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GENERAL ATOMICS

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ABSTRACT

A Fusion Nuclear Science Facility (FNSF) is necessary to make possible a DEMO of the Advanced Tokamak (AT) type after ITER. One candidate, Fusion Nuclear Science Facility-AT (FNSF-AT), should have neutron wall loading of 1-2 MW/m^2, continuous operation for periods of up to two weeks, a duty factor goal of 0.3 on a year and neutron fluence of 3-6 MW-yr/m^2 in ten years to enable development of blankets suitable for tritium and electricity production while demonstrating all the critical elements necessary for the qualification and design of a DEMO. FNSF-AT, also called FDF, will be designed using conservative implementations of all elements of AT physics to produce 150-300 MW fusion power with modest energy gain (Q<7) in a modest sized normal conducting coil device. It will demonstrate and its results will help in the selection of the DEMO tritium breeding blanket concept. It will demonstrate the tritium fuel cycle, the behavior of candidate plasma facing materials, design, and the design and cooling of the first wall chamber and divertor components. It will also provide experience in safe operation and remote maintenance necessary for the DEMO design.

I. INTRODUCTION

For the long-range strategic plan of the US Magnetic Fusion Energy (MFE) program, and with the construction and operation of ITER, the referenced study examined the question of what else in addtion to ITER is necessary for the construction of a net electric producing Demonstration Power Reactor (DEMO) (Ref. 1). In addition to the ongoing confinement research and the burning plasma research on ITER, a Fusion Nuclear Science Program (FNSP) is needed to close the identified gaps. We will need to build a high power steady-state neutron source that can enable the Fusion Nuclear Science development. A candidate for the high power steady-state neutron source is the normal aspect ratio copper magnet tokamak, the Fusion Nuclear Science Facility-Advanced Tokamak (FNSF-AT), also called Fusion Development Facility (FDF), described in this paper.

II. MISSION

The mission for the FNSF-AT is the following. On the physics operation:

- Demonstrate advanced physics operation of a tokamak in steady-state with burn.
- Utilize conservative expressions of all elements of Advanced Tokamak physics to produce 150-300 MW fusion power with modest energy gain (Q<7) in a modest sized device.
- Utilize full noninductive, high-bootstrap operation to achieve continuous operation for >2 weeks.
- Further develop all elements of Advanced Tokamak physics, qualifying them for an advanced performance DEMO.

For the development of fusion nuclear technology:

- Test materials and components to high neutron fluence (3-6 MW-yr/m^2) with duty factor of 0.3 per year with maintainability.
- Demonstrate tritium self-sufficiency.
- Develop helium-cooled fusion blankets that produce tritium and electricity at 1-3 MW/m^2 neutron fluxes.
- Develop advanced fusion blankets for generation of high temperature process heat.
- With ITER and material irradiation facility, provide the database for a fusion DEMO Power Plant.

The definition of AT operation includes the following: integration of high performance steady-state operation, maintenance and controlling of high-performance burning plasmas, avoidance and mitigation of off-normal events plasma modification by auxiliary systems and expanding the predictability of integrated models.

III. DESIGN OF FNSF-AT

For the design of FNSF-AT, although it does not attempt for net electric power production from its full blankets, it is aiming for self-sufficiency of tritium production. With neutron flux at the outer midplane of 1–2 MW/m^2 and a goal of a duty factor on a year of 0.3, it can produce neutron fluences of 3–6 MW-yr/m^2 in 10 years of operation with the full blanket structures and/or material sample testing volumes of about 1 m^3. This enables irradiation qualification of materials and structural assemblies in port test module stations. This level of fluence should enable qualification of at least the first few years of DEMO operation.

To facilitate the necessary change out for the testing of different nuclear components and to minimize the capital cost of the FNSF-AT, a copper coil tokamak with a normal aspect ratio was selected with a fusion power output of 150–300 MW as shown in Fig. 1. This is essentially a research device, enabling fusion blanket and nuclear research and development. A systems approach with the EXCL spreadsheet was used to develop different design options.[2] The baseline design and corresponding variations are summarized in Table I.

