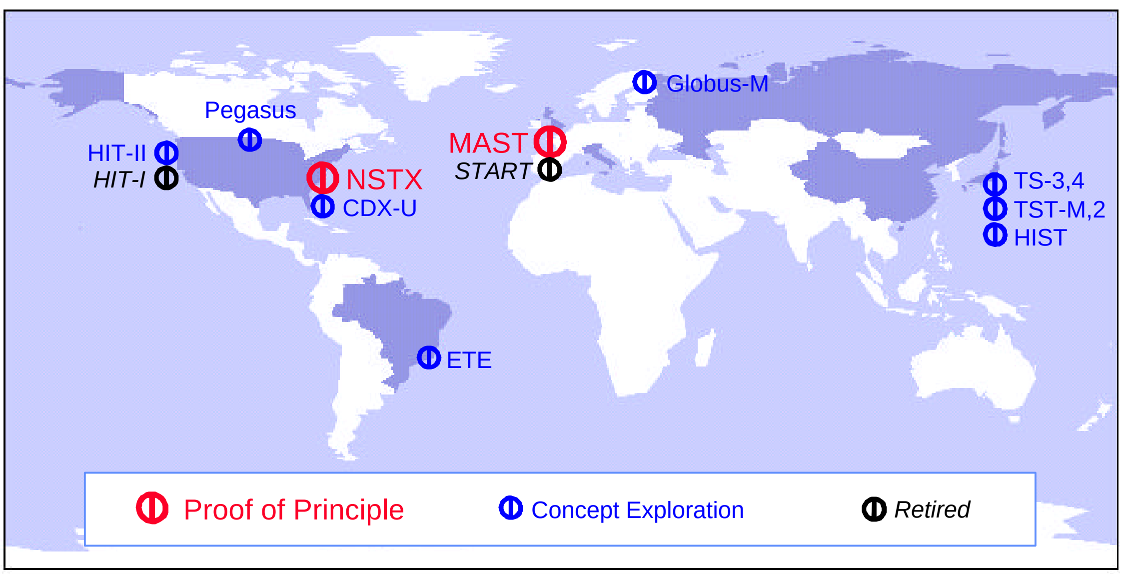
Appendix 2-1. Spherical Torus (ST) Devices and Designs - June 1999

Data sheets for the ST devices and designs listed in Table 2-1 are provided here as backup information to the White Paper, since these are relatively new to the fusion energy sciences research community.

Existing and Retired Experiments

At the level of Concept Exploration (CE) and Proof of Principle (PoP), a large number of experiments have emerged, with some already replaced, during this decade. (See Figure below). In addition, a number of pioneering tests emerged in the late 1980's by inserting a current-carrying central rod into an existing Spheromak, and indicated encouraging equilibrium and stability results even when the plasmas were limited to low temperatures (~10's eV). These included the Heidelberg Spheromak [BRUHNS, 1987], the Lucas Heights Rotamak-ST [COLLINS et al, 1988], and the SPHEX Rodamak [BROWNING, 1992]. The experiments indicated here have produced or are expected to produce plasmas with temperatures of order ~100 eV or higher (e.g., a few keV in NSTX and MAST).



Existing and Retired ST Experiments in the World

The following data sheets for the existing and retired ST experiments (Table 2-1), in order of increasing R_0, are included:

A. HIT-I (Jarboe, U. Washington, U.S.), retired
B. START (Sykes, Culham, U.K.), retired
C. TST-M (Toyama, U. Tokyo, Japan), retired
D. TS-3 (Katsurai, U. Tokyo, Japan), operating
E. HIT-II (Jarboe, U. Washington, U.S.), operating

F. HIST (Nagata, Himeji Inst. Technology, Japan), operating
G. CDX-U (Majeski, PPPL, U.S.), operating
H. ETE (Ludwig, INPE, Brazil), under construction
I. TST-2 (Takase, U. Tokyo, Japan), under construction
J. TS-4 (Ono, U. Tokyo, Japan), under construction
K. Pegasus (Fonck, U. Wisconsin, U.S.), achieved first plasma
L. Globus-M (Gusev, Ioffe Inst., R.F.), achieved first plasma
M. MAST (Cox, Culham, U.K.), commissioned
N. NSTX (Peng, Ono, U.S.), achieved first plasma

Design Concepts for the Future

Assuming the theoretical projections so far for the ST plasma, a number of design concepts for future ST devices have recently been examined in varying levels of detail. Such studies are directed to future stages of ST fusion development (Performance Extension, Fusion Energy Development, Pilot Plant or DEMO, and Power Plant) to help clarify the physics database, technology capabilities, engineering solutions, and facility features required to reach these stages. Highlights for the following examples are included:

O. DTST (Peng, U.S.), for Performance Extension, Optimization, and Burning Plasma Testing
P. CTF (Hender, U.K.), for Component Testing (Fusion Energy Development)
Q. VNS (Cheng, U.S.), for Fusion Energy Development testing
R. FDF (Stambaugh, U.S.), for Fusion Development encompassing Performance Extension testing, Fusion Energy Development testing, and Pilot Plant testing
S. ST Power Plant (Hender, Voss, U.K.), for central station electricity production
T. ARIES-ST (Najmabadi, U.S.), for central station electricity production

References (Appendix 2-1)

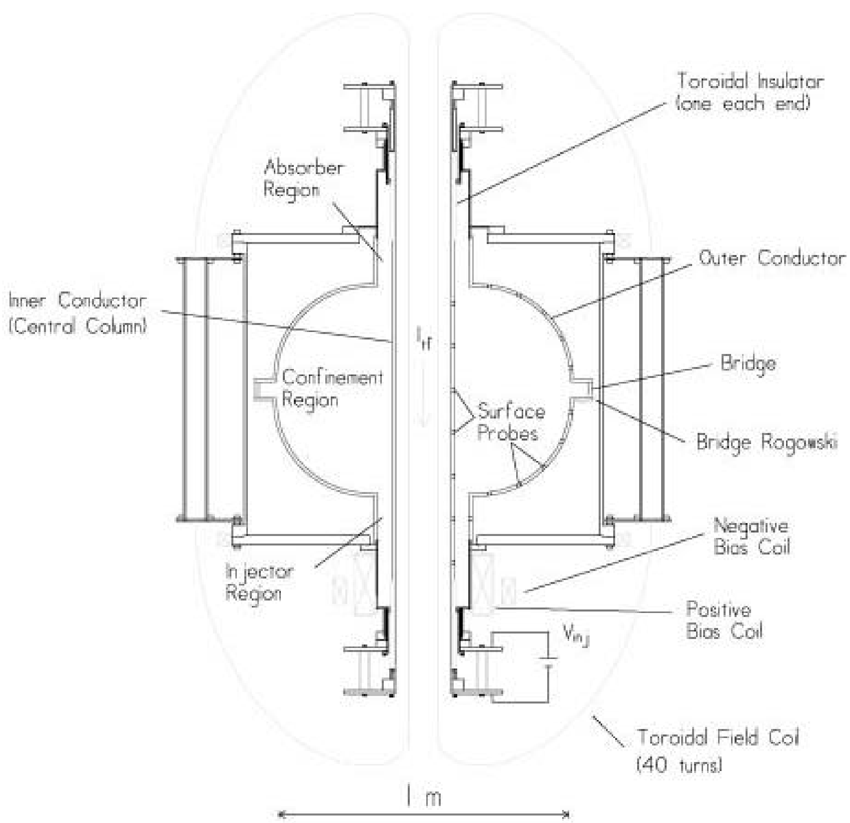
BRUHNS, H. et al, 1987, Nucl. Fusion, **27** (1987) 2178.
COLLINS, G. A. et al, 1988, Nucl. Fusion, **28** (1988) 255.
BROWNING, P. K. et al, 1992, Phys. Rev. Lett., **68** (1992) 1722.

