Overview of the Principles and Challenges of Fusion Nuclear Technology

One of a number of lectures given at the Institute for Plasma Research (IPR) at Gandhinagar, India, January 2007

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Overview of the Principles and Challenges of Fusion Nuclear Technology

Outline

- Definitions
- FNT components and functions, impact of vacuum vessel on the blanket
- World supply of tritium
- Fusion D-T fuel cycle and issues, T breeding, neutron multipliers, structural materials
- Types of blankets, solid breeder blanket concepts and issues, liquid breeder blanket concepts and issues
- Stages of FNT testing
- Role of ITER and why it is important but not sufficient for FNT DEMO development

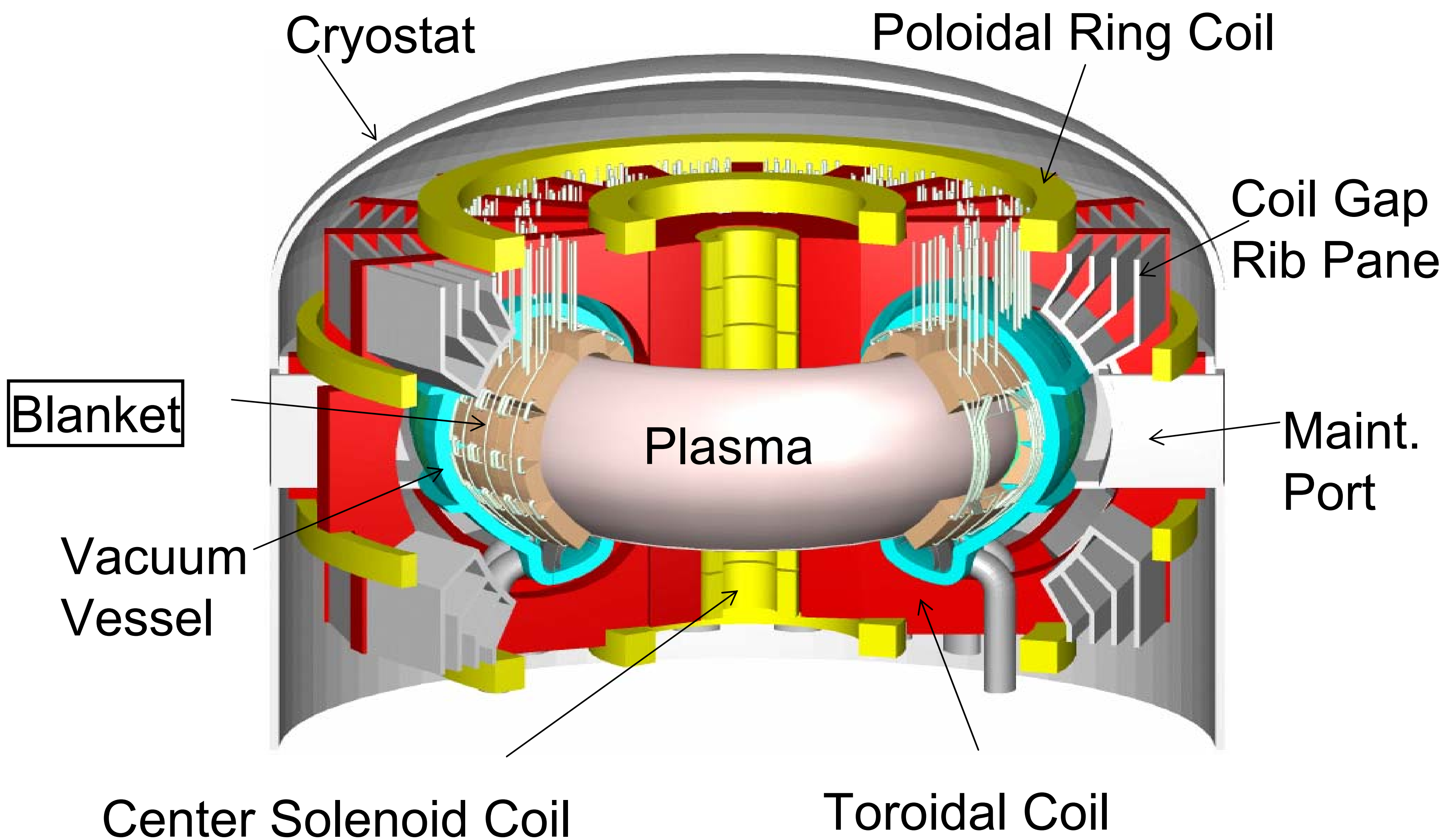
Incentives for Developing Fusion

- Fusion powers the Sun and the stars
  - It is now within reach for use on Earth
- In the fusion process, lighter elements are “fused” together, making heavier elements and producing prodigious amounts of energy
- Fusion offers very attractive features:
  - Sustainable energy source

    (for DT cycle; provided that Breeding Blankets are successfully developed)
  - No emission of Greenhouse or other polluting gases
  - No risk of a severe accident
  - No long-lived radioactive waste
- Fusion energy can be used to produce electricity and hydrogen, and for desalination

FNT: Components from Edge of Plasma to TFC

Blanket / Divertor immediately circumscribe the plasma



JAEA DEMO Design

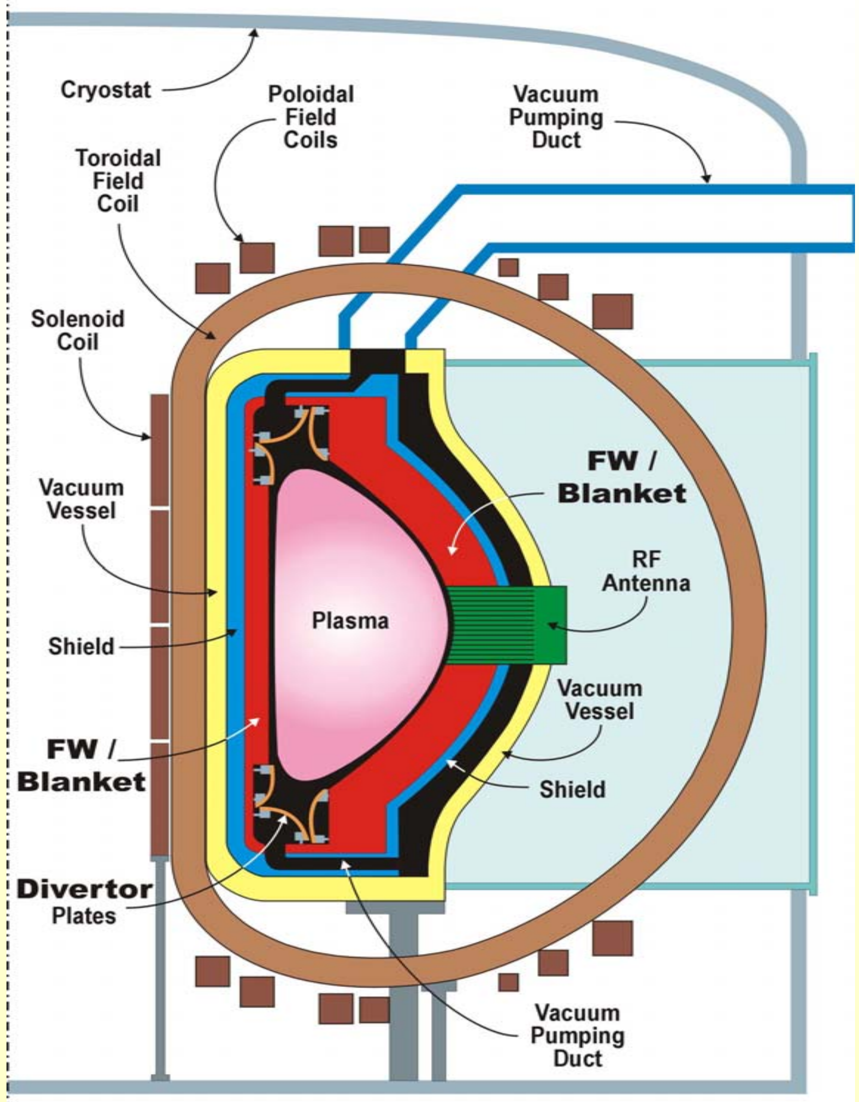
Fusion Nuclear Technology (FNT)
Fusion Power & Fuel Cycle Technology

FNT Components from the edge of the Plasma to TF Coils (Reactor "Core")

1. Blanket Components
2. Plasma Interactive and High Heat Flux Components
   a. divertor, limiter
   b. rf antennas, launchers, wave guides, etc.
3. Vacuum Vessel & Shield Components

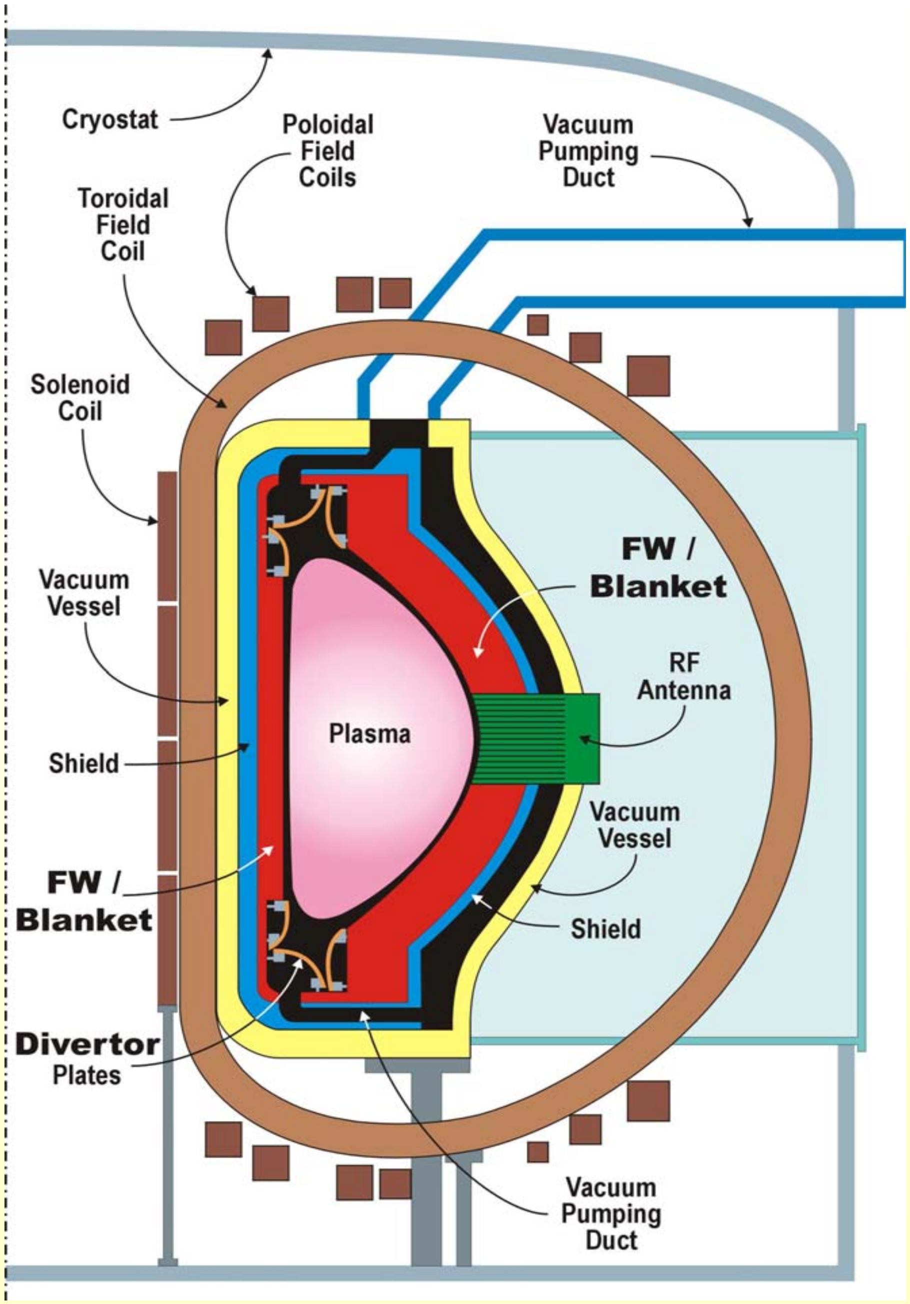
Other Components affected by the Nuclear Environment

4. Tritium Processing Systems
5. Instrumentation and Control Systems
6. Remote Maintenance Components
7. Heat Transport and Power Conversion Systems



Notes on FNT:

- The Vacuum Vessel is outside the Blanket (Shield). It is in a low-radiation field.
- Vacuum Vessel Development for DEMO should be in good shape from ITER experience.
- The Key Issues are for Blanket / PFC.
- Note that the first wall is an integral part of the blanket (ideas for a separate first wall were discarded in the 1980's). The term "Blanket" now implicitly includes the first wall.
- Since the Blanket is inside of the vacuum vessel, many failures (e.g. coolant leak from module) require immediate shutdown and repair/replacement.



Adaptation from ARIES-AT Design

The Deuterium-Tritium (D-T) Cycle

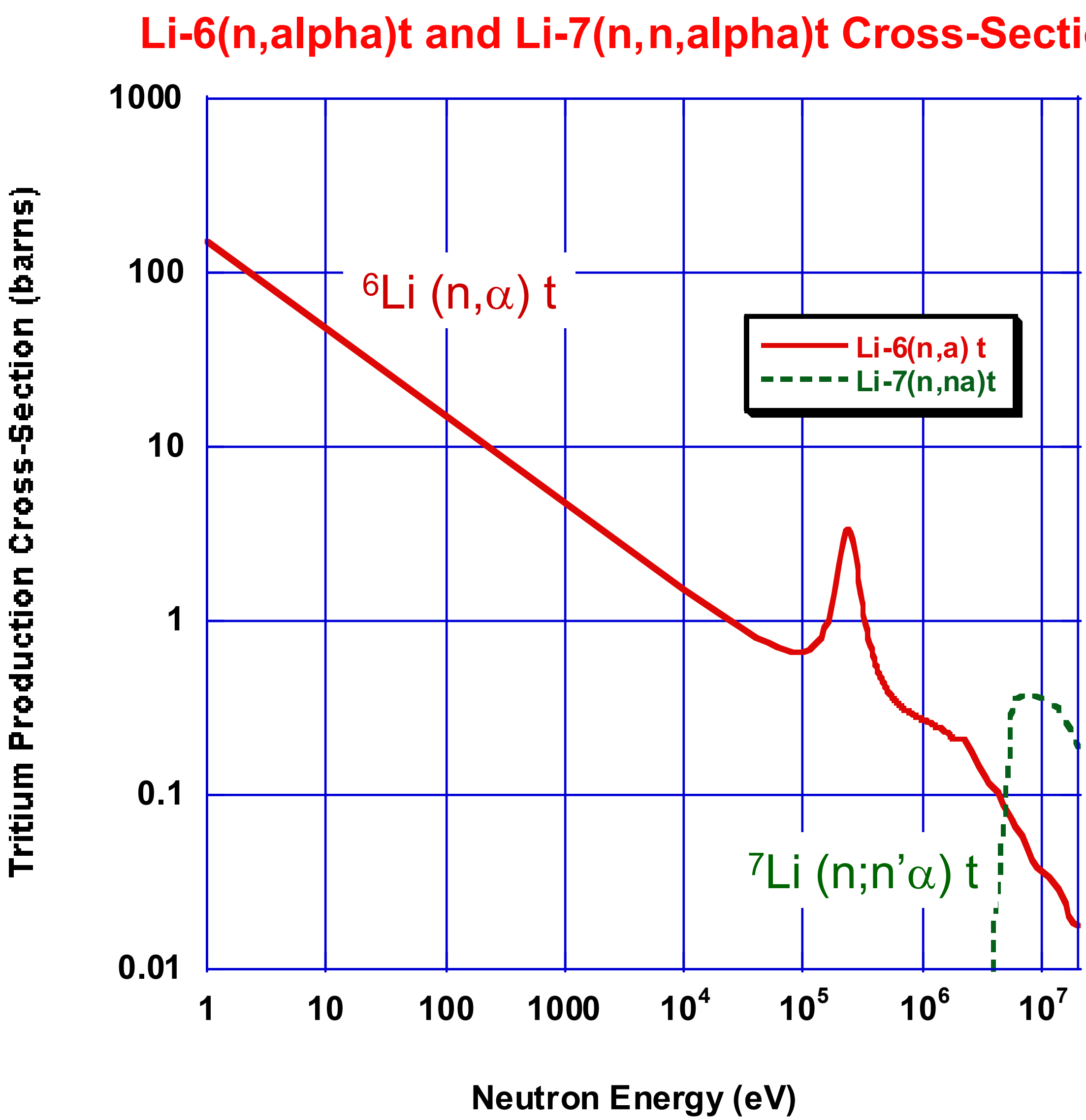
- World Program is focused on the D-T cycle (easiest to ignite):

$$D + T \rightarrow n + \alpha + 17.58 \text{ MeV}$$

- The fusion energy (17.58 MeV per reaction) appears as Kinetic Energy of neutrons (14.06 MeV) and alphas (3.52 MeV)
- Tritium does not exist in nature! Decay half-life is 12.3 years

  (Tritium must be generated inside the fusion system to have a sustainable fuel cycle)
- The only possibility to adequately breed tritium is through neutron interactions with lithium
  - Lithium, in some form, must be used in the fusion system

Tritium Breeding

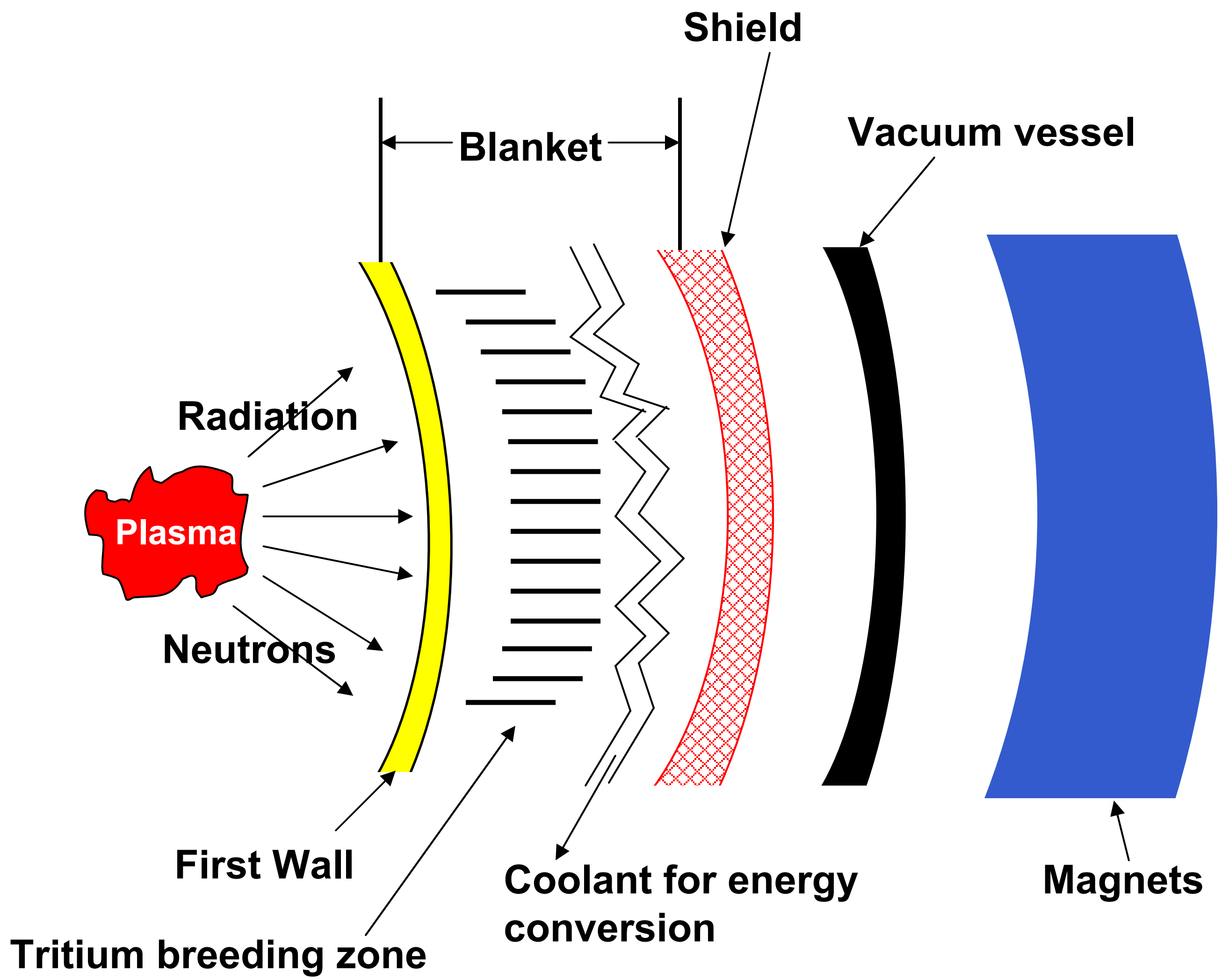


Natural lithium contains 7.42 % ^{6}Li and 92.58 % ^{7}Li.

$$^6Li + n \rightarrow t + \alpha + 4.78MeV$$

$$^7Li + n \rightarrow t + \alpha + n - 2.47MeV$$

The ^{7}Li(n;n'α)t reaction is a threshold reaction and requires an incident neutron energy in excess of 2.8 MeV.


Shield
Blanket
Vacuum vessel
Radiation
Plasma
Neutrons
First Wall
Tritium breeding zone
Coolant for energy conversion
Magnets

Blanket (including first wall)

Blanket Functions:

A. Power Extraction

- Convert kinetic energy of neutrons and secondary gamma rays into heat
- Absorb plasma radiation on the first wall
- Extract the heat (at high temperature, for energy conversion)

B. Tritium Breeding

- Tritium breeding, extraction, and control
- Must have lithium in some form for tritium breeding

C. Physical Boundary for the Plasma

- Physical boundary surrounding the plasma, inside the vacuum vessel
- Provide access for plasma heating, fueling
- Must be compatible with plasma operation
- Innovative blanket concepts can improve plasma stability and confinement

D. Radiation Shielding of the Vacuum Vessel

The Blanket (and PFCs) 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. 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**.

