# APPENDIX

**to main presentation of Seminar at General Atomics**

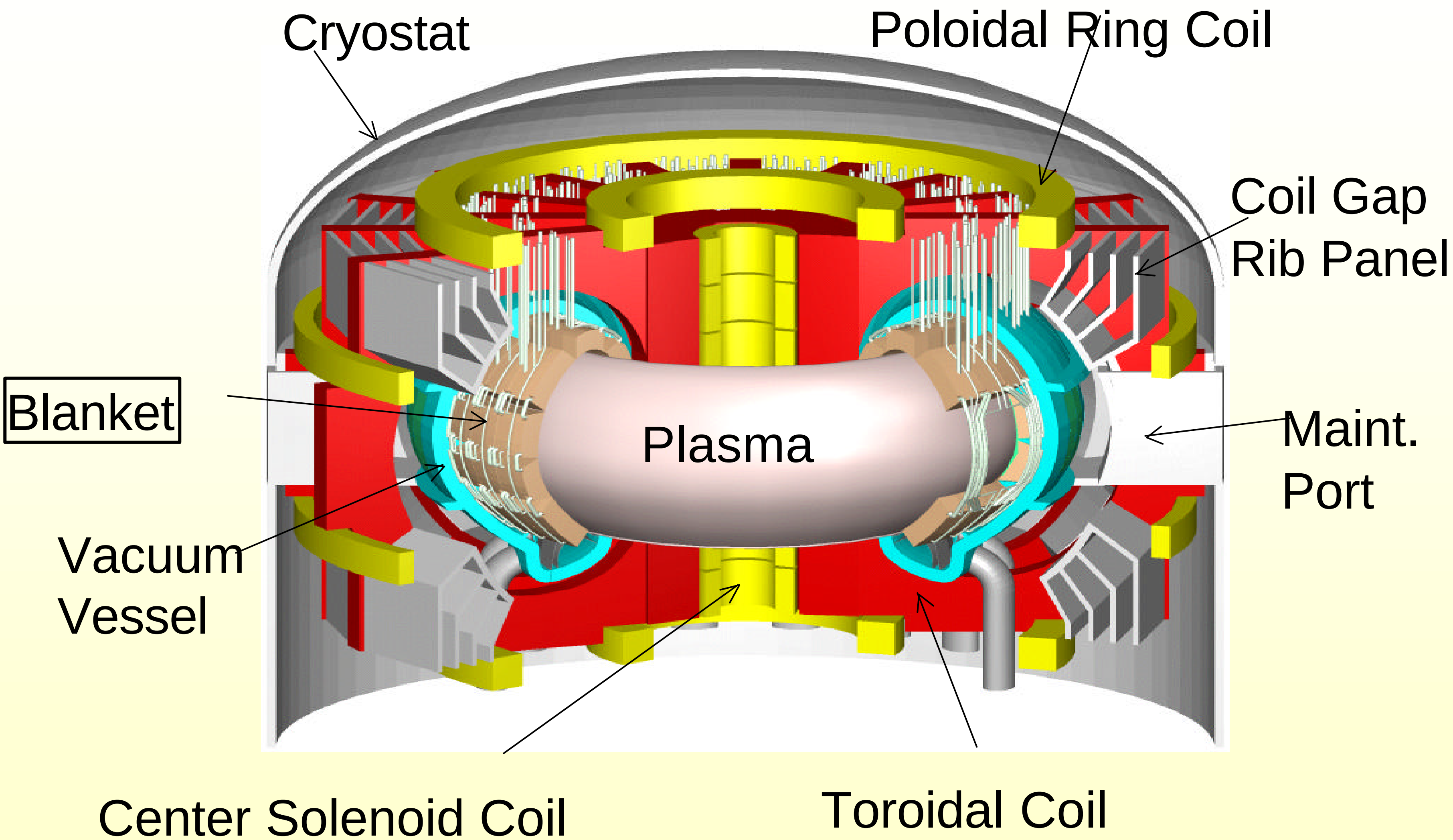
**December 9, 2002**

**Mohamed Abdou**

# *Demonstration*

**The US fusion demonstration power plant (Demo) is the last step before commercialization of fusion. It must open the way to commercialization of fusion power, if fusion is to have the desired impact on the world energy system. Demo is built and operated in order to assure the user community (*i.e.,* general public, power producers, and industry) that fusion is ready to enter the commercial arena. As such, Demo begins the transition from science and technology research facilities to a field-operated commercial system. Demo must provide energy producers with the confidence to invest in commercial fusion as their next generation power plant, *i.e.,* demonstrate that fusion is affordable, reliable, profitable, and meets public acceptance. Demo must also convince public and government agencies that fusion is secure, safe, has a low environmental impact, and does not deplete limited natural resources. In addition, Demo must operate reliably and safely on the power grid for long periods of times (*i.e.,* years) so that power producers and industry gain operational experience and public are convinced that fusion is a "good neighbor." To instill this level of confidence in both the investor and the public, Demo must achieve high standards in safety, low environmental impact, reliability, and economics.**

# JAERI DEMO Design




**FNT: Components from Edge of Plasma to TFC.**

Blanket / Divertor immediately circumscribe the plasma (often called Chamber Technology)

# Short Answers to Key Questions

That we have been asked the past few months

## 1. Can IFMIF do Blanket / FNT testing? ***No***

IFMIF provides data on "radiation damage" effects on basic properties of structural materials in "specimens".

Blanket Development is something **ELSE**

(IFMIF's role was explained by S. Zinkle. This presentation explains blanket/FNT development)

(No IFMIF report nor any of the material or blanket experts ever said this.)

## 2. What do we need for Blanket/PFC Development?

A – Testing in non-fusion facilities (laboratory experiments plus fission reactors plus accelerator based neutron sources)

AND B – Extensive Testing in Fusion Facilities

Conclusion from previous international studies (e.g. FINESSE, ITER Test Blanket Working Group, IEA-VNS):

*"The feasibility, operability, and reliability of blanket/FNT systems cannot be established without testing in fusion facilities."*

# Short Answers to Key Questions (Cont'd)

3. What are the Fusion Testing Requirements for Blankets/FNT?

Based on extensive technical international studies, many published in scholarly journals, the testing requirements are:

Neutron wall load of >1 $MW/m^2$ with prototypical surface heat flux, steady state (or long pulse > 1000 s with plasma duty cycle >80%), surface area for testing >10 $m^2$, testing volume > 5 $m^3$, neutron fluence > 6 $MW{\cdot}y/m^2$

4. Can the present ITER (FEAT) serve as the fusion facility for Blanket/FNT Testing? ***No***

- ITER (FEAT) parameters do not satisfy FNT testing requirements

  Short plasma burn (400 s), long dwell time (1200 s), low wall load (0.55 $MW/m^2$), low neutron fluence (0.1 $MW{\cdot}y/m^2$)

- ITER short burn/long dwell plasma cycle does not even enable temperature equilibrium in test modules, a fundamental requirement for many tests. Fluence is too low.

# Short Answers to Key Questions (Cont'd)

5. Is it prudent to impose FNT testing requirements on ITER? ***No***
   - Tritium consumption/tritium supply problem, complete redesign is costly, schedule is a problem.
   - The optimum approach is two fusion devices: one for plasma burn; the other for FNT testing. (Conclusion of many studies.)

6. What is CTF?
   - The idea of CTF is to build a small size, low-fusion power DT plasma-based device in which Fusion Nuclear Technology experiments can be performed in the relevant fusion environment at the smallest possible scale and cost.
     - In MFE: small-size, low fusion power can be obtained in a low-Q plasma device.
     - Equivalent in IFE: reduced target yield and smaller chamber radius (W. Meier Presentation).
   - This is a faster, much less expensive approach than testing in a large, ignited/high Q plasma device for which tritium consumption, and cost of operating to high fluence are very high (unaffordable!, not practical).

# Short Answers to Key Questions (Cont'd)

7. Is CTF Necessary? Most Definitely, ***but this is not the right question***. The right question is:

   *Will ITER plus CTF as the only DT Fusion Facilities be sufficient to have a successful DEMO?*

   Maybe, but we know for sure that, at a minimum, we need:

   - extensive developmental programs on ITER, CTF, and non-fusion facilities.
   - this work to begin sooner rather than later, before the tritium supply window closes, to have any hope that DEMO starts in 35 years.

   [And remember how many fission test reactors were built.]

# Blanket/PFC Concepts, FNT Issues, and Testing Requirements

# Blanket and PFC Serve Fundamental and Necessary Functions in a DT Fusion System

- **TRITIUM BREEDING at the rate required to satisfy tritium self-sufficiency**
- **TRITIUM RELEASE and EXTRACTION**
- **Providing for PARTICLE PUMPING (plasma exhaust)**
- **POWER EXTRACTION from plasma particles and radiation (surface heat loads) and from energy deposition of neutrons and gammas at high temperature for electric power production**
- **RADIATION PROTECTION**

Important Points

- All in-vessel components (blankets, divertor, vacuum pumping, plasma heating antenna/waveguide, etc.) impact ability to achieve **tritium self-sufficiency**.
- **High temperature** operation is necessary for high thermal efficiency. And for some concepts, e.g. SB, high temperature is necessary for tritium release and extraction.
- All the above functions must be performed **safely** and **reliably**.