A. HIT-I (Jarboe, U. Washington, U.S.), Retired

Stage	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
CE	0.3	0.2	1.85	0.01	0.5	0.25	CHI	Y

Accomplishments

- Achieved plasma current up to 250 kA using only Coaxial Helicity Injection (CHI)
- Measured electron temperature ~100 eV
- Demonstrated Transient Internal Probe (TIP) measurements of internal magnetic fields and fluctuations

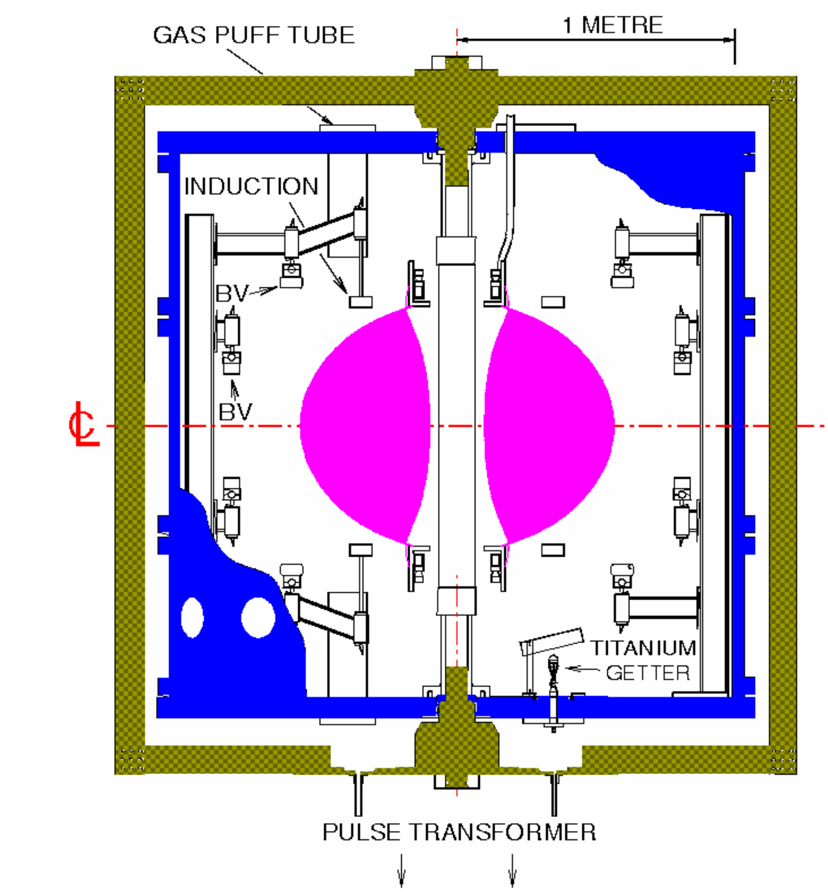


B. START (Sykes, Culham, UK), Retired

Stage	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div ?
CE	0.32	0.25	≥3	0.05	0.3	0.31	1.0	Y

Accomplishments

- Demonstration of formation and equilibrium concepts for the ST, e.g. induction-compression; existence of vertically stable high elongation plasmas, 'natural' and double null divertor configurations. Demonstration of favourable properties – high T_e, T_i (few hundred eV), operation up to Greenwald density limit, and down to $q_{95} \sim 2$
- Demonstration of 'disruption resilience' whereby 'IREs' largely replace major disruptions
- First studies of auxiliary heating in an ST (using NBI loaned from ORNL)
- Obtained World record β values: NBI heated plasmas, $\beta_T \sim 40\%$, $\beta_N \sim 5.5$
- Ohmic plasmas, $\beta_T \sim 24\%$, $\beta_N \sim 3.5$
- Studies of energetic particle-driven instabilities (TAEs etc.) during NBI
- First measurements of τ_E in a ST, using extensive diagnostics including $T_e(r)$, $T_i(r)$, $n_e(r)$, and EFIT magnetic reconstruction (collaboration with GA, USA). Established ST confinement database; proposed modified neo-Alcator scaling to represent Ohmic ST plasmas
- First demonstration of H-mode and improved confinement in an ST
- First measurement of SOL and divertor loading in a ST
- First pellet injection experiments in a ST (collaboration with ENEA Frascati)



C. TST-M (Toyama, U. Tokyo, Japan), Retired

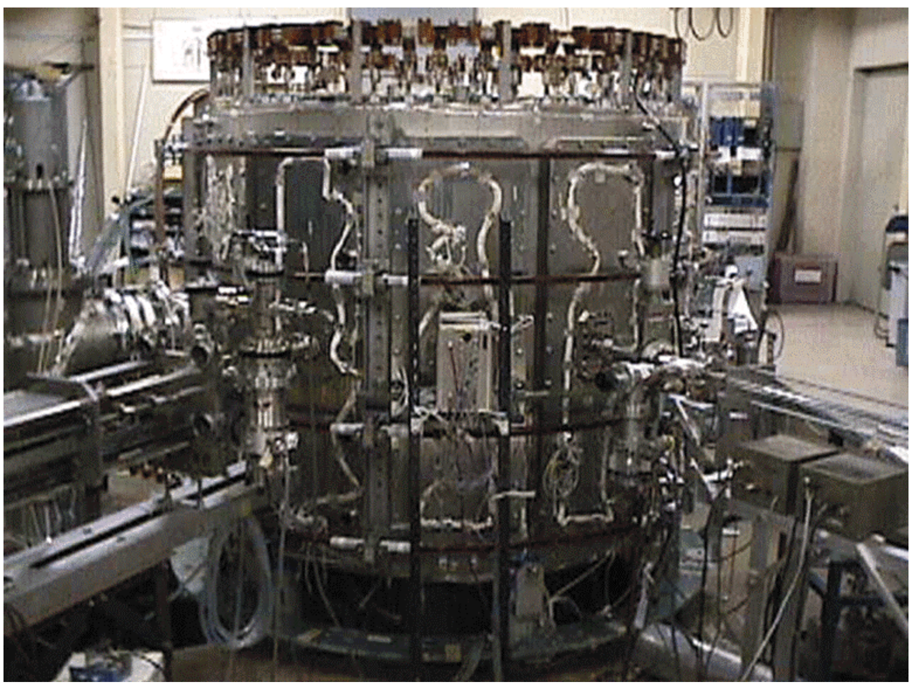
Stage	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
CE	0.35	0.25	1.5	0.01	0.3	0.1	Ohmic	N