R&D Tasks to be Accomplished Prior to Demo

1) Plasma

- Confinement/Burn
- Disruption Control
- Current Drive/Steady State
- Edge Control

2) Plasma Support Systems

- Superconducting Magnets
- Fueling
- Heating

3) Fusion Nuclear Technology Components and Materials
[Blanket, First Wall, High Performance Divertors, rf Launchers]

- Materials combination selection and configuration optimization
- Performance verification and concept validation
- Show that the fuel cycle can be closed (tritium self-sufficiency)
- Failure modes and effects
- Remote maintenance demonstration
- Reliability growth
- Component lifetime

4) Systems Integration

Where Will These Tasks be Done?!

- Burning Plasma Facility (ITER) and other plasma devices will address 1, 2, & much of 4
- Fusion Nuclear Technology (FNT) components and materials can only be PARTIALLY tested in ITER and will require dedicated fusion facilities prior to DEMO, e.g. VNS/CTF

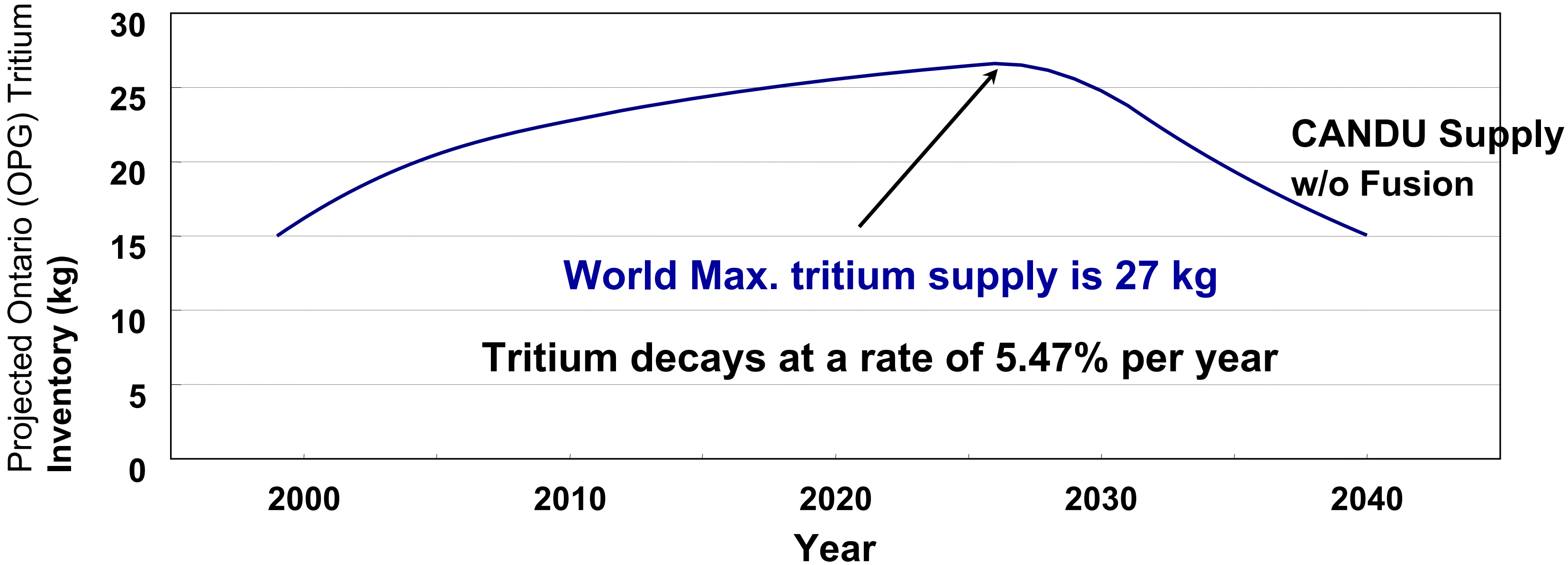
Challenging
Fusion Nuclear Technology Issues

1. Tritium Supply &
   Tritium Self-Sufficiency
2. High Power Density
3. High Temperature
4. MHD for Liquid Breeders / Coolants
5. Tritium Control (Permeation)
6. Reliability / Maintainability / Availability
7. Testing in Fusion Facilities

Tritium Consumption

AND Tritium Supply Issue

Projections for World Tritium Supply Available to Fusion Reveal Serious Problems



Tritium Consumption in Fusion is HUGE! Unprecedented!

55.8 kg per 1000 MW fusion power per year

Production & Cost:

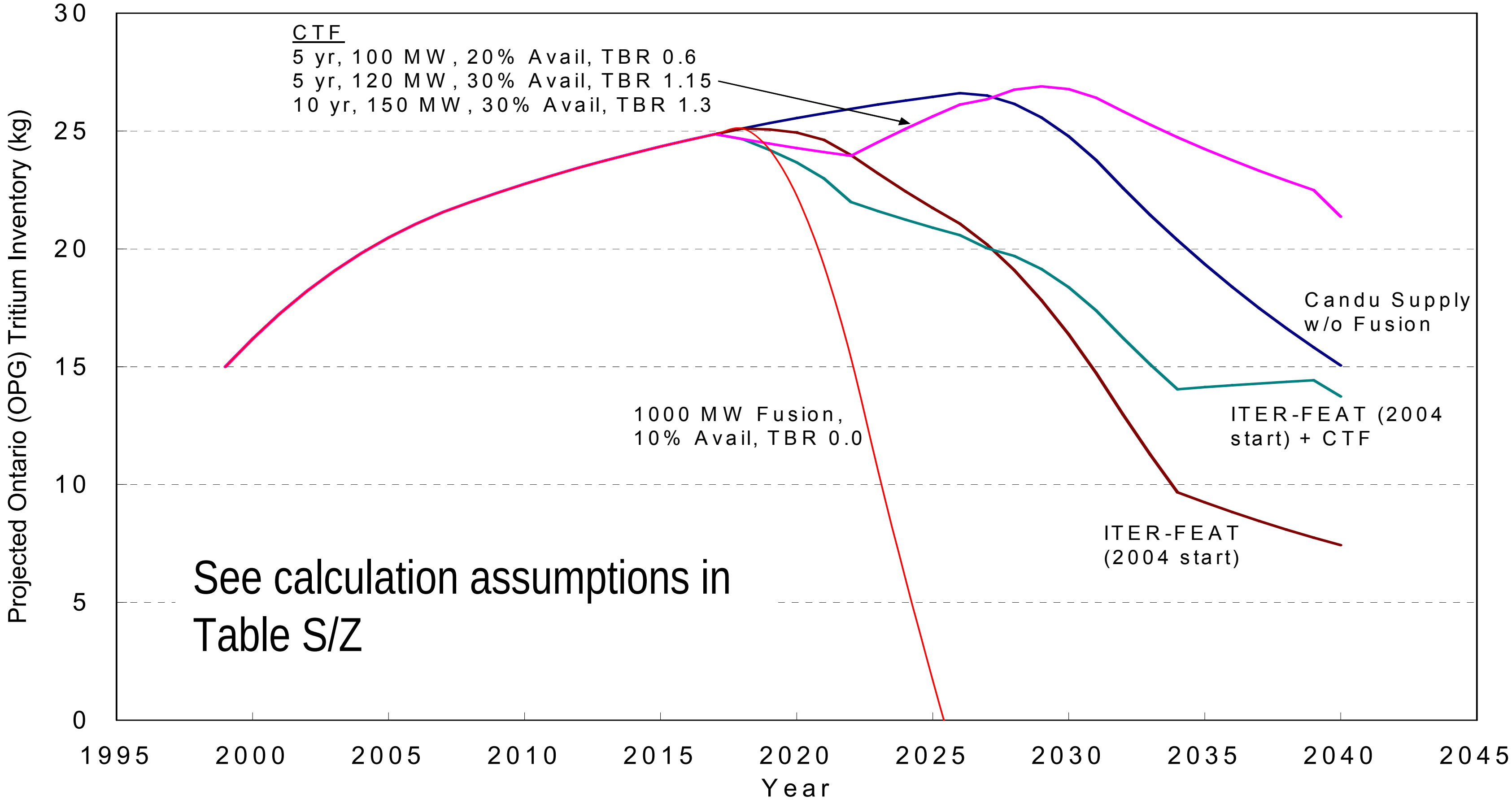
CANDU Reactors: 27 kg from over 40 years, $30M/kg (current)

Fission reactors: 2–3 kg per year.
It takes tens of fission reactors to supply one fusion reactor.

$84M-$130M per kg, per DOE Inspector General*

*DOE Inspector General's Audit Report, "Modernization of Tritium Requirements Systems", Report DOE/IG-0632, December 2003, available at www.ig.doe.gov/pdf/ig-0632.pdf

Projections for World Tritium Supply Available to Fusion for Various Scenarios



- World Tritium Supply would be exhausted by 2025 if ITER were to run at 1000 MW fusion power with 10% availability.
- Large Power DT Fusion Devices must breed their own tritium; not practical for blanket development.
- We need 5-10 kg of tritium as "start-up" inventory for DEMO (can be provided from CTF operating with TBR > 1 at later stage of operation).
- Blanket development and ITER-TBM are necessary in the **near term** to allow continued development of D-T fusion.

Table S/Z

(data used in Fig. for Tritium Supply and Consumption Calculations)

Tritium Supply Calculation Assumptions:

- Ontario Power Generation (OPG) has seven of twenty CANDU reactors idled
- Reactors licensed for 40 years
- 15 kg tritium in 1999
- 1999 tritium recovery rate was 2.1 kg/yr
- Tritium recovery rate will decrease to 1.7 kg/yr in 2005 and remain at this level until 2025
- After 2025, reactors will reach their end-of-life and the tritium recovery rate will decrease rapidly
- OPG sells 0.1 kg/yr to non-ITER/VNS users
- Tritium decays at 5.47 % / yr

It is assumed that the following will NOT happen:

- Extending CANDU lifetime to 60 years
- Restarting idle CANDU's
- Processing moderator from non-OPG CANDUs (Quebec, New Brunswick)
- Building more CANDUs
- Irradiating Li targets in commercial reactors (including CANDUs)
- Obtaining tritium from weapons programs of "nuclear superpowers"
- Premature shutdown of CANDU reactors

Table S/Z (cont'd)

(data used in Fig. for Tritium Supply and Consumption Calculations cont'd)

ITER-FEAT Assumptions:

- Construction starts in 2004 and lasts 10 years.
- There are four years of non-tritium operation.
- This is followed by 16 years of tritium operation. The first five years use tritium at a linearly increasing rate reaching 1.08 kg T used per year in the fifth year. Tritium usage remains at this level for the remainder of tritium operations.
- There is no tritium breeding (TBR=0).
- There is no additional tritium needed to fill materials and systems.

CTF Assumptions:

- Begins burning tritium in 2017
- 5 yr, 100 MW, 20% availability, TBR 0.6
- 5 yr, 120 MW, 30% availability, TBR 1.15
- 10 yr, 150 MW, 30% availability, TBR 1.3

Heat and Radiation Loads on First Wall

- Neutron Wall Load ≡ P_{nw}

  P_{nw} = Fusion Neutron Power Incident on the First Wall per unit area

  $= J_w E_o$

  J_w = fusion neutron (uncollided) current on the First Wall

  E_o = Energy per fusion neutron = 14.06 MeV

- Typical Neutron Wall Load ≡ 1-5 MW/m^2

  At 1 MW/m^2: $J_w = 4.43 \times 10^{17}\ n \cdot m^{-2} \cdot s^{-1}$

- Note the neutron flux at the first wall (0-14 MeV) is about an order of magnitude higher than J_w

- Surface heat flux at the first wall

  This is the plasma radiation load. It is a fraction of the α-power

  $q_w = 0.25\ P_{nw} \cdot f_\alpha$

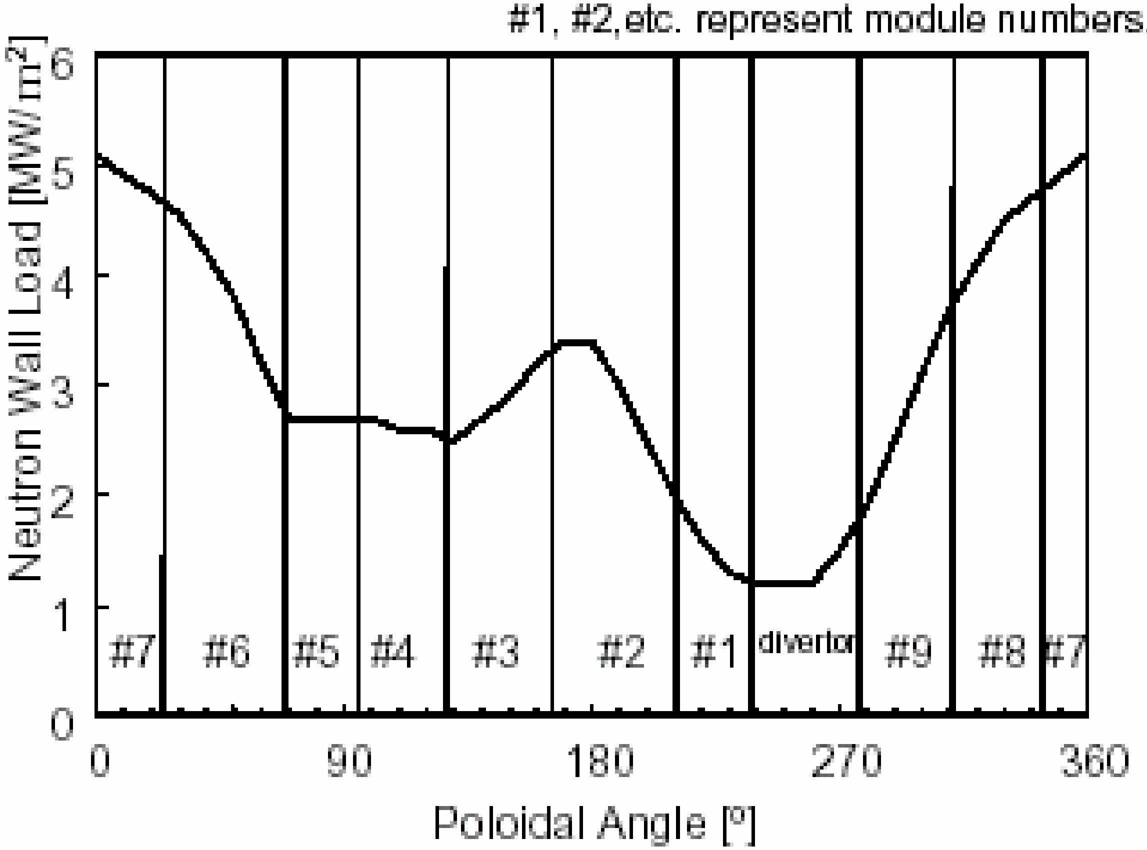
  where f is the fraction of the α-power reaching the first wall

  (note that the balance, 1 – f, goes to the divertor)

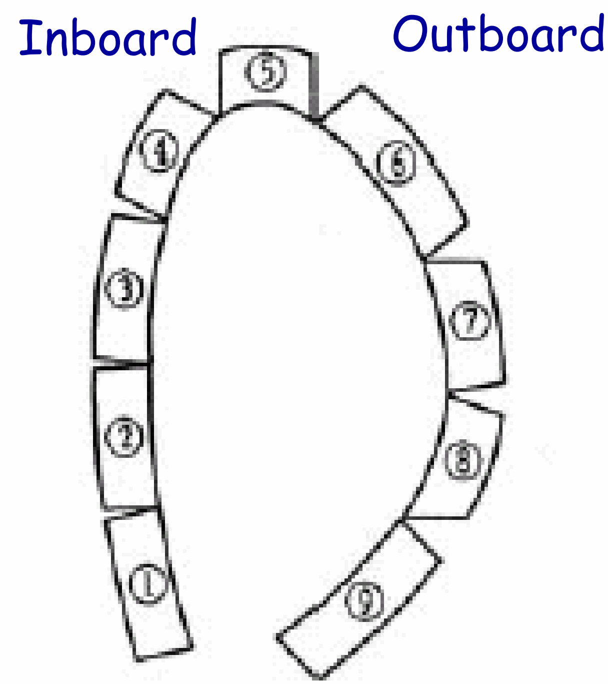
Poloidal Variation of Neutron Wall Load

– Neutron wall load has profile along the poloidal direction (due to combination of toroidal and poloidal geometries)

– Peak to average is typically about 1.4

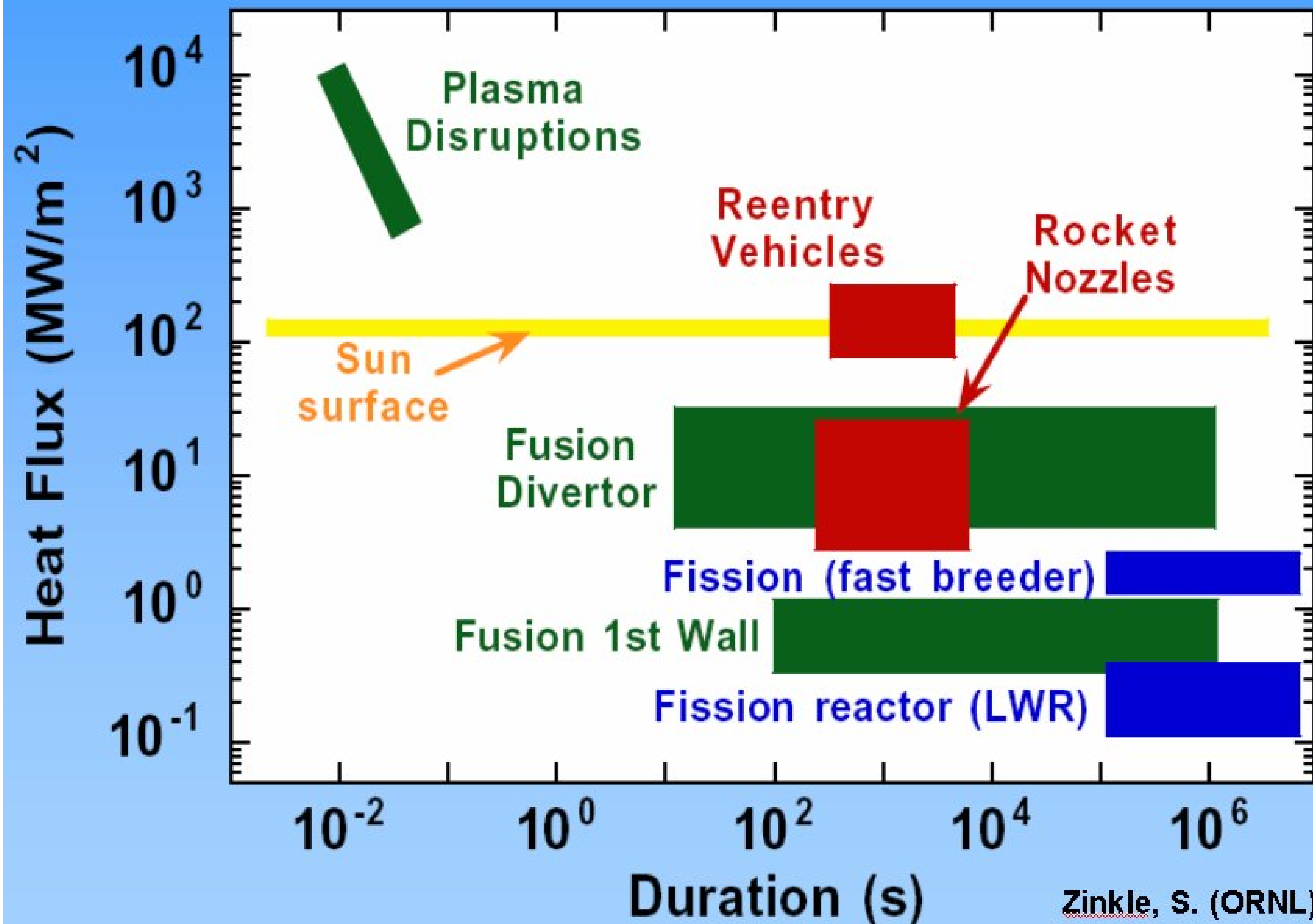


Profile of Neutron Wall Load
(equatorial plane, outboard, in 0º)



Module #

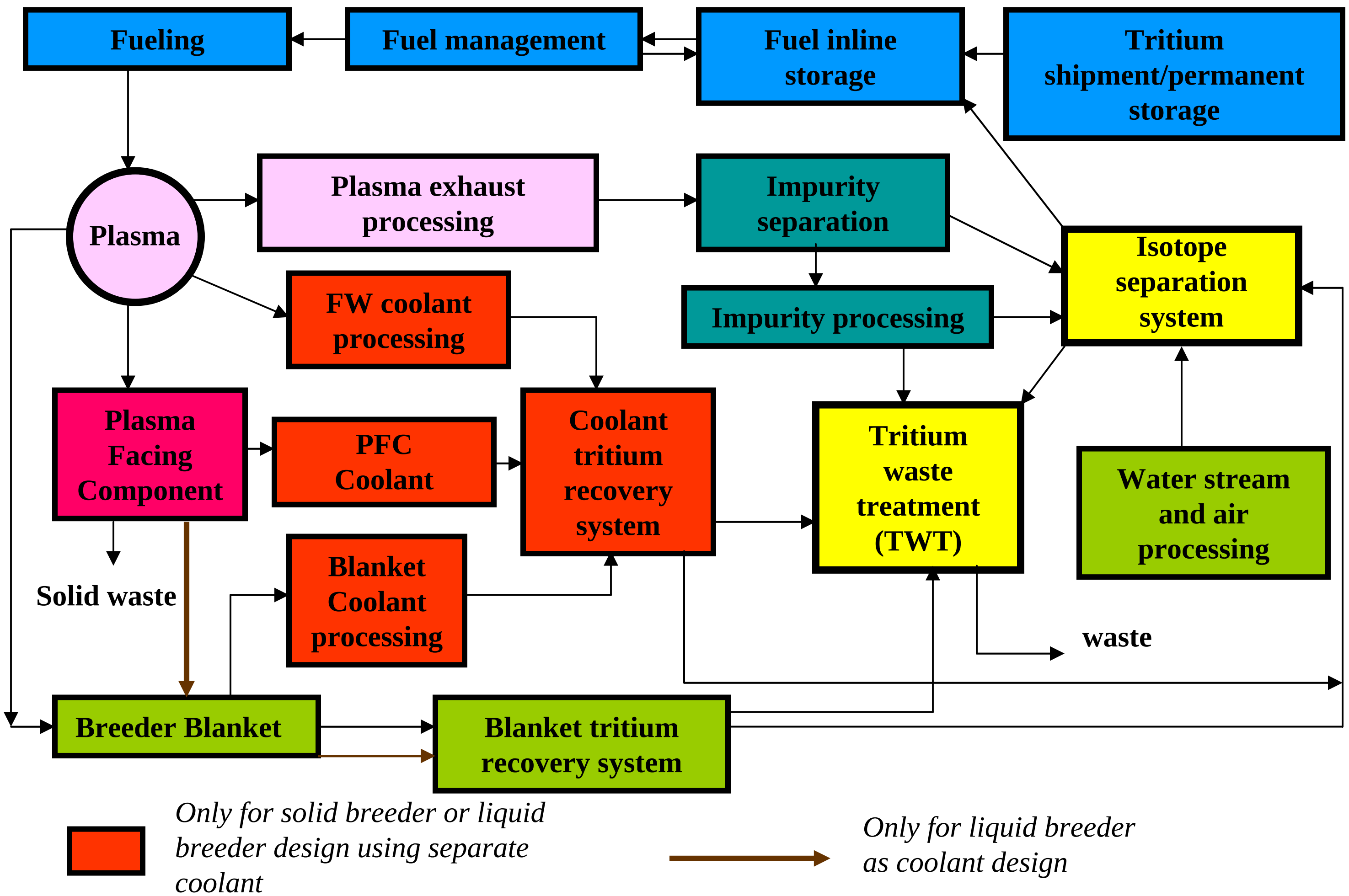
Comparison of Heat Fluxes



Zinkle, S. (ORNL)

Fuel Cycle Dynamics

The D-T fuel cycle includes many components whose operation parameters and their uncertainties impact the required TBR



Examples of key parameters:

- *ß: Tritium fraction burn-up*
- *T_i: mean T residence time in each component*
- *Tritium inventory in each component*
- *Doubling time*
- *Days of tritium reserves*
- *Extraction inefficiency in plasma exhaust processing*

Tritium Self-Sufficiency

- TBR ≡ Tritium Breeding Ratio = $\dot{N}^{+} / \dot{N}^{-}$

  $\dot{N}^{+}$ = Rate of tritium production (primarily in the blanket)

  $\dot{N}^{-}$ = Rate of tritium consumption (burnt in plasma)

Tritium self-sufficiency condition: $\Lambda_a > \Lambda_r$

Λ_r = Required tritium breeding ratio

Λ_r is 1 + G, where G is the margin required to: a) compensate for losses and radioactive decay between production and use, b) supply inventory for start-up of other fusion systems, and c) provide a hold-up inventory, which accounts for the time delay between production and use as well as reserve storage. Λ_r is dependent on many system parameters and features such as plasma edge recycling, tritium fractional burn up in the plasma, tritium inventories, doubling time, efficiency/capacity/reliability of the tritium processing system, etc.