Figure 2 shows the vertical maintenance approach of the FNSF-AT machine. It shows the vertical removal of the upper sections of the toroidal field (TF) coil, the top divertor coil and the top of vacuum vessel. In order to access the blanket structure, the blanket segments are removed as toroidally continuous rings. This unique design approach has the benefits of strong blankets to withstand electromagnetic (EM) loads and toroidal alignment is assured to sub-mm scale. The obvious difficulties of this design are the provision of services (coolant) to blanket rings near the midplane through blankets from above.

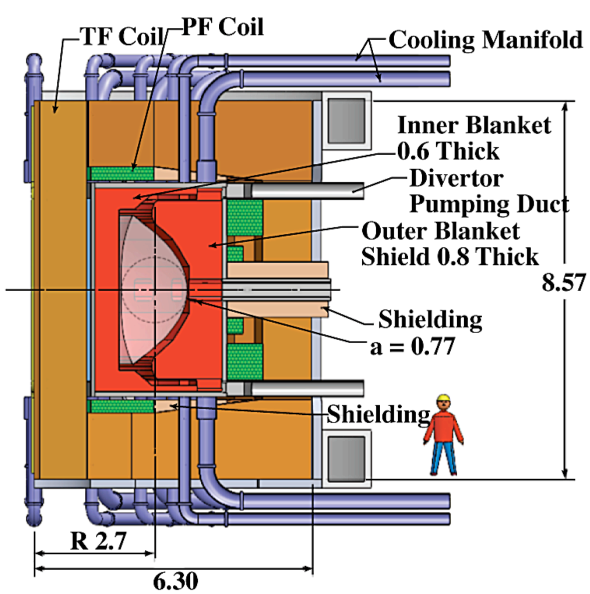

Fig. 1. FNSF-AT schematic showing major radius at 2.7 m, minor radius at 0.77 m, and a machine height of 8.57 m.

Table I
FNSF-AT Baseline and Variations of Key Parameters

		Baseline (2 MW/m^2)	Lower B, f_{bs} (1 MW/m^2)	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_o	Plasma major radius (m)	2.70	2.70	2.70	2.70	2.70
κ	Plasma elongation	2.31	2.31	2.31	2.31	2.31
J_c	Centerpost current density (MA/m^2)	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/m^2)	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	4.59	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	6.389	5.56	7.09	7.43
B_o	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	15.4
n(0)	Electron density (x10^{20}/m^3)	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
HITER98 Y2	H factor over ELMy H	1.60	1.60	1.36	1.60	1.60
P_{SOL}/A_{div}	Peak divertor heat flux (MW/m^2)	6.7	5.2	6.8	7.3	7.6

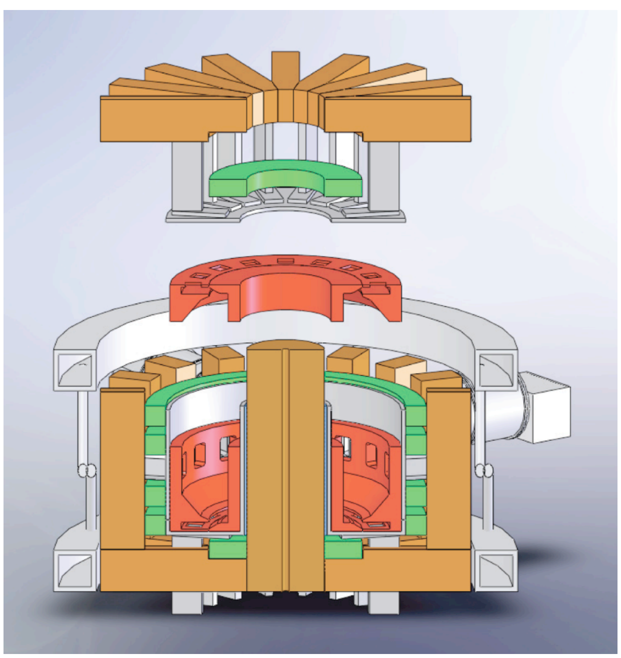

Fig. 2. FNSF-AT baseline maintenance scheme allows crane lift of toroidally continuous ring structures, assuring strength of blankets and precision toroidal alignment of the divertor surface. Red structures are the full blanket assemblies.

