A Fusion Development Facility on the Critical Path to Fusion Energy

V.S. Chan 1), R.D. Stambaugh 1), A.M. Garofalo 1), J. Canik 2), J. Kinsey 1), J.M. Park 2), M.Y.K. Peng 2), T.W. Petrie 1), M. Porkolab 3), R. Prater 1), M. Sawan 4), J.P. Smith 1), P.B. Snyder 1), P.C. Stangeby 5), and C.P.C. Wong 1)

1) General Atomics, P.O. Box 85608, San Diego, California 92186-5608, USA
2) Oak Ridge National Laboratory, PO Box 2008, Oak Ridge, TN 37831, USA
3) Massachusetts Institute of Technology, 77 Massachusetts Ave., Cambridge, MA 02139, USA
4) University of Wisconsin-Madison, 1500 Engineering Dr., Madison, WI 53706, USA
5) University of Toronto Institute for Aerospace Studies, 4925 Dufferin St., Toronto M3H 5T6, Canada

e-mail contact of main author: chanv@fusion.gat.com

Abstract. A Fusion Development Facility (FDF) based on the tokamak approach with normal conducting magnetic field coils is presented. FDF is envisioned as a facility with the dual objective of carrying forward advanced tokamak (AT) physics and enabling development of fusion's energy applications. AT physics enables the design of a compact steady-state machine of moderate gain that can provide the neutron fluence required for FDF's nuclear science development objective. A compact device offers a uniquely viable path for research and development in closing the fusion fuel cycle because of the demand to consume only a moderate quantity of the limited supply of tritium fuel before the technology is in hand for breeding tritium.

To make possible a fusion demonstration power plant (DEMO) of the ARIES-AT [1] type as the next step after ITER [2], a Fusion Development Facility (FDF) is needed. Here, an FDF based on the Advanced Tokamak (AT) approach with normal conducting magnetic field coils (sketch in Fig. 1) is presented [3]. As a candidate for a Fusion Nuclear Science Facility (FNSF), it is sometimes referred to as FNSF-AT. FDF would comprehensively address the scientific and technical gaps between ITER and an AT DEMO identified in recent studies [4,5].

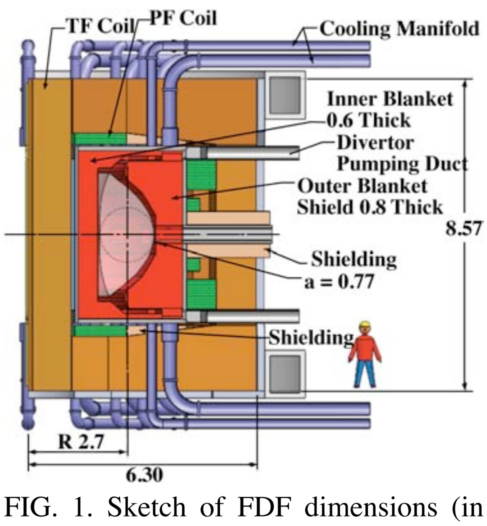

FIG. 1. Sketch of FDF dimensions (in meters) for reference. Nominal baseline parameters are $I_p = 6.6$ MA, $B_0 = 5.4$ T, $R = 2.7$ m, $A = 3.5$, $\kappa = 2.3$, $\beta_N = 3.7 = 0.67\,\beta_N^{ideal}$, $P_{fus} = 290$ MW.

Nuclear Science Mission

The main nuclear science mission of FDF (Table I) is to

- Demonstrate fusion fuel self-sufficiency
- Develop fusion blankets that make both tritium and electricity
- Obtain first data on Reliability, Availability, Maintainability and Inspectability (RAMI) of fusion nuclear components
- Test materials with high neutron fluence (2–6 MW-yr/m^2, 20–60 dpa)

Closing the fusion fuel cycle is a major goal of FDF. Net tritium production must be demonstrated before a DEMO can be committed. It is not practical to make this demonstration in the initial phase of DEMO operation, owing to the high tritium consumption rate. FDF will deploy blankets on its full first wall to enable a definitive demonstration of net tritium production. Operating durations of up to 2 weeks will enable demonstration of actual continuous closed loop tritium extraction for fusion systems. FDF will demonstrate for DEMO the whole fuel cycle including extraction, accountability, and safety in a steady-state DT device.