Λ_a = Achievable breeding ratio

Λ_a is a function of FW thickness, amount of structure in the blanket, presence of stabilizing shell materials, PFC coating/tile/materials, material and geometry for divertor, plasma heating, fueling and penetration.

Tritium Self-Sufficiency

Tritium self-sufficiency condition: $\Lambda a > \Lambda r$

Λr = Required tritium breeding ratio

Λr is 1 + G, where G is the margin required to account for tritium losses, radioactive decay, tritium inventory in plant components, and supplying inventory for start-up of other plants.

Λr is dependent on many system physics and technology parameters:

– plasma edge recycling, tritium fractional burn-up in the plasma
– tritium inventories (release/retention) in components
– efficiency/capacity/reliability of the tritium processing system, etc.

Λa = Achievable tritium breeding ratio

Λa is a function of technology, material and physics.

– FW thickness, amount of structure in the blanket, blanket concept (ITER detailed engineering is showing FW may have to be much thicker than we want for T self sufficiency)
– Presence of stabilizing/conducting shell materials/coils for plasma control and attaining advanced plasma physics modes
– Plasma heating/fueling/exhaust, PFC coating/materials/geometry
– Plasma configuration (tokamak, stellerator, etc.)

Need for High Power Density *Capability*

A. To improve potential attractiveness of fusion power compared to other energy sources (e.g., fission)

	PWR	BWR	LMFBR	ITER-Type
Average core power density (MW/m^3)	96	56	240	**0.4**

B. Some plasma confinement schemes have the potential to produce burning plasmas with high power density

– FW/Blanket/Divertor concepts developed in the 1970s and '80s have limitations on power density capability (wall load and surface heat flux)

– Substantial **PROGRESS** has been made in this area over the past several years (e.g. in the **APEX** Study in the US)

Blanket Materials

1. Tritium Breeding Material (Lithium in some form)

   Liquid: Li, LiPb (^{83}Pb ^{17}Li), lithium-containing molten salts

   Solid: Li_2O, Li_4SiO_4, Li_2TiO_3, Li_2ZrO_3

2. Neutron Multiplier (for most blanket concepts)

   Beryllium (Be, $Be_{12}Ti$)

   Lead (in LiPb)

3. Coolant

   – Li, LiPb – Molten Salt – Helium – Water

4. Structural Material

   – Ferritic Steel (accepted worldwide as the reference for DEMO)

   – Long-term: Vanadium alloy (compatible only with Li), and SiC/SiC

5. MHD insulators (for concepts with self-cooled liquid metals)
6. Thermal insulators (only in some concepts with dual coolants)
7. Tritium Permeation Barriers (in some concepts)
8. Neutron Attenuators and Reflectors

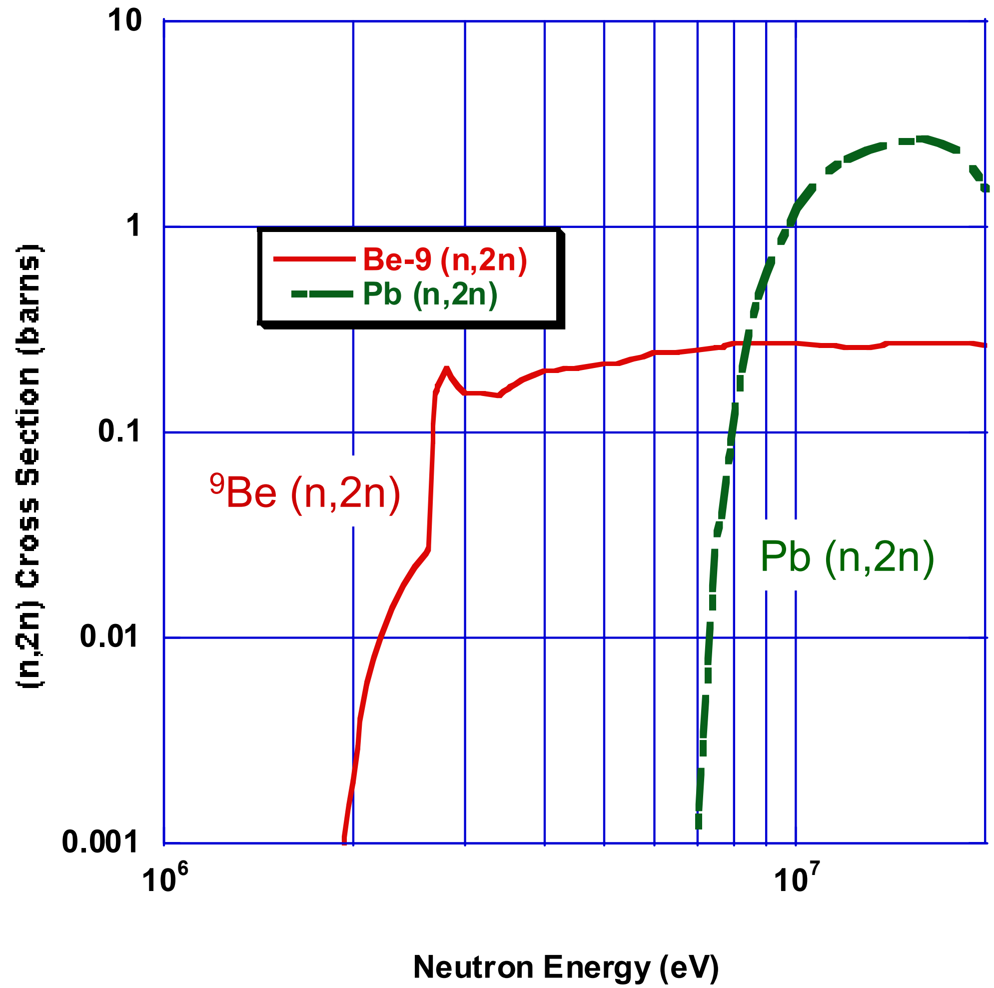
Neutron Multipliers

- Almost all concepts need a neutron multiplier to achieve adequate tritium breeding.

  (Possible exceptions: concepts with Li and Li_2O)
- Desired characteristics:
  - Large (*n*, 2*n*) cross-section with low threshold
  - Small absorption cross-sections
- Candidates:
  - Beryllium is the best (large *n*, 2*n* with low threshold, low absorption)
  - $Be_{12}Ti$ may have the advantage of less tritium retention, less reactivity with steam
  - Pb is less effective except in LiPb
  - Beryllium results in large energy multiplication, but resources are limited

Examples of Neutron Multipliers Beryllium, Lead

Be-9 (n,2n) and Pb(n,2n) Cross-Sections- JENDL-3.2 Data



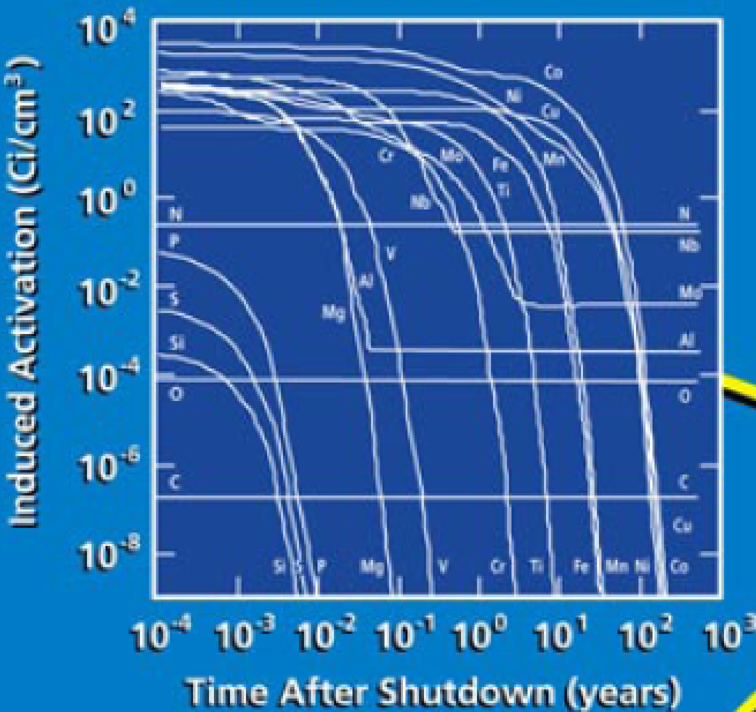
Structural Materials

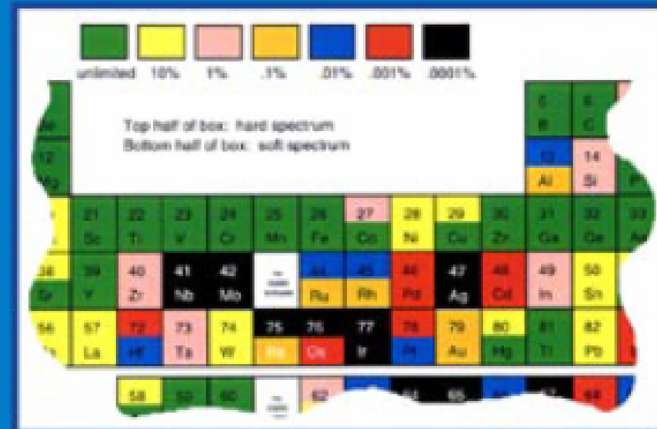
- Key issues include thermal stress capacity, coolant compatibility, waste disposal, and radiation damage effects
- The 3 leading candidates are ferritic/martensitic steel, V alloys and SiC/SiC (based on safety, waste disposal, and performance considerations)
- **The ferritic/martensitic steel is the reference structural material for DEMO**

(Commercial alloys (Ti alloys, Ni base superalloys, refractory alloys, etc.) have been shown to be unacceptable for fusion for various technical reasons)

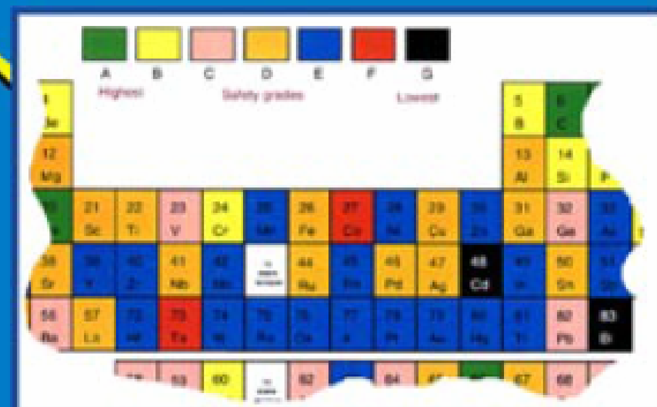
Structural Material	**Coolant/Tritium Breeding Material**					
	Li/Li	He/PbLi	H_2O/PbLi	He/Li ceramic	H_2O/Li ceramic	FLiBe/FLiBe
Ferritic steel						
V alloy						
SiC/SiC						

Structural Materials





Class C Waste Disposal



Safety

SiC

V Cr Ti Si

Fe Cr W V Ta

Low Activation Ferritic Steels for First Wall/Blanket Structures

Advantages

- Well-developed technology for nuclear and other advanced technology applications
- Fusion materials program has developed low activation versions with equivalent or superior properties
- Resistant to radiation-induced swelling and helium embrittlement
- Compatibility with aqueous, gaseous, and liquid metal coolants permits range of design options

Issues

- Upper operating temperature limited to ~ 550°C by loss of creep strength
- Potential for radiation-induced embrittlement at temperatures <400°C
- Possible design difficulties due to ferromagnetic properties

CURRENT APPROACH

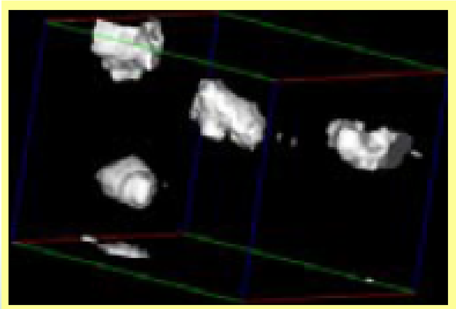
Expand Low Temperature Operating Window

- Pursue collaborative fission reactor irradiation program with EU and Japan
  - Investigate micro- mechanics of fracture and radiation-induced reductions in fracture toughness
  - Understand the role of helium on fracture and crack propagation
  - Develop Master Curve approach to examine deformation modes and fracture resistance

Expand High Temperature Operating Window

- Explore potential of nanocomposited ferritic (NCF) materials to expand upper operating temperature to ~800°C
  - Develop radiation-stable, high toughness microstructures

3-D atom probe image; clusters of ~100 atoms of Y, Ti, and O responsible for high strength of NCF materials

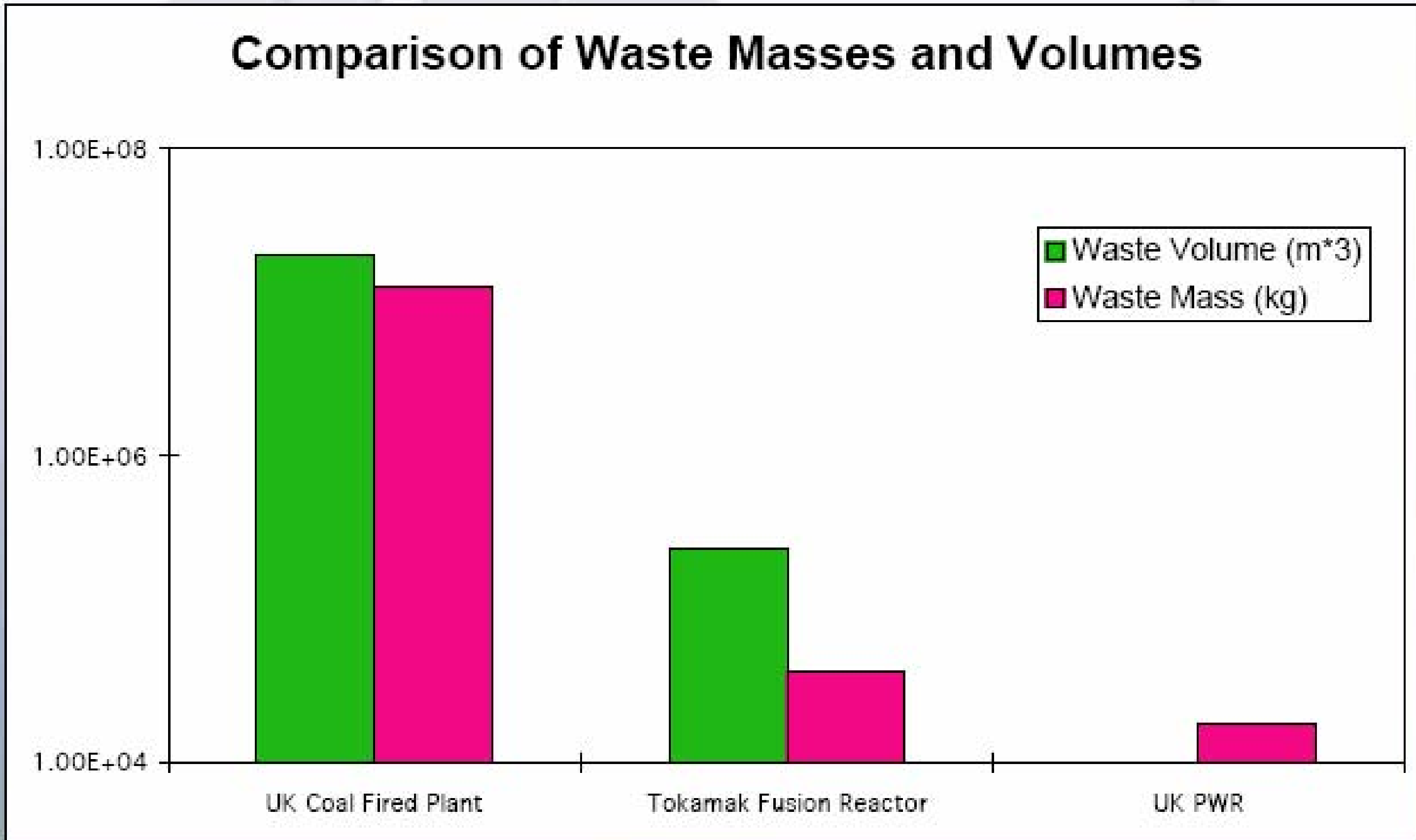


Comparison of fission and fusion structural materials requirements

	Fission (Gen. I)	Fission (Gen. IV)	Fusion (Demo)
Structural alloy maximum temperature	**<300˚C**	**600-850˚C (~1000˚C for GFRs)**	**550-700˚C (1000˚C for SiC)**
Max dose for core internal structures	**~1 dpa**	**~30-100 dpa**	**~150 dpa**
Max transmutation helium concentration	**~0.1 appm**	**~3-10 appm**	**~1500 appm (~10000 appm for SiC)**

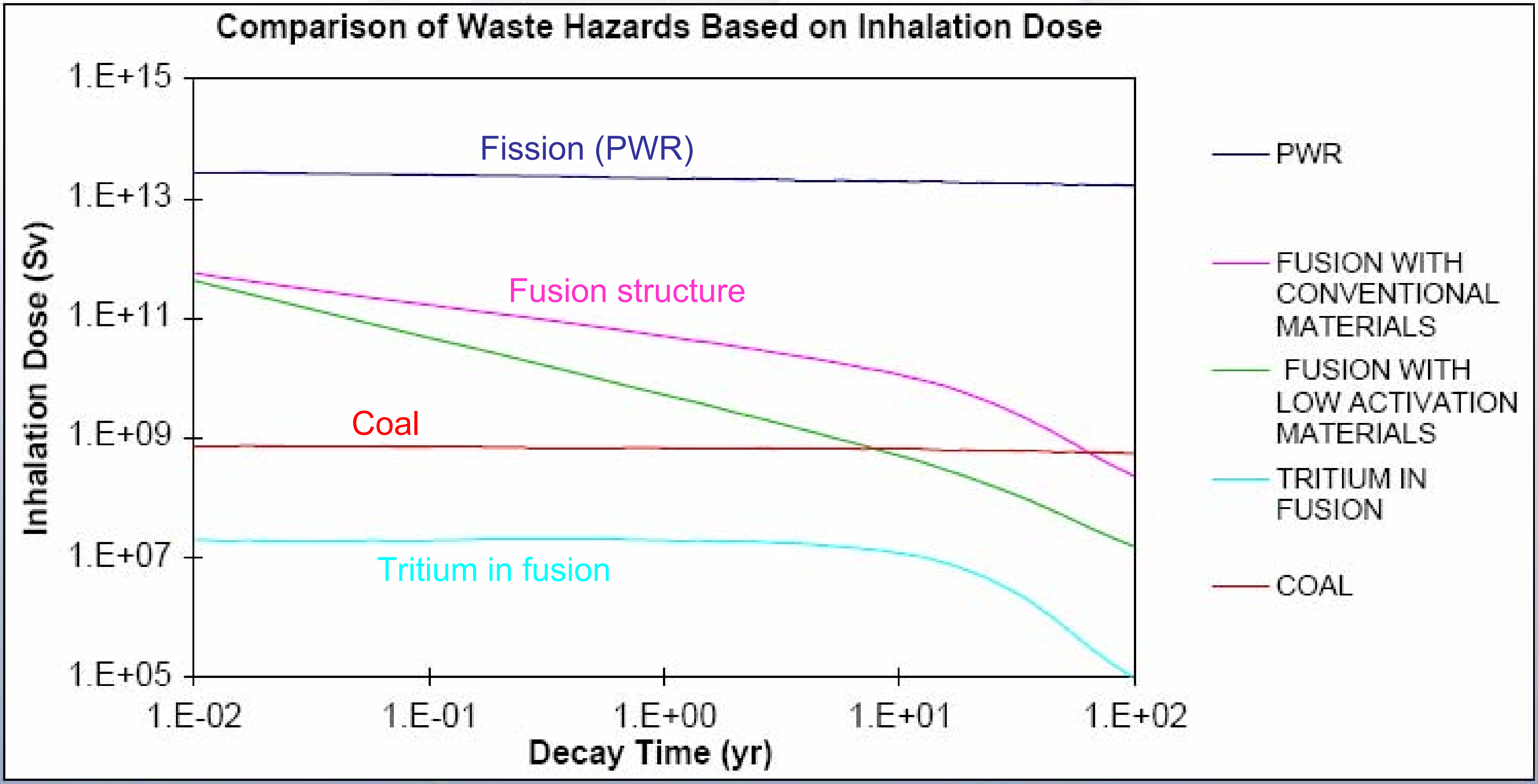
- Fusion has obtained enormous benefits from pioneering radiation effects research performed for fission reactors
  - Although the fusion materials environment is very hostile, there is confidence that satisfactory radiation-resistant reduced activation materials can be developed

Fusion radioactive waste volumes are more than fission but much less than coal for power plants of equal size.



