Proposed Liquid Blanket for CFETR

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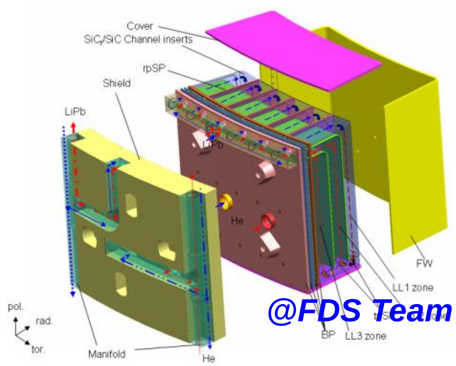
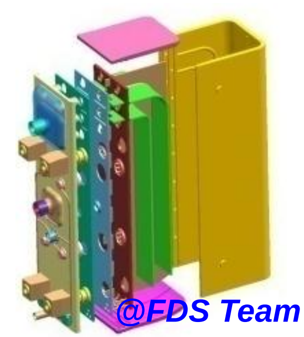
Contents

Introduction

CFETR Blanket Concept

- **CFETR blanket module is designated to check and validate relevant fusion DEMO technologies.**
  - **Validation of tritium and energy generation technology**
  - **Validation of relevant analysis tools, codes and database**
  - **Integrative testing for blanket system in different operations**
  - **Material testing under neutron irradiation**



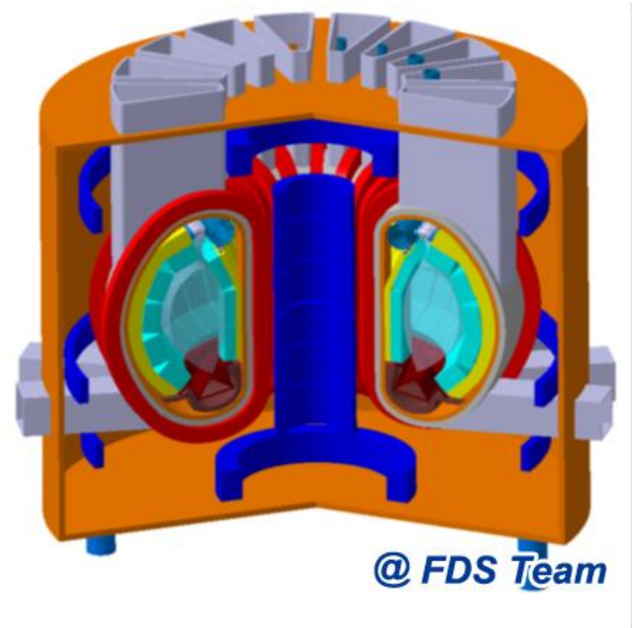


DEMO Blanket → Test Blanket Module

CFETR: Options for Liquid Blanket

Multi-Types-of-Blankets

Multi-Testing-Phases



- **Option I :** **Liquid PbLi-based blanket for tritium breeding and energy production**
  --- **SLL/DLL/DFLL**
- **Option II :** **Uranium-loaded hybrid blanket for energy production**
- **Option III :** **Spent fuel-loaded hybrid blanket for energy production and waste transmutation**

SLL/DLL

Demo Blanket Modules

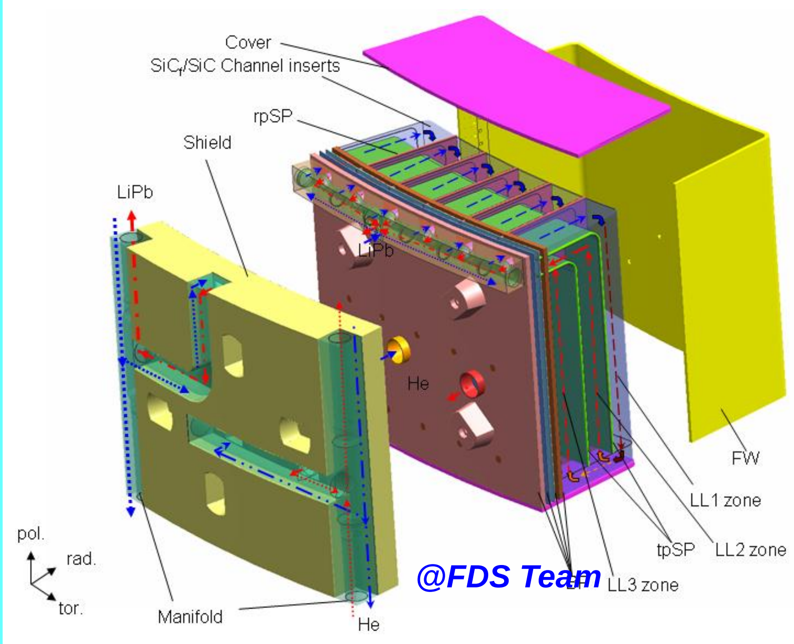
SLL/DLL Demo Blanket Module

SLL: He-cooled Quasi-Static Lead Lithium Blanket

- **Single Coolant:** He-gas (R-T + P-directions)
- **T-Breeder:** Quasi-Static PbLi: (slowly flowing in P-direction, outlet temp.~450 ºC)
- **Coating:** to protect the steel structure and to reduce T-permeation and MHD effects.

DLL: He/PbLi Dual-cooled Lead Lithium Blanket

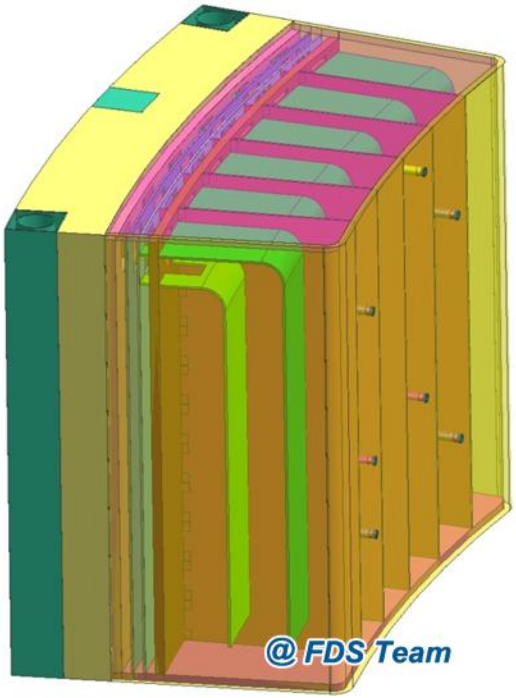
- **Coolant 1:** He-gas (R-T + P-directions)
- **Coolant 2 & T-Breeder:** PbLi (quickly flowing in P-direction, outlet temp.~700 ºC)
- **Thermal and electric insulators:** to avoid RAFM working at high temp. 700 ºC

