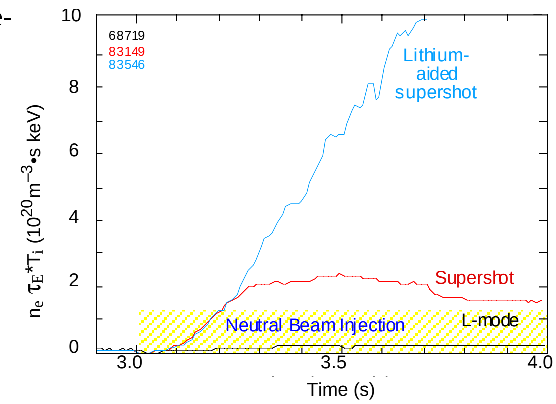


Fig. 1. Effect of wall conditioning in TFTR.

Status

Much progress has been made in characterizing the plasma boundary properties under a variety of conditions. Inside the separatrix, transport processes and their transition due to the formation of transport barriers (i.e., from L-mode to H-mode) have been characterized and correlated with changes in boundary plasma parameters. We are beginning to understand the interplay of parallel transport phenomena such as convection, conduction, and plasma flows in the SOL. A very prominent part in the plasma boundary program has been the development of wall conditioning methods to control plasma-surface interactions for optimum plasma performance. Experience with tritium retention in the walls of existing devices and the development of models such as REDEP and BBQ have increased our understanding of the underlying processes and highlighted the urgent need to develop efficient tritium removal methods. Transport processes in the boundary plasma have been characterized and described with complex models such as B2-EIRENE, DEGAS, or UEDGE. These models include the main working gas, hydrogen, as well as the prominent intrinsic or extrinsic impurities. However, currently there is no reliable physics-based model for the turbulent cross-field transport rates of heat and particles. Instead, these codes employ adjustable transport coefficients that are fitted to match experiments. Deliberately introduced impurities, such as neon, radiate predominantly in the plasma edge and are being used to create a stable radiating boundary for even distribution of exhaust power.

Current Research and Development (R&D)

R&D Goals and Challenges

- Establish detailed causal correlations between edge transport barriers and plasma edge parameters.
- Investigate stable radiating boundary plasmas as a mode of operation for fusion reactors.
- Provide an experimental database for further developing and coupling materials codes and plasma edge codes.
- Develop new wall conditioning techniques for high-performance plasmas applicable to long-pulse devices.
- Develop ways to avoid peaking of heat flux in space and time [edge-localized modes (ELMs) and disruptions] to permit the use of noncarbon materials, a necessary step for a tritium self-sufficient reactor.
- Develop empirical scaling relations between cross-field transport and local/global transport parameters.
- Apply knowledge base of plasma-surface interactions to materials processing and other nonfusion activities.

Related R&D Activities

- Strong international collaborations exist in all areas of boundary plasma physics. For example, wall conditioning techniques developed at TEXTOR are being applied and tested on most major machines.
- The Virtual Laboratory for Technology is developing high heat flux materials for plasma-facing components.

Recent Successes

- Stable radiating edge layers have been sustained with good confinement and low core impurity content.
- The role of neutrals in the plasma edge of the Doublet III-D (DIII-D) has been shown to play a role in the L-H transition.
- Tritium retention in beryllium has been shown to be much lower than expected at reactor conditions.

Anticipated Contributions Relative to Metrics

Metrics

The boundary plasma has to provide the buffer between core plasma and wall. In this capacity, it must fulfill a variety of tasks: (1) maintain strong gradients just inside the separatrix to support transport barriers, (2) maintain a stable radiating mantle with low core impurity concentrations, (3) shield the main plasma from wall impurities, (4) provide sufficient parallel transport for particle and power removal in the divertors or limiters, and (5) provide adequate perpendicular transport for sufficient radial power flux distribution. The plasma-surface interface has to be designed, with respect to materials choice and plasma parameters, for (1) minimum impurity production and erosion, (2) controllable wall recycling for plasma fueling and inventory control, and (3) minimum tritium inventory.

Near Term ≤5 years

It is known that wall conditions play an important role in establishing transport barriers and improving plasma performance, but the underlying physics phenomena that connect wall conditions with transport barrier formation are not well understood. Therefore, additional efforts are needed:

- Investigate the detailed atomic and molecular physics of hydrogen and impurities in the plasma boundary inside and outside the separatrix.
- Study techniques for establishing stable radiating boundary plasmas.
- Establish clear causal correlations between edge plasma parameters and transport barriers.
- Investigate the effects of impurities and neutrals on the formation of edge transport barriers.
- Develop and implement with dedicated machine time in-situ time-dependent diagnostics to provide the detailed database necessary for understanding plasma material interactions.

The database and modeling tools developed in the area of plasma-surface interactions should be adapted for broader use in nonfusion applications. Examples of nonfusion applications include materials processing, flat panel plasma display, and plasma spray coating.

Midterm ~20 years

- Build a new long-pulse machine with detailed wall/edge diagnostics dedicated to exploring long-pulse issues such as wall saturation, erosion, codeposition, and dust generation that are not significant in existing machines but pose challenges that will have to be solved for a high-duty-cycle fusion reactor.
- Develop predictive understanding of physical and chemical properties of mixed materials generated by plasma-surface interactions.
- Develop techniques to control wall erosion.
- Develop procedures for sustainable tritium inventory.

Long Term >20 years

- Develop boundary plasma control techniques for minimum wall erosion, manageable tritium inventory, control of transport barrier properties, and good helium exhaust.
- Build on the scientific and engineering understanding gained in the previous 20 years to design and build an attractive fusion reactor that incorporates solutions to the long-standing issues of the first wall.

