Blanket Technology, Fuel Cycle and Tritium Self Sufficiency

Mohamed Abdou

Distinguished Professor of Engineering and Applied Science (UCLA)
Director, Fusion Science and Technology Center (UCLA)
President, Council of Energy Research and Education Leaders, CEREL (USA)

With input from members of the US FNST community

Related publications can be found at www.fusion.ucla.edu

JASON Study on Tritium Production in Fusion
San Diego, CA, June 27-28, 2011

Summary Points

- The importance of tritium breeding blankets has been recognized since the beginning of fusion reactor studies in 1970. Many (>50) blanket design studies carried out to date.
- The blanket is a complex system, with multiple functions and multiple materials in multiple-field environment.
  - Multiple effects and synergistic phenomena will be dominant.
  - True simulation will be possible only in DT fusion facility.
- Many design studies, modelling activities, and experiments (primarily separate-effect) have been carried out.
- The key issues have been identified and characterized. The required R&D to resolve these issues have been well defined. We understand the path forward.
- We have identified a “phase space” of physics and technology parameters and conditions in which tritium self sufficiency can be attained. Our R&D in plasma physics, blanket technology, and fuel cycle is aimed at ensuring tritium self sufficiency.
- We have analyzed the engineering issues of blanket and fuel cycle system practicality and attractiveness. There are many challenges, but the field is now positioned to move forward with an extensive R&D Program toward multiple-effect and integrated testing in fusion facilities (ITER TBM and FNSF).

Blanket Technology, Fuel Cycle and Tritium Self Sufficiency

OUTLINE

Tritium Breeding Blanket

- Functions, environmental conditions, integration with nuclear components
- History of Blanket design and FNST studies
- Currently favored blanket design options; representative designs and operating parameters; and representative modelling & experimental results

Tritium Fuel Self Sufficiency

- Fuel Cycle key elements, features, interfaces, and technological issues
- Dynamic Modelling and Analysis of Fuel Cycle
- “Phase Space” for attaining tritium fuel self sufficiency

Summary of ISSUES and Path Forward

- Summary of key blanket/FNST issues
- Science–Based R&D framework
- ITER Test Blanket Module (TBM) Study
- Needed R&D in non-fusion facilities and DT fusion facilities (ITER and FNSF)

We have addressed the Blanket together with the tritium fuel cycle as an integral element of Fusion Nuclear Science and Technology (FNST).

Fusion Nuclear Science & Technology (FNST)

FNST is the **science**, **engineering**, **technology** and **materials**
for the fusion nuclear components that
generate, control and utilize neutrons, energetic particles & tritium.

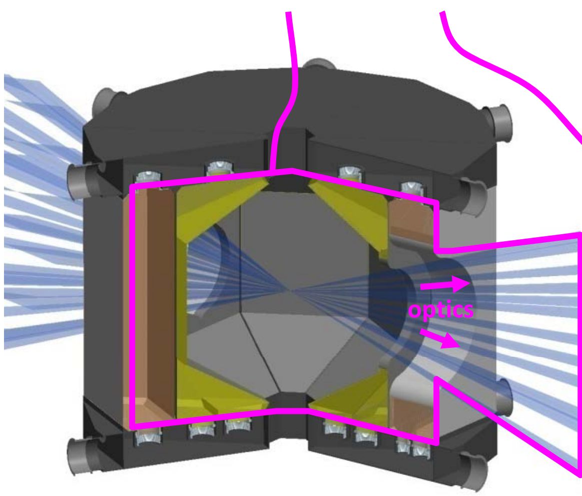
In-vessel Components

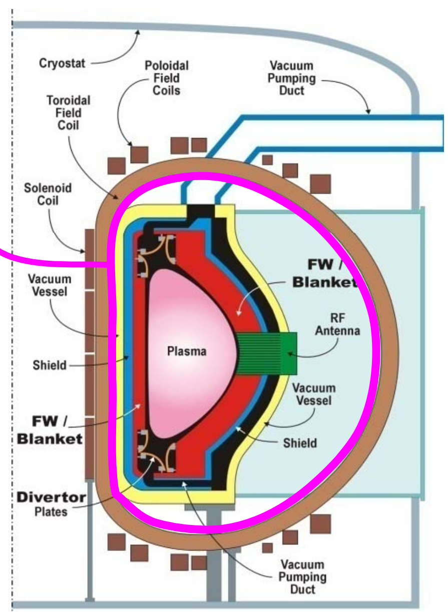
- **Plasma Facing Components**
  divertor, limiter, heating/fueling and final optics, etc.
- **Blanket** and **Integral First Wall**
- **Vacuum Vessel** and **Shield**

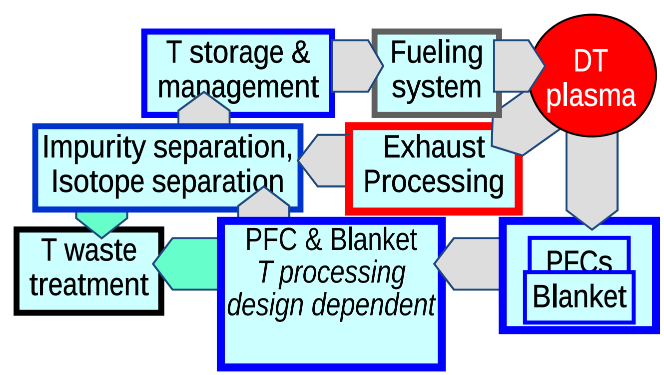
The nuclear environment also affects

- **Tritium Fuel Cycle**
- **Instrumentation & Control Systems**
- **Remote Maintenance Components**
- **Heat Transport & Power Conversion Systems**

These are the FNST Core
for IFE & MFE







Fusion Nuclear Environment is Complex & Unique

Neutrons *(flux, spectrum, gradients, pulses)*
- Radiation Effects
- Tritium Production
- Bulk Heating
- Activation and Decay Heat

Heat Sources *(thermal gradients, pulses)*
- Bulk (neutrons)
- Surface (particles, radiation)

Particle / Debris Fluxes *(energy, density, gradients)*

Magnetic Fields *(3-components, gradients)*
- Steady and Time-Varying Field

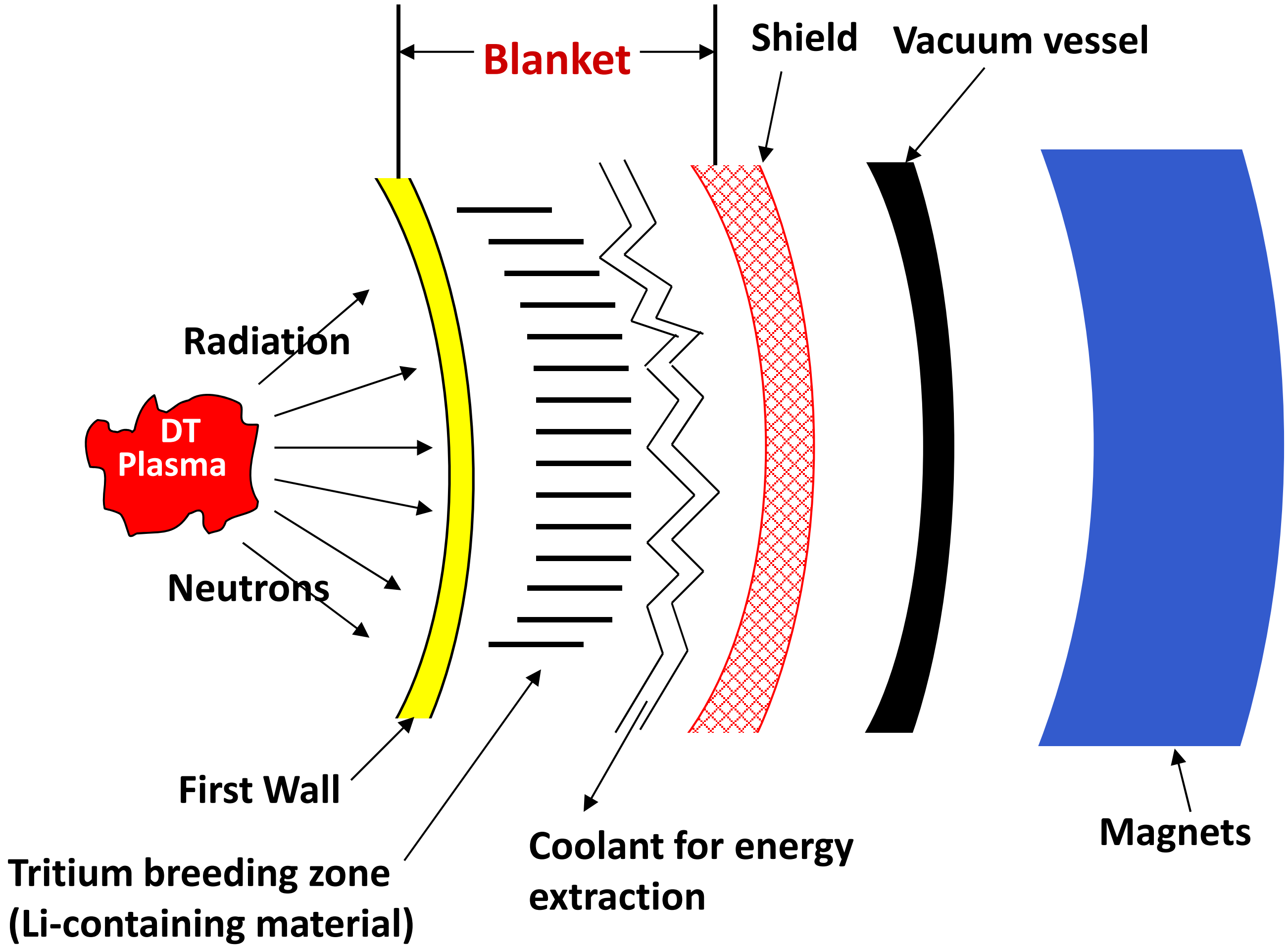
Mechanical Forces
- Normal *(steady, cyclic)* and Off-Normal *(pulsed)*

Multiple functions, materials, and many interfaces in highly constrained system

Combined Loads, Multiple Environmental Effects
- Thermal-chemical-mechanical-electrical-magnetic-nuclear interactions and synergistic effects
- Interactions among physical elements of components

Non-fusion facilities (Laboratory experiments) need to be substantial to simulate multiple effects
Simulating nuclear **bulk heating in a large volume** is the most difficult and is most needed
Most phenomena are temperature (and neutron-spectrum) dependent– it needs DT fusion facility
The full fusion Nuclear Environment can be simulated only in DT plasma–based facility

The primary functions of the blanket are to provide for: Power Extraction & Tritium Breeding



- Liquid metals (Li, PbLi) are strong candidates as breeder/coolant.
- Ceramic Breeders with He cooling are also strong candidates.

There are many material and configuration options for the blanket

Material or Configuration	Options
Structural Materials	Reduced Activation Ferritic Steel Alloys (including ODS), Vanadium Alloys, SiC Composites
Coolant Media	Helium, Water, Liquid Metals, Molten Salts
Breeder Media	Lithium-Bearing: Ceramic Breeders (Li_4SiO_4, Li_2TiO_3, Li_2O); Liquid Metals (Li, PbLi, SnLi); Molten Salts (FLiBe, FLiNaBe); Varying enrichments in Li-6
Neutron Multiplier Materials	Beryllium, $Be_{12}Ti$, Lead
MHD/Thermal Insulator Materials	SiC composites and foams, Al_2O_3, CaO, AlN, Er_2O_3, Y_2O_3
Corrosion and Permeation Barriers	SiC, Al_2O_3, others
Plasma Facing Materials	Beryllium, Carbon, Tungsten alloys, others
HX or TX Materials	Ferritic Steels, Ni-based alloys, Refractory Alloys, SiC, Direct Gas Contact
Blanket Configurations	He or Water Cooled Ceramic Breeder/Be; Separately Cooled, Self-Cooled, Dual-Coolant LM or MS
Ceramic Breeder Configurations	Layered, Mixed, Parallel, Edge-On (referenced to FW), Breeder-In-Tube
Liquid Breeder Configurations	Radial-Poloidal Flow, Radial-Toroidal Flow, others
MHD/Thermal Insulator Config.	Flow Channel Inserts, Self-Healing Coatings, Multi-Layer Coatings
Structure Fabrication Routes	HIP; TIG, Laser and E-beam Welding; Explosive Bonding; Friction Bonding; Investment Casting; and others

But there are only a few compatible combinations possible

(because of considerations of chemical compatibility, safety, etc)

Evolution of Fusion Blanket Concepts during 40-yr History of Blanket Studies

- **The importance of blanket was recognized from Day 1 of fusion reactor studies.** In the first reactor study in 1970, UWMAK-I at Univ of Wisconsin, the Breeding blanket and neutronics tools/analysis/design to predict and ensure sufficient tritium breeding were central elements of the Study.

- **UWMAK-I selected self-cooled liquid lithium with SS as the simplest concept.**

  key points from this blanket design study were:

  a) Li at natural enrichment can breed well with no neutron multiplier.

  b) Tritium can be recovered with acceptable tritium inventory (1 Kg at that time.)

  c) Li can be circulated on the OB of the reactor, but not on the IB because of MHD effects.

  d) Insulating coatings should be used to minimize MHD issues.

- **In 1974, Victor Maroni (ANL) demonstrated that Tritium can be extracted from liquid lithium using a molten salt process with a potential to keep tritium inventory in lithium < 1 kg**

  (US Patent No 3957597 , May 18,1976), V. Maroni, et. al., Nuclear Technology, Vol. 25 (1975)

- **UWMAK-II introduced ceramic breeders as possibly attractive option for non-mobile blanket with low Li and T inventories.**

  Key points:

  - **BATCH Processing of tritium,** as commonly done in fission reactors, **is not acceptable** for fusion (large tritium inventory: for 3000 MW fusion power T production is ~ 170 Kg per year!)
  - **In –SITU T recovery is necessary.** A scheme of flowing low-pressure He gas in the ceramic breeder was introduced.
  - Neutron multiplier is required for adequate T breeding. Be was used , issue of limited Be reserves was raised.
  - SS is not adequate as structural material

Evolution of Fusion Blanket Concepts during 40-yr History of Blanket Studies (cont'd)

- **From the mid 1970's many blanket and reactor studies were performed in ANL, ORNL, LLNL, UW.**

- **UWMAK-III proposed liquid Li for Outboard, ceramic breeder for the Inboard (to eliminate MHD), and TZM structure to obtain high temperature**

  Key points:

  - minimizes MHD effects and eliminate the need for MHD insulators

  - Refractory Materials are expensive.- High thermal efficiency can not offset the cost of piping

  (although TZM is not suitable for current strategy of low activation, the **conclusion** was shown later to be applicable to all refractory alloys and other expensive structural materials)

- **In 1979, the TRIO experiment was carried out by ANL in the HFIR reactor at ORNL: it demonstrated the feasibility of in-situ tritium release using low pressure helium gas in ceramic breeder**

  (reference : Journal of Nuclear Materials, Volumes 133-134, August 1985, Pages 171-175)

- **Common conclusions in the early studies: The Vacuum Vessel must be outside the Blanket** (one of the major decisions affecting RAMI until now and will have major impact on fusion development). **Batch processing of tritium is not acceptable, In-Situ T recovery necessary**

- Europe and Japan started their own blanket studies. These studies explored a range of options of breeder, multiplier, coolant, structure as well as configurations and coolant flow schemes.

  EU introduced Dual coolant blanket with He-cooled FW/blanket ferritic steel structure, and self-cooled Lead Lithium breeding zone employing alumina coatings as electrical insulation

Evolution of Fusion Blanket Concepts during 40-yr History of Blanket Studies (cont'd)

- By the early 1980's there were ~ 50 blanket concepts proposed!!!!!!

- **BLANKET Comparison and Selection Study (BCSS)** 1983-1985 :

  Major Study initiated by DOE, Led by ANL, involved many national labs, universities, and experts from Aerospace Industry and Nuclear Fission

  The primary objectives were :

  - Define a limited number of Blanket Concepts that that should provide the focus of Blanket R&D

  - Identify and prioritize critical issues for the leading blanket concepts

  The STARFIRE and MARS reactor designs were used as reference designs with a neutron wall loading of $5 MW/m^2$ as nominal reference value

- **BCSS selected:**

Lithium/Lithium/Vanadium Alloy	(self-cooled LM)
Li_2O/Helium/Ferritic Steel	(ceramic Breeder)
LiPb/LiPb/Vanadium Alloy	(self-cooled LM)
Lithium/Helium/Ferritic Steel	(separately cooled LM breeder)

Evolution of Fusion Blanket Concepts during 40-yr History of Blanket Studies (cont'd)

- **EU carried out their own BCSS in 1995.** They evaluated and compared:

  a) Water cooled PbLi blanket
  b) Helium cooled ceramic breeder blanket with breeder in tubes (BIT)
  c) Helium cooled ceramic breeder blanket with breeder out of tube tubes (BOT)
  d) Dual coolant blanket with He-cooled FW/blanket steel structure, and self-cooled Lead Lithium breeding zone employing alumina coatings as electrical insulation

  EU BCSS selected two concepts : Helium-cooled ceramic breeder with FS and Water Cooled PbLi (later changed to He cooled)

- Evolution of blanket concepts continued in Blanket /FNST studies.

- **An innovative Dual Coolant PbLi (DCLL) blanket concept was proposed by Siegfried Malang.** It has helium cooled FW/steel structure and a self-cooled lead lithium breeding zone as his earlier DCLL, but the arrangement of flow channel inserts made of SiC between the flowing LM and the duct walls enables a LM exit temperature about 200 K higher than the maximum ferritic steel temperature (to be described later)

- ARIES power plant studies in the US and EU Power Plant Conceptual Studies (PPCL) explored a number of blanket concepts and evaluated their performance in the fusion power plan environment.
- **The FINESSE Study (1983 – 1986) identified and characterized the issues, requirements for experiments and non fusion and fusion facilities for all FNST component and disciplines.** The study involved experts from the Aerospace and fission industry, experts on technology development , and strong participation by Japan and EU. The study showed that extensive R&D program is required for Blanket/FNST development including new facilities for multiple-effect experiments and dedicated DT fusion facility for integrated testing in the fusion nuclear environment.

