A Perspective on Design and Material Issues for Fusion Systems

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Past 28 Years

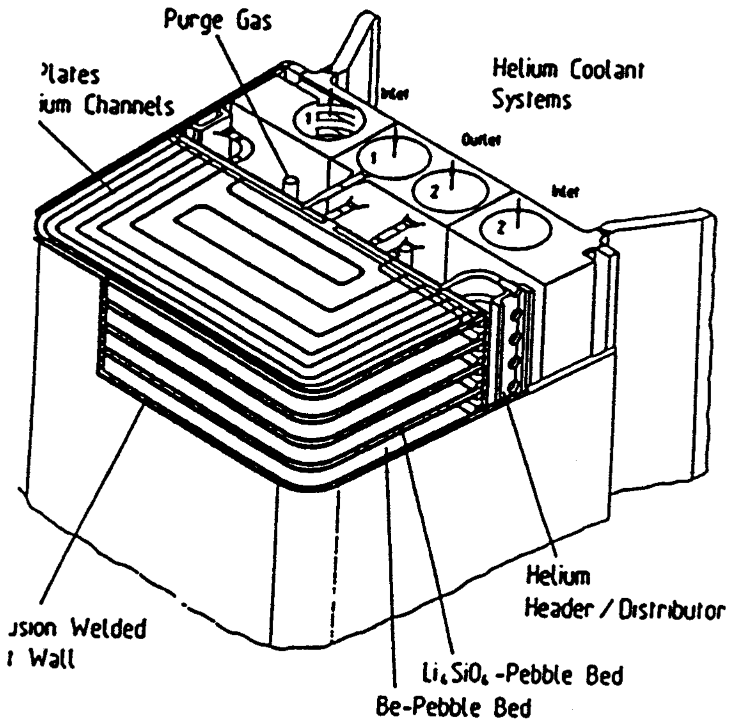
- The world explored and adopted specific materials and concepts.
- The design concepts for in-vessel system are essentially the same as UWMAK-I (1971) and UWMAK-II (1974):
  - solid first wall facing the plasma
  - configurations for solid breeder, separately cooled LM, self-cooled liquid metal

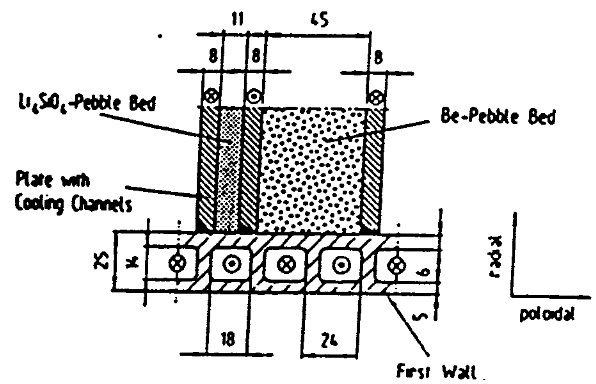
Main Line Blanket Concepts by Various Parties (1997)

	EU	Japan	USA
Helium/Solid Breeder/Ferritic Steel	X	X	X
Water/Solid Breeder/Ferritic Steel		X	
Water/Li Pb/Ferritic Steel	X		
Li/Li/Vanadium			X

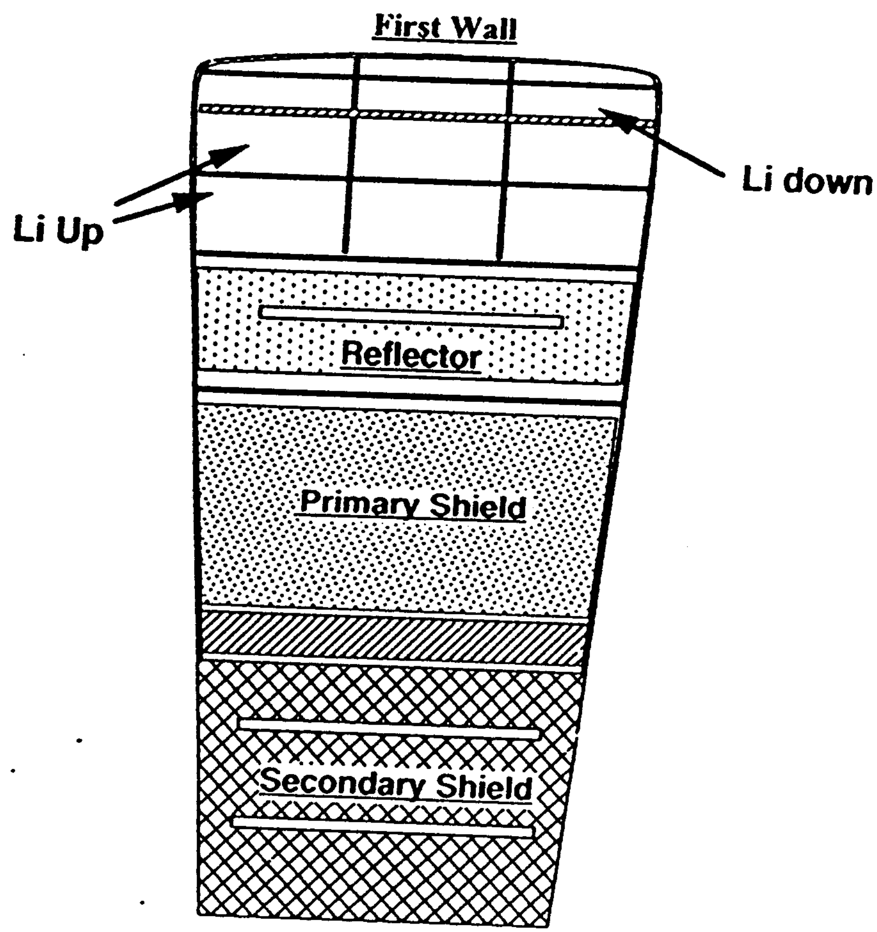
*RF is <u>not</u> devoting resources to R&D

EU Helium-cooled pebble bed DEMO blanket design





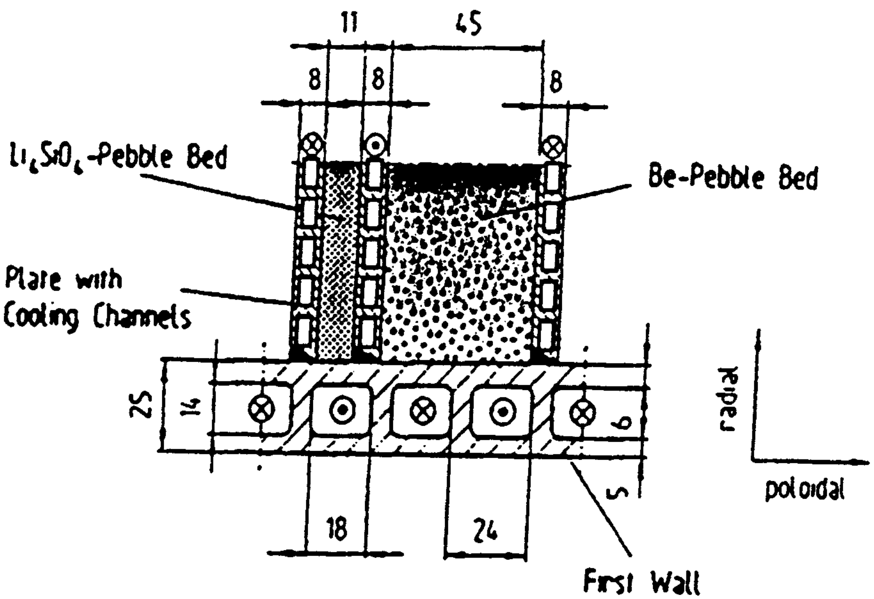
Cross Section of U.S. Li-V DEMO Blanket Module