IV. BLANKET DESIGN AND ASSESSMENT

Three-dimensional neutronics analysis was performed using the CAD model of the baseline design of FNSF-AT. Two blanket concepts were considered: Dual Coolant Lead Lithium (DCLL), and Helium-Cooled Ceramic Breeder (HCCB) (Ref. 3). Based on the neutron wall loading results, the fusion power can be reduced from the system calculation of 290 MW fusion to 240 MW fusion while yielding a peak outboard (OB) neutron wall load (NWL) of 2 MW/m^2 and a fluence of 6 MW-yr/m^2. IB and OB blanket/shield thicknesses of 0.6 and 0.8 m, respectively, are used. The tritium breeding ratio was shown to be adequate for both blankets at values of 1.09 for the HCCB and 1.0 for the DCLL, respectively. This includes a minor blanket configuration addition in the divertor region behind the heat removal surface and is without the insertion of tritium breeding blankets in all of the 16 test ports. Lost coverage in the 16 ports amounts to a ~6% penalty in tritium breeding ratio (TBR). Since several of these ports will be utilized for advanced breeding blanket testing, we expect these test modules to contribute to the overall TBR. It is clear that FNSF-AT has the potential for achieving tritium self-sufficiency. The design would allow construction of the TF and ohmic heating (OH) coils with conventional organic insulators; this drives the required inboard (IB) blanket/shield thickness. The FNSF-AT allows for neutron wall loadings as high as 2 MW/m^2 and fluences of 3-6 MW-yr/m^2 in 10 years of operation to enable the achievement of its mission. It is assumed that the blanket, vacuum vessel (VV), and OH coil are replaced at one-third the machine lifetime, while the TF magnet is a lifetime component. The ferritic steel (FS) VV is re-weldable during the lifetime of the machine. Modest nuclear heating, atomic displacements and conductor resistivity increase would occur in the TF, OH, and poloidal (PF) coils. While the cumulative end-of-life organic insulator dose levels in the TF and OH coils are acceptable, it is recommended that the PF coil in the divertor region be moved vertically farther away from the midplane to allow adding ~15 cm of shielding to reduce the peak insulator dose to an acceptable level.

V. CONTRIBUTION TO NUCLEAR SCIENCE

For nuclear science development, FNSF-AT is a DT device that generates fusion power and contributes to the progress toward fusion energy in ways listed below:

- Produce significant fusion power in true steady state (<3 dpa).
- Shows fusion can make its own fuel with initial results at <3 dpa and full blanket development program at 10-20 dpa.
- Extracts high-grade process heat from fusion reactions with innovative blanket designs, with initial results at <3 dpa and full results at 10–20 dpa.
- Shows electricity production of 300 kW from one of the first test blanket modules (<3 dpa).
- Shows fusion chambers can survive high plasma, neutron and helium fluence with results obtained for at least three blanket types (10–20 dpa for each).
- Develops plasma measurement diagnostic systemss suitable for a DEMO (10–20 dpa).
- Obtains high fluence irradiation data on materials, assemblies, welds, etc. (30–60 dpa).

The selected relatively low fluence of <3 dpa is based on the present available reliable data base for the reference reduced activation ferritic martensitic (RAFM) steel fusion structural material for the first series of full blanket and test modules. Higher dpa exposure will take place with the increase of the high dpa material database. FNSF-AT will operate as a user facility and support large National User Teams, with the goal of 1000 users.