Classes of Blanket Concepts
(many concepts proposed worldwide)

A. Solid Breeder Concepts

- Solid Breeder: Lithium Ceramic (Li_2O, Li_4SiO_4, Li_2TiO_3, Li_2ZrO_3)
- Neutron Multiplier: Be or $Be_{12}Ti$
- Coolant: Helium or Water

B. Liquid Breeder Concepts

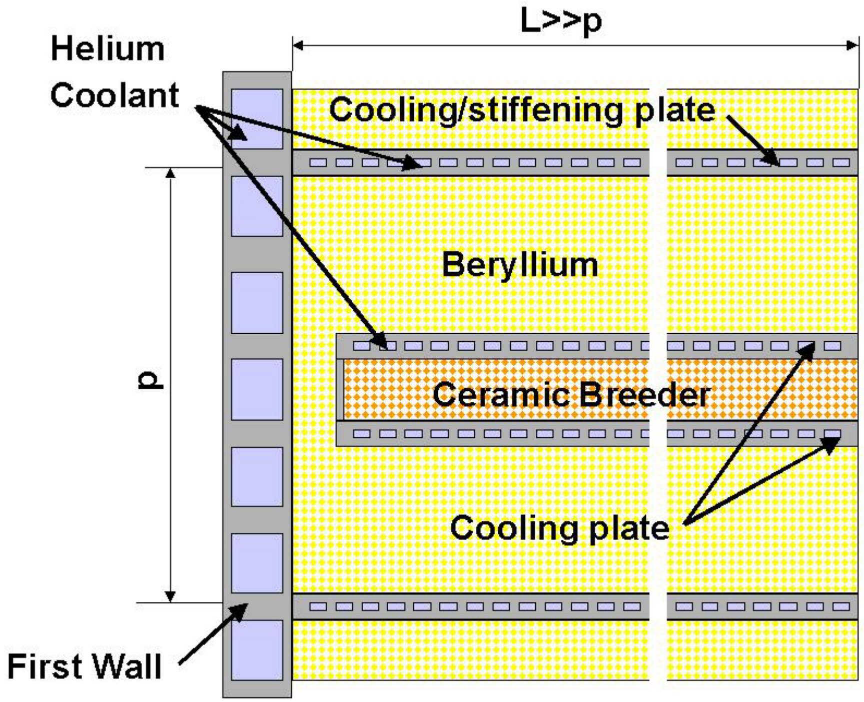
Liquid breeder can be:

a) **Liquid metal** (high electrical/thermal conductivity, low viscosity):
Li, or PbLi

b) **Molten salt** (low electrical/thermal conductivity, high viscosity):
Flibe $(LiF)_n \cdot (BeF_2)$, Flinabe ($LiF\text{-}BeF_2\text{-}NaF$)

A Helium-Cooled Li-Ceramic Breeder Concept : Example

- **High pressure Helium** cooling in structure (ferritic steel)
- **Ceramic breeder** (Li_4SiO_4, Li_2TiO_3, Li_2O, etc.) for tritium breeding
- **Beryllium** (pebble bed) for neutron multiplication
- **In-situ tritium removal* with Helium purge** (low pressure) to remove tritium through the "interconnected porosity" in ceramic breeder

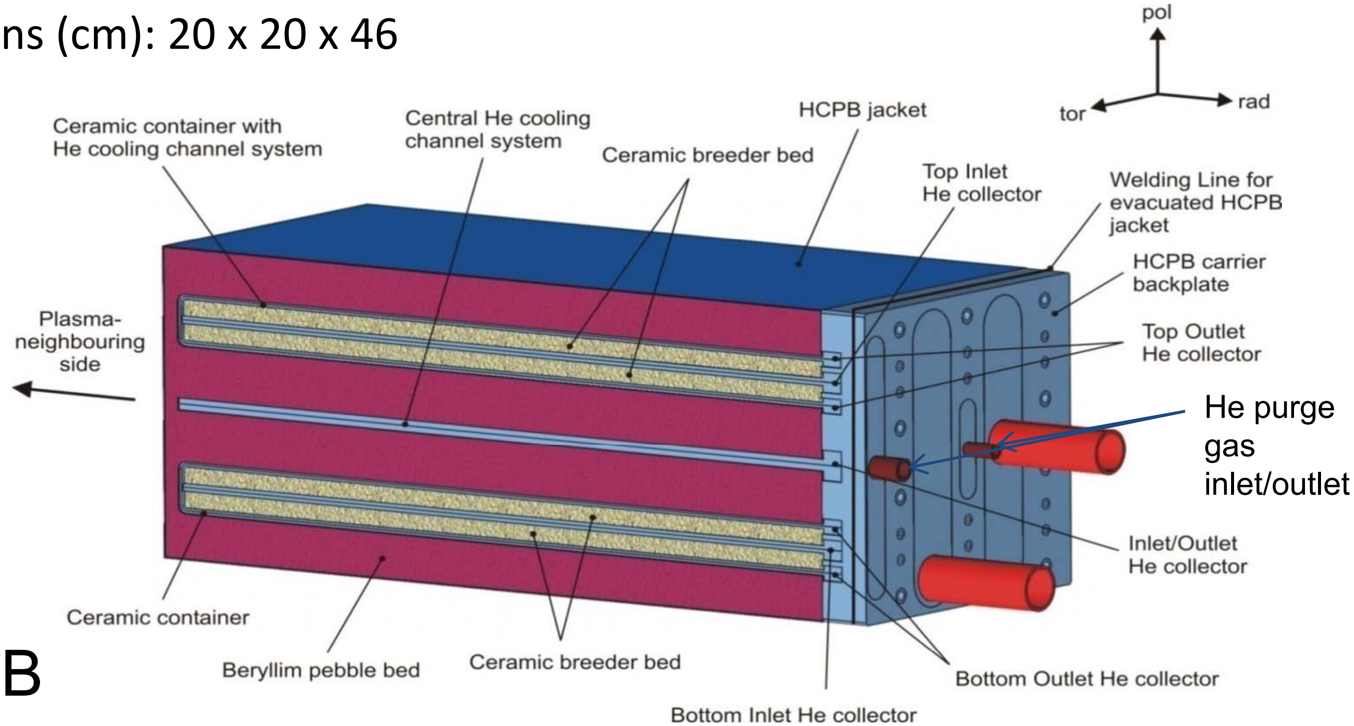


Several configurations exist (e.g. wall parallel or "head on" breeder/Be arrangements)

* "In-situ" is necessary to keep tritium inventory in the system low.
"Batch" processing is not appropriate for fusion (>150 kg/yr 1000MWe fusion power plant).

Breeder Unit for Helium-Cooled Pebble Bed Concept (HCPB)

Dimensions (cm): 20 x 20 x 46



EU-HCPB

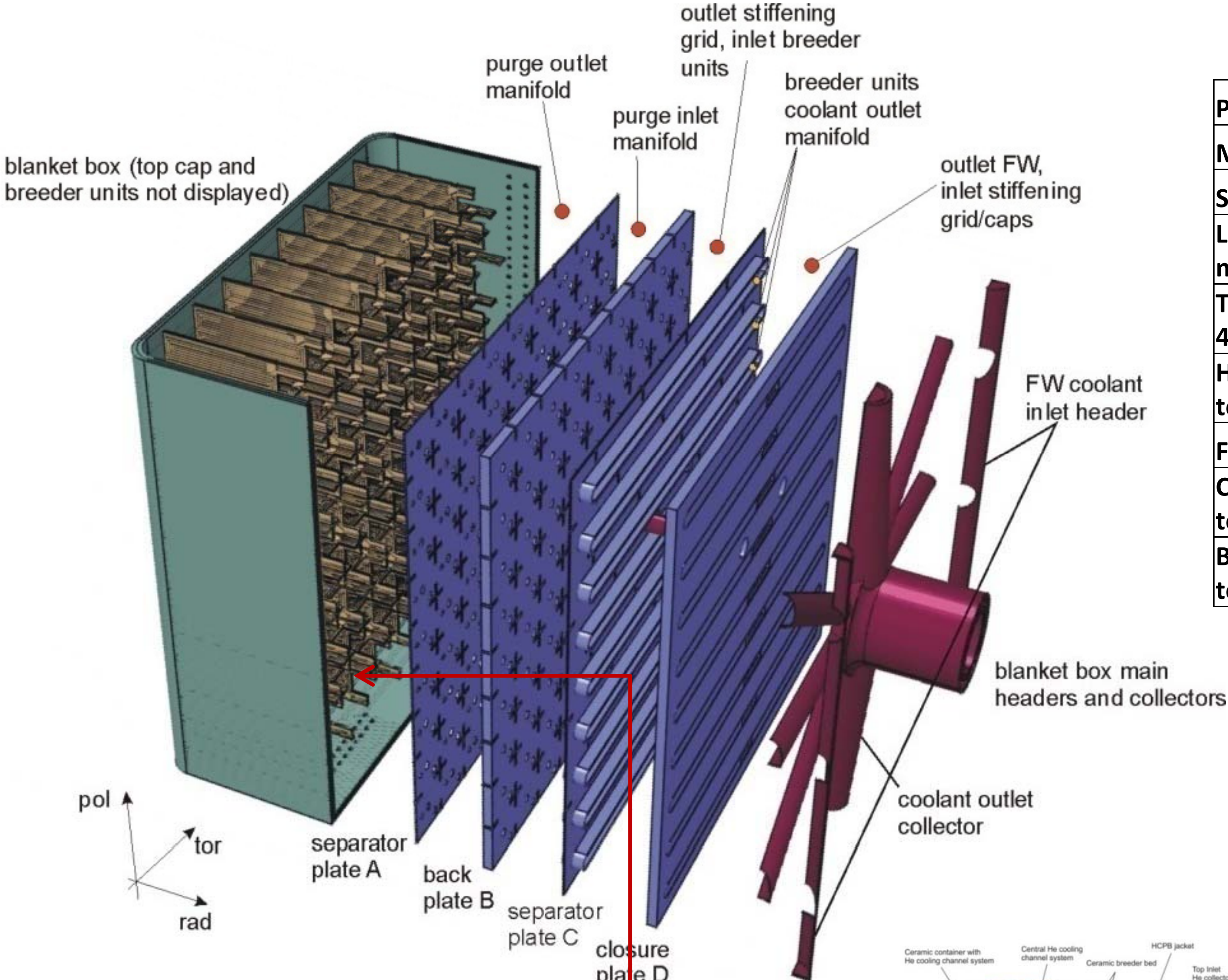
Design* variant	^{6}Li enrichment	TBR	Be mass [tons]	Li_4SiO_4 mass [tons]
Variant A	40%	1.14	412	147
Variant B (less Be)	40% and 60% for rear breeder zone	1.14	284	242

Main blanket features

- Li_4SiO_4 breeder pebbles (0.2-0.6 mm)
- Be pebbles (1 mm)
- He-gas coolant (8MPa)
- He purge (0.1-0.2 MPa) with 0.1% H_2
- Reduced activation ferritic steel

*U. Fischer, et.al., EU Blanket Design Activities and Neutronics Support Efforts, 16th Topical Meeting on the Technology of Fusion Energy, Madison, Wisconsin, USA, Sep. 14-16, 2004

Helium-Cooled Pebble Bed Module Structural Configuration

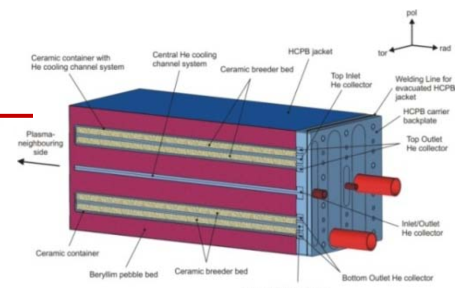


EU HCPB DEMO

Plant fusion power	3300 MW
Mid-plane neutron wall load	2.24 MW/m^2
Surface heat flux	0.5 MW/m^2
Local blanket energy multiplication	1.25
Tritium Breeding Ratio (with 40% ^{6}Li enrichment and 46 cm)	1.14
Helium coolant inlet/outlet temperature	300 - 500^{o}C
FW maximum temperature	550^{o}C
Ceramic breeder pebble bed temperature	400-920^{o}C
Beryllium pebble bed temperature	400-650^{o}C

Tritium Inventory*
Ceramic: earlier estimation gave ~250 g in Li_4SiO_4
Beryllium: Low production of T, but high uncertainties in the effective release rate. It is still an open issue, R&D is ongoing in EU.

Breeder Unit to be inserted into the space between the grid plates

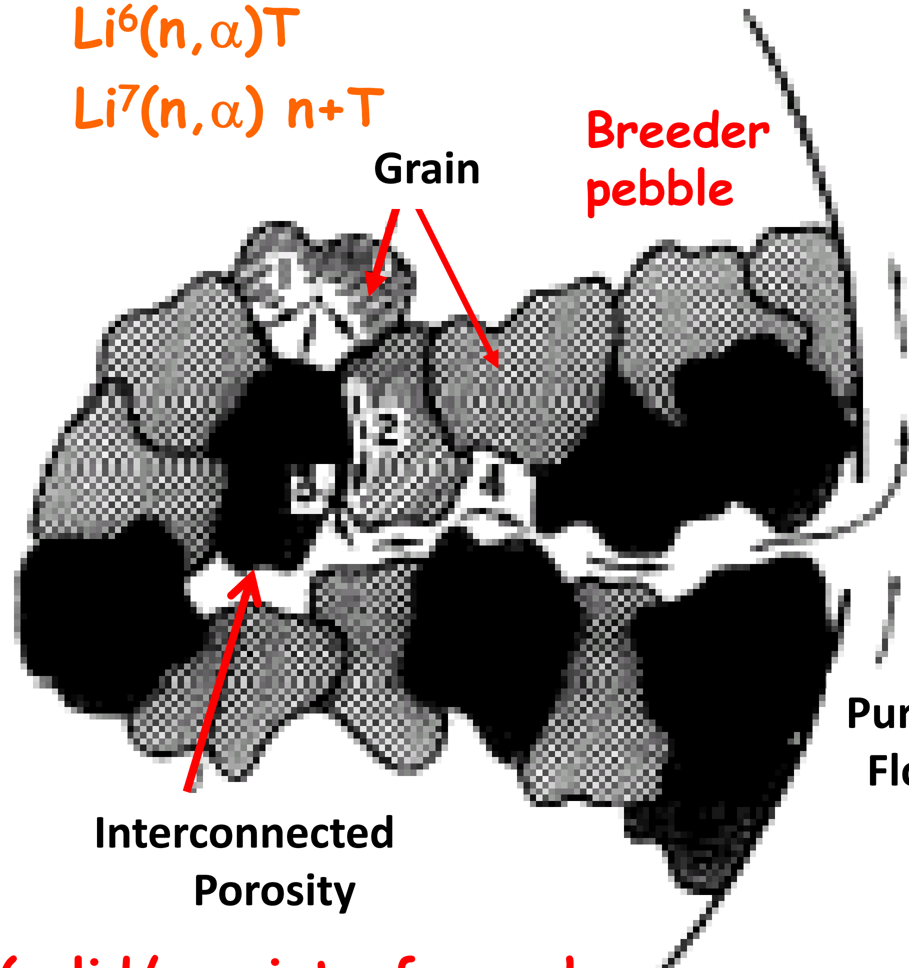


*L.V. Boccaccini, The concept of the breeding blanket for T-self sufficiency, comparison of different schemes, SOFT 25, Sep. 18, 2008

Mechanisms of tritium transport (for solid breeders)

$Li^6(n,\alpha)T$

$Li^7(n,\alpha)\ n+T$



(solid/gas interface where adsorption/desorption occurs)

Mechanisms of tritium transport

1) Intragranular diffusion
2) Grain boundary diffusion
3) Surface Adsorption/desorption
4) Pore diffusion
5) Purge flow convection

Purge gas composition:

$He + 0.1\%\ H_2$

Tritium release composition:

T_2, HT, T_2O, HTO

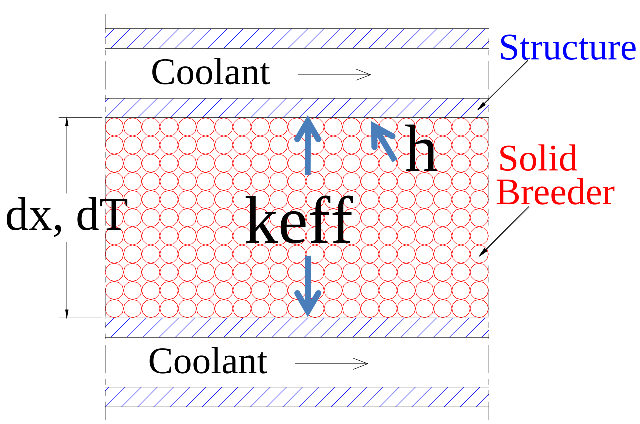
Reference: G. Federici, Ph.D. Dissertation, University of California at Los Angeles (October 1989); UCLA-FNT-30 Report (November 1989).