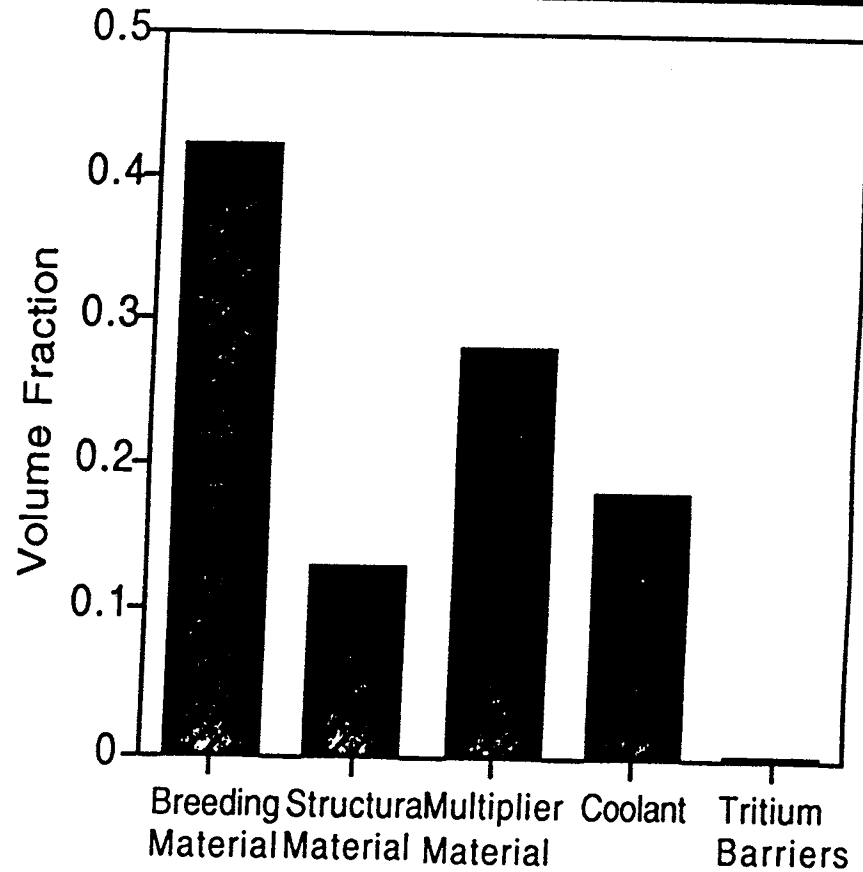


In-Vessel Material System Has Many Materials

-Structural Material
- Cooling Fluid
- Breeding Material
- Tritium Barriers
- Multiplier Material
- Insulators

- These Materials are Equally Important Because
  - Each material has its own critical technical issues
  - Interactions among materials are critical, involve challenging science
  - Can not have low activation structure and HIGH Activation Breeder Material

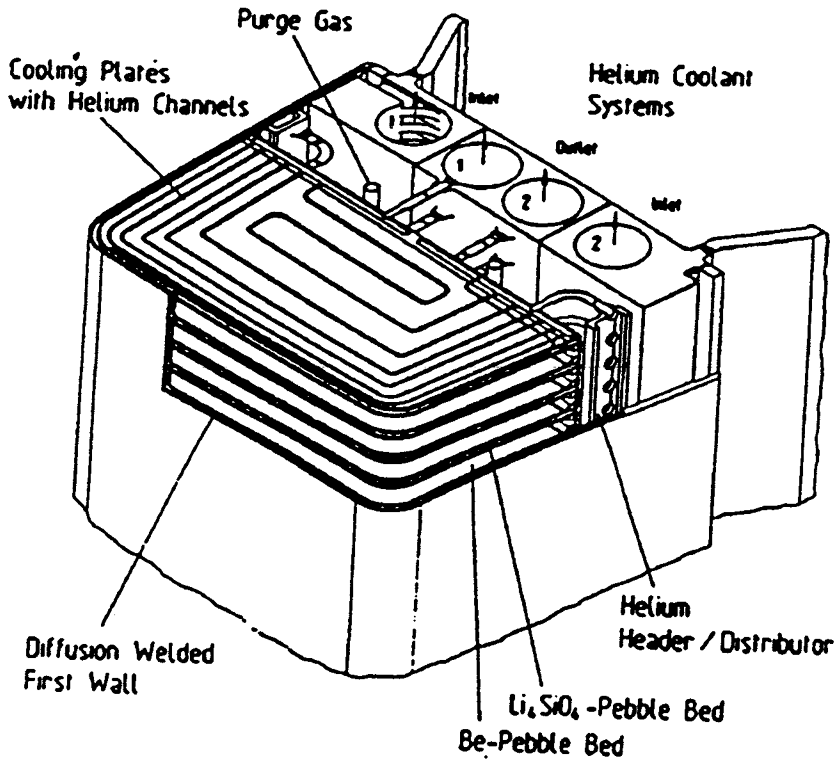


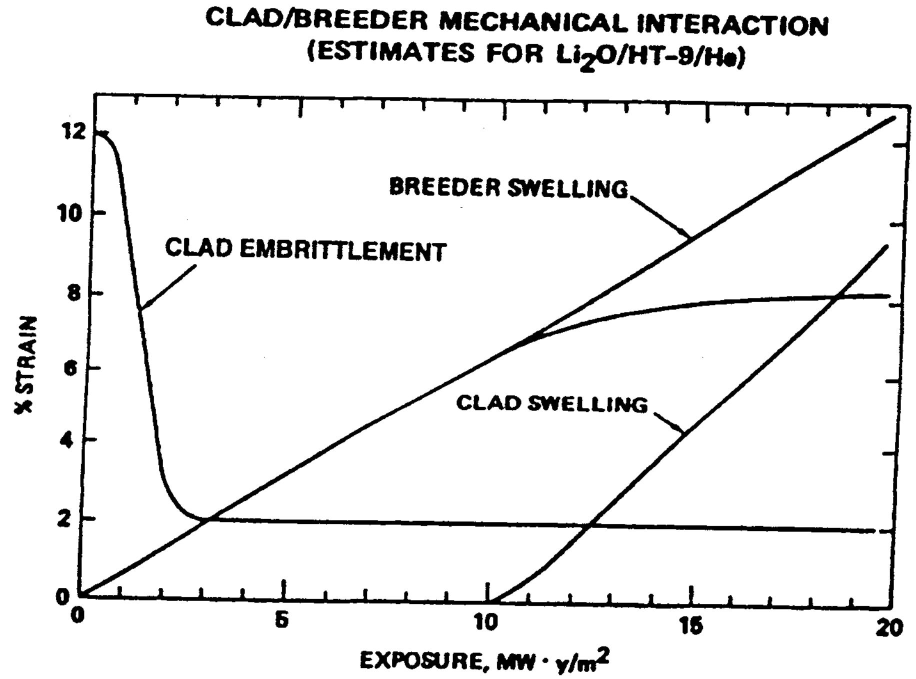


- BALANCE in Research among these materials is Necessary

The interactions among materials are very important:

- often the most critical feasibility issues
- often the most interesting scientific issues





EXAMPLES OF INTERACTIONS BETWEEN PHYSICAL REGIONS

Major Events during the Past Two Years (1997/98)

1. Major cuts in US funding
2. Criticism of fusion became intense

   They say – We do not have an attractive product

   – We do not know how to get there
3. ITER detailed design revealed that the In-Vessel system was one of the most difficult. Design kept changing.

   – This is at 1 MW/m^2
4. "Internal" technical analysis and realistic assessment revealed that current FW/blanket/divertor concepts are <u>not</u> likely to lead to an attractive product.