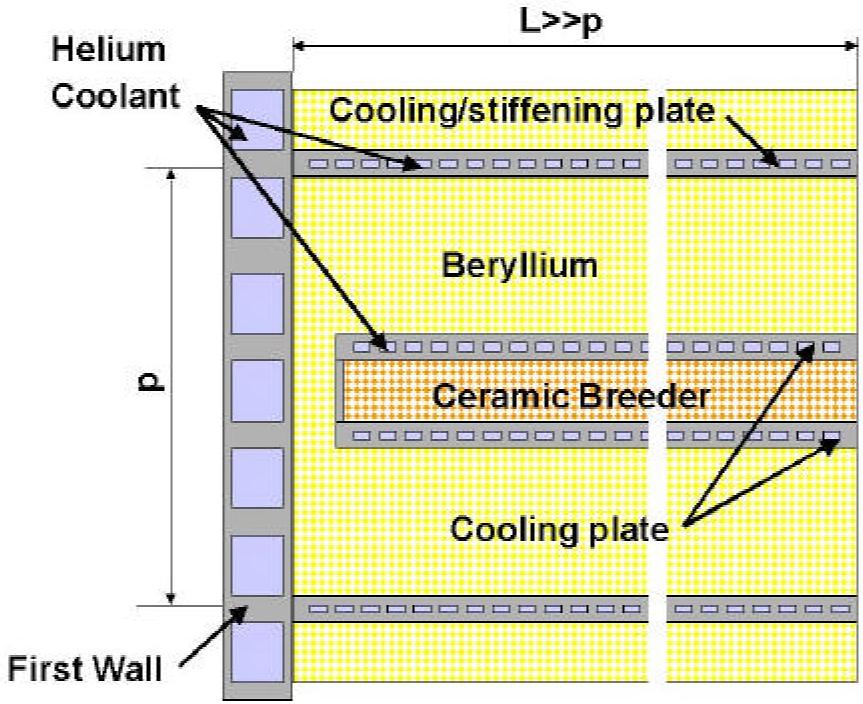
# Specific Blanket Options (Worldwide)

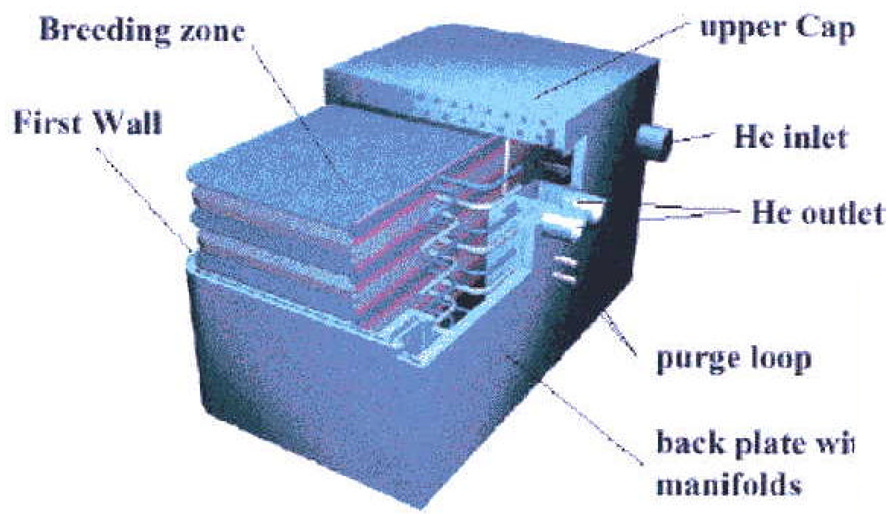
| Options | Breeder/Multiplier | Coolant | Purge | Structure | Insulator |
|---|---|---|---|---|---|
| **EU** Demo & 1st generation plants | Pb-17Li | He (8 MPa) | --- | Ferritic+ | |
| | Li-Ceramic/Be | He (8 MPa) | He 0.13 MPa | Ferritic | |
| 2nd generation plants | Pb-17Li | Pb-17Li & He | --- | Ferritic | SiC Insert |
| | Li-Ceramic/Be | He | He | SiC/SiC | |
| | Pb-17Li | Pb-17Li | --- | SiC/SiC | |
| **JA** Demo | $Li_2O(Li_2TiO_3)$/Be | $H_2O$ & He | He | Ferritic | |
| LHD (Univ.) | Flibe | Flibe | | Ferritic | |
| **USA** APEX* Studies | Li | Li | --- | Ferritic/V | Coating |
| | Flibe(Flinabe)/Be | Flibe/Flinabe | | Ferritic | |
| | Li-Ceramic/Be | He | He | Ferritic | |
| ARIES Studies | Pb-17Li | Pb-17Li | --- | SiC/SiC | |
| | Pb-17Li | He | --- | Ferritic | SiC Insert |

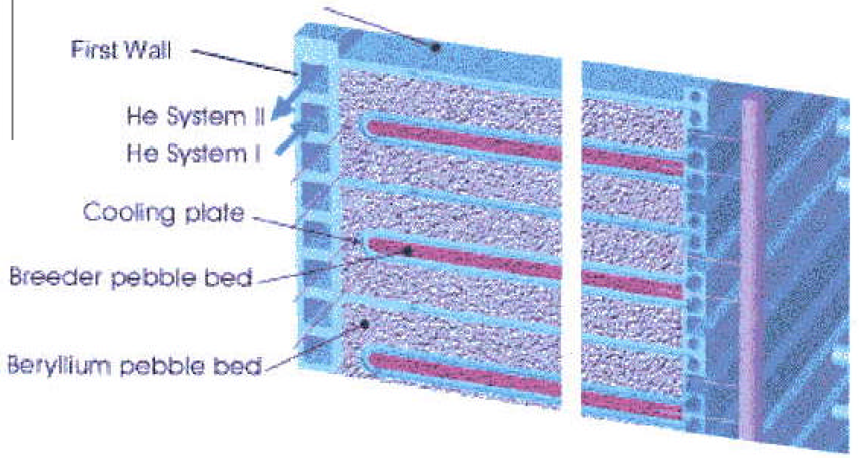
*** APEX considers both bare solid wall and thin (2 cm) plasma-facing liquid on first wall and divertor**

**+ Advanced Ferritic Steels are often proposed for designs using ferritic**

# A Helium-Cooled Li-Ceramic Breeder Concept is Considered for EU (Similar Concept also in Japan, USA)




## Material Functions

**Beryllium** (pebble bed) for neutron multiplication

**Ceramic breeder**($Li_4SiO_4$, $Li_2TiO_3$, $Li_2O$, etc.) for tritium breeding

**Helium purge** to remove tritium through the “interconnected porosity” in ceramic breeder

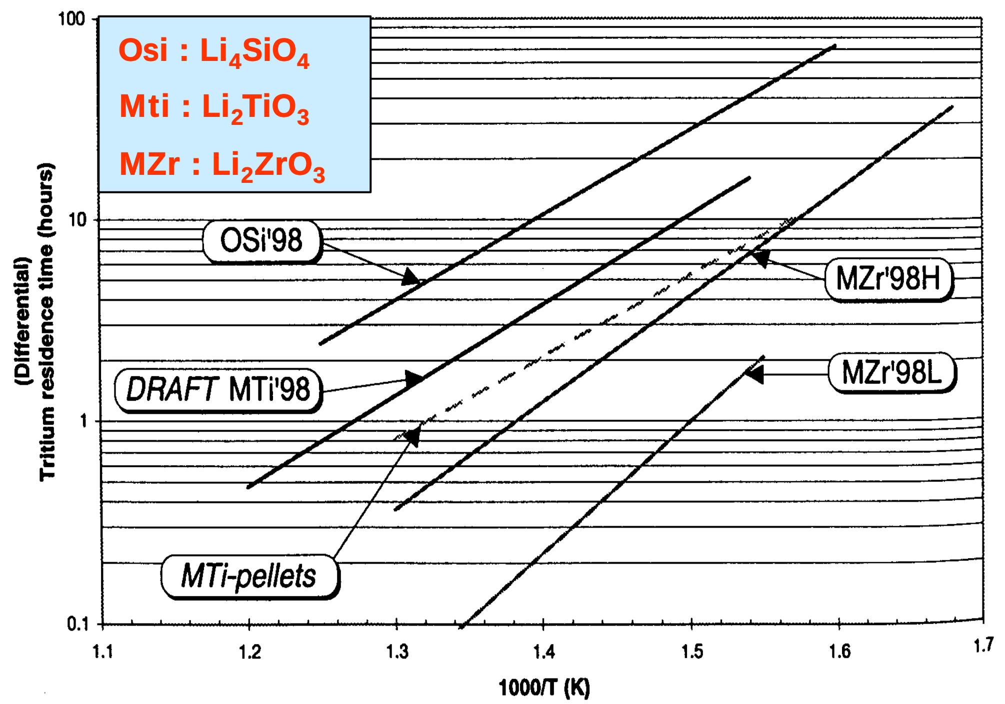
**High pressure Helium** cooling in structure (advanced ferritic)