Proponents' and Critics' Claims

Proponents claim that the advances made in boundary plasma control over the past two decades have been a key contribution to the success of magnetic fusion energy research. Further accomplishments, needed for the full realization of magnetic confinement, can be predicted from reasonable extrapolation of past success.

Critics claim that the boundary plasma has traditionally been viewed in the realm of "kitchen physics," that is, governed by the application of purely empirical techniques that lack a firm scientific foundation. Extending high-performance discharges from the present second time scale to hours or days poses insuperable materials problems.

M-16. BURNING PLASMA SCIENCE

Description

Fusion energy is released by burning light elements using nuclear reactions that consume mass and release large amounts of energy in the form of extremely energetic charged particles or neutrons. The most reactive fusion fuel is a 50/50 mix of deuterium (D) and tritium (T) that requires fuel temperatures of ~100,000,000°K and the product of the fuel density n and energy confinement τ_E such that the Lawson product $n\tau_E > 2 \times 10^{20}$ m^{-3} s. The energetic charged particles (3.5-MeV alphas for the D-T reaction) are confined by the magnetic field and deposit their energy in the plasma. Other reactions are possible, but higher temperatures and better confinement are required, ultimately resulting in much lower fusion power density for a given plasma pressure. In magnetic fusion, plasmas are heated to reaction conditions using external auxiliary power (Paux) and produce fusion power (Pfusion). The most fundamental metric is the fusion gain, Q = Pfusion/Paux. Magnetic fusion reactors will require Q ≥ 25 to be economically attractive. With Lawson product only ~20% higher, “ignition” is obtained, where the plasma is self-sustained purely by its alpha particle heating. The science of burning plasmas consists of (1) the physics of magnetic confinement in the “dimensionlessly large” reactor regime [transport, magnetohydrodynamic (MHD) stability and edge plasma parameters]; (2) the behavior of the plasma in the presence of energetic alpha particles (alpha confinement, induced instabilities, and energy transfer to the bulk plasma); (3) dynamic control of the self-heated plasma; and (4) power and particle exhaust, especially, removal of helium ash.

Status

Burning plasma science has been studied initially with deuterium plasmas that have provided understanding of plasma behavior near burning conditions and have allowed the first studies of energetic particle behavior at near burning plasma conditions. A series of experiments using 50/50 D-T fuel has been carried out on the Tokamak Fusion Test Reactor (TFTR) and the Joint European Torus (JET), which have produced fusion powers of 11 to 16 MW and Q values of 0.3 to 0.6 for durations of about 1 s. These experiments have confirmed many of the physics models for burning plasmas, but experiments are needed to study the science of a strongly burning plasma (Q ≥ 10) for longer time durations.

Current Research and Development (R&D)

R&D Goals and Challenges

- Obtain and explore controlled fusion burning plasmas.
- Investigate plasma confinement phenomena in the reactor regime of a large size relative to gyro radius, in an experiment that integrates confinement and stability in the core and edge plasma.
- Explore and understand the phenomena associated with reactor levels of fusion alpha-particles in a magnetically confined plasma, especially the instabilities that may be excited by their presence.
- Establish the practical feasibility of controlled burn in a self-heated burning plasma through passive and active influences on the overall power balance.
- Explore the compatibility of self-heating with the pressure and current profiles required for optimal stability, transport, and steady-state (long-pulse high-duty cycle) sustainment.
- Establish the practical feasibility of power and particle (especially helium) exhaust.

These goals reflect the progress that has been achieved in many topical areas during the past. However, no existing facility possesses sufficient confinement to take this next step to a burning plasma experiment that addresses these goals.

Related R&D Activities

- Inertial fusion burning plasma studies such as those proposed for the National Ignition Facility (NIF).
- Stellar dynamics (e.g., the marginal stability model for the sun).
- Fusion program research and technology in support of obtaining a burning plasma.

Recent Successes

- Theory, modeling, and experiments on energetic particle-induced instabilities.
- Theory, modeling, and experiments on transport and MHD (e.g., disruptions).
- Development of comprehensive diagnostics (J, Er, alphas, ...) for burning plasma experiments.
- Confirmation of the physics of a weakly burning D-T plasma and production of significant fusion power.

Budget

The next step to an experimental burning plasma requires a capital construction project at least of the magnitude of one of the major fusion facilities (TFTR, JET, JT-60) built in the 1970s. However, the U.S. fusion budget at its current level is insufficient to support such a project domestically. Refer to M-17. Burning Plasma Experimental Options for more details.

Anticipated Contributions Relative to Metrics

Metrics

- Science
  - Enter and diagnose the regime where alpha heating dominates the power balance and defines the plasma pressure profile $Q \geq 10$ and preferably $Q \rightarrow \infty$.
  - Determine the transport properties, MHD stability, and edge-plasma characteristics of a reactor-size plasma.
  - Demonstrate quantitative predictive understanding of alpha particle dynamics, both single particle and collective effects, in the strongly burning plasma regime.
  - Demonstrate global thermal stability at high Q.
  - Demonstrate high bootstrap-fraction burning plasma operation.
  - Demonstrate power exhaust and alpha ash removal for at least ten energy confinement times.
- Energy
  - Demonstrate scientific feasibility of high-Q D-T controlled fusion reactions using magnetic confinement.
- Technology
  - Integrate safe handling of tritium into a burning plasma environment.
  - Demonstrate fueling and heating technologies for a reactor-scale plasma.
  - Establish plasma-facing component technologies for reactors.
  - Demonstrate remote handling technologies.