Research Goals

- Turbulence and transport
- Helicity injection current drive

Accomplishments

- Measurement of electrostatic and electromagnetic fluctuations by Langmuir and magnetic probes for aspect ratios A = 1.3 – 2.5
  - Electromagnetic fluctuations decrease with A, while electrostatic fluctuations remain unchanged
  - Transport dominated by electrostatic fluctuations
- Current drive effects observed using an electron emitter, a plasma gun, a pair toroidal symmetric electrodes
- Plasma current ramp-up to 150 kA using a pair of auxiliary induction coils



D. TS-3 (Katsurai, U. Tokyo, Japan), Operating

Stage	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
CE	0.2	0.19	0.8-2	0.001	0–0.2	0.08	Ohmic	N

Research Goals

- Comparison of ST, Spheromak, FRC, and Compact RFP equilibrium and stability in a single device
- Formation of high-β ST via merging of two spherical plasmas
- Exploration of low-q limit of ST and transition between ST and Spheromak
- MHD and kinetic investigations of transition between ST and FRC

Accomplishments

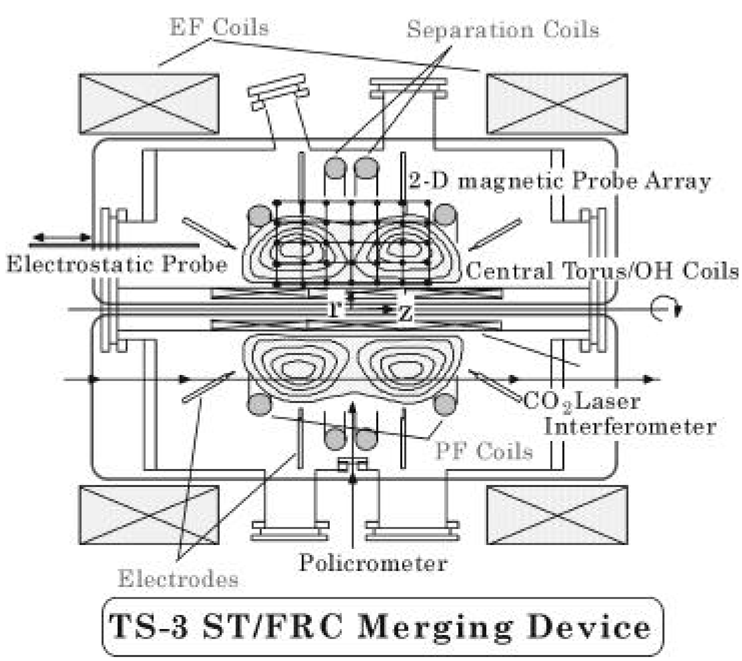
- Formation and stabilization of tilt and shift of ultra-low aspect ratio ST (A = 1.05)
- Merging of two ST plasmas and the associated heating of ions (to 100s eV)
- Formation of ultra-high-β (strongly diamagnetic) ST plasma (⟨β⟩ above 50%) via merging of two Spheromaks with opposing magnetic helicity

Proposed Research (in 3–5 years)

Through December 1999: Magnetic reconnection and plasma heating mechanisms during plasma merging

Through March 2001: MHD and kinetic study of ultra-high β ST transformed from FRC stabilized by applied toroidal field

Through March 2001: Comparison of equilibrium and stability of ST, Spheromak, FRC, and compact RFP and the transition between these configurations



E. HIT-II (Jarboe, U. Washington, U.S.), Operating

Stage	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Di
CE	0.3	0.2	1.85	0.035	0.5	0.2	CHI, Ohmic	Y

Research Goals

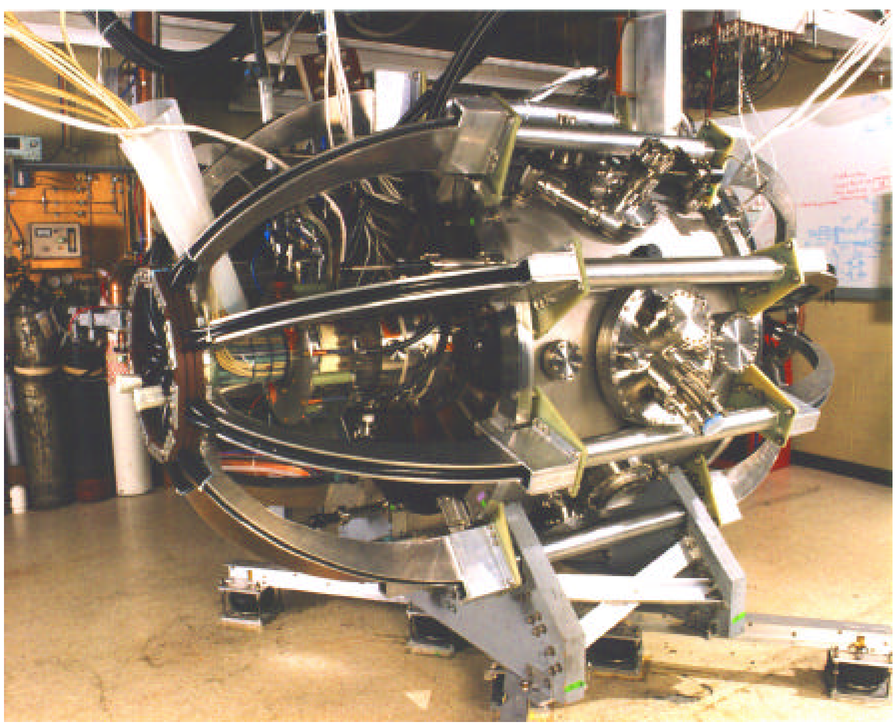
- Demonstrate CHI current drive with a thin stainless steel shell
- Combine CHI current drive with transformer current drive for current profile control
- Study effects of CHI on confinement

Accomplishments

- Achieved over 200 kA by both CHI and transformer current drive methods
- Sustained CHI plasma for several R/L wall times
- Added CHI current drive sustainment to end of transformer current drive

Proposed Research (in 3–5 years)

- Further operation mixing CHI and transformer current drive for current profile control and stability studies
- Install multi-point Thomson scattering for confinement studies
- Develop constant helicity injection using inductive methods



F. HIST (Nagata, Himeji Inst. Technology, Japan), Operating

Stage	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
CE	0.3	0.24	2.0	0.005	0.2	0.15	Ohmic	N

