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MISSION AND OVERVIEW OF A FUSION DEVELOPMENT FACILITY

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ABSTRACT

The mission of the proposed Fusion Development Facility (FDF) is to fill the gaps between ITER and current experiments and a fusion demonstration power plant (DEMO). FDF should carry forward Advanced Tokamak (AT) physics and enable development of fusion's energy applications. Near term advanced tokamak physics and non-superconducting magnet technology will be used to achieve steady-state with burn, producing 100-250 MW fusion power with modest energy gain ($Q < 5$) in a modest sized device (between DIII-D and JET). FDF will further develop all elements of AT physics for an advanced performance DEMO. With neutron flux at the outboard midplane of $1–2\ \mathrm{MW/m^2}$, continuous operation for periods up to two weeks, and a goal of a duty factor of 0.3 per year, FDF can produce fluences of $3–6\ \mathrm{MW\text{-}yr/m^2}$ in ten years of operation, for fusion nuclear component research and development. The development of blankets suitable for tritium, electricity, and hydrogen production will be done first in port modules. Then, the most promising candidates will be deployed as full blankets in FDF. Two to three full blankets and about a dozen port blanket types could be tested. A goal of FDF is to demonstrate closure of the fusion fuel cycle, producing its own tritium. FDF, ITER, IFMIF, and other AT devices will provide the basis for a fusion DEMO power plant of the ARIES-AT type.

I. INTRODUCTION

The proposed Fusion Development Facility (FDF) has a dual mission of carrying forward advanced tokamak (AT) physics and enabling development of fusion's energy applications. The two elements of the mission are interrelated. 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 an attractive (perhaps the only) 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. Without advanced physics this would not be possible. In concert with existing and planned research tokamaks, ITER [1], and the International Fusion Materials Irradiation Facility (IFMIF) [2], FDF will provide the basis for a fusion demonstration power plant (DEMO) of the ARIES-AT [3] type.

In order to carry forward AT physics, FDF should demonstrate advanced physics operation of a tokamak in steady-state with burn. FDF will be designed using already proven and conservative implementations of all elements of AT physics to produce 100–250 MW fusion power with modest energy gain ($Q < 5$) in a modest sized device. The many advances made in the last decade must be captured in a next step device in order to make progress toward the even more advanced physics called for by ARIES-AT. Modest size, hence modest Q, is needed to minimize the cost consistent with the mission. FDF with $Q < 5$ does not compete with ITER for the high-energy gain burning plasma mission.

A conservative expression of AT physics refers to values of normalized beta, β_N, relative to the limiting β_N value, significantly higher than for the ITER baseline operation, and already demonstrated with sustained high performance discharges in existing devices. High β_N is required in order to utilize full noninductive, high bootstrap current scenarios 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. Besides using AT-physics for its baseline operating modes, FDF must be capable of further developing all elements of AT physics, qualifying them for an advanced performance DEMO. In practical terms this means striving for operating modes with β_N up to 5.

By realizing the volume neutron source described above, FDF will be able to develop fusion's energy applications. With neutron wall loading at the outer midplane of 1-2 $\mathrm{MW/m^2}$ and a goal of an annual duty factor of 0.3, FDF can produce fluences of 3-6 $\mathrm{MW\text{-}yr/m^2}$ in ten years of operation onto complete blanket structures and/or material sample volumes of about one m^3. This level of fluence should enable qualification of at least the first few years of DEMO operation.

Before a DEMO project can be committed, net tritium production to close the fusion fuel cycle must be demonstrated and assured. Making a first demonstration of this in the initial phase of DEMO operation is impractical, owing to the high tritium consumption rates. This assurance of tritium supply must be made first in a more modest device. FDF will have a goal of producing its own tritium and building a supply to start up DEMO using a blanket.