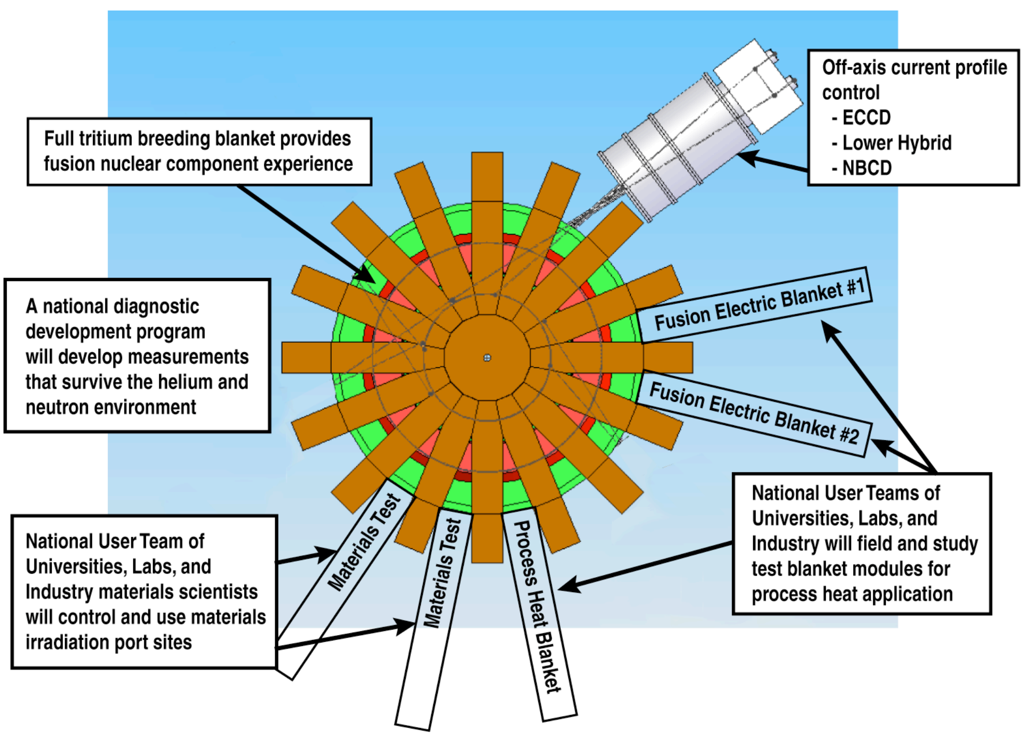

Fig. 3. Current drives and arrangement of different test blankets and test ports.

Figure 3 shows the possible arrangements of off-axis current plasma profile control hardware and different locations of test modules for tritium breeding blanket, diagnostics development and the testing of different innovative blanket designs.

VI. DIAGNOSTIC DEVELOPMENT

DT fusion diagnostics development is a key element in the operations of FNSF-AT. Not only are robust diagnostics necessary during the testing phase of FNSF- AT, but they are also an absolute necessity for feedback control for the operation of a steady state FNSF-AT and the corresponding extension to DEMO. Figure 4 shows the necessary diagnostics development sets during the first 23 years of FNSF-AT operation.

Year	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23

	START UP			FIRST MAIN BLANKET			SECOND MAIN BLANKET			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	2.8		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 ODS Ferritic Steel	
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 ($MW\text{-}yr/m^2$)			0.06		1.2			3.7			7.6
				ITER-like set (start)			Reduced set			DEMO-like set	

Fig. 4. FNSF-AT operation schedule[3] and diagnostics sets.

The first set is ITER-like, covering the need to validate physics, verify plasma and first wall blanket performance and optimize the plasma. The second is a reduced set covering reduced profile and first wall blanket information. It should generate reliability data for different diagnostics and first wall blanket components which require detailed physics and engineering predictive models. The third set provides a DEMO-like environment for testing the optimized, minimal, reliable set needed for steady-state plasma and first wall blanket operation of DEMO.

