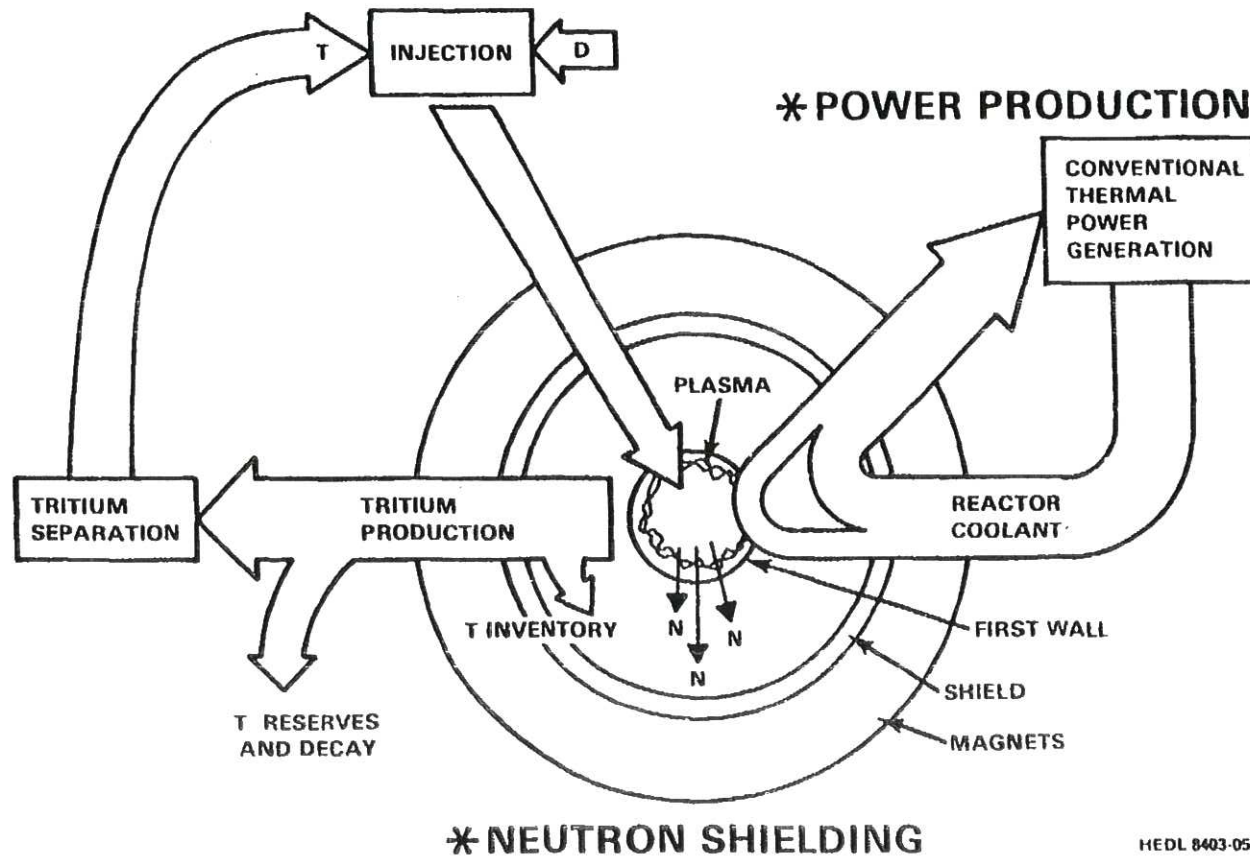


FIG. 1
Ref. G. W. Hollenberg, et al.
Tritium Breeding Materials,
CONF-8404137-1, DE84-
010521.

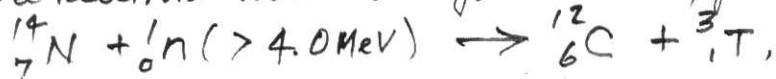
FUNCTIONS OF A FUSION BLANKET

* TRITIUM PRODUCTION



Tritium Production and Handling/Blanket :

Sources of tritium are from natural production via interaction of cosmic rays with atmosphere and from nuclear reactions; for example, interaction of protons and neutrons with nitrogen and oxygen atoms in the atmosphere,



and nuclear reactions as listed below.

Reference: J. E. Phillips + C. E. Easterly, Sources of Tritium, ORNL/TM-6402 (1980).

Table 3.2. Effective microscopic cross-sections for reactions producing tritium or precursors^a

Reaction	$\sigma_{\text{eff}} (10^{-28} \text{ m}^2)$
${}^2\text{H}(n, \gamma)\text{T}$	0.000316
${}^6\text{Li}(n, \alpha)\text{T}$	693
${}^7\text{Li}(n, n\alpha)\text{T}$	0.0516
${}^{10}\text{B}(n, \alpha)\text{Li}$	3060
${}^{10}\text{B}(n, 2\alpha)\text{T}$	1.27

^aFrom ref. 37. [Bell, 1973]

Also^b, ${}^3\text{He}(n, p){}^3\text{H}$ production 5237 barn

^bFrom ref. 53 [Fischer, 1977]

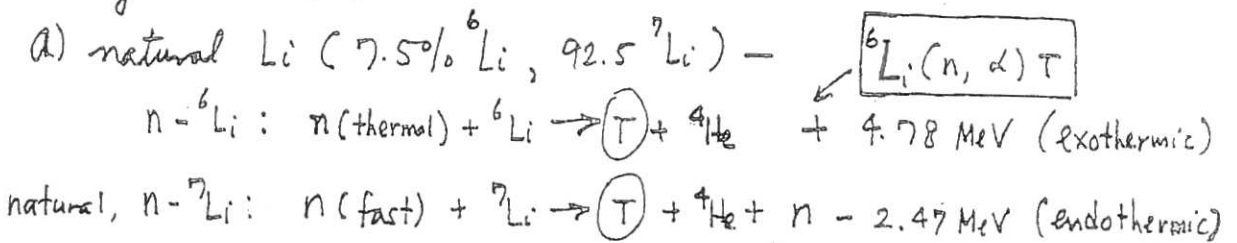
Tritium decays into ${}^3\text{He}$ with a beta decay with $T_{1/2} = 12.32 \text{ yrs}$,
i.e., ${}^3_1\text{H} \rightarrow {}^3_2\text{He}^+ + e^- + \bar{\nu}_e + 18.6 \text{ keV}$.

Blanket Materials (for D-T Reactors)

- should be "tritium-breeding" materials (e.g. liquid Li)
- should be "structural" materials (w/ 1-2 m thick)
- should be good "coolants" (e.g., water)

But, liquid lithium and water don't get along - bad choice!
 → Careful selection of materials needed!

Breeding Materials:



${}^7\text{Li}(n, n'\alpha)\text{T}$

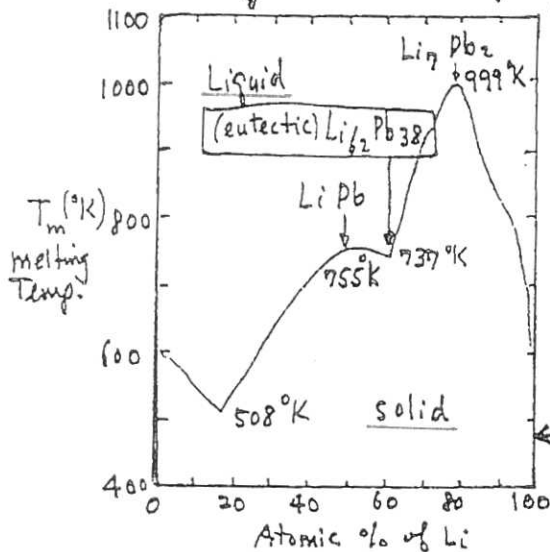
TBR $\equiv \frac{\text{tritium atoms bred}}{\text{tritium atoms used}}$
 $= \frac{\text{tritium atoms}}{\text{incident neutrons}}$

Tritium Breeding Ratio (TBR) ≥ 1.6

depends on structural material and blanket thickness
 [DEMO: TBR = 1.127]

Solid Li Compounds:

(Li-Pb) - Has higher melting temp. than that of natural Li; $(T_m = 450^\circ\text{K})$ & TBR ~ 1.6



- (Li_2O) - $T_m \sim 1840^\circ\text{K}$
 - has low thermal conductivity
 - can operate at 700-950 $^\circ\text{K}$
- (Molten Salts) - $\text{LiF} - \text{BeF}_2$ ("FLIBE")
 - Li_2BeF_4
 - $T_m \sim 637^\circ\text{K}$
 - has good chemical stability
 - TBR ~ 1.07 (Marginal!)