Additionally, in port blanket modules, the development of blankets suitable for both tritium production and hydrogen production will be made. FDF will provide a facility to test different blanket concepts or variants in 2–3 ports over a ten year time period. FDF will be the necessary facility to learn how to evaluate, design, and fabricate blankets that support high temperature, high thermodynamic efficiency for power conversion for electric power production. Although FDF will not attempt electric power production from its full blankets, demonstrations of both electricity production (300 kW) and of hydrogen production should be made on port blankets that are sufficiently successful to warrant that effort.

Other component test facilities have been proposed [4,5] that would take on some elements of the FDF mission. However, FDF would comprehensively address the scientific and technical gaps identified in the recent report by the DOE Fusion Energy Science Advisory Committee (FESAC) [6].

II. INTEGRATION OF MISSION, DESIGN, AND OPERATING MODES

FDF is envisioned as an aspect ratio 3.5 tokamak whose technical and physics basis is sufficiently in hand to allow proceeding to design and then construction in a few years time. It would be built as a direct follow-on of DIII-D [7] and Alcator C-Mod [8] with the construction features of those two machines. As indicated in the sketch of Fig. 1, FDF is between the DIII-D tokamak (major radius 1.8 m) and the Joint European Torus (major radius 3 m) [9] in size. The outermost element is a massive copper toroidal coil, steady-state water cooled and capable of 6 T on axis. The TF coil is constructed of plates like DIII-D and Alcator C-Mod, which enables easy steady-state water or oil cooling and a TF coil joint somewhere in the upper inner corner or inboard leg. This demountable coil will allow the top to be taken off the machine for full remotely operated crane lift type maintenance and changout of the blanket and divertor structures. A construction approach for the blanket/shield, vessel, and divertor based on toroidally continuous ring structures assures strength of blankets and precision toroidal alignment of the divertor surfaces.

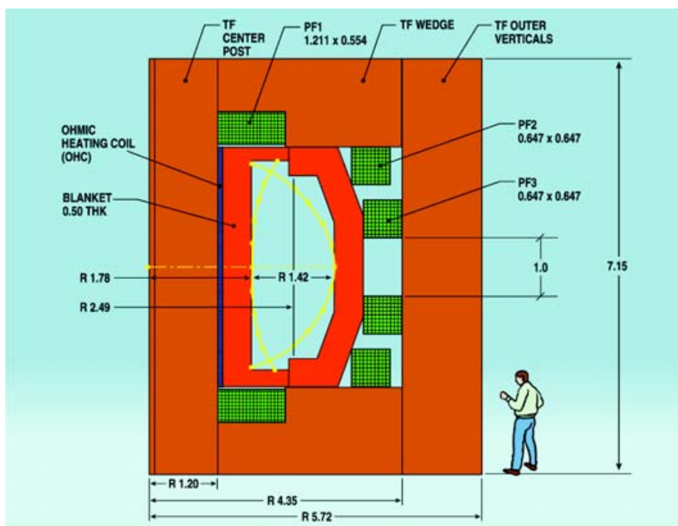

Fig. 1. FDF Dimensions for Reference. FDF is intermediate in size between DIII-D and JET and about 40% the linear dimension of ITER.

The PF coils will be inside the TF coil; their proximity to the plasma allows higher elongation and triangularity for higher performance and an overall smaller size machine. The OH coil is wound on the TF coil, allowing a small OH coil to produce enough volt-seconds to run the plasma current up to full value half-swung so that in steady-state the OH coil is near zero current. The OH coil can be made with organic insulators if the inboard 50 cm region is an optimized stainless steel and water shield; however in that case, although a tritium breeding ratio (TBR) for the whole machine can closely approach 1.0, it probably will not be possible to show $\mathrm{TBR} > 1$. If the inboard 50 cm area is a breeding blanket, then $\mathrm{TBR} > 1$ can be achieved but the lower shielding effectiveness of the breeding blanket will require the OH coil to use ceramic insulators. The OH coil is viewed as a disposable component replaced on a regular interval, perhaps with the inboard blanket and perhaps as an integral part of the inboard blanket assembly. Auxiliary systems are also planned to be similar to those on DIII-D and Alcator C-Mod with the challenge of extension to steady-state positive ion neutral beam injection (NBI), electron cyclotron heating and current drive (ECH/ECCD), and lower hybrid current drive (LHCD).