Table I. Key technology challenges addressed by capabilities envisioned in FDF

Key Technology Challenges	**Target**	**Capabilities**	**Technology Assumptions**
Closing the fuel cycle	TBR>1. Build T supply for DEMO 0.4 – 1.3 kg/yr.	Dedicated axisymmetric breeding blanket, changed out 3 times during FDF operating life time. Demonstrate entire T extraction cycle.	Inner blanket/shield 60 cm, change out 3-5 yrs. Outer blanket/shield 80 cm, change out 3-5 yrs. Divertor blanket/shield 50 cm, change out 3-5 yrs. Candidate blankets: He cooled solid blanket (FS), dual coolant Pb-Li blanket (FS), ODS-FS.
Fusion blanket development in test blanket module program	Test 2 blankets every 2 years as test blanket modules. In 10 years test 10 blanket approaches.	1-2 MW/m^2 14 MeV neutron flux. Continuous operation for 2 weeks. Large area, relevant gradients (heat, neutrons). Integrated testing with fluence 1-2 $MW\text{-}yr/m^2$.	Test blanket types: Solid breeders (3), liquid breeders (2), various coolants (2), advanced low activation structural materials (2). Each blanket operate for 6 months, change out every 2 yrs.
Fusion electric blanket development	Produce 300 kW from one port blanket.	High temperature (500–700°C) heat extraction. DEMO-like environment (hot, corrosive, neutrons). Demonstrate reliability to off-normal events.	Brayton cycle turbine tested at high end of temperature.
Material testing	Up to 60 dpa of DT fusion neutron irradiation in controlled environment.	Provide material test ports for: • First wall chamber materials • Structural materials • Breeders • Neutron multipliers • Tritium permeation barriers • Composites • Electric and thermal insulators • Compatibility tests in an integrated environment • Flow channel inserts for DCLL blanket option • Chamber components and diagnostics development	Capable of testing 3 batches of samples at 30,000/batch, irradiated up to 20 dpa. Material samples in ports removed at regular intervals, 10 yrs integrated exposure possible.

FDF will be designed for neutron nominal wall loading $\Gamma_n \sim 2\ MW/m^2$ into the midplane port test blanket modules and will have a goal of duty factor 0.3 for a ten year integrated fluence of 3–6 MW-yr/m^2. These are the essential capabilities for fusion nuclear technology development. FDF will test whole, full size first wall/blanket structures as well as relatively large, fully integrated and engineered components with significant neutron fluxes and fluences, relevant first wall heat and plasma fluxes, and in a real system with disruptions and other challenges. FDF will be designed with the flexibility and maintainability to allow ten test blanket variations to be tested in ten years and 1–2 change-outs of the main full tritium producing blanket. It is desirable to take one of the best performing test blanket modules and actually make a small demonstration of electricity and hydrogen production.

The challenges of tritium handling and safety will be fully engaged in FDF, as in ITER. The FDF Program will be a test bed for learning how to engineer reliable first wall/blanket structures and gain first information on reliability growth. Because FDF is research machine with the intention to make planned changes in the entire blanket structure, it will be designed with a jointed copper TF coil so the entire machine can be taken apart readily (Fig. 2), any component remotely serviced, and in particular the nuclear components (red) accessible by simple crane lift. The machine must be sufficiently reliable to achieve two week continuous operation and to achieve a 30% duty factor on a year. It must be maintainable because it is a research environment and failures must be repairable and the blankets must be changeable. Inspection of the components is an integral part of the research; we need to find out what is happening to these components in real time and in scientifically well-equipped hot cells .