Several configurations exist to overcome particular issues

**Geometric Configurations** and **Material Interactions** among breeder/Be/coolant/structure represent critical feasibility issues that require testing in the fusion environment

- Configuration (e.g. wall parallel or "head on" breeder/Be arrangements) affects TBR and performance
- Tritium breeding and release
  - **Max. allowable temp. (radiation-induced sintering in solid breeder inhibits tritium release; mass transfer, e.g. LiOT formation)**
  - **Min. allowable Temp. (tritium inventory, tritium diffusion**
  - **Temp. window (Tmax-Tmin) limits and $k_e$ for breeder determine breeder/structure ratio and TBR**

Tritium release characteristics are highly temperature dependent




- Thermomechanics interactions of breeder/Be/coolant/structure involve many feasibility issues (cracking of breeder, formation of gaps leading to big reduction in interface conductance and excessive temperatures)

# Tests for Thermomechanics Interactions of Be/Breeder/He-purge/Structure require "volumetric" heating in complex geometry (fission then fusion)

**A Case Study** ⇨ **HICU Project: A High Fluence Irradiation on Ceramic Breeder Pebble Beds with Mechanical Constraints in Fission Reactor**

## Project goals:

"the investigation of the impact of neutron spectrum and the influence of constraint conditions on the thermo-mechanical behavior of breeder pebble-beds in a high fluence irradiation"

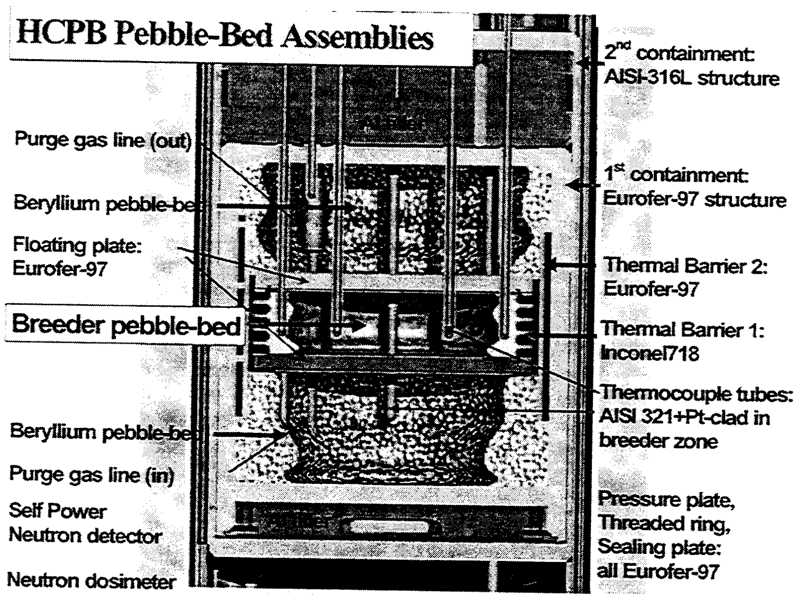
**$Li_2O$ ceramic breeder**

## Main critical issues for the "project"

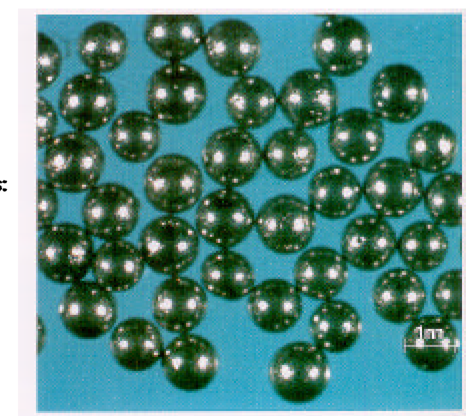
concern the specimen size and the geometry(limited test volume in fission reactor)

Instrumentation

**(neutron dosimeter, thermocouples, tritium monitor)**



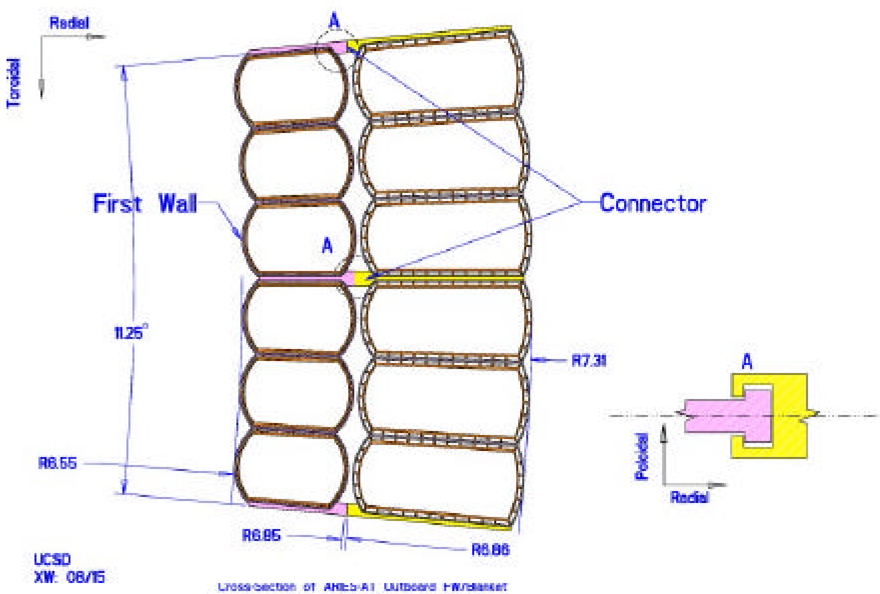

**Schematic view of pebble-bed assembly, showing cross-section of test-element, second containment and instrumentation**

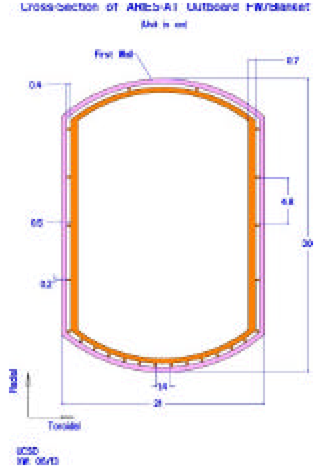


**Beryllium pebble**

# ARIES-AT blanket with SiC composite structure and Pb-17Li coolant and tritium breeder

Cross-Section of ARIES-AT Outboard FW/Blanket
(One Segment)



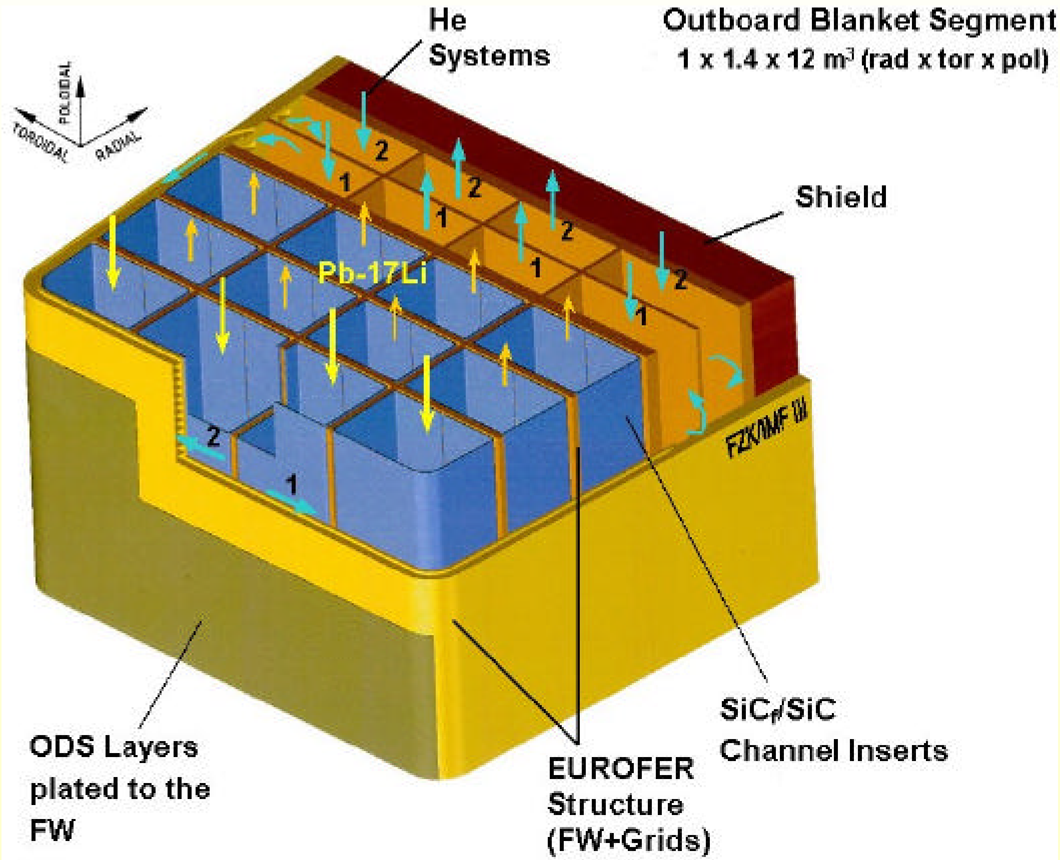
ARIES-AT Outboard First Wall and Blanket Segment