From "A Study of the Environmental Impact of Fusion" (AERE R 13708)

Radiotoxicity (inhalation) of waste from fusion is less than fission and similar to that from coal at 100 years.



- From "A Study of the Environmental Impact of Fusion" (AERE R 13708).
- Coal radiotoxicity is based on Radon, Uranium, Thorium, and Polonium in coal ash
- Inhalation represents major pathways for uptake of material by the human body
- Dose hazard used here is a relative measure of radiotoxicity of material

Blanket Concepts

(many concepts proposed worldwide)

A. Solid Breeder Concepts

- Always separately cooled
- Solid Breeder: Lithium Ceramic (Li_2O, Li_4SiO_4, Li_2TiO_3, Li_2ZrO_3)
- Coolant: Helium or Water

B. Liquid Breeder Concepts

Liquid breeder can be:

a) **Liquid metal** (high conductivity, low Pr): Li, or $^{83}Pb\ ^{17}Li$

b) **Molten salt** (low conductivity, high Pr): Flibe $(LiF)_n \cdot (BeF_2)$, Flinabe (LiF-BeF_2-NaF)

B.1. Self-Cooled

- Liquid breeder is circulated at high enough speed to also serve as coolant

B.2. Separately Cooled

- A separate coolant is used (e.g., helium)
- The breeder is circulated only at low speed for tritium extraction

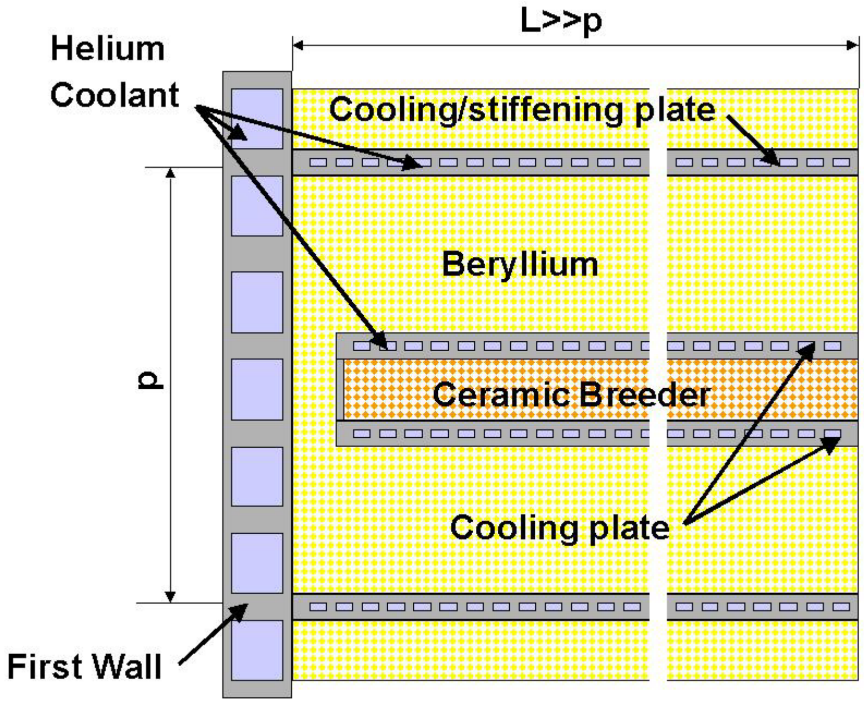
B.3. Dual Coolant

- FW and structure are cooled with separate coolant (He)
- Breeding zone is self-cooled

A Helium-Cooled Li-Ceramic Breeder Concept: Example

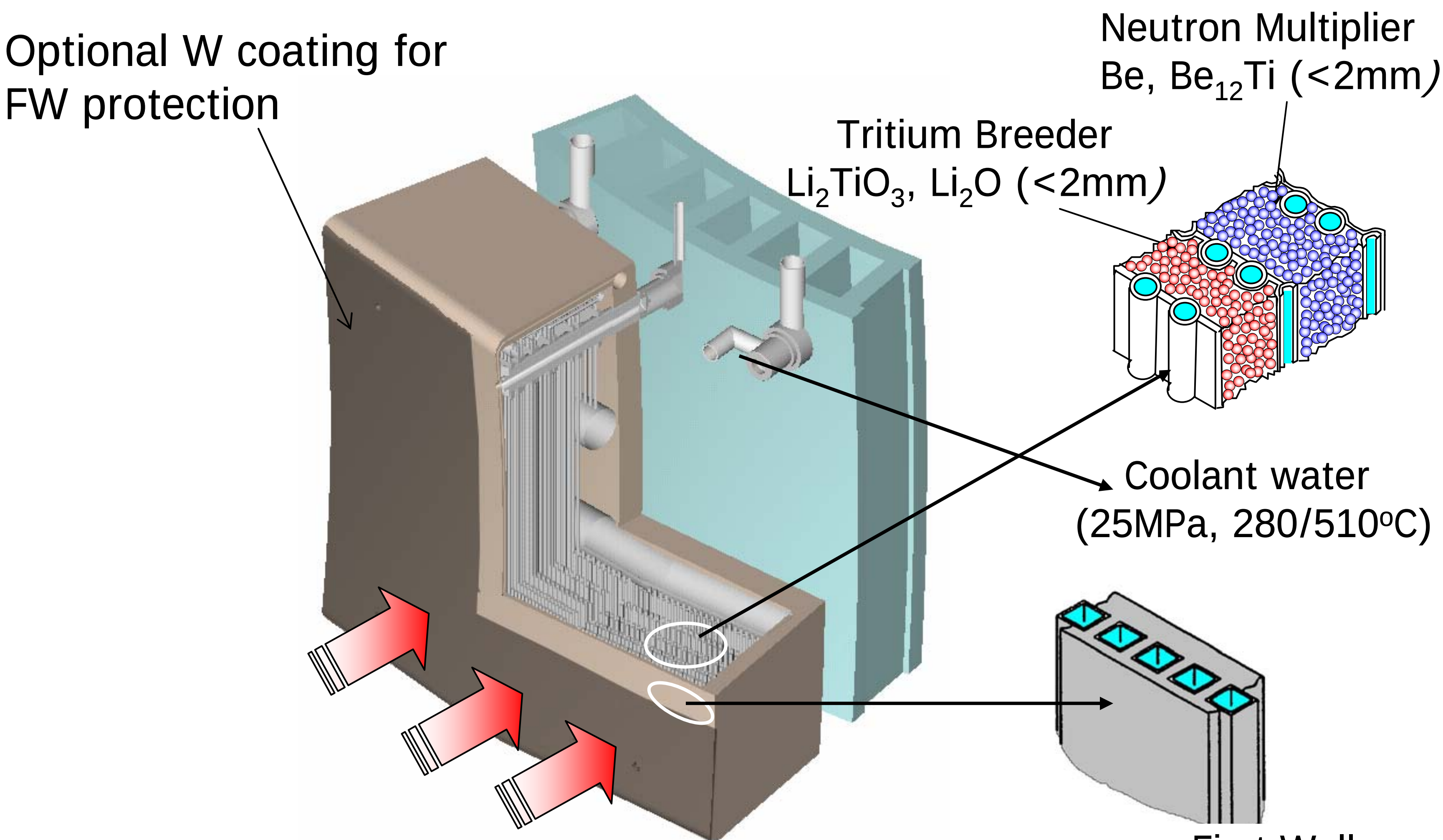
Material Functions

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



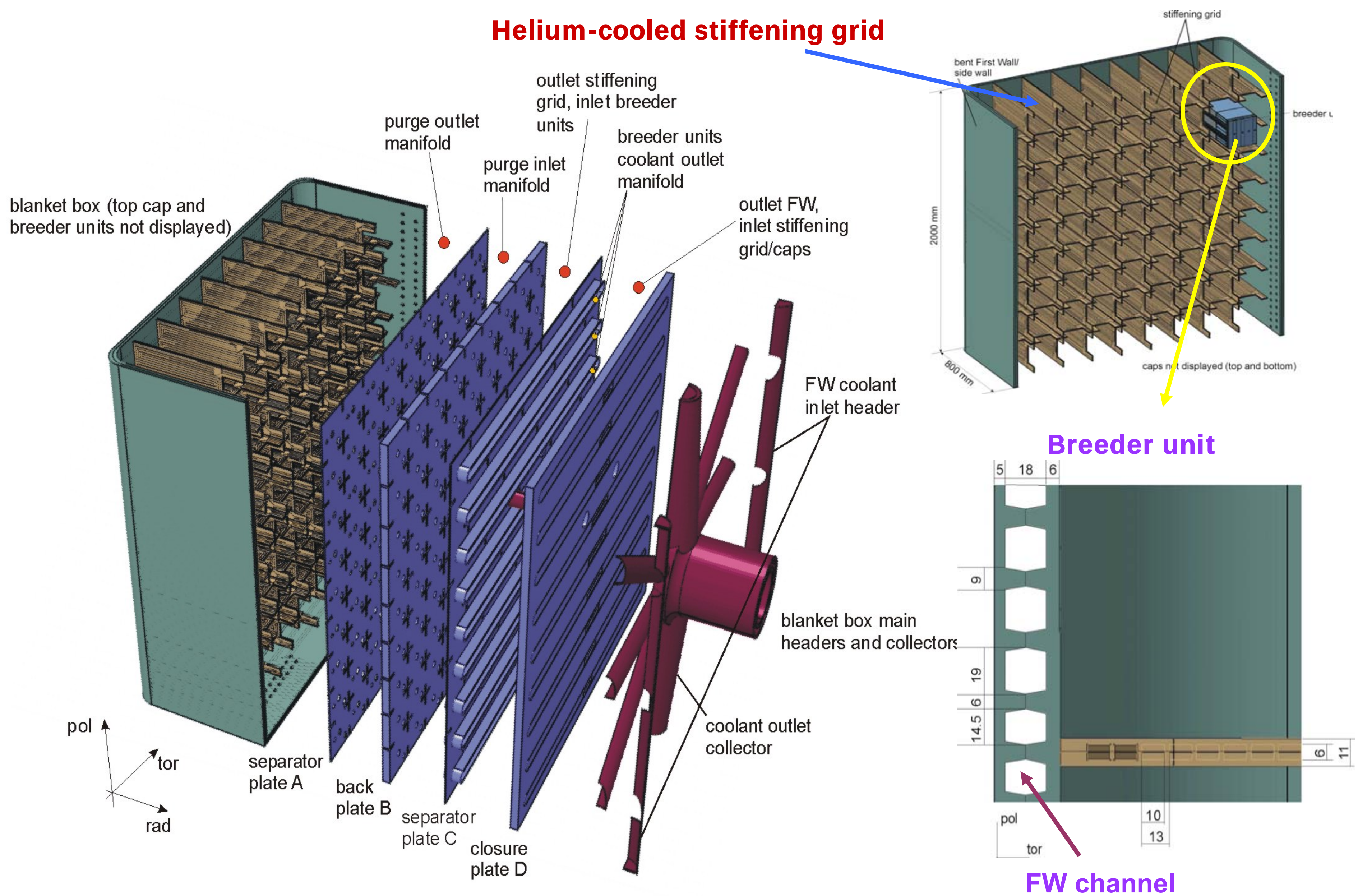
Several configurations exist (e.g. wall parallel or "head on" breeder/Be arrangements)

JA Water-Cooled Solid Breeder Blanket

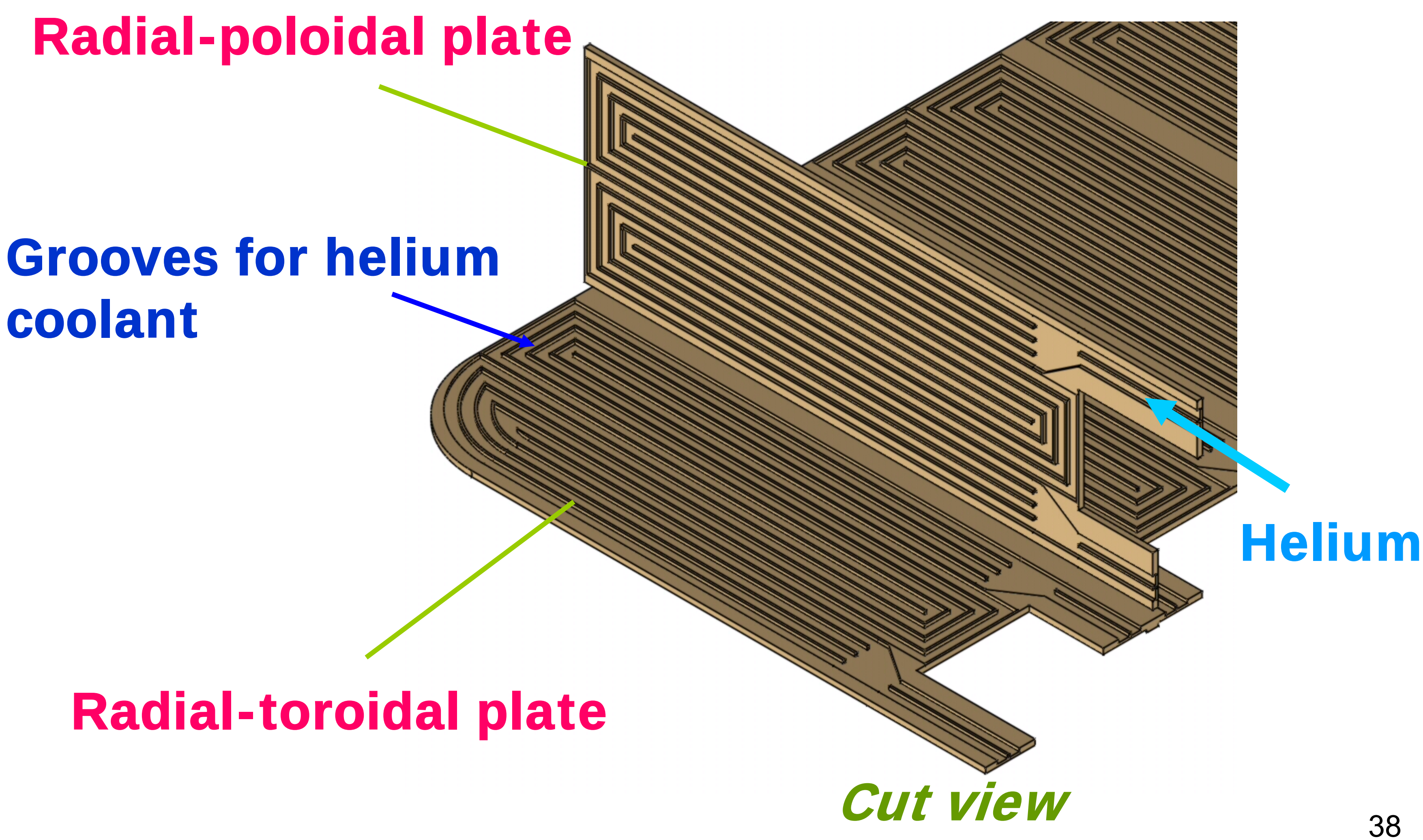


Surface Heat Flux:1 MW/m^2
Neutron Wall Load: 5 MW/m^2(1.5×10^{15}n/cm^{2}s)

Helium-Cooled Pebble Breeder Concept for EU



Stiffening plate provides the mechanical strength to the structural box



Solid Breeder Blanket Issues

- Tritium self-sufficiency
- Breeder/Multiplier/structure interactive effects under nuclear heating and irradiation
- Tritium inventory, recovery and control; development of tritium permeation barriers
- Effective thermal conductivity, interface thermal conductance, thermal control
- Allowable operating temperature window for breeder
- Failure modes, effects, and rates
- Mass transfer
- Temperature limits for structural materials and coolants
- Mechanical loads caused by major plasma disruption
- Response to off-normal conditions

Liquid Breeders

- Many liquid breeder concepts exist, all of which have key feasibility issues. Selection can not prudently be made before additional R&D results become available.
- Type of Liquid Breeder: Two different classes of materials with markedly different issues.

a) Liquid Metal: Li, ^{83}Pb ^{17}Li

High conductivity, low Pr number

Dominant issues: MHD, chemical reactivity for Li, tritium permeation for LiPb

b) Molten Salt: Flibe $(LiF)_n \cdot (BeF_2)$, Flinabe $(LiF\text{-}BeF_2\text{-}NaF)$

Low conductivity, high Pr number

Dominant Issues: Melting point, chemistry, tritium control

Liquid Breeder Blanket Concepts

1. Self-Cooled
   - Liquid breeder circulated at high speed to serve as coolant
   - Concepts: Li/V, Flibe/advanced ferritic, flinabe/FS
2. Separately Cooled
   - A separate coolant, typically helium, is used. The breeder is circulated at low speed 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 idea is to keep the temperature of the structure (ferritic steel) below 550ºC, and the interface temperature below 480ºC.
   - The liquid breeder is self-cooled; i.e., in the breeder region, the liquid serves as breeder and coolant. The temperature of the breeder can be kept higher than the structure temperature through design, leading to higher thermal efficiency.

Flows of electrically conducting coolants will experience complicated magnetohydrodynamic (MHD) effects

What is magnetohydrodynamics (MHD)?

- Motion of a conductor in a magnetic field produces an EMF that can induce current in the liquid. This must be added to Ohm's law:

$$\mathbf{j} = \sigma(\mathbf{E} + \mathbf{V} \times \mathbf{B})$$

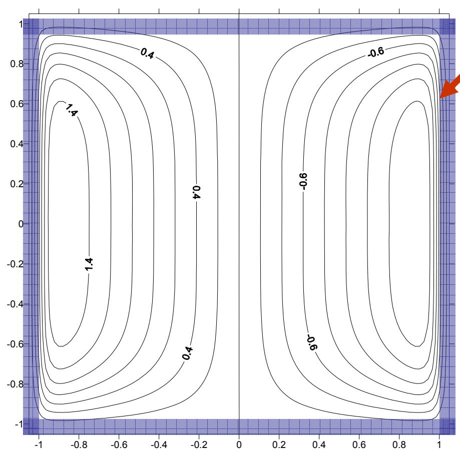
- Any induced current in the liquid results in an additional body force in the liquid that usually opposes the motion. This body force must be included in the Navier-Stokes equation of motion:

$$\frac{\partial \mathbf{V}}{\partial t} + (\mathbf{V} \cdot \nabla)\mathbf{V} = -\frac{1}{\rho}\nabla p + \nu\nabla^2\mathbf{V} + \mathbf{g} + \frac{1}{\rho}\mathbf{j} \times \mathbf{B}$$

- For liquid metal coolant, this body force can have dramatic impact on the flow: e.g. enormous MHD drag, highly distorted velocity profiles, non-uniform flow distribution, modified or suppressed turbulent fluctuations

Large MHD drag results in large MHD pressure drop

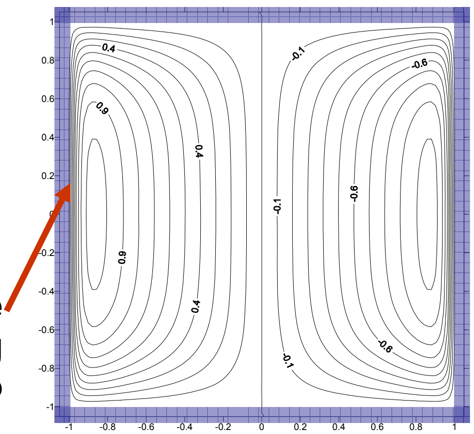
Conducting walls



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 drag from jxB force is zero