Research Goals

- Investigation of helicity injection physics related to MHD relaxation, magnetic reconnection, and plasma flow on ST and Spheromak
- Current profile control by Helicity Injection Current Drive (HICD) combined with Ohmic Heating Current Drive (OHCD)
- Fueling and density profile control by Compact Toroid (CT) injection into ST

Accomplishments

- Coaxial Helicity Injection (CHI) formation and sustainment of ST plasmas for 4 ms with peak toroidal current I_t = 150 kA
- Verification of internal magnetic field structure of ST sustained by CHI
- Comparisons of magnetic fluctuations related to current drive mechanism between ST and Spheromak

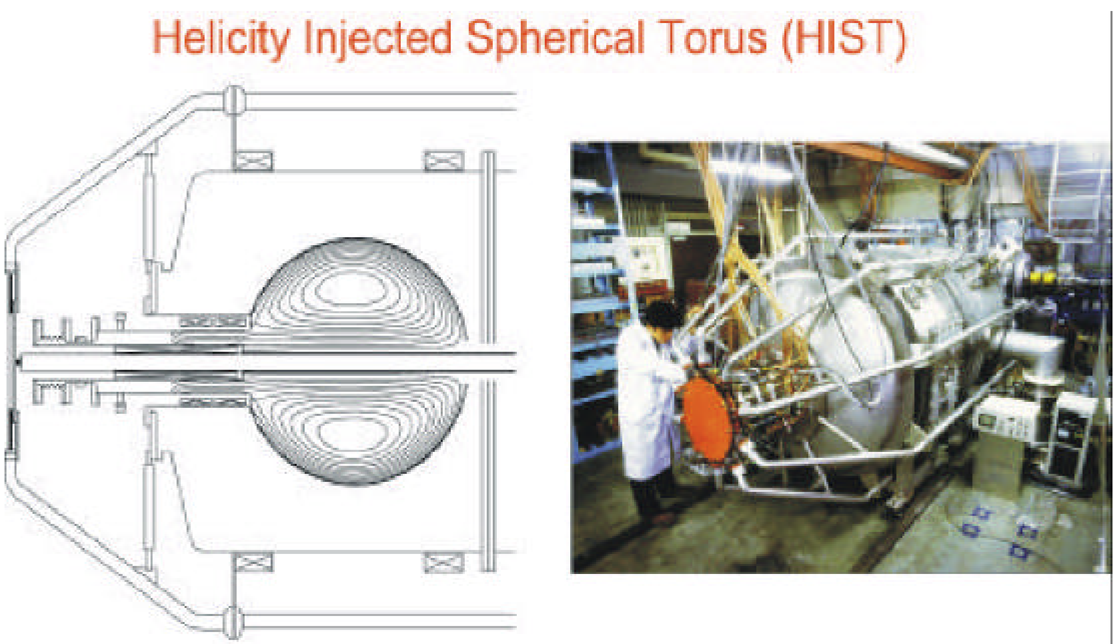
Proposed Research (in 3–5 years)

<u>July 1999 – March 2000</u>:

- Investigation of HICD mechanism on ST
- Formation of ST by combination of CHI and OHCD

<u>April 2000 - March 2002</u>:

- Investigation of current profile control in ST by CHI and OHCD
- CT injection into ST plasmas



G. CDX-U (Majeski, PPPL, U.S.), Operating

Stage	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
CE	0.34	0.22	1.6	0.05	0.2	0.2	0.3	Y

Research Goals

Explore the scientific boundaries of the ST plasma, such as in rf heating and current drive, rotating magnetic field current drive, novel rf techniques for core plasma measurements, and novel FRC formation techniques

Accomplishments

- Initial measurements of very small halo currents (≤5% of plasma current) during forced vertical-displacement disruption of ST plasma
- Tests of Ultra-Soft X-Ray tomography diagnostic technique to measure time and space characteristics of ST plasma edge region oscillations
- Initial verification of HHFW launching and propagation in ST plasma at $P_{HHFW} < P_{Ohmic}$

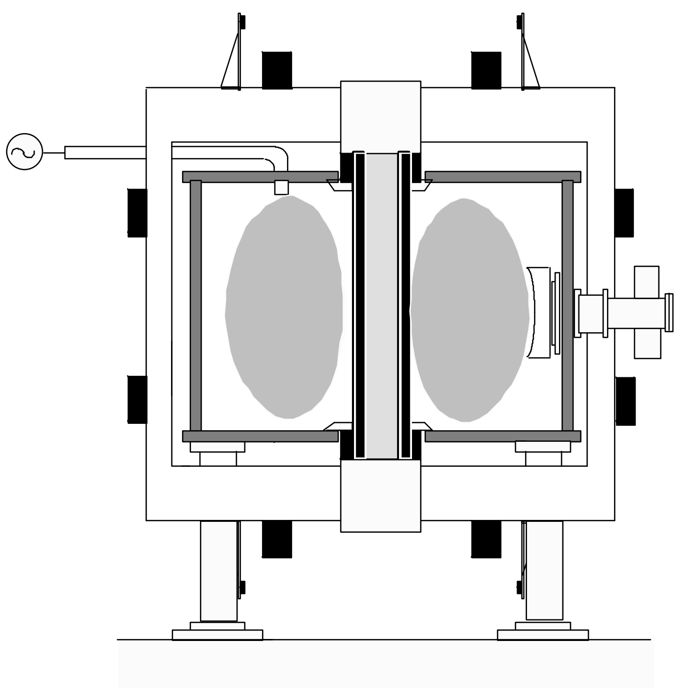
Proposed Research (in 3–5 years)

Through September 1999:

- Test of HHFW launching and propagation at $P_{HHFW} \sim P_{Ohmic}$
- Tests of Electron Bernstein Wave (EBW) emission as a possible time-resolved T_e diagnostic method

October 1999 - September 2001:

- Investigation of ST plasma-Lithium divertor interactions
- Investigation of ST edge current drive via rotating magnetic field
- Investigate rf techniques for FRC/FRM (Field-Reversed Mirror) plasma formation, stabilization, and sustainment, tilt-stabilized with small externally applied toroidal field