Pb-17Li Operating Temperature

Inlet: 654 $^{o}$C

Outlet: 1100 $^{o}$C

# A Dual-Coolant Concept for EU 2nd Generation Plants (similar to ARIES-ST)




- **Dual coolant: He and Pb-17Li**
- **Coolant temperature (inlet/outlet, °C)**
  - 460/700 (Pb-17Li)
  - 300/480 (He)
- **SiC/SiC inserts to allow Pb-17Li operated at temperature greater than the allowable ODS/Pb-17Li corrosion temperature limit**

# MHD and Insulators are Critical Issues

## Engineering Feasibility will be proven only through Integrated Tests

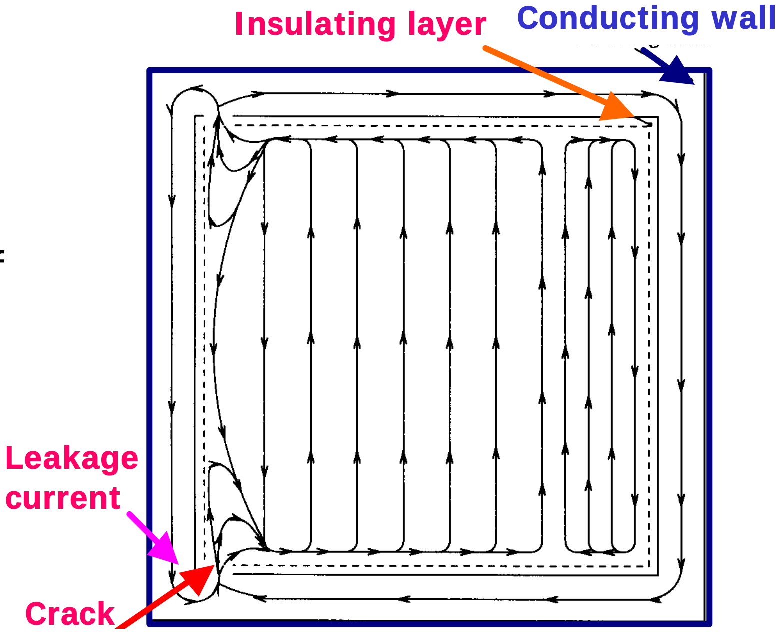
**MHD** is critical issue for liquid-metal-cooled blankets and PFC's

**Insulators** are required: Ceramic coatings have been proposed

**Key issue:** disparate thermal expansion coefficient, low tensile strength and poor ductility of ceramic coatings compared to pipe wall heated under cyclic operations will lead to significant cracking of the coating. Once a crack is generated it forms an electrical circuit for leakage current – leading to critical increase MHD pressure drop.

**Therefore,** rapid self-healing of coating is mandatory. Healing speed will depend on the details of crack generation rate and size – currently unknown and unpredictable.

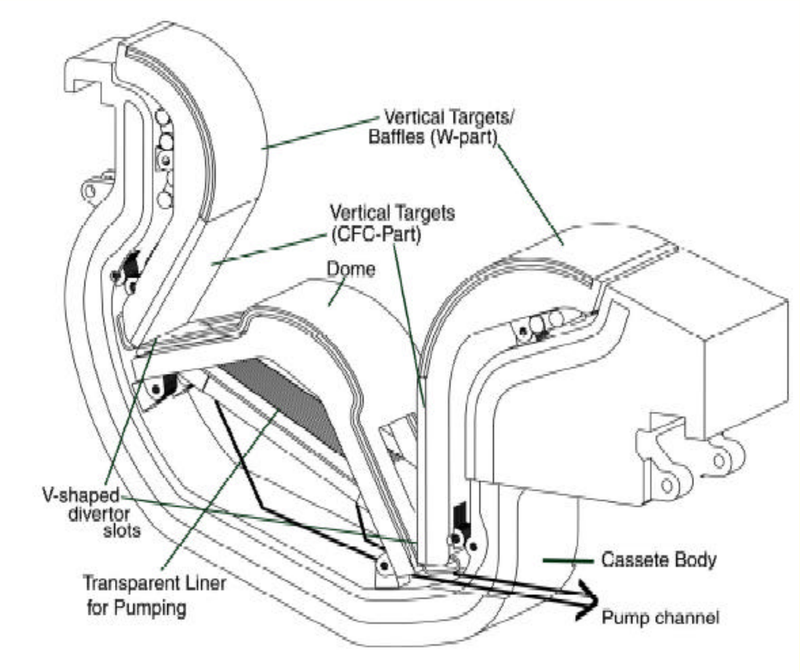
**Meaningful testing** of the performance of this thin insulating layer **can only be performed in a multi-effect environment** with: (1) high temperature and strong temperature gradients (volumetric nuclear heating), (2) electric and magnetic fields, (3) stress and stress gradients, (4) prototypic material and chemical systems and geometry, and (5) radiation effects.




Leakage of Electric currents in 2D channel with cracked insulator coating

# PFC Development

- Highest heat flux component in a fusion device (10-20 MW/m$^2$)
- Closely coupled to plasma performance
- Cyclic Power excursions (ELMs & Disruptions) erosion lifetime
- Limited materials choices (W, Mo, Ta, Nb?, C?, Liquids: Li, Ga, Sn)
- High neutron fluence
- Tritium retention (C)
- Joining, fabrication, and coolant compatibility issues

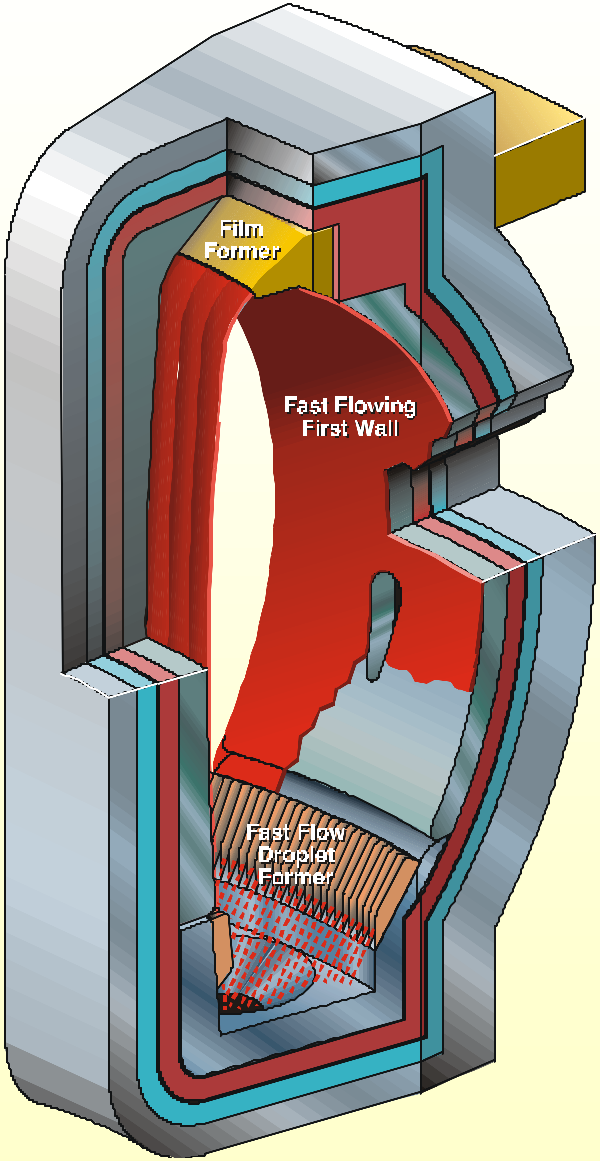



ITER-FEAT Divertor Cassette

Note: PFC, Blanket, rf antennas, and other in-vessel components in reactor “core” must be compatible and they collectively play a major role in key FNT issues, e.g. Tritium Self-Sufficiency.