“Temperature Window” for Solid Breeders

- The operating temperature of the solid breeder is limited to an acceptable “temperature window”: T_{min}– T_{max}
  - T_{min}, lower temperature limit, is based on acceptable tritium transport characteristics (typically bulk diffusion). Tritium diffusion is slow at lower temperatures and leads to unacceptable tritium inventory retained in the solid breeder
  - T_{max}, maximum temperature limit, to avoid sintering (thermal and radiation-induced sintering) which could inhibit tritium release; also to avoid phase change/mass transfer (e.g., LiOT vaporization)
- Low *k (thermal conductivity)*, combined with the allowable operating “temperature window” for solid breeders, results in:
  - Limitations on power density, especially behind first wall and next to the neutron multiplier (limits on wall load and surface heat flux)
  - Limits on achievable tritium breeding ratio (beryllium must always be used; still TBR is limited) because of increase in structure-to-breeder ratio
  - Higher “effective” k is obtainable with a homogenous mixture of ceramic breeder (low k) and Be_{12} Ti (high k)

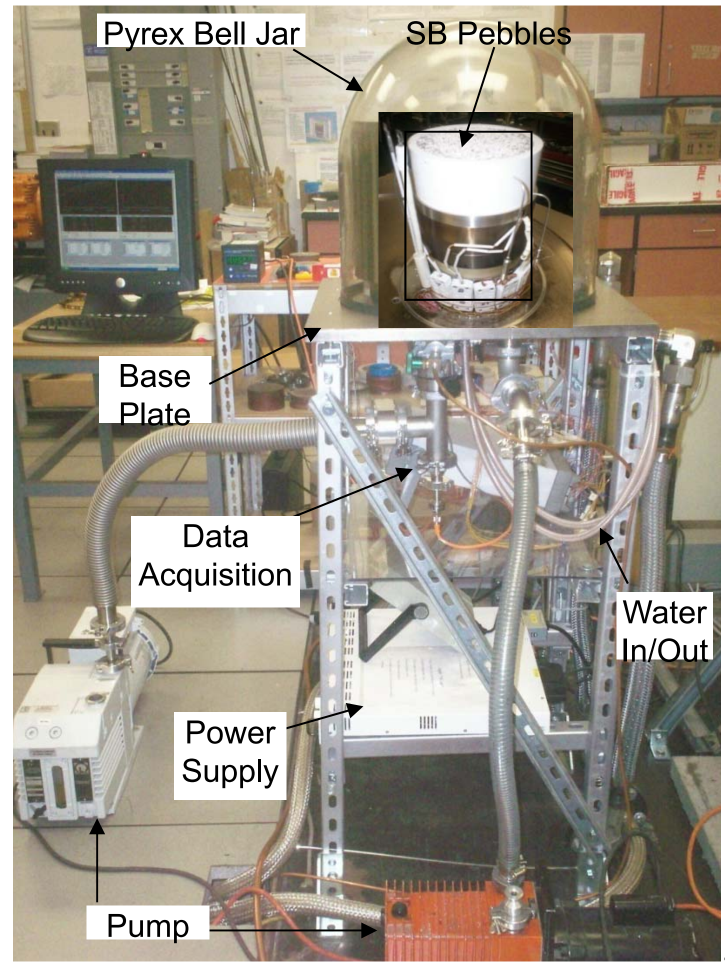
Breeder operating temperature depends on the thermo-physical properties as well as thermo-mechanical state of the pebble bed

- *Effective thermal conductivity (**k_{eff}**)* **& interface conductance (*h*) of ceramic solid breeder pebble beds have significant impact on thermal performance of fusion blanket.**
- **Thermal and irradiation induced stresses (σ) and mechanical constraints imposed by the structural wall can alter packing state (φ) and pebble/pebble and pebble/wall contact characteristics (ε) and thus thermal transport properties.**
- **Most of these properties (mechanical, thermal, etc.) cannot be deduced directly or precisely from the properties of the base single pebble material; therefore, dedicated experiments were performed.**



$$\kappa_{eff} = f\,(k_{CB}, k_{He}, T, \varepsilon/\sigma, \varphi, \text{cyclic})$$

One experimental apparatus at UCLA Fusion Technology Laboratory

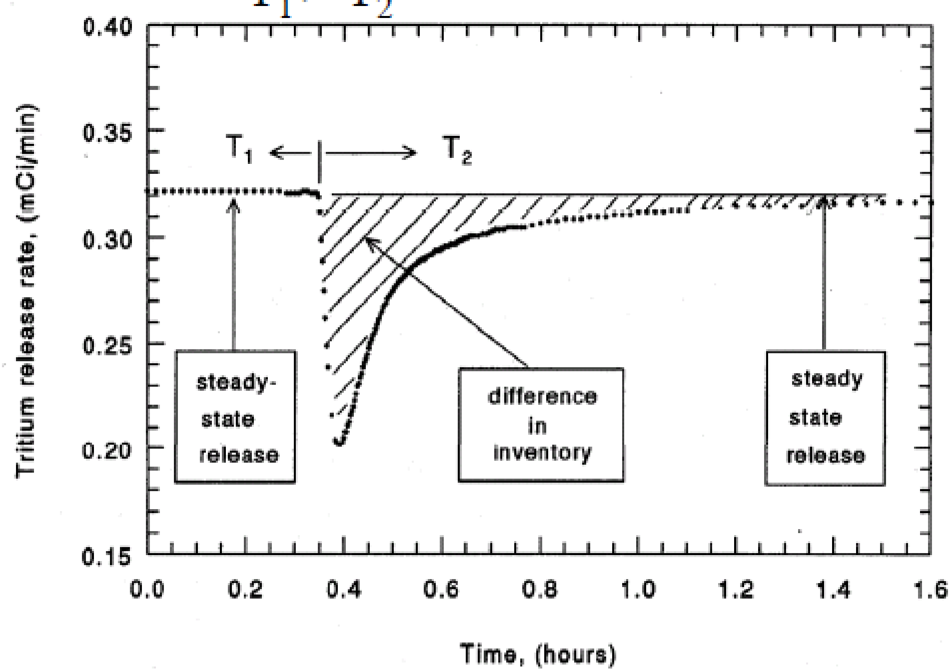


Many irradiation experiments were performed to quantify tritium release characteristics for various ceramic breeders

Recent experiment: EXOTIC 9/1 (EXtraction Of Tritium In Ceramics) in HFR-Petten with in-pile gas purge to quantify tritium release behavior. (The average total ^{6}Li burn-up is 3%. The total measured activity from tritium during irradiation is 220.42 Ci.)

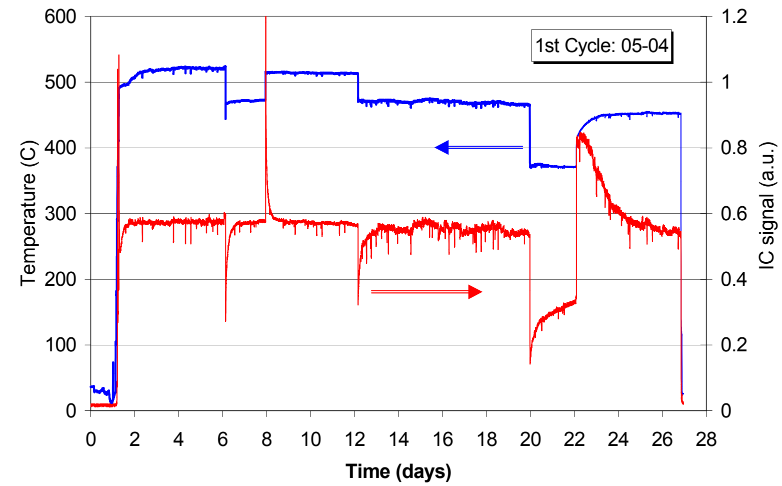
Method (The **temperature step** technique is usually adopted to study in-pile tritium release kinetics)

$T_1 > T_2$



In-pile tritium release data

Temperature varies between 340 and 580 °C

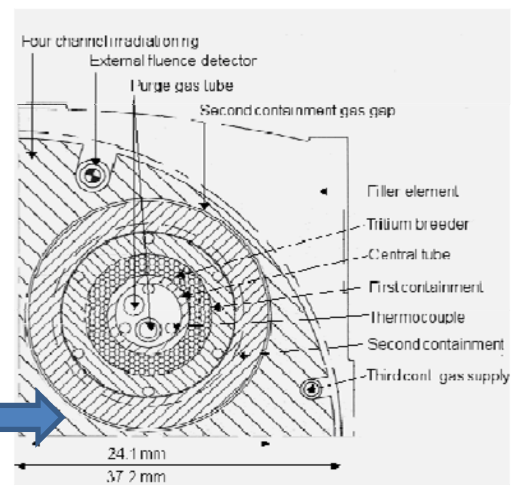


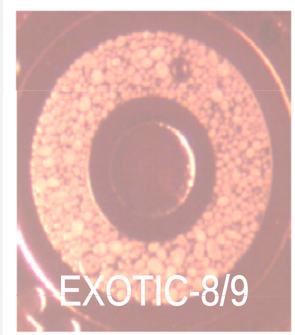
determine Tritium residence (τ):

$$\tau = \frac{I}{G}$$

I = tritium inventory (mCi)
G = tritium production rate (mCi/min)

Annular breeder pebble-bed, modest radial temperature gradient, 120 mm stack height





Representative results on tritium release from ceramic breeder experiments in fission reactors

- **Tritium residence time depends on**
  - Temperature
  - Microstructure/grain size
  - Open and closed porosity
- **Typical range for mean Residence Time:**

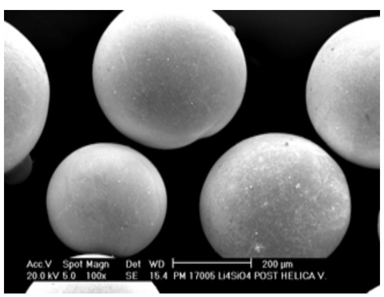
1 – 30 Hours

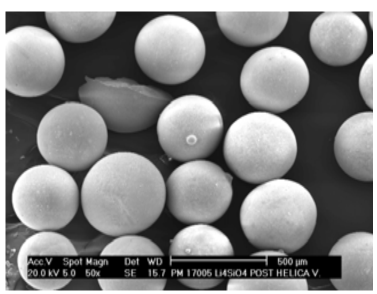
- Hence, Tritium Inventory in Ceramic Breeder Blanket is <1kg
- Radiation-induced sintering would increase T inventory considerably. Hence, the upper operating temperature must be limited to avoid sintering.

Progress on ceramic breeder fabrication R&D in EU and JA

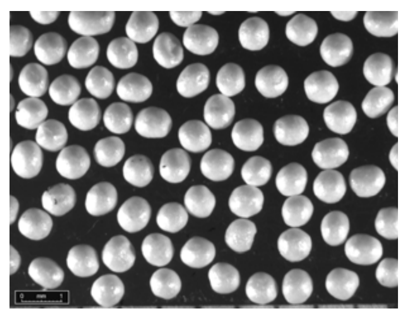
- Pebble selected as the reference material form instead of sintered block
  - a better margin against thermal cracking, easily fit into complex blanket geometries, and better accommodate volumetric swelling and expansion.
- Fabrication routes and quality control steps for ITER materials have been developed
  - Microstructure, phase analysis, density and porosity, specific surface area, mechanic strength (crush load)
  - Examples: FZK Li_4SiO_4 pebbles produced by melt-spraying, CEA Li_2TiO_3 pebbles produced by extrusion-spheronization-sintering, and JAEA Li_2TiO_3 pebbles produced by a wet process.
- Advanced material fabrication R&D are progressing
  - A modified melt-spraying process to reduce micro cracks and pores, and thus enhance mechanical properties of the pebbles as well as the yield of the process
  - Oxide-doped Li_2TiO_3 for better control of grain size and enhancement of chemical stability

0.2- 0.4 mm Li_4SiO_4 pebbles (FZK) melt-spraying fabrication technique



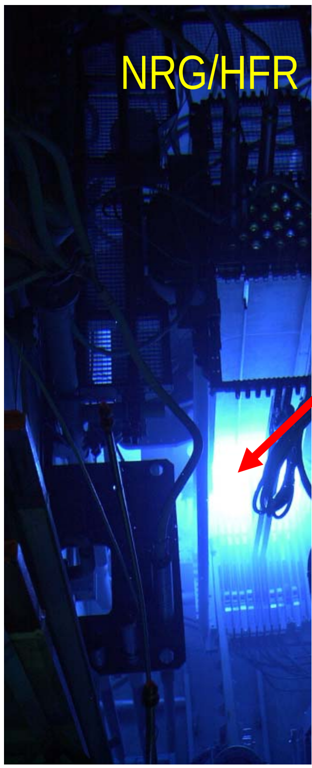


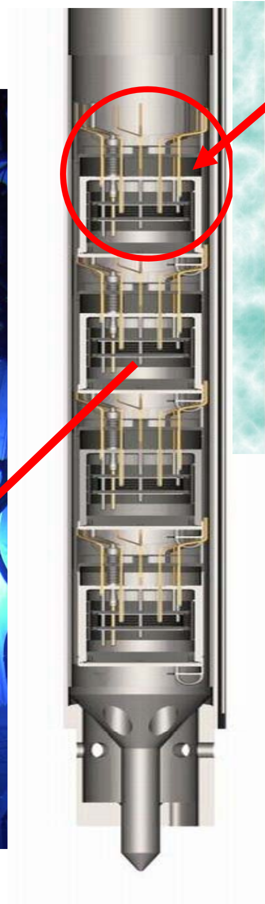
0.6 – 0.8 mm Li_2TiO_3 pebbles (CEA) Extrusion-spheronization-sintering

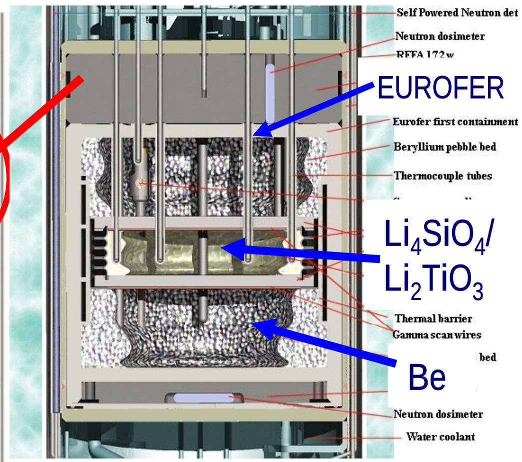


Neutron irradiation experiments were also performed to study thermal-mechanical behavior of EU HCPB unit cell at DEMO relevant temperatures and mechanical constraints

Example: Pebble bed assembly (PBA) test







A HCPB unit cell
6.75 cm D x 12.5 cm H

Total 4 HCPB Unit Cells were tested
2- Li_4SiO_4 beds (650°C and 850°C)
2- Li_2TiO_3 beds (650°C and 850°C)

End of Irradiation of the PBA (ITER testing EOL)

- PBA has been operated in-pile for 12 irradiation cycles, 300 FPD
- Accumulate in 12 cycles, or 7200 hours:
  - **8 x10^{22} at T production**
  - **Lithium burn ups 2 to 3%**
  - **~2 dpa in Eurofer**

- Experimental results with Li_4SiO_4 pebble bed qualitatively benchmarks FEM predicted stress/strain gradients.

Liquid Breeder Blanket Concepts

1. Self-Cooled
   - Liquid breeder circulated at high speed (V ~ 0.5-1 m/s) to serve as coolant
   - Concepts: Li/V, Flibe/advanced ferritic, Flinabe/FS, PbLi/SiC
2. Separately Cooled
   - A separate coolant, typically Helium, is used. The breeder is circulated at low speed (V ~ 1 mm/s) for tritium extraction
   - Concepts: LiPb/He/FS, Li/He/FS
3. Dual Coolant
   - First Wall (highest heat flux region) and structure are cooled with a separate coolant (Helium).
   - The liquid breeder is self-cooled; i.e., in the breeder region, the liquid serves as breeder and coolant (V ~ 5-10 cm/s)
   - Concepts: PbLi/FS, High-Temperature (700°C) and Moderate-Temperature (500°C) options

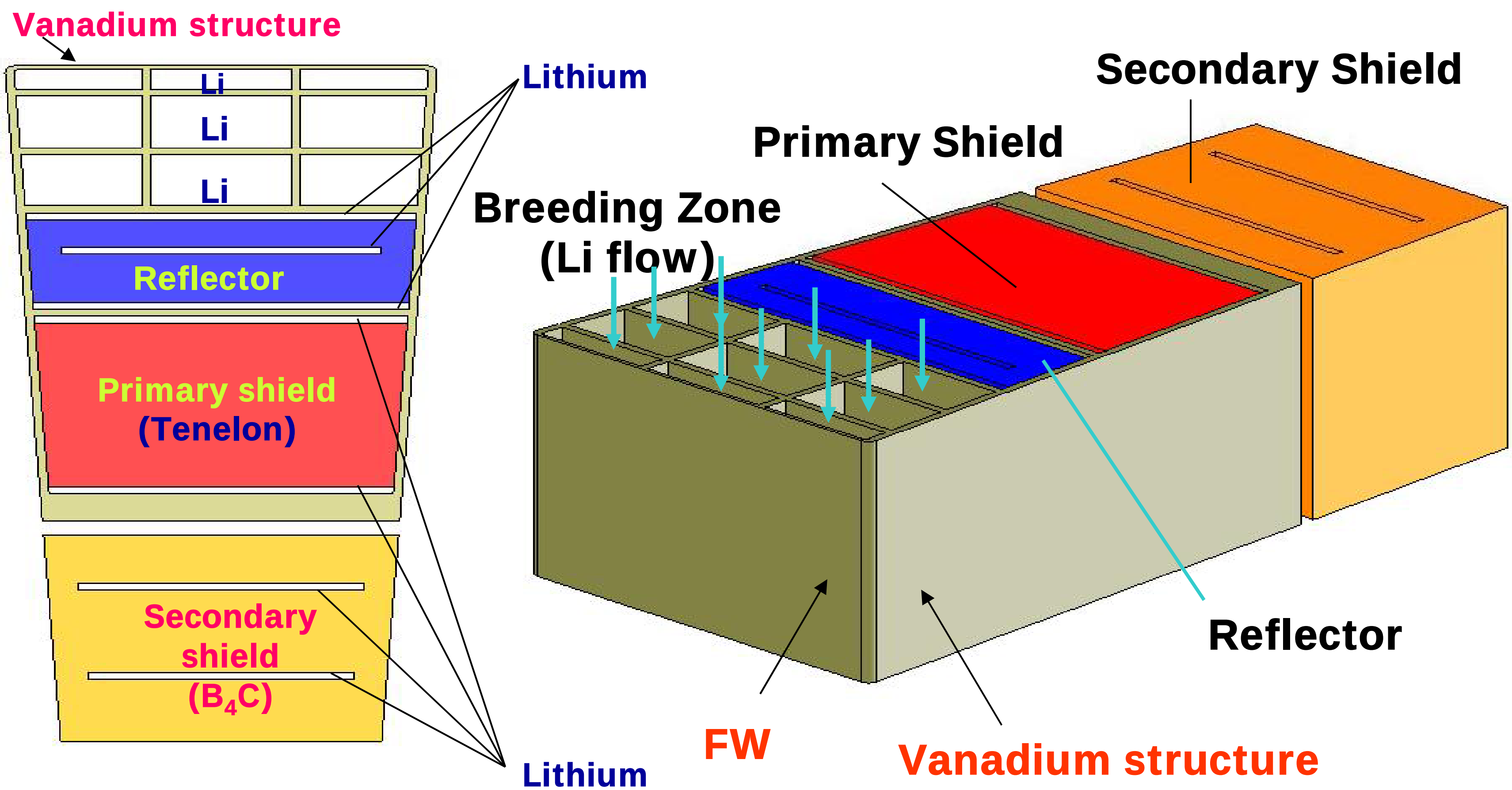
Self-Cooled Lithium/Vanadium Blanket Concept

Motivation

- Simplicity: flowing Li serves as breeder and coolant
- Tritium solubility is high: reduces tritium permeation
- Low activation structural material

Issues

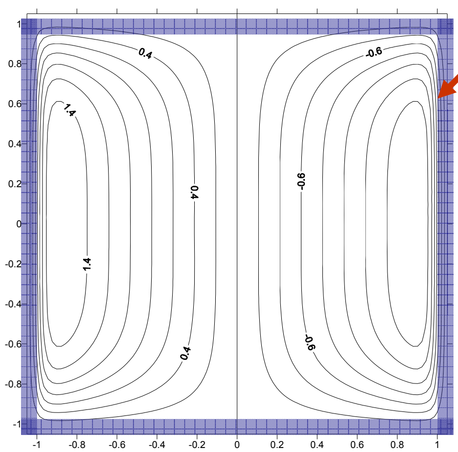
- High reactivity of Li with water and air
- High MHD pressure drop



Self-Cooled Liquid Metal Blankets are NOT feasible now because of MHD Pressure Drop.