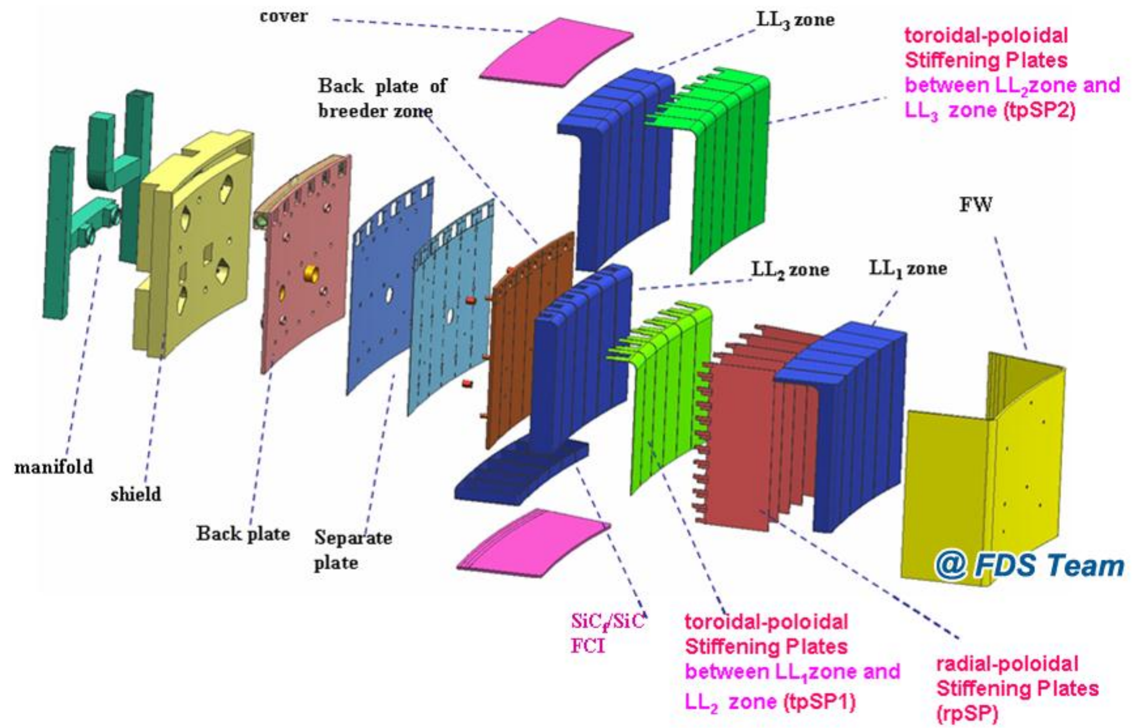


- **The basic blanket structure using RAFM steel e.g. the CLAM steel.**
- **DLL blanket as the main candidate blanket scheme: using Flow channel insert (FCI).**
- **SLL blanket without FCI as backup scheme: relatively mature material technology, use quasi-static PbLi flow instead of fast moving PbLi in DLL.**

[1] Conceptual design of the China fusion power plant FDS-II. *Y. Wu, FDS Team*. **Fusion engineering and design**. 2008 (83), 1683–1689

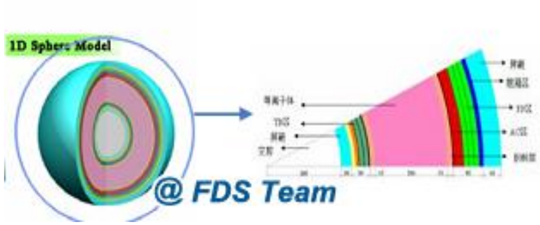
DLL Blanket Structure



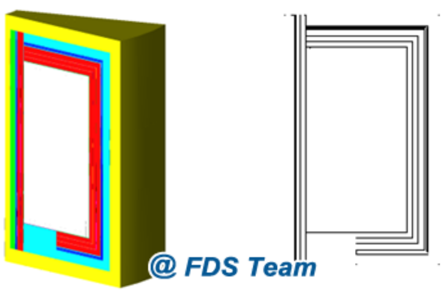


- Features a big rectangular steel box enclosed by **U-shape FW, covers, and BPs. SPs strengthen the structure**
- **3 (rad.) × 6(tor.) rectangular PbLi channel** inside breeder zone
- **Dimension :** Outboard: ~ 2 m (Pol.)×2m (Tor.) ×**1.2m (Rad.)**
  Inboard: ~ 2 m (Pol.)×2m (Tor.) ×**0.8m (Rad.)**