Note: "Advanced fuel" reactors do not need TBR-blankets
 [Advanced fuels = D-D, D- ${}^3\text{He}$, not D-T]

Ref. G. W. Hollenberg, et al. (ibid)

TABLE 1
CANDIDATE BLANKET MATERIALS

<u>Breeding Materials</u>	<u>Coolants</u>	<u>Structure</u>	<u>Neutron Multiplier</u>
<u>Liquid Metals</u>	Water	Austenitic Stainless Steel	Be
Li	(H ₂ O, D ₂ O)	Ferritic Steels	BeO
Li-Pb	<u>Liquid Metals</u>	Nickel-base Alloys	Pb
Li-Pb-Bi	Li	Refractory Alloys (e.g., V)	PbO
<u>Molten Salts</u>	Li-Pb		Bi
Flibe	Li-Pb-Bi		Zr
<u>Intermetallic Compounds</u>	<u>Molten Salts</u>		Zr ₅ Pb ₃
Li ₇ Pb ₂	Flibe		PbBi
<u>Solid (Ceramic)</u>	HTS		
Li ₂ O	<u>Gases</u>		
LiAlO ₂	He		
Li ₂ SiO ₃	Steam		
Li ₂ ZrO ₃			
Li ₂ TiO ₃			

Until about 1980, liquid-metal blankets were considered almost worldwide to be the leading option. In the course of the STARFIRE study,⁽¹⁾ the safety problems associated with liquid lithium received a great deal of attention and an advisory panel from electric utilities judged the safety problems as a serious objectional flaw in reactors using liquid lithium. Subsequent to the STARFIRE study, a significant R&D program on solid breeding materials was initiated as activities on liquid metals were reduced in the United States. Japanese R&D activities have had a singular emphasis on Li₂O throughout the 1970's and 1980's, while activities in Europe, until only recently, have solely emphasized liquid metals. Today, in programs such as the Blanket Comparison and Selection Study (BCSS)⁽²⁴⁾ a reassessment of critical issues is causing a reevaluation of both liquid metal concepts and solid breeder concepts.

Tritium Production Rate

$$\text{Tritium "production" rate} = \underbrace{(\text{tritium breeding ratio})}_{R_B} \cdot (\text{tritium consumption rate}),$$

$$\text{where tritium "consumption" rate} = \frac{P(\text{reactor thermal power including blanket reactions})}{W'_{DT}}$$

$$\text{with } W'_{DT} = 1.6 \times 10^{-13} (3.5 + 14.1 M) \text{ (J)}$$

$$= 3.2 \times 10^{-12} \text{ J}$$

$M = \text{blanket energy gain} \approx 1.2$

$$\text{If } P = 2.5 \text{ GWth, tritium consumption rate} = 7.8 \times 10^{20} \text{ atoms/s}$$

$$= 0.34 \text{ kg (Tritium)/day,}$$

$$\text{then, tritium production rate} = \begin{cases} (1.2) (7.8 \times 10^{20} \text{ atoms/s}) \\ = 9.4 \times 10^{20} \text{ atoms/s} \\ = 0.41 \text{ kg/day} \end{cases}$$

$R_B = 1.2$

Note that the tritium decays radioactively, λ , is

$$\lambda = \ln 2 / t_{1/2} = 0.693 / 12.3 \text{ yr} = 1.787 \times 10^{-9} \text{ s}^{-1}$$

The total number of tritium atoms in the system,

$$\frac{dN}{dt} = \underbrace{R_B \frac{P}{W'_{DT}}}_{\text{production rate}} - \underbrace{\frac{P}{W'_{DT}}}_{\text{consumption rate}} - \underbrace{\lambda N}_{\text{radioactive decay}}$$

→ want to find out N and doubling time (t_2)

Eq. (12.7)

$$\frac{dN}{dt} = \left(R_B \frac{P}{W_{DT}} - \frac{P}{W_{DT}} \right) - \lambda N$$

set $R = \frac{P}{W_{DT}} (R_B - 1)$

$$dN = R dt - \lambda N dt$$

$$[dN + \lambda N dt = R dt] \times e^{-\lambda t}$$

$$e^{-\lambda t} dN + \lambda N e^{-\lambda t} dt = R e^{-\lambda t} dt$$

$d(N e^{-\lambda t})$

$$\int_{N=N_0}^{N=N} d(N e^{-\lambda t}) = R \int_{t=0}^{t=t} e^{-\lambda t} dt$$

$$N e^{-\lambda t} \Big|_{t=0}^{t=t} = \frac{R}{\lambda} e^{-\lambda t} \Big|_{t=0}^{t=t}$$

$$N e^{-\lambda t} - N_0 e^{-\lambda(0)} = \frac{R}{\lambda} (e^{-\lambda t} - 1)$$

$$N = N_0 e^{-\lambda t} + \frac{R}{\lambda} (1 - e^{-\lambda t})$$

$$\boxed{\frac{N}{N_0}} = e^{-\lambda t} + \left(\frac{R}{\lambda N_0} \right) (1 - e^{-\lambda t}), \dots \text{Eq. (12.8)}$$

set $\frac{1}{x}$

$$\text{where } x = \frac{\lambda N_0}{R} = \lambda N_0 \cdot \frac{W_{DT}}{P(R_B - 1)} \dots \text{Eq. (12.9)}$$

Note: Activity, $\mathcal{A} = \lambda N$,

$$\mathcal{A} = \mathcal{A}_0 e^{-\lambda t} + R (1 - e^{-\lambda t})$$

Solving Eq. (12.8) for t , one gets

$$t = \frac{1}{\lambda} \ln \left[\frac{1-x}{1-x \frac{N}{N_0}} \right] \text{ (sec)} \dots \text{Eq. (12.10)}$$

If $R_B > 1$, tritium increases.

Define $t = t_2$ @ $\frac{N}{N_0} = 2$:

($t_2 =$ doubling time)

$$t_2 = \frac{1}{\lambda} \ln \left[\frac{1-x}{1-x \frac{N}{N_0}} \right] \Big|_{\frac{N}{N_0} = 2}$$

$$= \frac{1}{\lambda} \ln \left[\frac{1-x}{1-2x} \right]$$

Note: $\ln(1-x) = -x - \frac{x^2}{2} - \frac{x^3}{3} \dots$

$$t_2 = \frac{1}{\lambda} \left[\ln(1-x) - \ln(1-2x) \right]$$

$$= \frac{1}{\lambda} \left[\left(-x - \frac{x^2}{2} - \frac{x^3}{3} \dots \right) - \left(-2x - \frac{(2x)^2}{2} - \frac{(2x)^3}{3} \dots \right) \right]$$

$$= \frac{1}{\lambda} \left[(-x + 2x) - \left(\frac{x^2}{2} \right) + \left(\frac{4x^2}{2} \right) - \frac{x^3}{3} + \frac{8x^3}{3} \dots \right]$$

$$= \frac{1}{\lambda} \left[x + \frac{3x^2}{2} + \frac{7x^3}{3} + \dots \right]$$

$$= \frac{x}{\lambda} \left[1 + \frac{3x}{2} + \frac{7x^2}{3} + \dots \right] \dots \text{Eq. (12.11)}$$

Note: For comparison with fission decay, e.g., Pu-239 & U-233 , set $\lambda \approx 0$ as decay rates are small (long $t_{1/2}$):

$$\frac{N}{N_0} = 1 + \frac{Pt}{N_0 W} (R_B - 1),$$

$$t_2 = N_0 W / [P(R_B - 1)] \text{ (s)} \dots \text{Eq. (12.12)}$$

where $W = 200 \text{ MeV} = 3.2 \times 10^{-11} \text{ J}$

Example Prob. 12.2

Estimate the fuel doubling times of (a) a 2.5 GW_{th} fusion reactor with 10 kg of Tritium, and (b) a 2.5 GW_{th} fission breeder with 10⁴ kg of fuel, assuming both having breeding ratio, R_B = 1.2.