# Role of Liquid Walls in Blanket and PFC Development

- Liquid Walls are being pursued in the US for many potential benefits (removal of high surface heat flux, increased potential for disruption survivability, reduced thermal stresses in structural materials, possible improvements in plasma confinement and stability, etc.)
- The focus of the on-going R&D Program in laboratory experiments and plasma devices is on a thin liquid wall (~2 cm) on the plasma-facing side of the first wall and divertor
- No major changes in Fusion Nuclear Technology Development Pathways are necessary for thin liquid walls. If thin liquid walls prove feasible (e.g. from NSTX liquid surface module), they can be easily incorporated into CTF (and also, hopefully, into ITER at later stages) and DEMO



# Table XV*: Capabilities of Non-Fusion Facilities for Simulation of Key Conditions for Fusion Nuclear Component Experiments

| | Neutron Effects(1) | Bulk Nuclear Heating(2) | Non-Nuclear(3) | Thermal / Mechanical / Chemical / Electrical(4) | Integrated Synergistic |
|---|---|---|---|---|---|
| Non-Neutron Test Stands | no | no | partial | partial | no |
| Fission Reactor | partial | partial | no | no | no |
| Accelerator-Based Neutron Source | partial | no | no | no | no |

(1) radiation damage, tritium and helium production, transmutations

(2) nuclear heating in a significant volume

(3) magnetic field, surface heat flux, particle flux, mechanical forces

(4) thermal-mechanical-chemical-electrical interactions (normal and off normal)

* From Fusion Technology, Vol. 29, pp 1-57, January 1996

# Table XX.*

*Characteristic Time Constants in Solid Breeder Blankets*

| Process | Time Constant |
|---|---|
| **Flow** | |
| Solid breeder purge residence time | 6 s |
| Coolant residence time | 1 to 5 s |
| **Thermal** | |
| Structure conduction (5-mm metallic alloys) | 1 to 2 s |
| Structure bulk temperature rise | |
| 5 mm austenitic steel / water coolant | ~1 s |
| 5 mm ferritic steel / helium coolant | 5 to 10 s |
| Solid breeder conduction | |
| $Li_2O$ (400 to 800ºC) | |
| 10 $MW/m^3$ | 30 to 100 s |
| 1 $MW/m^3$ | 300 to 900 s |
| $LiAlO_2$ (300 to 1000ºC) | |
| 10 $MW/m^3$ | 20 to 100 s |
| 1 $MW/m^3$ | 180 to 700 s |
| Solid breeder bulk temperature rise | |
| $Li_2O$ (400 to 800ºC) | |
| 10 $MW/m^3$ | 30 to 70 s |
| 1 $MW/m^3$ | 80 to 220 s |
| $LiAlO_2$ (300 to 1000ºC) | |
| 10 $MW/m^3$ | 10 to 30 s |
| 1 $MW/m^3$ | 40 to 100 s |
| **Tritium** | |
| Diffusion through steel | |
| 300ºC | 150 days |
| 500ºC | 10 days |
| Release in the breeder | |
| $Li_2O$ 400 to 800ºC | 1 to 2 h |
| $LiAlO_2$ 300 to 1000ºC | 20 to 30 h |

* From Fusion Technology, Vol. 29, pp 1-57, January 1996

# Table XXI.*

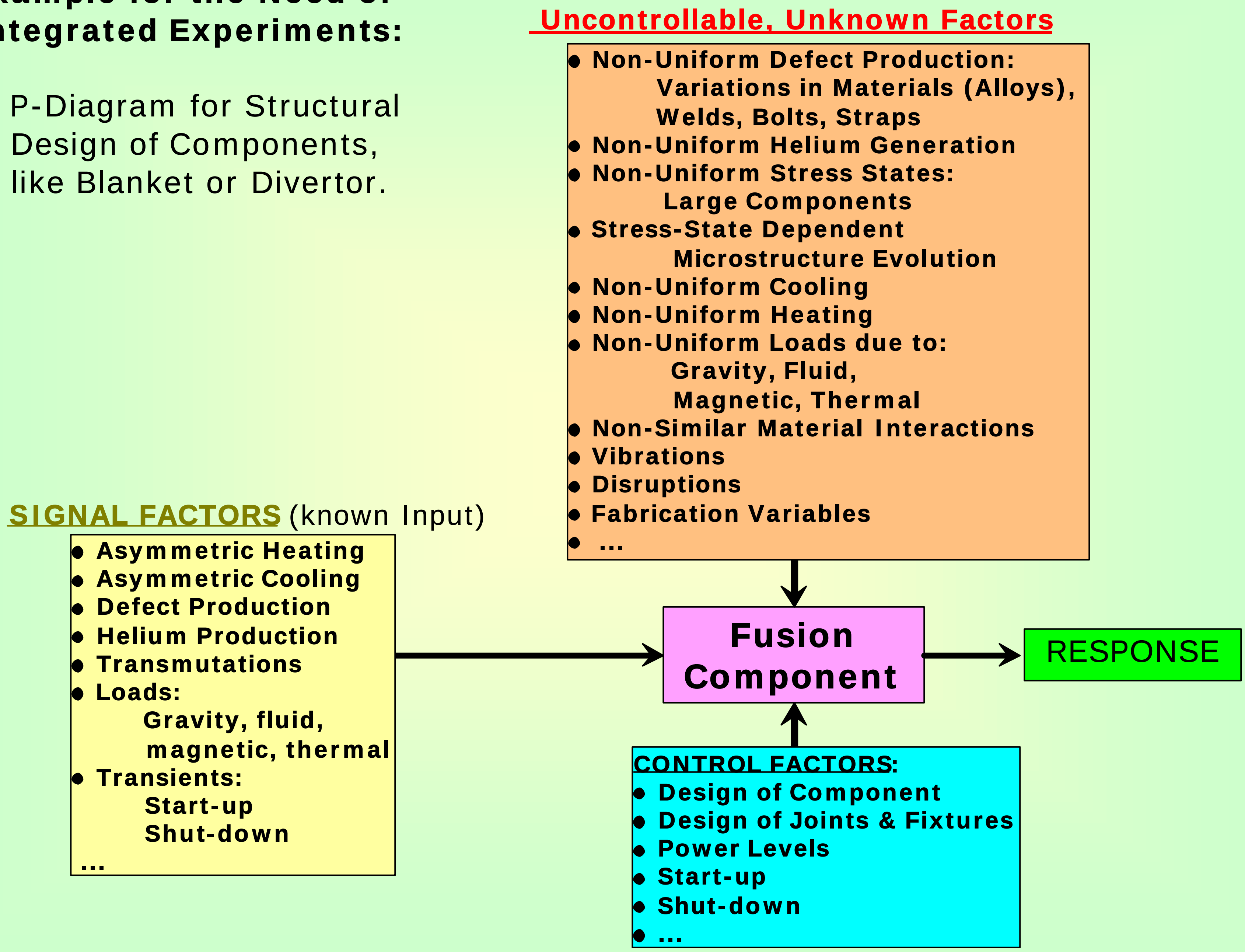
*Characteristic Time Constants in Liquid-Metal Breeder Blankets*

| Process | Time Constant |
|---|---|
| **Flow** | |
| Coolant residence time | |
| First wall (*V*=1 m/s) | ~30 s |
| Back of blanket (*V*=1 cm/s) | ~100 s |
| **Thermal** | |
| Structure conduction (metallic alloys, 5mm) | 1 to 2 s |
| Structure bulk temperature rise | ~4 s |
| Liquid breeder conduction | |
| Lithium | |
| Blanket front | 1 s |
| Blanket back | 20 s |
| LiPb | |
| Blanket front | 4 s |
| Blanket back | 300 s |
| **Corrosion** | |
| Dissolution of iron in lithium | 40 days |
| **Tritium** | |
| Release in the breeder | |
| Lithium | 30 days |
| LiPb | 30 min |
| Diffusion through: | |
| Ferritic Steel | |
| 300°C | 2230 days |
| 500°C | 62 days |
| Vanadium | |
| 500°C | 47 min |
| 700°C | 41 min |

* *From Fusion Technology, Vol. 29, pp 1-57, January 1996*

## Example for the Need of Integrated Experiments:

P-Diagram for Structural Design of Components, like Blanket or Divertor.



# FW-Mock Up Fatigue Testing at FZK

## Shows an example of unexpected failure modes that cannot be predicted by models.

*(Information from Eberhard Diegele at FZK)*

- Thermo-mechanical fatigue test were performed for FW-mock ups from SS 316 L.
  - Loading conditions: about 0.7 MW/m2 heat flux (Fig. 1)
- The specimens were pre-cracked (notched) perpendicular to the coolant tubes at different locations with different sizes (Fig. 2)
- After 75,000 cycles the notched cracks grew to the sizes as indicated.
- **However, unexpectedly there were longitudinal cracks that were initiated in *every channel* - and these cracks grow under fatigue and would have led to failure if the experiment continued.**

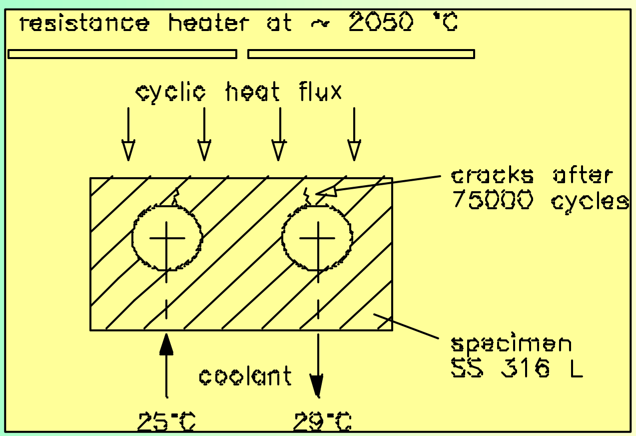



Fig.1: Schematic of FW-Mock Up

**From elastic-plastic fracture mechanics modeling:**

- **Expected the large pre-cracks at the crown of the channel to fail.**
- **Initiation and growth of the longitudinal cracks were not and can not be predicted by models.**