VII. DISRUPTION AND ELM HANDLING TARGET

A key operational issue for tokamaks is the handling of transient events like disruptions and edge localized modes (ELMs). FNSF-AT is designed to mitigate the serious impacts from these events. For disruption the following will be applied:

- Real-time stability calculations in the control loop.
- Active instability avoidance and suppression of – resistive wall modes and neoclassical tearing modes.
- Control system good enough to initiate soft shutdowns and limit firing the disruption mitigation system to no more than 20 times per year.
- Disruption mitigation system 99% reliable.

For ELM suppression the following will be applied:

- Resonant magnetic perturbation coils and/or pellet injection.
- QH-mode and/or other ELM-free high-performance plasma regime.

Since transient event could be most damaging for tokamak operation, targets for disruption handling were developed as given in Table II.

Table II
Targets for Disruption Handling

Device	ITER	FNSF-AT	DEMO
Pulse length (s)	400	1×10^6	3×10^7
No. of pulses/yr	1000	10	1
Fast shutdowns/yr	100	20	5
Time between fast shutdowns (s)	4×10^3	5×10^5	6×10^6
Unmitigated disruptions/yr	5	1	0.3

VIII. PFC ISSUES

Critical issues of plasma wall interactions and Plasma Facing Components (PFC) will be addressed in FNSF-AT. These include:

- Hot wall operation (>400°C) will be an entirely new regime.
- Erosion
  - A significant contribution of FNSF-AT is its operation time, 10^7 s/yr, and corresponding high neutron and particle fluence (100 times ITER).
  - Divertor gross annual erosion estimates are in the mm range for tungsten and up to cm range for carbon.
  - Tons of material per year will erode and redeposit. Questions on deposited material removal and tritium inventory will be addressed.
  - Questions on suitable plasma facing materials such as tungsten, Si-W, and operation with detached divertor plasmas and other options will be addressed.
- Tritium retention
  - Cannot be allowed to prevent TBR > 1.
- Heat flux handling
  - Axisymmetric maintenance scheme allows precision alignment of surfaces to hide edges and allow maximal use of flux expansion, progressing through the options of x-divertor, snowflake divertor, super x-divertor and other configurations.
  - High plasma density helps promote neutral radiation and containment of neutrals in the divertor, helping to separate the core and divertor plasmas.
- Other issues
  - Fast plasma shutdowns, first wall heat fluxes in fault conditions, EM forces.

IX. INTEGRATED NUCLEAR TECHNOLOGY RESULTS

The key nuclear technology development issue will be provided through the operation of FNSF-AT. Much of the necessary development data will be supplied in an integrated manner:

- Show high performance, steady-state, burning plasmas operating for weeks.
- Demonstrate net tritium production and fusion fuel sufficiency with main breeder blankets.
- Develop advanced tritium breeding blankets.
- Show diagnostics and control of plasma and high performance blankets for weeks to months.
- Develop plasma and component measurements suitable for DEMO.
- Show significant fusion power can be produced with a significant duty factor.
- Show avoidance and mitigation of off-normal and transient events.
- Develop auxiliary systems for true steady-state operation.
- Develop candidate plasma facing materials and robust PFC.
- Demonstrate the production of high-grade heat for power conversion.
- Show electricity production from fusion with selected test blankets.
- Establish remote handling and maintenance technology.
- Develop and perform irradiation tests on low activation, high strength, high temperature PFC, structural and functional materials under high neutron and helium fluence.
- Provide component operation database, including failure modes, failure rates and mean times to failure for DEMO.
- Establish fusion facility safety and operational database.

X. CONCLUSION

For the US fusion program to prepare for the design of DEMO, FNSF-AT is a necessary complement to ITER and as a dedicated component irradiation facility for DEMO development with high helium and neutron fluence. FNSF-AT will provide the necessary physics and technology operational database for DEMO and will demonstrate and provide results to enable the selection of the DEMO tritium breeding blanket and divertor design. It will demonstrate helium coolant technology and associated power production and also provide data on safe operation and remote maintenance necessary for DEMO.

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