GA–A26280

MISSION AND OVERVIEW OF A FUSION DEVELOPMENT FACILITY

by
R.D. STAMBAUGH, V.S. CHAN, A.M. GAROFALO,
J.P. SMITH and C.P.C. WONG

SEPTEMBER 2008

GENERAL ATOMICS

DISCLAIMER

GA–A26280

MISSION AND OVERVIEW OF A FUSION DEVELOPMENT FACILITY

by
R.D. STAMBAUGH, V.S. CHAN, A.M. GAROFALO,
J.P. SMITH and C.P.C. WONG

This is a preprint of a paper to be presented at the 18th Topical Meeting on the Technology of Fusion Energy, San Francisco, California on September 28–October 2, 2008 and to be published in the *Proceedings.*

Work supported by
General Atomics Internal Funding

GENERAL ATOMICS PROJECT 40010
SEPTEMBER 2008

GENERAL ATOMICS

1.0 MISSION AND SCOPE SUMMARY

A Fusion Development Facility (FDF) is proposed to make possible a fusion demonstration power plant (DEMO) as the next step after ITER. To make possible an advanced DEMO of the ARIES-AT type the mission of the FDF should be:

To carry forward Advanced Tokamak physics and enable development of fusion's energy applications.

This two part mission may be further elaborated. For AT physics, FDF should ***demonstrate advanced physics operation of a tokamak in steady-state with burn***. FDF must be the first tokamak designed using already proven and conservative implementations of all elements of Advanced Tokamak 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 (we envision a device between DIII-D and JET in size) is needed to minimize the cost consistent with the mission. Even so, the cost will be substantial and the ambition of the mission must match the cost. Modest size means modest Q; in tokamaks size and Q are strongly coupled. FDF with $Q<5$ does not compete with ITER for the high energy gain burning plasma mission.

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. 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***.

By realizing the volume neutron source described above, FDF will be able *to* ***develop fusion's energy applications***. 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 ten years of operation onto complete blanket structures and/or material sample volumes of about 1 m^3. This level of fluence should enable qualification of at least the first few years of DEMO operation. This fluence is less than the 15 MW-yr/m^2 to show lifetime irradiation of materials in IFMIF, but IFMIF will only irradiate a 0.5 liter volume of samples and not with realistic heat and neutron dpa gradients possible in FDF.

Before a DEMO project can be committed, ***net tritium production must be demonstrated and assured***. We do not believe it is practical to first make this demonstration in the initial phase of DEMO operation, 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. The approach taken will be to engineer a first full blanket with the simplest technology that just produces net tritium.

All other design requirements are secondary. Then in parallel, more advanced blankets will be tested in port blanket modules and successful ones will then be engineered into second generation full blankets. FDF will be designed to facilitate changeout of the full first wall/blanket structures and will do so 1–2 times in the life of the project.

In the port blanket modules, ***the development of blankets suitable for both tritium production and electricity production will be made***. FDF will provide the necessary facility to test perhaps ten 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 make blankets that support high temperature, high thermodynamic efficiency for power conversion for electric power production. Another port site should be devoted to ***the development of blankets that can support hydrogen production***, which can require even more demanding temperatures of extracted coolant, over 900°C. Although FDF will not attempt electric power production from its full blankets, actual demonstrations of both electricity production (300 kW) and of hydrogen production (one metric ton per week) should be made on port blankets that are sufficiently successful to warrant that effort.

The above mission elements for FDF, with ITER and IFMIF, and other AT devices, will ***provide the basis for a fusion DEMO power plant of the ARIES-AT type*** (Fig. 1).

