Feasibility of Flibe/V-alloy Blanket

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**National Institute for Fusion Science*
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Fusion Power Plants and Related Advanced Technologies
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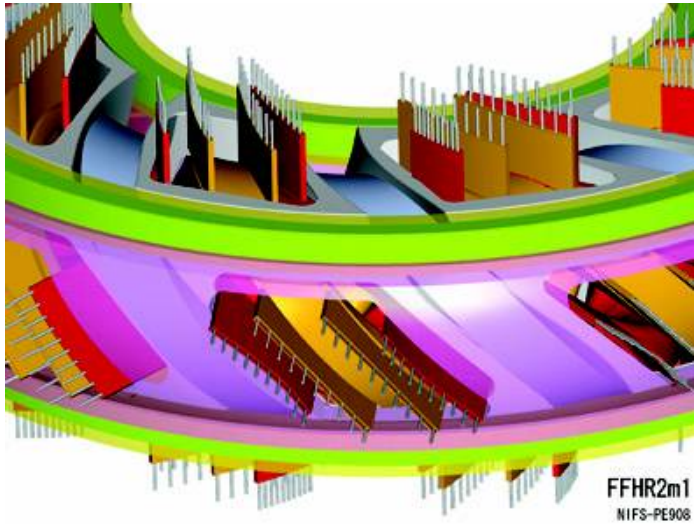
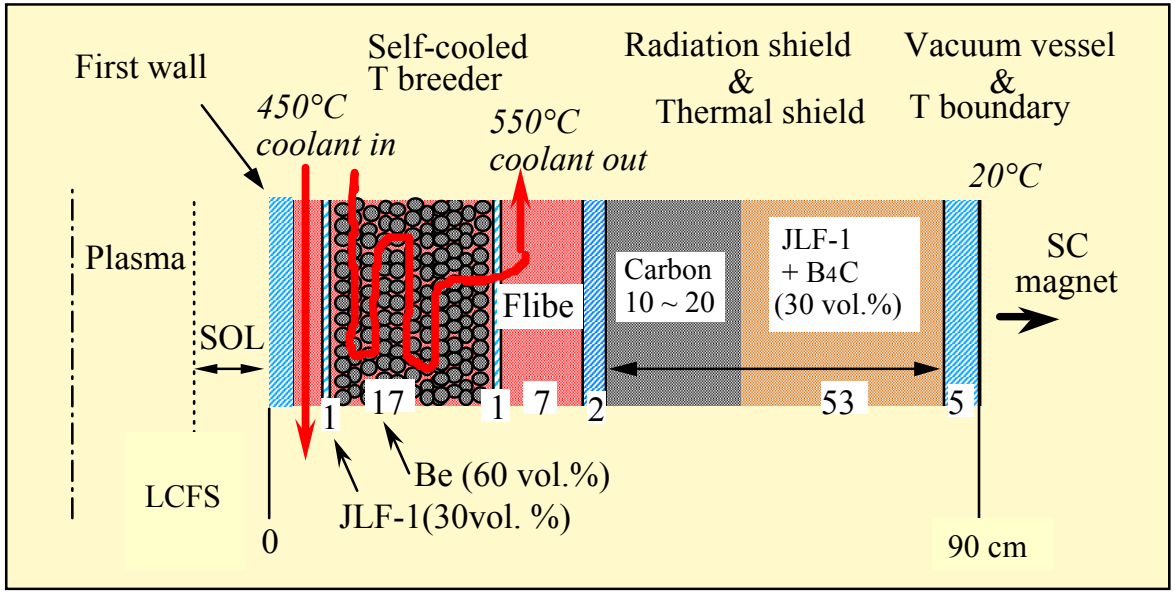
Original Design for Force Free Helical Reactor (FFHR-2)





Blanket structure of original **FFHR2**

Force Free Helical Reactor conceptual design FFHR2

- Blanket space: 90cm
- Structural material: JLF-1 (Reduced Activation Ferritic steel)
- Breeder/Coolant: Flibe + Beryllium
  - Neutron multiplication
  - Suppression of wall corrosion by REDOX control (TF => T_2)

Neutronics performance for original FFHR2

- Neutron wall load: 1.5MW/m^2
- Local TBR: 1.37
- Fast neutron flux (>0.1MeV) at SC magnet: 6.3 x 10^{10}n/cm^2/s.