(a) For the fusion reactor,

$$N_0 = 10 \text{ kg} / (5.008 \times 10^{-27} \text{ kg/atom}) = 2.00 \times 10^{27} \text{ atoms (tritium)}$$

$$\text{Then } X = \frac{W_{DT}' \lambda N_0}{P (R_B - 1)} = \frac{(3.2 \times 10^{12} \text{ J}) (1.787 \times 10^{-9} \text{ Bq}) (2.00 \times 10^{27} \text{ atoms})}{(2.5 \times 10^9 \text{ J/s}) (1.2 - 1)} = 2.29 \times 10^{-2}$$

where $W_{DT}' = 3.2 \times 10^{12} \text{ J}$ (including the blanket energy multiplication),

$$\text{and } \lambda = \frac{\ln 2}{t_{1/2}} = \frac{0.693}{12.3 \text{ yr}} = 1.787 \times 10^{-9} \text{ Bq},$$

$$\text{and the doubling time, } t_2 = \frac{X}{\lambda} \left(1 + \frac{3X}{2} + \dots \right) \dots \text{Eq. (12.11)}$$

$$= \frac{2.29 \times 10^{-2}}{1.787 \times 10^{-9} \text{ Bq}} \left(1 + \frac{3 \times 2.29 \times 10^{-2}}{2} + \dots \right) = 1.32 \times 10^7 \text{ sec} = \text{5.1 months}$$

(b) For the fission reactor,

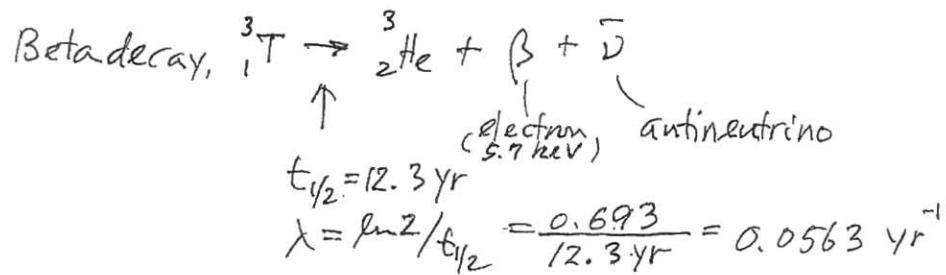
$$N_0 = 10^4 \text{ kg} / [(238 \text{ u}) (1.66 \times 10^{-27} \text{ kg/u})] = 2.53 \times 10^{28} \text{ atoms}$$

$$t_2 = N_0 W / [P (R_B - 1)] = \frac{2.53 \times 10^{28} \text{ atoms} \cdot (3.2 \times 10^{11} \text{ J})}{2.5 \times 10^9 \text{ J/s} (1.2 - 1)} \dots \text{Eq. (12.12)}$$

$$= 1.62 \times 10^9 \text{ sec}$$

$$= \text{51 yrs}$$

Prob. 12.1 Pure tritium in a non-permeating, sealed bottle is stored for 1 year. Find the fraction of the tritium in helium at the end of the year.



$$N = N_0 e^{-\lambda t}, \quad t = 1 \text{ yr}$$

$$\frac{N}{N_0} = \exp[-(0.0563 \text{ yr}^{-1})(1 \text{ yr})] = 0.9452$$

$$N = 0.9452 N_0 \quad (\# \text{ of tritium atoms remaining after 1 yr.})$$

$$\text{Let } N' = (1 - 0.9452) N_0 = \# \text{ of He-atoms produced after 1 yr.}$$

$$N' = 0.0548 N_0$$

$$\therefore \frac{\# \text{ of He-atoms present}}{\text{total \# of atoms}} = \frac{N'}{N_0} = 0.0548 = \textcircled{5.48\%}$$

Tritium permeation rates, J (atoms/m².s)

Consider tritium permeation through a wall of thickness x with tritium pressures, P_1 and P_2 , and the tritium concentration $C(x)$ within the wall.

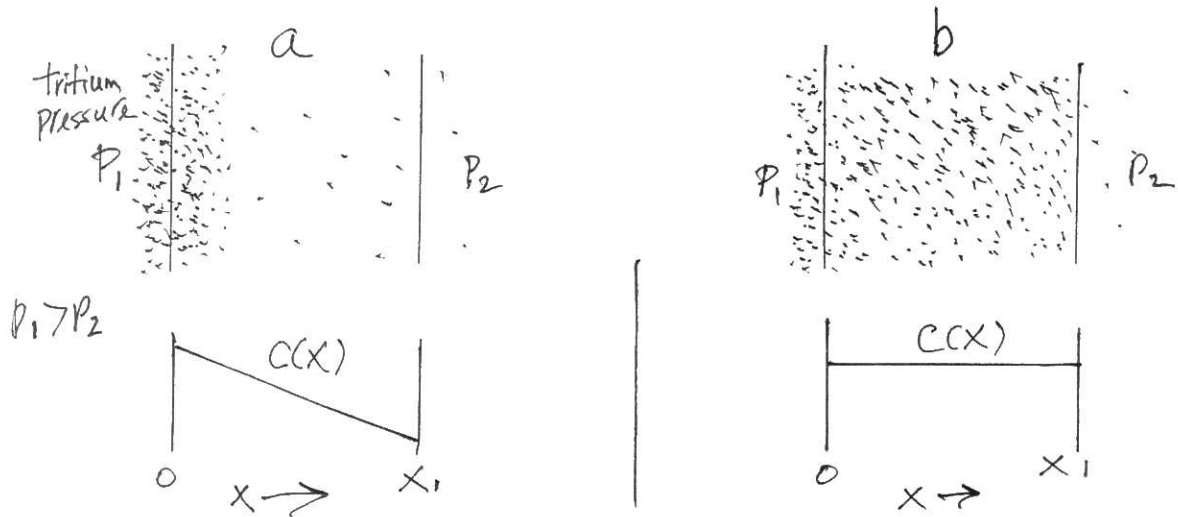


Fig. 12.2 Distribution of tritium atom concentration $C(x)$ in a tube wall. a) permeation rate limited by diffusion in the wall; b) permeation rate limited by surface recombination.

At high pressure P_1 , or with the fluids (like liquid metals), diffusion through the wall is the flow rate-limiting process:

$$C(x_1) = S P_2^{1/2} \text{ (atoms/m}^3\text{)}$$

$$J_{diff} = -D \left(\frac{dC}{dx} \right) = -D \left[\frac{C(x_1) - C(0)}{x_1} \right] = DS (P_1^{1/2} - P_2^{1/2}) / x_1 \text{ (atoms/m}^2\text{.s)}$$

with D = diffusion coeff. (m²/s),
 S = solubility of tritium in a given metal (atoms/m³Pa^{1/2})

and DS = permeation coeff. or permeability

[with $D \approx D_0 \exp[-E_d/kT]$ (m²/s) ... (12.19)
 $S \approx S_0 \exp[-E_s/kT]$ (atoms/m³Pa^{1/2})
 "activation energies"
 $E_d, E_s < 1 \text{ eV}$... (12.20)]

At low pressure, P_1 , surface molecular recombination is tritium flow rate-limiting process:

(cf. ch. 8 - spontaneous desorption flux = $(K C^2(0,t))$ (atoms/m².s)
 ↑ recombination rate (m⁴/s)

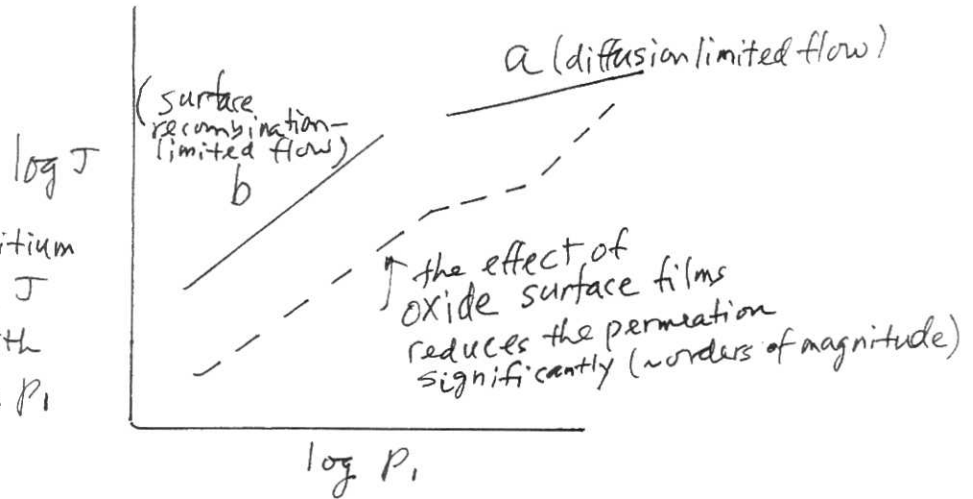
Note: $C(0) = (S) P_1^{1/2}$ (atoms/m³)
 ↑ solubility of tritium in given metal

$J_{rec} = (K) C^2(x_1) \leq K C^2(0)$
 " $K S^2 P_1$ (atom/m².s)
 ↑ surface recombination coefficient

with $K = \frac{\alpha}{M} (C_1 d / D_0^2) \exp[(2E_s - E_x)/kT]$ (M/s)
 [atomic weight (inamu) of metal] (Dolan, 2.22)
 $C_1 = 2.60 \times 10^{-24} K^{1/2} \mu^{1/2} / \text{Pa} \cdot \text{s} \cdot \text{m}^2$... (12.23)

and $E_x = \begin{cases} E_s + E_d & \text{if } E_s + E_d > 0 \\ 0 & \text{otherwise} \end{cases}$... (12.24)
 [$\alpha = 0.5$ sticking coefficient]

Fig. 12.3
Variation of tritium permeation rate, J (atoms/m².s) with driving pressure P_1



Example 12.3: A steam generator with total tube surface area 2000 m² has Ni tubes 2-mm thick at 800 °K, with a partial pressure of 10⁻⁵ Pa tritium on the primary coolant side and a negligible tritium pressure on the steam side. Estimate the leakage of tritium (Ci/day) into the steam system.