<u>Major Issues</u>

1. Heat removal capability at high temperature and power density
2. Failure rate (MTBF) is too short
3. Time to recover from failure (MTTR) is too long
4. Tritium fuel self-sufficiency margin

Most Challenging Issues for FPT

1. Heat removal at high temperature and high wall load
2. Failure rate
3. Time to recover from a failure
4. Tritium fuel self sufficiency

Power Density and Heat Flux in Fission Reactors

	PWR	BWR	HTGR	LMFBR	ITER-Type
Equivalent Core Diameter(m) Core Length (m)	3.6 3.8	4.6 3.8	8.4 6.3	2.1 0.9	30 15
Average Core Power Density (MW/m^3)	**96**	**56**	**9**	**240**	**0.4**
Peak-to-Average Heat Flux Coolant (MW/m2)	2.8	2.6	12.8	1.43	50

Suggested Fusion Goals

- Neutron Wall Load > 10 MW/m^2
- Minimize Peak - to - Average Power Density

Current Design Concepts and Materials for First Wall / Blanket Do NOT Have the Capability to Meet the Fusion Challenge

Concept	Wall Load Capability * MW/m^2	Other Observations
Ferritic / He / Breeder **Ferritic / H_20 / Li Pb**	2	• Magnetic material • Fracture toughness
Vanadium Alloy / Lithium	2.5	• V works only with lithium • Is lithium acceptable? • Not feasible until a self healing coating is found
SiC / SiC / He / Breeder	1.5	• Serious feasibility issues • Do <u>NOT</u> know how to design • Poor thermal conductivity

* Average wall load based on 5-mm thick wall, surface heat flux with peaking factor of 2, and high-temperature limits

Goals for MTBF & MTTR Can be Easily Derived

Availability = A

A (Plant) = 75%

A (BOP) = 85%

A (Reactor) = 88%

Reactor

Assume 6 major components with equal outage risk

An example of such a component is FW / Blanket

A (Blanket) = 97.8 %

A (FW / Blanket)

$$A = \frac{MTBF}{MTBF + MTTR}$$

$$\boxed{\frac{MTBF}{MTTR} = 43.8}$$

Note: It is the Mean Time Between Failure which is the issue.
It is NOT lifetime

What MTBF Can Be Achieved?

Several Studies

- R. Bünde et al. (several articles, 1990-95)
- Abdou & Ying (1994)
- Detailed EU Blanket Evaluation (1994)

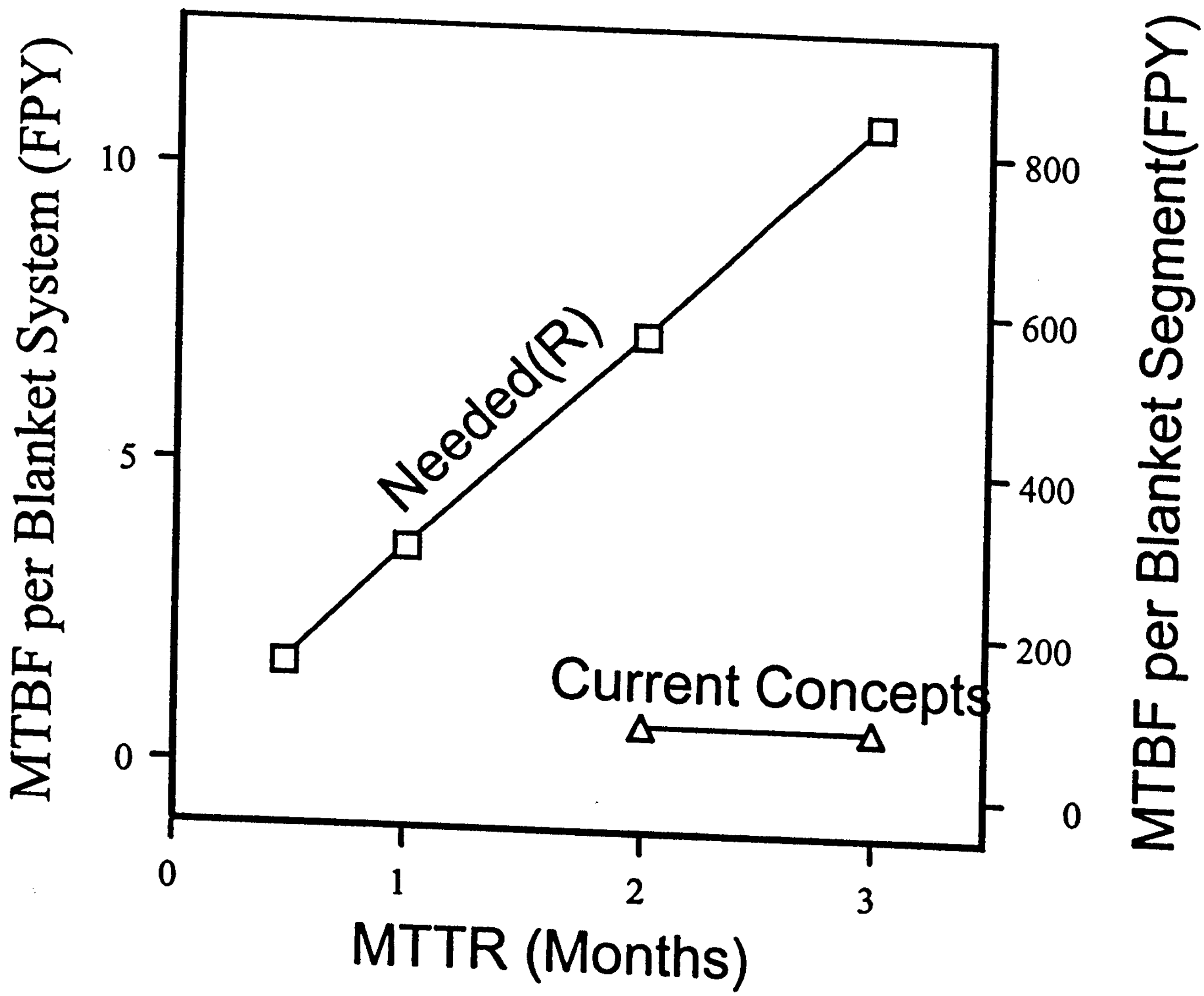
Methodology

- Compile Relevant Failure Rate from Mature Technologies (e.g. fission)
- Estimate Failure Frequency For the Best FW/Blanket Designs Available
    - ◊ Include Failures for Pipes and Welds
    - ◊ IGNORE (DO NOT Include) Fusion Specific Failure Modes

Failure Modes (FW)	Failure Rate $hr^{-1}.m^{-1}$	Length
Diffusion weld	1×10^{-9}	4.56 km
EB Weld	1×10^{-8}	2.93 km
Longitudinal weld	1×10^{-9}	19 km

Failure Modes (BLKT)	Failure Rate $hr^{-1}.m^{-1}$	Length
Longitudinal weld	1×10^{-9}	4.8 km
Butt weld	1×10^{-9}	2.58 km
Pipe bend (90°)	5×10^{-9}	1152 bends
Straight pipe	1×10^{-10}	2.9 km

Current FW/B Design Concepts are NOT Capable of Meeting the Challenging Reliability and Maintenance Requirements