Insulated walls



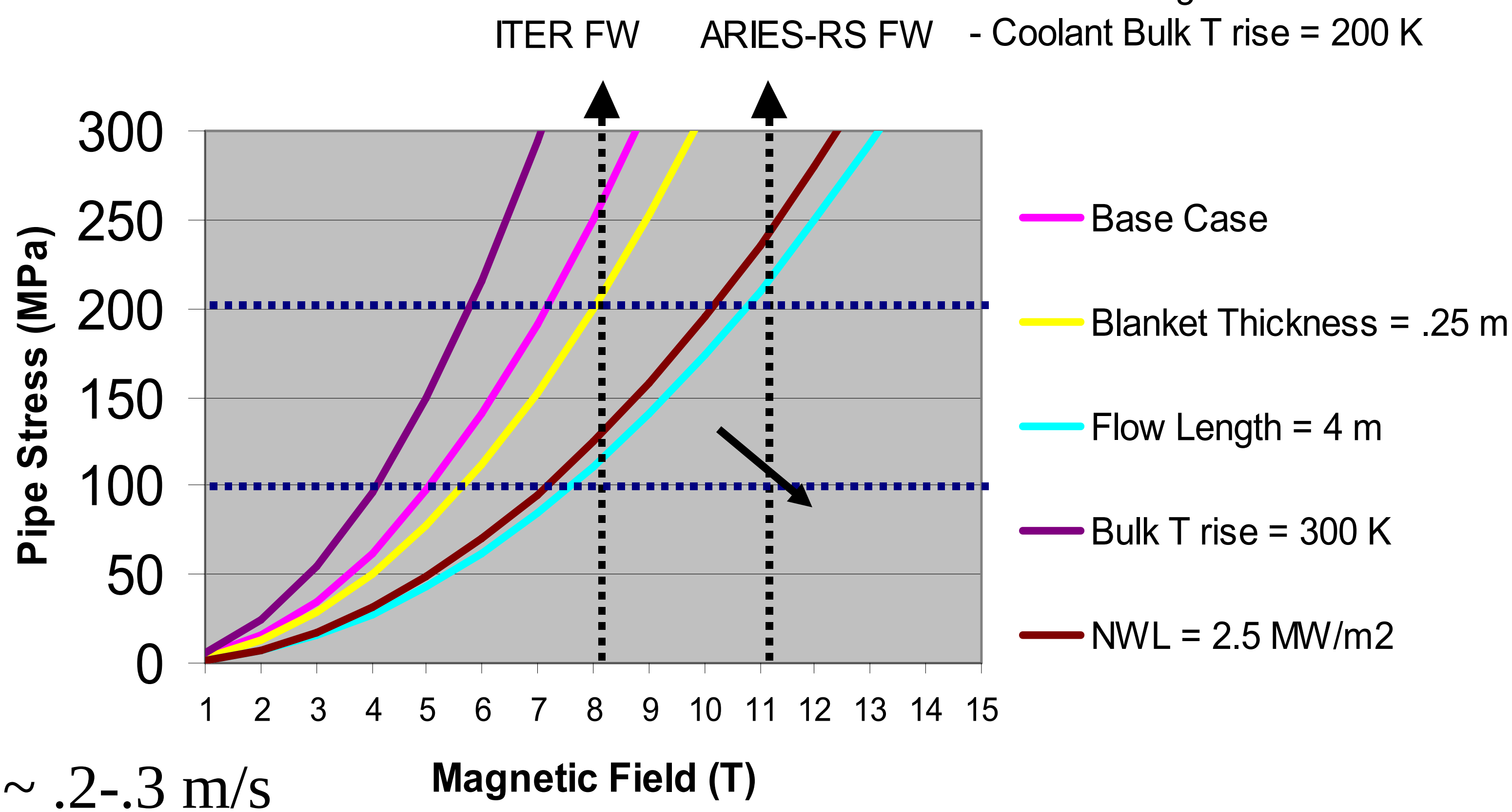
- Net JxB body force $\nabla p = c\sigma VB^2$ where $c = (t_w \sigma_w)/(a\ \sigma)$
- For high magnetic field and high speed (self-cooled LM concepts in inboard region) the pressure drop is large
- The resulting stresses on the wall exceed the allowable stress for candidate structural materials

- Perfect insulators make the net MHD body force zero
- But insulator coating crack tolerance is very low ($\sim 10^{-7}$).
  - It appears impossible 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)

LM-MHD pressure drop window for inboard channels is closed!

$$S = \frac{(\mathrm{NWL})L^2 B^2 \sigma_w}{\delta \rho c_p \Delta T}$$ (Sze, 1992)

Lithium Inboard Base Case
- NWL = 5 MW/m2
- Blanket Thickness = 0.2 m
- Blanket Length = 6.0 m
- Coolant Bulk T rise = 200 K



U ~ .2-.3 m/s

So a strategy is needed to reduce MHD pressure drop for liquid metals

$$\boxed{\Delta P_{MHD} = KL\sigma_l UB^2}$$

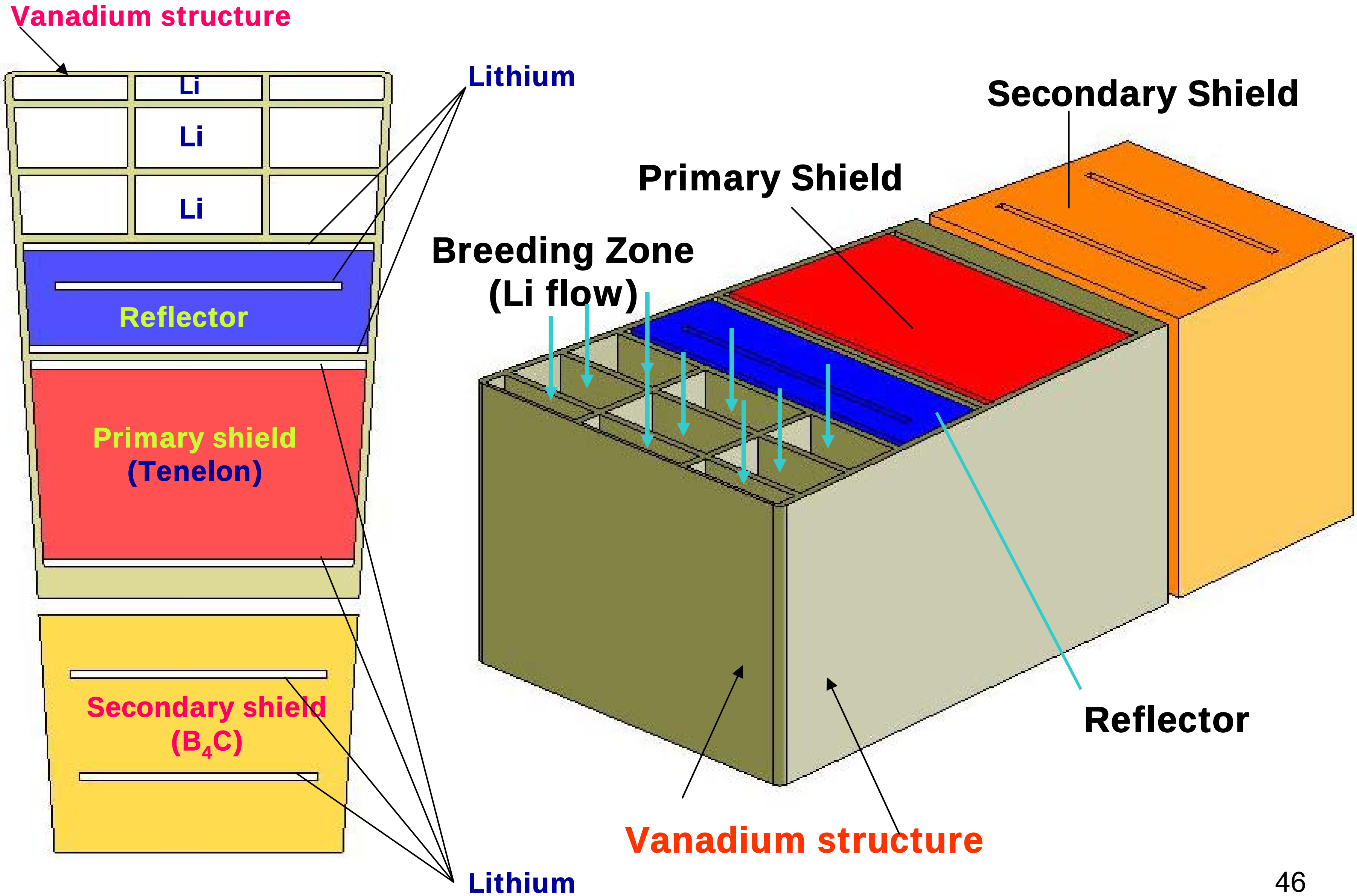
"K" factor represents a measure of relative conductance of induced current closure paths

- **Lower K**
  - Insulator coatings/Laminated walls
  - Flow channel inserts
  - Elongated channels with anchor links or other design solutions

Main options considered: Break electrical coupling to load bearing walls so pipe walls can be made thick for more strength without also increasing pressure drop!

- **Lower Velocity: U**
  - Heat transfer enhancement or dual/separate coolant to lower velocity required for first wall/breeder zone cooling
  - High temperature difference operation to lower mass flow
- **Lower Magnetic field and flow length: B,L**
  - Outboard blanket only, with poloidal segmentation
- **Lower electrical conductivity: σ (molten salt)**

Li/Vanadium Blanket Concept



Issues with the Lithium/Vanadium Concept

- Li/V was the U.S. choice for a long time, because of its perceived simplicity. But negative R&D results and lack of progress on serious feasibility issues have eliminated U.S. interest in this concept as a near-term option.

<u>Issues</u>

- Insulator
  - Insulator coating is required
  - Crack tolerance (10^{-7}) appears too low to be achievable in the fusion environment
  - "Self-healing" coatings can solve the problem, but none has yet been found (research is ongoing)
- Corrosion at high temperature (coupled to coating development)
  - Existing compatibility data are limited to maximum temperature of 550ºC and do not support the BCSS reported corrosion limit of 5µm/year at 650ºC
- Tritium recovery and control
- Vanadium alloy development is very costly and requires a very long time to complete

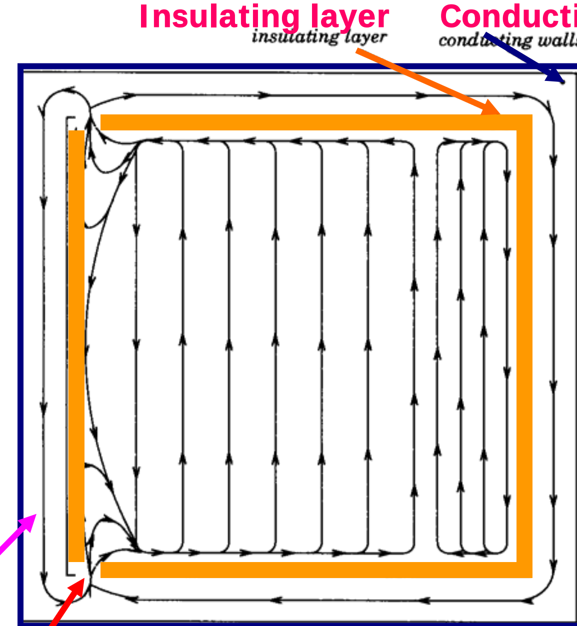
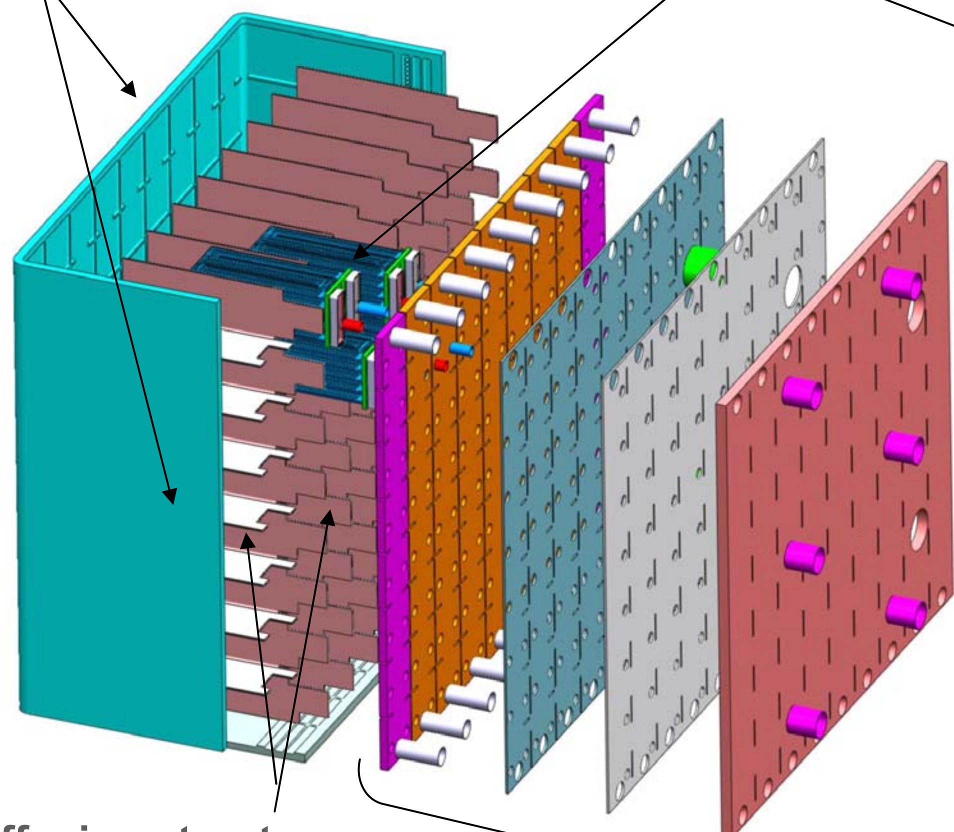


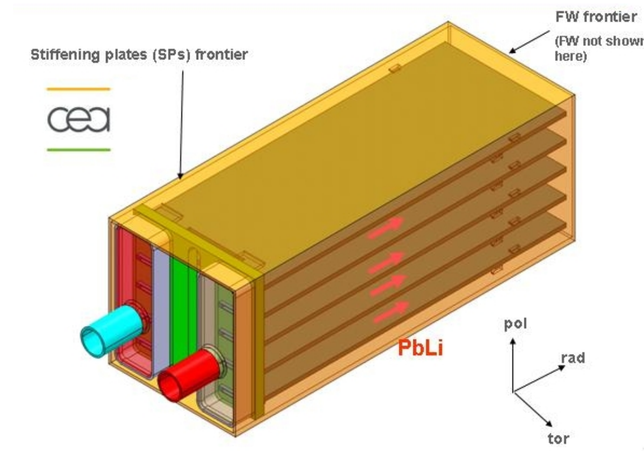
Fig. 8. Electric current lines for the case of two side-wall cracks near the corner.

EU – The Helium-Cooled Lead Lithium (HCLL) DEMO Blanket Concept

Module box
(container & surface heat flux extraction)

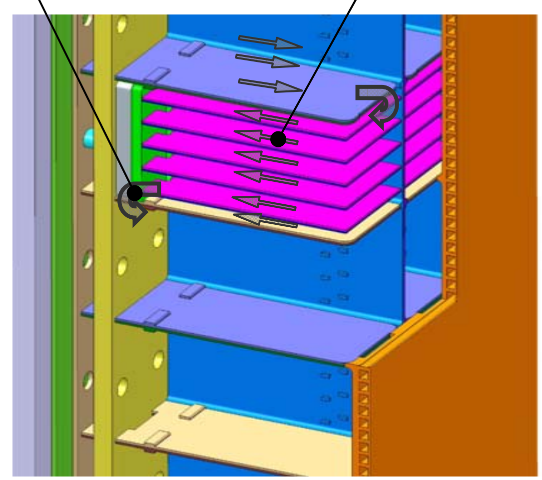
Breeder cooling unit (heat extraction from PbLi)





[0.5-1.5] mm/s

[18-54] mm/s



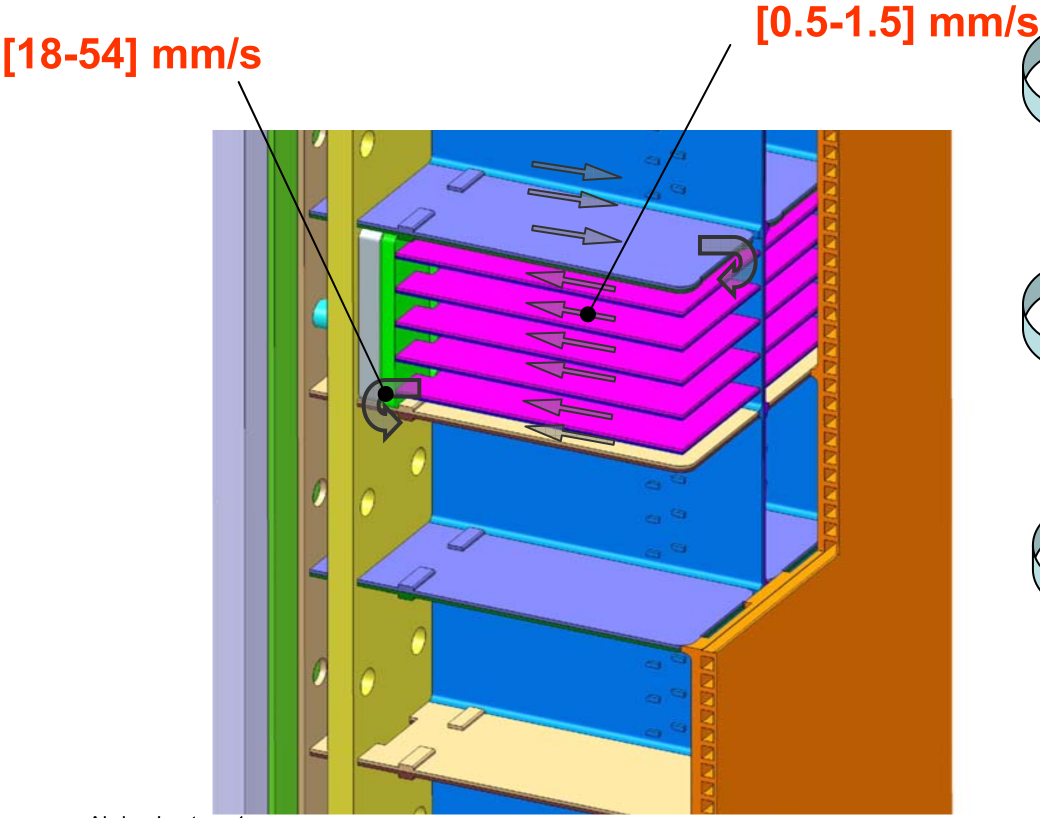
Stiffening structure
(resistance to accidental in-box pressurization i.e He leakage)

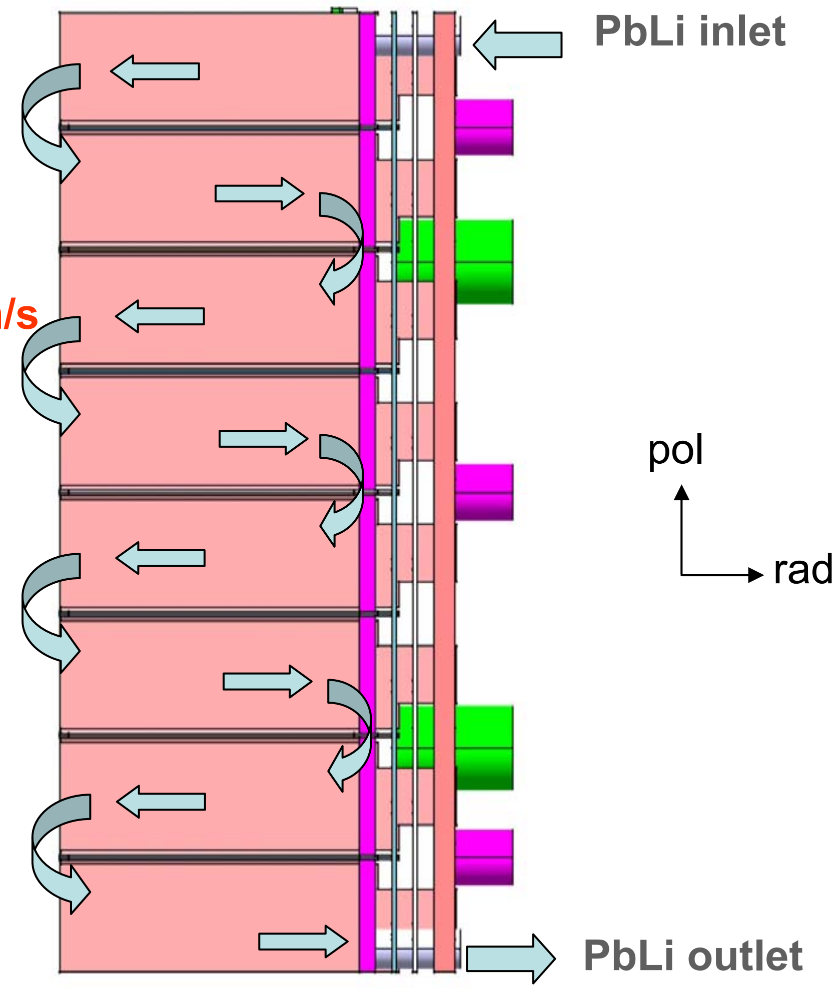
He collector system
(back)

HCLL PbLi flow scheme

He-Cooled PbLi Flow Scheme

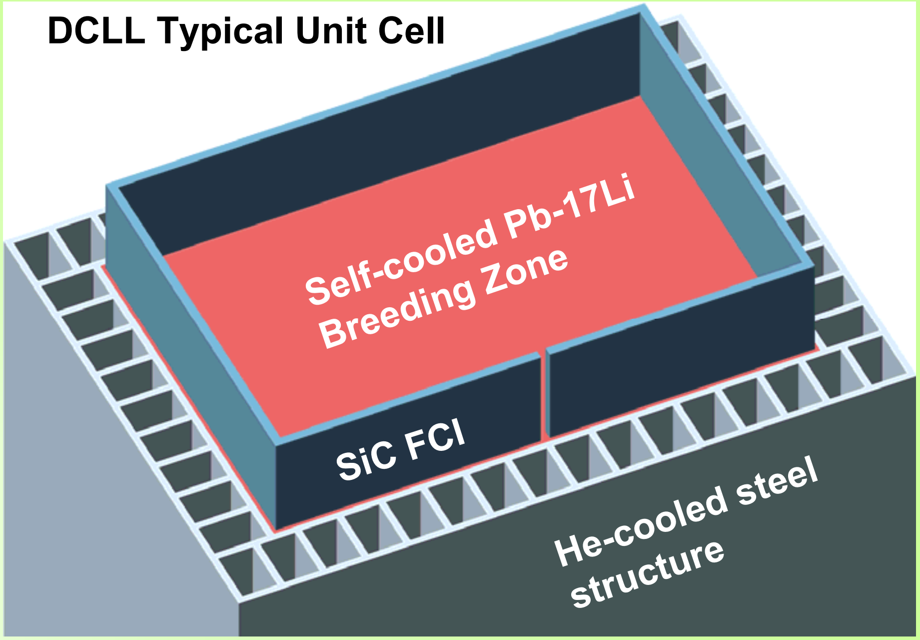
- PbLi is fed at the top and collected at the back
- Meandering PbLi flows in vertical columns delimited by vertical SPs
- Alternative flow holes at front/back of horizontal SPs





Pathway Toward Higher Temperature Through Innovative Designs with Current Structural Material (Ferritic Steel): *Dual Coolant Lead-Lithium (DCLL) FW/Blanket Concept*

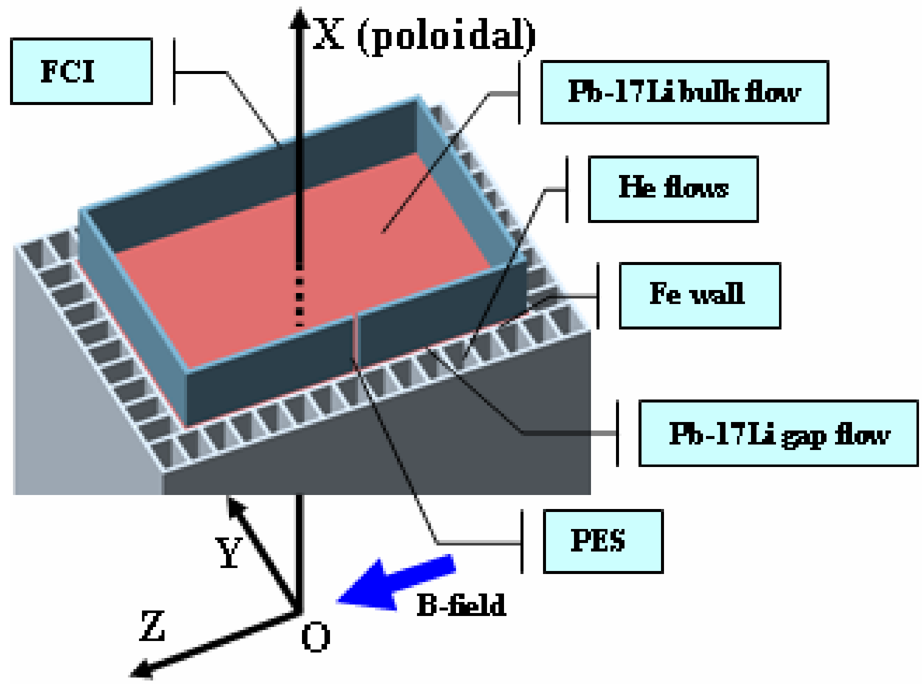
- First wall and ferritic steel structure cooled with helium
- Breeding zone is self-cooled
- Structure and Breeding zone are separated by SiCf/SiC composite *flow channel inserts (FCIs)* that
  - Provide thermal insulation to decouple PbLi bulk flow temperature from ferritic steel wall
  - Provide electrical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



Pb-17Li exit temperature can be significantly higher than the operating temperature of the steel structure ⇒ High Efficiency

WHAT IS FCI ?