H. ETE (Ludwig, INPE, Brazil), Under Construction

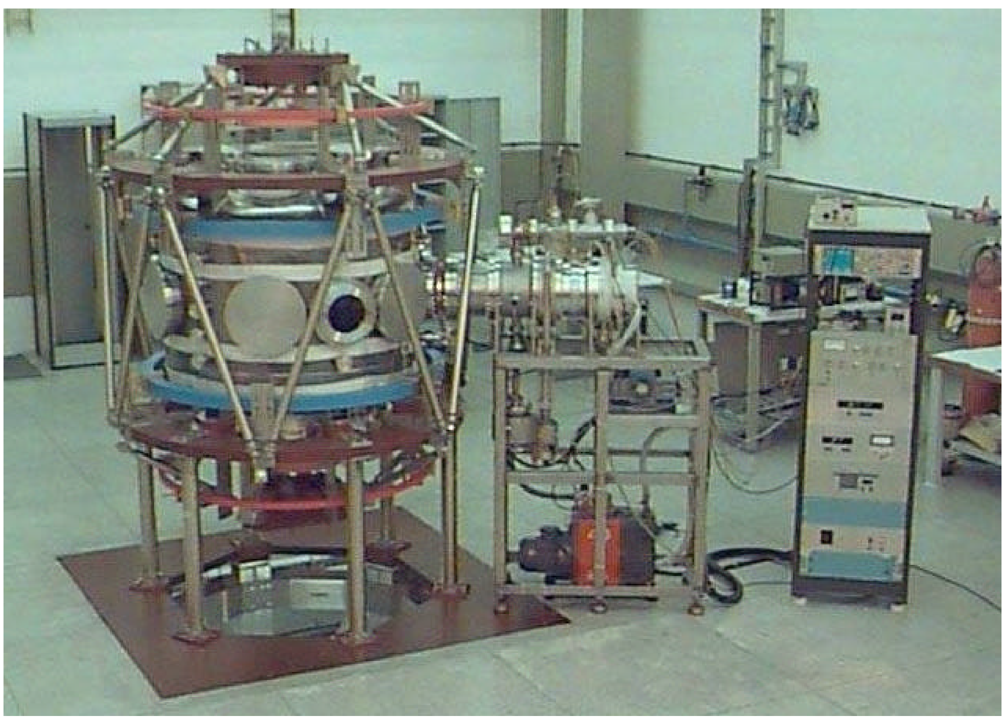
Stage	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
CE	0.3	0.2	1.8	0.02	0.6	0.4	Ohmic	N

Research Goals

- Investigation of Ohmic plasma operational space
- Studies of plasma edge
- Diagnostics development (Lithium beam probe, multipass CO2 laser interferometer)
- Tests of new current drive ideas

Proposed Research (in 3–5 years)

1999 – 2000: Basic ST physics, operational parameter space investigation
2001 – 2002: Machine upgrade to maximum parameters, operational space, plasma edge studies
2003 – 2004: Current drive development (e.g., rotating magnetic field)



Tests and conditioning of the vacuum vessel of the ETE experiment (June 1999)

I. TST-2 (Takase, U. Tokyo, Japan), Under Construction

Stage	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
CE	0.37	0.23	1.5	0.1	0.4	0.2	Ohmic	N

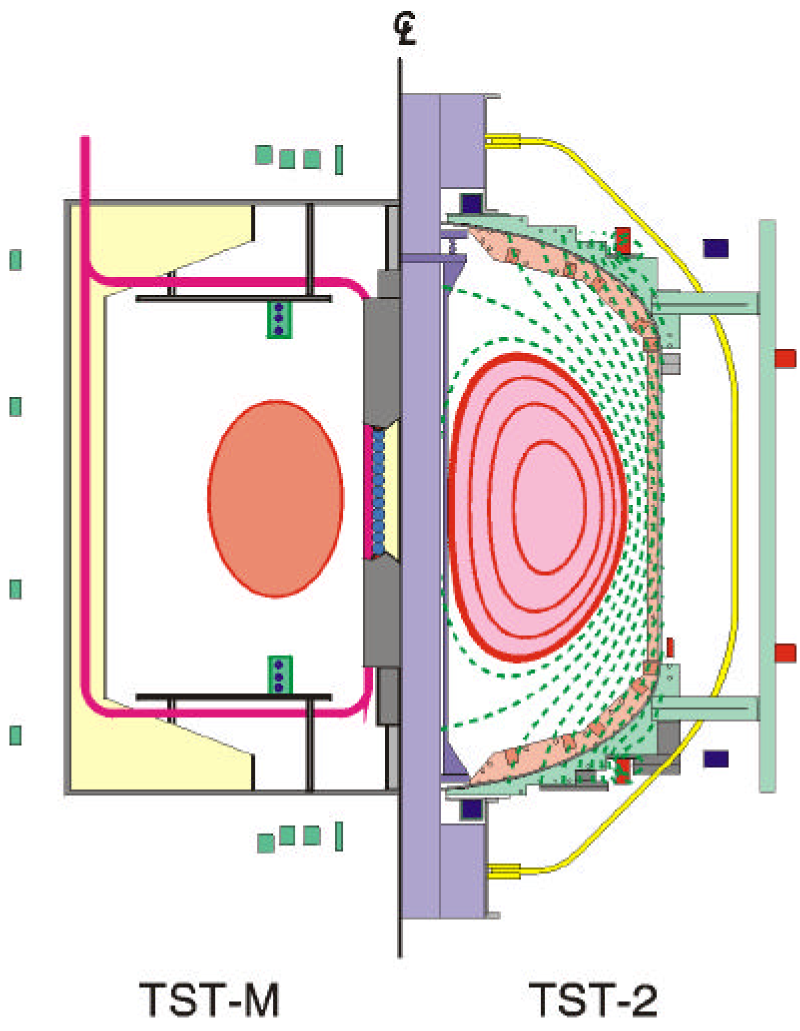
Research Goals

- Development of ST plasma startup techniques
- RF wave physics
- Turbulence and transport

Proposed Research (in 3–5 years)

<u>August 1999 – August 2000</u>: Plasma startup improvement, ohmic plasma, rf wave physics experiments at low power

<u>September 2000 – March 2002</u>: RF heating experiments, turbulence and transport control experiments



J. TS-4 (Ono, U. Tokyo, Japan), Under Construction

Stage	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
CE	0.5	0.45	0.8–3	0.005	0–0.5	0.3	Ohmic	N

Research Goals

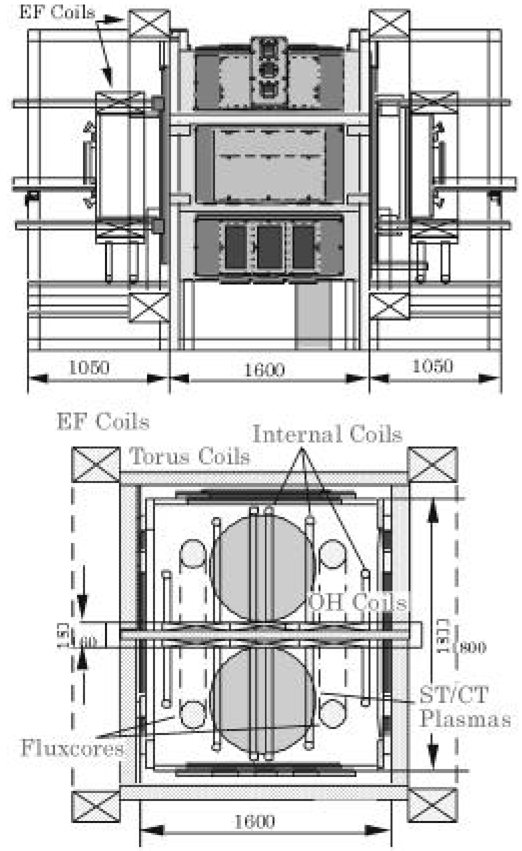
- Comparison of ST, Spheromak, FRC, and Compact RFP plasma stability, transport, and kinetic properties in a single device
- Stability, Transport, and Kinetic properties during transition between ultra-high-β ST and FRC plasmas
- Formation and heating of ST plasma via merging Spheromaks with opposing magnetic helicity
- Application of laser plasma to formation of high-β ST plasma