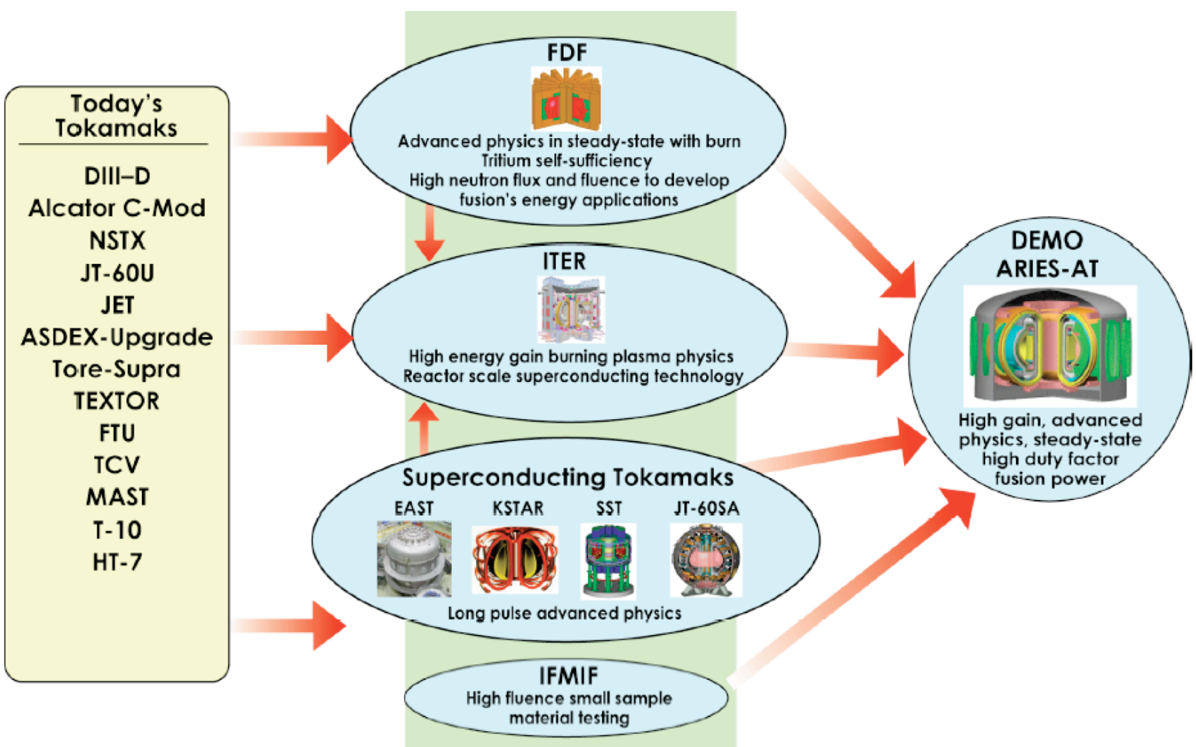

Fig. 1. FDF integrates Advanced Tokamak physics, burning plasmas, and fusion nuclear technology for DEMO.

2.0 FDF'S ROLE IN FILLING RESEARCH GAPS TO DEMO

FDF will fill in all the gaps between ITER, the new superconducting tokamaks, and IFMIF and the DEMO. Figure 2 summarizes the issues that need to be addressed to be able to build a DEMO and what today's existing or committed tokamaks (including the new superconducting tokamaks), ITER, FDF, and IFMIF can and are expected to do to resolve these issues. This package of research facilities should be sufficient to resolve all issues to proceed to a DEMO. We first discuss those issues in which the primary contribution will come from ITER and then those issues to which FDF will make the primary contribution.

Issue	Today's Exp'ts	ITER	FDF	IFMIF	ITER +IFMIF +FDF	DEMO
High Gain Q > 10		3	2		3	R
Alpha Containment & Physics	1	3	2		3	R
Confinement at Large Size	1	3	1		3	R
Pulsed Heat Loads	1	3	1		3	R
Reactor Scale Superconducting Technology	1	3			3	R
Exhaust Power Handling (~10 MWm^{-2})	1	3	3		3	R
Tritium Handling and Safety	1	3	3		3	R
Integrated Plasma Performance in SS	1	2	2		3	R
Steady-State @ High Beta (β_N, f_{bs})	1	2	3		3	R
High Neutron Wall Loading (Γ_n ~ 2 MWm^{-2})	1	2	3		3	R
Tritium Self-Sufficiency (TBR > 1)		1	3		3	R
PFC and Divertor Materials Lifetime	1	2	3		3	R
FW/Blanket Materials/Components Lifetime		1	3	1	3	R
Materials Characterisation (>100 dpa)		1	2	3	3	R
High Temperature Blankets (electricity, H_2)		2	3		3	R

Key:

1	Will help to resolve the issue
2	Will contribute significantly to resolution of the issue
3	Should resolve the issue
R	Solution is essential

Today's Expt's = DIII-D, C-Mod, NSTX, JT-60U, JET, ASDEX-U, Tore Supra, JT-60 SA, KSTAR, EAST, SST-1

Fig. 2. ITER, FDF, IFMIF, and today's experiments enable a DEMO.