Advantages and Issues for the present FFHR-2 Blanket (Flibe/ RAFM)

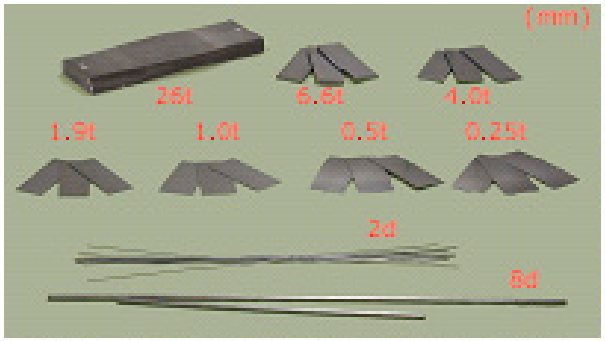
Advantage

- Low pressure, low reactivity, easy T recovery
- Smaller risk in materials development
- Corrosion suppression by REDOX control
  - (being verified in JUPITER-II)

Key Issues

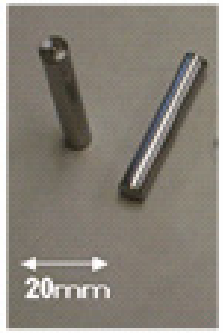
- Thermofluid, heat control
- Tritium permeation
- Low max. temperature, small ΔT (because of max. operation temp. of RAFM)
  - Increase in the operation temperature by using advanced materials seems to be the only solution

V-4Cr-4Ti Alloy Development

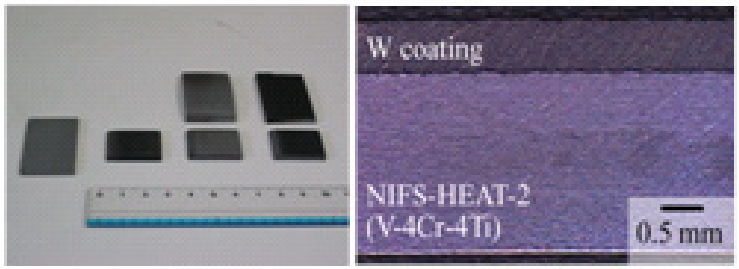


Plates, Sheets, Wires and Rods

Thin pipes

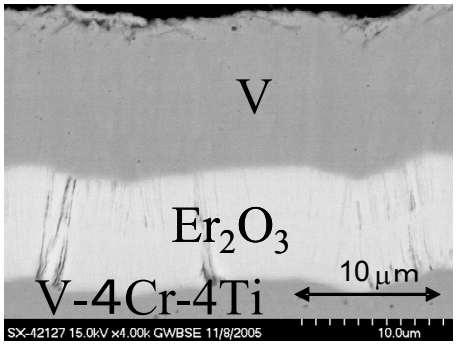


Creep tubes



W coating by plasma spraying

Laser weld joint



Leading concept of MHD insulator coating:

Products from high purity V-4Cr-Ti ingot (NIFS-HEAT-2)

V/Er_2O_3 double coating

What will happen if we change the structural materials from RAFM to V-alloy?

Improvements

Some increase in TBR because of (n 2n) reactions of V

increased potentiality not to use Be as multiplier (Part 1)

Higher max. temperature, larger ΔT

increase in efficiency

decrease in viscosity – improvement in pressure drop and thermofluid (Part 2)