Failure Rates and Maintainability Are More Pressing Issues Than Lifetime

Table 1, Availability as a function of lifetime

Lifetime	Availability
1 year	92 %
2 year	96 %
5 year	98 %

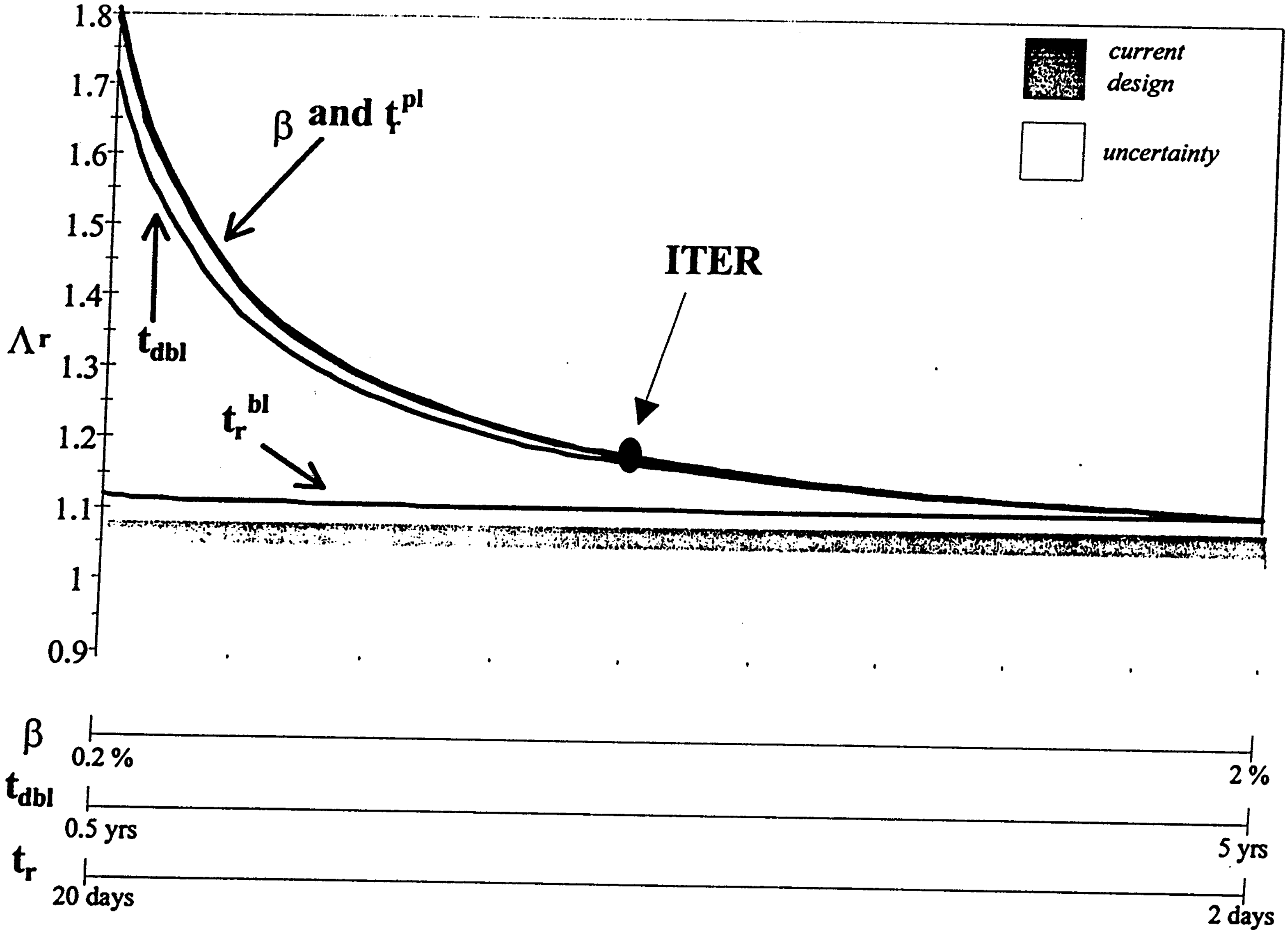
MTTR = 1 month

Table 2, Availability as a function of mean time between failure (MTBF)

MTBF	Availability
1 hour	0.14 %
1 day	3.2 %
1 week	18.6 %

MTTR = 1 month

Tritium Self Sufficiency is a Serious Issue



Summary of FPT most challenging issues

1) Economic competitiveness requires much higher power density than we have been working on. Current first wall/blanket concepts are limited to about 2 or 2.5 MW/m^2 neutron wall load. Comparison to fission reactors reveals that much more higher neutron wall loads should be the goal for fusion R & D.
2) Tritium self-sufficiency is highly uncertain with present concepts.
3) Failure rates as extrapolated from current technologies are too high with present first wall/blanket concepts (and due to the nature of present magnetic confinement schemes)
4) Maintainability is a serious issue with current concepts. Specifically, MTTR (mean time to recover from failure) is very long. Such long MTTR (>2 months) seriously reduces reactor availability and make requirements on MTBF impractical.

Path to Improving Fusion

- All the above four issues need to be addressed (ultimately).
- We need concepts that
  1) can handle much higher wall loads than we have been working on,
  2) can provide better margins for insuring self-sufficiency,
  3) have lower failure rate (longer MTBF), and
  4) faster maintenance (shorter MTTR)

New Directions in the US

- Two studies were initiated in 1998 to explore how to improve fusion
  - APEX (in-vessel system)
  - ALPS (Divertor)

APEX Objective

Explore new (and possibly revolutionary) concepts that can provide the capability to efficiently extract heat from systems with ***high neutron (>7 MW/m²) and surface heat (>1.5 MW/m²) loads*** *while satisfying all FPT functional requirements and maximizing reliability, maintainability, safety and environmental attractiveness*

The Motivation for APEX taken from the New Vision of Restructured Fusion Program

- Emphasize science (including engineering sciences) as basis for innovation
- Take the long term view, **Key is Improving Fusion**

New Concepts

Several are being explored

- Thin Liquid Wall
- Thick Liquid Blanket (no solid first wall)
- Li_2O particulate flow with no structural first wall
- High-temperature refractory alloy (e.g. W or Ta alloy) first wall with helium cooling
- Others

Functional Requirements of Fusion Power Technology

1) provision of VACUUM environment

2) EXHAUST of plasma burn products

3) POWER EXTRACTION from plasma particles and radiation (surface heat loads)

4) POWER EXTRACTION from energy deposition of neutrons and secondary gamma rays

5) TRITIUM BREEDING at the rate required to satisfy tritium self sufficiency

6) TRITIUM EXTRACTION and processing

7) RADIATION PROTECTION

General Criteria for Attractiveness of (Fusion) Energy System

1. ECONOMICS
   a) cost per unit thermal power
   b) thermal conversion efficiency
   c) mean time between failure (MTBF)
   d) mean time to repair (MTTR)
   e) lifetime
2. SAFETY
   a) chemical reactivity
   b) decay heat
   c) tritium inventory and tritium permeation
   d) off-site dose
   e) biological hazard potential
   f) radioactive inventory of volatile materials
   g) etc.
3. ENVIRONMENTAL
   a) waste disposal
   b) routine releases (e.g. tritium)
   c) material resources utilization
   d) etc.

Liquid First Wall Concepts

Use thin/fast liquid layer to remove "surface" heat and peak nuclear heat in FW (and divertor) at low surface temperatures.