Near Term ≤5 years

- Possibly further ($Q \sim 1$) experiments on JET briefly in 2002.
- No alpha-dominated experiments are possible using the existing magnetic fusion energy (MFE) facilities.
- Carry out design activities and related experimental investigations on prototypical facilities to establish optimal configurations and expectations for a next step burning plasma experiment.
- Develop the engineering basis for a next step burning plasma experiment.
- Develop a scientific consensus worldwide in support of the burning plasma physics mission.

Midterm ~5 to 20 years (2004 to 2019)

- Explore burning plasma physics on an alpha-heated magnetically confined plasma.
- Demonstrate efficient alpha energy transfer to bulk plasma with alphas providing at least 65% of heating power ($Q > 10$).
- Demonstrate sustained burn for at least five energy confinement times.
- Respond appropriately to a decision in ~2000 by the international participants on the construction of the International Thermonuclear Experimental Reactor (ITER).

Long Term >20 years

- Demonstrate steady state burning plasma at $Q > 25$ and provide the burning plasma science basis for an MFE reactor.

Proponents' and Critics' Claims

Critics claim that present MFE concepts, especially the tokamak, do not project to reactor systems with economics competitive with natural gas, fossil fuels, and fission. High Q plasmas based on today's most advanced magnetic configuration cannot be controlled at advanced performance levels, and the plasma may constantly disrupt and result in a reactor concept with low reliability. The science of the final MFE reactor may be quite different from the concepts investigated during the last 50 years, and the burning plasma science developed in today's magnetic concepts may not be generic.

Proponents state that the leading MFE configurations (tokamak, stellarators and spherical torus) project a power plant cost of electricity within a factor of 2 of existing fission power plants if a high Q plasma can be sustained with high reliability. Tokamaks are sufficiently advanced that burning plasma physics can be accessed with a modest extension of existing science and facilities. Thus far, there appear to be no technical showstoppers; therefore, the burning plasma physics step can be taken with reasonable assurance. Burning plasma physics should first be demonstrated to establish scientific feasibility and to understand the scientific issues of a strongly burning plasma, many of which are generic to all MFE reactor concepts. Subsequently, the magnetic configuration can be optimized to develop an attractive MFE reactor with favorable economics and high reliability.

M-17. BURNING PLASMA EXPERIMENTAL OPTIONS

Description

The options for a next step burning plasma experiment are defined by the overall strategic pathways available for the development of magnetic fusion energy (MFE) as shown in Fig. 1. The one step to demonstration reactor (DEMO) would be undertaken by a single large facility, such as the International Thermonuclear Experimental Reactor (ITER), with multiple missions of developing and integrating burning plasma physics, long-pulse physics and technology, and fusion technologies. The enhanced concept innovation pathway would delay burning plasma experiments to emphasize first, experimentation on, for example, stellarators, spherical tori, reversed field pinches, spheromaks, and multipoles. However, none of these configurations could be ready for a burning plasma test at $Q \geq 10$ in less than a decade. The modular pathway employs multiple facilities each focused on resolving a key MFE issue at conditions approaching those expected in an MFE system. The modular pathway addresses the key technical issues, high-Q burning plasmas and steady-state operation, separately.

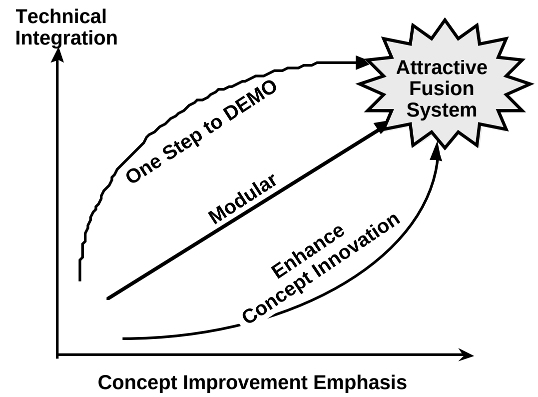


Fig. 1. Pathways for the Development of Magnetic Fusion

Status

The burning plasma issues for MFE are discussed in M-16. The status of the leading magnetic configurations to address MFE burning plasma issues is given in terms of the extrapolation required from parameters achieved in laboratory experiments to those required in that concept's reactor assessment by the Advanced Reactor Innovation and Evaluation Studies (ARIES) (Table 1).

Table 1. Required extrapolation

	Tokamak	Stellarator	Spherical torus	ARIES
$n\tau_E T_i$	10	1,000	100,000	1
Plasma pressure	3	100	>100	1
Neutron wall load (MW/m^{-2})	50	>1,000	>1,000	1
Duty cycle	50	>1,000	>10,000	1

Only the tokamak is sufficiently advanced to address the burning plasma physics issues of MFE within the next decade. The crucial issues of understanding the science of plasma transport and magnetohydrodynamic (MHD) stability in advanced configurations where the profiles are defined by alpha heating can be studied thoroughly in the tokamak configuration, and this knowledge can be used to understand and predict burning plasma physics phenomena in other magnetic configurations. A next step MFE experiment capable of achieving $Q \geq 10$ in deuterium-tritium (D-T) plasmas would serve both as proof of performance and as a facility to explore, understand, and optimize burning plasmas for MFE in parallel with the National Ignition Facility (NIF)/LMJ experiments for IFE.

Current Research and Development (R&D)

R&D Goals and Challenges

Previous design studies of next step burning plasma experiments (TFCX, CIT, BPX, and ITER) have all produced technically credible designs but have not garnered the required scientific and financial support to proceed with construction. The challenge is to develop a design proposal with a more focused mission that will address the critical burning plasma issues within a constrained budget profile.