Ans. Calculate both diffusion-limited flow and recombination-limited flow to find out which is lower.

For diffusion-limited flow:

$$J_{diff} = DS (P_1^{1/2} - P_2^{1/2}) / x_1 \quad (\text{atoms/m}^2\text{.s})$$

where $D \approx D_0 e^{-E_d/kT}$ (m²/s)

$$S \approx S_0 e^{-E_s/kT} \quad (\text{atoms/m}^3 \text{Pa}^{1/2})$$

↓ algebra

For recombination-limited flow:

$$J_{rec} = K c^2(x_1) \leq K c^2(x_0) = K S^2 P_1 \quad (\text{atoms/m}^2\text{.s})$$

where $K \approx \sqrt{\frac{8}{\pi}} \left(\frac{c_1 \alpha / S_0^2}{M T} \right) e^{(2E_s - E_x)/kT}$ (m⁴/s)

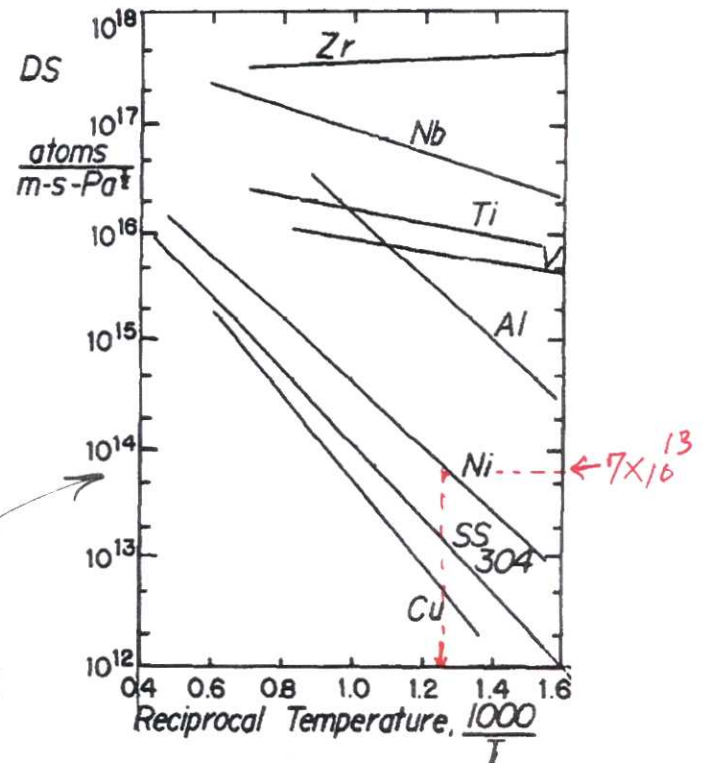
↓ algebra

Table 12.4 Diffusion and solubility coefficients for hydrogen in various metals. Some values of S_0 and D_0 have large errors (~50%) (Baskes 1980; Cecchi 1979; Perkins 1973)

Metal	S_0 10 ²³ atoms/ m ³ Pa ^{1/2}	E_s eV	D_0 10 ⁻⁷ m ² /s	E_d eV
Ni	9.7	0.16	4.0	0.41
Fe(bcc)	6.3	0.28	0.78	0.08
Fe(fcc)	14.0	0.32	6.7	0.47
Al	5.6	0.66	110.0	0.43
Ti	4.7	-0.50	18.0	0.54
Zr	20.0	-0.63	4.2	0.41
Mo	4.2	0.22	4.8	0.39
304 SS	1.1	0.061	4.7	0.56
Inconel 625	2.2	0.13	7.6	0.50
Cu	4.8	0.40	11.0	0.40

→ $DS = 9.5 \times 10^{13}$ atoms/mPa^{1/2} (hydrogen) → $K \approx 4.8 \times 10^{-28}$ (m⁴/s)
 $= 5.5 \times 10^{13}$ (tritium)
 $\therefore J_{diff} = (8.7 \times 10^{13})$ (atoms/m².s) ($w/P_1 \ll P_2$) $\therefore J_{rec} \leq (4.4 \times 10^{13})$ (atoms/m².s)

Fig. 12.4 Permeation coefficient versus reciprocal temperature, for hydrogen in various metals. T is in Kelvin (Fraas 1975)



Note the DS @ 800 °K for Ni to read 7×10^{13} ($\frac{\text{atoms}}{\text{m-s-Pa}^{1/2}}$)
 [However, $8.7 \times 10^{13} \frac{\text{atoms}}{\text{m}^2 \cdot \text{s}}$ from J_{diff} calculation is more accurate.]

By comparison between J_{diff} (i.e., $8.7 \times 10^{13} \text{ atoms/m}^2 \cdot \text{s}$) and J_{rec} (i.e., $4.4 \times 10^{13} \text{ atoms/m}^2 \cdot \text{s}$), the flow is recombination limited (lower value).

$$\text{The total atom flow rate, } J_{rec} \cdot A \leq (4.4 \times 10^{13}) \cdot (2 \times 10^3 \text{ m}^2) \\ = 8.8 \times 10^{16} \text{ (atoms/s)}$$

$$\text{Hence, the activity, } \lambda J_{rec} A \leq 1.6 \times 10^8 \text{ Bq/s} = \underline{360 \text{ Ci/day (ck algebra)}}$$

Note: For more accurate estimate for the tritium permeation rate, one finds $\frac{1}{J} \approx \frac{1}{J_{diff}} + \frac{1}{J_{rec}}$ (since two rate-limiting processes are in series) $\approx 2.9 \times 10^{13} \text{ (atoms/m}^2 \cdot \text{s)}$, but use the "smaller" of J_{diff} and J_{rec} for a conservative estimate, i.e., $4.4 \times 10^{13} \frac{\text{atoms}}{\text{m}^2 \cdot \text{s}}$

Note: Sources of parameter uncertainties — cracks in materials might have caused higher permeation rates, but at the same time oxide films could reduce permeation rate by orders of magnitude.

"Barrier" coolant loop may be used to reduce tritium leakage into the steam generator (see Fig. 12.5 next page).