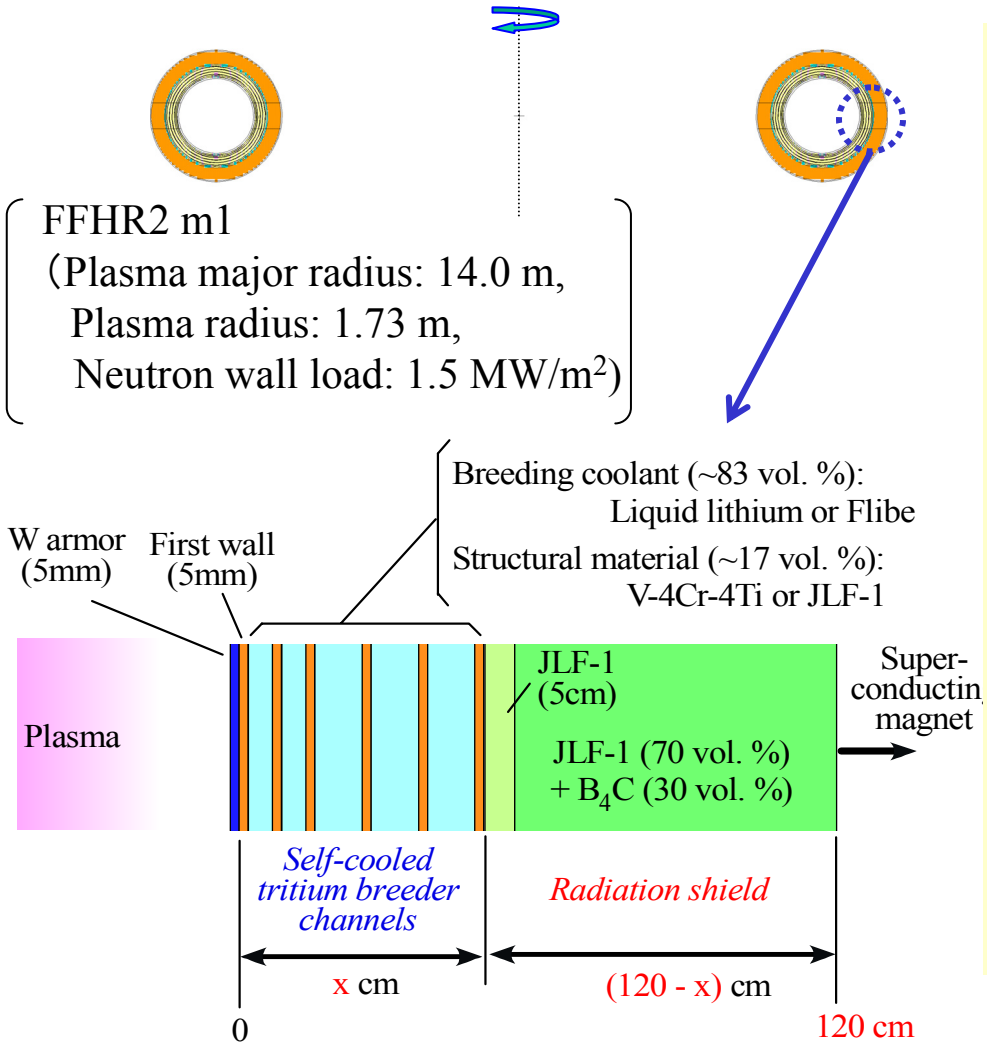
New Issues

Corrosion by impurities would be more serious

Extreme increase in T inventory in the structural material

alternative REDOX control will be proposed (Part 3)