- Provides high heat flux removal capability
- Eliminates thermal stress in the first structural wall
- Accommodates plasma erosion (disruption, etc.) without increasing FW thickness or damage

Pump liquid back through self-cooled blanket to:

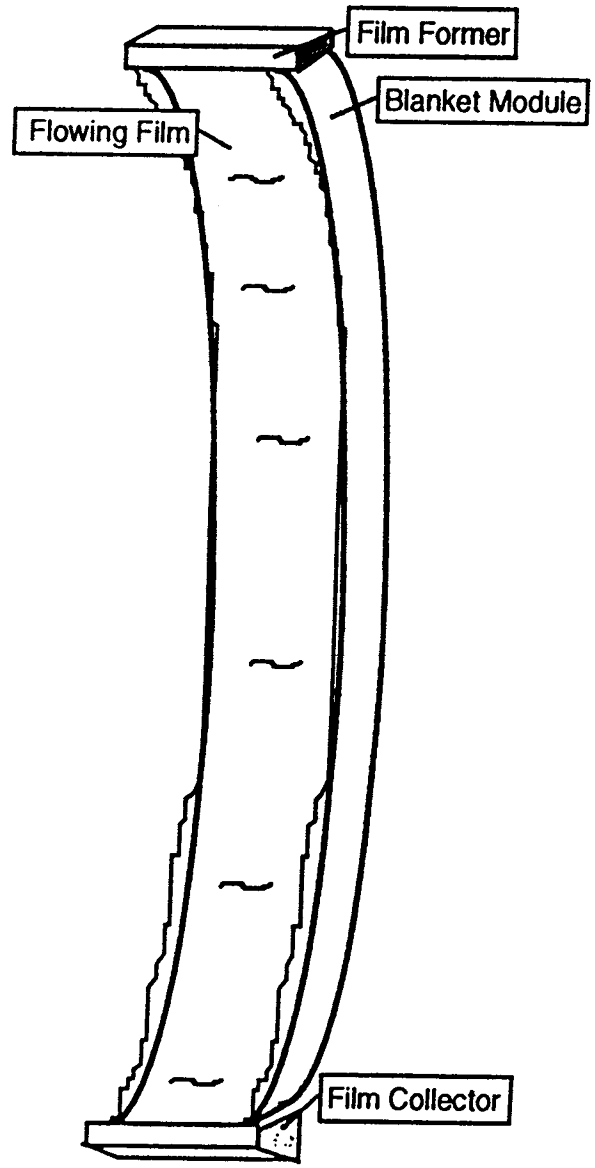
- achieve bulk heating for energy conversion
- breed sufficient tritium
- attenuate neutron flux

<u>Other advantages</u>

➢ FW surface breeds tritium

➢ Reduces failure rates in FW

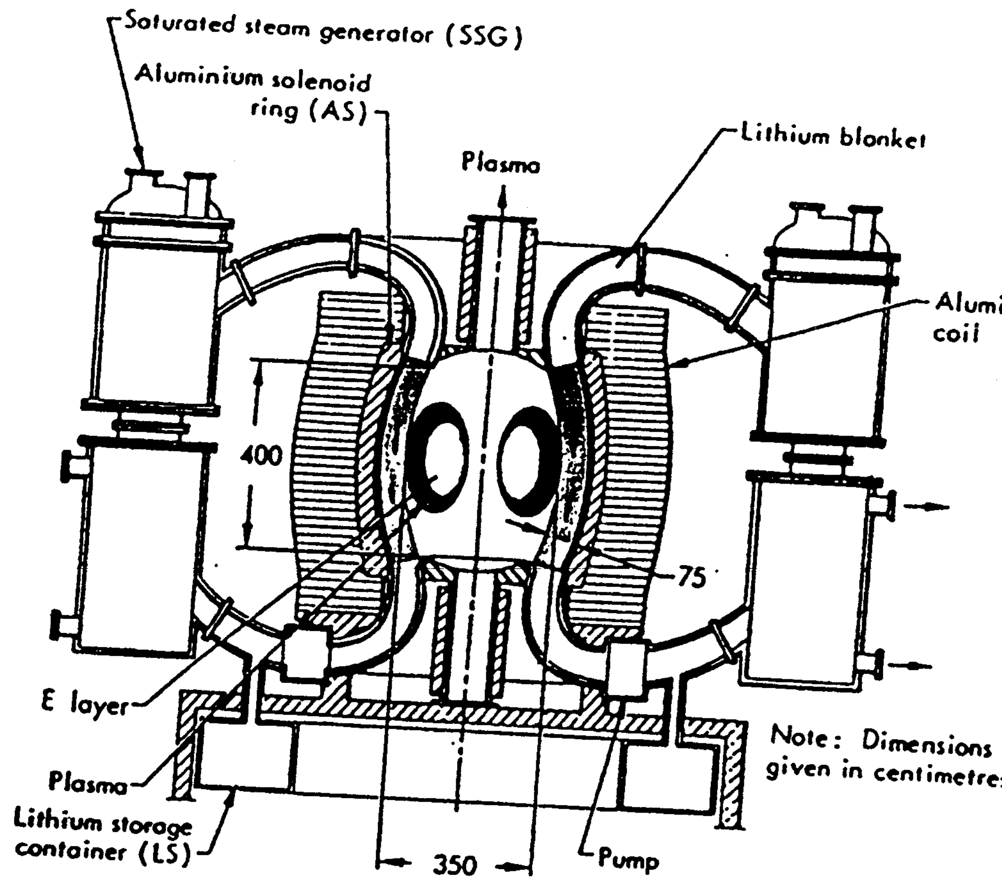
➢ Flowing liquid helps pump plasma impurities



Thick Liquid FW and Blanket

- Fast Flowing Thick Liquid FW/Blanket Flow catches all surface and nuclear heating
- Applicable to alternative confinement schemes (here an FRC)
- Easier maintainability of in-vessel components

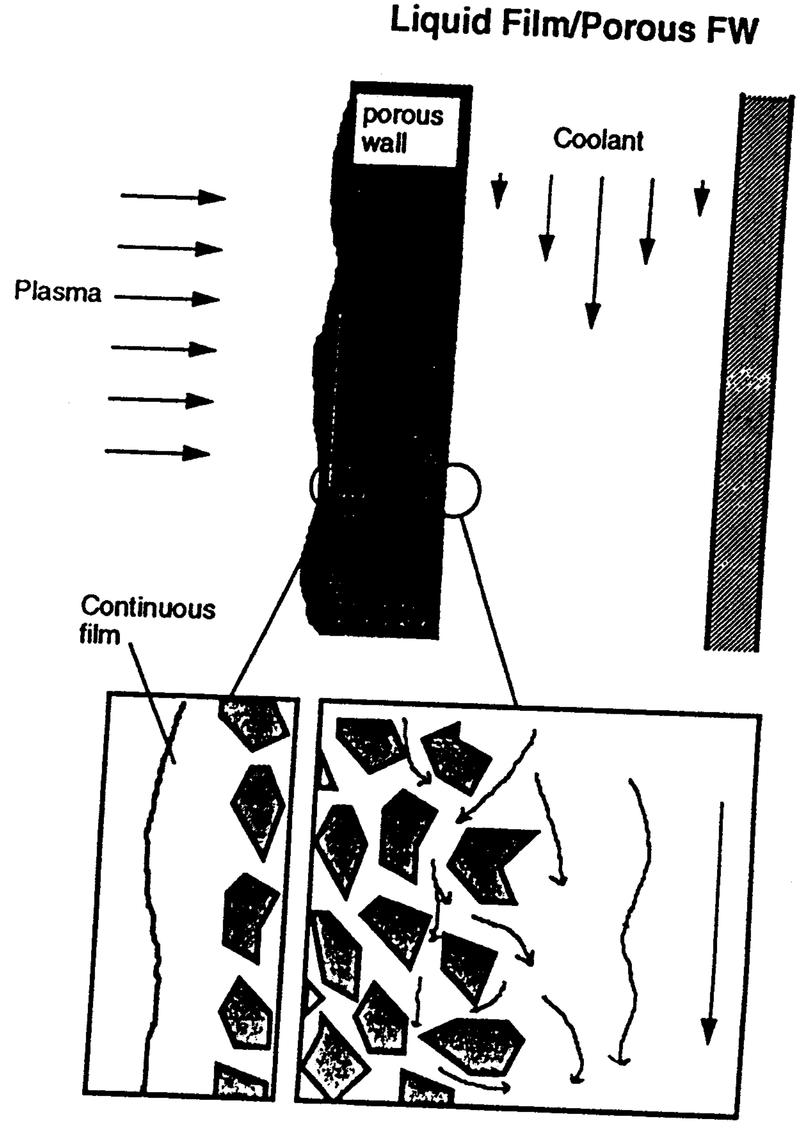
Other advantages
- ➢ Renewable FW surface
- ➢ Improved tritium breeding characteristics
- ➢ Impurity pumping
- ➢ Reduced neutron damage and activation of structures