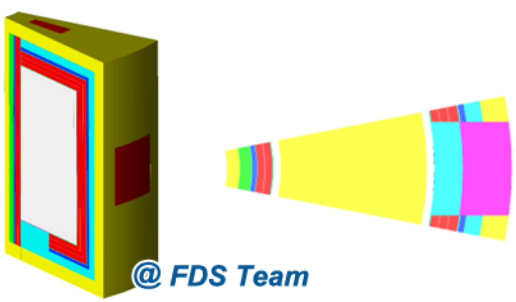
Neutronics Analysis



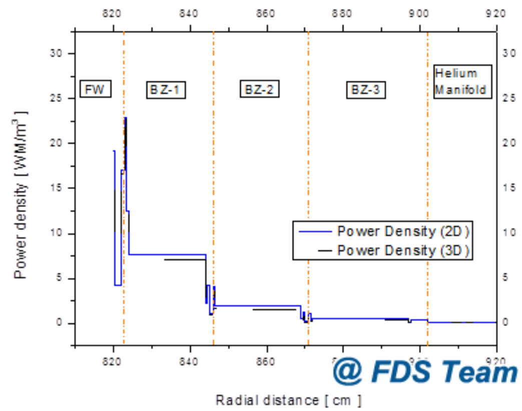
1D model



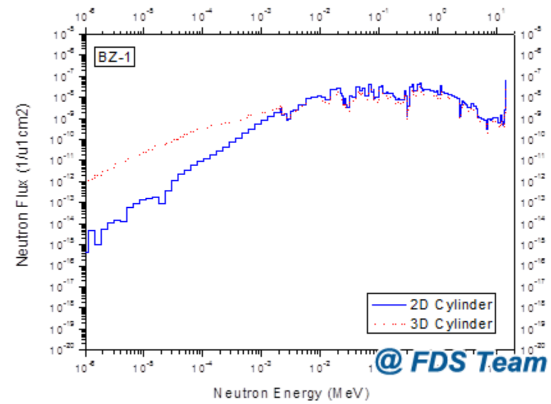
2D model



3D model



Nuclear power density distributions

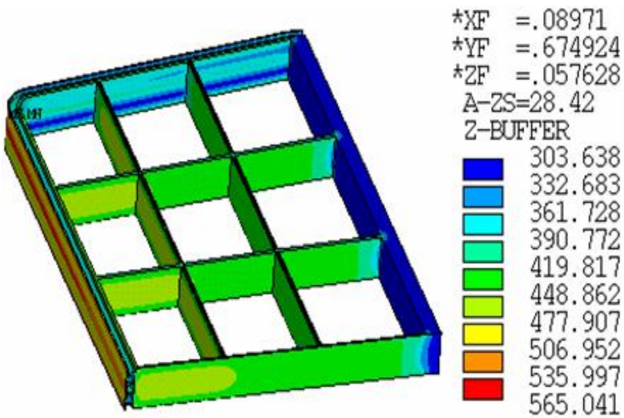


The neutron flux distribution in tritium breeding zone-1

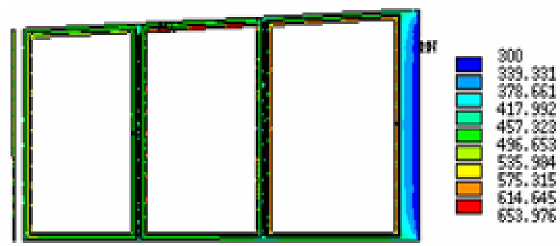
The TBR of the DLL blanket

	TBR		
	1D	2D	3D
Tritium Breeding Zone (IB/OB)	0.33/1.00	0.23/1.05	0.21/0.75
Shielding (IB/OB)	0.04/0.02	0.03/0.02	0.03/0.02
Port (breeding zone)	-	-	0.28
Total	1.38	1.34	1.29-x

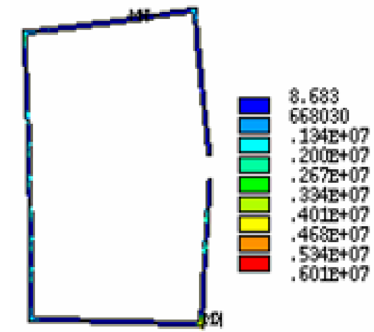
Thermal Stress Analysis



Temperature distribution on DLL



Temperature distribution on FCI



Stress distribution on FCI

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When 2 mm ODS thin layer is fabricated on plasma facing FW surface :

- **The maximum temperature on first wall amounts to 557 ºC, well below the limit of 650 ºC for ODS.**
- **The maximum Von Mises stress is 333MPa, less than allow. 3Sm of 476MPa at 370 ºC.**
- **The maximum temperature and stress on FCI amounts to 653 ºC and 6 MPa, less than allowable limit of 1000 ºC and 190 MPa for SiC material.**

MHD Analysis

Empirical relations:

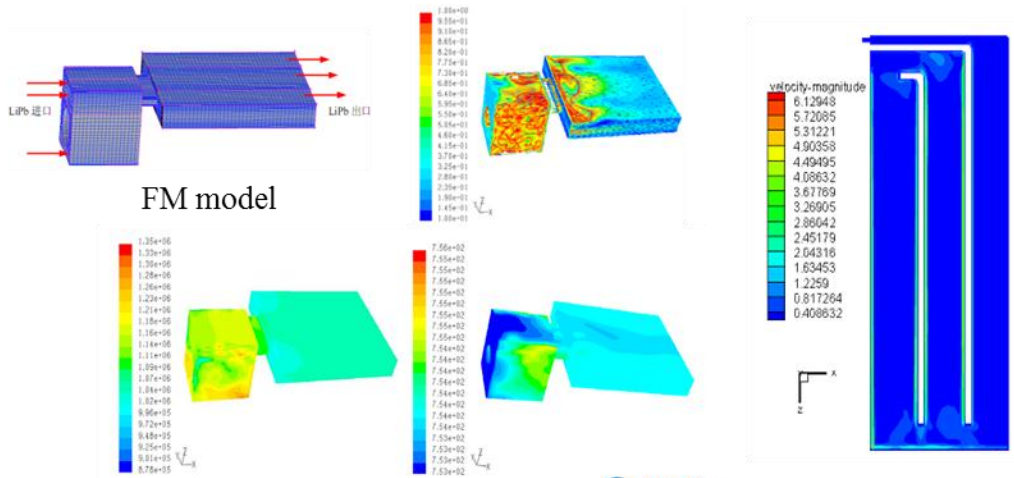
1. Flow in the long poloidal channel

$$\frac{dp}{dx} = \sigma V B^2 \frac{1}{1+\dfrac{\rho_i t_i M}{\rho_i t_i + 2bM\rho}}$$

2. Considering 3D geometry effect, which happens at following zones:

- From inlet to LL1 channel;
- From LL1 channel tune to LL2 channel;
- From LL2 channel tune to LL3 channel;
- From LL3 channel to outlet.

$$\Delta p = \zeta \frac{1}{2} \rho v^2 \text{ , with } \zeta = f(N,M)$$



FM model

Pressure distributon

Temperature distributon

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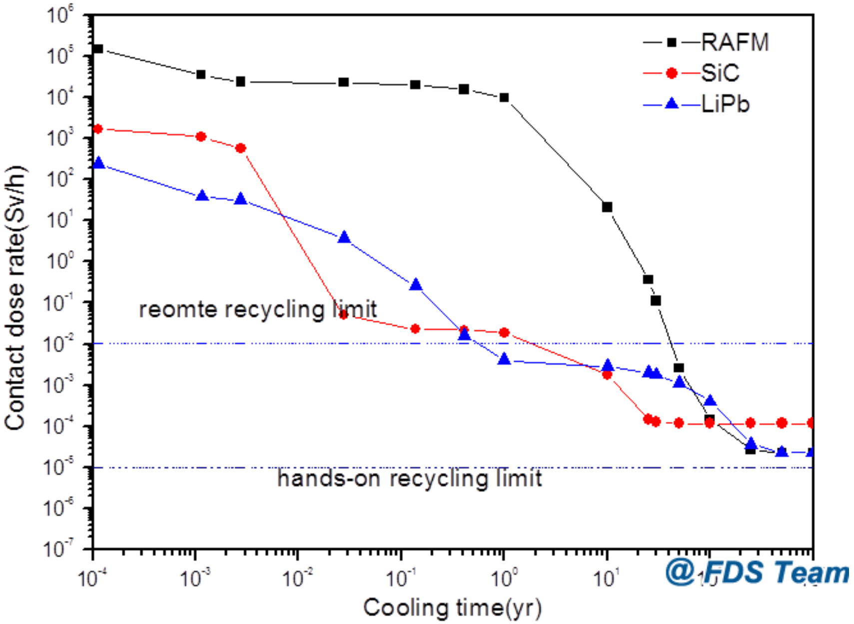
Velocity distribution at breeding zone