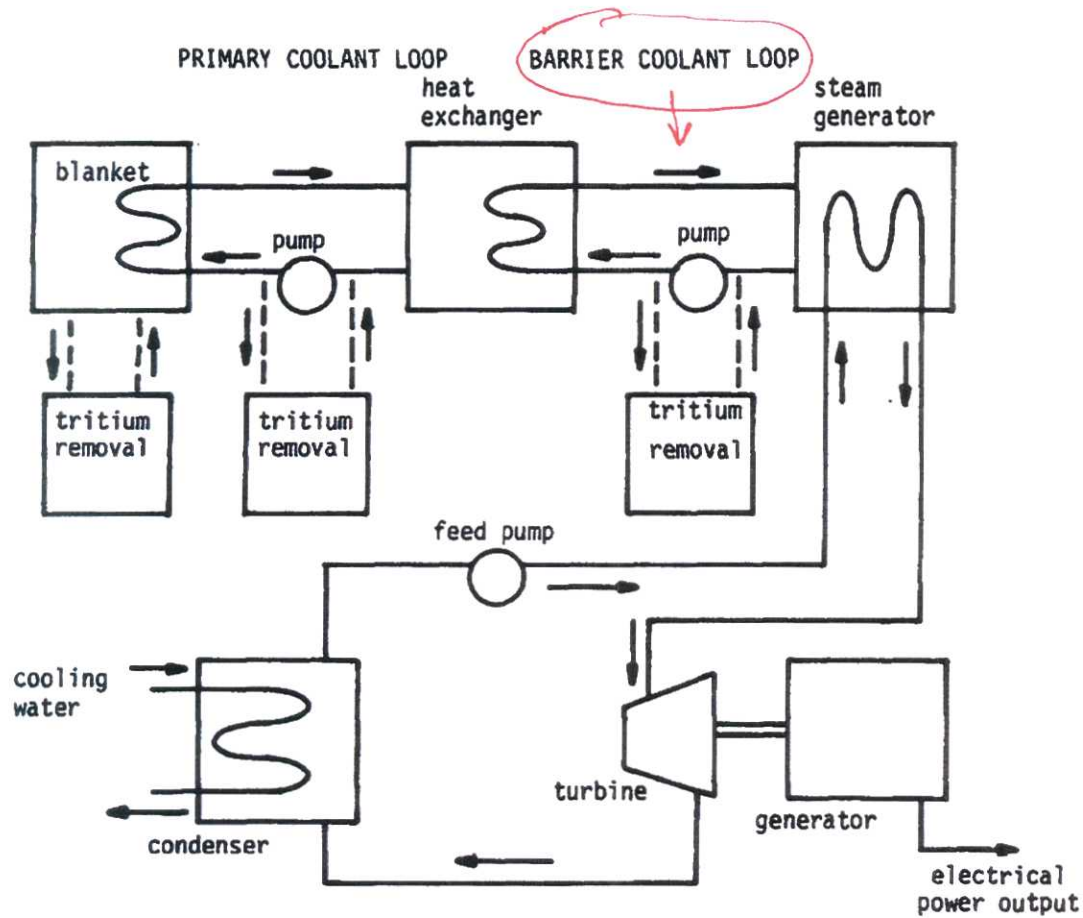


Fig. 12.5 Flow diagram for a system using an intermediate "barrier" coolant loop to reduce tritium leakage into the steam system

Tritium Recovery systems : Tritium can be removed from the vacuum system by cryogenic distillation or by diffusion through permeable membranes.

Table 12.5 Some tritium removal methods

Fluid	Removal method
Water	Boiling temperature difference
Water (ITER)	Electrolysis and catalysis
PbLi	Diffusion into He gas and extraction from He
PbLi	Permeation through window (such as Nb) into tritium removal tubes
He	Oxidation by O ₂ , then condensation in cold trap
He	Diffusion through Nb or Pd window into vacuum (only if high tritium pressures are allowable)
He	Solid sorbents, if high tritium pressures are allowable
Molten salt	Spray droplets into vacuum
Molten salt	Bubble He through molten salt

Tritium Facilities (References included)

1. "Tritium Technology Programs in the United States,"
(James L. Anderson, Los Alamos National Laboratory)
(1991) - LANL Technology since 1988 advancement,
TSTA (Tritium Systems Test Assembly) at LANL;
Savannah River Site (SRS), EG & G Mound Laboratory,
Lawrence Livermore National Laboratory (LLNL), Sandia
National Laboratory at Livermore, California, and also
in Los Alamos National Laboratory (Weapons Engineering
Tritium Facility (WETF) in 1991, Tritium Salt Facility
(TSF), Tritium Systems Test Assembly (TSTA) in early 1980s
2. "Fusion tritium research facilities in KAERI" (2012)
(H. Chung, et al.) - HANARO (High-flux Advanced Neutron
Application Reactor), ELLI (Experimental Loop for Liquid Breeder)
3. "Tritium Technology for ITER" [Full Report attached]
(S. Willms, Karlsruhe, 2016)

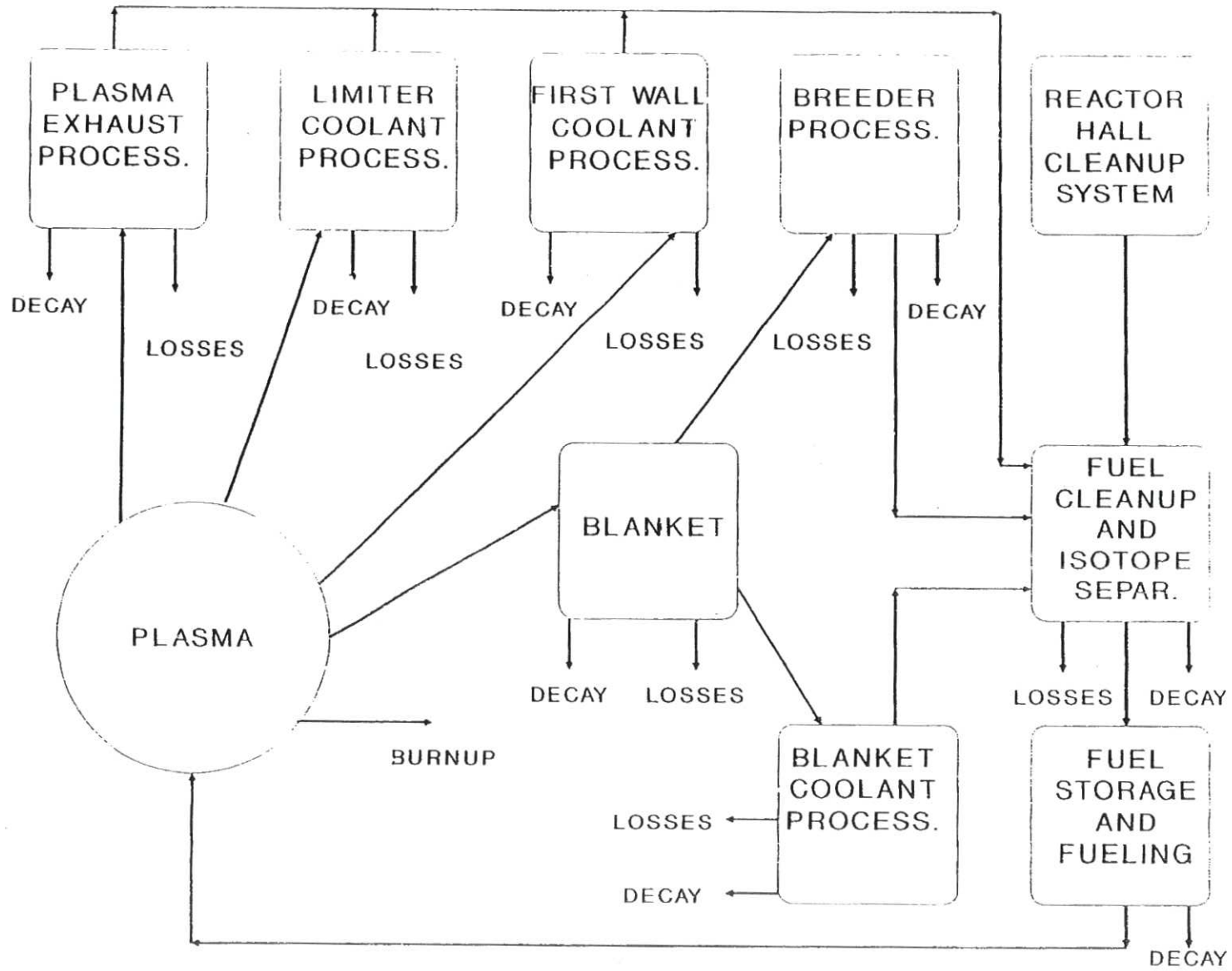
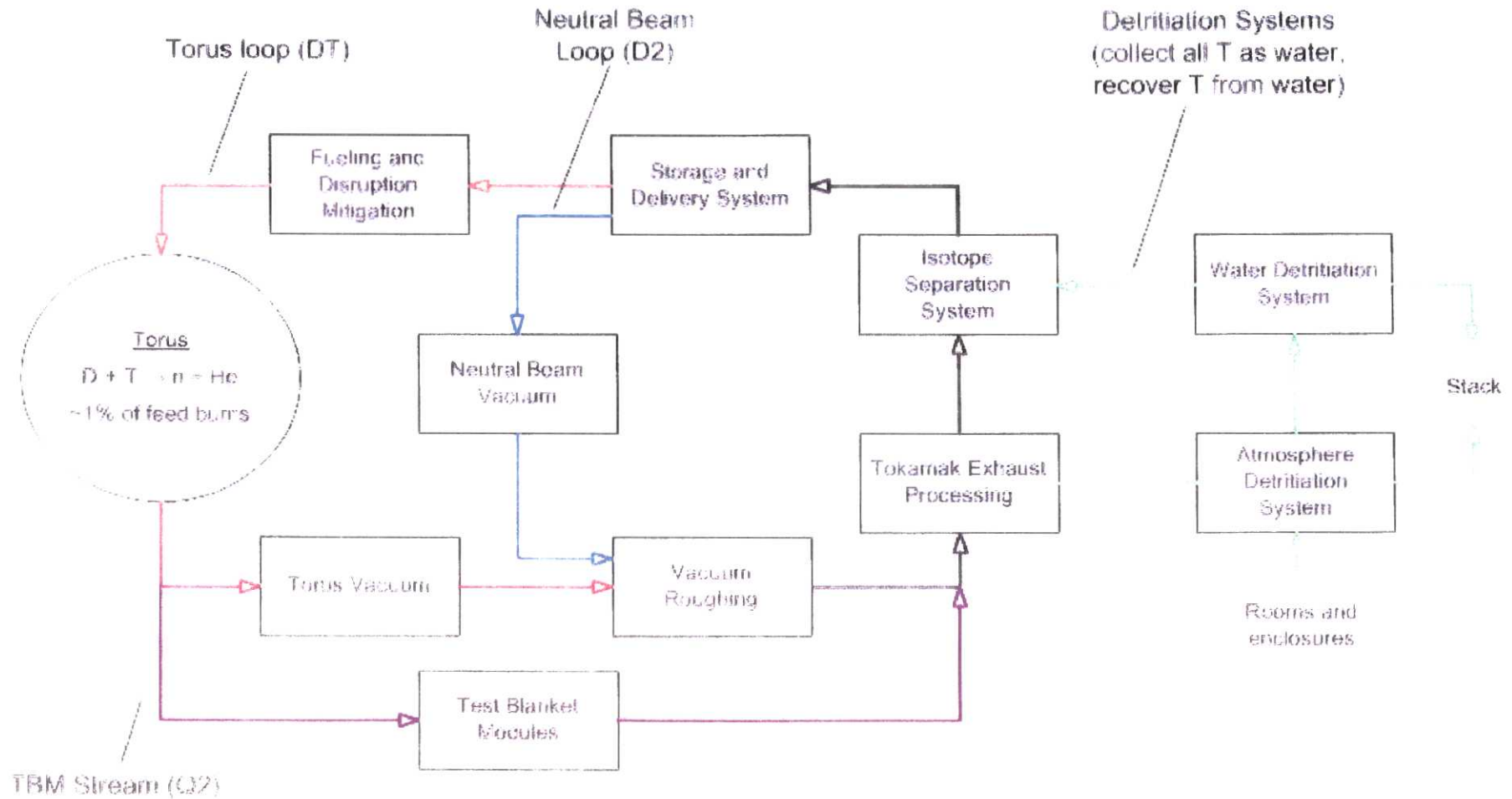