A perfectly insulated "WALL" can mitigate the problem, but is it practical?

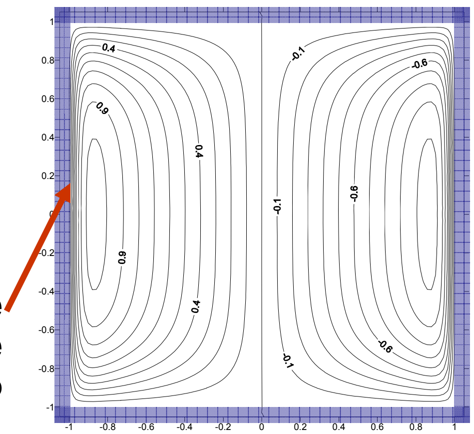
Conducting walls



Lines of current enter the low resistance wall – leads to very high induced current and high pressure drop

All current must close in the liquid near the wall – net force from jxB force is zero

Insulated walls



- Net JxB body force

  $\nabla p = VB^2\, t_w\, \sigma_w / a$

- For high magnetic field and high speed (self-cooled LM concepts in inboard region) the pressure drop is large
- The resulting stresses on the wall exceed the allowable stress for candidate structural materials

- Perfect insulators make the net MHD body force zero
- But insulator coating crack tolerance is very low ($\sim 10^{-7}$).
  - We have not been able to develop practical insulators under fusion environment conditions with large temperature, stress, and radiation gradients
- Self-healing coatings have been proposed but none has yet been found (research is on-going)

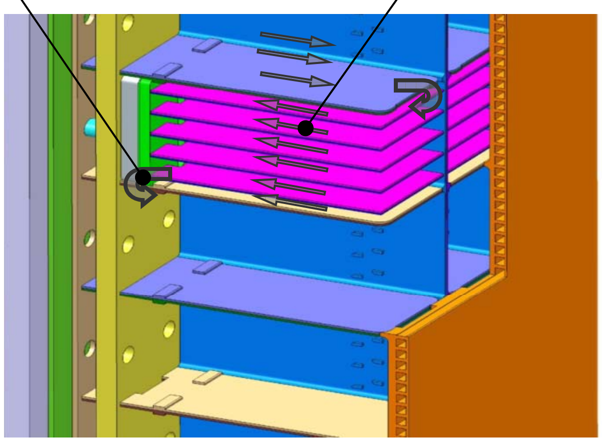
Impact of MHD: No self-cooled blanket option without practical insulators

Separately-cooled LM Blanket

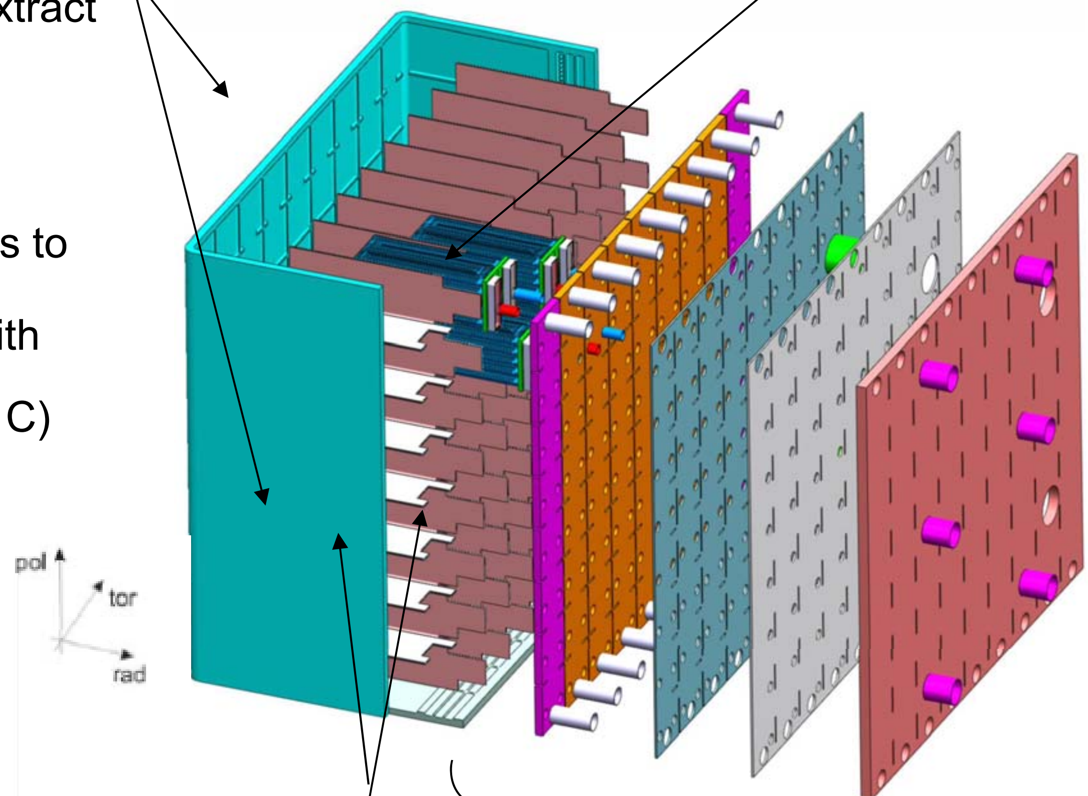
Example: PbLi Breeder / Helium Coolant with RAFM

- **All energy removed by separate Helium coolant**
- ***The idea is to avoid MHD issues***
  But, PbLi must still be circulated to extract tritium

- ISSUES:
  - Low velocity of PbLi leads to high tritium partial pressure, which leads to higher tritium permeation
  - T_{out} limited by PbLi compatibility with RAFM steel structure ~ 470 C (and also by limit on Ferritic, ~550 C)



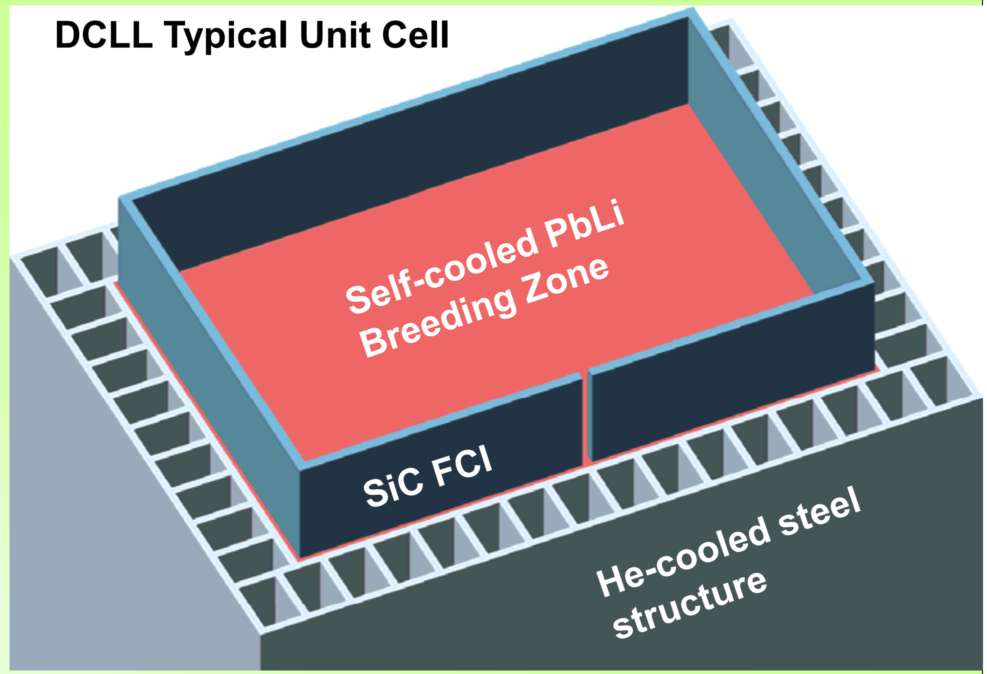
HCLL PbLi flow scheme



Dual Coolant Lead-Lithium (DCLL) FW/Blanket Concept

- **First wall and ferritic steel structure cooled with helium**
- **Breeding zone is self-cooled**
- **Structure and Breeding zone are separated by SiC/SiC composite or foam *flow channel inserts (FCIs)* that:**
  - **Provide thermal insulation to decouple PbLi bulk flow temperature from ferritic steel wall**
  - **Provide electrical insulation to reduce MHD pressure drop in the flowing breeding zone**

FCI does not serve structural function

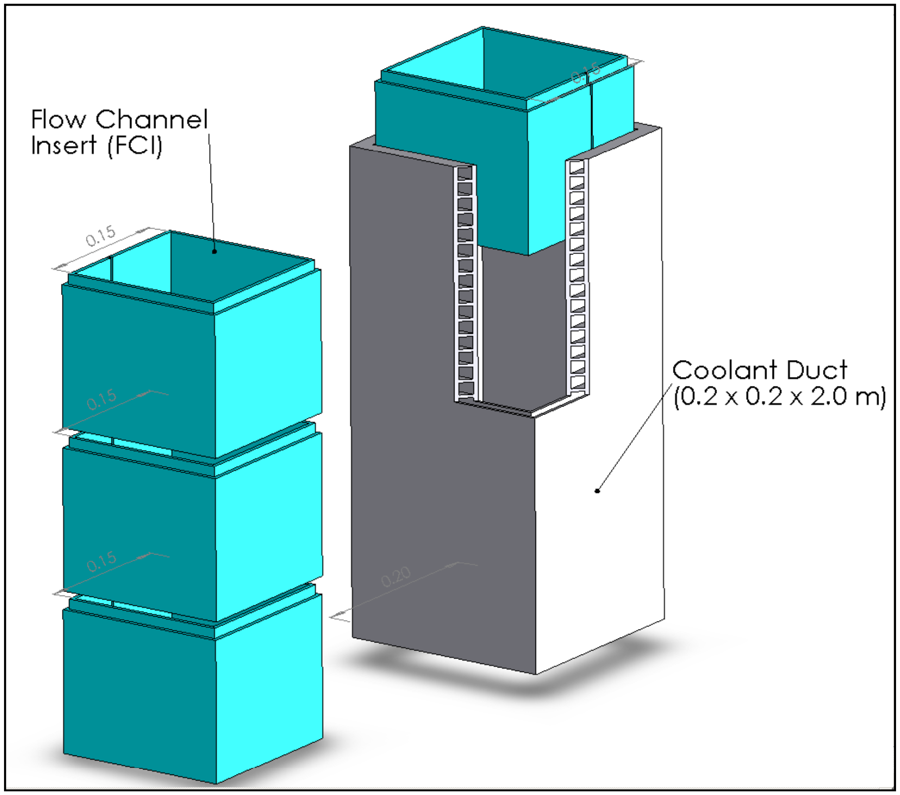


Fusion power: **2200 MW**	PbLi velocity: **10 cm/s**
NWL: **3.7 MW/m^2**	PbLi Tin/Tout: **500°/700°**
HF: **0.5 MW/m^2**	He Tin/Tout: **350°/450°**
Mass flow rate per duct/module: **40/160 kg/s**	Thermal efficiency: **45%**

PbLi exit temperature can be significantly higher than the operating temperature of the steel structure ⇒ High Efficiency

FCI is a critical element of the high outlet temperature DCLL blanket

- **FCIs are roughly box channel shapes made from either SiC/SiC composites or SiC foams, having low electrical (1-50 S/m) and thermal (1-3 W/m-K) conductivity**
- **They will *slip* inside the He Cooled RAFS structure, but not be rigidly attached**
- **They will slip fit over each other, but not be sealed**
- **FCIs may have a thin slot or holes in one wall to allow better pressure equalization between the PbLi in the bulk flow and in the gap region**
- **FCIs in front channels, back channels, and access pipes will be subjected to different thermal and pressure conditions; and will have different designs / material tailoring**



R&D ISSUES of PbLi BLANKETS

- MHD pressure drop and flow distribution / balancing
- T permeation
- SiC FCI related issues (e.g., insulation, thermal stress, degradation of thermophysical properties under neutron irradiation)
- Compatibility between PbLi and structural and functional materials in the presence of a strong magnetic field
- Limits on operating temperature, re-deposition of radioactive corrosion products in the transport/HX system; clogging of the LM tract with corrosion products

Progress in MHD Experiment

- Many MHD studies have been performed for the last several decades (FZK, UCLA, ANL, Riga, Grenoble)
- R&D Scope: fundamental MHD, MHD pressure drop, insulating techniques, flow distribution, coupling between MHD and heat transfer, mock up testing
- The obtained results can be used for predicting many MHD flows under blanket-relevant conditions and also for benchmarking existing and new CFD codes
- What is still missing – coupling between MHD and heat & mass transfer processes

UCLA MTOR facilities for studying blanket-relevant MHD flows

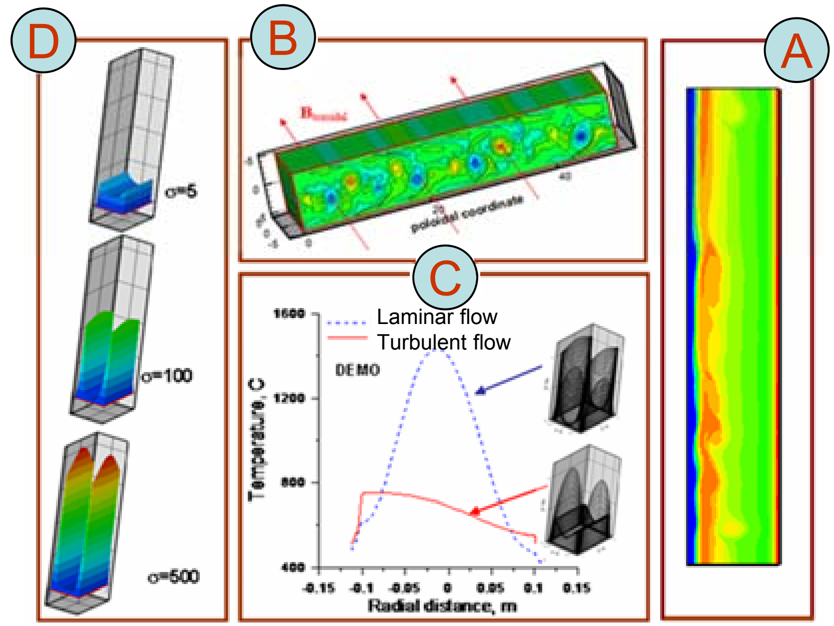
Magnet: 2.0 T pulsed, 1.7 T steady state
Liquid metals: Hg, InGaSn, PbLi

Impressive Progress on MHD Fluid Flow

- Much better understanding and advances of phenomenological models for LM fluid flow in the fusion environment with magnetic field and nuclear heating.
- Major progress in developing computer codes for MHD fluid flow
  - 2-D codes Ha ~ 10^4 capability
  - **3-D codes for complex geometry:**

    Ha ~ 10^3

    (compared to Ha ~ 8 in 1988)
- Progress on MHD experiment: Good, but limited by relatively poor capabilities of existing facilities

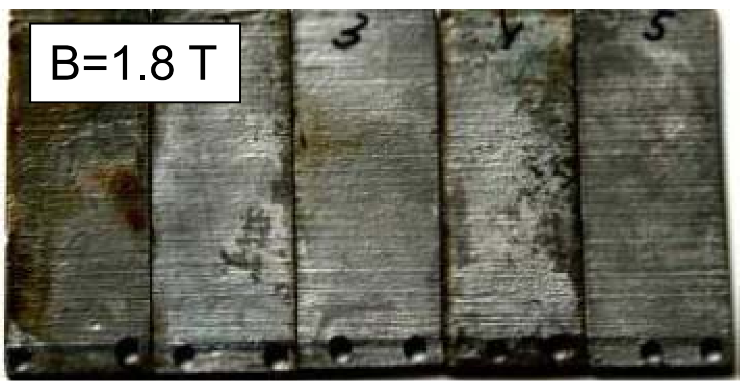


A. Buoyancy forces associated with neutron heating cause intensive thermal convection.
B. MHD turbulence in blanket flows takes a special quasi-two-dimensional form.
C. Strong effect of turbulence on temperature in liquid and solid.
D. Typical MHD effect is formation of special "M-type" velocity profiles.