3D MHD effect simulation

MHD pressure drop has been analyzed with the empirical relations and simulation:

- ➢ **MHD pressure losses in the poloidal channels are much smaller than the 3D MHD pressure drops, associated with the contraction/expansion.**
- ➢ **The total pressure drop is ~3MPa.**
- ➢ **The pumping power is about 0.136MW assuming pumping efficiency of 80%.**

Activation Calculation



Dose rates of RAFM steel , SiC and LiPb in the DLL blanket

Management options for activated materials of the DLL functional blanket

Management option	Cooling time 50a	Cooling time 100a	limit
Permanent disposal waste	20%	0	-
Complex recycle material([a]RHR)	31%	0	20mSv/h
Simple recycle material(RHR)	15%	66%	2mSv/h
Simple recycle material([b]HOR)	34%	34%	10 μSv/h
Non-activated Waste	0	0	[c]I_c<1
Mass inventory(t)	9678	9678	

- **The level of afterheat, dose rate, activity and biological hazard potential for the different regions of DLL breeder blanket are evaluated.**
- **Code and library: VisualBUS code and multi-group working library FENDL2.1/MG.**

DFLL

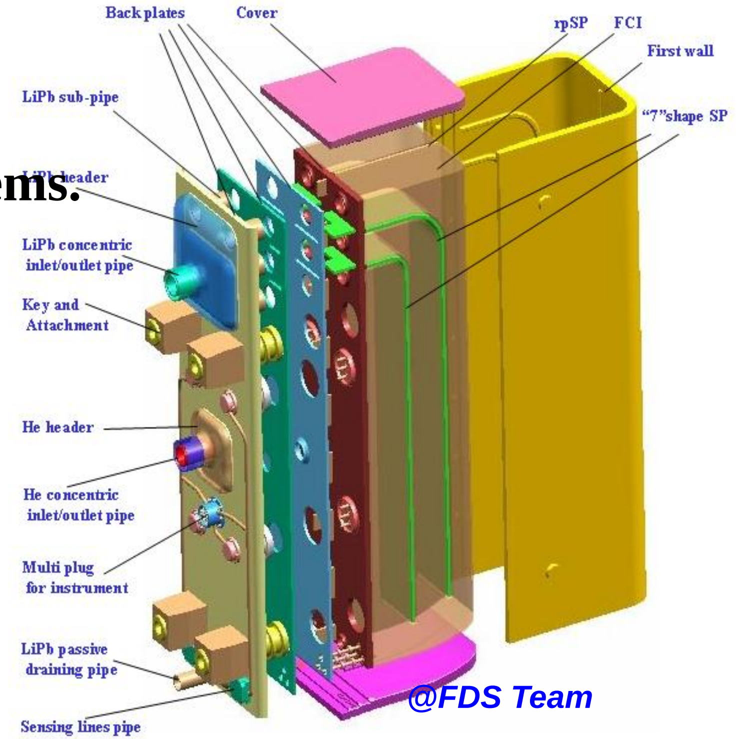
Testing Blanket Module

DFLL Testing Blanket Module

Objectives: **DFLL Blanket for test, to demonstrate the technologies of SLL/DLL lead lithium DEMO blankets.**

Flexible design :

- Similar structure and auxiliary systems.



SLL mode at the earlier stage

- **PbLi: 1mm/s, 450℃,**
- **Without FCI in PbLi channels**

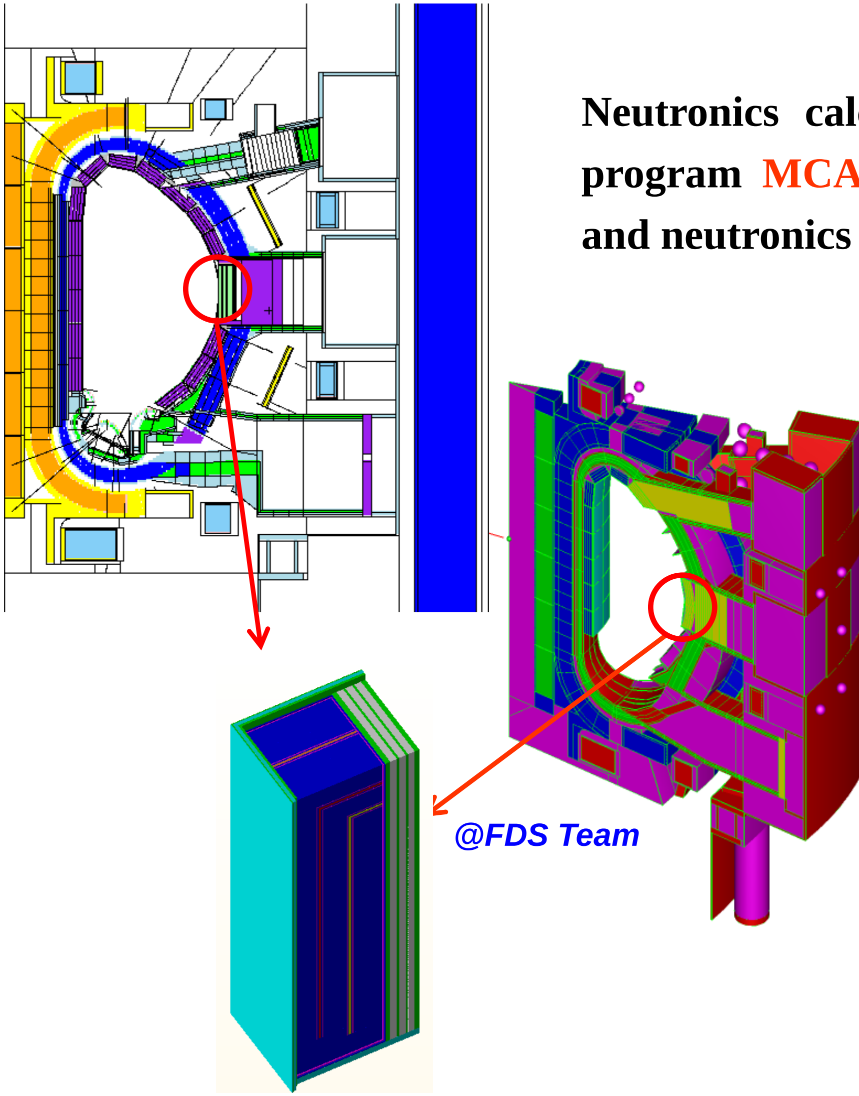
DLL mode at the later stage

- **PbLi: ~10mm/s, ~700 ℃,**
- **With FCI in PbLi channels**

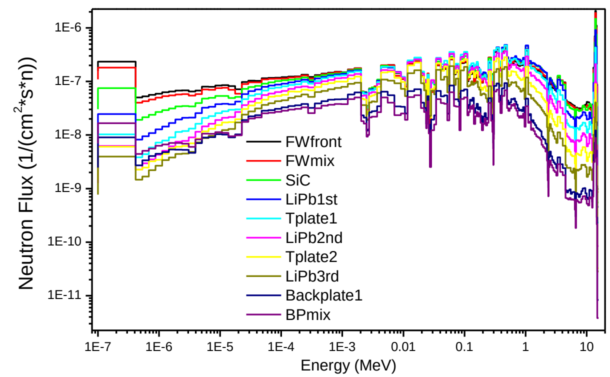
[2] *Y. Wu and the FDS Team.* **Nuclear Fusion**, 2007, 47(11): 1533-1539.
[3] *Y. Wu, the FDS Team.* **Fusion Engineering and Design**, 2007, 82: 1893-1903.