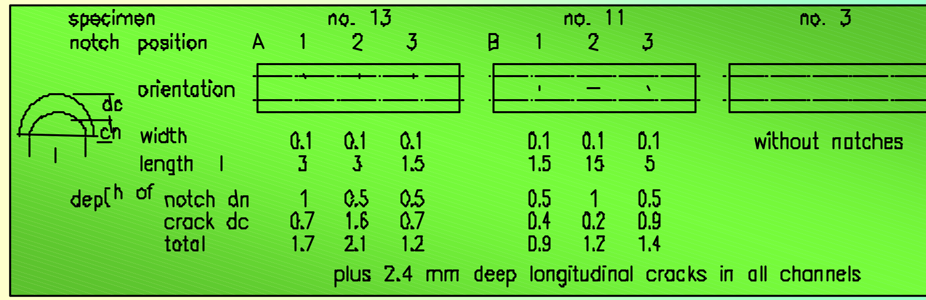



Fig.2:Spark eroded notches and cracks after 75,000 cycles

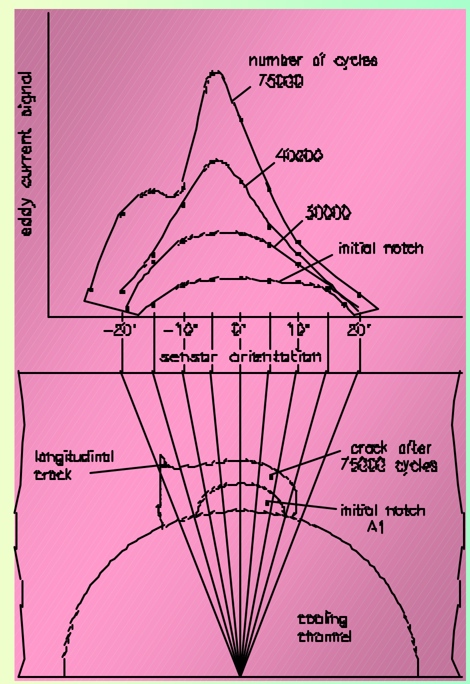



Fig.3: Crack measurements

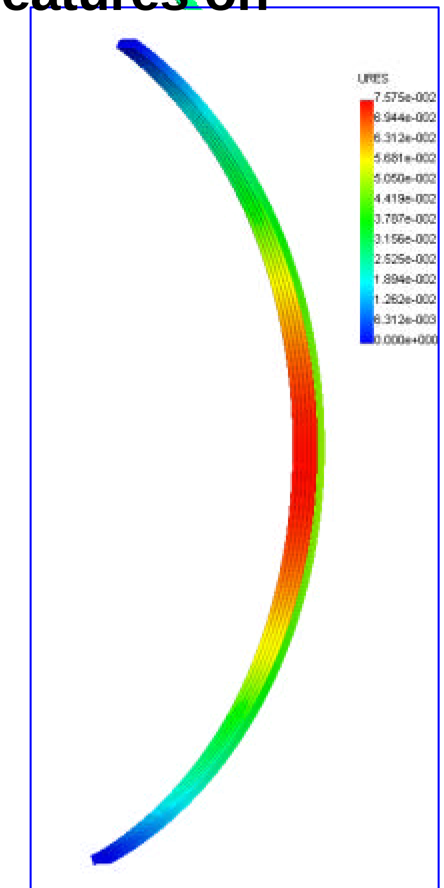
# FW-Panel Displacement:

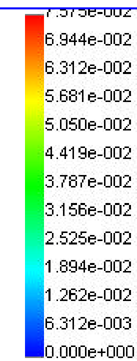
FW_all_channels_DS-Fixed Vertex & No rotation :: Static Displacement

The Movie shows the displacement at a **1:1 Scale**

**Effects of 3-D Geometric Features on Displacement:**

**FW Central Portion Experiences largest Displacement**




**BC:**

Bottom and Top Face are Fixed
No Rotational Freedom along the back

Max Displacement at Center ~ 7.3 cm with no back support. With back support, these displacements must be accommodated through higher stresses

- **To Achieve DEMO Availability = 48%**

| | Required Blanket Availability |
|---|---|
| R. Buende (1989) | 97% |
| IEA-VNS (1996) | 90% |

- **To Achieve DEMO Availability = 30%**

J. Sheffield (2002): Required blanket availability = 88%

(Assuming Major MTTR = 800 h, Minor MTTR = 100 h)

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Required MTBF for DEMO Blanket

Depends on availability requirements and MTTR

| DEMO Availability | Required Blanket Availability | Required MTBF for a Blanket Module (100 modules, MTTR=1 month) |
|---|---|---|
| **30%** | **88%** | **60 yr** |
| **48%** | **90%** | **75 yr** |

Notes from M. Abdou and D. Sze in response to a question received on 10/25/2002.

# Is "Batch" Processing together with "low temperature blanket" a good "transition" option?

**Batch Processing**

--Evaluated in the 1970s

--Conclusion: Not Practical for the "complex" fusion devices

1. In large systems like a tokamak: It takes a long time to remove/reinsert blankets. You still have to go through the vessel, the shield, and the magnet support. (for example: several months in ITER); therefore you cannot do it frequently (once every two years?!).
2. In 1000 MW Fusion Power Device, the tritium consumption is 55.8 kg per full power year. So, for 20% availability, tritium inventory accumulated in 2 years is >22 kg (in addition to the "hold up" inventories in PFCs and other in-vessel components).
3. Safety experts have suggested much lower targets for tritium inventory (~2 kg). Note also that tritium will decay at 5.47%/year and you will have to provide external start up inventory, plus inventory for duration of "first batch".
4. And "there is really no effective way to recover tritium from the blanket using a batch process."

# Low-Temperature Blanket?

**Evaluated during INTOR, ITER-CDA, ITER-EDA**

**Assessment:**

-- It is still high risk because we use technologies unvalidated in the fusion environment.

-- There is no good low-temperature breeding blanket option. You can have only “partly” low-temperature.

-- “Partly” low-temperature breeding blankets have their added complications and issues for which an additional R&D program is needed.

# Options for Low-Temperature Blanket?

- **All self-cooled liquid metal options require high temperature (>300°C) because of high melting point. We do not know if any of them are feasible in the fusion environment because of issues such as insulators, tritium barriers, etc.**

- **Separately-cooled LiPb requires either Helium or water, both above 300°C. Practically all feasibility issues for "reactor-type" blankets are the same and must be resolved by extensive testing first in the fusion environment.**

# Options for Low-Temperature Blanket? (cont'd)

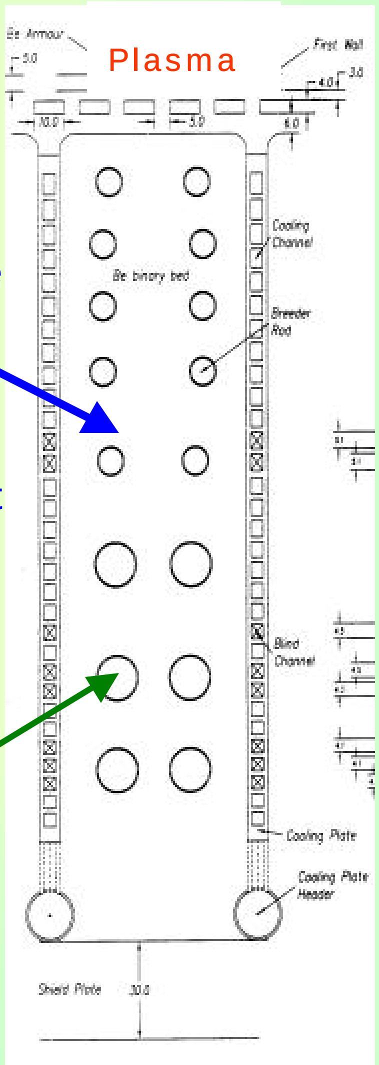
- **Solid Breeder Options were evaluated in INTOR, and ITER-CDA, ITER-EDA**

-- Breeder must run at high temperature

-- Only the coolant can be low temperature

-- All the feasibility issues with the breeder and multiplier are essentially the same as those for reactor-type blanket. But with the added complexity of providing "thermal resistance" between the low-temperature coolant and the hot solid breeder.

-- Both stainless steel and ferritic steel have severe embrittlement problems at low-temperature (ITER can use low-temperature coolant in the present non-breeding design only because of the very low fluence).