- FCI (Flow Channel Insert) is the key element of the DCLL blanket concept
- Both ITER and DEMO
- Made of 5-10 mm SiC_f/SiC composite
- Pressure equalization openings (slot or holes) to nearly eliminate primary stress. Secondary (thermal) stress still exists
- The main functions are:
  - to reduce the MHD pressure drop (electrical insulation);
  - to reduce heat leakage into He (thermal insulation);
  - to separate hot PbLi (650°C) from Fe
- No serious feasibility issues have been identified yet. However tailoring SiC properties and fabrication of complex shape FCIs is still an issue.



Pb-17Li exit temperature can be significantly higher than the operating temperature of the steel structure => ***High Efficiency***

SiC$_f$/SiC FCI REQUIREMENTS

- **$\sigma_{\perp SiC}$=1-100 S/m**: 10^1-10^3 reduction of MHD pressure drop
- **$k_{\perp SiC}$=1-10 W/m-K**: heat leakage is <10% of the total power (DEMO)
- The optimal ($\sigma_{\perp SiC}$,$k_{\perp SiC}$)* is strongly dependent on the thermofluid MHD and should be determined by design tradeoffs, taking into account:
  - *ΔP (<1-2 MPa)*
  - *heat leakage (<10-15% of the total power)*
  - *temperature gradient (<150-200 K per 5 mm FCI)*
  - *PbLi-Fe interface temperature (<470-500°C)*
- Suggested (DEMO): $k_{\perp SiC}$~2 W/m-K; $\sigma_{\perp SiC}$~100 S/m
(S.Smolentsev, N.Morley, M.Abdou, *MHD and Thermal Issues of the SiCf/SiC FCI*, FST, July 2006)

* Only $k_{\perp}$ and $\sigma_{\perp}$ (across the FCI) are important

FCI RELATED R&D

Material science

- Development of low-conductivity grade 2-D woven SiC_f/SiC with a thin surface sealing layer to avoid soaking of PbLi into pores (*e.g.* using CVD)
- Improvement of crack resistance
- Reliable measurements of SiC_f/SiC properties at 300 to 800°C, including effect of irradiation
- Fabrication of complex shape FCIs with pressure equalization openings and overlap sections

Thermofluid MHD

- Effectiveness of FCI as electrical and thermal insulator
- Pressure equalization (slot or holes ?)
- Effect of FCI on flow balancing in normal and abnormal (cracked FCI) conditions
- Optimal location of the FCIs in the module

Key features of Dual Coolant Lead-Lithium Concept

(One of the primary concepts considered by the U.S. for ITER TBM)

- **Cool the ferritic steel FW and structure with separate coolant – He (also used for FW/blanket preheating and possible tritium control)**
  - The idea is to keep the structure temperature below 550ºC, the allowable temperature for ferritic steel, and the interface temperature below 480ºC
- **Breeding zone is self-cooled PbLi**
  - PbLi can be moving at slow velocity, since the heat generation rate in the breeding zone is lower than the surface heat flux at the FW
  - PbLi can be operated at temperatures higher than the structure, for higher thermodynamic efficiency
  - Some type of thermal/MHD insulator, e.g., FCIs, is required, but the requirements are more relaxed than for "all self-cooled" concepts
- **Use flow channel inserts (FCIs), wherever possible to:**
  - Provide electrical insulation to reduce MHD pressure drop
  - Provide thermal insulation to decouple PbLi bulk flow temperature from wall temperature
  - Provide permeation barrier to reduce T permeation into the He system
  - Provide additional corrosion resistance since only stagnant PbLi is in contact with the ferritic steel structural walls

Molten Salt Blanket Concepts

- Lithium-containing molten salts are used as the coolant for the Molten Salt Reactor Experiment (MSRE)
- Examples of molten salt are:
  - Flibe: $(LiF)_n \cdot (BeF_2)$
  - Flinabe: $(LiF\text{-}BeF_2\text{-}NaF)$
- The melting point for flibe is high (460ºC for n = 2, 380ºC for n = 1)
- Flinabe has a lower melting point (recent measurement at SNL gives about 300ºC)
- Flibe has low electrical conductivity, low thermal conductivity

Molten Salt Concepts: Advantages and Issues

<u>Advantages</u>

- Very low pressure operation
- Very low tritium solubility
- Low MHD interaction
- Relatively inert with air and water
- Pure material compatible with many structural materials
- Relatively low thermal conductivity allows dual coolant concept (high thermal efficiency) <u>without</u> the use of flow-channel inserts

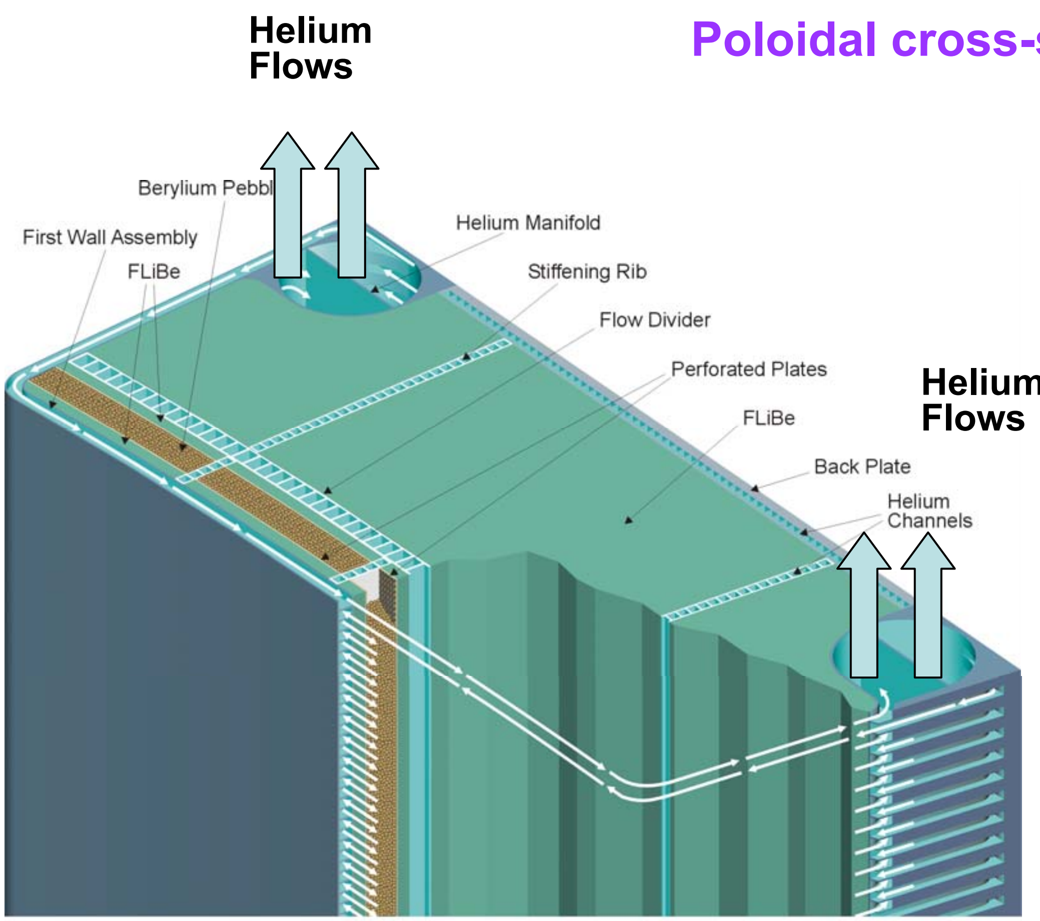
<u>Disadvantages</u>

- High melting temperature
- Need additional Be for tritium breeding
- Transmutation products may cause high corrosion
- Low tritium solubility means high tritium partial pressure (tritium control problem)
- Limited heat removal capability, unless operating at high Re (not an issue for dual-coolant concepts)

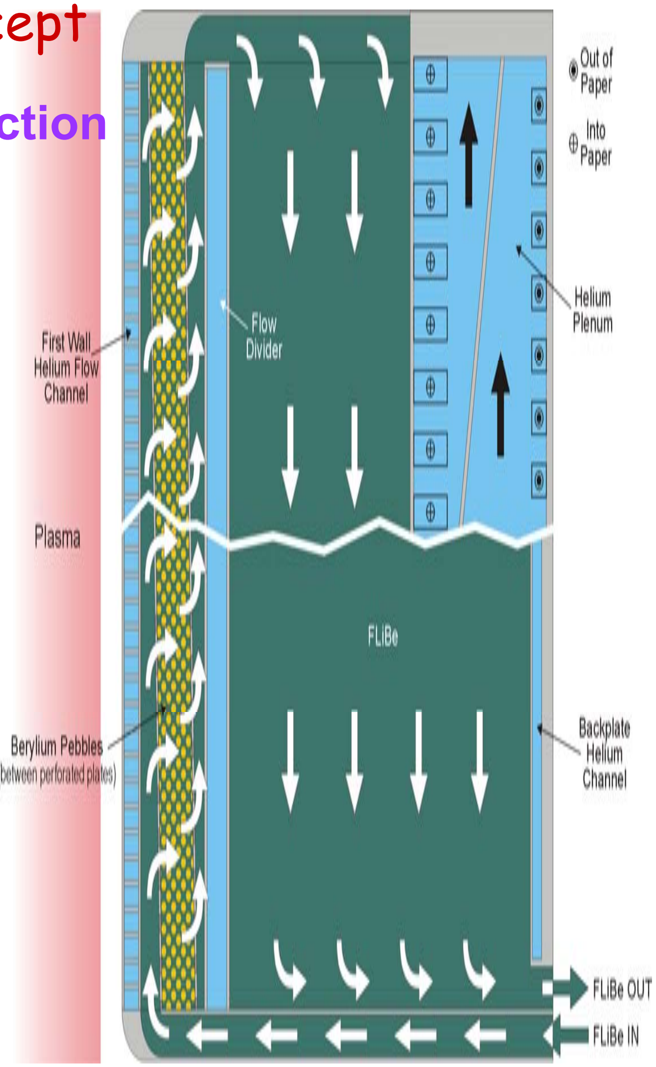
Dual Coolant Molten Salt Blanket Concepts

- He-cooled First Wall and structure
- Self-cooled breeding region with flibe or flinabe
- No flow-channel insert needed (because of lower conductivity)

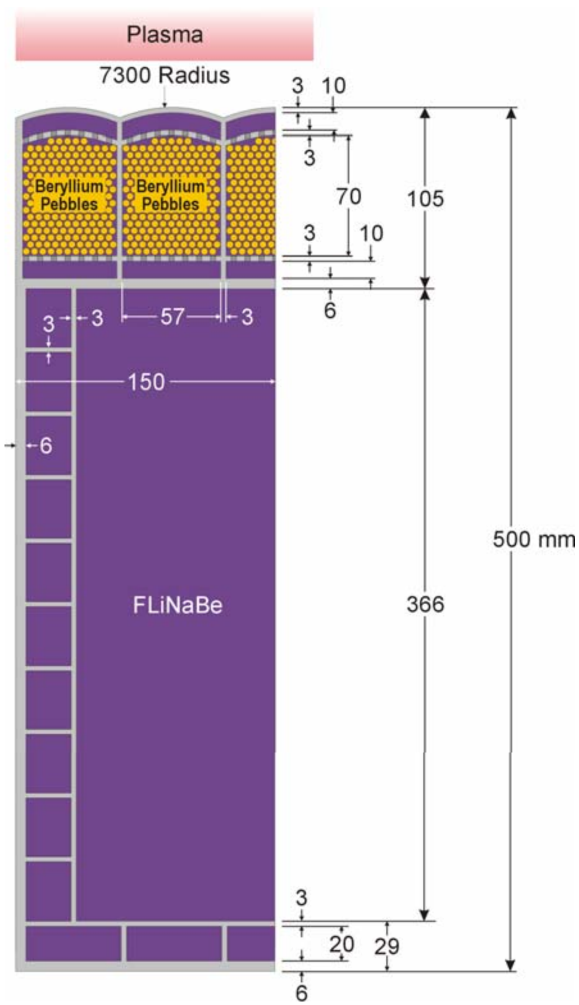
Example: Dual-Cooled FLiBe + Be Blanket Concept

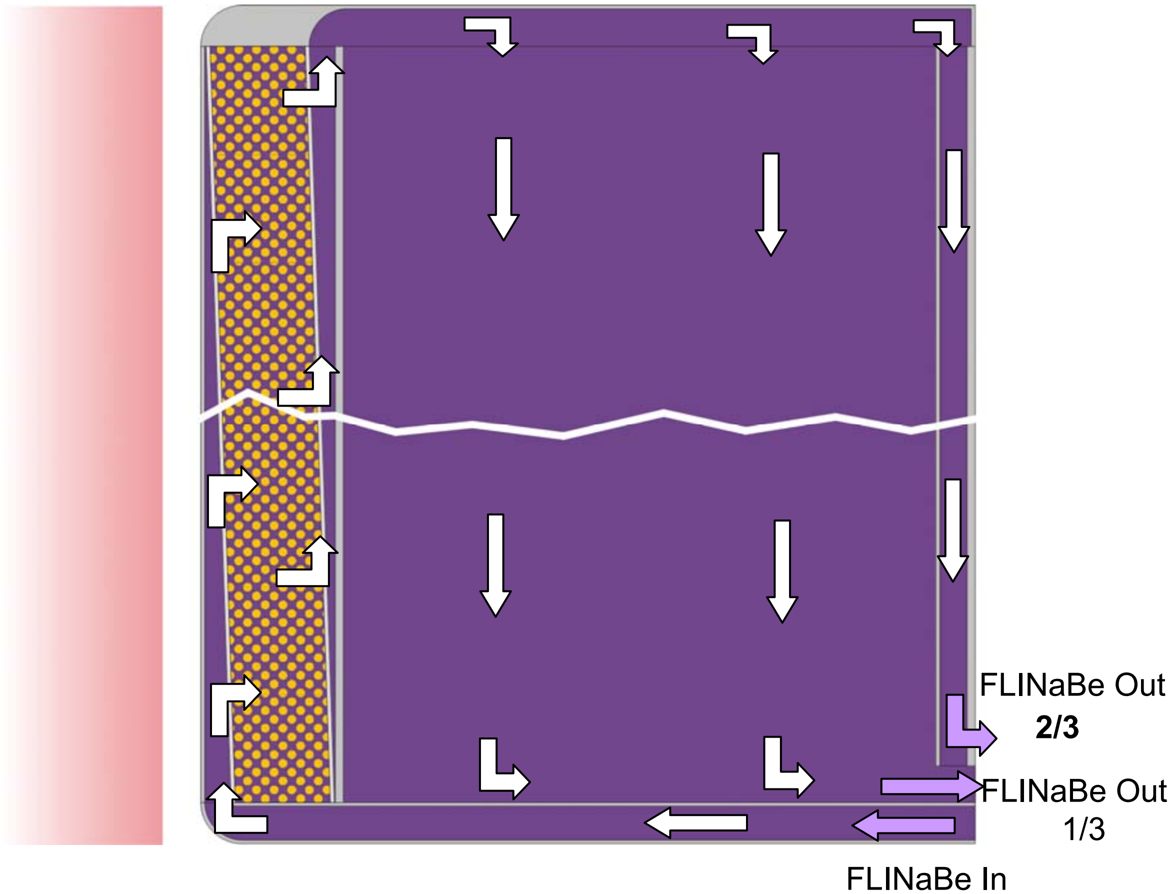


Poloidal cross-section



Self-cooled – FLiNaBe Design Concept
Radial Build and Flow Schematic





Issues and R&D on Liquid Metal Breeder Blankets

- Fabrication techniques for SiC Inserts
- MHD and thermalhydraulic experiments on SiC flow channel inserts with Pb-Li alloy
- Pb-Li and Helium loop technology and out-of-pile test facilities
- MHD-Computational Fluid Dynamics simulation
- Tritium permeation barriers
- Corrosion experiments
- Test modules design, fabrication with RAFS, preliminary testing
- Instrumentation for nuclear environment

Lessons learned: The most challenging problems in FNT are at the *INTERFACES*

- Examples:
  - **MHD insulators**
  - **Thermal insulators**
  - **Corrosion** (liquid/structure interface temperature limit)
  - **Tritium permeation**
- Research on these interfaces **must be done jointly by blanket and materials researchers**

Will the development of high-temperature structural material lead to more attractive blankets?

Not necessarily (unless we can solve the interface problems)

1. **Vanadium alloys** (high temperature capability)
   - V is compatible only with liquid lithium
   - Flowing liquid Li requires **MHD insulators**
   - Tolerable crack fraction is estimated to be very low (< 10^{-7})—much lower than can be achieved with real coatings
   - "Self-healing" coating R&D results are negative for non-isothermic systems
   - Laminated layer insulators (alternating layers of insulator and metallic protection layer) were proposed, but V layer needs to be too thin

2. **High-temperature advanced ferritic steels (ODS, NFS)**
   - May potentially operate up to ~700ºC (compared to 550ºC for EUROFER and F82H)
   - At present, we **cannot** utilize such advanced high-temperature ferritics
   - Li and LiPb interface temperature (T_{int}) is limited by corrosion to ~500ºC **Unless corrosion temperature limit is improved, EUROFER and F82H are satisfactory**
   - Solid breeder/structure interface cannot be increased much above 400–500ºC (to have adequate temperature window for T-release and TBR)

Tritium Control and Management

- Tritium control and management will be one of the most difficult issues for fusion energy development, both from the technical challenge and from the "public acceptance" points of view.
- Experts believe the T-control problem is underestimated (maybe even for ITER!)
- ***The T-control problem in perspective:***
  - ***The scale-up from present CANDU experience to ITER and DEMO is striking:***
    The quantity of tritium to be managed in the ITER fuel cycle is much larger than the quantities typically managed in CANDU (which represents the present-day state of practical knowledge).
  - ***The scale-up from ITER to DEMO is orders of magnitude:***
    The amount of tritium to be managed in a DEMO blanket (production rate ~400 g/day) is several orders of magnitude larger than that expected in ITER, **while the allowable T-releases could be comparable.**

For more details, see:

- W. Farabolini et al, "Tritium Control Modelling in an He-cooled PbLi Blanket…" paper in ISFNT-7 (this conference)
- Papers and IEA Reports by Sze, Giancarli, Tanaka, Konys, etc.

Why is Tritium Permeation a Problem?

- Most fusion blankets have high tritium **partial pressure**:

  LiPb = 0.014 Pa Flibe = 380 Pa
  He purge gas in solid breeders = 0.6 Pa

- The **temperature** of the blanket is high (500–700ºC)
- Surface area of heat exchanger is high, with thin walls
- Tritium is in **elementary form**

These are perfect conditions for tritium permeation.