Neutronics Performances

Neutronics calculations were performed with the interface program MCAM converting models between CAD software and neutronics models



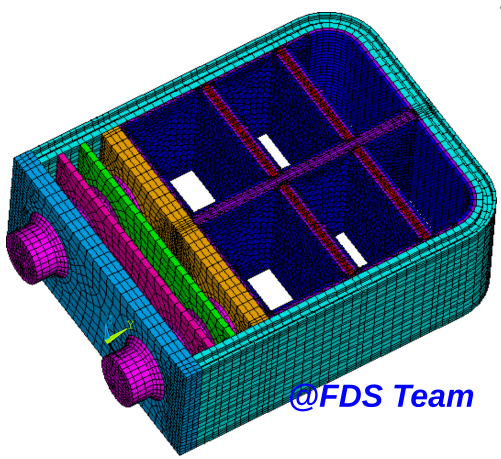
DFLL testing blanket module



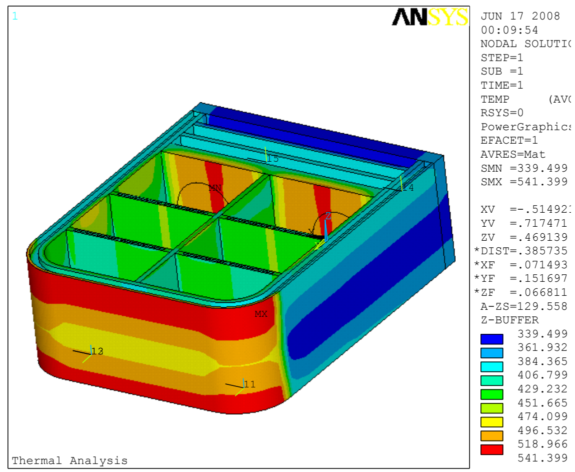
Main calculation results

Li enrichment	TBR	T production (mg/FPD)	Total nuclear heat (MW)
Natural Li	4.68E-03	19.73	0.42
90% ^{6}Li	1.34E-02	56.42	0.47

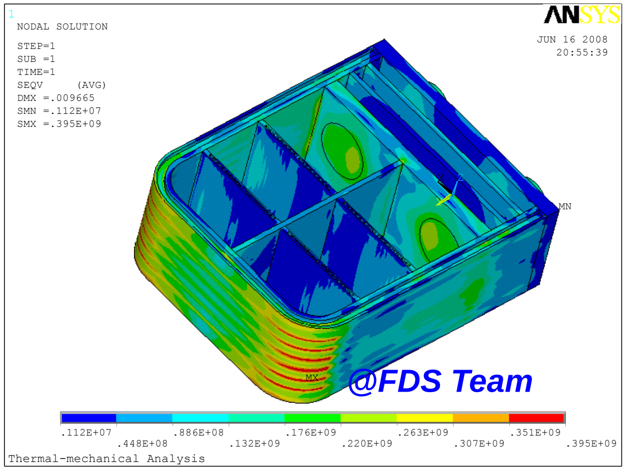
Thermal-Mechanical Performances



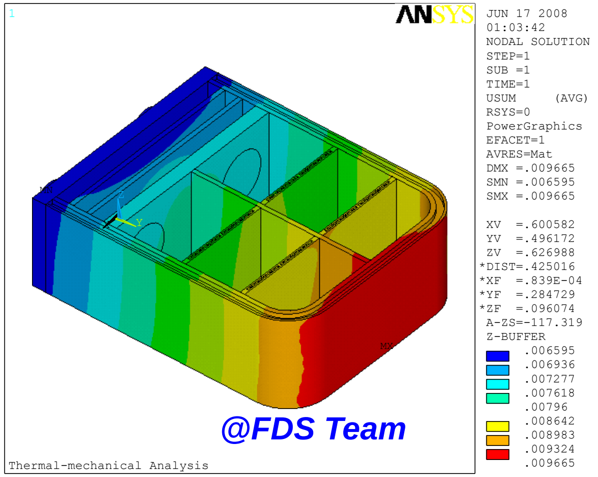
3-D model of DFLL blanket



Temperature distribution



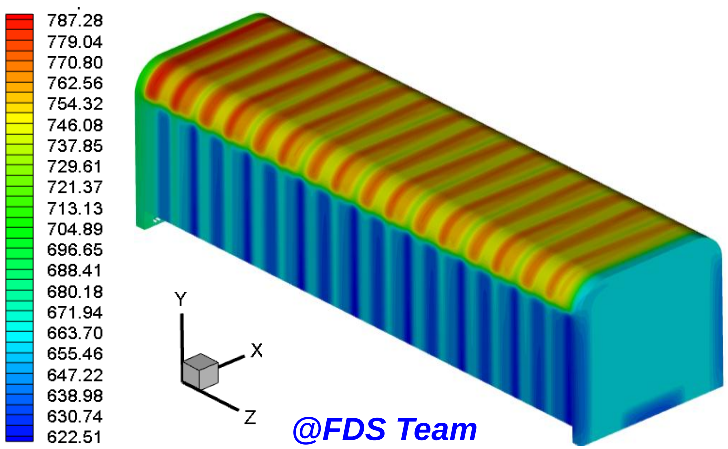
Von Mises stress



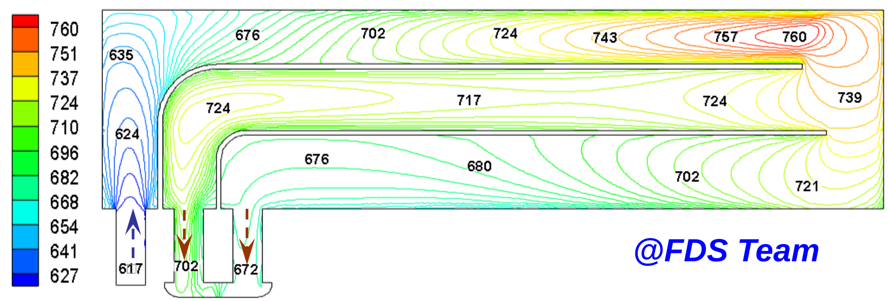
Displacement in radial direction

- **The max. temperature of 541℃ at the plasma facing side of the FW under the specification limitation of FAFM.**
- **The max. Von Mises stress of 395MPa (<allow 3Sm of 417MPa for RAFM at 500℃).**