Related R&D Activities

The base theory, modeling, and confinement program will interact closely with the burning plasma experiment. Enabling technology development in plasma heating, current drive, and fueling (pellet injection) will be needed for the burning plasma experiment. Fusion technology development (especially tritium handling and remote handling) will be integrated with the experiment.

IFE burning plasma experiments on NIF/LMJ will be complementary to the MFE burning plasma experiment(s).

Recent Successes

Recent experiments and the ITER design study have produced a well-documented physics basis for analyzing burning plasma performance.

Budget

IGNITOR is viewed as an Italian project with potential European Community (EC) support. ITER-RC is viewed as a Japanese, European, and Russian project with potential U.S. support. FIRE is viewed as a U.S. project with potential international support (Table 2). The construction budgets for the representative next-step options are estimated in Table 3.

Anticipated Contributions Relative to Metrics

Metrics

Table 2. Extrapolation required

	IGNITOR	FIRE	ITER-RC	ARIES
$n\tau_E T_i$	~1	~1.5	<1.5	1
Plasma pressure	~0.6	~0.8	2	1
Neutron wall load (MW/m^{-2})	~1	~1	8	1
Duty cycle	>1000	>1000	~10	1

In the modular pathway the duty cycle metric will be addressed in a separate steady-state advanced toroidal facility.

Near Term ≤5 years

- Comprehensive technical assessment of all approaches to fusion and identification of key metrics (1999).
- Performance optimization and cost reduction, design activities with supporting physics, and technology R&D.
- Proposal ready for technical review and decision by end of 2000 in concert with international decision on ITER.

Several representative options for a next step burning plasma experiment in MFE have been identified during the next step options study that followed the Madison Forum with parameters in the ranges illustrated in Table 3.

Table 3. Options with parameters

	R(m)	B(T)	Coils	Ip (MA)	Gain	Pfusion (MW)	Exhaust	Burn time (s)	Cost ($M)
IGNITOR	1.32	13	30°K Cu	12	>10	~200	Limiter	5	<500
FIRE	~2.0	~10	70°K BeCu	~7	~10	~200	DND	≥10	<1000
ITER-RC	6.2	5.5	NbSn S/C	13	10	~500	S-DND	≥400	<6000

ITER-Reduced Cost (RC) is very similar to the ITER EDA with the same overall program objective—to establish the scientific and technological feasibility of magnetic fusion—but with slightly reduced size and performance to reduce the construction cost by 50%. ITER-RC would have superconducting coils capable of allowing up to steady state under driven plasma conditions. IGNITOR is a very compact high-field moderately shaped tokamak with cryogenically cooled copper coils. The plasma is heated to high Q by ohmic and ion cyclotron range of frequencies (ICRF), and the plasma power and particles are exhausted using the first wall as a limiter. The Fusion Ignition Research Experiment (FIRE) is based on previous U.S. compact copper-conductor burning plasma experiment designs (CIT, BPX, BPX-AT), but responds to recent tokamak physics developments. FIRE is a compact high-field tokamak similar to IGNITOR but with higher triangularity and a double-null closed-divertor configuration.

Midterm ~20 years

- Initiate construction of next step burning plasma experiment by 2002 with first operation by 2009 with high-Q D-T plasmas by 2012 (if compact high field) or first plasma by 2012 and high-Q D-T by 2016 (if ITER-RC).
- Make a major programmatic decision in 2015 to 2020 time frame on the selection of potential concept(s) for further development as an Advanced Integrated Experimental Reactor or for burning plasma tests of additional concepts.

Long Term >20 years

If successful, the key burning plasma issues would be addressed and resolved by 2020.

Proponents' and Critics' Claims

The tokamak is favored by a vast majority of the world MFE programs for a next step burning plasma experiment; the issue is whether the tokamak will lead directly to an economical reactor. The JA, EC, and Russian Federation fusion programs favor the one step to DEMO strategy, and it has been central to their official program plans. A majority of the U.S. fusion community (e.g., the Madison Forum) favor the modular pathway, which was the MFE pathway prior to the ITER initiative and is similar to the IFE pathway. The enhanced concept innovation pathway would delay initiation of a burning plasma experiment to develop the optimum magnetic configuration at small size and cost prior to large-scale testing. This approach will extend the time scale and possibly the cost if difficulties arise at the proof of performance and burning plasma phase with dominant alpha heating.

Critics claim that the cost of ITER and even ITER-RC is too high, indicating that the tokamak will not be a cost-effective power plant. They further dispute the general applicability of burning plasma science from the tokamak, particularly to the more self-organized plasma systems.

Description

Integrated Fusion Science and Engineering Technology Research integrates the physics of burning plasmas with engineering technologies needed to design a commercial demonstration fusion power plant, often called the demonstration reactor (DEMO). Integrating research in a single facility is often called one-step-to-DEMO strategy. This approach, with international collaboration, is the fastest and least expensive route to a commercial fusion power plant; it is also the most reliable because the physics data returned will require little extrapolation to DEMO. Several variations are now under active study, ranging from the Reduced Technical Objectives/ Reduced Cost International Thermonuclear Experimental Reactor (ITER-RC), to the less capable JT-60 Super Upgrade. These devices all support an operational deuterium-tritium (D-T) burning capability. Their superconducting magnets are essential to developing the plasma physics of burning steady-state tokamaks and will assure that investments in magnet technology will benefit DEMO. A second common feature, shown in Fig. 1, is a poloidal field coil set outside the toroidal field magnets with a segmented central solenoid. This coil arrangement offers flexibility consistent with the tokamak reactor concept and can support both single-null and double-null divertor operation as well as Advanced Tokamak (AT) research. Much of the technology data, such as high-heat-flux components, magnet fabrication, tritium breeding, neutral beam and gyrotron auxiliary power systems, as well as erosion of plasma-facing components, will be generic to all toroidal magnetic fusion energy approaches.