But, inadequate progress on modelling and experiments for mass transfer and the entire area of interfacial phenomena (fluid-material interactions)

Experiments in Riga (funded by Euratom)
Show Strong Effect of the Magnetic Field on Corrosion
(Results for Ferritic Steel in PbLi)

Macrostructure of the washed samples after contact with the PbLi flow



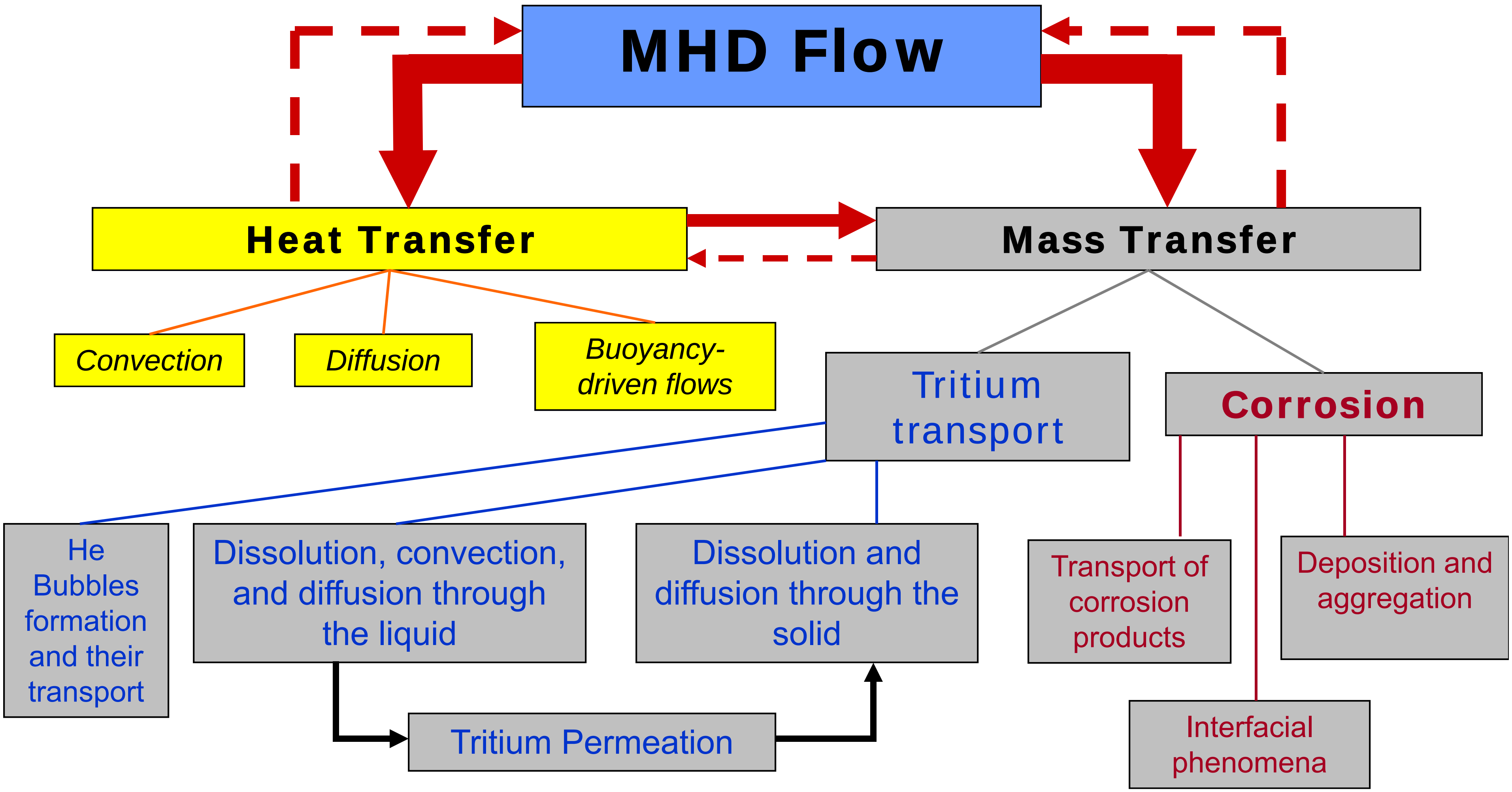
From: F. Muktepavela et al. *EXPERIMENTAL STUDIES OF THE STRONG MAGNETIC FIELD ACTION ON THE CORROSION OF RAFM STEELS IN Pb17Li MELT FLOWS*, PAMIR 7, 2008

Corrosion rate h_n for samples without and with magnetic field

n	h_n , μm/year	
	$B_o = 0$	$B_o = 1.8$ T
1	523	967
2	458	877
3	381	694
4	293	846
5	388	726

Strong experimental evidence of significant effect of the applied magnetic field on corrosion rate. The underlying physical mechanism has not been fully understood yet.

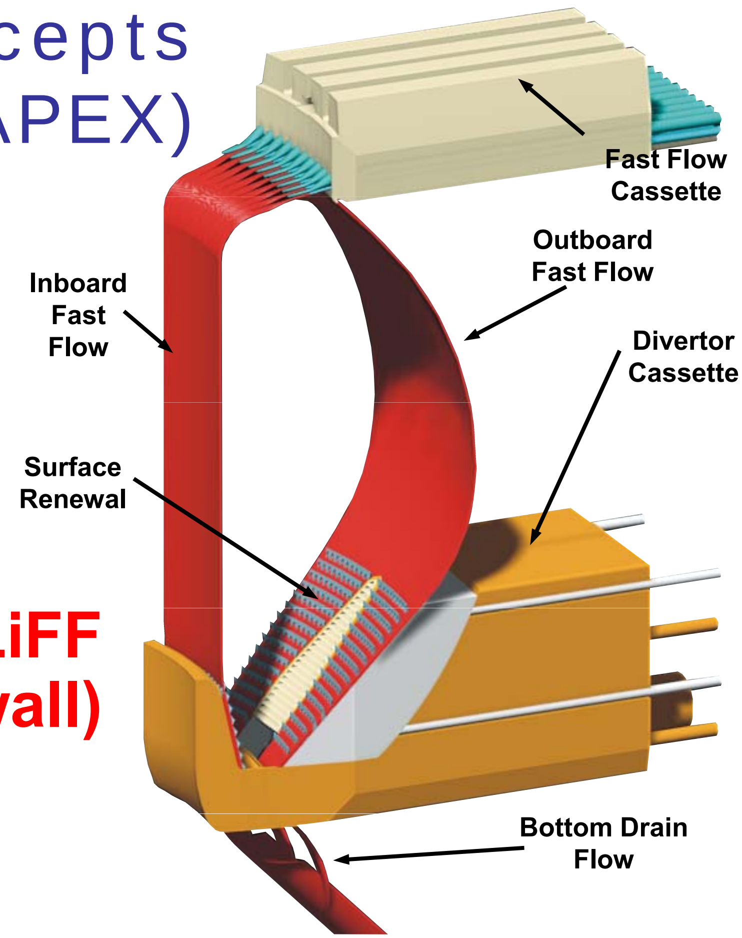
Current and future R&D for liquid metals address momentum, heat, and mass transfer in an integrated approach



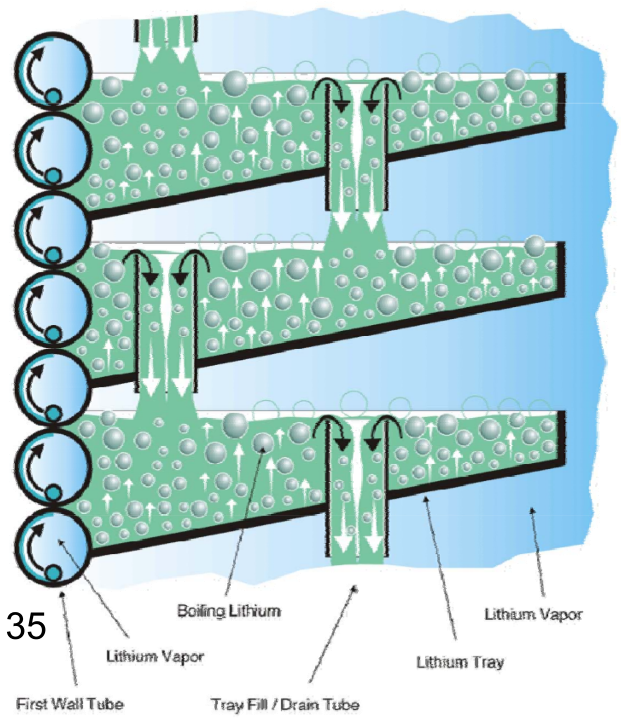
Coupling through the source / sink term, boundary conditions, and transport coefficients

Other innovative blanket concepts have been studied recently (APEX)

- Liquid wall concepts
  - Improve disruption survivability and heat removal
  - Increase TBR
  - potentially improve plasma performance



CLiFF (Thin liquid wall)



35

EVOLVE (2 phase lithium/W)

- High temperature solid wall concept with Li phase change
  - Increase power density and efficiency
  - Favorable TBR

Blankets for IFE

<u>Covered in W. Meier presentation</u>

- Geometry not constrained by burn physics/magnets
- No magnetic field; hence no MHD effects. Self-cooled LM blankets (as well as molten salts) are viable
- **Higher surface heat and particle flux on the first wall, but proposed liquid surfaces mitigate the problem (similar concepts have been proposed for MFE)**
- **No transient heat fluxes comparable to those in tokamaks such as ELMS and disruptions. Hence, there seems to be no large uncertainties associated with first wall thickness (as now exists for tokamaks).**
- **Chamber and driver separated – seem to lead to easier maintenance**
- **Cyclic operation may negatively impact failure modes and rates in First Wall and Blanket of IFE**

Tritium Issues

1. Available External Tritium Supply
2. Tritium Fuel Cycle and Tritium Self-Sufficiency
   a. Conditions for attaining tritium self sufficiency
   b. Achievable TBR (M. Sawan)
   c. Dynamic modeling of the tritium fuel cycle
   d. Tritium Burn-up Fraction (in the plasma) and fueling efficiency
   e. Tritium Inventories and Start-up requirements
   f. Required TBR
   g. Phase Space of Physics and Technology Parameters and Conditions
   h. How, where, and when tritium self sufficiency can be demonstrated
3. Tritium Extraction and Processing (S. Willms)
4. Tritium Permeation and Control (B. Merrill)

Tritium Consumption and Production

Physical constants

- Half life of tritium: 12.32 years
- Mean life of tritium: 17.77 years
- Tritium decay rate: 5.47 %/yr

Tritium Consumption in Fusion Systems

55.8 kg per 1000 MW fusion power per year

For 3000 MW Fusion Power Plant (~1000 MWe)

167.4 kg/year

0.459 kg/day 0.019 kg/hour

Production and Cost in Fission Reactors

Fission Reactor (with special designs for T production): **~0.5-1 kg/year**

$84M-$130M/kg (per DOE Inspector General*)

**www.ig.energy.gov/documents/CalendarYear2003/ig-0632.pdf*

CANDU Reactors: 27 kg from over 40 years,
$30M/kg (current)

Successful ITER will consume almost all externally available tritium supply from CANDUs

Tritium self-sufficiency condition:
$TBR_a \geq TBR_r$

TBR_a = Achievable tritium breeding ratio

TBR_a is a function of **technology, material** and **physics**.

TBR_r = Required tritium breeding ratio

TBR_r *should exceed unity by a margin required to:*

1) *Compensate for losses and radioactive decay (5.47% per year) of tritium between production and use*
2) *Supply tritium inventory for start-up of other reactors (for a specified doubling time)*
3) *Provide a "reserve" storage inventory necessary for continued reactor operation under certain conditions (e.g. a failure in a tritium processing line)*

TBR_r depends on many system physics and technology parameters.

Achievable TBR_a Summary (Details in M. Sawan presentation)

- Achievable TBR is a function of technology, materials, and physics choices, parameters, and conditions
- The largest uncertainties in achievable TBR are due to shortcomings in design definition associated with uncertainties in what is achievable in plasma physics and technological components
- **Present blanket designs in conceptual tokamak power plant studies have calculated TBR values <1.15**

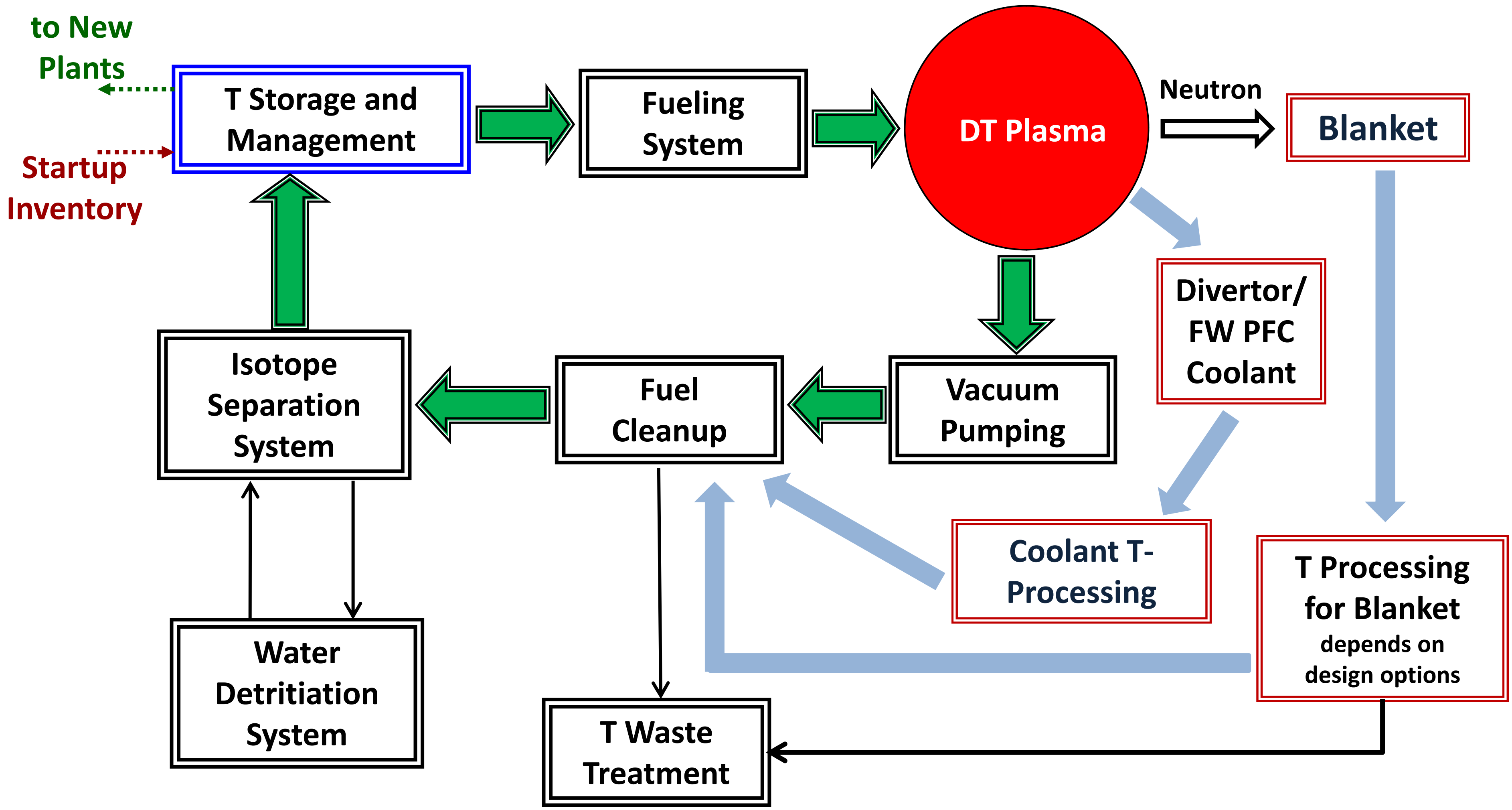
Accounting for Uncertainties

- **At present there are uncertainties in predicting the Achievable TBR and the Required TBR. (Both are currently based on calculations and modelling, not measured in experiments)**
- **A thorough statistical treatment of uncertainties in tritium fuel self sufficiency is a complex area (See Ref.1 at the end of this section)**
- **At this early stage of fusion development, we propose that fusion physics and technology R & D should have the following guideline: Estimated Achievable TBR should exceed the estimated Required TBR by a margin, Δ. Current estimates suggest Δ of ~10%**

Dynamic fuel cycle models were developed to calculate time-dependent tritium flow rates and inventories and required TBR

(Dynamic Fuel Cycle Modelling: Abdou/Kuan et al. 1986, 1999; **See Refs 1-3**)

Simplified Schematic of Fuel Cycle



Key Parameters Affecting Tritium Inventories, and Hence, Required TBR

1) Tritium burn-up fraction in the plasma (f_b)
2) Fueling efficiency (η_f)
3) Time(s) required for tritium processing of various tritium-containing streams (e.g. plasma exhaust, tritium-extraction fluids from the blanket), t_{tp}
4) "Reserve Time", i.e. period of tritium supply kept in "reserve" storage to keep plasma and plant operational in case of any malfunction in a part (q) of any tritium processing system
5) Parameters and conditions that lead to significant "trapped" inventories in reactor components (e.g. in divertor, FW, blanket)
6) Inefficiencies (fraction of T not usefully recoverable) in various tritium processing schemes, ε
7) Doubling time for fusion power plants (time to accumulate surplus tritium inventory sufficient to start another power plant)

Tritium Burn-up Fraction (f_b)

f_b = fusion reaction rate / tritium fueling rate

tritium injection rate = $\dfrac{\text{fueling rate}}{\text{fueling efficiency } (\eta_f)} = \dfrac{\text{fusion reaction rate}}{f_b \eta_f}$

η_f = fueling efficiency = fraction of injected fuel that enters and penetrates the plasma

Need to minimize tritium injection rate: Need high η_f and high f_b

- An expression for f_b can be derived as $\boxed{f_b = 1 / \left(1 + \dfrac{2}{n \tau^* < \sigma v >}\right)}$

$\tau^* = \tau / (1 - R)$ where R = recycling coefficient from the edge (that penetrates the plasma)

τ = particle confinement time

Status

η_f: Recent results: gas fueling is not efficient (η_f ~5%).