FDF has a range of operating modes. Table I shows the nominal parameters for some of the operating modes evaluated from a 0-D system optimizer model which selected the optimal aspect ratio to be 3.5, with columns for the ITER steady state scenario and ARIES-AT for comparison. The column headed Wall Load 2 $\mathrm{MW/m^2}$ is the FDF baseline case. Energy gain, Q, is a modest 4.2. Normalized beta, β_N, is 3.7, equivalent to a conservative $\beta_N = 3.3$ in DIII-D because of the elongation and aspect ratio dependence of the β_N stability limit [10]. Bootstrap current is 60%, requiring 59 MW of rf current drive to drive the remaining 40% of the current. Toroidal field is 6 T and the plasma current is 6.7 MA. Density is relatively high but still just 57% of the Greenwald limit, to increase current drive efficiency. The confinement factor $\mathrm{H_{98Y2}}$ is 1.6, comparable to what DIII-D achieves on very long pulse plasmas. Total power to run the entire facility is 507 MW in this mode.

TABLE I. FDF supports a variety of operating modes for developing fusion nuclear technology.

		Wall Load 2 MW/m2	1.0 MW/m2, Lower B, fbs	High Gain Inductive	Very Advanced	Very Advanced	ITER-SS	ARIES-AT
A		3.5	3.5	3.5	3.5	3.5	3.4	4
a	m	0.71	0.71	0.71	0.71	0.71	1.85	1.30
Ro	m	2.49	2.49	2.49	2.49	2.49	6.35	5.20
Elongation		2.31	2.31	2.31	2.31	2.31	1.85	2.20
Fusion Power	MW	246	123	231	301	401	356	1755
Plant Power	MW	507	362	395	482	536		
Pn/Awall	MW/m2	2.0	1.0	1.9	2.5	3.3	0.5	4.8
Qplasma		4.2	2.5	11.5	4.5	6.1	6.0	45.0
BetaT		5.8%	7.6%	9.2%	7.9%	7.4%	2.8%	9.2%
BetaN	mT/MA	3.7	3.7	3.3	4.5	4.5	3.0	5.4
fbs		60%	46%	30%	65%	70%	48%	91%
Pcd	MW	59	50	20	65	66		35
Paux	MW	59	50	20	67	66	59	36
Ip	MA	6.7	6.5	9.3	6.8	7.0	9.0	12.8
Bo	T	6.0	4.4	4.7	5.4	6.0	5.2	5.8
q		5.0	3.8	2.8	4.4	4.8	5.3	3.7
Ti(0)	keV	19	20	16	18	18	19	31
n(0)	E20/m3	3.0	2.0	3.5	3.5	4.1	0.7	2.9
nbar/nGR		0.57	0.40	0.47	0.66	0.74	0.82	0.96
Zeff		2.1	2.1	2.1	2.1	2.1	2.1	1.7
W	MJ	70	50	67	77	89	287	640
TauE	sec	0.6	0.7	1.0	0.6	0.6	3.1	2.0
HITER98Y2		1.60	1.60	1.36	1.59	1.60	1.57	1.40
PTotal/R	MW/m	43	30	27	51	59	21	74
Peak Heat Flux	MW/m2	8.2	6.1	3.8	9.3	10.2	10.0	9.3

The column headed 1.0 $\mathrm{MW/m^2}$ shows that reducing the toroidal field and the bootstrap current results in a reduced performance case that still delivers 1.0 $\mathrm{MW/m^2}$

into the test blankets. The nuclear technology mission is still secure in this reduced wall load case. In this reduced wall load case, 362 MW is needed to run the facility. The columns labeled Very Advanced look at raising the β_N and bootstrap fractions to move toward ARIES-AT. Achievement of these modes is an open-ended research goal for FDF; the machine hardware will be capable of such modes if the physics allows them.