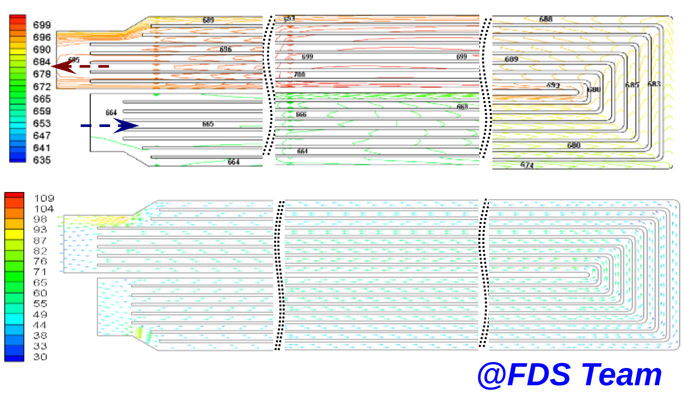
3D Dual-flow Fields Analysis



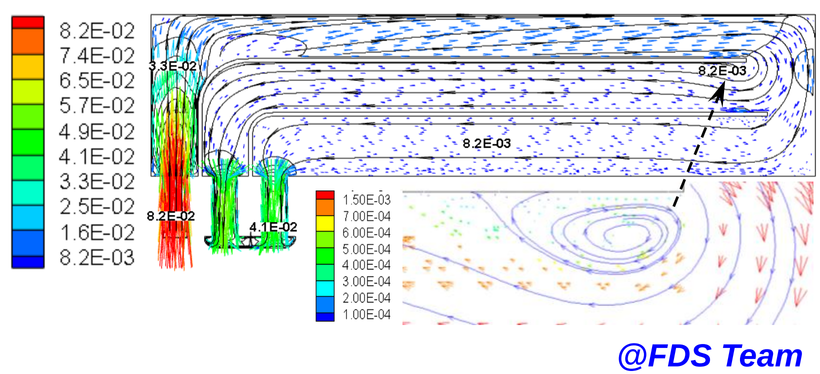
Blanket structure temperature contour



PbLi temperature contour



Temp. and velocity distribution of the helium gas in tpSP

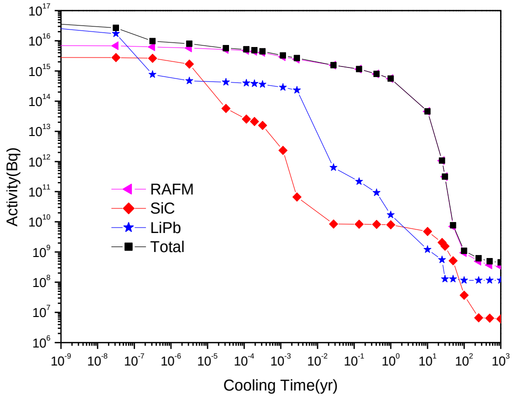


Velocity distribution of the PbLi(m/s)

The hydraulic parameter fields can satisfy the design requirements

Activity and Decay Heat Inventory

- **Activity Inventory**



Total activity generated in the TBM and contribution from each material

Total activity (Bq):
(at several shutdown time point)

- 0 sec: 4.9E16
- 10 sec: 9.6e15
- 1000 sec: 5.6E15
- 1 h: 5.2E15
- 1 day: 2.7E15
- 10 day: 1.6E15

Dominant Paths :
(at the shutdown time):

■ RAFM

1. Fe56(n,p)Mn56
2. W186(n,g)W187
3. Fe56(n,2n)Fe55

■ LiPb

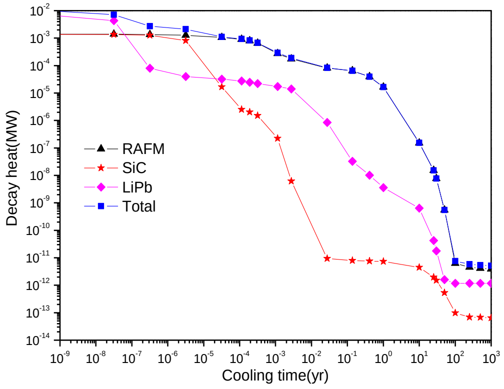
1.Pb208(n,2n)Pb207m

2.Pb204(n,2n)Pb203m

■ SiC

1. Si28(n,p)Al28
2. Si29(n,p)Al29

- **Decay Heat Inventory**



Total afterheat generated in the TBM and contribution from each material

Total decay heat (MW):
(at several shutdown time point)

- 0 sec: 0.013
- 10 sec: 2.9e-3
- 1000 sec: 1.1E-3
- 1 h: 9.4E-4
- 1 day: 1.9E-4
- 10 day: 8.2E-5

Dominant Paths
(at the cooling of 1h)

■ **RAFM**

1. Fe56(n,p)Mn56
2. W186(n,g)W187
3. Ta181(n,g)Ta182

■ **LiPb**

1. Pb204(n,2n)Pb203
2. Pb204(n,n')Pb204m

■ **SiC**

1. Si30(n,g)Si31
2. Si29(n,p)Al29

Accident Analysis: Conditions

Reference events	Assessment objective	Enveloped PIEs	Model and conditions
In-vessel blanket coolant leaks	Small pressurisation of the first confinement (VV) Passive removal of decay heat Limited chemical reactions and hydrogen formation	In-vessel loss of blanket coolant-He In-vessel loss of blanket coolant-PbLi	**Model:** 2-D radial-poloidal model as Fig.1 **Code:** ANSYS **Source:** heat flux on FW, nuclear heat and decay heat **Conditions:** Decay heat removed only by radiation to frame and the shielding blanket Frame maintaining temperature at 135°C Shielding blanket temperature as Fig.2 Initial blanket temperature. 550 °C.
In- blanket breeder box coolant leaks	Pressurization of the module and purge gas system. Limited chemical reactions and hydrogen formation Subsequent in-vessel leakage	Loss of Flow -PbLi because of pump seizure Ex-vessel loss of blanket coolant-PbLi In- blanket loss of coolant-He into PbLi	
Ex-vessel blanket ancillary coolant leaks	Pressurization of the port cell, vault, assembly cask Limited chemical reactions and hydrogen formation	Loss of Flow -He because of circulator seizure Ex-vessel loss of blanket coolant-He Ex-vessel loss of blanket coolant-He	