'The Immaculate' Astron, Field Reversed Configuration
(Christofilos, J. Fus. Energy **8**, 1989)

Liquid-Filled Porous Wall Concepts

- Liquid from cooling channel flows through porous wall to the plasma-facing surface
- Surface heat removed by combination of conduction to main coolant channels, convection along wall, and/or heat of vaporization

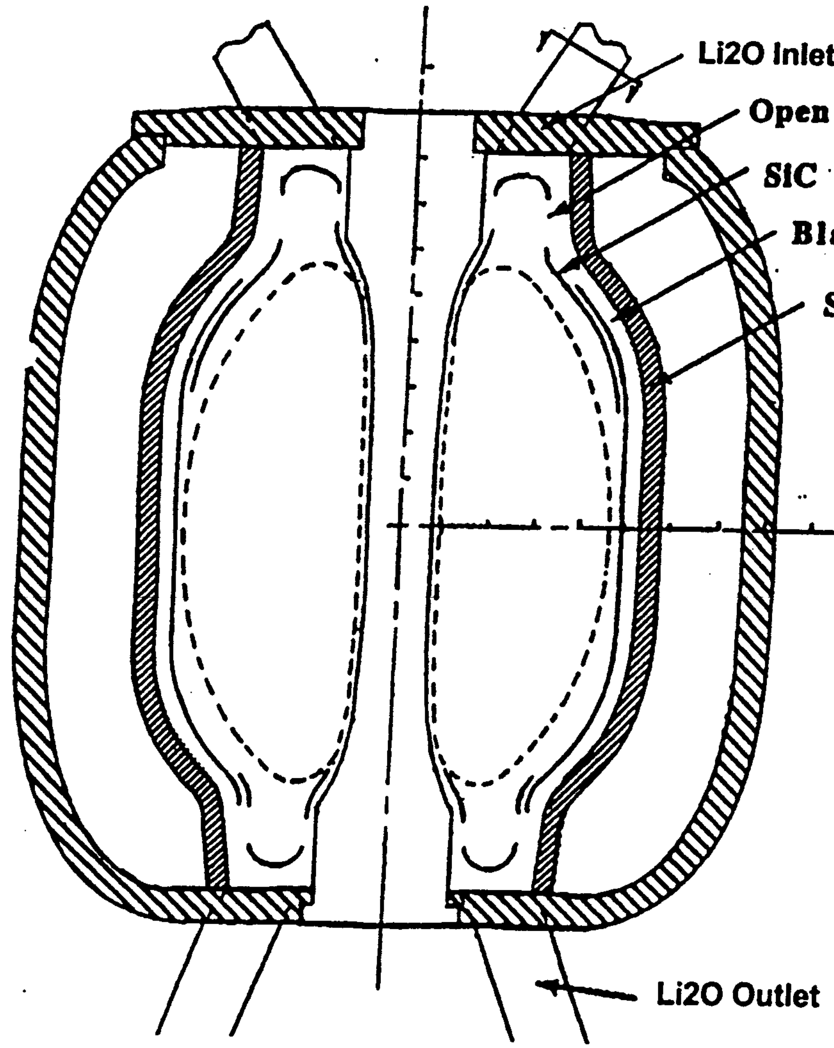


Advantages

- Renewable FW surface
- Reduced elastic modulus of porous material FW
- High thermal conductivity (even under irradiation) and thermal inertial due to filling porous FW with liquid
- Reduced coolant-side film drop and thermal contact resistance by direct liquid contact to bulk coolant and mixing

Lithium Oxide Particulate FW/Blanket

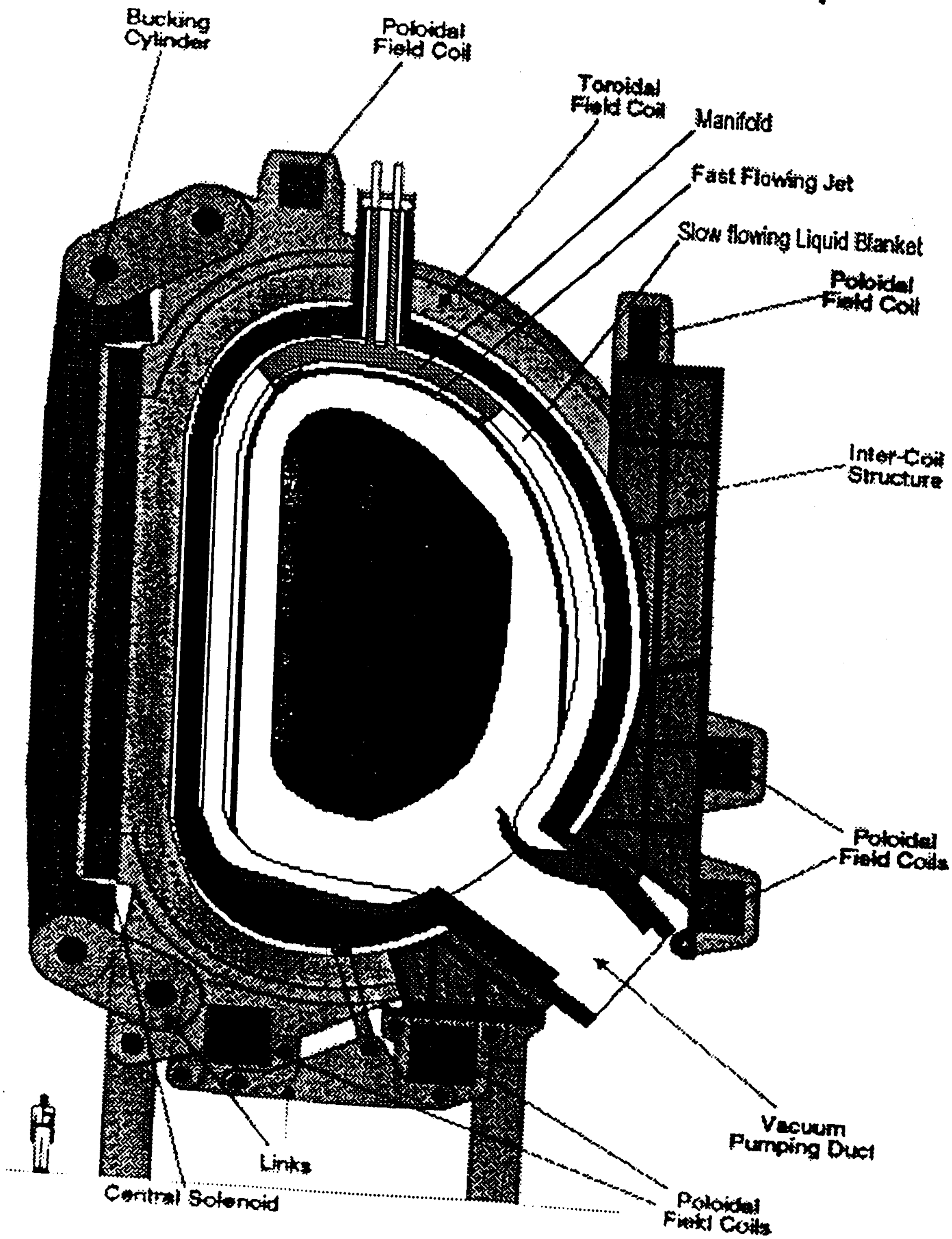
- Free flowing Li_2O particles in direct contact with the plasma act as first wall and divertor
- Li_2O particles flowing in channels behind the first structural wall act as blanket
- Particles are recirculated by means of mechanical conveyors



Other Advantages

- ➢ Li_2O has low vapor pressure up to 1000°C
- ➢ High temperature and large ΔT for power conversion

Thick Liquid FW/Blanket Concept



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Possible Implications of the New Innovative Designs

- Liquid walls and/or liquid blankets will have tremendous effects on the operating environment and requirements for the solid first wall.