III. NUCLEAR SCIENCE RESEARCH PLAN

The FDF nuclear science research plan is to close the fusion fuel cycle by using the FDF main tritium breeder blankets, and demonstrate electricity production and possibly hydrogen production by using the test port blankets. With neutron fluence at the outer midplane of 1–2 MW/m^2 and a goal of a duty factor on a year of 0.3, FDF can produce fluences of 3–6 MW-yr/m^2 in 10 years of the DT operation onto complete blanket structures and/or material sample volumes of about 1 m^3. FDF can therefore enable irradiation qualification of materials and components from the main blankets, test port blankets and material sample exposure stations. This level of fluence should enable qualification of at least the first few years of DEMO operation.

A possible schedule of the FDF nuclear science program is given in Table II. An initial four-year commissioning type period is envisioned in which the working fuel will progress from H to D to DT. Fusion power will rise to 150 MW in 10 minute pulses. The basic operating modes of the machine can be developed in this phase without dependence on the fusion power since the installed auxiliary power will be sufficient. A helium cooled solid breeder blanket will be installed from the start and the TBR will gradually be improved to 1.2 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, later to be returned. By the end of this First Main Blanket phase, true steady-state operation will have been developed with duty factor 0.2 and fusion power 250 MW and wall loading 2 MW/m^2. Net tritium produced will be 0.56 kg per year. In the port blanket sites, the first two test blanket modules (TBMs) will have been tested.

TABLE II. Operational and blanket development schedule of FDF.

Year	1–4			5–9		10–11	12–16		17–18	19–23	
	START UP H	START UP D	START UP DT	FIRST MAIN BLANKET			SECOND MAIN BLANKET			THIRD MAIN BLANKET	
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

At the end of the First Main Blanket phase there will be a two-year shutdown to change to the Second Main Blanket phase. This blanket is envisioned to be the dual coolant lead-lithium blanket. 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.

Another two-year shutdown will allow changing to the Third Main Blanket phase. 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 with fusion power reaching 400 MW and wall loading 3.2 MW/m^2. Net tritium production per year will reach 1 kg. TBMs 7, 8, 9, and 10 will have been tested. Accumulated fluence for the machine lifetime to date would reach 7.8 MW-yrs/m^2 in this plan.

IV. MATERIALS SCIENCE RESEARCH PLAN

With fluences of 3–6 MW-yr/m^2, FDF can make a significant material science research contribution on relatively large fully integrated and engineered components such as the main tritium module and the test port module. For high fluence of potential lifetime irradiations of materials >15 MW-yr/m^2 (>150 dpa) only IFMIF can produce the required fluence. In comparison, IFMIF can only irradiate a 0.5 liter volume of samples. For multiple samples material science research, FDF can take two test ports and fill each with about a cubic meter of samples including welds and small assemblies and expose them in controlled conditions for ten years to accumulate a fluence of 3–6 MW-yr/m^2 (30-60 dpa). Samples can be removed periodically to accumulate data as a function of fluence and specific control temperatures and fluid flow conditions.

Basic material damage phenomena for metallic structural materials as indicated in Table III can be addressed. Advanced fusion relevant high performance alloys like oxide dispersion-strengthened ferritic steel (ODFS), V-alloy, refractory-alloys, SiC/SiC composite, and SiC-foam as flow channel insert, tritium barriers and electrical insulators can be tested to high fluence and under controlled exposure conditions. Depending on the specific material testing requirements, key issues of thermal stress, compatibility, safety, nuclear waste and disposal, radiation damage, and lifetime limits for the first wall (FW) and blanket components can also be addressed in the relative large available testing volume from FDF.

TABLE III. Damage phenomenon for metallic structural materials in the fusion environment. (Courtesy of R. Kurtz, PNL).