Part 1 : Nuclear Calculation

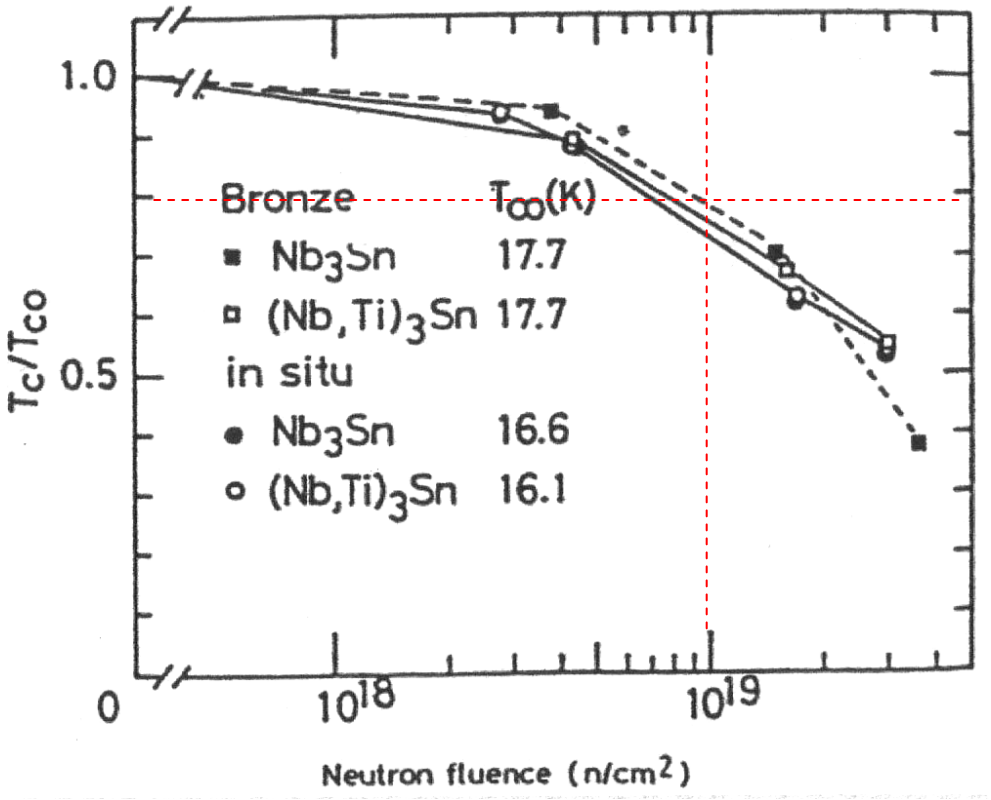


Geometric model of self-cooled liquid blanket system for nuclear calculation

- 3D calculation of torus fully covered with uniform blanket
- MCNP-4C code
  + JENDL 3.2 nuclear data
- Local TBR
  Fast neutron flux (> 0.1 MeV) at outside of radiation shield

  Investigation of compatibility in blanket space of 120 cm
- Radiation shield:
  JLF-1(70 vol.%) + B_4C(30 vol.%)
  the fraction will be the parameter

Target for the Shielding Performance



Fluence dependence of T_c of super-conducting materials*
(Research reactor)

Target in Present Calculation

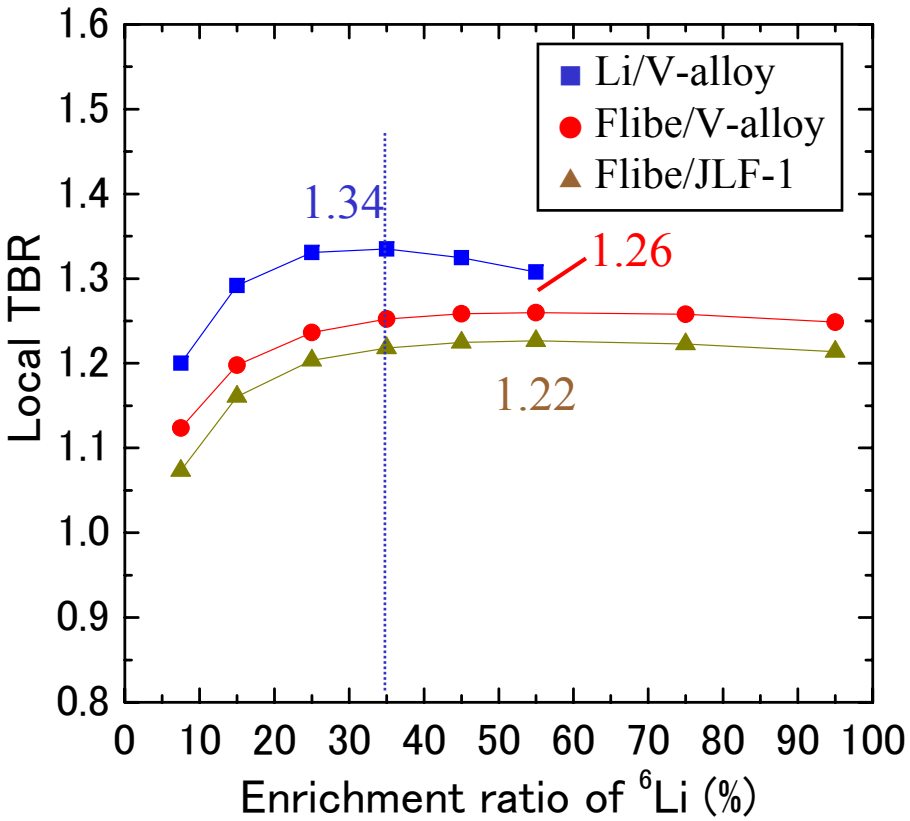
- Local TBR:
  Comparable to original FFHR2 (~1.37)
- Neutron Shielding:
  After 30 years operation, Tc/Tco > 0.8