The broad international tokamak database serves as the baseline design for ITER-RC. This will project, with margin, to nominal D-T burning with $Q \geq 10$ and will permit investigations of fusion science issues such as β- and density-limits and their control as well as provide helium exhaust data and neutrons for technology assessments. The flexibility inherent in the PF system, the fueling system, and the auxiliary heating will support operation in advanced modes now foreseen as reactor compatible and provide AT design data for DEMO. With contemporary tokamak devices, it is impossible to simultaneously achieve reactor-like core plasmas and reactor-like edge plasmas. Thus, in any tokamak strategy, a ITER-RC class device is required to obtain the robustly reliable data needed for a steady-state AT with a D-T energy source to qualify as the design basis for DEMO/Advanced Reactor Innovation and Evaluation Studies (ARIES)-RS.

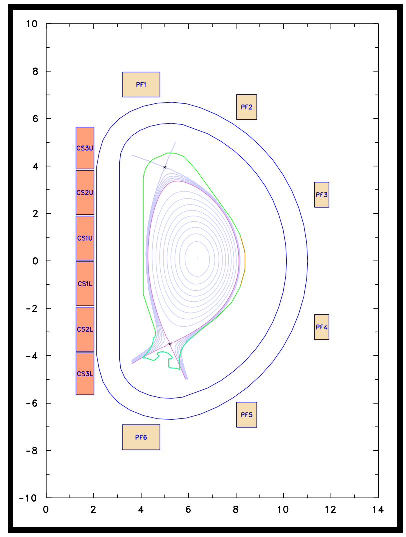


Fig. 1. A candidate ITER-RC design with AT capabilities.

Status

The July 1998 ITER Final Design Report (FDR) established the engineering feasibility of a ITER-RC-class facility with detailed, disruption-tolerant engineering solutions for all design elements: first wall, divertor, magnets, poloidal field control, and tritium inventory. The engineering design is supported by $800M of international engineering research and development (R&D) over 6 years with prototype testing. Key successes of the ITER/ engineering design activity (EDA) physics program were to establish common, scaleable physics and simulation codes for the critical physical processes: divertors, disruptions, β-limits, power thresholds, and confinement scaling. A 400-page issue of *Nuclear Fusion* documenting the physics basis for both ITER/FDR and ITER-RC will appear in June 1999. Now, an ITER-RC design program, incorporating staged construction and operation to level cash flow, is being developed by the Three-Party Joint Central Team. Cost reductions (~50%) are achieved by ~50% reductions in the fusion power, inductive pulse length, and plasma volume. The new designs have higher triangularity plasma shapes, achieve $Q \geq 10$ with conventional ELMy H-mode physics, and are capable of full ignition with AT operation.

Current Research and Development (R&D)

R&D Goals and Challenges

Demonstration of self-consistent profiles of α-heating, pressure, and bootstrap/driven current density as well as active control of $n = 1$ magnetohydrodynamic (MHD) modes are seen as crucial for steady-state operations at reactor β-values. For ELMy H-modes, increases in long-pulse β_N-values and operation at densities above the Greenwald value will increase the tokamak fusion power. Neoclassical tearing mode control by electron-cyclotron current drive and inside pellet launch fueling are respective proposed control techniques.

Anticipated Contributions Relative to Metrics

Energy Concept Metrics

The ITER-RC approach fulfills reactor-level metrics and also serves as a flexible test bed to improve β, density, reliability, fusion power, and wall-loading metrics, thus optimizing both the standard tokamak and ARIES-RS-like AT approaches to fusion power. Fusion power exceeding 500 MW for durations exceeding 500 s will enable nuclear testing.

Science Metrics

The tokamak science established by ITER-RC experiments will provide reactor-scale physics data not accessible to present machines. More generally, tokamak physics played a major motivating role in the development of the science of magnetized plasmas, which underlies all toroidal fusion concepts, as well as space, solar, and astrophysical plasma dynamics. Interdisciplinary examples include current sheets in reconnection and particle energization by Alfven waves in space and the galaxy. The science concepts are shared, but each field develops its own simulation codes and facilities. Computationally, tokamaks will remain the principal quantitative comparison between five-dimensional turbulence simulation codes and experiment. Tokamak MHD codes will be at the frontiers of simulating current sheets and other examples of spatial intermittency.

Nonfusion Applications

ITER/EDA technology development has already produced state-of-the-art advances. Model test coils will make available large-volume, high-field test facilities for superconducting magnet technology. High heat flux components are important to many high-temperature processes. Gyrotrons have applications in plasma and materials processing, and potentially in radars.

Near Term ≤5 years

Already the ITER/EDA project, through physics expert groups, has been very effective at determining the key areas where physics research will define the design basis and requirements for a reactor-scale facility. This has identified conceptually new processes such as the role of recombination in divertors. But much remains to be done, and the discipline associated with a design project is needed to assure that nominal performance requirements are met and flexibility is retained.

Midterm ~20 years

This time frame will see operation of the ITER-RC facility and attainment of reliable fusion burn at the 500–1000 MW level over pulses of at least 500-s duration. The case for the reality of a fusion energy option will be made! Experimental fusion physics will enter reactor-scale experiments where the balance between different physical processes will differ from that of present experiments, leading to original tokamak physics data. Reactor blanket technology demonstrations will benefit from the neutron fluxes with initial blanket testing.