Damage Phenomenon	Temperature Range, Fraction of Melting Point	Dose Level, dpa
Hardening & Embrittlement	<0.3	0.1
Phase Instabilities	0.3 - 0.6	>1
Irradiation Creep	<0.45	>10
Volumetric Swelling	0.3 - 0.6	>10
He Embrittlement	>0.5	>10
Volumetric Swelling, Irradiation Creep, & Thermal Conductivity Change in SiC	<0.4	0.01

V. PHYSICS RESEARCH PLAN

FDF uses "conservative" assumptions of AT physics to achieve physics performance that will produce 100–250 MW of fusion power ($Q < 5$), neutron fluence of $3-6$ MW-yr/m^2, and achieve continuous operation for > 2 weeks in a compact device. Table I in Sec. 1.3 shows the nominal parameters for FDF baseline operation evaluated from a 0-D spreadsheet analysis. The Physics basis for the performance indicated can be available from experiments and theory/simulation in the next few years.

The FDF device is designated as a double null plasma with very high elongation ($\kappa \sim 2.3$) and triangularity ($\delta >\sim 0.6$). Stability of $\beta_N = 3.7$ in FDF translates to a $\beta_N = 3.3$ on DIII-D based on a theoretically justified β_N scaling of $\mathrm{K}/A^{1/2}$ with K increasing with shaping and A the aspect ratio [10]. In DIII-D, $\beta_N = 3.5$ has been sustained in fully noninductive operation with a fully aligned current profile [11]. DIII-D has also sustained $\beta_N \sim 4$ with no current driven by the Ohmic heating transformer [12], but in that scenario about 20% of the plasma current was driven by ramping the toroidal field to obtain significant off-axis current drive. 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 developed in present devices for Resistive Wall Modes (RWM), Edge Localized Modes (ELM) and Neoclassical Tearing Modes (NTM) can be straightforwardly adapted to FDF.

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-free scenario. 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 without disruptions. Disruption mitigation techniques with high reliability will be implemented to mitigate 99% of those disruptions that do happen, in case of off-normal events.

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. DIII-D has many long-pulse discharges with $\mathrm{H}_{98Y2} > 1.6$ [13] which exceeds the FDF requirement. 1-D transport modeling of FDF using physics-based GLF23 transport model corroborates that $\mathrm{H}_{98Y2} \approx 1.6$ is achievable and that the confinement quality is robust.

The FDF baseline assumes 60% of bootstrap current with the remainder supplied by external non-inductive current drive. The strong shaping of the FDF plasma cross section is predicted to allow pedestal heights in the range required for an ARIES-AT class device $\left(\beta_{N,ped} > 1\right)$. High pedestal density and temperature prevent LHCD waves and positive ion NBI from penetrating deeper than $\rho < 0.85$. Transport modeling shows that 75 MW of ECCD deposited at $\rho = 0.6$ can provide 20% of the total current with 80% of bootstrap current concentrated near the edge. The total current profile is favorable for stability. Alternative operating modes using internal transport barriers would be attractive to lower the pedestal density for more efficient LHCD to penetrate deeper in the plasma.

The power exhaust for FDF as estimated by a published ITER scaling is somewhat lower than that of ITER. The techniques used in ITER design such as core and divertor radiation should be directly applicable for FDF. Additional tools such as resonant magnetic perturbation (RMP) coils can further spread the heat flux.

VI. SUMMARY

To enable a DEMO to be built after ITER, a Fusion Development Facility is needed to develop fusion's energy applications, close the fusion fuel cycle, develop blankets for electric power from fusion, develop blankets for hydrogen production from fusion, address nearly all gaps identified by the FESAC Planning Panel, motivate the needed, large, supporting fusion nuclear science program, provide a materials irradiation research facility, and qualify Advanced Tokamak physics for DEMO. This FDF will provide the leading element in a fusion nuclear science "laboratory" broadly construed as the collection of national resources in people, intellectual assets, institutions, laboratories, and facilities needed to carry out fusion research's nuclear science era. ITER will provide high energy gain burning plasma physics and power plant scale superconducting technology. IFMIF could provide high neutron fluence materials data on small samples. These three research facilities, supported by current experiments, are the necessary combination of resources to proceed to a DEMO after ITER.

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