Pellet fueling: η_f ~90% on high-field side, 50% for low-field side injection.

Results on ineffectiveness of gas fueling imply significantly smaller R, and hence lower f_b.

$f_b \eta_f$: ITER: ~0.3% selected to size the tritium system; 0.5% expected

Reactors: depends on assumptions on R (subject of current research)

Impact of Tritium Burn-up Fraction and Tritium Processing Time on Tritium Inventory

$$I = I_{fe} + I_c$$

$I_{fe} \equiv$ Tritium inventory in systems associated with the plasma (fueling, exhaust, etc.)

$$I_{fe} = f\left(t_p / f_b \eta_f\right)$$

$\boldsymbol{t_p}$ is the time for tritium processing (to go through the vacuum pumping, impurity separation, ISS, fuel fabrication and injection). Function of technology, design/cost trade-off

$I_c =$ Tritium inventory in other components, e.g. blanket, PFC

Fusion Program is aiming at minimizing tritium inventories

Why large tritium inventory is unacceptable

- Safety
- "Start-up" inventory becomes large (not available from external sources)
- Required tritium breeding ratio becomes much higher

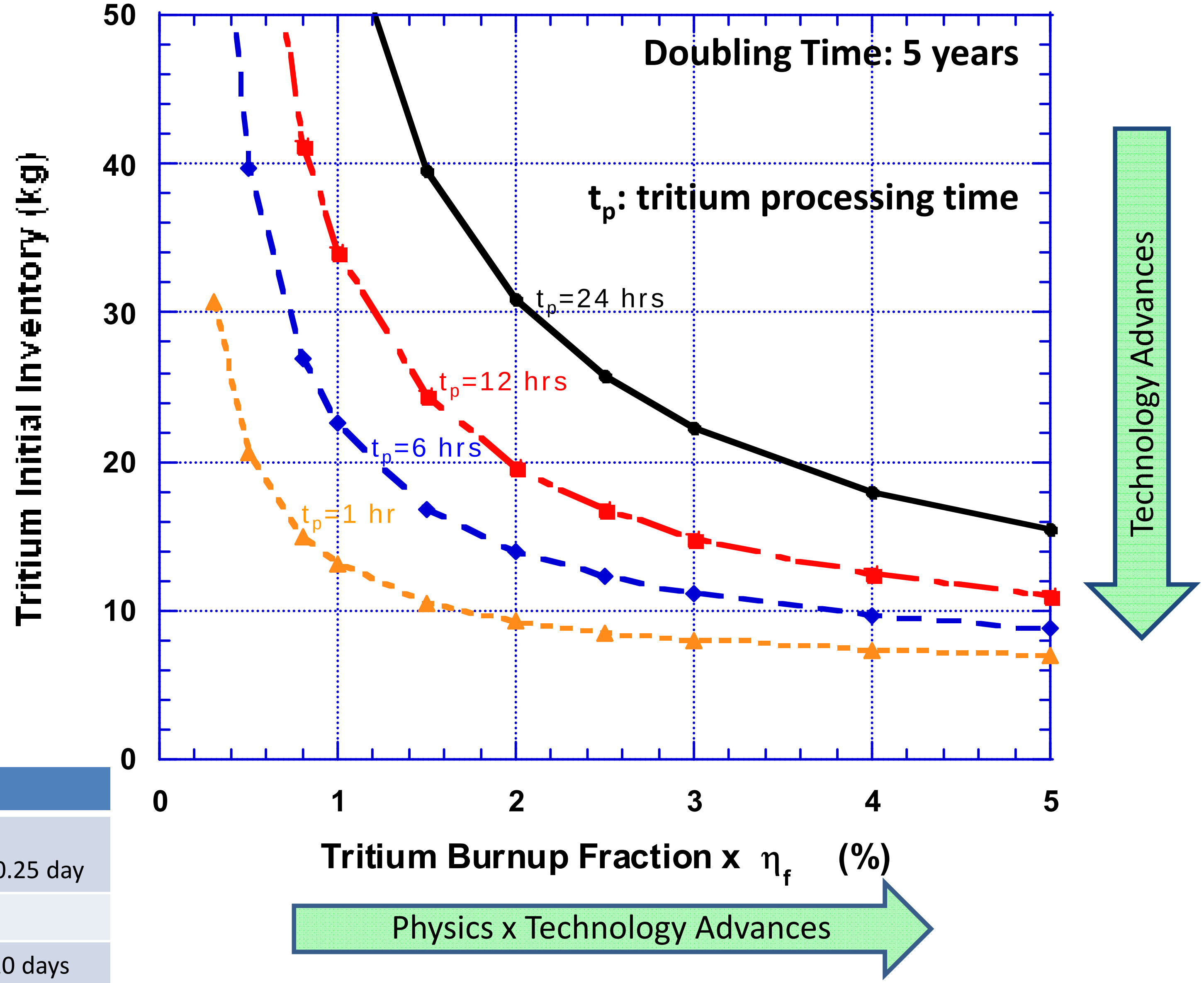
Status on Tritium Processing Time, t_p

1970's-80's Reactor Designs (ANL, FED, etc) : 24 hours ; 1986 TSTA demonstrated < 24 hours

2010 ITER Goal is t_p ~ 1 hour

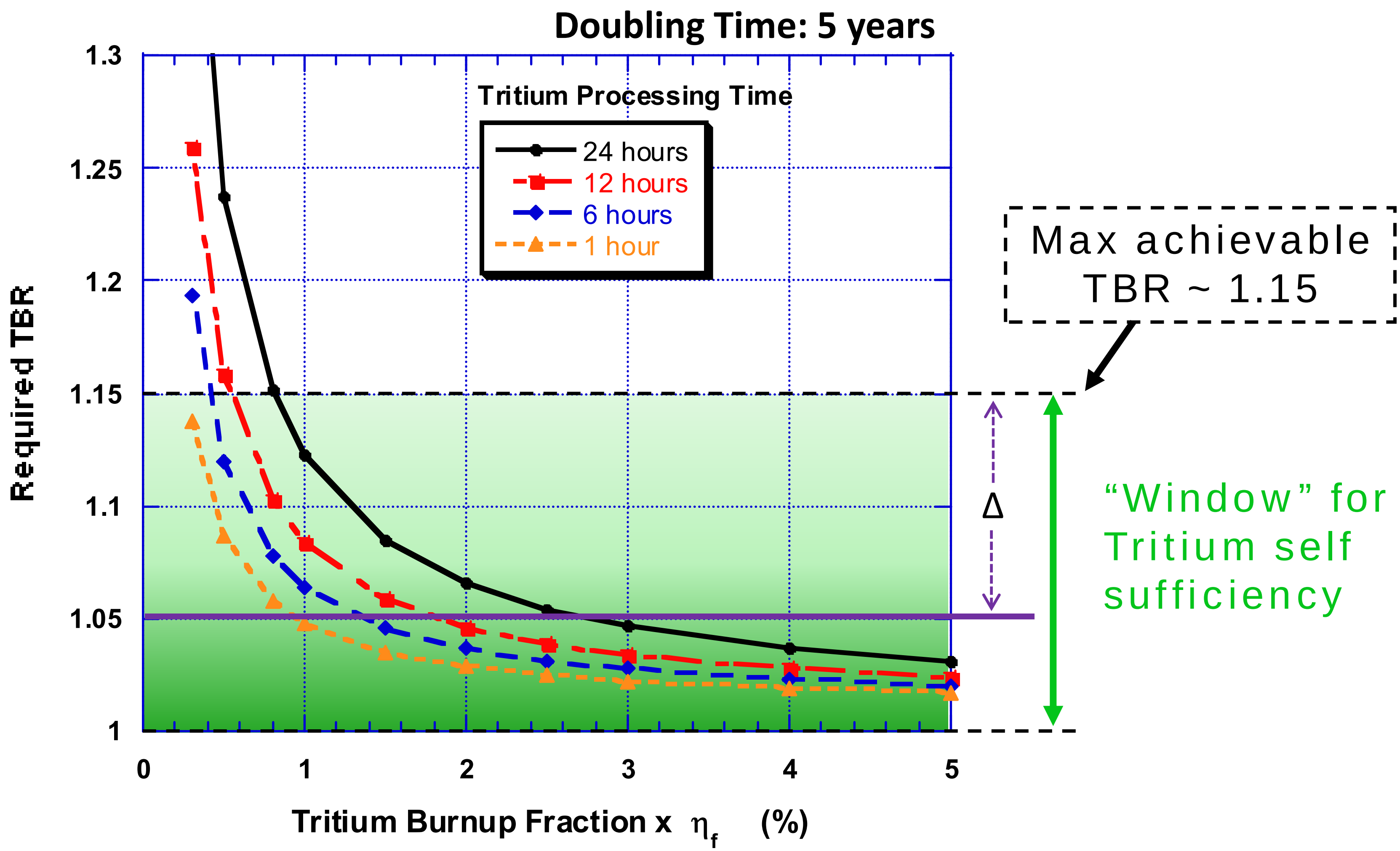
Reactor: no reliable estimate yet, probably somewhere between ITER and TSTA

Tritium inventories depend strongly on tritium burnup fraction (f_b), tritium fueling efficiency (η_f), and tritium processing time (t_p)



- Fusion Power = 3000 MW
- Reserve time for outage x fraction of tritium plant failing= 0.25 day
- Inefficiency, ε = 0.01%
- Blanket mean residence time = 10 days

Variation of Required TBR with $f_b \times \eta_f$, and t_p

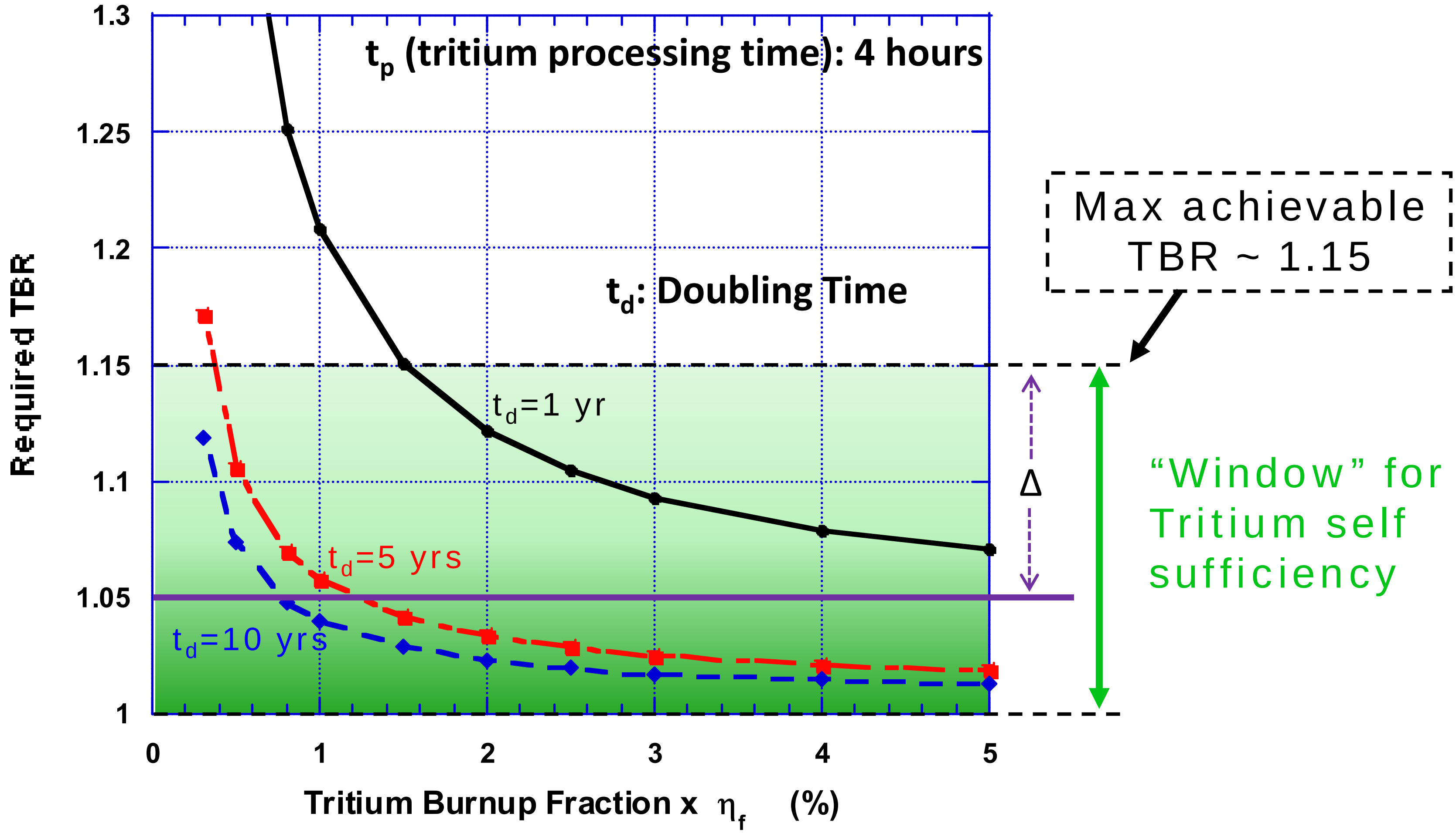


Attaining Tritium Self Sufficiency in DT Fusion Imposes Key Requirements on Physics and Technology. **The goal for R & D should be to achieve:**

T burnup fraction (f_b) x fueling efficiency (η_f) > 5% (not less than 2%)

T processing time (in Plasma exhaust/fueling cycle) < 6 hours

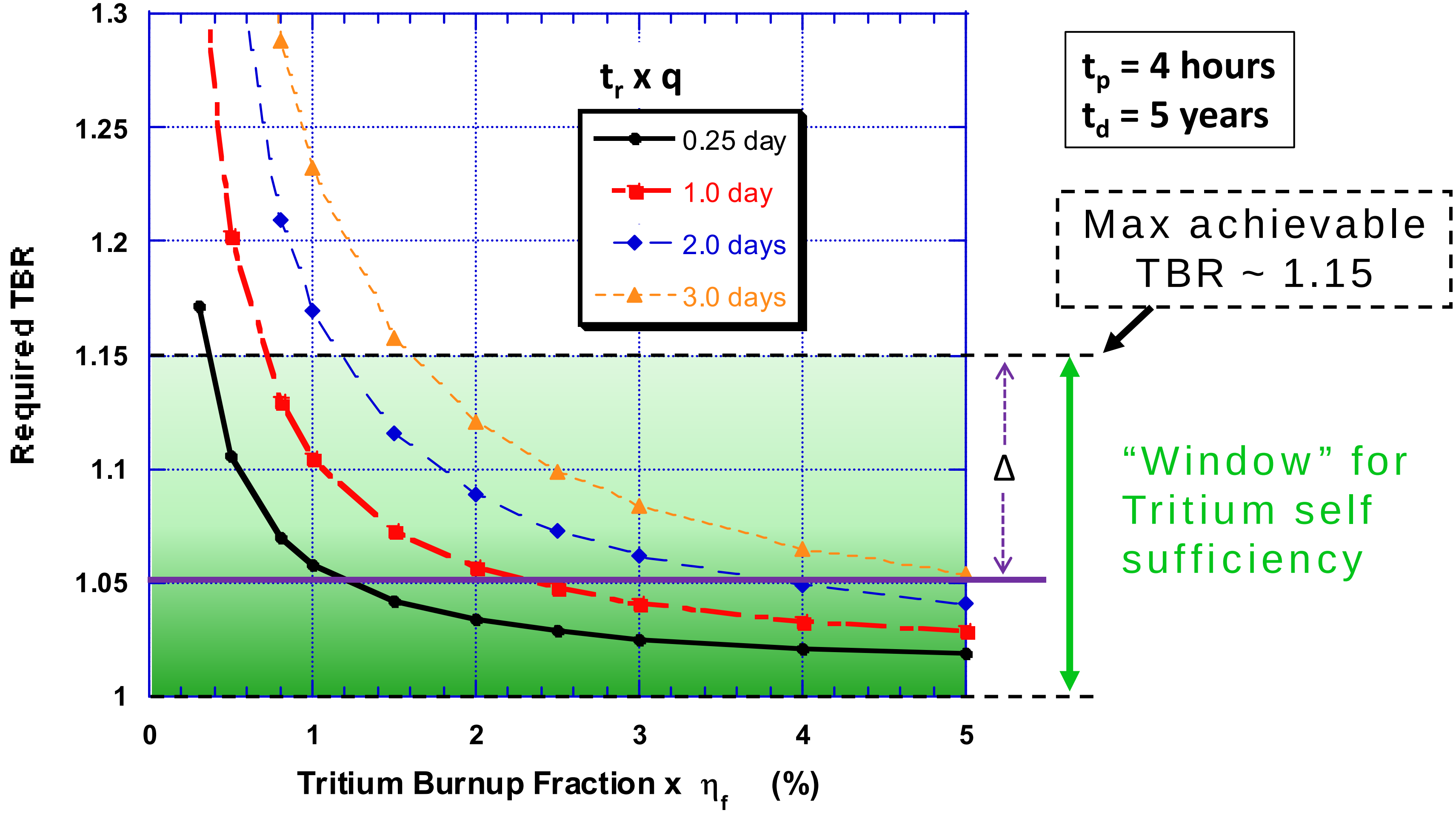
Variation of Required TBR with $f_b \times \eta_f$ and doubling time for short tritium processing time



A “reserve” storage tritium inventory is necessary for continued reactor operation under certain conditions, e.g. failure of a tritium processing line