Proposed Research (in 3–5 years)

<u>October 1999 – March 2000</u>:
Initial operation

<u>January 2000 – December 2000</u>:
MHD and kinetic properties of ultra-high-β ST formed from FRC stabilized by applied toroidal field

<u>January 2000 – March 2002</u>:
Comparison of stability and transport properties of ST, Spheromak, FRC, and Compact RFP plasmas



TS-4 ST/FRC Merging Device

K. Pegasus (Fonck, U. Wisconsin, U.S.), Achieved First Plasma

Stage	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
CE	0.45	0.40	2–4	0.04	0.1	0.3	2	Y

Research Goals

- Minimize central column while maintaining stability and confinement
- Stability limits for A < 1.3, I_p/I_{TF} » 1 in OH and RF heated, diverted or limited plasmas
- Access to near-unity β_T without a close-fitting conducting shell as A → 1
- Explore relaxation stability boundaries between ST and Spheromak or Compact Toroids
- Startup and sustainment techniques development (HHFW, EBW, Gun Current Injection)

Accomplishments

- MEDUSA: j(R) profiles measured in ST
- Pegasus: first plasma achieved @ B_T = 0.07 T

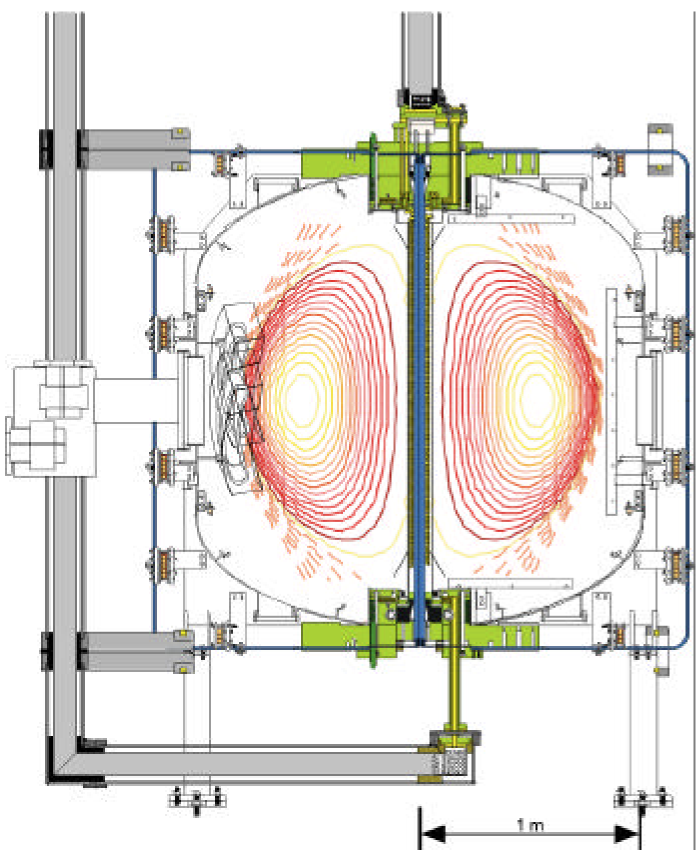
Proposed Research (in 3–5 years)

November 1999 – October 2000:

- Low-q/low-TF stability studies in OH plasmas; high I_p/I_{TF} operation as A → 1
- Exploration of β-limit as A → 1; effects of configuration (κ, A) on stability limits
- Explore minimal B_T needed for low-q stability against relaxation instability
- Tokamak-Spheromak transition studies

November 2002 – October 2004:

- Reduction of center column to minimum: operation at minimal TF
- Noninductive startup and sustainment via plasma guns, EBW, and HHFW



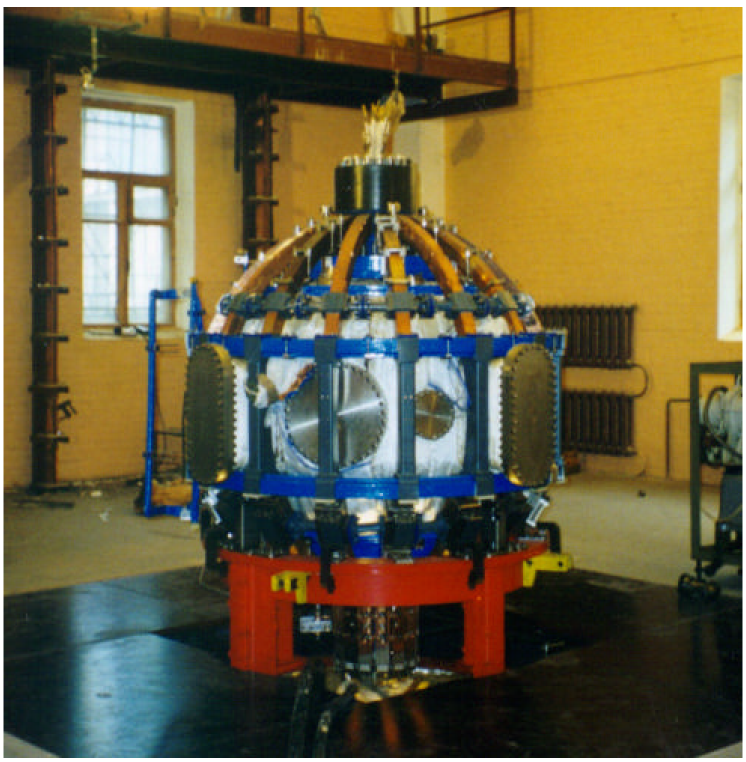
L. Globus-M (Gusev, Ioffe Inst., R.F.), Achieved First Plasma

Stage	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
CE	0.36	0.24	2.2	<1	0.65	0.5	5	Y

Research Goals

Accomplishments

Proposed Research (in 3–5 years)



M. MAST (Cox, Culham, UK), Commissioned

Stage	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div ?
PoP	≤ 0.85	≤ 0.65	2.5	5	0.5	2	6.5	Y

<u>Research Goals</u>: Use the ST to advance understanding of fusion devices in general, and in particular to establish the potential of the ST as a future VNS or power plant