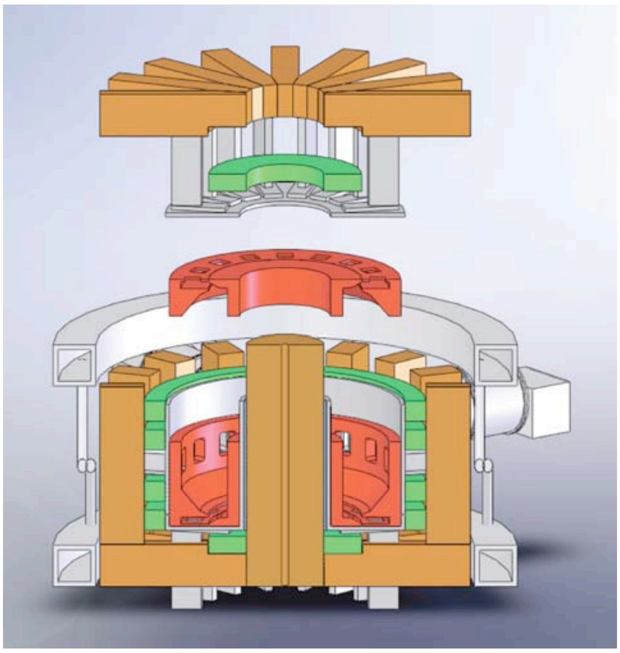

FIG. 2. FDF baseline maintenance scheme allows crane lift of toroidally continuous ring structures, assuring strength of blankets and precision toroidal alignment of the divertor surface [3].

FDF is capable of studying a broad spectrum of nuclear science issues under a broader fusion nuclear science program. Basic research is needed on test specimens and components to build a fusion nuclear database on such issues as:

- Materials compositions
- Activation and transmutation
- Materials properties, especially in the irradiated condition
- Thermo-hydraulics
- Thermal expansion and stress
- Mechanical and electromagnetic stresses
- Tritium breeding

- Tritium retention
- Tritium solubility, diffusivity, and permeation
- Liquid metals crossing magnetic fields
- Flow channel inserts
- Coolant properties
- Chemical compatibilities
- Insulators
- Leak tightness
- Foams, and more

FDF could take two ports and fill each with about a cubic meter of samples including welds and small assemblies and leave them there in controlled conditions for ten years to accumulate a fluence of 3–6 MW-yr/m^2. Samples can be removed periodically to accumulate data versus dose. Alternatively, those materials irradiation ports could be used for broad surveys at the 20 dpa level on 30,000 samples in 2 years. This basic work would support buildup of fusion core components such as: first wall and blanket assemblies, plasma interactive high heat flux components, vacuum vessels, shields, magnet components including insulators. This work is also needed on other systems and components affected by the nuclear environment such as: tritium processing systems; remote maintenance systems; instrumentation, diagnostic, and control systems; heat transport and power conversion systems; and safety and waste disposal systems.

Advanced Tokamak Mission

FDF is envisioned as a facility that can achieve the dual objective of carrying forward AT physics and enabling development of fusion's energy applications. The two elements of the objective are interrelated. Conservative AT physics enables the design of a compact steady-state machine of moderate gain that can provide the neutron fluence required for FDF's nuclear science development objective. A compact device offers a uniquely viable path for research and development in closing the fusion fuel cycle because of the demand to consume only a moderate quantity of the limited supply of tritium fuel before the technology is in hand for making net tritium. FDF is conceived with the potential to further enhance the integrated physics performance towards that of the ARIES-AT DEMO.

Scientific understanding from existing fusion facilities is providing the basis for conceptual design of an AT DEMO. However, operating parameters of existing facilities are typically different from those of an AT DEMO, e.g. operating in a fast ion mode instead of an electron-ion equilibrated high density regime; pulse durations that are orders of magnitude shorter; current profiles quite different from the strong negative shear profile expected in ARIES-AT, with 90% bootstrap current that peaks near the plasma edge.

The new superconducting tokamaks in Asia will extend the AT operating space to much longer pulse durations, and develop start-up and control methodologies that will be valuable for an AT DEMO. However, since they do not have a nuclear science mission imposing operation at high absolute plasma pressure, that particular aspect of scientific information will have to be obtained elsewhere.