High Gain (Q > 10). Exploration of this burning plasma physics regime is a mission unique to ITER. FDF makes a meaningful contribution with Q up to 5.

Alpha Containment and Physics. Here again, ITER provides the essential burning plasma information. But FDF makes a meaningful contribution with its modest gain, significant fusion power, reactor level alpha beta and ratio of alpha speed/Alfven speed.

Confinement at Large Size. ITER will make the unique contribution of confinement data at low ρ^*.

Pulsed Heat Loads. Since the plasma stored energy in ITER will be about five times that of FDF, ITER has more challenges in such pulsed heat loads as disruptions and ELMs.

Reactor Scale Superconducting Technology. Here is another unique ITER contribution with its reactor size all superconducting coils. FDF is a copper coil machine to keep the size down and enable effective maintenance.

Exhaust Power Handling. Here the contributions of ITER and FDF are comparable since the peak heat fluxes expected onto divertor components range up to 10 MW/m^2. Even at 400 seconds, ITER must engineer steady-state heat removal. FDF's P/R values span from ITER to ARIES-AT, depending on how FDF is operated, but its nominal peak heat fluxes stay below 10 MW/m^2.

Tritium Handling and Safety. The challenges in this area are shared equally by ITER and FDF.

Integrated Plasma Performance in Steady-State. Here both ITER and FDF are rated as contributing significantly to resolution of the issue and the combination of both efforts is needed to provide the necessary basis for DEMO. ITER will look at very long pulse issues at high energy gain and FDF will look at true steady-state but at modest energy gain.

Steady-State at High Beta (High β_N and Bootstrap Fraction). Here the main contribution will come from FDF since it plans to fully embrace reactor level β_N operation through an optimally designed RWM stabilization coil system and with substantial plasma rotation to enable high bootstrap fraction operation with significant fusion gain. Auxiliary H&CD systems will be optimized for plasma rotation and current drive. FDF aims to show operation for arbitrary time durations, days to two weeks. While ITER will certainly make an important contribution here, how much it can contribute will depend on the result of the ongoing design review which will decide whether RWM coils will be implemented; whether the ITER plasma can be rotated fast enough; whether there will be sufficient off-axis current drive for AT modes; whether ITER startup can support AT modes; and whether ITER can implement ELM suppression.

High Neutron Wall Loading (Γ_n ~ 2 MW/m^2). FDF will make the definitive contribution here since it will be designed for Γ_n ~ 2 MW/m^2 into the midplane port blanket modules and will have a goal of duty factor 0.3 for an integrated fluence of 3–6 MW-yr/m^2. These are essential capabilities for fusion nuclear technology development. ITER's goals are 0.5 MW/m^2 midplane neutron flux and a lifetime fluence of 0.3 MW-yr/m^2. FDF will be about 1/10 and ITER about 1/100 of reactor fluence.

Tritium Self-Sufficiency (TBR>1). FDF has this squarely as a major goal. Net tritium production must be demonstrated before a DEMO can be committed. The ITER Test Blanket Module Program is outside the project scope and under current consideration. In any case, the TBM program will be far short of a large area tritium production demonstration. The limited pulse length on ITER (perhaps as high as 3000 seconds) may not allow an adequate demonstration of continuous extraction of tritium from the test blanket modules. FDF will develop blankets in port modules at 1–2 MW/m^2 neutron fluxes and will deploy blankets on the 130 m^2 area of its first wall (or at least the outer wall) to enable a demonstration of net tritium production. Operating durations of up to 2 weeks will enable demonstration of the kind of actual continuous closed loop tritium extraction to be used in fusion systems. FDF will demonstrate the whole fuel cycle including extraction, accountability, and safety issues of a steady-state DT device to pave the way for DEMO.

PFC and Divertor Materials Lifetime. The issue here is erosion of plasma facing surfaces. With ten times greater plasma fluence onto surfaces, FDF will make the major contribution. ITER will contribute significantly.

FW/Blanket Materials and Components Lifetime. This issue could be phrased much more broadly. Fusion has yet to capture its first fusion neutron in a blanket. Everything in combined first wall/blanket development remains to be done experimentally. FDF will test whole, real size first wall/blanket structures 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 changeouts of the main full tritium producing blanket. Further, first wall materials and structures and near first wall components like rf launchers and diagnostics will be developed in a fusion relevant environment. FDF will be a test bed for learning how to engineer reliable first wall/blanket structures and make first efforts on reliability growth.