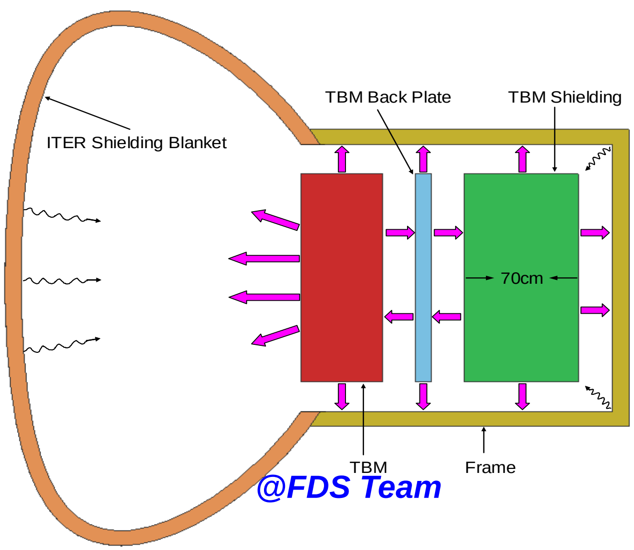


Fig.1 2-D ANSYS radial-poloidal model

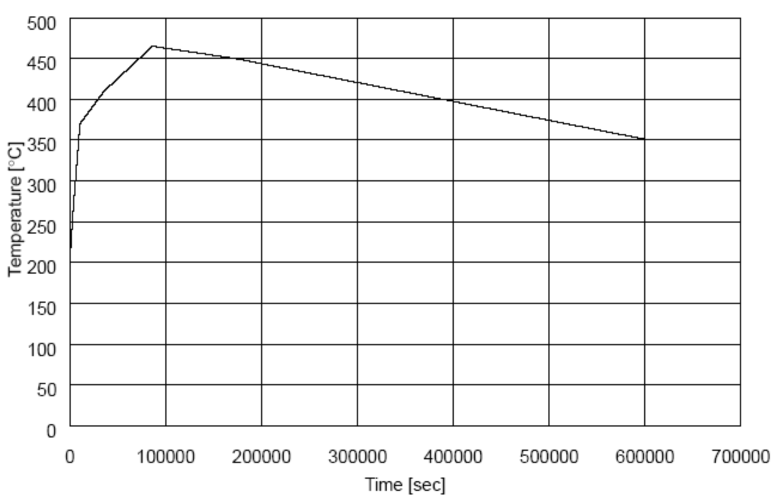


Fig.2 Shielding blanket temperature evolution

Accident Analysis: Results

Accident Category		VV Pressurization	Vault Pressurization	Temperature Evolution	Decay Heat Removal Capability	Hydrogen Generation	Tritium Release
In-vessel loss of coolant	Plasma disruption	22.4kPa	no	Max.temp.647^{o}C	Decrease to 224^{o}C in 10 days	<2.5 kg	<0.133mg
In-blanket loss of coolant	with detection	No	no	Max.temp.577^{o}C	Decrease to 224^{o}C in 10 days	no	<1.723mg
	with no detection	22.4kPa	no	Max.temp.1395^{o}C	Decrease to 224^{o}C in 10 days	<2.5 kg	
Ex-vessel loss of coolant	with detection	No	1158Pa	Max.temp.577^{o}C	Decrease to 224^{o}C in 10 days	no	<1.904mg
	with no detection	No	1158Pa	Max.temp.1395^{o}C	Decrease to 224^{o}C in 10 days	<2.5 kg	

- **The max. VV pressurization is ~22kPa, which is less than the limit of 200kPa.**
- **The max. vault pressure buildup is ~1.2kPa, which is much lower than the limit of 200kPa.**
- **The decay heat can be removed by radiation heat transfer.**
- **The blanket structure temperature is still less than the melting points of RAFM.**
- **The hydrogen production is less than limit of 2.5 kg.**
- **The maximum tritium release is about 1.9mg**

Testing Strategy

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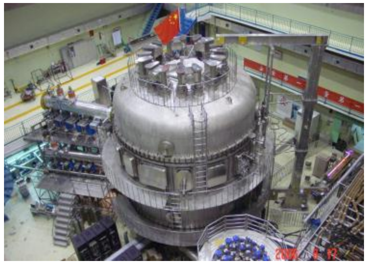
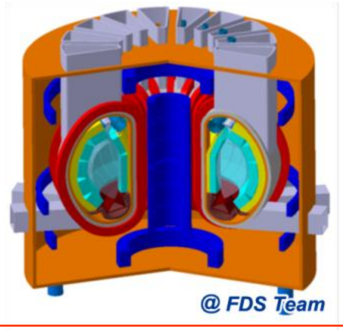
Testing Strategy of DFLL Blanket

Stage I: Out-of-pile Test (1/3 size)

Stage II: Test in EAST (1/2 size)

Stage III: Test in CFETR (full size)

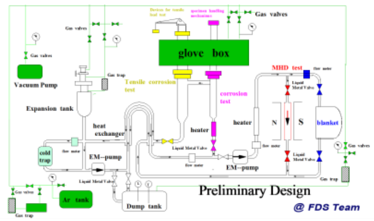


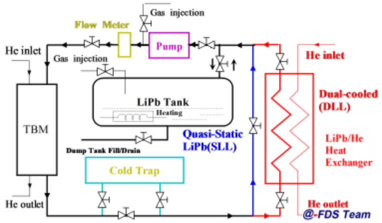


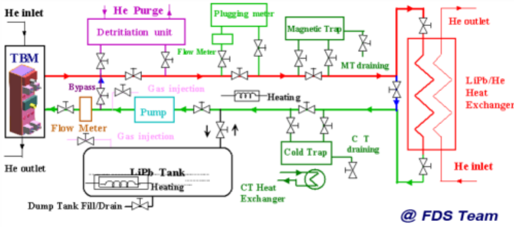
Thermal convection loop
Forced convection loop

PbLi/He system for test blanket in EAST

PbLi/He system for test blanket in CFETR







- R&D on materials (RAFMs, coatings and FCI) and fabrication technology
- Out-of-pile test of 1/3 mockup etc.
- Thermaldynamics and MHD
- Diagnostic and measurement

- EM and thermo-mechanics
- Partially neutronics performances
- Influence on plasma confinement
- Thermaldynamics and MHD
- Diagnostic and measurement

- To confirm results of EM, thermo-mechanics test in EAST
- To test neutronics, tritium production, integration performan--ces in CFETR

[2] *Y. Wu and the FDS Team.* **Nuclear Fusion**, 2007, 47(11): 1533-1539.