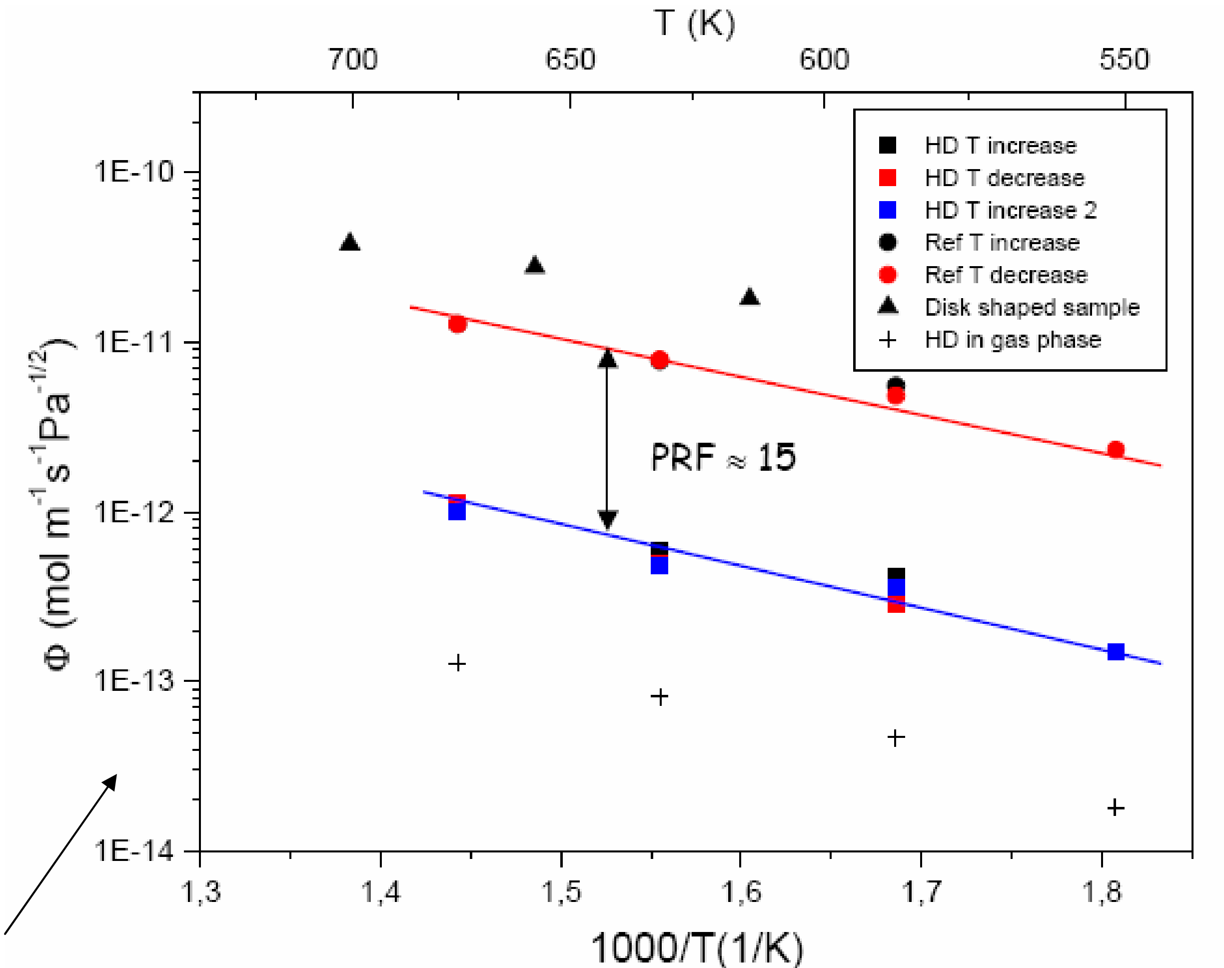
- The allowable tritium loss rate is very low (~10 Ci/day), requiring a partial pressure of ~10^{-9} Pa.

Challenging!

- *Even a tritium permeation barrier with a permeation reduction factor (PRF) of 100 may be still too far from solving this problem!*

Tritium Permeation Barrier Development in EU

- Tritium permeation barrier development is a key to tritium leakage and inventory control
- Development and tests of tritium permeation barrier coatings (up to 2003, in the EU) have **not yet been conclusive**



Comparisons of permeability of HDA (Hot Dip Aluminization) coated tubes in H_2 gas and Pb-17Li

Key R&D Items for Tritium Control

Test Blanket Modules (TBMs) in ITER (and DT operation in ITER) will give us the first quantitative real tests of the tritium control and management issue.

Key R&D required toward successful demonstration:

- **Sophisticated modeling tools capable of predicting the T-flows in different blanket system and reactor components**
  - accounting for complexities from geometric factors, temperature dependent properties, convection effects
- **Continue to develop high performance tritium diffusion barrier and clarify the still existing technological questions:**
  - understanding the sensitivity of the PRF to the quality of coating
  - crack tolerance and irradiation experiments on coatings
  - compatibility studies of coatings in flowing conditions at elevated temperatures
- **Continue to develop efficient tritium recovery system for both the primary and the secondary coolants**
  - efficiency to 99.99%
- **Develop instrument capable of detecting tritium on-line down to a very low concentration**

Reliability/Maintainability/Availability is one of the remaining "Grand Challenges" to Fusion Energy Development. Chamber Technology R&D is necessary to meet this Grand Challenge.

Need High Power Density/Physics-Technology Partnership

-High-Performance Plasma

-Chamber Technology Capabilities

Need Low Failure Rate

$$COE = \frac{C \cdot i + \boxed{\text{replacement cost}} + O\,\&\,M}{P_{fusion} \cdot \boxed{\text{Availability}} \cdot M \cdot \eta_{th}}$$

Energy Multiplication

Need High Temp. Energy Extraction

Need High Availability / Simpler Technological and Material Constraints

$$\frac{(1/\text{failure rate})}{1/\text{failure rate} + \text{replacement time}}$$

- **Need Low Failure Rate:**
  - Innovative Chamber Technology
- **Need Short Maintenance Time:**
  - Simple Configuration Confinement
  - Easier to Maintain Chamber Technology

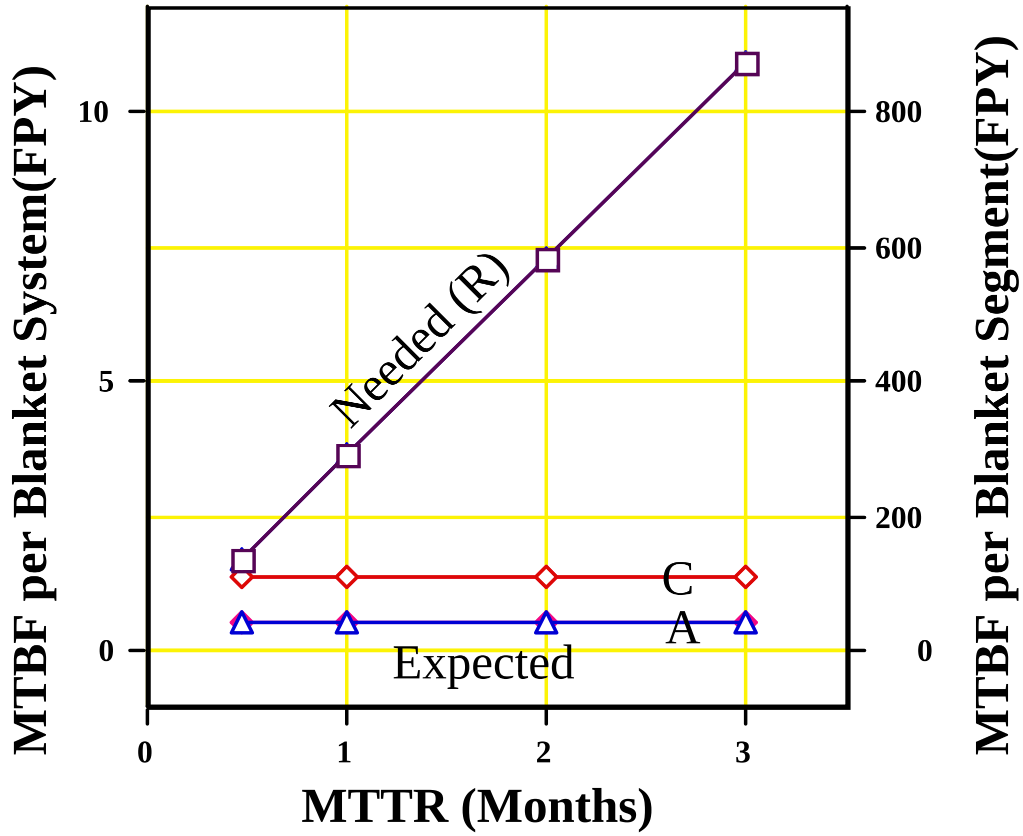
$$\text{Availability} = \frac{\text{MTBF}}{\text{MTBF} + \text{MTTR}}$$

Current plasma confinement schemes and configurations have:

- Relatively long MTTR (weeks to months)
  - ➔Required MTBF must be high
- Large first wall area
  - ➔Unit failure rate must be very low

    MTBF ~ 1/(area • unit failure rate)

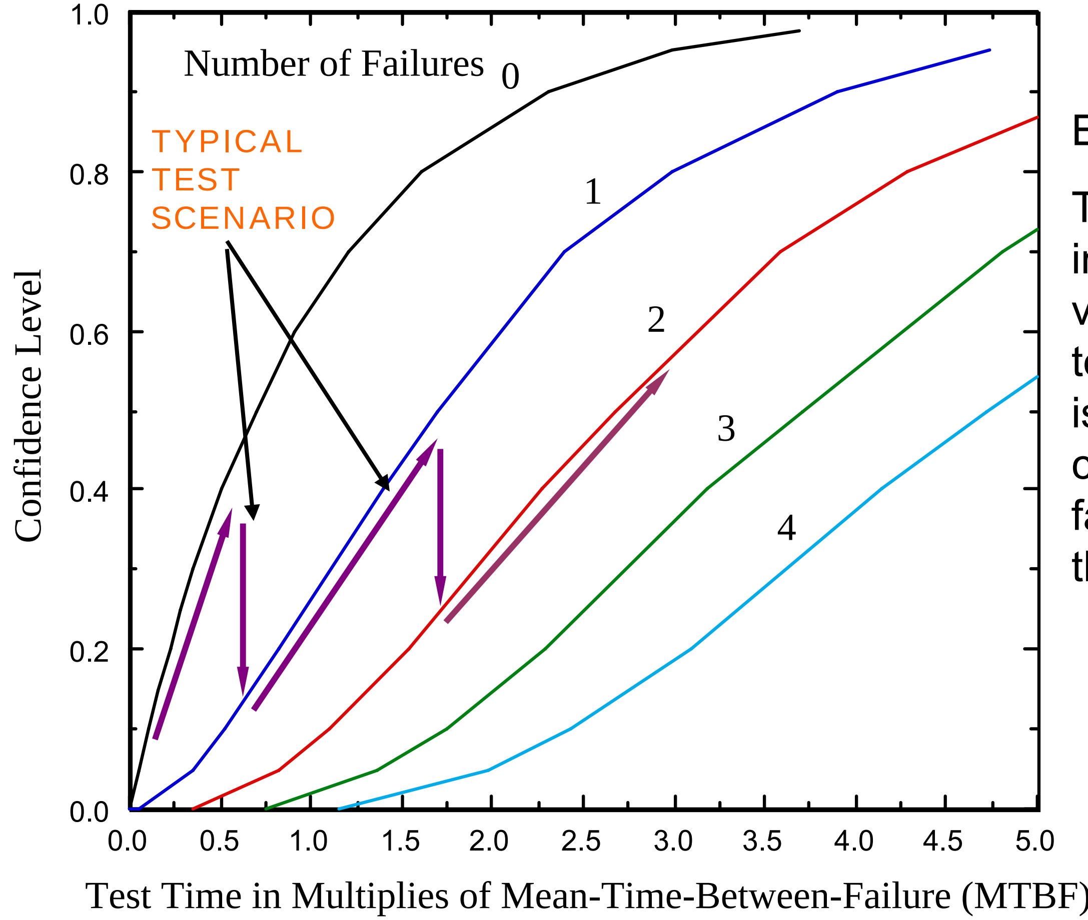
Reliability requirements are more demanding than for other non-fusion technologies



The **reliability requirements** on the Blanket/FW (in current confinement concepts that have long MTTR > 1 week) are **most challenging** and pose critical concerns. These must be seriously addressed as an integral part of the R&D pathway to DEMO. Impact on ITER is predicted to be serious. **It is a DRIVER for CTF.**

“Reliability Growth”

Upper statistical confidence level as a function of test time in multiples of MTBF for time terminated reliability tests (Poisson distribution). Results are given for different numbers of failures.



Example,

To get 80% confidence in achieving a particular value for MTBF, the total test time needed is about 3 MTBF (for case with only one failure occurring during the test).

Reference: **M. Abdou et. al., "FINESSE A Study of the Issues, Experiments and Facilities for Fusion Nuclear Technology Research & Development, Chapter 15 (Figure 15.2-2.) Reliability Development Testing Impact on Fusion Reactor Availability", Interim Report, Vol. IV, PPG-821, UCLA,1984. It originated from A. Coppola, "Bayesian Reliability Tests are Practical", RADC-TR-81-106, July 1981.**

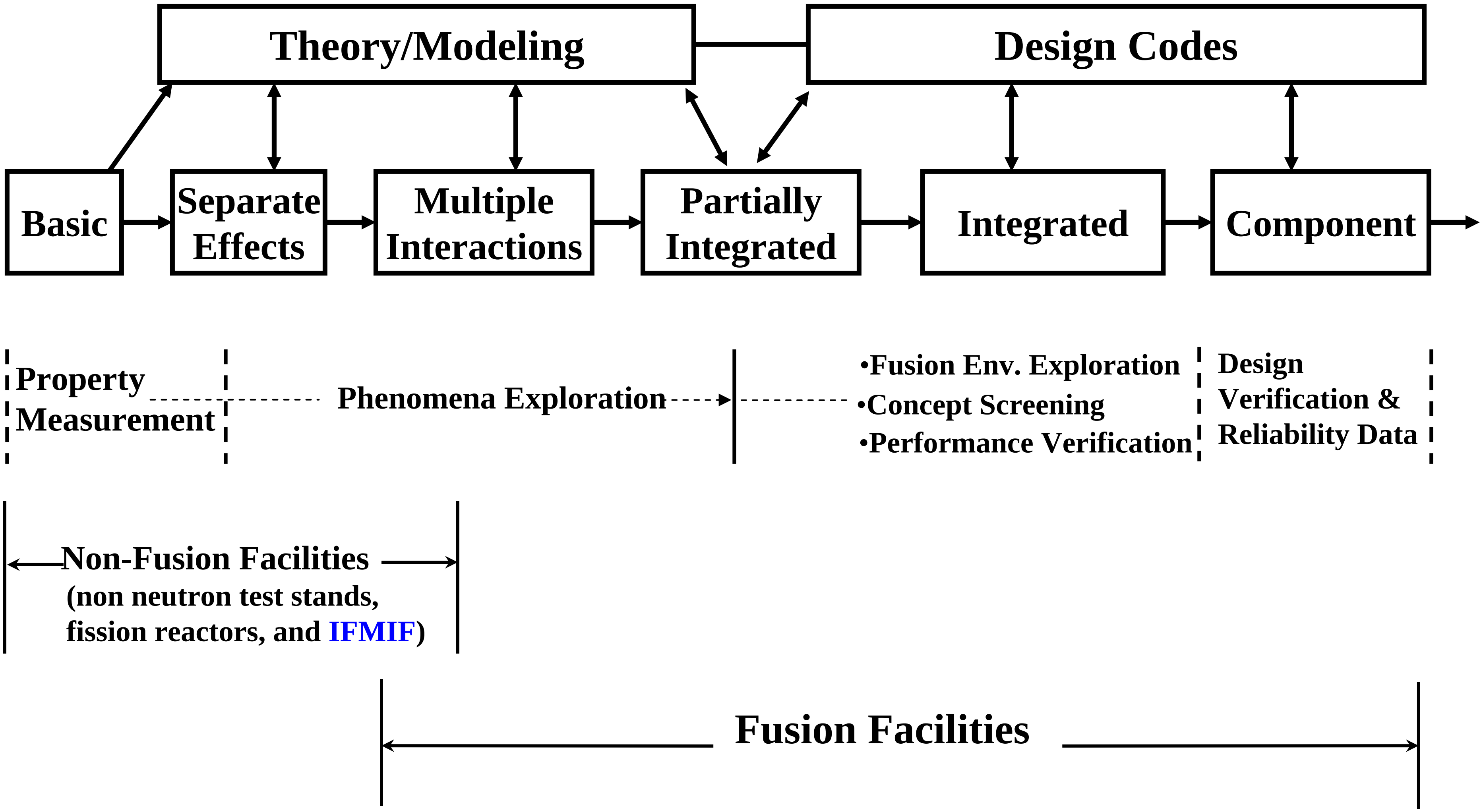
Summary of Critical R&D Issues for Fusion Nuclear Technology

1. D-T fuel cycle **tritium self-sufficiency** in a practical system

   depends on many physics and engineering parameters / details: e.g. fractional burn-up in plasma, tritium inventories, FW thickness, penetrations, passive coils, etc.

2. **Tritium extraction and inventory** in the solid/liquid breeders under actual operating conditions
3. **Thermomechanical** loadings and response of blanket and PFC components under normal and off-normal operation
4. Materials **interactions and compatibility**
5. Identification and characterization of **failure modes, effects, and rates** in blankets and PFC's
6. Engineering feasibility and reliability of electric (MHD) **insulators** and **tritium permeation barriers** under thermal / mechanical / electrical / magnetic / nuclear loadings with high temperature and stress gradients
7. **Tritium permeation, control and inventory** in blanket and PFC
8. **Lifetime** of blanket, PFC, and other FNT components
9. **Remote maintenance** with acceptable machine shutdown time

Types and roles of experiments, facilities, and modeling for FNT



- Non fusion facilities (e.g. fission reactors and IFMIF) have useful but limited roles
- Fusion testing facilities are NECESSARY for multiple interactions, partially integrated, integrated, and component tests

Key Fusion Environmental Conditions for Testing Fusion Nuclear Components

Neutrons (fluence, spectrum, spatial and temporal gradients)

- Radiation Effects
  (at relevant temperatures, stresses, loading conditions)
- Bulk Heating
- Tritium Production
- Activation

Heat Sources (magnitude, gradient)

- Bulk (from neutrons)
- Surface

Particle Flux (energy and density, gradients)

Magnetic Field (3-component with gradients)

- Steady Field
- Time-Varying Field

Mechanical Forces

- Normal
- Off-Normal

Thermal / Chemical / Mechanical / Electrical / Magnetic Interactions

Synergistic Effects

- Combined environmental loading conditions
- Interactions among physical elements of components

Stages of FNT Testing in Fusion Facilities

	Fusion "Break-in"	Design Concept & Performance Verification	Component Engineering Development & Reliability Growth	DEMO
Stage:	**I**	**II**	**III**	
Required Fluence ($MW\text{-}y/m^2$)	**~ 0.3**	**1 - 3**	**> 4 - 6**	
Size of Test Article	**Sub-Modules**	**Modules**	**Modules / Sectors**	

I — Fusion "Break-in"

- Initial exploration of performance in a fusion environment
- Calibrate non-fusion tests
- Effects of rapid changes in properties in early life
- Initial check of codes and data
- Develop experimental techniques and test instrumentation
- Narrow material combination and design concepts
- 10-20 test campaigns, each is 1-2 weeks

II — Design Concept & Performance Verification

- Tests for basic functions and phenomena (tritium release / recovery, etc.), interactions of materials, configurations
- Verify performance beyond beginning of life and until changes in properties become small (changes are substantial up to ~ 1-2 $MW \cdot y/m^2$)
- Data on initial failure modes and effects
- Establish **engineering feasibility** of blankets (satisfy basic functions & performance, 10 to 20% of lifetime)
- Select 2 or 3 concepts for further development

III — Component Engineering Development & Reliability Growth

- Identify failure modes and effects
- Iterative design / test / fail / analyze / improve programs aimed at improving reliability and safety
- Failure rate data: Develop a data base sufficient to predict mean-time-between-failure with sufficient confidence
- Obtain data to predict mean-time-to-replace (MTTR) for both planned outage and random failure
- Develop a data base to predict overall availability of FNT components in DEMO

FNT Requirements for Major Parameters for Testing in Fusion Facilities with Emphasis on Testing Needs to Construct DEMO Blanket

- These requirements have been extensively studied over the past 20 years, and they have been agreed to internationally (FINESSE, ITER Blanket Testing Working Group, IEA-VNS, etc.)
- Many Journal Papers have been published (>35)
- Below is the Table from the IEA-VNS Study Paper (Fusion Technology, Vol. 29, Jan 96)

Parameter	Value
Neutron wall load[a] (MW/m^2)	1 to 2
Plasma mode of operation	Steady State[b]
Minimum COT (periods with 100% availability) (weeks)	1 to 2
Neutron fluence at test module ($MW \cdot y/m^2$)	
Stage I: initial fusion break-in	0.3
Stage II: concept performance verification (engineering feasibility)	1 to 3
Stage III[c]: component engineering development and reliability growth	4 to 6[c]
Total neutron fluence for test device ($MW \cdot y/m^2$)	>6
Total test area (m^2)	>10
Total test volume (m^3)	>5
Magnetic field strength (T)	>4

a - Prototypical surface heat flux (exposure of first wall to plasma is critical)

b - If steady state is unattainable, the alternative is long plasma burn with plasma duty cycle >80%

c - Note that the fluence is not an accumulated fluence on “the same test article”; rather it is derived from testing “time” on “successive” test articles dictated by “reliability growth” requirements

Type of Integrated Facility Needed for FNT Development (blanket/FW, PFC, materials, tritium, safety) in addition to ITER

- Need fusion environment, i.e., plasma-based facility
- Testing requirements are: (see IEA study)
  - NWL > 1 MW/m^2, steady state, test area ~ 10 m^2, test volume ~ 5 m^3
  - Fluence requirements > 6 $MW•y/m^2$
    (engineering feasibility: 1–3 $MW•y/m^2$; reliability growth > 4 $MW•y/m^2$)
- What is needed is:
  small power < 100 MW fusion power
  long fluence > 6 $MW•y/m^2$ (for reliability growth testing)
- There is no external supply of tritium to run a large-power device such as ITER for such fluences
- **A device with small fusion power (~100MW) and moderate wall load (~1 MW/m^2) is a driven-plasma device (Q~2) and is most suitable for FNT testing. This is often called VNS or CTF.**

FNT Testing in Fusion Facilities

- ITER operation and conducting the Test Blanket Module (TBM) Program in ITER will provide the first real experimental results on the performance and issues of FNT components and materials in the integrated fusion environment.
- **But are ITER tests sufficient to proceed to DEMO?**
  - Technical studies say **other fusion test facilities (e.g. VNS/CTF) are necessary.**
  - But official plans of some Parties still show ITER as the only fusion facility from now to DEMO.
- Probably the biggest issue the International Community needs to seriously address is **how many major fusion FNT experimental devices are needed** to develop practical fusion energy by the middle of the century.