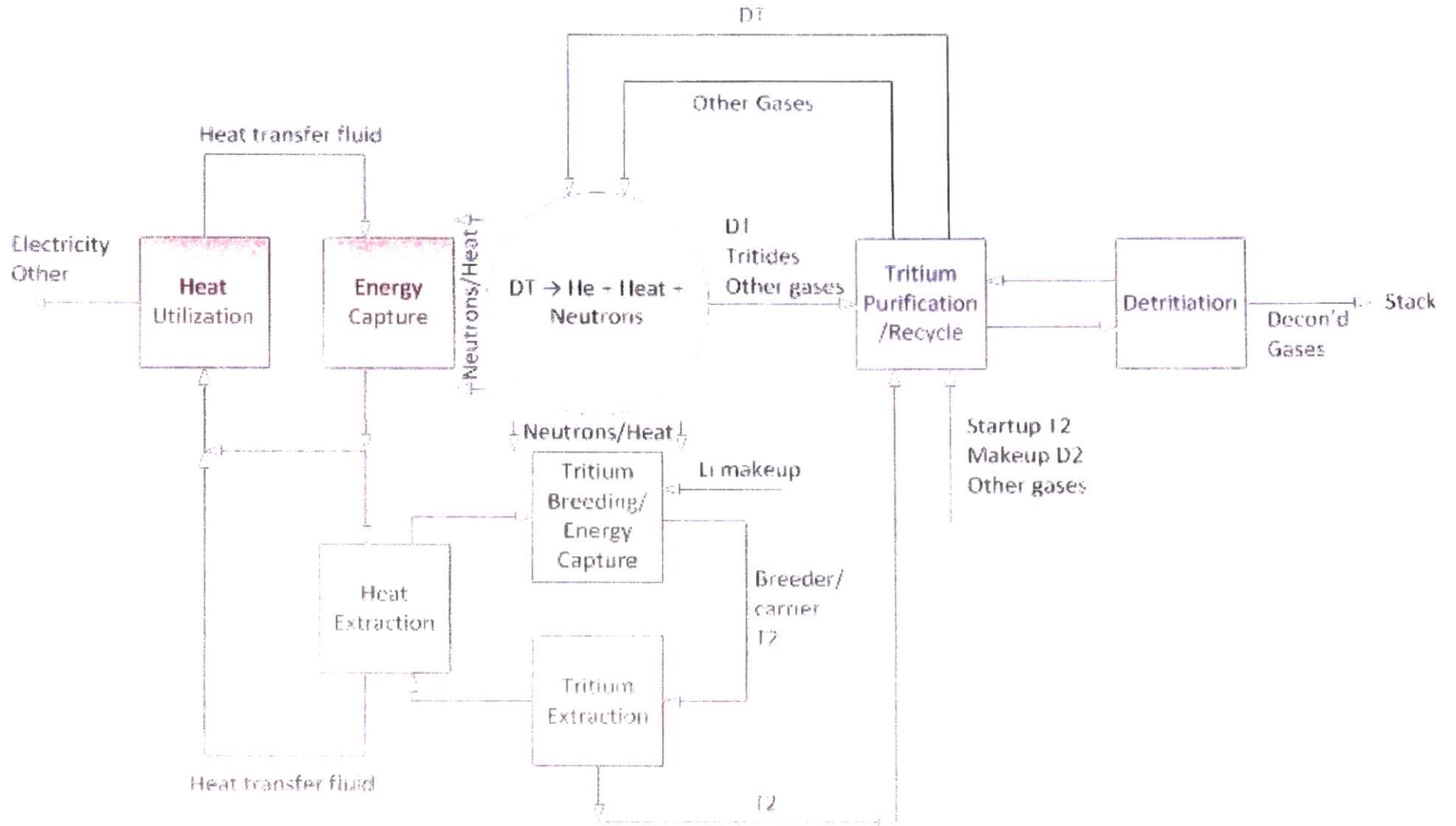
Figure 1. Schematic arrangement of a generic, DT fusion power reactor tritium processing system.

ITER Fuel Cycle

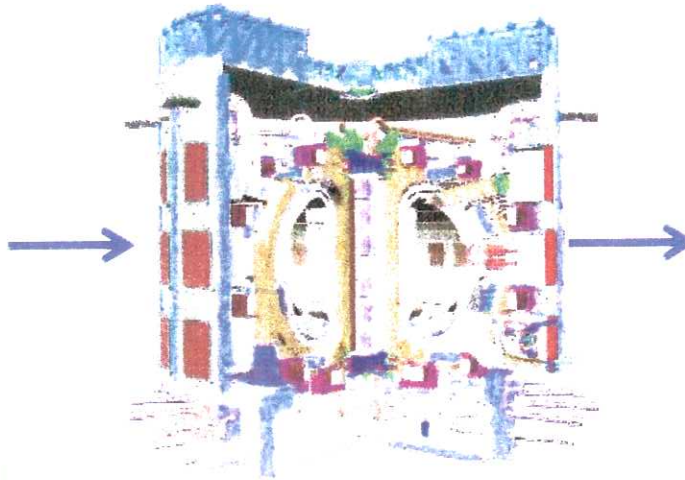
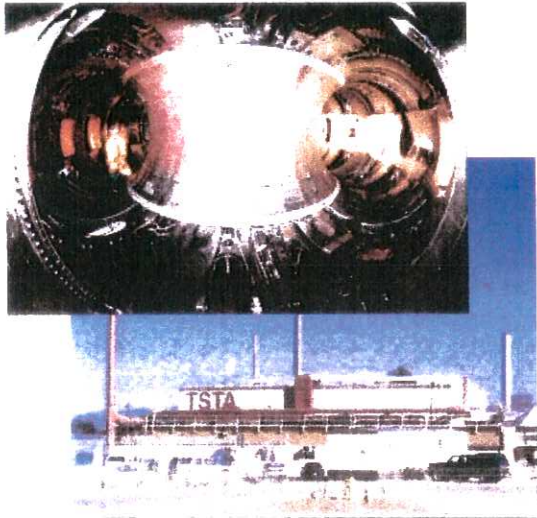
- DT loop fuel tokamak with T2 and D2
- D2 loop supplies neutral beams with D2 (and some H2)
- Effluent detritiation oxidizes all hydrogen isotopes to water, recovers tritium from the water and exhausts the detritiated gases.
- TBM Stream recovers T and other gases from TBM