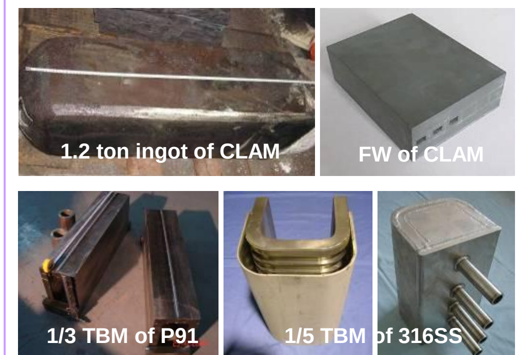
Stage I: Out-Of-Pile Small Mockup Test

Objectives:

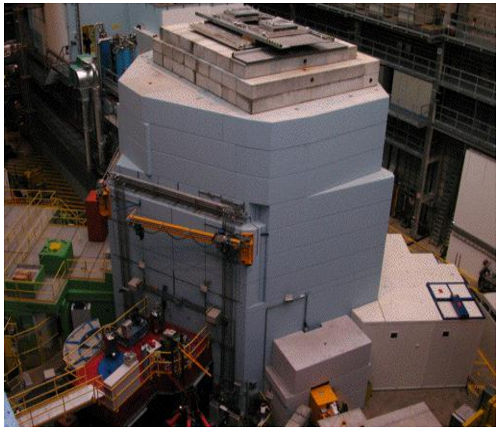
- Validation of the fabrication route and techniques
- Validation of performances
- Assessment of reliability and safety with regard

Test Items:

- Leak and pressure test.
- MHD and heat removal from FW.
- Mock-up connected to PbLi loop
- Hydrogen control and extraction to simulate tritium extraction
- Irradiation performance

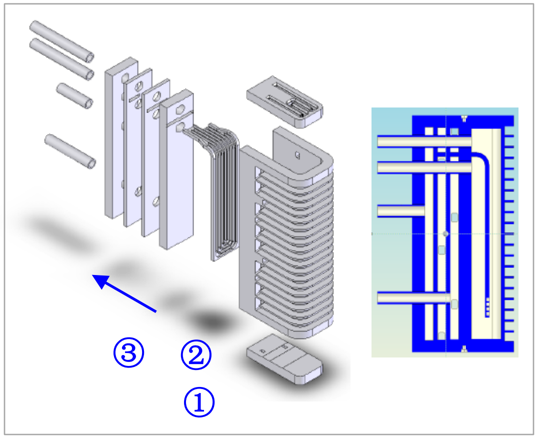


R&D on 1/3 size-reduced mockup



Spallation neutron source at PSI

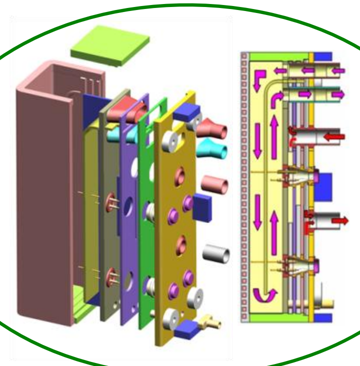
Out-of-pile integrated test loops



1/3 size-reduced mockup

Stage II: Test in EAST Tokamak

(1/2 Size-reduced Test Blanket)



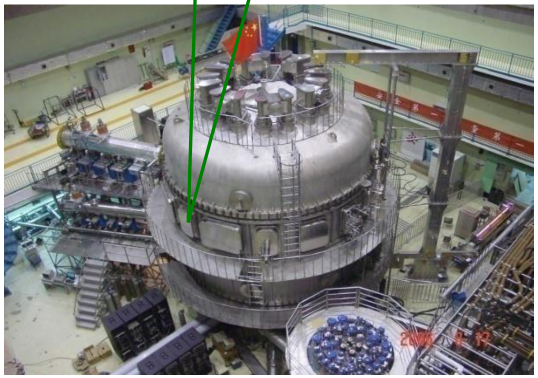
Objectives:

- Preliminary validation of design codes and data
- Checking of feasibility & availability of auxiliary system
- FM Influence on Plasma

Test Blanket Test in EAST:

- ElectroMagnetic performance (MHD pressure drop, influence on plasma)
- Thermo-mechanics/Thermofluid dynamics performances
- Partially neutronics performance (DD neutrons), Diagnostic instruments

Device	EAST	CFETR
Phase	DD	DT
R (m)	1.95	5.5
A (m)	0.46	1.6
Bt (T)	3.5-4.0	5.3
Neutron rate (n/s)	10^{15}~10^{17}	
Avg.HF(MW/m^2)	0.1~0.2	
Port Size	0.97m x 0.53m	
Pulse (sec)	~1000	



Stage III Test in CFETR

(Full size Test Blanket)

3 years			2 years		3 years			2 years		3 years		
			HH	HH	DD	DD	DD	DT	DT	DT	DT	DT

TBM installation

Qualification test

EM-TBM
(SLL-function):

EectroMagnetic Test:
- Confirm EM/Thermo-mechanics test
- Auxiliary system

NT-TBM
(SLL-Function)

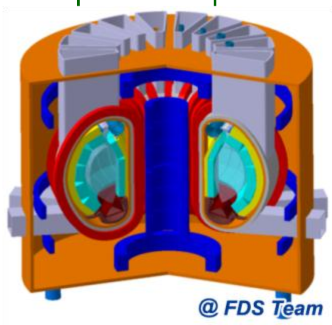
Neutronics Test:
- Neutronics performance
- N-fluence&gamma spectra
- Nuclear heat
- T-breeding
- Code &Data

TT-TBM
(DLL-Function)

ThermoDynamics&Tritium Behavior Test:
- Tritium recovery
- FCI performance
- Thermofluid MHD
- Code &Data

IN-TBM
(DLL-Function)

Integrated Test:
- On-line tritium extraction
- Synergic nuclear effects
- MHD integrated effects
- Reliability and Safety



Summary

Summary: Proposed Liquid Blanket Module for CFETR

■ Testing Blanket Module

◆ **DFLL-Test Blanket: He/PbLi Dual-Fuctional lead lithium Test Blanket Module**

- **To demonstrate the technologies of SLL/DLL lead lithium DEMO blanket Module.**

■ DEMO Blanket Modules

◆ **SLL: He-cooled Quasi-Static Lead Lithium Blanket**

- **Single Coolant:** He-gas (R-T + P-directions)
- **T-Breeder:** Quasi-Static PbLi: (outlet temp.~450 ^{o}C)

◆ **DLL: He/PbLi Dual-cooled Lead Lithium Blanket**

- **Coolant 1:** He-gas (R-T + P-directions)
- **Coolant 2 & T-Breeder:** PbLi (outlet temp.~700 ^{o}C)

The End

Thanks for your attention !

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