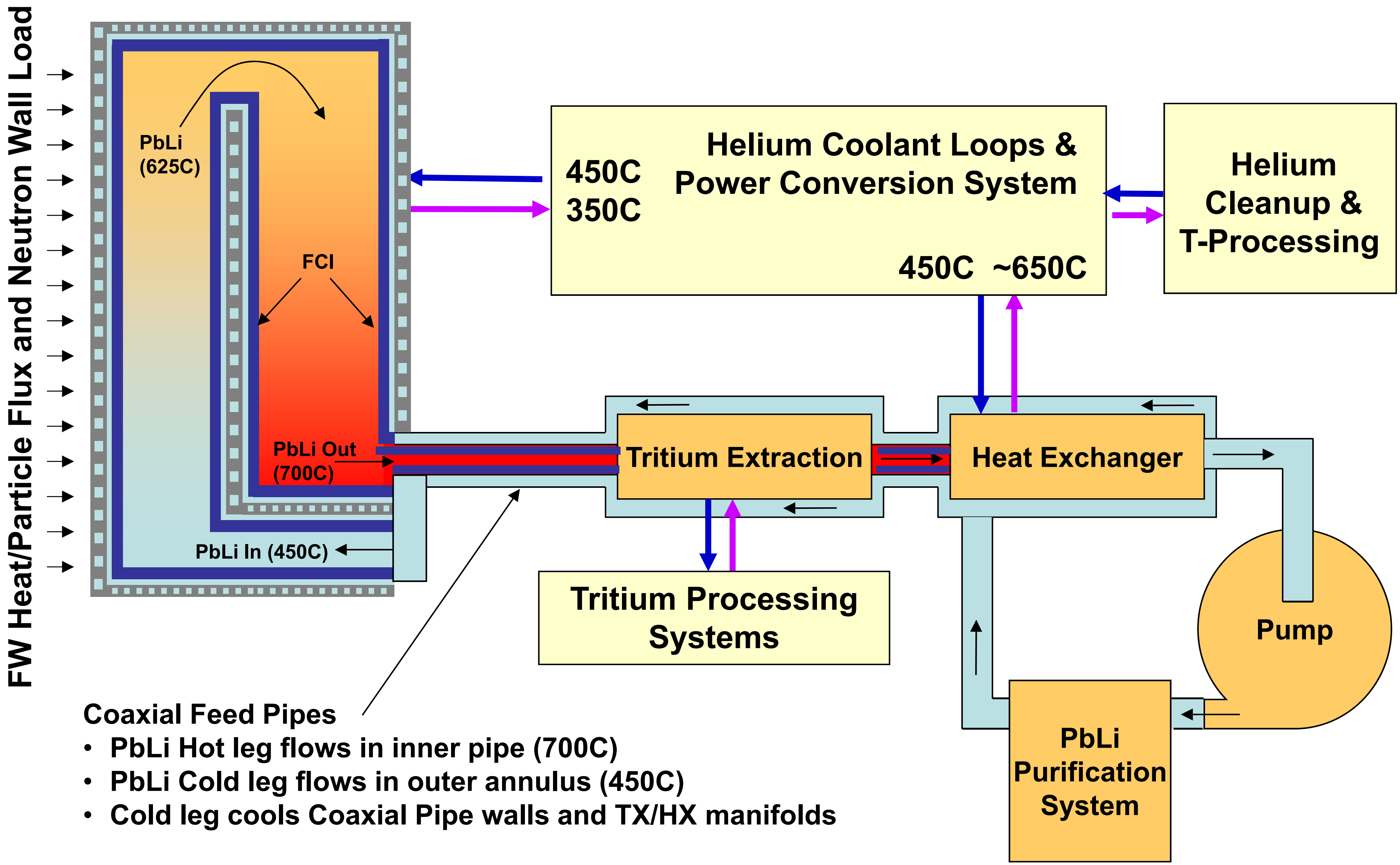
Variation of Required TBR as a function of f_b x η_f for different t_r x q values

t_r = reserved time for outage (days) q = fraction of the T processing system that has failure



- Higher f_b and η_f mitigate the problems with T processing system outage
- T processing systems must be designed with high reliability and redundancy

A Simplified DCLL PbLi Transport System



Coaxial Feed Pipes

- PbLi Hot leg flows in inner pipe (700C)
- PbLi Cold leg flows in outer annulus (450C)
- Cold leg cools Coaxial Pipe walls and TX/HX manifolds

Blanket Tritium Inventory, Breeder & Coolant Processing time; PFC Tritium inventories and coolants processing; and processes other than plasma exhaust/fuel T processing

Blanket/Breeder/Coolant

- Tritium Inventory in Breeding Blanket is <1 kg
  - This is based on calculations and some experiments
  - Radiation- induced sintering for ceramic breeder may increase T inventory to ~ 5 kg
- There are proposals/designs for the tritium processing systems from breeders, helium purge, and coolants. But no detailed engineering design or experimental data/verifications for such systems
- Based on available information, tritium inventories in such systems are < 1 kg and tritium processing time < 24 hours
  - Much smaller impact on Required TBR compared to impact of plasma exhaust/fueling cycle

PFC (First Wall, Divertor)

- T trapping inventories in solid materials can be large for some materials (e.g. C), but the Fusion Program is moving away from such materials
- Tritium Permeation to First Wall and Divertor coolants from the plasma side can be large resulting in significant T inventories.
  - But the impact on Required TBR appears insignificant since such inventories would come out of the plasma exhaust/processing system (which is already accounted for in detail)

Comments on IFE T Self Sufficiency

- Achievable TBR in IFE is estimated to be somewhat higher than in MFE for present conceptual designs. Liquid wall concepts in IFE increase the achievable TBR.
- No detailed dynamic modelling analysis has been performed to predict the value of Required TBR in IFE systems.
- There was a study in 1993 which briefly analyzed the Required TBR for IFE using modified version of the MFE Tritium Dynamic Modelling code. This study showed that tritium inventories and required TBR are very sensitive to :
  - mean residence time for manufacturing targets in the Target Factory which must process huge number of targets per day
  - time and efficiencies of processes to extract unburnt fuel from the target debris and plasma chamber exhaust. (This area must receive attention to make sure it does not offset the advantage of higher burn up in IFE)

Conclusions on Tritium Self Sufficiency

We have identified a "phase space" of physics and technology conditions in which tritium self sufficiency can be attained. Our R & D in plasma physics, blanket technology, and fuel cycle must aim at ensuring tritium self sufficiency. In particular, our R & D Goals should:

Minimize Tritium Inventories and Reduce Required TBR

- T burnup fraction x fueling efficiency > 5%   (not less than 2%)
- Tritium processing time (in plasma exhaust/fueling cycle) < 6 hours
- Minimize Tritium Inventories in Blanket, PFC, other components
- Minimize tritium processing time in breeder and coolants cycles

Ensure Achievable TBR is not significantly below the currently calculated value of 1.15

- Avoid Design choices that necessitate use of large neutron absorbing materials in blanket and divertor regions (challenges: thickness of first wall and divertors and blankets structure to handle plasma off-normal conditions such as disruptions, and ELMS; passive coils inside the blanket region for plasma stabilization and attaining advanced plasma physics mode)
- Aim the R & D for subsystems that involve penetrations such as impurity control/exhaust and plasma auxiliary heating to focus on design options that result in minimum impact on TBR

When Can We Accurately Predict , Verify, and Validate Achievable TBR?

After we have:

1. Detailed, accurate definition of the design of the in-vessel components (PFC, First Wall/Blanket, penetrations, etc.)
2. Prototypical accurate integral neutronics experiments:
   - This can be achieved only in DT-plasma-based facility
   - Current integral experiments are limited to point neutron source with S < 5 x 10^{12} n/s. Does not allow a) accurate simulation of angular neutron flux, b) complex geometry with subsystem details and heterogeneity. (Efforts on such experiments showed that calculations differ from experiments by ~10%)

Analysis has shown that at least a "full sector" testing in fusion facility is required for accurate measurement of achievable TBR. (Uncertainties in extrapolation in the poloidal direction from module is larger than the required accuracy.)

- **ITER TBM will provide very important information on achievable TBR (initial verification of codes, models, and data).**
- **FNSF is essential in providing more definitive validation of codes, models, and data and the predictability of achievable TBR. (Total tritium production will be measured directly in addition to local measurements). FNSF is essential to validating the design of blanket, divertor, and other in-vessel components.**

Role of ITER in Resolving Tritium Fuel Cycle Issues and Demonstrating the Principles of Tritium Self-Sufficiency

- We will learn from ITER (and other physics devices) what tritium burn-up fraction and fueling efficiency are achievable.
  - **ITER must explore methods to increase f_b and η_f.**
- Work on ITER fuel processing systems will help quantify inventories, flow rates, and processing times required in fusion at near reactor scale (for plasma exhaust/fueling cycle).
- **ITER TBM will provide important data on some key aspects of tritium breeding and extraction.**

Demonstration of tritium self-sufficiency requires another DT fusion facility (FNSF), in addition to ITER, in which full breeding blankets, or at least "complete sectors", efficient plasma fueling, fast plasma exhaust processing, and fully integrated tritium processing systems can be tested.

References for Tritium Self Sufficiency

1. M. Abdou, et. al, "Deuterium-Tritium Fuel Self-Sufficiency in Fusion Reactors", Fusion Technology, 9: 250-285 (1986).
2. W. Kuan and M. Abdou, "A New Approach for Assessing the Required Tritium Breeding Ratio and Startup Inventory in Future Fusion Reactors", Fusion Technology, 35: 309-353 (1999).
3. William Kuan, PhD Thesis, UCLA-ENG-98-193.
4. M. Sawan, M. Abdou, "Physics and Technology Conditions for attaining Tritium Self-Sufficiency for the DT Fuel Cycle", Fusion Engineering & Design, 81:(8–14), 1131–1144 (2006).
5. W. Kuan and M. Abdou, R. Scott Willms, “Dynamic Simulation of a Proposed ITER Tritium Processing System”, Fusion Technology, VOL. 28 OCT. 1995.
6. D.K. Murdoch, et. al, “Tritium Inventory Issues for Future Reactors: Choices, Parameters, Limits”, Fusion Engineering & Design, 46 (1999) 255-271.
7. David Murdoch, et. al, “Strategy for Determination of ITER In-Vessel Tritium Inventory”, Fusion Engineering & Design, 75-79 (2005) 667-671.
8. M. Glugla, et. al, “The ITER Tritium Systems”, Fusion Engineering & Design, 82 (2007) 472-487.
9. Yoshiyuki Asaoka, et. al, “Requirement of Tritium Breeding Ratio for Early Fusion Power Reactors”, Fusion Technology VOL. 30 DEC. 1996.
10. Yong Song, et. al, “Tritium Analysis of Fusion-Based Hydrogen Production Reactor FDS-III”, Fusion Engineering and Design 85 (2010) 1044-1047.
11. L.R. Baylor at. al., “Comparison of fueling efficiency from different fueling locations on DIII-D”, Journal of Nuclear Materials, 313-316 (2003) 530-533.

What we have done so far:

Many blanket design studies, modelling activities, and experiments, primarily single effect, have been carried out.

What is needed to show the practicality of the blanket and associated nuclear systems:

This question has been extensively studied over many years.

This will be the focus of the remainder of this presentation.

Top-Level Technical Issues for FNST/Blanket (set 1 of 2)

(Details of these issues published in many papers, Last update: December 2009)

Tritium

1. "Phase Space" of practical plasma, nuclear, material, and technological conditions in which tritium self sufficiency can be achieved
2. Tritium extraction, inventory, and control in solid/liquid breeders and blanket, PFC, fuel injection and processing, and heat extraction systems

Fluid-Material Interactions

3. MHD Thermofluid phenomena and impact on transport processes in electrically-conducting liquid coolants/breeders in both electrically conducting and insulated ducts
4. Interfacial phenomena, chemistry, compatibility, surface erosion & corrosion

Materials Interactions and Response

5. Structural materials performance and mechanical integrity under the effect of radiation and thermo-mechanical loadings in blanket/PFC
6. Functional materials property changes and performance under irradiation and high temperature and stress gradients (including HHF armor, ceramic breeders, beryllium multipliers, flow channel inserts, electric and thermal insulators, tritium permeation and corrosion barriers, etc.)
7. Fabrication and joining of structural and functional materials

Top-Level Technical Issues for FNST/Blanket

(set 2 of 2)

Plasma-Material Interactions

8. Plasma-surface interactions, recycling, erosion/redeposition, vacuum pumping
9. Bulk interactions between plasma operation and blanket and PFC systems, electromagnetic coupling, and off-normal events

Reliability, Availability, Maintainability (RAMI)

10. Failure modes, effects, and rates in blankets and PFC's in the integrated fusion environment
11. System configuration and remote maintenance with acceptable machine down time

All issues are strongly interconnected:

- they span requirements
- they span components
- they span many technical disciplines of science & engineering

Reliability/Availability/Maintainability/Inspectability (RAMI)

Availability required for each component needs to be high

Component	#	failure rate (1/hr)	MTBF (yrs)	MTTR/type Major (hrs)	MTTR/type Minor (hrs)	Fraction Failures Major	Outage Risk	Component Availability
Toroidal C	16	5 x10^{-6}	23	10^4	240	0.1	0.098	0.91
P C								
Magnet supplies	4	1 x10^{-4}	1.14	72	10	0.1	0.007	0.99
Cryogenics	2	2 x10^{-4}	0.57	300	24	0.1	0.022	0.978
Blanket	100	1 x10^{-5}	11.4	800	100	0.05	0.135	0.881
Divertor	32	2 x10^{-5}	5.7	500	200	0.1	0.147	0.871
Htg/CD	4							0.884
Fueling	1							0.998
Tritium System	1							0.995
Vacuum	3							0.998
Conventional equi								0.952
TOTAL SYSTEM	(Due to unscheduled maintenances)						0.624	0.615

Two key parameters:

MTBF – Mean time between failures

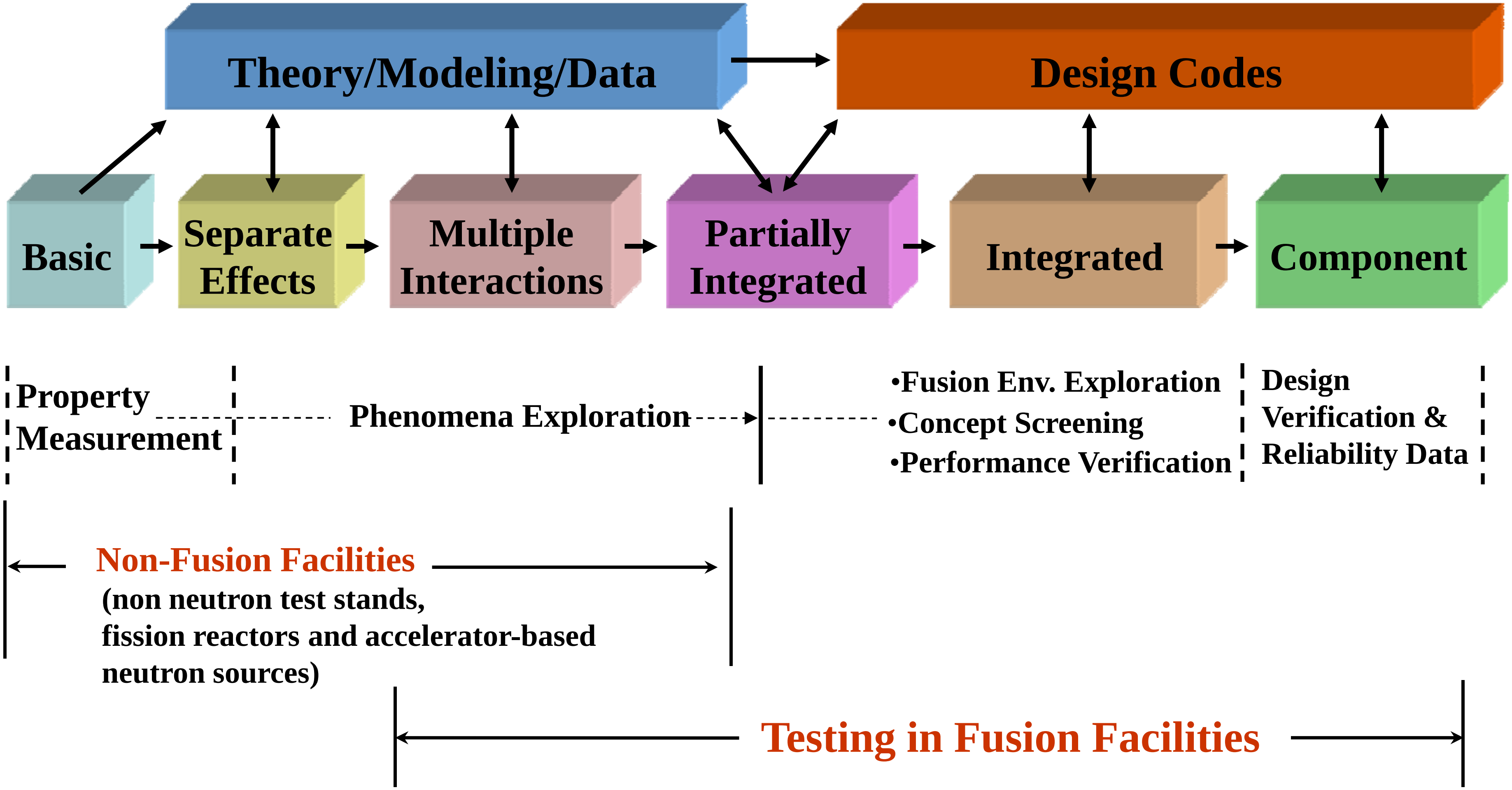
MTTR – Mean time to repair

DEMO availability of 50% requires:

- **Blanket/Divertor Availability ~ 87%**
- **Blanket MTBF >11 years**
- **MTTR < 2 weeks**

These are challenging requirements. They require extensive R&D, particularly a "reliability growth" program through testing in DT fusion facilities

Science-Based Framework for FNST R&D involves modeling and experiments in non-fusion and fusion facilities



- Experiments in non-fusion facilities are essential and are prerequisites to testing in fusion facilities.
- Testing in Fusion Facilities is NECESSARY to uncover new phenomena, validate the science, establish engineering feasibility, and develop components.