Generic overview of fusion fuel cycle mass and energy flows



Current and planned key fusion tritium fuel cycle development points



DEMO

State-of-the-art

Fusion power: 10's MW
Burn fraction: Nil

Pulse length: seconds
Annual duty cycle: ~5%

Breeding: None

ITER/ITER-TBM

Fusion power: 500 MW
Burn fraction: 0.3%

Pulse length: 3000 s
Annual duty cycle: 5%

Breeding: TBM testing

DEMO

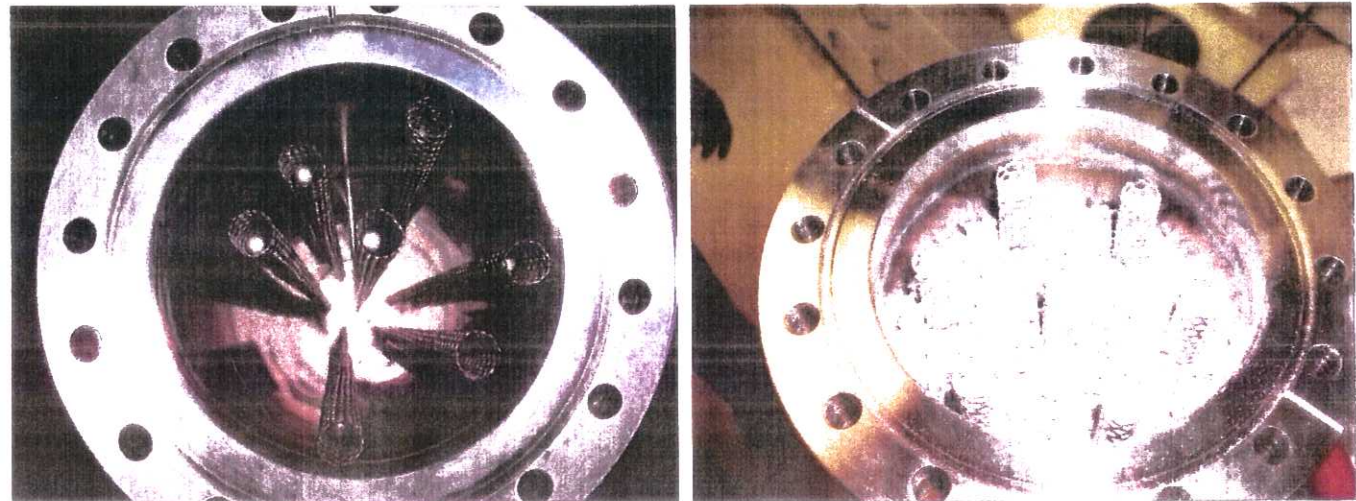
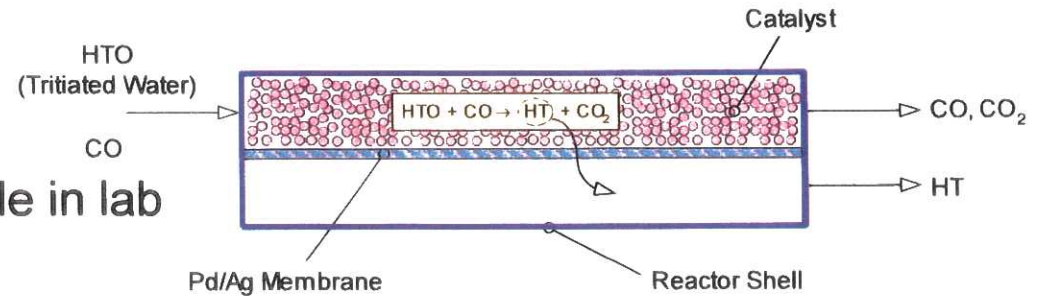
Fusion power: 2000 MW
Burn fraction: >3%?

Pulse length: Continuous
Annual duty cycle: 50%

Breeding: Self-sufficient

Palladium membrane reactor

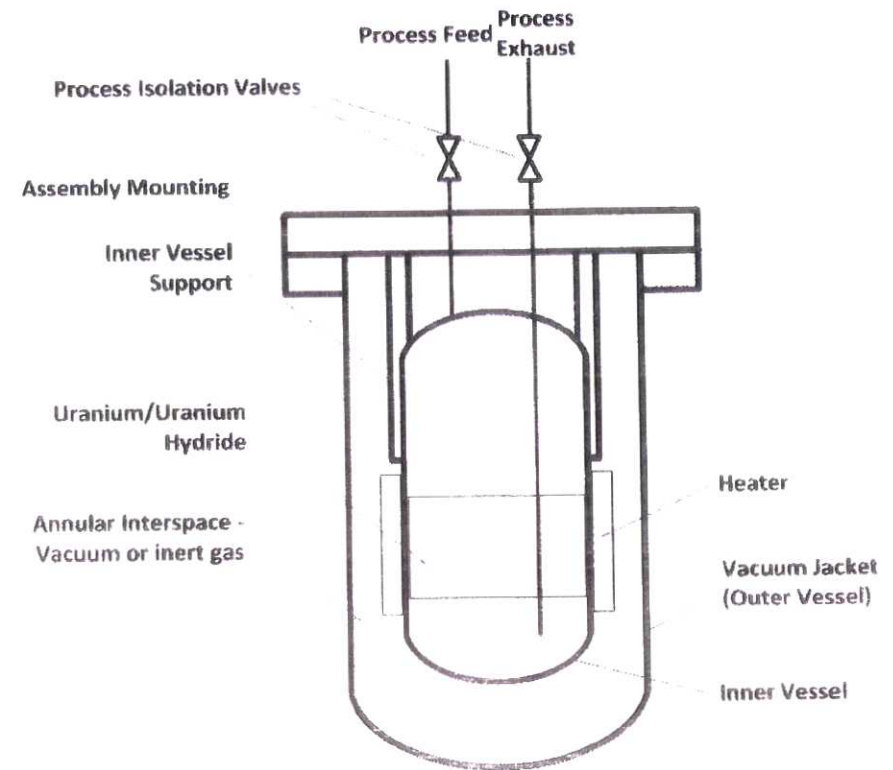
- Without generating solid waste, recover hydrogen isotopes from:
 - Water
 - Hydrocarbons
 - Ammonia
- Successfully tested at ITER scale in lab
- Considering tests at JET



Seven tube PMR before and after loading catalyst

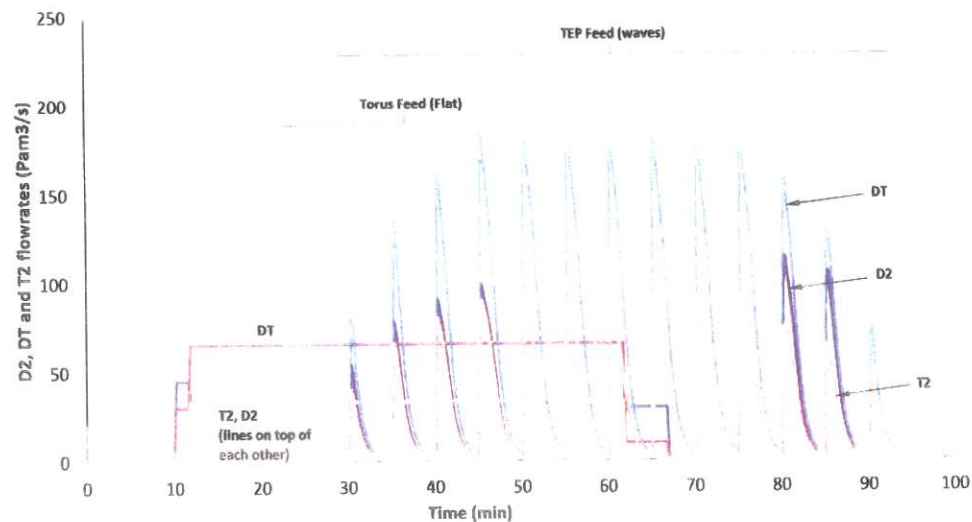
Uranium hydride beds

- Used successfully to store hydrogen isotopes including tritium
- Store tritium at room temperature with negligible gas pressure, deliver hydrogen isotopes at 400 – 500 C.
- Generates gas pressure with no moving parts
- Tested with tritium at ITER scale

