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Candidates for a Fusion Nuclear Science Facility (FDF and ST-CTF)

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A Fusion Nuclear Science Facility (FNSF) is needed to make possible a DEMO of the ARIES-AT type after ITER. One candidate, a conventional aspect ratio Fusion Development Facility (FDF), should have neutron flux of 1-2 MW/m^2, continuous operation for periods up to two weeks, a duty factor goal of 0.3 on a year and fluences of 3-6 MW-yr/m^2 in ten years to enable development of blankets suitable for tritium, electricity, and hydrogen production. A second candidate is the Spherical Tokamak-Component Test Facility (ST-CTF).

FDF and ST-CTF share many aspects of mission scope. FDF will develop fusion's energy applications and the operating modes needed in DEMO. FDF should be used to learn how to close the fusion fuel cycle and make electricity and hydrogen from fusion. FDF will have a goal of producing its own tritium and building a supply to start up DEMO. The size of FDF (R = 2.7 m) and the significant level of fusion power produced (290 MW) require that FDF be self-sufficient in tritium. The ST-CTF's small size (R = 1.3 m) and lower fusion power (110 MW) mean the provision of 20% of its tritium from external supply may be feasible.

In test blanket modules in ports, the development of blankets suitable for both tritium production and electricity production will be made. Both FDF and ST-CTF will provide the necessary facility to test perhaps ten different blanket concepts or variants in 2-3 ports over a 10 year time period. Actual demonstrations of electricity production (100-300 kW) should be made on the most successful test blanket modules.

With neutron fluence of 3-6 MW-yr/m^2 (30-60 dpa in 10 years) onto complete blanket structures and port material sample exposure stations (1 m^3), FDF and ST-CTF can enable irradiation qualification of materials to qualify the first years of DEMO operation.

FDF will demonstrate advanced physics operation of a tokamak in steady-state with burn. FDF will be designed using conservative implementations of all elements of Advanced Tokamak physics to produce 100-300 MW fusion power with modest energy gain (Q < 7) in a modest sized device. Conservative AT physics will enable full non-inductive, high bootstrap operation to demonstrate continuous operation of a tokamak for periods up to two weeks, a necessary step before DEMO and essential to a blanket development mission. The ST-CTF will also operate steady-state, but with very conventional physics and the majority of the plasma current driven by auxiliary power. FDF must be capable of further developing all elements of AT physics, qualifying them for an advanced performance DEMO.

To evaluate FNSF candidates, a 0-D model was constructed that incorporates relevant physics constraints and engineering constraints. A nonlinear optimizer is used to create a range of optimized machine designs over aspect ratio [1] holding constant key parameters like proximity to the beta and elongation limits, bootstrap fraction, magnet stress (<276 MPa), neutron wall loading and blanket/shield thickness, OH flux needed and constraints like upper limits on confinement H-factor (<1.6) and density ratio to the Greenwald limit (<0.8). The major radii of the optimized FDF and ST-CTF machines is shown in Fig. 1 and the fusion power in Fig. 2. For FDF, A = 3.5 (Fig. 3) was chosen to limit total facility power to 500 MW. The optimum ST-CTF was at A = 1.7. Copper TF coils with

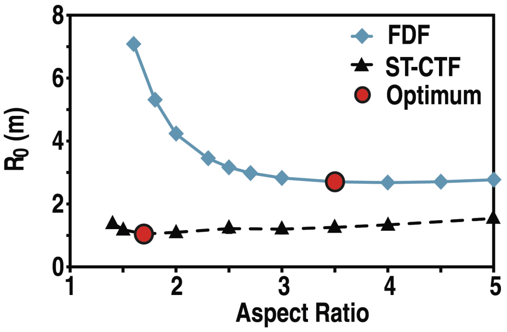

Fig. 1. Major radius vs aspect ratio for machines of the FDF and ST-CTF types.

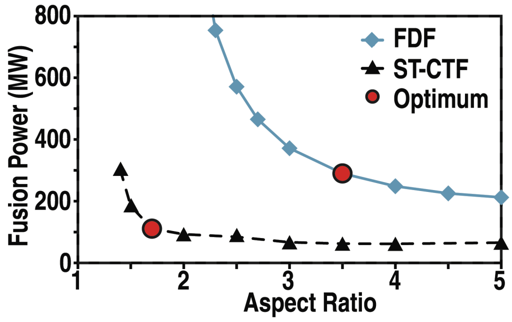

Fig. 2. Fusion power vs A.