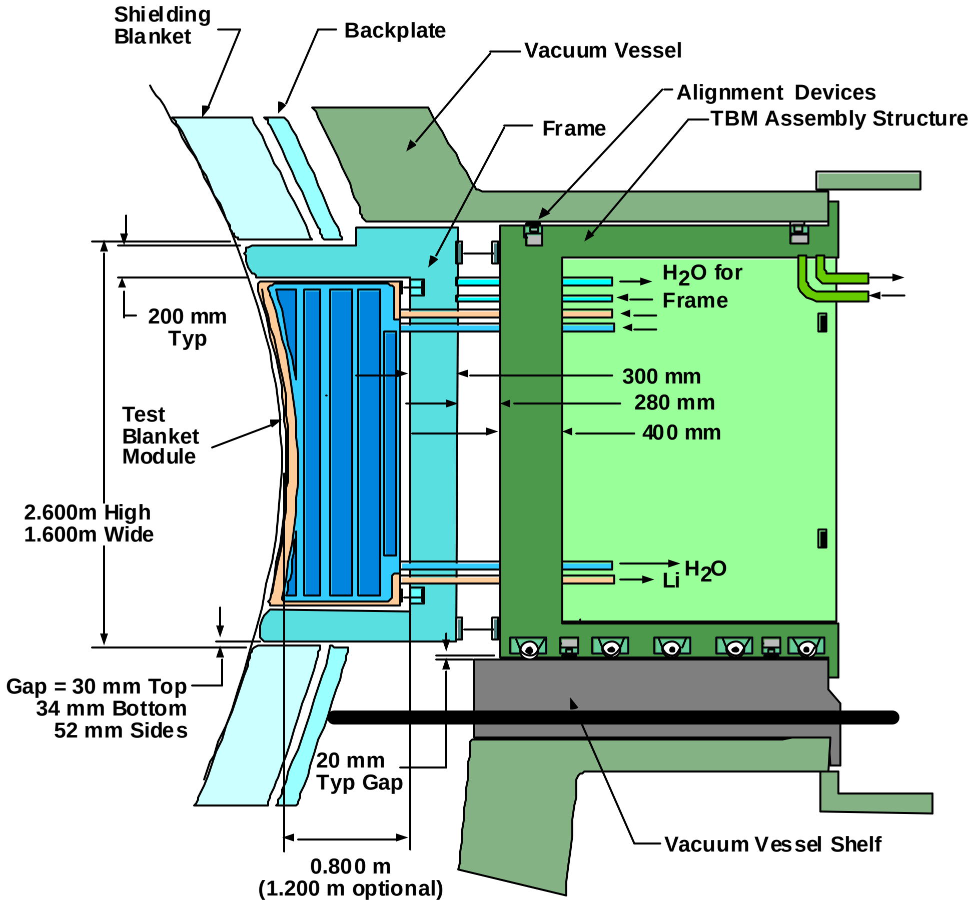

Beryllium pebble bed is used as a temperature barrier in a low temperature breeding blanket design

Breeder pebble bed rod

# Engineering Requirements for CTF Test Program

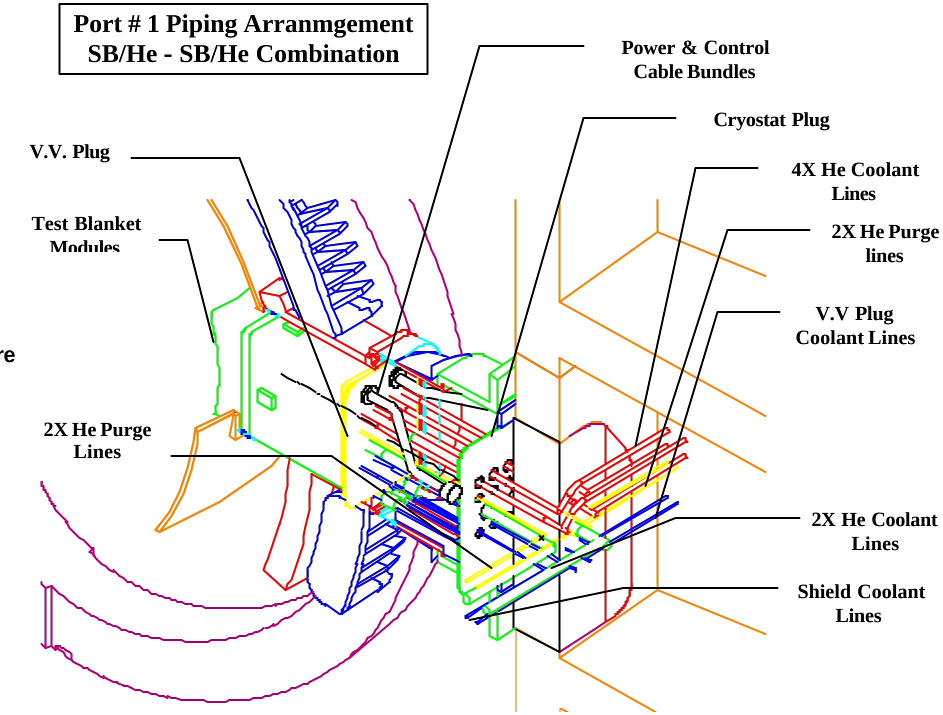
- **Exposure of test module first wall to plasma**
  - **Surface heat flux is crucial for blanket test**
  - **Thickness of first wall is crucial for tritium self sufficiency, stress, etc.**
- **Easy and fast access to place and remove test articles**
  - **access to inside of vacuum vessel without welding and rewelding**
- **Sufficient space at the first wall**
  - **Adequate dimensions in the poloidal and toroidal directions for test articles**
  - **Space around test modules for boundary conditions**
- **Space outside the reactor for ancillary equipment and control**
- **Space for manifolds, access lines, and instrumentation**

# Example Test Program Modules

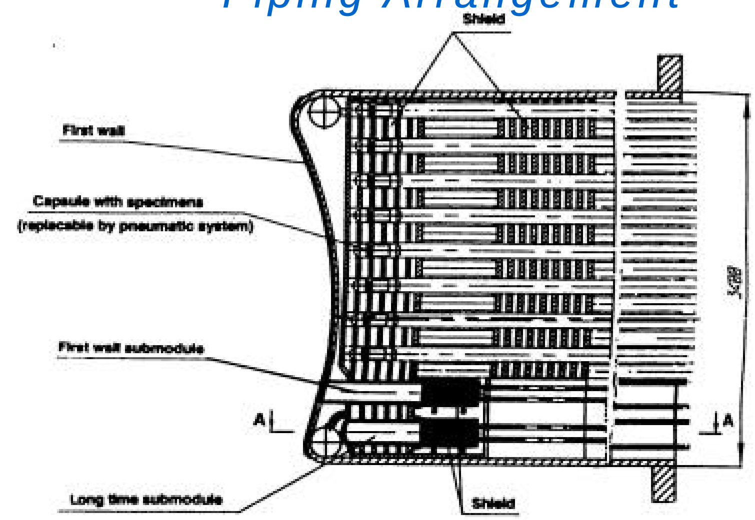



Prepared by the US Test Blanket Working Group

*Liquid Lithium Blanket Modules in Horizontal Port*




*Solid Breeder Blanket Module and Piping Arrangement*



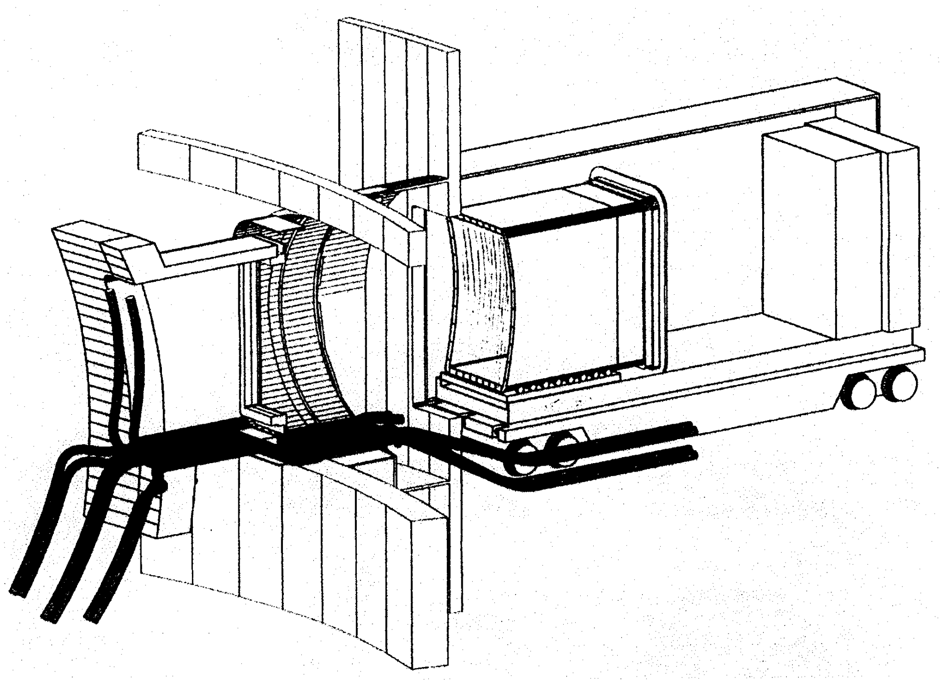

*Material Module*

# Test Module Design Strategy

- **Because of the reduced operating conditions of CTF v.s. Demo (i. e. neutron and surface wall loads), an engineering scaling test module design approach is necessary**
  - **calculate Demo key performance parameters**
  - **design test module to reproduce these parameters such as resizing wall thickness, coolant spacing, etc.**
- **3 Types of Test Module Designs:**
  - **Demo Act-Alike (majority of tests)**
  - **Demo Look-Alike (useful for neutronics)**
  - **CTF optimized component concepts**
- **Multiple integrated modules exposed to the plasma are proposed for initial fusion break-in tests**
  - **fully-integrated tests can only be done in fusion testing facility, and should take higher priority**
  - **issue specific tests can be carefully designed into small scale submodules**