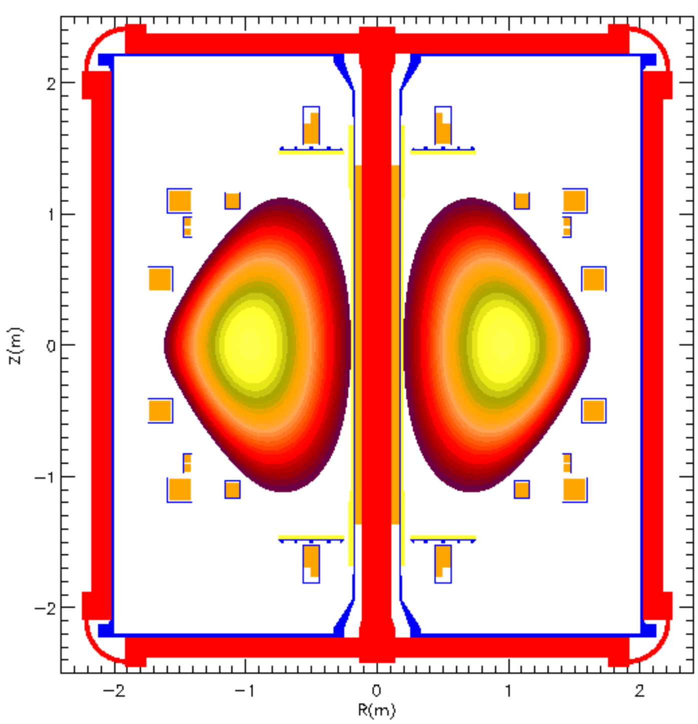
- **Operational:** investigate techniques for start - up and current drive in the ST (incl. flux consumption; induction/compression methods; NBI and RF current drive; high bootstrap fraction). Establish the location and origin of the operation limits (n,q,β). Demonstrate quasi-steady-state operation at high pressure with large non-inductive current fraction.
- **Confinement:** Test and extend the Ohmic and Aux. Heated confinement scalings, L-H transitions etc derived by START, and establish database to predict performance of ST VNS and power plant devices
- **Exhaust and Divertor:** SOL properties and wall loading will be investigated in a wide range of configurations and divertor scenarios made possible by the large vacuum tank in MAST.
- **MHD:** Test the 'disruption resilience' and low halo currents exhibited by START. Investigate whether the low levels of turbulence and absence of adverse neo-classical tearing mode effects (which were key to the high beta results in START) apply in the low-collisionality plasmas in MAST.

<u>Accomplishments</u>

- First toroidal plasmas (Dec 1998).
- First tests of NBI equipment (on loan from ORNL/US DoE) (March 99).

<u>Proposed Research Programme</u>

- Commence in Autumn 1999 with studies of both Ohmic and auxiliary heated plasmas (NBI and ECRH).



N. NSTX (Ono, Peng, U.S.), Achieved First Plasma

Stage	R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
PoP	0.86	0.68	2	5	0.3	1	11	Y

Research Goals

Investigate ST physics principles and establish database needed by the Performance Extension experiment in:

- MHD stability for high toroidal average beta
- Turbulence suppression to approach neoclassical ion confinement
- Noninductive start-up, current sustainment and profile control with large pressure-gradient-driven current fractions
- Scrape-off layer and divertor plasmas for wide dispersion of exhaust fluxes

Accomplishments

- Achieved first plasmas up to ~300 kA plasma current in inductive pulses up to ~0.15 s

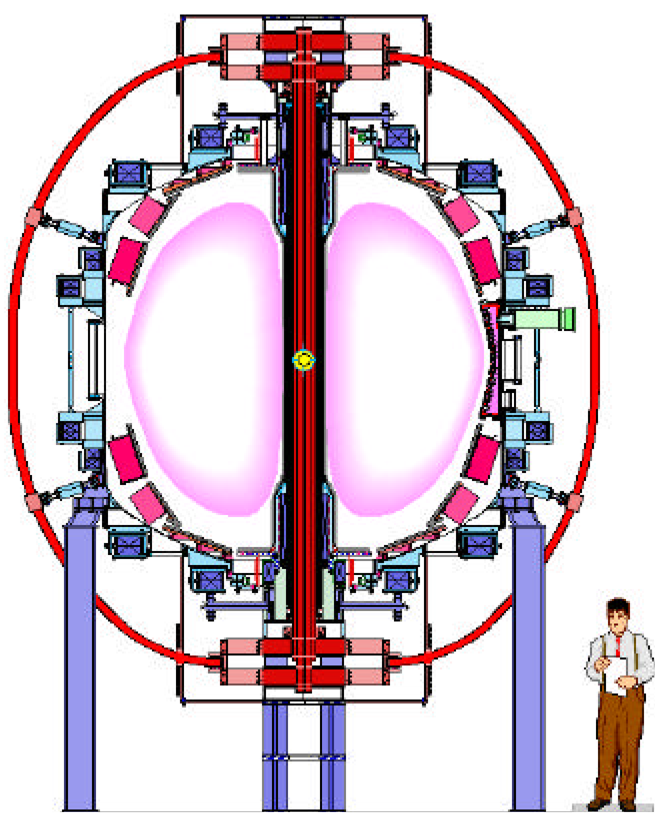
Proposed Research (in 3–5 years)

August 1999 - June 2000: Ohmic plasma properties, Tests of Coaxial Helicity Injection (CHI) current drive and High Harmonic Fast Wave (HHFW) heating

July 2000 - September 2001: Properties of ST plasmas up to the "no-wall" beta limits (β_T up to 25% and ∇p-driven current fractions up to 40%) using HHFW, NBI, and EBW

October 2001 - September 2003: Properties of ST plasmas up to the "wall-stabilized" beta limits (β_T up to 40% and ∇p-driven current fractions up to 70%) using HHFW, NBI, and EBW

October 2003 - January 2004: Install high-performance center stack for κ = 3 ST plasma experimentation at the Proof of Principle level



O. DTST (Peng, U.S.) for Performance Extension, Optimization, and Burning Plasma Testing

R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
1.17	0.83	3	50–150	1.7–0.9	10–5	40	Y

Research Goals

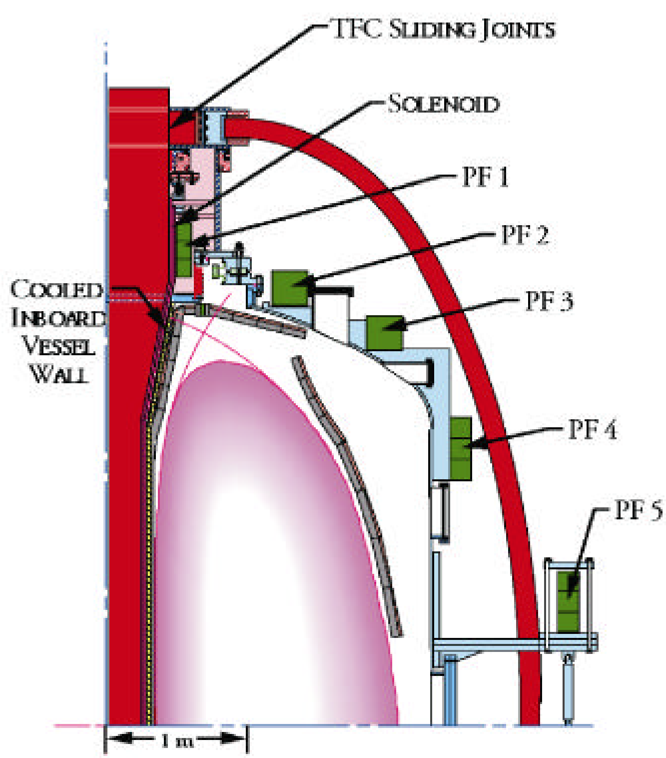
- For Performance Extension: explore ST physics at or near parameters appropriate for D-T burn to provide data needed for a cost-effective FED device
- For Burning Plasma: investigate and demonstrate fusion burn at high Q (~10), the condition that can lead to economic fusion power