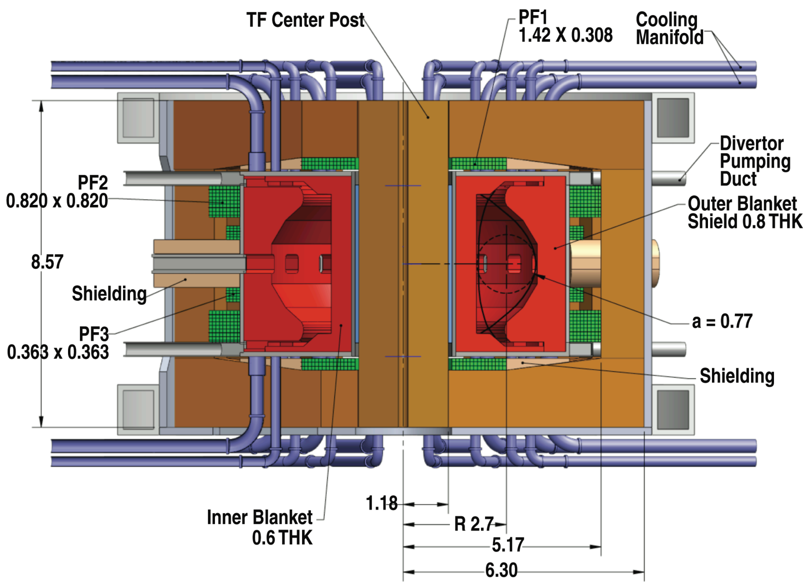

Fig. 3. Re-baselined FDF incorporating increased blanket/shield, realistic divertor geometry, plasma wall gaps.

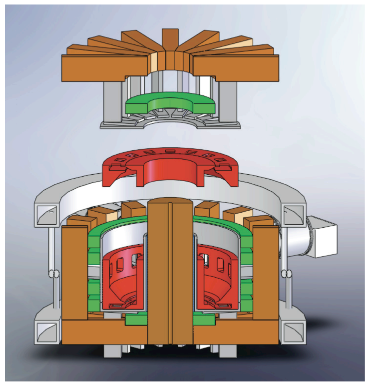

Fig. 4. FDF baseline maintenance scheme allows crane lift of toroidally continuous ring structures, assuring strength of blankets and precision toroidal alignment of the divertor surface. Red structures are the full blanket assemblies.

joints are used so that a vertical maintenance scheme (Fig. 4) can changeout axisymmetric ring blanket and divertor structures, enabling the blanket research and precision alignment of divertor surfaces.

Nominal operating points (column 1) and various operating modes are shown in Tables 1 (FDF) and 2 (ST-CTF). Common factors for FDF are elongation 2.31 at 95% of the limit, centerpost current density <17 MA/m^2, and $Z_{eff} \sim 2$. For FDF, the second and third columns are two of many lesser performance modes that preserve the nuclear mission at wall loading 1 MW/m^2. The fourth and fifth columns explore the potential of the machine to reach for ARIES-AT type parameters by increasing the fraction of the beta limit and the bootstrap fraction up to 90% (column 5); the device will support reaching for such modes if the physics allows, but the nuclear mission does not depend on such advanced modes.

For the ST, the second column is our reconstruction of the somewhat larger ST being proposed by ORNL [2,3] with more conservative physics than we assume. Common factors are elongation 2.98 at 95% of the limit, centerpost current density <60 MA/m^2, $Z_{eff} \sim 2$. Column 3 shows the performance of that machine with the more advanced physics we assume. The fourth and fifth columns explore the possible technical reach of this device if the fraction of the beta limit and the bootstrap fraction are taken up to 90% (Column 5).

Table 1. FDF Operating Modes.

		Baseline (2 MW/m²)	**Lower B, fbs (1.0 MW/m²)**	**Lower β_N fbs, H98**	**Advanced**	**Very Advanced**
A	Aspect ratio	3.5	3.5	3.5	3.5	3.5
a (m)	Plasma minor radius	0.77	0.77	0.77	0.77	0.77
R_0 (m)	Plasma major radius	2.70	2.70	2.70	2.70	2.70
P_f (MW)	Fusion power	290	145	159	476	635
$P_{internal}$ (MW)	Power to run plant	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} (MW/m^2)	Neutron power at blanket	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 (mT/MA)	Normalized beta	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} (MW)	Current drive power	42	41	54	39	32
I_p (MA)	Plasma current	6.60	6.39	6.56	7.09	7.43
B_0 (T)	Field on axis	5.44	3.90	5.44	5.44	5.44
q	Safety factor	5.00	3.70	5.02	4.65	4.43
$T_i(0)$ (keV)	Ion temperature	16.4	18.2	16.4	15.0	15.5
$n(0)$ (E20/m^3)	Electron density	3.14	1.96	2.22	4.32	5.11
W (MJ)	Stored energy in plasma	73	51	52	96	112
τ_E (s)	τ_E	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} (MW/m^2)	Peak divertor heat flux	6.7	5.2	6.8	7.3	7.6

Table 2. ST-CTF Operating Modes.