- These effects depend on the liquid (lithium, LiPb, or flibe) and the thickness.

- Effects
  - Eliminate thermal stress as a limiting factor
  - Lower Flux
  - Markedly softer spectrum (No 14 MeV peak anymore)
  - Lower activation
  - Lower helium
  - Lower dpa
  - Big change in helium-to-dpa-ratio. It will be much lower (easier to test in fission reactor)

- For thick liquids, the problem may shift more toward making the vacuum vessel thin, better reweldability limit, and lower failure rate.

Vacuum Vessel As The First Solid Wall

- It is desirable to have only liquids inside the vacuum vessel.

- It takes only 40-50 cm of liquid to reduce dpa in 30 year (i.e. lifetime) to < 200 in stainless steel.

- Reweldability becomes a major issue.
  It takes a lot more liquid to reduce helium to < 1 appm in 30 years.

<u>Interesting Material Issues</u>

- Criteria for reweldability of the vacuum vessel

- Material candidates for vacuum vessel?

- Material candidates for flow guiders, MHD guiders, etc.

Liquid Walls/Blankets Will Reduce He, Dpa, He/Dpa Ratio In Solid First Walls

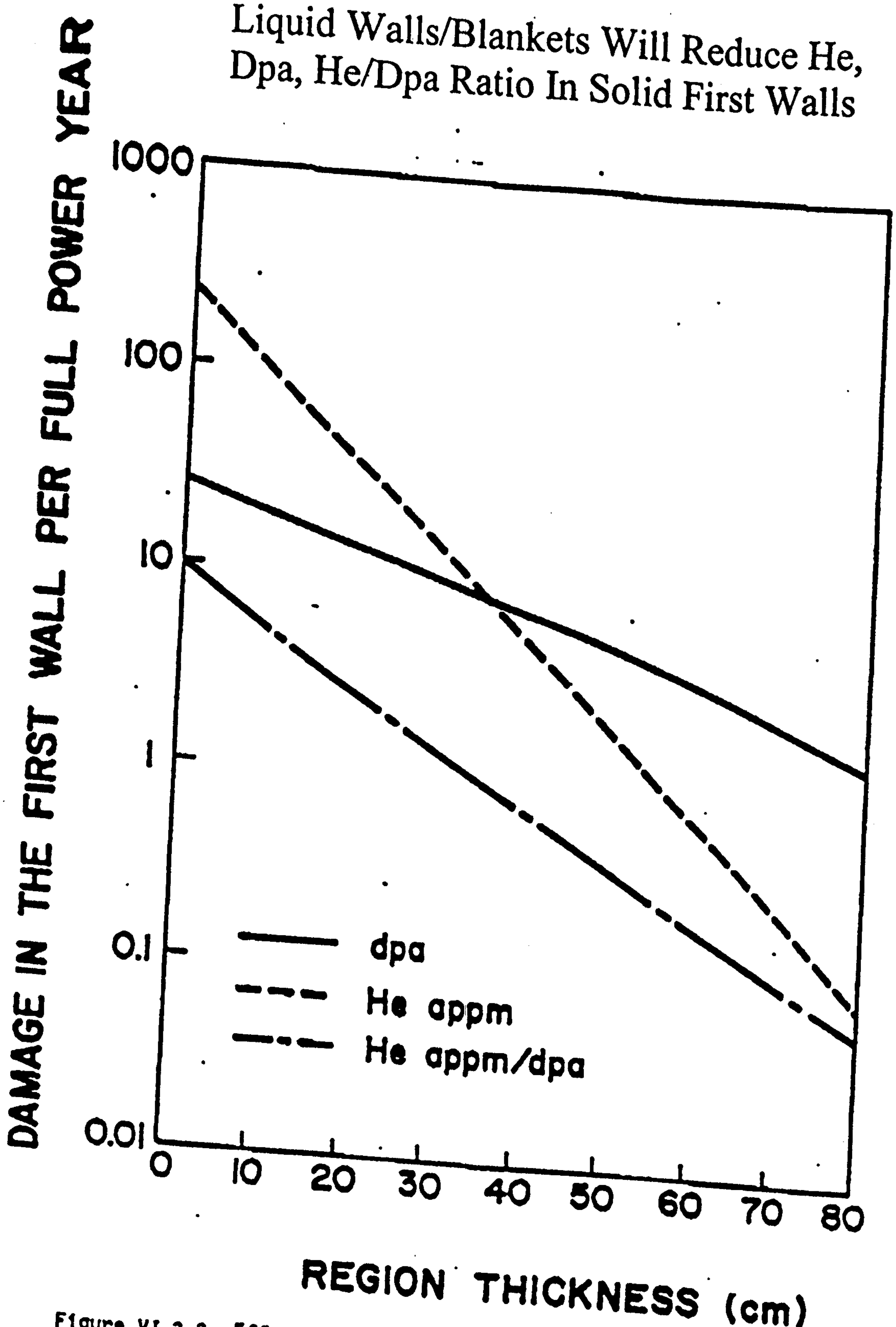

Figure VI.3-8 Effect of INPORT tubes on Damage in HT-9 first wall.

→ Effective LiPb (not Li) thickness in front of wall

From HIBALL (M. Sawan's Calculations)
Neutron Wall Load = 4.54 MW/m^2

APEX (M. Youssef)

Maximum Rate of Displacement per Atom in the Vaccum Vessel Versus Convective Layer Thickness