Other important topics for which no time is available to discuss in detail in this lecture

There are many other important issues in Fusion Nuclear Technology Development that have not been discussed in detail in this lecture because of time constraints, e.g.:

- Reliability/Maintainability/Availability challenging requirements and DEMANDS on testing

- FNT testing requirements in fusion facilities
- Limitations of ITER for FNT testing
- Need for another low fusion-power (<100MW) facility to test and develop FNT consistent with realistic constraints on world tritium supply. It has a low plasma Q and operates steady state. It is called Volumetric Neutron Source (VNS) , and more recently Component Development Facility (CTF)

- The slides in the appendix are limited examples. See Papers and Presentations on the web site www.fusion.ucla.edu/abdou

APPENDIX

Reliability / Maintainability / Availability
Critical Development Issues

$$Unavailability = U(total) = U(scheduled) + U(unscheduled)$$

This you design for

This can kill your DEMO and your future

Scheduled Outage:

Planned outage (e.g. scheduled maintenance of components, scheduled replacement of components, e.g. first wall at the end of life, etc.).

This tends to be manageable because you can plan scheduled maintenance / replacement operations to occur simultaneously in the same time period.

Unscheduled Outage: (This is a very challenging problem)

Failures do occur in any engineering system. Since they are random they tend to have the most serious impact on availability.

This is why "reliability/availability analysis," reliability testing, and "reliability growth" programs are key elements in any engineering development.

Availability (Due to Unscheduled Events)

Availability: $= \dfrac{1}{1 + \sum_i \text{Outage Risk}}$ i represents a component

$(\text{Outage Risk})_i = (\text{failure rate})_i \bullet (\text{mean time to repair})_i = \dfrac{MTTR_i}{MTBF_i}$

MTBF = mean time between failures = 1/failure rate

MTTR = mean time to repair

- **A Practical Engineering System must have:**

1. **Long MTBF**: have sufficient reliability

   - MTBF depends on reliability of components.

     One can estimate what MTBF is NEEDED from "availability allocation models" for a given availability goal and for given (assumed) MTTR. **But predicting what MTBF is ACHIEVEABLE requires real data from integrated tests in the fusion environment.**

2. **Short MTTR**: be able to recover from failure in a short time

   - MTTR depends on the complexity and characteristics of the system (e.g. confinement configurations, component blanket design and configuration, nature of failure). Can estimate, but need to demonstrate MTTR in fusion test facility.

An Example Illustration of Achieving a Demo Availability of 49%

(Table based on information from J. Sheffield's memo to the Dev Path Panel)

Component	Num ber	Failure rate in hr^{-1}	MTBF in years	MTTR for Major failure, hr	MTTR for Minor failure, hr	Fraction of failures that are Major	Outage Risk	Component Availability
Toroidal Coils	16	5×10^{-6}	**23**	10^4	240	0.1	**0.098**	0.91
Poloidal Coils	8	5×10^{-6}	**23**	5×10^3	240	0.1	**0.025**	0.97
Magnet supplies	4	1×10^{-4}	**1.14**	72	10	0.1	**0.007**	0.99
Cryogenics	2	2×10^{-4}	**0.57**	300	24	0.1	**0.022**	0.978
Blanket	100	1×10^{-5}	**11.4**	800	100	0.05	**0.135**	0.881
Divertor	32	2×10^{-5}	**5.7**	500	200	0.1	**0.147**	0.871
Htg/CD	4	2×10^{-4}	**0.57**	500	20	0.3	**0.131**	0.884
Fueling	1	3×10^{-5}	**3.8**	72	--	1.0	**0.002**	0.998
Tritium System	1	1×10^{-4}	**1.14**	180	24	0.1	**0.005**	0.995
Vacuum	3	5×10^{-5}	**2.28**	72	6	0.1	**0.002**	0.998
Conventional equipment- instrumentation, cooling, turbines, electrical plant ---							**0.05**	0.952
TOTAL SYSTEM							**0.624**	0.615

Assuming 0.2 as a fraction of year scheduled for regular maintenance.

Availability = 0.8* [1/(1+0.624)] = 0.49

Reliability/Availability is a challenge to fusion, particularly blanket/PFC, development

- Fusion System has many major components (TFC, PFC, plasma heating, vacuum vessel, blanket, divertor, tritium system, fueling, etc.)
  - Each component is required to have high availability
- All systems except the reactor core (blanket/PFC) will have reliability data from ITER and other facilities
- There is NO data for blanket/PFC (we do not even know if any present blanket concept is feasible)
- Estimates using available data from fission and aerospace for unit failure rates and using the surface area of a tokamak show:

PROBABLE MTBF for Blanket ~ 0.01 to 0.2 yr

compared to REQUIRED MTBF of many years

Aggressive "Reliability Growth" Program

We must have an aggressive "reliability growth" program for the blanket / PFC (beyond demonstrating engineering feasibility)

1) All new technologies go through a reliability growth program

2) Must be "aggressive" because extrapolation from other technologies (e.g. fission) strongly indicates we have a serious CHALLENGE

ITER Provides the First Integrated Experimental Conditions for Fusion Technology Testing

- Simulation of all Environmental Conditions

Neutrons	Plasma Particles
Electromagnetics	Tritium
Vacuum	Synergistic Effects

- Correct Neutron Spectrum (heating profile)
- Large Volume of Test Vehicle
- Large Total Volume, Surface Area of Test Matrix

ITER Blanket Testing is Essential to:

- **Achieve a key element of the "ITER Mission"**
- **Achieve the most critical milestone in blanket and material research: testing in the integrated fusion environment**

  (ITER construction and operation is for the next 30 years. Without such fusion testing, material and blanket research loses "focus", relevance: Why are we doing any research in these areas then?)
- **Develop the technology necessary to install breeding capabilities to supply ITER with tritium for its extended phase of operation**
- **Resolve the critical "tritium supply" issue for fusion development**

But testing TBM in ITER alone is not sufficient. Need VNS/CTF for full FNT development prior to DEMO

ITER (FEAT) Parameters Relevant to Blanket Testing

Overall Schedule

- 10 yr construction
- H and D operation: 4 yr
- DT operation (First DT Plasma Phase): 6 yr

Parameters

Neutron Wall Load: 0.55 MW/m^2
Plasma Burn Time: 400 s
Plasma Dwell Time: 1200 s
Plasma Duty Cycle: 0.25
Neutron Fluence: ~ 0.1 $MW{\bullet}y/m^2$

Note: "possibility of second DT Phase will be decided following a review of results of first 10 yr operation"

ITER-FEAT as designed can NOT perform many of the important tests for FNT

- ITER (FEAT) has been designed as a burning plasma experiment
  - The changes in design from ITER-EDA to ITER-FEAT have reduced the usefulness of ITER facility for important FNT tests

- ITER-FEAT Parameters do not satisfy many of the FNT testing requirements:
  - Neutron wall load: 0.55 MW/m^2 (compared to required 1-2 MW/m^2)
  - Neutron fluence: 0.1 $MW \cdot y/m^2$ (compared to required >6 $MW \cdot y/m^2$)
  - Plasma duty cycle makes it impossible to adequately perform the FNT testing mission
    - FNT testing requires steady state (or at least plasma duty cycle > 80%)
    - ITER-FEAT has short plasma burn (400S), long dwell time (1200S) resulting in a plasma duty cycle of 25%
    - The ITER-FEAT short burn/long dwell plasma cycle does not enable temperature equilibrium in test modules, a fundamental requirement for many tests (most FNT tests are highly temperature dependent)

Mode of Plasma Operation is critical to Most of FNT Testing

- This issue was investigated extensively in several studies including the ITER Test Blanket Working Group in both ITER-CDA and ITER-EDA, IEA-VNS. The conclusion reached: need steady state (or if unattainable, long burn/short dwell with plasma duty cycle >80%).
- Extensive Investigation of Blanket Testing Requirements using detailed engineering scaling to preserve phenomena, etc. show that:

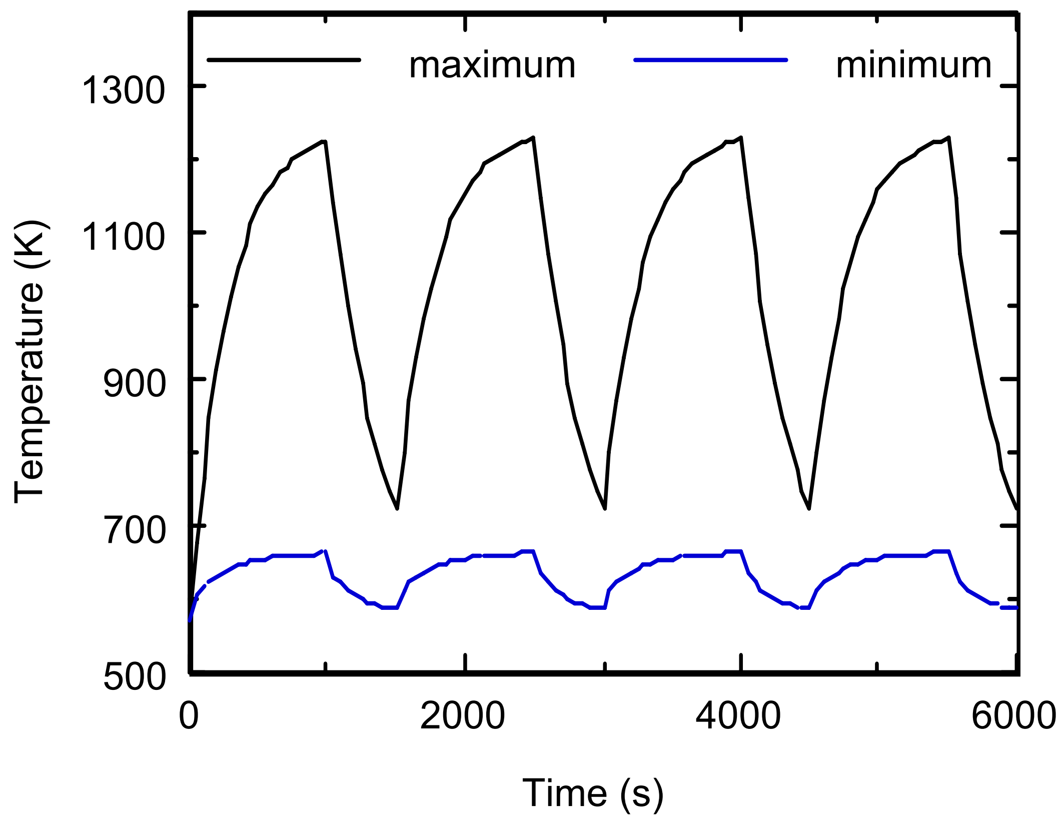
  plasma burn time (t_b) > 3 τ_c

  plasma dwell time (t_d) < 0.05 τ_c

  Where τ_c is a characteristic time constant (for a given phenomena)
- Characteristic time constants for various responses/phenomena in the blanket range from a few seconds to a few hours (even days for some phenomena). See Tables in Appendix.

  - Thus the burn time needs to be hours and the dwell time needs to be a few seconds.
- Example of Difficulty: In ITER-FEAT scenario of 400 s burn and 1200 s dwell time, even temperature equilibrium can not be attained. Most critical phenomena in the blanket have strong temperature dependence. Tests for phenomena such as tritium release and recovery, failure modes, etc. can yield the wrong answer.

Example of many calculations in the literature of the adverse effects of low plasma cycle* on the usefulness of FNT tests



First unit cell breeder temperature response (burn time = 1000 s, dwell time = 500 s)

***Not long enough burn time and not short enough dwell time**

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*

International studies and experts have concluded that extensive testing of fusion nuclear components in FUSION testing facilities is REQUIRED prior to DEMO

--Non-fusion facilities cannot fully resolve any of the critical issues for blankets or PFC's

--There are critical issues for which no significant information can be obtained from testing in non-fusion facilities (An example is identification and characterization of failure modes, effects and rates)

--The Feasibility of Blanket/PFC Concepts can NOT be established prior to testing in fusion facilities

Note: Non-fusion facilities can and should be used to narrow material and design concept options and to reduce the costs and risks of the more costly and complex tests in the fusion environment. Extensive R&D programs on non-fusion facilities should start now.

Critical Factors in Deciding where to do Blanket / PFC / FNT Testing

- Tritium Consumption / Supply Issue
- Reliability / Maintainability / Availability Issue
- Cost
- Risk
- Schedule

Fundamental Considerations in Deciding where to do Blanket / PFC / FNT Fusion Testing

- The FNT Testing Requirements are

  Fusion Power only 20-30 MW

  Over about 10m^2 of surface area (with exposure to plasma)

  With Steady State Plasma Operation (or plasma cycle >80%)

  Testing Time on successive test articles equivalent to neutron fluence of 6 MW • y/m^2

- Tritium Consumption / Tritium Supply issue dictates that any fusion facility that performs FNT testing must internally breed all (or most) of its own tritium

  - If TBR <1, Larger Power Devices require larger TBR

  - For a given TBR, the FW area required for breeding is much larger than for small devices

- FNT Testing involves RISKS to the fusion testing device

  - unvalidated technology with direct exposure to plasma

  - frequent failures are expected

  - considerable amounts of tritium and activated materials

  - These risks are much greater for large power devices because of the much larger area for tritium breeding

- Cost

  - Frequent failures will require frequent replacements: COST will be much higher for the larger power, larger area devices

  - COST of operation to higher fluence is larger for larger devices

Selected Publications of Fundamental and Key Information for Students / Scientists / Engineers interested in studying Fusion Nuclear Technology

Copies of all publications below can be downloaded from the following website:
www.fusion.ucla.edu/abdou
This website has many additional publications and presentations.

The UCLA website (www.fusion.ucla.edu) also has posted many papers, presentations and major reports (e.g. BCCS, FINESSE, APEX, etc.)

M. Abdou and C.W. Maynard, "Calculational Methods for Nuclear Heating—Part I: Theoretical and Computational Algorithms", Nuclear Science and Engineering, 56: 360–380 (1975).

M. Abdou and C.W. Maynard, "Calculational Methods for Nuclear Heating—Part II: Applications to Fusion-Reactor Blankets and Shield", Nuclear Science and Engineering, 56: 381–398 (1975).

M. Abdou, L.J. Wittenberg, and C.W. Maynard, "A Fusion Design Study of Nonmobile Blankets with Low Lithium and Tritium Inventories", Nuclear Technology, 26: 400–419 (1975).

M. Abdou, "Nuclear Design of the Blanket/Shield System for a Tokamak Experimental Reactor", Nuclear Technology 29: 7–36 (1976).

M. Abdou, "Important Aspects of Radiation Shielding for Fusion Reactor Tokamaks", Atomkernenergie, 30(4): 308–312 (1977).

M. Abdou, "Radiation Considerations for Superconducting Fusion Magnets", Nuclear Materials, 72(1/2): 147–167 (1978).

M. Abdou, "Key Issues of FED INTOR Impurity Control System", Nuclear Technology/Fusion, 4: 654-665 (1983).

M. Abdou, "Critical Issues and Required Facilities", Journal of Fusion Energy, 4: 133-138 (1985).

P. Gierszewski, M. Abdou, et. al., "Engineering Scaling and Quantification of the Test Requirements for Fusion Nuclear Technology", Fusion Technology, 8: 1100-1108 (1985).

M. Abdou, et. al, "A Study of the Issues and Experiments for Fusion Nuclear Technology," Fusion Technology, 8: 2595-2645 (1985).

M. Abdou, et. al, "Deuterium-Tritium Fuel Self-Sufficiency in Fusion Reactors," Fusion Technology, 9: 250-285 (1986).

M. Abdou, et. al., "Blanket Material and Engineering Issues, and Requirements for Experiments and Facilities", Journal of Nuclear Materials, 141-143: 10-18 (1986).

M. Abdou, et. al, "Technical Issues and Requirements of Experiments and Facilities for Fusion Nuclear Technology" Nuclear Fusion, 27, No. 4: 619-688 (1987).

K. Fujimura, A. Raffray and M. Abdou, "Analysis of Helium Purge Flow in a Solid Breeder Blanket", Fusion Engineering & Design, 8: 109-114 (1989).

K. McCarthy, M. Tillack and M. Abdou, "Analysis of Liquid Metal MHD Flow using an Iterative Method to Solve the Core Flow Equations", Fusion Engineering & Design, 8: 257-264 (1989).

M. Abdou, M. Tillack, and A. Raffray, "Thermal, Fluid Flow and Tritium Release Problems in Fusion Blankets," Fusion Technology, 18,No.2: 165-200 (1990).

G. Federici, A. Raffray, M. Abdou, "MISTRAL: A Comprehensive Model for Tritium Transport in Lithium-Base Ceramics. Part I: Theory and Description of Model Capabilities", Journal of Nuclear Materials, 173: 185-213 (1990).

G. Federici, A. Raffray, M. Abdou, "MISTRAL: A Comprehensive Model for Tritium Transport in Lithium-Base Ceramics. Part II: Comparison of Model Predictions with Experimental Results", Journal of Nuclear Materials, 173: 214-228 (1990).

A. Ying, A. Raffray and M. Abdou, "Transient Flow Behavior for a Helium-Cooled Ceramic Breeder Blanket," Nuclear Engineering & Design, 126: 137-145 (1991).

K. McCarthy and M. Abdou, "Analysis of Liquid Metal MHD Flow in Multiple Adjacent Ducts using an Iterative Method to Solve the Core Flow Equation," Fusion Engineering & Design, 13: 363-380 (1991).

G. Federici, A. Raffray, M. Billone, C. Wu, S. Cho and M. Abdou, "An Assessment of Models for Tritium Release from Ceramic Breeders for Blanket Analysis Applications," Journal of Nuclear Materials, 212-215: 1003-1009 (1994).

S. Cho, A. Raffray and M. Abdou, "Modeling of Tritium Release from Beryllium in Fusion Blanket Applications," Journal of Nuclear Materials, 212-215: 961-965 (1994).

F. Tehranian, M. Abdou and M. Tillack, "Effect of External Pressure on Particle Bed Effective Thermal Conductivity," Journal of Nuclear Materials, 212-215: 885-890 (1994).

M. Abdou, "A Volumetric Neutron Source for Fusion Nuclear Technology Testing and Development," Fusion Engineering and Design, 27: 111-153 (1995).

S. Cho and M. Abdou, "Analysis of Tritium Transport in Irradiated Beryllium," Fusion Engineering and Design, 28: 265-270 (1995).

M. Xu, M. Abdou, and A. Raffray, "Thermal Conductivity of a Beryllium Gas Packed Bed," Fusion Engineering and Design, 27: 240-246 (1995).

F. Tehranian and M. Abdou, "Experimental Study of the Effect of External Pressure on Particle Bed Effective Thermal Properties," Fusion Technology, 27: 298-313 (1995).

M. Abdou, et al, "Japan Atomic Energy Research Institute/United States Integral Neutronics Experiments and Analyses for Tritium Breeding, Nuclear Heating, and Induced Radioactivity," Fusion Technology, 28,No.1: 5-38 (1995).

M. Abdou, et al, "Results of an International Study on a High-Volume Plasma-Based Neutron Source for Fusion Blanket Development," Fusion Technology, 29: 1-57 (1996).

R. Abelson and M. Abdou, "Experimental Evaluation of the Interface Heat Conductance between Roughened Beryllium and Stainless Steel Surfaces", Journal of Nuclear Materials, 233-237: 847-851 (1996).

L. Zhang, M. Abdou, "Kerma factor evaluation and its application in nuclear heating experiment analysis", Fusion Engineering and Design, 36: 479-503 (1997).

A. Ying, M. Abdou, "Analysis of Thermomechanical Interactions and Properties of Ceramic Breeder Blankets", Fusion Engineering and Design, 39-40: 651-657 (1998).

M. Abdou, A. Ying, and Z. Lu, "Thermal and Mechanical Properties of Ceramic Blanket Particle Bed Materials: Numerical Derivation", Journal of Nuclear Materials, 258-263: 576-581 (1998).

M. Abdou and the APEX Team, "Exploring Novel High Power Density Concepts for Attractive Fusion Systems", Fusion Engineering & Design, 45, No.1: 145-167 (1999).

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).

S. Cho, M. Abdou, "Analysis of Tritium Kinetics of SIBELIUS Beryllium", Fusion Engineering and Design, 51-52: 85-91 (2000).

M. Abdou, et. al. "On the Exploration of Innovative Concepts for Fusion Chamber Technology", Fusion Engineering and Design, 54: 181-247 (2001).

R. Abelson and M. Abdou, "Experimental Measurement of the Interface Heat Conductance between Nonconforming Beryllium and Type 316 Stainless Steel Surfaces Subjected to Nonuniform Thermal Deformations", Fusion Technology, 39: 157-188 (2001).

Z. Lu, M. Abdou, A. Ying, "3-D Micromechanical Modeling of Packed Beds", Journal of Nuclear Materials, 299: 101-110 (2001).

M. Ni, M. Abdou, "Temporal Second-Order Accuracy SIMPLE-Type Method for Unsteady Incompressible Flows", Numerical Heat Transfer B, 46: 529-548 (2004).

M. Abdou, et. al., "Overview of Fusion Blanket R&D in the U.S. over the Last Decade," Nuclear Engineering and Technology, 37:5 (2005).

S. Smolentsev, N. Morley, M. Abdou, "Code Development for Analysis of MHD Pressure Drop Reduction in a Liquid Metal Blanket using Insulation Technique", Fusion Engineering and Design, 73, 83-93 (2005).

S. Smolentsev, M. Abdou, N. Morley, M. Sawan, S. Malang, C. Wong, "Numerical Analysis of MHD Flow and Heat Transfer in a Poloidal Channel of the DCLL Blanket with a SiCf/SiC Flow Channel Insert", Fusion Engineering & Design, 81:(1–7), 549–553 (2006)

A. Ying, M. Abdou, P. Calderoni, S. Sharafat, M. Youssef, Z. An, A. Abou-Sena, E. Kim, S. Reyes, S. Willms, R. Kurtz, "Solid Breeder Test Blanket Module Design and Analysis", Fusion Engineering & Design, 81:(1–7), 659–664 (2006)

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)

A. Abou-Sena, A. Ying, M. Abdou, "Experimental Measurements of the Effective Thermal Conductivity of a Lithium Titanate (Li2TiO3) Pebbles-Packed Bed", Journal of Materials Processing Technology, 181: 206-212 (2007).