We have studied and defined the FNST/blanket testing requirements in DT fusion facilities

Non-fusion facilities

Preparatory R&D

Modeling and experiments in non-fusion facilities

- Basic property measurement
- Understand issues through modeling and single and multiple-effect experiments

None of the top level technical issues can be resolved before testing in the fusion environment

FNST Testing in Fusion Facilities

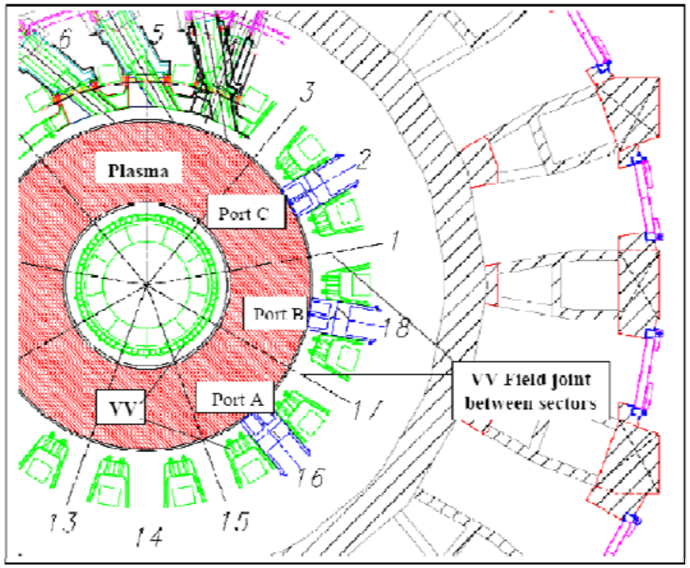
Scientific Feasibility	Engineering Feasibility	Engineering Development
Stage I	Stage II	Stage III
0.1 - 0.3 MW-y/m²	**1 - 3 MW-y/m²**	**> 4 - 6 MW-y/m²**
≥ 0.5 MW/m² burn > 200 s	**1-2 MW/m² steady state or long burn COT ~ 1-2 weeks**	**1-2 MW/m² steady state or long burn COT ~ 1-2 weeks**
Sub-Modules/Modules	Modules (10-20m²)	Modules/Sectors (20-30m²)
• Establish scientific feasibility of basic functions under prompt responses and under the impact of rapid property changes in early life	• Establish engineering feasibility of blankets/PFC/materials (satisfy basic functions & performance, up to 10 to 20% of MTBF and of lifetime)	• RAMI: Failure modes, effects, and rates and mean time to replace/fix components and reliability growth • Verify design and predict availability of FNST components in DEMO

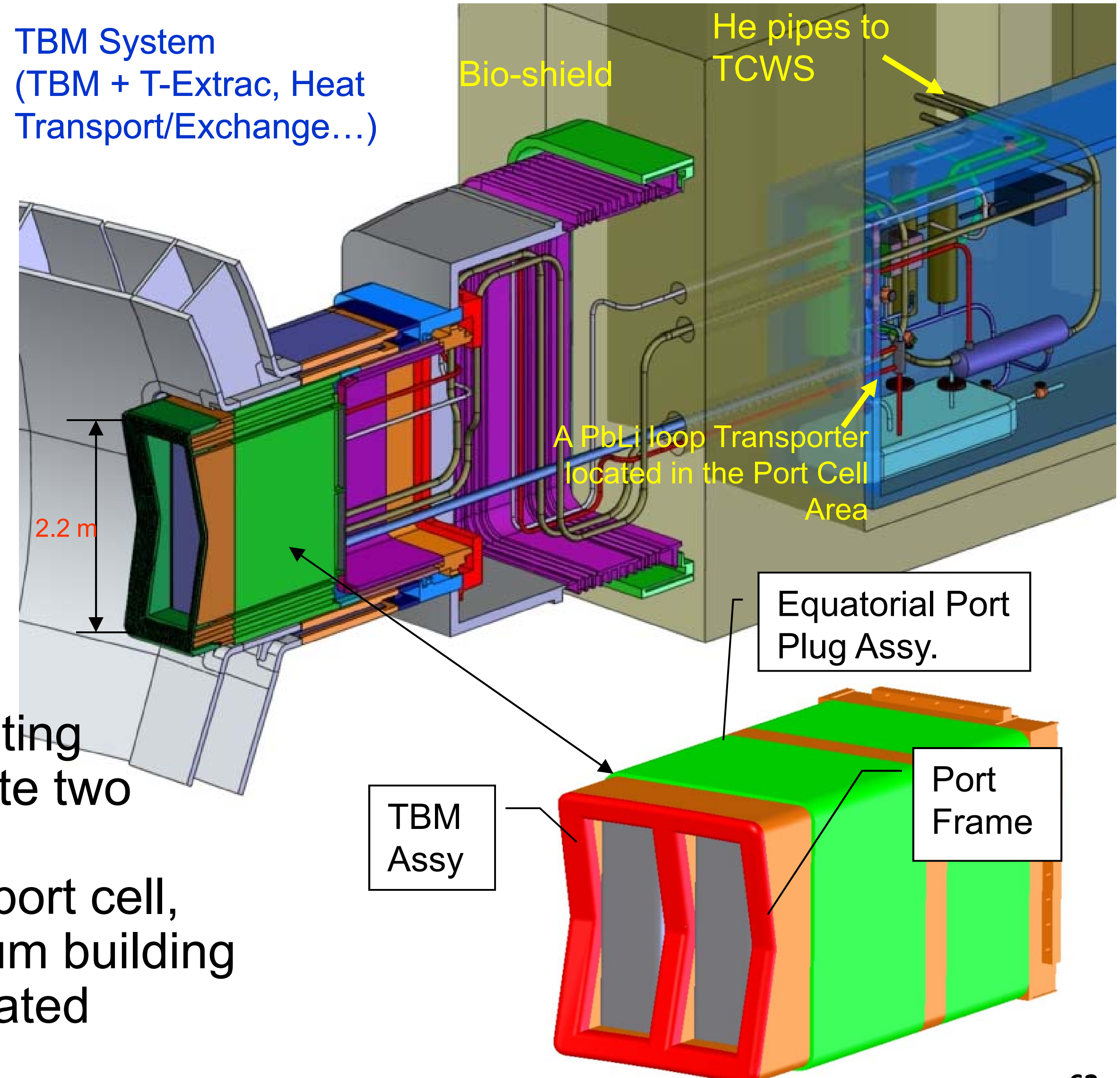
DEMO

Where can such testing be done?

ITER and FNSF are proposed.

ITER Provides Substantial Hardware Capabilities for Testing of Tritium Breeding Blanket Modules (TBM) and *Systems*

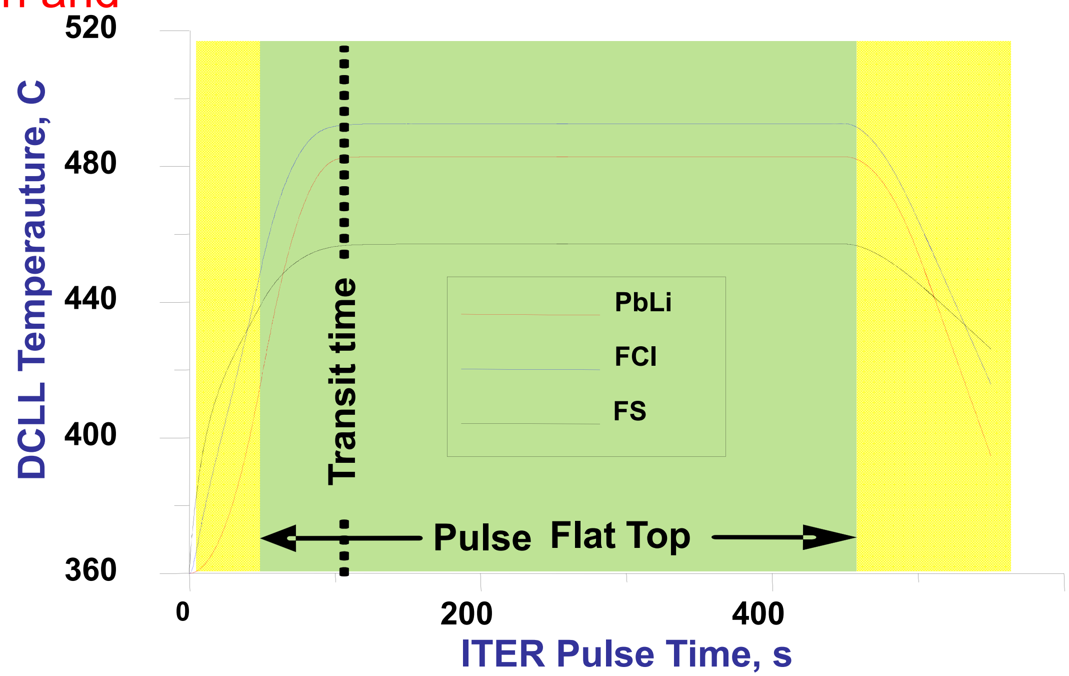




- ITER has allocated **3 equatorial ports** (1.75 x 2.2 m^2) for TBM testing
- Each port can accommodate two modules **(i.e. 6 TBMs max)**
- Space and services in the port cell, cooling water building, tritium building and hot cells are also allocated

ITER TBM experiments can be used to explore "prompt" responses, tritium breeding, nuclear heating, early irradiation effects

– The ITER test module size, neutron and magnetic fields, and pulse length are all significant
  - Especially the combined strong, spatially complex, nuclear heating and magnetic field
– "Prompt" phenomena that reach near steady state during the ITER burn (minutes to an hour)
  - Tritium production profiles
  - Nuclear heating profiles
  - MHD thermofluid behavior
  - Thermomechanical state and temperature profiles
– Cyclic equilibrium over many pulses
  - Tritium concentration and permeation
  - Corrosion and activated product transport
  - Impact of beginning of life radiation damage in ceramic breeders and insulators



DCLL TBM rear channel temperature reaches steady state after about 1 PbLi transit time through the module

US Planning for ITER Test Blanket Experiments

Based on DOE request, the FNST community spent 2 years formulating a TBM technical plan and cost estimate.**

- Focus tests on 2 concepts (1. LM, 2. Ceramic Breeder) with substantially different feasibility issues
- Capitalize on international collaboration with other ITER parties (strong interest world-wide in blankets using ceramic breeders or PbLi based blankets)

Cost was estimated at $10M /yr for 10 years

- To deliver first TBM test modules and ancillary systems
- About 50% was for R&D to build and qualify for nuclear operation


REPORT NO. UCLA-FNT-216

US ITER TEST BLANKET MODULE (TBM) PROGRAM

VOLUME I: TECHNICAL PLAN
AND COST ESTIMATE SUMMARY

BY:

M.A. ABDOU, N.B. MORLEY, A.Y. YING, C.P.C WONG, T. MANN, S. TOURVILLE
AND THE US ITER TEST BLANKET MODULE TEAM

KEY CONTRIBUTORS:

M.A. ABDOU, N.B. MORLEY, A.Y. YING, S. SMOLENTSEV, S. SHARAFAT, M. DAGHER, M. YOUSSEF - UNIVERSITY OF CALIFORNIA, LOS ANGELES
C.P.C. WONG, D. SONN, C. BAXI - GENERAL ATOMICS
T. MANN, Y. KATOH, B. PINT, S. ZINKLE, P. FOGARTY - OAK RIDGE NATIONAL LABORATORY
R.J. KURTZ - PACIFIC NORTHWEST NATIONAL LABORATORY
B. MERRILL, P. SHARPE, P. CALDERONI, D. PETTI - IDAHO NATIONAL LABORATORY
R. NYGREN, M. ULRICKSON, T. TANAKA - SANDIA NATIONAL LABORATORY
M. SAWAN, G. SVIATOSLAVSKY - UNIVERSITY OF WISCONSIN
R.S. WILLMS - LOS ALAMOS NATIONAL LABORATORY
S. REYES - LAWRENCE LIVERMORE NATIONAL LABORATORY
D.K. SZE - UNIVERSITY OF CALIFORNIA, SAN DIEGO
S. TOURVILLE, S. MALANG, A. ROWCLIFFE - CONSULTANTS

FIRST ISSUED: JULY 2006
REVISED: APRIL 2007

UCLA

** complete reports available at www.fusion.ucla.edu

This planning study forced consideration of the practical engineering R&D required to build, qualify and operate a practical blanket system. Such R&D are not typically addressed in our scientific studies

A key feature of the R&D identified for US ITER-TBM was the emphasis on the engineering requirements for qualification and licensing

R&D on Fabrication, Joining, Diagnostics, Mockup Testing, Tritium Control, and QA'd codes accepted by the Licensing Agency, etc.
are all required to be able to build and get a TBM accepted by ITER

US DCLL & CB modules

Test Modules

1. Thermofluid MHD
2. SiC FCI Fabrication and Properties
3. SiC/FS/PbLi Compatibility & Chemistry
4. RAFS Fabrication & Materials Prop.
5. Be Joining to FS for First Wall
6. Helium System Subcomponents Tests
7. PbLi/H2O Hydrogen Production Rates
8. TBM Diagnostics
9. Ceramic Breeder Thermomechanics
10. Breeder/Multiplier Pebble Specifications
11. Partially Integrated Mockups Testing

Tritium Systems

1. Transport Model Development
2. Tritium Extraction from PbLi
3. Tritium Extraction from He
4. Purge gas composition & flow conditions

System Integration

Predicative Capability Integration

Fusion Nuclear Science Facility (FNSF)

- The idea of FNSF (also called VNS, CTF) is to build a small size, low fusion power DT plasma-based device in which Fusion Nuclear Science and Technology (FNST) experiments can be performed and tritium self sufficiency can be demonstrated in the relevant fusion environment:

  1- at the smallest possible scale, cost, and risk, and

  2- with practical strategy for solving the tritium consumption and supply issues for FNST development.

  In MFE: small-size, low fusion power can be obtained in a low-Q (driven) plasma device, with normal conducting Cu magnets

  - Equivalent in IFE: reduced target yield (and smaller chamber radius)

- Design options for FNSF are being explored

Reduced activation Ferritic/Martensitic Steel (FS) is the reference structural material option for DEMO

- **FS should be used for TBMs in ITER and therefore for mockup tests prior to ITER**
- **FS should be the structural materials for both base and testing breeding blankets on FNSF.**
- **FS irradiation data base from fission reactors extends to ~ 80 dpa, but it generally lacks He (only limited simulation of He in some experiments).**
  - ✓ **There is confidence in He data in fusion typical neutron energy spectrum up to 100 appm (~ 10 dpa).**

FNSF Strategy/Design for Breeding Blankets, Structural Materials, PFC & Vacuum Vessel

Day 1 Design

- Vacuum vessel – low dose environment, proven materials and technology
- **Inside the VV** – **all is "experimental."** Understanding failure modes, rates, effects and component maintainability is a crucial FNSF mission.
- Structural material - reduced activation ferritic steel for in-vessel components
- Base breeding blankets - conservative operating parameters, ferritic steel, 10 dpa design life (acceptable projection, obtain confirming data ~10 dpa & 100 ppm He)
- Testing ports - **well instrumented, higher performance blanket experiments (also special test module for testing of materials specimens)**

Upgrade Blanket (and PFC) Design, Bootstrap approach

- Extrapolate a factor of 2 (standard in fission, other development), 20 dpa, 200 appm He. Then extrapolate next stage of 40 dpa…
- Conclusive results from FNSF (real environment) for testing structural materials,
  - no uncertainty in spectrum or other environmental effects
  - prototypical response, e.g., gradients, materials interactions, joints, …

Summary Points

- The importance of tritium breeding blankets has been recognized since the beginning of fusion reactor studies in 1970. Many (>50) blanket design studies carried out to date.
- The blanket is a complex system, with multiple functions and multiple materials in multiple-field environment.
  - Multiple effects and synergistic phenomena will be dominant.
  - True simulation will be possible only in DT fusion facility.
- Many design studies, modelling activities, and experiments (primarily separate-effect) have been carried out.
- The key issues have been identified and characterized. The required R&D to resolve these issues have been well defined. We understand the path forward.
- We have identified a “phase space” of physics and technology parameters and conditions in which tritium self sufficiency can be attained. Our R&D in plasma physics, blanket technology, and fuel cycle is aimed at ensuring tritium self sufficiency.
- We have analyzed the engineering issues of blanket and fuel cycle system practicality and attractiveness. There are many challenges, but the field is now positioned to move forward with an extensive R&D Program toward multiple-effect and integrated testing in fusion facilities (ITER TBM and FNSF).

References: The proceedings of the International Symposium on Fusion Nuclear Technology (ISFNT) over the past 23 years provide scholarly publications on advances in Blankets, Tritium, PFC, Materials, Neutronics, Modeling and Experiments, Design and System Studies

- **ISFNT-1** (Tokyo, Japan – April 10-19, 1988)
  Proceedings published in *Fusion Engineering and Design*, Vol. 8, 9, 10 (1989)

- **ISFNT-2** (Karlsruhe, Germany – June 2-7, 1991)
  Proceedings published in *Fusion Engineering and Design*, Vol. 16, 17, 18 (1991)

- **ISFNT-3** (Los Angeles, CA, USA – June 26-July 1, 1994)
  Proceedings published in *Fusion Engineering and Design*, Vol. 27, 28, 29 (1995)

- **ISFNT-4** (Tokyo, Japan – April 6-11, 1997)
  Proceedings published in *Fusion Engineering and Design*, Vol. 39, 40, 41, 42 (1998)

- **ISFNT-5** (Rome, Italy – September 19-24, 1999
  Proceedings published in *Fusion Engineering and De*sign, Vol. 49, 50, 51, 52 (2000)

- **ISFNT-6** (San Diego, CA, USA – April 7-12, 2002)
  Proceedings published in *Fusion Engineering and Design*, Vol. 61, 62, 63, 64 (2002)

- **ISFNT-7** (Tokyo, Japan – May 22-27, 2005)
  Proceedings published in *Fusion Engineering and Design*, Vol. 81(1-14):1-1706 (2006)

- **ISFNT-8** (Heidelberg, Germany – September 30-October 5, 2007)
  Proceedings published in *Fusion Engineering and Design*, Vol. 83(7-12):785-1902 (2008)

- **ISFNT-9** (Dalian, China – October 11-16, 2009)
  Proceedings published in *Fusion Engineering and Design*, Vol. 85(7-12):963-2348 (2010)

- **ISFNT-10** (Portland, OR, USA – September 11-16, 2011)
  http://www.isfnt-10.org/