ITER, being a superconducting tokamak with plans for a steady-state AT burning plasma campaign, comes closest to reactor conditions. Even here, because of the size difference, ITER would not replicate the ARIES-AT conditions. ARIES-AT operates at 80% higher β_N and requires much stronger shaping. The six times larger neutron power requirement for ARIES-AT also means a higher absolute pressure. Last but not least ARIES-AT will have a higher edge pedestal and a broad negative shear profile that impact both stability and transport.

FDF will complement ITER by approaching the ARIES-AT physics performance with valuable opportunities to fully diagnose and control the unexplored regime. Specifically, FDF can bridge the following gaps between ITER and an ARIES-AT type DEMO (Table II):

- Integration of high performance, steady-state, burning plasmas
- Maintenance and controlling of high-performance, burning plasmas
- Avoidance and mitigation of off-normal events
- Plasma modification by auxiliary systems
- Expanding predictability of integrated models

Table II. Key scientific challenges addressed by the capabilities envisioned in FDF

Key Scientific Challenges	Target	Capabilities	Technology Assumptions
Operate in true steady-state	Up to 2 weeks continuous operation, up to 10^7 s/year, 0.3 duty factor on a year.	f_{bs} 0.5-0.75 augmented by ECCD and LHCD.	TF coil stress 40 ksi, life of facility. Jointed copper TF coils for effective maintenance. OH coil stress 33 ksi, change out with inner blanket. OH half-swung provides 100% of flux need to reach full I_p. Up to 50 MW of EC, 20 MW of LH, and 5 MW of positive ion neutral beam. Launchers, couplers upgrades as desired.
Avoidance and mitigation of off-normal events	Only one unmitigated disruption per year. ELM-free operation. Operation from below to just above the no-wall limit for baseline nuclear science mission.	Active instability avoidance and suppression of RWMs, NTMs. Control system capable of initiating soft shutdowns and limiting firing the disruption mitigation system more than 20 times per year. Disruption mitigation system 99% reliable. Resonant magnetic perturbation coils or QH-mode for ELM suppression.	DIII-D -like PCS system. Feedback controlled 3-D coils and ECCD. Massive gas or pellet injection system. Disruption mitigation actuators to be developed on existing facilities. ELM-suppression and robust QH-mode operation to be developed on existing facilities.
Advanced tokamak performance	β_N extendable up to 4.5, H_{98} up to 1.6, f_{bs} up to 0.75.	Strong shaping, κ=2.3, δ=0.7 Strong NCS.	Vertical stability with feedback justified from calculations. Low l_i makes feedback control easier.
DEMO divertor heat handling	Divertor heat flux < 10 MW/m^2.	Well-aligned divertor tiles. 50% core radiation. Innovative concept testing.	Radiative divertor possible. X-divertors part of design. Super-X divertor an option.

FDF uses “conservative” assumptions of AT physics to achieve physics performance that will produce 150–300 MW of fusion power (Q<7), neutron fluence of 3–6 MW-yr/m^2, and achieve continuous operation for > 2 weeks in a compact device with 2.70 m major radius.

Table III shows the nominal parameters for FDF baseline operation (as well as back-down and more advanced scenarios) evaluated from a 0-D spreadsheet analysis [3]. The physics basis for the baseline performance indicated is either available or will be available from experiments and theory/simulations in the next few years.

Table III. FDF supports a variety of operating modes for developing fusion nuclear technology [3]