Major Issues for Performance Extension Testing (Q ~ 1) in Support of Fusion Energy Development Device

- Noninductive startup to full current
- Achievement of β_T ~25% in "no-wall" regime with ∇p-driven current fractions ~50%
- Impact of NTM and ELMs, and energetic particle (NBI and fusion α) driven TAE, "fishbone" modes, etc.
- Scaling of confinement with ρ* and ν* for achieving H ~ 1.2 (ITER98H)
- Maintenance, for several energy confinement times, of these conditions at I_p ~ 10 MA and Q ~ 1
- Maintenance, for a magnetic flux diffusion time, of these conditions at I_p ~ 5 MA and B_T ~ 0.9 T in D-D operation
- Compatibility of Naturally Diverted (ND) configuration and non-carbon tiles with the above conditions
- Design and construction of demountable jointed center stack assembly

Additional Issues for Burning Plasma Testing (Q ~ 10)

- Scaling of confinement with ρ* and ν* for achieving H ~ 3 (ITER98H) in plasma dominated by α heating, via formation of internal transport barriers and approach to neoclassical ion confinement
- Impact of TAE, "fishbone" modes, driven by dominating fusion α particles
- Compatibility of dominating α heating profile (large α "banana" width) with neoclassical ion transport, pressure profile, and MHD stability, inductively maintained for several energy confinement times
- Compatibility of Double-Null (DN) and Single-Null (SN) divertor configurations with the above



P. CTF (Hender, U.K.), for Component Testing (Fusion Energy Development)

R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
0.8	0.53	2.3	s.s.	3.3	13.4	69	Y

Research Goals

Major Issues

(device figure)

Q. VNS (Cheng et al, U.S.), for Fusion Energy Development Testing

R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
1.1	0.78	3	s.s.	2.1	10	50	N

Research Goals

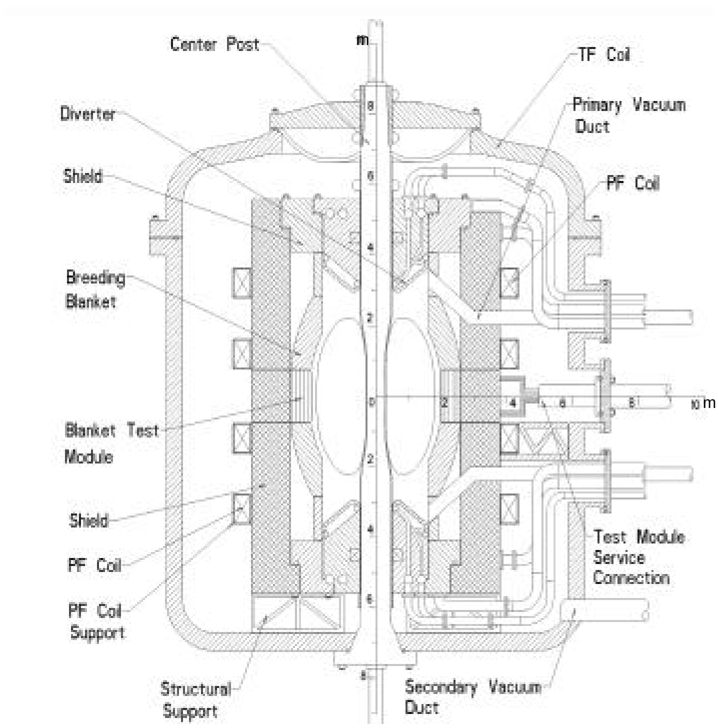
– Test fusion materials, blanket, divertor in a reactor-relevant environment
– Obtain lifetime data on materials integrated in components
– Develop component capabilities at low extrapolation risks
– Demonstrate operation of a safe, reliable, and environmentally attractive fusion system
– Capability to achieve 1-2 weeks continuous operation with neutron $W_L \geq 1$ MW/m^2
– Capability to handle total neutron fluence of 4–6 MW-yr/m^2 over 10 m^2 total testing area

Major Issues

– Steady-state driven burn (Q~1) plasma operation in the "no-wall" regime characterized by $\beta_T \leq 30\%$, ∇p-driven current fractions ~50%, confinement according to ITER98H, and driven current fraction ~50%
– Steady-state power and particle handling at ~50 MW total heating (with 60 MW fusion power) possibly via inboard limited Naturally Diverted (ND) configuration
– Steady-state plasma heating and current drive systems, such as NBI at ~400 kV, HHFW at up to ~80 MHz for a total power of ~40 MW
– Manufacturing, operation, reliability, and lifetime of demountable, water-cooled, single-turn Glid-Cop center leg for TF without significant shielding
– Safe and environmentally sound design and operation of water-cooled jointed magnets in presence of various blanket test modules cooled by liquid-metal, molten salt, or gas.

Additional Issues for Nearly Self-Sustained and Ignited Operation Like Power Plant

– Steady-state high Q (>10) plasma operation in the "wall-stabilized" regime characterized simultaneously by $\beta_T \leq 50\%$, ∇p-driven current fractions ~90%, and nearly neoclassical confinement, requiring detailed plasma profile optimization and control
– Steady-state power and particle handling, and systems operation at ~80 MW total heating (with ~260 MW fusion power) using ND, DN, or SN configuration
– Safety, environment, reliability, and lifetime of all major components at ~260 MW fusion power and ~4 MW/m^2 in neutron wall loading



R. FDF (Stambaugh, U.S.), for Fusion Development Encompassing Performance Extension, Fusion Energy Development, and Pilot Plant Testing

R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
1.12	0.7	3	s.s.	2.9	13	25	Y

Research Goals

Major Issues

(device figure)

S. ST Power Plant (Hender, U.K.), for Central Station Electricity Production

R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
3.9	2.8	3	s.s.	1.9	37	100	Y

Assumed Operational Conditions

Major Issues

(device figure)

T. ARIES-ST (Najmabadi, U.S.), for Central Station Electricity Production

R(m)	a(m)	κ	Pulse(s)	B_T(T)	I(MA)	P_{AUX}(MW)	Div?
3.2	2.0	3.4	s.s.	3.0	28	120	Y

Assumed Operational Conditions

Major Issues

(device figure)