Long Term ≥20 years

The operational data acquired from ITER-RC in this period will form the basis of reactor utilization of advanced and/or steady-state tokamak operations with internal transport barriers. The required robust demonstrations for DEMO will be carried out by staged upgrades of the ITER-RC facility, based on reactor-scale data from its initial operations. Tritium breeding and other blanket technologies will be tested at moderate neutron fluences in test modules that fit into ITER-RC ports. Within 30 years, design of a DEMO will be nearly complete, and fusion power will be at the threshold of commercial realization.

Proponents' and Critics' Claims

Proponents claim that the ITER-RC approach employs a single facility for the necessary integrated physics and technology investigations at a minimum integrated cost. Since tokamak science at the reactor scale differs from that of present devices, ITER-RC will provide the relevant science basis for fusion energy. ITER-RC is no higher risk than modular approaches because all the technological goals must be met in any event. It further provides for reactor-scale development of advanced and steady-state burning plasma scenarios with as much flexibility as a tokamak can implement. At the present time, it is the only next-step tokamak for which international collaboration on construction can be foreseen and can lead to a commercial fusion power by ~2050. A large advantage is derived by international cost sharing.

Critics claim that the ITER-RC facility cost places it out of reach for an energy technology demonstration. Moreover, ITER-RC specializes to the tokamak concept too early in the development of fusion energy; researchers should wait for the possible development of an alternative approach before moving to reactor-scale research. Science carried out in present tokamak facilities may permit a more judicious and optimized choice of reactor configuration.

M-19. VOLUMETRIC NEUTRON SOURCE

Description

- A Volumetric Neutron Source (VNS) is a deuterium-tritium (D-T) plasma-based facility that simulates the fusion environment and produces neutrons at a neutron wall load (~1 MW/m^2) and on a surface area (~10 m^2) sufficient to test in-vessel components: first wall, blanket, divertor, and vacuum vessel. The facility is designed so that the in-vessel components of the basic device are also part of the tests, especially for identifying failure modes, obtaining data on failure rates, testing remote maintenance, and determining shutdown times required to recover from failures.
- The D-T plasma is highly driven (Q ~ 1) and operated at steady state or long pulse.
- Several designs have been proposed including high aspect ratio tokamak, spherical torus (ST), and mirrors. In general, they have resistive copper magnets and a fusion power <150 MW to minimize cost and tritium consumption (in some options, the facility breeds its own tritium).
- The purpose of tests in VNS are (1) to examine experimentally the scientific and engineering feasibility and attractiveness issues for various design and material system options for the first wall/blanket/divertor/vacuum vessel (e.g., solid and liquid walls); (2) to obtain critical data on basic phenomena and performance of in-vessel components concerning power and particle extraction, tritium self-sufficiency, failure modes and rates, maintainability, and other key issues; (3) to advance the engineering sciences necessary to conduct powerful D-T plasma physics experiments; and (4) to provide the engineering science knowledge base for the in-vessel system, which is a necessary element in the fusion program mission to identify an attractive fusion product.

Status

- The VENUS study at the University of California—Los Angeles (UCLA) (1994) and the International Energy Agency (IEA) study on HVPNS (1995) have made progress in defining the major requirements on VNS parameters and design features [see M. Abdou et al., *Fusion Technology*, **29** (1996)]. These include (1) neutron wall load ~1 MW/m^2; (2) total test area at the first wall ~10 m^2; (3) steady-state or long-pulse plasma operation; (4) configuration, remote maintenance, and other design features to emphasize the reliability of the basic device and rapid replacement of device components and test articles; (5) device availability >25%; (6) cumulative fluence (in sequential test articles) $\geq$6 MW year/m^2; and (7) fusion power <150 MW.
- Two Small Business Innovation Research (SBIR) studies [e.g., E. T. Cheng et al., "Progress of the ST-VNS Study," *Fusion Technology*, in press) and other studies [e.g., Ho and Abdou, *Fusion Eng. & Design*, **31** (1996)] have identified two classes of possible VNS facilities. One is based on high A (~3.5), and the other is based on ST. Both satisfy the VNS requirements and are cost-effective. Some researchers in Japan and Russia proposed mirror-based facilities.

Current Research & Development (R&D)

R&D Goals and Challenges

- Identify the best design option for VNS that (1) is cost-effective, (2) can meet the testing requirements, and (3) requires modest extrapolation of current database.
- Perform the physics and engineering R&D necessary to build VNS by ~2015–2020. Establish the physics and engineering bases for steady-state/long-pulse-driven (Q ~ 1) D-T plasma operation with high device availability (~25%) and rapid insertion and removal of test articles.
- Establish close interaction and coordination between the physics and technology community.
- Develop international framework for collaboration on VNS.

Related R&D Activities

- The National Spherical Torus (NSTX) program for establishing the physics database for ST plasmas.
- SBIR projects on design of VNS.
- Considerable experience relevant to VNS gained from the International Thermonuclear Experimental Reactor (ITER) Test Program.
- Several international workshops and an IEA study on VNS.

Recent Success

- Results of the IEA study on High Volume Plasma-Based Neutron Source have shown clear consensus among international technology experts of the critical need for VNS and have identified the mission for and major requirements on VNS.
- SBIR and other studies have identified attractive and cost-effective options for VNS.

Budget

Currently, only one company is funded through the SBIR program at $750K for 3 years. Some activities are being carried out on VNS-ST under the NSTX program.

Anticipated Contributions Relative to Metrics

Metrics

- Neutron wall loading ~1–2 MW/m^2.
- Steady-state D-T plasma operation or long burn length (>1000 s) with a duty cycle >80%.
- Cumulative neutron fluence in successive test article ≥6 MW year/m^2.
- Total test area at the first wall ≥10 m^2, a minimum test area per test article of ~0.36 m^2.
- Device availability >25%.
- Capability for continuous operation during test campaigns of >1–2 weeks.
- Magnetic field in the test region >2 T.