Materials Characterization (> 100 dpa). If this issue is put as obtaining potential lifetime irradiations of materials (>150 dpa), then only IFMIF can produce the required fluence, albeit only into a 0.5 liter volume of test articles. However, with fluences of 3–6 MW-yr/m^2 (dpa of about 30–60), FDF can make a significant contribution on relatively large, fully integrated and engineered components. IFMIF can irradiate a 0.5 liter volume of samples; FDF could take one port and fill one cubic meter with samples including welds and small assemblies and leave them there for ten years to accumulate a fluence of 3–6 MW-yr/m^2.

High Temperature Blankets (Electricity, Hydrogen Production). FDF will have reactor relevant neutron fluxes and fluences to develop such blankets in port test modules. FDF should take one of the best performing electric and hydrogen producing blankets and actually make a small demonstration of electricity and hydrogen production.

The recent FESAC Planning Panel identified 15 gaps between ITER, with current international program elements, and a DEMO. Figure 3 shows that FDF addresses nearly all those gaps except the two specifically aimed at the stellarators and superconducting coil machines.

Before further elaborating on the mission elements, it is useful to introduce the device concept in order to make the ensuing discussions more concrete.

How Initiatives Could Address Gaps

Legend

Major Contribution	3
Significant Contribution	2
Minor Contribution	1
No Important Contribution	

	G-1 Plasma Predictive capability	G-2 Integrated plasma demonstration	G-3 Nuclear-capable Diagnostics	G-4 Control near limits with minimal power	G-5 Avoidance of Large-scale Off-normal events in tokamaks	G-6 Developments for concepts free of off-normal plasma events	G-7 Reactor capable RF launching structures	G-8 High-Performance Magnets	G-9 Plasma Wall Interactions	G-10 Plasma Facing Components	G-11 Fuel cycle	G-12 Heat removal	G-13 Low activation materials	G-14 Safety	G-15 Maintainability
I-1. Predictive plasma modeling and validation initiative	3	2		2	2	3	1		2						
I-2. ITER – AT extensions	3	3	3	3	3		2		2	2	1	1		1	1
I-3. Integrated advanced physics demonstration (DT)	3	3	3	3	3	1	3	2	3	3	1	1	1	1	1
I-4. Integrated PWI/PFC experiment (DD)	2	1		1	2		2	1	3	3	1	1		1	1
I-5. Disruption-free experiments	2	1		2	1	3		1	1	1					
I-6. Engineering and materials science modeling and experimental validation initiative							1	3	1	3	2	3	3	2	1
I-7. Materials qualification facility							1			3	2	1	3	3	
I-8. Component development and testing			1				2	1		3	3	3	2	2	2
I-9. Component qualification facility	1	1	2	1	2		3	2	2	3	3	3	3	3	3
FDF	2	3	3	3	3		3		3	3	3	3	3	3	3

Fig. 3. Gaps chart from FESAC Planning Panel with our addition of the last line showing FDF addresses nearly all the gaps.

3.0 THE FDF CONCEPT

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 and Alcator C-mod with the construction features of those two machines. It is only about 50% larger than DIII-D (major radius 2.5 m) and about 40% the size of ITER. It is topologically equivalent to DIII-D. The outermost element is a massive copper toroidal coil, steady-state water cooled and capable of 6 T. 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 changeout of all blanket and PF coil elements.

The PF coils will be inside the TF coil; their proximity to the plasma allows higher elongation and triangularity for higher performance and smaller size. 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 copper TF coil means the shielding can be minimal, only enough to protect the insulators in the coils. Neutronics calculations indicate a 50 cm shield is adequate but an inboard tritium producing blanket may have to be about 15 cm thicker.

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 beams, ECH, and lower hybrid.

3.1. FDF OPERATING MODES

FDF has a range of operating modes, not a single column of numbers. Table I shows some of the operating modes with columns for ITER and ARIES-AT for comparison. The column headed Wall Load 2 MW/m^2 is the baseline case from the study which selected the aspect ratio to be 3.5. The machine size is between DIII-D and JET. Energy gain is a modest 4.2. Normalized beta is 3.7, which is equivalent to 3.3 at the DIII-D aspect ratio and elongation. Bootstrap fraction is 60%, requiring 59 MW to drive the remaining 40% of the current. Toroidal field is 6 T and the plasma current is 6.7 MA. Density is high but still just 57% of the Greenwald limit to increase current drive efficiency. The confinement factor H98y2 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.

