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CLOSING THE FUSION FUEL CYCLE

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GENERAL ATOMICS

Closing the Fusion Fuel Cycle

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Within the general theme of

" **Harnessing fusion power:** *The state of knowledge must be sufficient to design and build, with high confidence, robust and reliable systems that can convert fusion products to useful forms of energy in a reactor environment, including a selfsufficient supply of tritium fuel."*

the FESAC Planning Panel Report recommends the need to
"Learn and test how to manage the flow of tritium throughout the entire plant, including breeding and recovery."

Closing the Fusion Fuel Cycle

For the development of fusion power, we must learn how to close the fusion fuel cycle from the fusion process. It is clear also that before a DEMO project can be committed, net tritium production must be demonstrated and assured. Therefore, a key goal for FDF is to produce its own tritium and build a supply for the start up of DEMO. Corresponding management of the flow, recovery and accountability of the tritium will need to be assured.

Main-Blanket Research Plan

To achieve the goal of closing the tritium fuel cycle, the initial FDF main-blanket modules that cover most of the chamber surface area, will be designed with only one primary constraint, net tritium production. It only has to keep the FDF in tritium supply. FDF may require about 1 kg tritium supply to get through its first years of initial pulsed DT operation. Efficient electricity production is not a requirement from the main blankets. FDF will be designed to facilitate change out of the first wall/blanket structures and will do so at least twice in the life of the project, thus allowing at least the investigation of three main-blanket options. 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.

Blanket Types

After reviewing different solid breeder and liquid breeder tritium breeding blanket options, and with inputs from the US fusion nuclear science and technology community, we decided to follow the recommendation from the US ITER TBM program [1]. Presently, the two tritium breeding blanket concepts preferred by the US, namely, the helium cooled ceramic breeder (HCCB) and dual coolant lithium lead (DCLL) concepts are recommended to be used. Both concepts use RAF/M structure and helium cooling.

Research Schedule

A possible schedule of the FDF nuclear science program is given in Fig. 1-1. An initial 4 year commissioning 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. A helium cooled solid breeder blanket will be installed from the start and the 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 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 TBMs will have been tested.

	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→ H	D	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

Fig. 1-1. Operational and blanket development schedule of FDF.

Then there will be a 2 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.

Then there will be a 2 year shutdown to change 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 may reach 7.6 MW-yrs/m^2.

Tritium Production

Neutronics calculations were performed on the DCLL blanket concept. The DCLL concept was selected for the convenience of available blanket design details. The option of HCCB design will be performed in the future. Eight different inboard shield/blanket options were evaluated, these included stainless steel and ferritic steel shield options and the choice of WC or Be behind the inboard first wall. Results are that the OH coil can be made with organic insulators if the inboard 50 cm region is an optimized non-breeding shield; however in that case, although a TBR for the whole machine can closely approach 1.0, it probably will not be possible to show TBR>1. If the inboard 50 cm area is a breeding blanket, then 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. Our current baseline position is to use an inboard breeding blanket in order to have net tritium breeding and investigate whether the OH coil can be made with ceramic insulators.

Tritium Loop

With the explicit goal of FDF to produce its own tritium and to build a supply for the start up of DEMO, it is imperative to have a high performance tritium breeding blanket and a highly efficient system for the extraction of the bred tritium. FDF will follow very closely the development of tritium handling development of the ITER TBM program [1]. The key element for the tritium extraction system for the DCLL blanket is the Pd/Ag permeator and will have a vacuum on the permeate side for the extraction of tritium from the PbLi stream. In this manner tritium in the PbLi will move to the permeate side, and this stream of tritium (and any other hydrogen isotopes) will be sent to the Tritium Plant. It is expected that due to permeation, tritium will also migrate into the blanket helium. Another vacuum permeator to recover this tritium in the helium loop will also be needed.

Tritium extraction from the solid breeder blanket

The approach for the extraction of tritium from solid breeder of the HCCB concept [2] is different from the DCLL concept. A purge flow of helium at ~0.1 MPa is passed through the solid breeder and the bred tritium is collected in the purged helium gas. The collected tritium will then be extracted from the helium purge flow external of the blanket system. This purge flow extraction approach for solid breeder blankets has been well established for solid breeder blanket designs.

Plant systems

To support the operation of the main-blanket system, three main interface systems between the blanket and the FDF will be needed. They are the hot cell system, the blanket heat removal system (BHRS) and tritium extraction system (TES).

Hot Cell Interface Area

Activities for the support of the main-blanket program require the following:

- Remote handling (RH) equipment
- Hot cell area for main-blanket components exchange and ancillary equipment maintenance
- Utilities for main-blanket and ancillary equipment tests

Blanket Heat Removal System (BHRS)

The second interface area is the BHRS system. Heat generated from the main-blanket will be removed by heat exchangers. High pressure lines and helium purge gas lines connecting the main-blanket to the BHRS and the tritium plant area will be needed. For various coolant line connections, requirements will be specified. The BHRS will also house the helium coolant purification system (CPS).

Tritium Extraction System (TES)

The third interface area is the tritium extraction system and the connections between the main-blanket and the tritium plant. These connections will be modified when we move from the solid breeder blanket design to the PbLi breeder blanket design. Key elements of the Coolant Purification System (CPS), Tritium Extraction System (TES) and the accountancy of the bred tritium will be needed for the two blanket options.

For the blanket bred tritium, there are two sources of tritium that will need to be extracted and examined.

1. From the breeder and its corresponding purge stream and/or extraction stream.
2. From the tritium that permeated through the wall and contaminated the main coolant helium stream.

Furthermore, for the operation of the main-blanket, and especially during maintenance, tritium containing lines will have to be handled safely. Therefore detailed operational procedures for the main-blanket tritium system will have to be developed, and the use of environmental control, local house vacuum line and/or other vending detritiation systems will have to be considered. It should be noted that The TES will also be responsible for the tritium fuel cycle of the plasma burn cycle, which includes the extraction and preparation of the DT fuel for the plasma and the handling and recycling of the plasma exhaust gases.

Heat Extraction

The heat generated from the main blankets will be removed at the BHRS by He/H_2O heat exchanges. In the following we will again use the DCLL blanket concept as an example. There are two heat sources from the DCLL blanket design. The first one is from the helium coolant that cools the first wall and blanket structure. Dedicated helium piping will be designed between the main-blanket and the helium/water heat exchanger at the BHRS area. The second one is the liquid breeder loop, which will be connected to a helium intermediate heat removal circuit located between the FDF tokamak hall and the BHRS area. Corresponding helium pipes will connect the PbLi/He exchanger to the BHRS heat removal system.

Reference:

[1] US TBM team, "DDD for the US Dual Coolant Pb-17Li Test Blanket Module," GA-C25027 Rev. 3, November 2005

[2] US TBM team, "DDD for the US Helium-cooled Solid Breeder Test Blanket Module," UCLA report FNT213, November 2005