Other metrics related to cost-effectiveness and experimental flexibility include fusion power <150 MW; configuration, remote maintenance, and other features that emphasize device reliability and rapid insertion and replacement (days) of test articles; relatively low power consumption (for normal copper coils, current drive) <400 MW; and safe operation.

Near Term ≤5 years

- Conceptual design of a VNS to meet its mission requirements at reasonably low cost, modest risk, and minimum extrapolation of physics and technology database.
- Physics database from the NSTX program for VNS-ST and from other world facilities for standard aspect ratio tokamaks.
- Engineering interface solutions for test article integration including instrumentation and remote maintenance. Examples of conceptual designs of test articles based on engineering scaling rules.

Midterm ~20 years

- Progress in physics database for VNS (plasma-driven) facility.
- Progress in establishing the engineering database for VNS.
- Database from nonfusion facilities for first wall/blanket/divertor options to be tested on VNS.
- Engineering design of VNS (most likely with international collaboration).
- Construction of VNS (around the year 2015?).

Long Term >20 years

- Use VNS to obtain database on first wall/blanket/shield, remote maintenance, and other systems.
- Identify the most attractive options of in-vessel components for an attractive and competitive fusion product.

Proponents' and Critics' Claims

Proponents see a VNS as the most practical and cost-effective option for obtaining critical data on in-vessel components and material systems. Proponents see VNS as a critical element in establishing the engineering feasibility of fusion systems (especially for key issues such as heat and particle removal capability, tritium self-sufficiency, failure rates and modes, maintainability, and safety and environmental features). Proponents also claim that conducting (nuclear technology) testing and plasma ignition in two separate facilities is the most cost-effective pathway because nuclear testing requires small power but high fluence while ignition requires large power and low fluence.

Critics claim that combining the mission of VNS with plasma ignition in a single facility such as ITER is a more desirable approach because it focuses world attention on one facility and provides full integration of all systems. [Proponents of VNS counter that VNS will still be needed prior to a demonstration reactor (DEMO) even if ITER is built. The IEA study on VNS calculates that the maximum availability achievable in DEMO is ~4% with the ITER-alone scenario; testing in-vessel components in VNS will allow the DEMO to reach its availability goal of 50%.] Critics also voice concern that VNS cannot achieve 25% availability and 6 MW·year/m^2 fluence because of the lack of an adequate engineering database. [Proponents of VNS counter that generating the engineering database is part of the mission of VNS. They say that engineering tests are expensive, complex, and time-consuming and that the only cost-effective approach is to perform these engineering tests on a small device (driven plasma Q ~ 1, small fusion power).] Proponents also argue that what matters is the cumulative fluence experience in sequential test articles, not only the fluence on a given test article. Proponents also ask that if building the small-size, small-power VNS with availability of 25% is high risk because of lack of engineering database, how can we assume that the much larger (20 times the volume), more expensive (10 times higher cost), and much more demanding (50% availability) DEMO could be built without VNS?

M-20. ADVANCED FUELS

Description

All important advanced fusion fuels produce their energy mainly as charged particles, in contrast to the 80% neutron power fraction produced by deuterium-tritium (D-T) fuel and ~50% produced by deuterium-deuterium (D-D) fuel. Anticipated engineering, safety, and environmental advantages of reduced neutron production motivate advanced-fuel research, despite greater physics obstacles compared to D-T (see following page). The two advanced fuels generally considered most important are D-^{3}He (1–5% of fusion power in neutrons from D-D reactions) and p-^{11}B (no neutrons). Although p-^{11}B and ^{3}He-^{3}He produce no neutrons, calculations indicate that Maxwellian (thermal) plasmas will produce bremsstrahlung radiation power less than or equal to fusion power. Burning of p-^{11}B, therefore, would require non-Maxwellian fusion concepts (e.g., inertial-electrostatic confinement or colliding-beam fusion) or very low charged-particle transport and other radiation losses. Fusion gain, even under these circumstances, is projected to be modest, requiring a large recirculating power fraction.

Status

Researchers have pursued advanced fuels since the earliest days of fusion energy because D-T fusion carries the liabilities of high radiation damage, large induced radioactivity and afterheat, a large radioactive waste volume, complex tritium-breeding blankets, and extensive tritium handling. Conceptual design studies show the following:

- Radiation damage in advanced-fuel power plant structures would be sufficiently low for them to be full-lifetime components, whereas tritium-breeding blankets and part of the shield must be replaced every few years for D-T.
- Induced radioactivity in advanced-fuel power plants would be sufficiently low for the waste to qualify at worst as low-level, essentially similar to hospital radioactive waste, and for the power plant to qualify as inherently safe (no possibility of significant radioactivity release or a meltdown).
- Fueling and exhaust handling systems in advanced-fuel power plants should be relatively simple in comparison to the complex safety and processing equipment required for the large D-T power plant tritium throughput.
- Surface heat fluxes on D-^{3}He first walls are manageable, particularly in geometries where the charged particle transport losses flow along magnetic fields outside the fusion core into a separate chamber.
- Sufficient ^{3}He has been identified on earth to conduct a D-^{3}He fusion research program up to and including the first 1000-MW(e) power plant.
- Advanced fuels such as p, D, and ^{11}B are plentiful on earth, but large-scale deployment of D-^{3}He power plants would require use of the large ^{3}He resource (~10^{9} kg) on the lunar surface.