			Baseline 2 MW/m²	**Lower B, f_{bs} 1.0 MW/m²**	**Lower β_N f_{bs}, H98**	**Advanced**	**Very Advanced**
A	Aspect ratio		3.5	3.5	3.5	3.5	3.5
a	Plasma minor radius	m	0.77	0.77	0.77	0.77	0.77
R_0	Plasma major radius	m	2.70	2.70	2.70	2.70	2.70
k	Plasma elongation		2.31	2.31	2.31	2.31	2.31
J_c	Centerpost current density	MA/m²	16.7	12.0	16.7	16.7	16.7
P_f	Fusion power	MW	290	145	159	476	635
$P_{internal}$	Power to run plant	MW	500	348	527	501	492
Q_{plasma}	P_{fusion} / p_{aux}		6.9	3.5	2.9	12.4	19.8
P_n / A_{wall}	Neutron power at blanket	MW/m²	2.0	1.0	1.1	3.3	4.4
β_T	Toroidal beta		0.058	0.078	0.041	0.076	0.088
β_N	Normalized beta	mT/MA	3.69	3.69	2.65	4.50	5.00
f_{bs}	Bootstrap fraction		0.75	0.56	0.54	0.85	0.90
P_{cd}	Current drive power	MW	42	41	54	39	32
I_p	Plasma current	MA	6.60	6.39	6.56	7.09	7.43
B_0	Field on axis	T	5.44	3.90	5.44	5.44	5.44
TF Stress	Stress in TF coil	MPa	276	142	276	276	276
q	Safety factor		5.00	3.70	5.02	4.65	4.43
$T_i(0)$	Ion temperature	keV	16.4	18.2	16.4	15.0	15.5
$n(0)$	Electron density	E20/m³	3.14	1.96	2.22	4.32	5.11
$\bar{n} / n_{GR}$	Ratio to Greenwald limit		0.60	0.38	0.42	0.76	0.86
Z_{eff}			2.00	1.98	1.96	2.02	2.03
W	Stored energy in plasma	MJ	73	51	52	96	112
P_{AUX}	Total auxiliary power	MW	42	41	54	39	32
τ_E	τ_E	s	0.73	0.73	0.61	0.72	0.70
HITER98Y2	H factor over ELMY H		1.60	1.60	1.36	1.60	1.60
P_{SOL} / A_{div}	Peak divertor heat flux	MW/m²	6.7	5.2	6.8	7.3	7.6

Stability at β_N of 3.7 for FDF translates to β_N of 3.3 on DIII-D based on a theoretically justified β_N scaling of $\kappa/A^{1/2}$. DIII-D has already achieved β_N of 3.5 in a 100% non-inductive discharge. Further advances from domestic and international experiments together with theoretical understanding will allow extrapolation of FDF relevant shapes and profiles to higher β_N. Active stability control techniques can be straightforwardly adapted to FDF. Rotational stabilization of Resistive Wall Modes (RWM) has been demonstrated on DIII-D and NSTX. Moderate positive NBI power can provide sufficient edge rotation for RWM stabilization on FDF. Neoclassical Tearing Modes (NTM) stabilization using ECCD is known to work for many existing experiments. ECCD and LHCD are planned on FDF to supplement the bootstrap current and ECCD can be used for NTM stabilization. Maintaining a negative magnetic shear profile with q_{min} above 2 is also beneficial in avoiding NTMs.

FDF has to operate in steady-state and high power density. Avoidance of ELMs and disruption is essential to prevent damage to plasma facing materials and machine structures. Resonant Magnetic Perturbation (RMP) coils have shown to be effective in eliminating ELMs on DIII-D. Operation in Quiescent H-mode has also shown promise as an ELM-stable scenario. Research in the next few years should lead to a better understanding of the physics that provides confidence in extrapolating either approach to FDF.

FDF will be optimally designed to avoid disruption. Given our comprehensive understanding of MHD stability boundaries, by operating FDF with sufficient stability margin and utilizing a state-of-the-art control system, FDF should be able to operate disruption-free barring unforeseen off-normal events. In the case of off-normal events, a multitude of disruption mitigation techniques using pellets and gas jets are being tested on existing facilities with encouraging degree of success.

Most of the AT modes achieved in present-day tokamaks have quite good confinement and are limited by stability from attaining higher pressure. With improved stability, similar confinement quality would allow FDF to deliver the required fusion gain. 1-D transport modeling of FDF using physics-based GLF23 transport model [4] shows that H_{98Y2}=1.4 is sufficient to achieve the target performance, and that the confinement quality is robust. The H-mode threshold based on available scaling is sufficiently low that the available auxiliary power is more than adequate to keep the plasma in good confinement. DIII-D has many long-pulse discharges with $H_{98Y2} > 1.5$ which exceeds the FDF requirement. Many are Hybrid Mode whose current profile differs from that of FDF. There are examples of a fully non-inductive off-axis ECCD discharges with H_{98Y2}=1.4, which provide an existence proof of good confinement quality for negative magnetic shear. JT-60U has also demonstrated good confinement with H_{98Y2}=2.2 in an off-axis NBCD negative shear discharge.