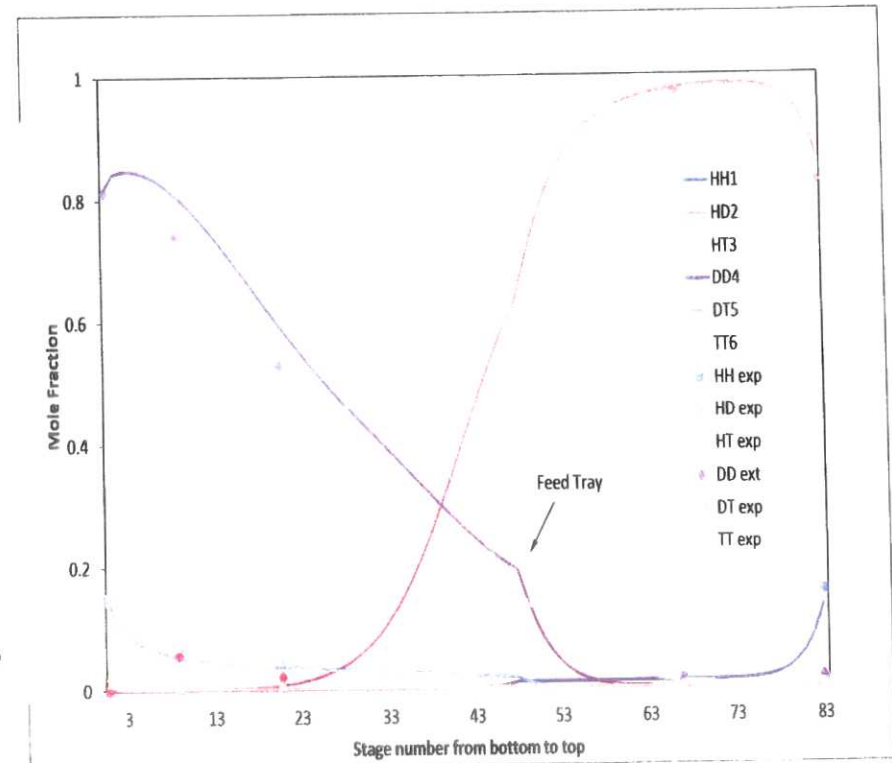


Time-dependent Fuel Cycle system modeling

- Commercially available chemical engineering software (Aspen) adapted to tritium system modeling. Highlights include:
 - Cryogenic distillation
 - Flow and composition determination throughout cycle



Torus feed (ramp up/flat top/ramp down) becomes highly variable feed following cryogenic pumping and regeneration



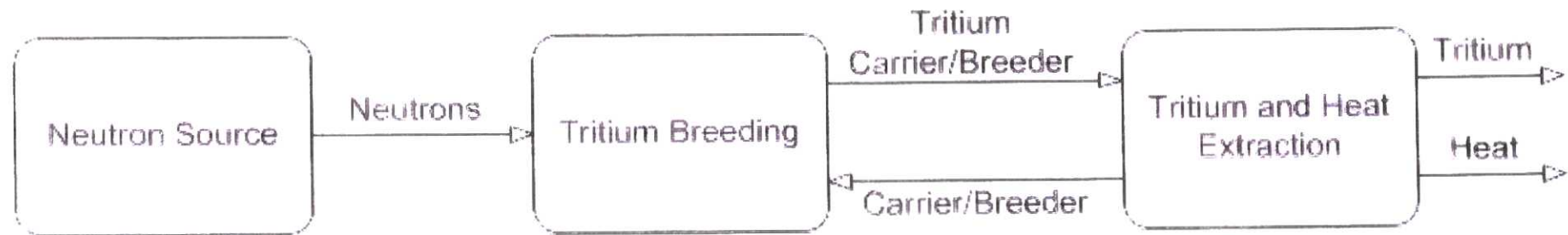
Cryogenic distillation column concentration profile matches favorably with experiments

ITER Test Blanket Module (ITER-TBM)

System characteristics

- Operate in extended burn and dwell with daily interruptions for 6 days
- Include tritium containment systems
- Large ITER tritium inventory and small TBM inventory

Sub-system characteristics



- 14 MeV source
- Varying neutron flux

- Small modules (~1 m²)
- Realistic fusion environment
- Must consider non-TBM interactions
- Modules changeable during maintenance periods

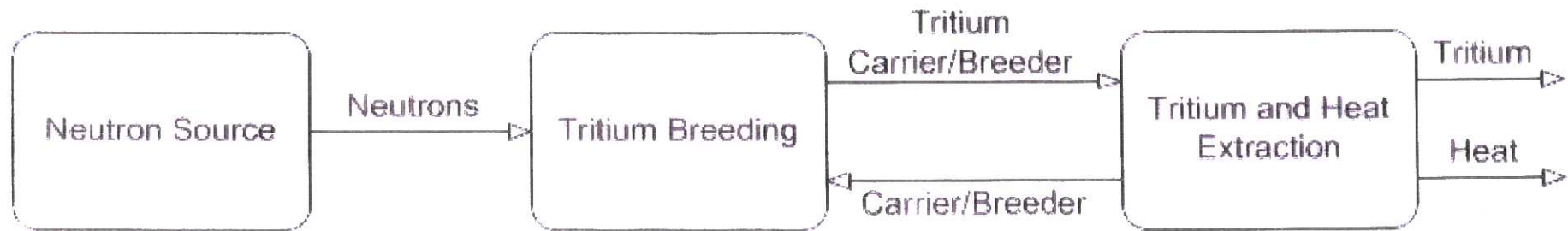
- Include tritium extraction and heat extraction technology
- Technology changeable during maintenance periods

Tritium Breeding and Extraction Facility (TBEF)

System characteristics

- Operate continuously until steady state achieved (1-4 wks)
- Include tritium containment systems
- Relatively small tritium inventory

Sub-system characteristics



- Not necessary to be 14 MeV source
- Controlled neutron flux

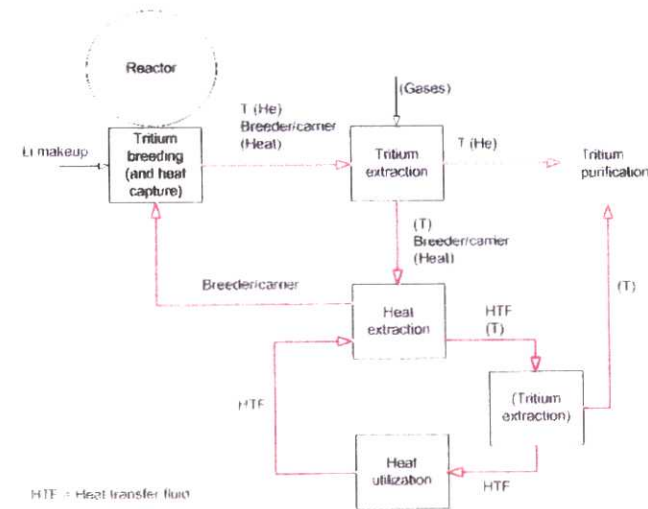
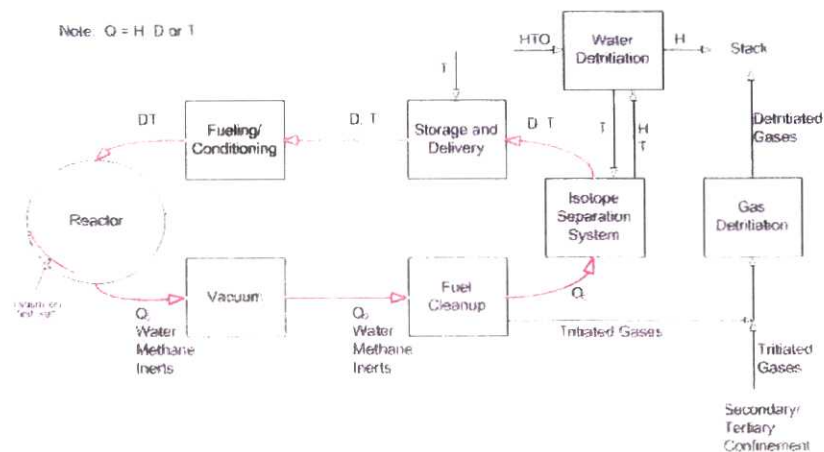
- Small modules, but large enough to display system behavior (~1 m²)
- Modules readily changeable
- Controlled temperature/heat input

- Include tritium extraction and heat extraction technology
- Technology readily changeable

Fuel Cycle Development Facility (FCDF)

System characteristics

- Include all purification/recycle and extraction sub-systems
- Constructed at approximate 1/5th DEMO scale
- Used to develop sub-systems as well as the overall system
- Conducive to low-cost technology development since
 - Not directly tied to support machine operations (ITER or FNSF)
 - Not operating with tritium
- Test safety systems without risking actual tritium release
- Can test beyond performance extremes (not tied to operating machine)
- Generate reliability
- Use for staff training



Conclusions

- The ITER Tritium Plant design and construction is a major undertaking
- Feasible technologies for each function have been identified
- Lessons being learned on ITER will be valuable for DEMO
- There is still tritium technology work beyond ITER that is needed to advance to DEMO.