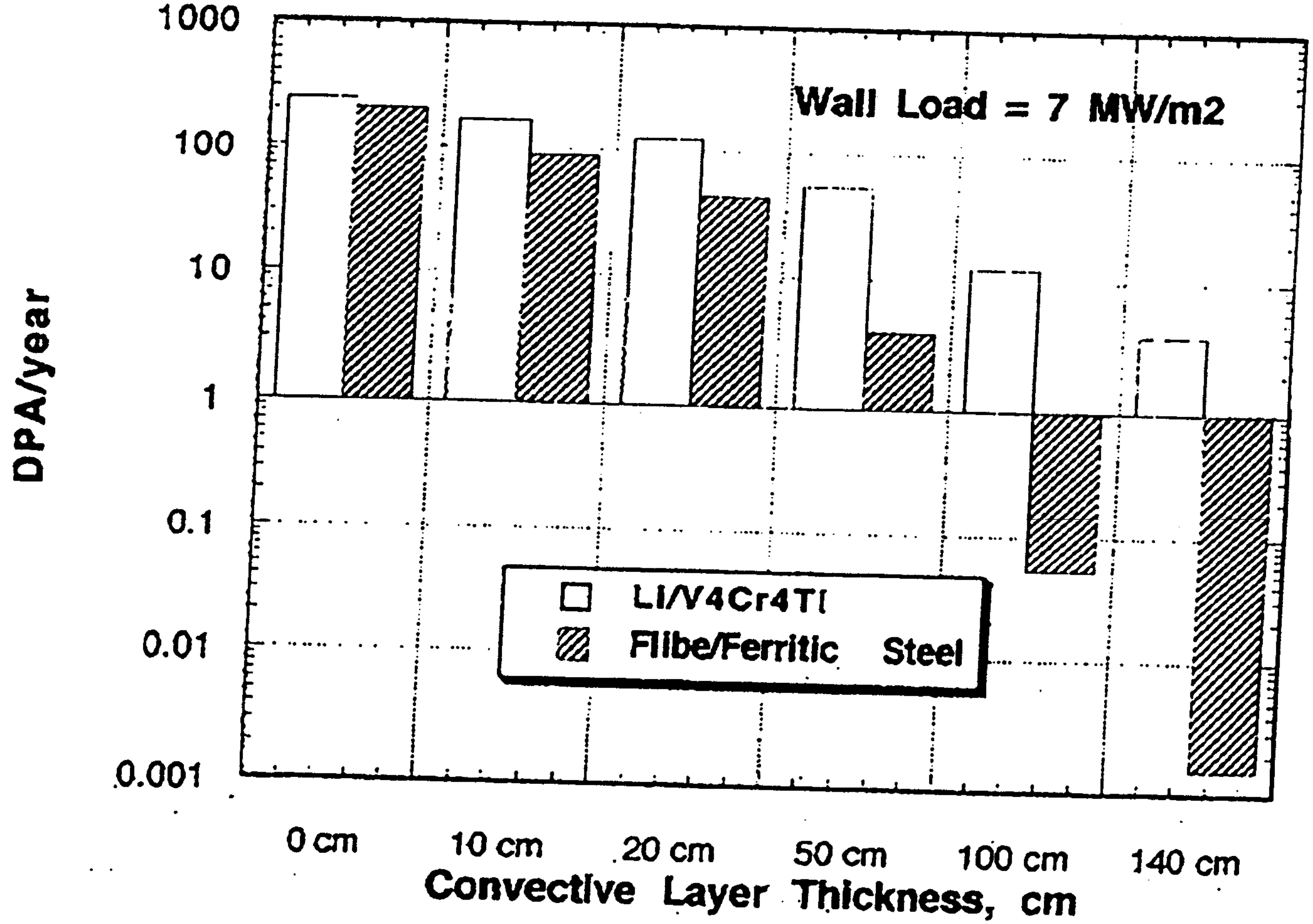

Fig 6

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APEX (M. Youssef)

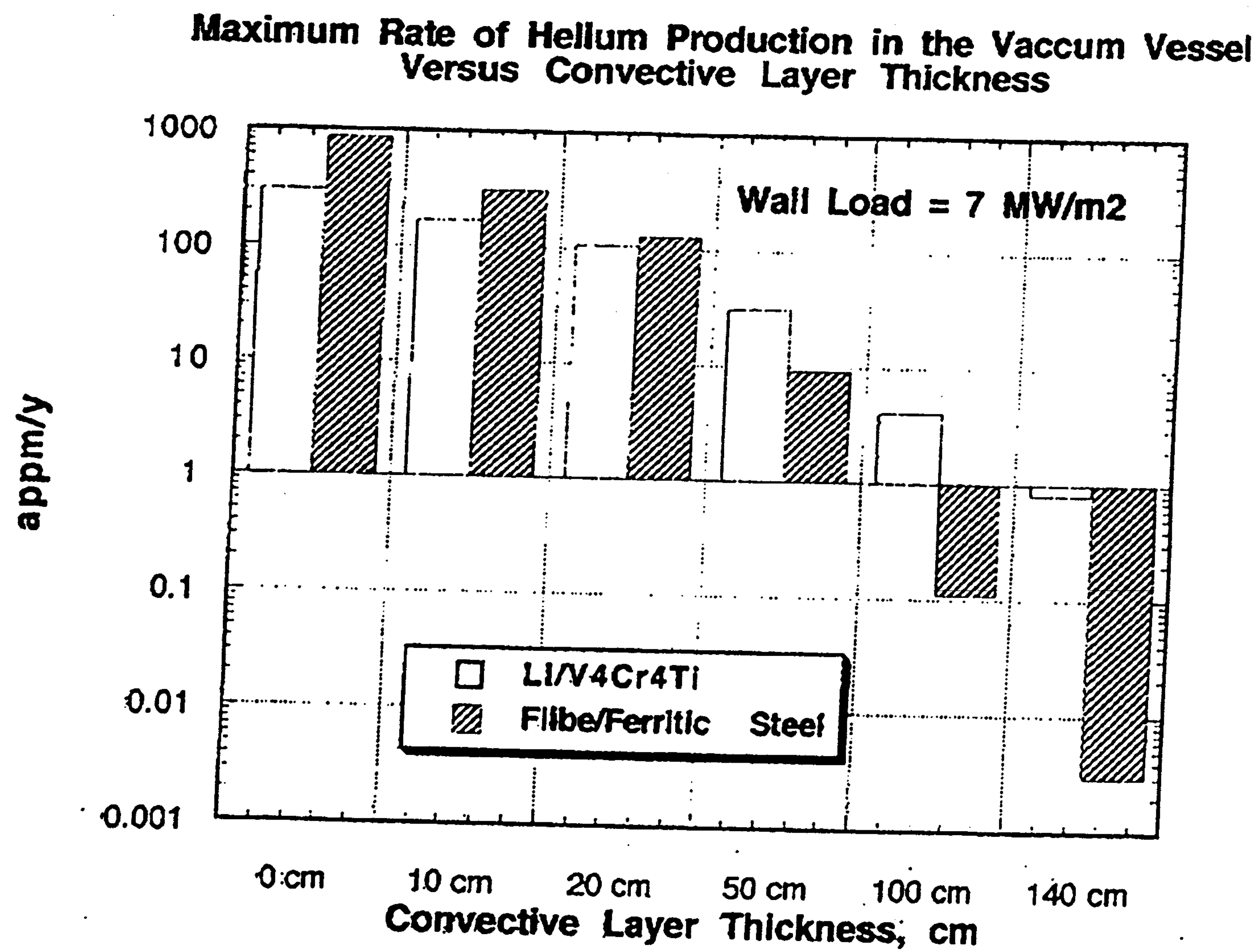

Fig 7

November, 1977

ANL/FPP/TM-99
MDCE-1743

THE ESTABLISHMENT OF ALLOY DEVELOPMENT GOALS IMPORTANT TO THE COMMERCIALIZATION OF TOKAMAK-BASED FUSION REACTORS*

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Authorized Reprint from
Special Technical Publication 570

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Comparison of Displacement and Gas Production Rates in Current Fission and Future Fusion Reactors

REFERENCE: Kulcinski, G. L., Doran, D. G., and Abdou, M. A., "Comparison of Displacement and Gas Production Rates in Current Fission and Future Fusion Reactors," *Properties of Reactor Structural Alloys After Neutron or Particle Irradiation, ASTM STP 570*, American Society for Testing and Materials, 1975, pp. 329–351.

ABSTRACT: The displacement, helium production, and hydrogen production rates in five candidate materials for controlled thermonuclear reactors (CTR) (Type 316 stainless steel, molybdenum, columbium, vanadium, and sintered aluminum product) were calculated for seven potential irradiation facilities. The damage rates were calculated for two fast fission reactors (fast flux test facility and Experimental Breeder Reactor-II), two thermal reactors (high flux isotope reactor and experimental test reactor), two accelerator neutron sources (Los Alamos Meson Production Facility and rotating target neutron source), and a typical CTR blanket. It was found that while fission reactors can easily duplicate displacement damage rates typical of CTR first walls, they fall short, sometimes by several orders of magnitude, of duplicating the helium and hydrogen production rates. The one exception to the latter statement is that helium production in stainless steel, due to the ^{59}Ni production, can even be higher in thermal fission reactors than in CTR. Accelerator sources produce damage which is more like that in CTR, but the absolute magnitude in current facilities is too low by at least an order of magnitude. It is concluded that the high flux isotope reactor is currently the best neutron facility to simulate fusion reactor damage.

KEY WORDS: radiation, irradiation, power reactors (nuclear), displacement, simulation, helium, hydrogen

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