FDF baseline is designed to have 75% bootstrap current with the remainder supplied by external non-inductive current drive. A special challenge in providing 25% of external non-inductive current is the compatibility of non-inductive current drive in the plasma core with a high edge pedestal. High pedestal density and temperature could prevent LH waves and positive NBI from penetrating deeper than $\rho < 0.85$. Further research is needed to alleviate the penetration issue. Transport modeling shows that 75 MW of ECCD deposited at $\rho = 0.6$ can provide 20% of the total current with 80% bootstrap current concentrated near the edge. The total current profile is favorable for stability. ECCD efficiency is not as high as LHCD so if some way of driving LHCD deeper in the plasma can be foreseen, that would be attractive. An issue that requires more research on existing experiments is start-up and plasma evolution to the desired steady-state target.

The power exhaust as estimated by a published ITER scaling is somewhat lower than that of ITER [5]. The techniques used by ITER, substantiated by results from DIII-D and Alcator C-Mod, to handle power exhaust such as core and divertor radiation should be directly applicable for FDF. Innovations such as X-divertors can further spread the heat flux. Collaborated effort among world-wide experiments to update the ITER scaling would provide further confidence in the anticipated heat flux.

FDF will also provide ample AT research opportunities to address DEMO issues toward the performance of ARIES-AT:

- The machine hardware and configuration in FDF can support extending β_N up toward 5.
- Robust, disruption-free operation will be done by knowing the stability boundary and the stabilization techniques with high accuracy and reliability.
- ITER, FDF and ARIES-AT have strongly coupled electron and ion transport. Alpha stabilization of electron turbulence at higher β_N can further improve confinement.
- Integration of high bootstrap fraction (> 80%) peaked near the edge with non-inductive current drive, high β_N and good confinement will be a feasibility demonstration of ARIES-AT physics operation.
- FDF provides an environment to study alpha particle physics uncoupled from other energetic particle drive.
- Controlling the shape in both DN and SN and optimizing stability in a plasma dominated by alpha-heating will produce valuable information for DEMO design.
- Plasma control system will show multiple-day operation of a burning plasma tokamak without interruption.
- 3-D magnetic field and rotational stabilization of edge instabilities with high pedestal β will expand control techniques to reactor relevant regimes.
- Techniques for start-up of a burning plasma leading to a steady-state with high pressure and well-aligned bootstrap current for stability can be developed on FDF.
- Innovative divertor configurations compatible with steady-state particle exhaust and shaping control can be tested.

Acknowledgment

This work supported in part by General Atomics internal funding and the U.S. Department of Energy under DE-FC02-04ER54698, DE-AC05-00OR22725, DE-FG02-90ER54084, DE-FG02-09ER54513 and supported by the Collaborative Research Opportunities Grant from the National Science and Engineering Research Council of Canada.

References

[1] NAJMABADI, F., and The ARIES Team, Fusion Engin. Design **80**, 3 (2006).

[2] AYMAR, R., BARABASCHI, P., and SHIMOMURA, Y., Plasma Phys. Control. Fusion **44**, 519 (2002).

[3] STAMBAUGH, R.D., *et al.*, “Fusion Nuclear Science Facility Candidates,” accepted for publication in Fusion Sci. Technol. 2010.

[4] CHAN, V.S., *et al.*, “Physics Basis of a Fusion Development Facility Utilizing the Tokamak Approach,” Fusion Science and Technology 57, 66 (2010).

[5] GAROFALO, A.M., CHAN, V.S., STAMBAUGH, R.D., SMITH, J.P., WONG, C.P.C., IEEE Trans. Plasma Science **38**, 461 (2010).