The column headed 1.0 MW/m^2 shows that reducing the toroidal field and the bootstrap current results in a reduced performance case that still delivers 1.0 MW/m^2 into the test blankets. The nuclear technology mission is still secure in this backdown case. In this backdown case, 362 MW is needed to run the facility.

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	5.9	4.4	2.7	6.7	7.3	10.0	9.3

The columns labeled Very Advanced look at turning up 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.

4.0 MISSION ELEMENTS IN MORE DETAIL

4.1 TRITIUM PRODUCTION

One has to have an approach to solving the fuel availability problem for fusion nuclear technology development. The approach FDF takes is that the first full blanket can be designed with only one primary constraint, net tritium production. It only has to keep the FDF in tritium supply. FDF may require about one kg tritium supply to get through its first years of initial pulsed DT operation.

The blanket community has to develop the blanket plan. They will be a principal developer and user of the FDF facility. They have identified two most attractive blanket concepts: helium-cooled solid breeder and dual coolant Pb-Li. These may be the first two full blankets deployed in FDF with a potential third type coming from the port blanket module testing program. Some blanket experts are interested in using significant fractions (toroidal sectors) of the outer wall to test large areas of blankets. It appears possible to engineer the FDF to enable such large area testing. A few of the upper TF joints could be dismounted and the wedge sectors of the TF coil removed enabling crane lift extraction of an entire outer blanket toroidal sector and its replacement without full disassembly. Temperature above 400°C may be necessary for efficient tritium extraction and/or because of the coolants or breeder materials chosen and temperatures up to 500°C may be desirable on the front face to limit tritium retention in the plasma facing materials. Such considerations may motivate a relatively advanced first full blanket.

4.2 PORT BLANKET TESTS FOR ELECTRICITY PRODUCTION

Meanwhile, while the full main blanket is keeping the FDF in tritium supply, in 2–3 port sites blankets with DEMO relevant materials and improved designs for higher temperatures will be developed toward the kind of blankets needed on DEMO. These research sites are needed to address the high temperatures needed (> 600°C – 700°C) for efficient power conversion; the complex neutronics issues in detail; the chemistry effects with hot, corrosive fluids and materials being transmuted; and all of these effects with TBR > 1 in the test blanket and in real geometry with realistic gradients of temperature and neutrons. In the tokamak environment, the blankets must survive disruptions and provide a plasma friendly front surface.

The magnitude of the test program required may be further gauged by the fact that for blankets there are potentially 3 solid breeder materials, 2 liquid breeder materials, 2 coolants imagined, and 2 different advanced low activation structural materials, although the blanket community mainly focuses on just a few of these options.

The blanket community has identified the desirable blanket development capabilities.

- 1–2 MW/m^2 neutron fluxes. FDF provides that.
- 10 m^2 test area. The ports in FDF provide about 1 m^2 each, but the full blanket could be up to 130 m^2 and full poloidal sectors could range up to 1/8 of the toroidal circumference or 10–15 m^2.
- Continuous on-time of two weeks. This requirement is set because the time constant to achieve steady-state diffusion of tritium into the carrier fluid stream is days. FDF provides two week runs.
- Integrated fluence 6 MW-yr/m^2. FDF will provide 3–6 MW-yr/m^2.

FDF can test two blankets every two years in two port sites for a total of 10 blanket concepts or variants of concepts tested in 10 years. Such a research program can prepare blankets for DEMO in which we can have confidence.

4.3 PORT BLANKET TESTS FOR HYDROGEN PRODUCTION

One port site should be devoted to the even more difficult problem of hydrogen production. Hydrogen production from electrolysis needs over 800°C. The more efficient Sulfur-Iodine cycle uses heat directly but requires blankets that can produce over 900°C outlet coolant streams, placing significant demands on materials and design configurations.

5.0 CONCLUSION

To enable a DEMO to be built after ITER, a Fusion Development Facility is needed.

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.

FDF is needed to qualify Advanced Tokamak physics for DEMO and to enable development of fusion's energy applications, in particular closing the tritium fuel cycle.

These three research facilities, supported by current experiments, are the necessary combination of resources to proceed to a DEMO after ITER.

ACKNOWLEDGMENT

This work supported by General Atomics internal funding.