	Summary	**ST-CTF Optimum**	**ORNL (2 MW/m²)**	**ORNL FDF Physics**	**ORNL Advanced**	**ORNL Very Advanced**
A	Aspect ratio	1.7	1.7	1.7	1.7	1.7
a (m)	Plasma minor radius	0.61	0.78	0.78	0.78	0.78
J_c (MA/m^2)	Centerpost current density	59.9	42.7	42.7	42.7	42.7
P_f (MW)	Fusion power	111	177	368	557	797
$P_{internal}$ (MW)	Power to run plant	355	475	442	450	452
Q_{plasma}	P_{fusion}/p_{aux}	2.6	2.9	7.7	12.1	19.6
P_n/A_{wall} (MW/m^2)	Neutron power at blanket	2.0	2.0	4.2	6.3	9.0
β_T	Toroidal beta	0.26	0.17	0.26	0.32	0.38
β_N (mT/MA)	Normalized beta	5.19	3.49	5.19	6.20	6.98
f_{bs}	Bootstrap fraction	0.75	0.50	0.75	0.85	0.90
P_{cd} (MW)	Current drive power	42	62	47	46	41
I_p (MA)	Plasma current	8.4	12.0	11.9	12.5	13.3
B_0 (T)	Field on axis	2.77	3.11	3.11	3.11	3.11
q	Safety factor	8.2	8.2	8.2	7.8	7.3
$T_i(0)$ (keV)	Ion temperature	10.7	13.8	16.0	15.3	15.5
$n(0)$ (E20/m^3)	Electron density	5.4	3.6	4.6	6.1	7.2
W (MJ)	Stored energy in plasma	25	44	65	81	97
τ_E (s)	τ_E	0.4	0.45	0.53	0.52	0.49
HITER98Y2	H factor over ELMY H	1.60	1.25	1.59	1.60	1.58
P_{SOL}/A_{div} (MW/m^2)	Peak divertor heat flux	6.8	8.4	8.7	9.3	10.5

A candidate research plan for the FDF is shown in Fig. 5 and is probably typical of the program on ST-CTF. The plan is to make three major changeouts of the full blanket structures, and necessary maintenance operations. An initial 4-year commissioning period is envisioned in which the working fuel will progress from H to D to DT. Fusion power rises during those years to 150 MW in 10-minute pulses. The auxiliary power alone can develop the basic operating modes. A helium cooled solid breeder blanket is installed from the start; TBR will gradually be improved from about 0.9 to ~1.1 by the end of the First Main Blanket phase. Until this first main blanket starts to produce net tritium, the facility will be a net tritium consumer with a need for about 1 kg of external supply. By the end of this first phase, true steady-state operation will have been developed with duty factor 0.2, fusion power 250 MW, and wall loading 2 MW/m^2. In port blanket sites, the first two TBMs will have been tested.

Year	1–4: START UP			5–9: FIRST MAIN BLANKET		10–11	12–16: SECOND MAIN BLANKET		17–18	19–23: THIRD MAIN BLANKET	
	H	D	DT								
Fusion Power (MW)	0	0	125	125	250		250	250		250	400
P_N/A_{WALL} (MW/m^2)			1	1	2		2	2		2	3.2
Pulse Length (Min)	1		10	SS	SS		SS	SS		SS	SS
Duty Factor	0.01		0.04	0.1	0.2		0.2	0.3		0.3	0.3
T Burned/Year (kg)			0.28	0.7	0.8		2.8	4.2		4.2	5
Net Produced/Year (kg)				–0.14	0.56		0.56	0.84		0.84	1
Main Blanket	He Cooled Solid Breeder Ferritic Steel						Dual Coolant Pb-Li Ferritic Steel			Best of TBMs RAFS?	
TBR				0.8	1.2		1.2	1.2		1.2	1.2
Test Blankets				1,2			3,4	5,6		7,8	9,10
Accumulated Fluence (MW-yr/m^2)			0.06		1.2			3.7			7.6

Fig. 5. Operational and blanket development schedule of FDF.

In a 2-year shutdown, the second main blanket, dual coolant lead-lithium, will be installed. By the end of this phase, the duty factor will be 0.3 and the tritium produced per year 0.84 kg. TBMs 3, 4, 5, and 6 will have been tested. Accumulated fluence on anything that has remained in the machine all 16 years will be 3.7 MW-yrs/m^2.

The third main blanket will be built from the best result of the first two TBMs. At the end of this phase, the machine will reach for its very advanced operating modes, perhaps fusion power 400 MW, wall loading 3.2 MW/m^2, and net tritium production 1 kg. TBMs 7, 8, 9, and 10 will have been tested. Accumulated fluence lifetime will reach 7.6 MW-yrs/m^2.

FDF is deliberately configured to be ready to move to construction soon and clearly has the potential to develop DEMO advanced operating modes. It’s construction features are based on the existing tokamaks DIII-D and Alcator C-Mod. It is a rather prosaic copper coil tokamak. Its physics basis for its nominal operations is either in hand or can be sufficiently in hand in the next few years.

The ST-CTF also is a machine we are nearly ready to construct and is the smallest and lowest cost next step of the two. The ORNL version is positioned to use only very conventional, already in-hand physics. However, it requires experimental demonstration of startup without an OH transformer and has higher peak divertor heat flux. It appears to us that the transformerless startup issue can be settled in the next few years. Our analysis suggests the peak heat flux problem may not be as severe as generally thought and we suggest pathways to AT operation that may be available in the ST-CTF as constructed.

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