Current Research & Development (R&D)

R&D Goals and Challenges

- <u>Develop fusion concepts for advanced fuels</u>. Several presently funded innovative magnetic fusion concepts would satisfy this challenge for D-^{3}He, while p-^{11}B would require a non-Maxwellian plasma approach.
- <u>Quantify the safety, environmental, and economic attributes of advanced-fuel power plants</u>. The importance of energy for the global environment and marketplace makes careful evaluation of all fusion options necessary.
- <u>Demonstrate the feasibility of lunar ^{3}He acquisition or terrestrial breeding</u>. The Apollo program developed much of the required rocket technology 30-years ago, but the use of terrestrial mining technology and its reliability on the moon remain unproven. Attractive ^{3}He breeding methods have not yet been identified.

Related R&D Activities

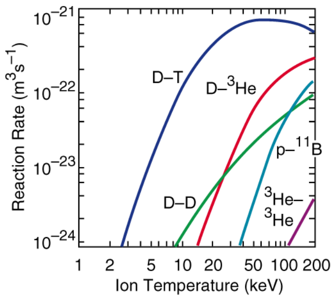
- A very small experimental and theoretical research program exists for the high-β (plasma pressure/magnetic field pressure) configurations best suited to burning advanced fusion fuels.
- Lunar ^{3}He geologic, economic, environmental, and legal characteristics have been analyzed and found to be favorable, although this activity is presently unfunded and dormant.
- Direct conversion of charged particle energy to electricity at high efficiency (>60%) was shown at Lawrence Livermore National Laboratory in the 1970s.

Recent Successes

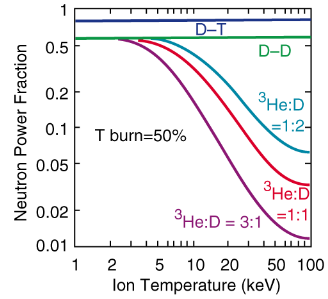
- Innovative concept research resurfaced several years ago after a decade that virtually eliminated all advanced configurations. The reinvigorated program has been producing encouraging theoretical results for the high-β concepts of primary interest for advanced fuels. Innovative concept experiments presently are coming on-line.
- The Joint European Torus (JET) produced 200 kW of D-^{3}He fusion power in 1993.
- Fusion power plant conceptual design analyses support the attractiveness of advanced-fuel fusion. Although the funding level of these studies is low, they give qualitatively encouraging results.
- National Aeronautics and Space Administration (NASA), NASDA (Japan), and other space agencies are seriously assessing the near-term return of humans to the moon, including the assessment of ^{3}He resources and mining technology.

Budget

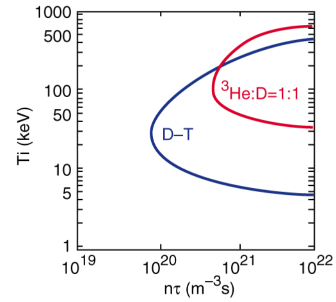
The present worldwide budget for advanced-fuel research is less than $1M.



Reaction rates (Maxwellian plasmas)



Neutron power production (Maxwellian plasmas)



Ignition temperature and confinement requirements

Anticipated Contributions Relative to Metrics

Metrics

- $\underline{\beta}$. Innovative confinement concepts must demonstrate sufficiently high values (>0.2) of this ratio of plasma pressure to magnetic-field pressure for advanced-fuel operation. The β value is fundamental to achieving high fusion power density, which strongly relates to cost and scales as β^2B^4, where B ≡ magnetic field magnitude.
- $\underline{n\tau}$. Sufficient values (>10^{21} m^{-3} s) of this “confinement parameter,” the product of plasma fuel–ion density and energy confinement time, must be demonstrated for plasma charged-particle energy losses to be manageable.
- Projected cost of electricity (COE). New power plant concepts must achieve competitive COE values to break into the anticipated future energy marketplace. Detailed conceptual design analyses for the high-β innovative concepts remain to be performed at the level required for confident COE projections.

Near Term <5 years

- Demonstration of proof-of-principle (PoP) for a high-β innovative concept would be a major step along the path of advanced-fuel fusion.
- Detailed conceptual designs could lay a firmer foundation for the anticipated benefits and challenges of coupling advanced fuels with innovative confinement concepts.

Midterm ~20 years

- PoP D-^{3}He operation in an innovative confinement concept appears likely given the pace of concept progress.
- A suitable non-Maxwellian fusion concept for third-generation fuels, such as p-^{11}B, might reach the PoP stage.
- Burning D-^{3}He in an integrated test facility could be accomplished with a concerted effort.
- Demonstration of the feasibility of returning ^{3}He economically from the moon to earth will be necessary for D-^{3}He power plants. Because of the modularity of the mining process, the required scale should be modest.

Long Term >20 years

- Because advanced fuels substantially relax engineering constraints, a physics PoP demonstration during the midterm research phase could lead to operating commercial power plants in this time frame.
- Solar system exploration and development will be in progress, and lunar operations for science and ^{3}He acquisition will most likely have begun.

Proponents’ and Critics’ Claims

Proponents claim that (1) the decades of testing required for developing reliable, low-activation materials will keep D-T fusion from entering the marketplace on any relevant time scale; (2) D-^{3}He plasmas can have higher fusion power densities than D-T plasmas, because neutron damage limits D-T neutron wall loads more than surface heat loads limit D-^{3}He plasmas; and (3) lunar ^{3}He acquisition requires essentially developed technology.

Critics claim that (1) advanced materials can mitigate the D-T neutron damage, activation, and afterheat problems; (2) D-T power plants can also make use of progress in β or $n\tau$ by the innovative concepts; and (3) the procurement of ^{3}He from the Moon is too speculative.