# CTF Test Port Engineering Considerations

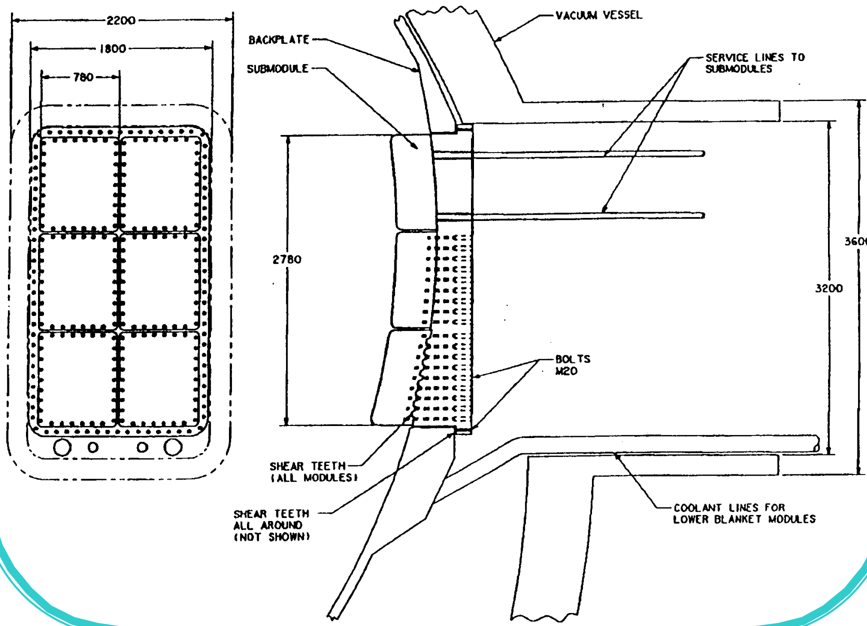
- **Minimal Impact on CTF Design**
  - **Use a Common Interface Design for RF, Diagnostic, Maintenance, and Test Ports**
- **Minimal Impact on CTF Operations**
  - **Access Test Modules only through Horizontal Test Ports**
  - **Employ Isolation Valve in Test Port Extension**
    - **Does not disturb chamber vacuum to change module or submodule**
  - **Use Dedicated Test Port Remote Handling Equipment**



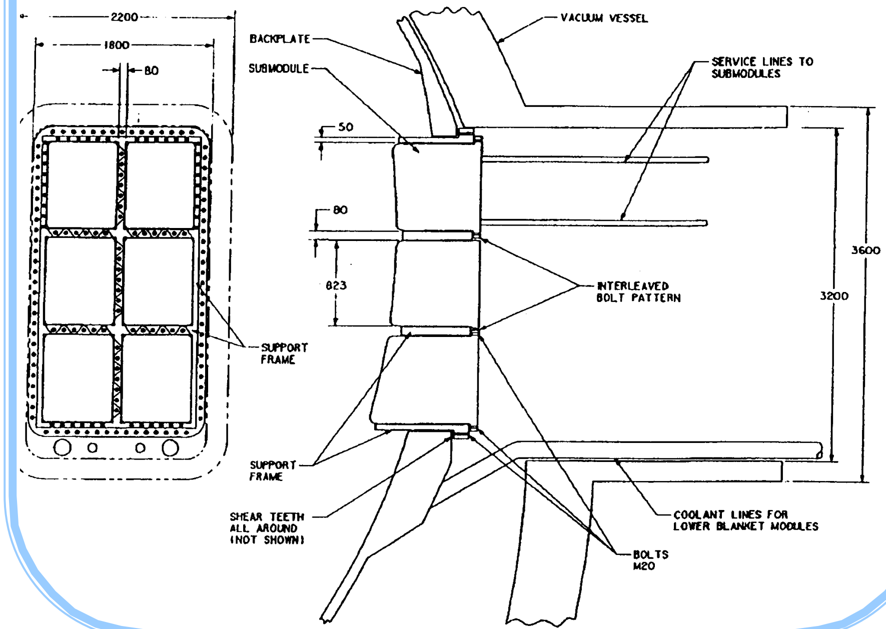
# Test Port Design Options

**Design Goal: To Seamlessly Interface with the Basic CTF Device such that the Design and Operation of CTF will be Minimally Impacted**

## Frameless Test Port Assembly
## Front Loading Approach

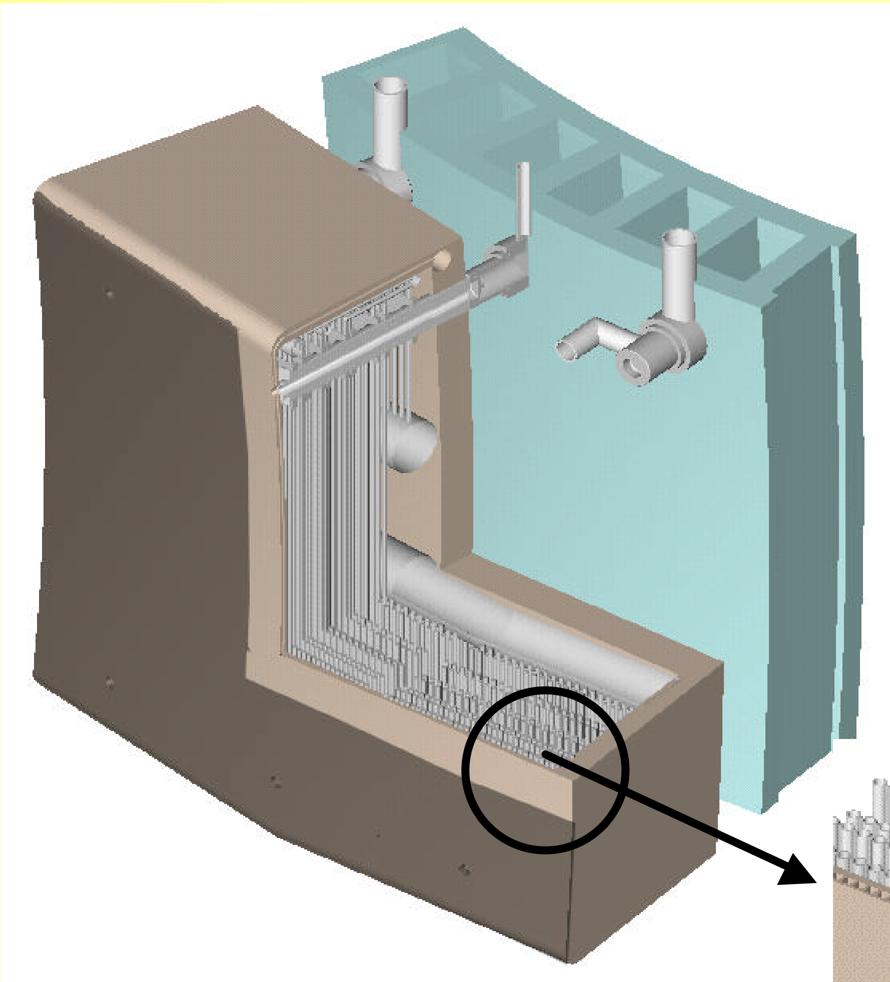



## Framed Test Port Assembly
## Rear Loading Approach

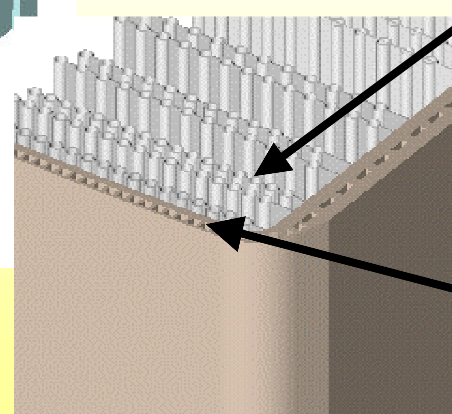


From Akiba, Japan: Typical Blanket Module in DEMO

# Schematic of Test Blanket Module



| Typical Blanket Module | |
|---|---|
| Weight | 4 ton |
| Height | 1 m |
| Width | 2 m |
| Thickness | 0.6 m |
| Number of modules | 256 |



# Tritium Self Sufficiency is a Serious Issue