↓

Fast neutron flux (>0.1 MeV) at outside of radiation shield
< 1.0 x 10^{10} n/cm^2/s

* T. Okada *et al.*, Proceedings of the 6th Japan-US workshop on high-field superconducting materials and standard procedures for high-field superconducting materials testing

Enhancement of Local TBR by ^{6}Li Enrichment



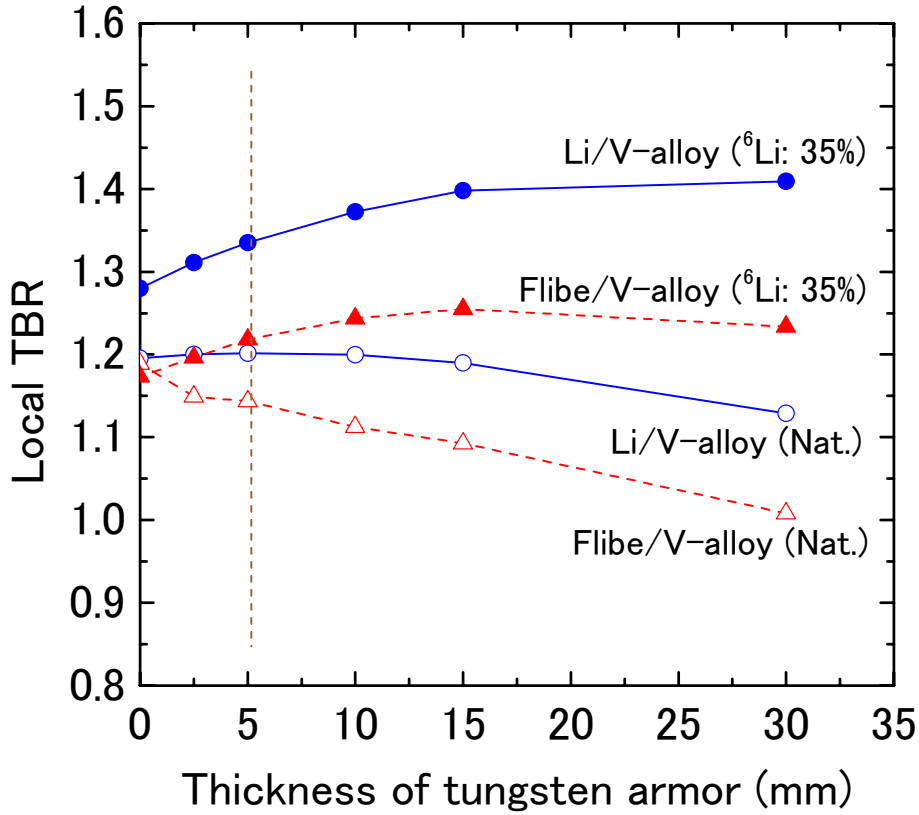
Effect of ^{6}Li enrichment on local TBR in blankets with breeding channels of 54 cm

- ^{6}Li enrichment to achieve comparable TBR with original FFHR2 (1.37)
- Li/V-alloy blanket
  ^{6}Li:35%, Local TBR: 1.34

  Flibe/V-alloy blanket
  ^{6}Li:55%, Local TBR: 1.26

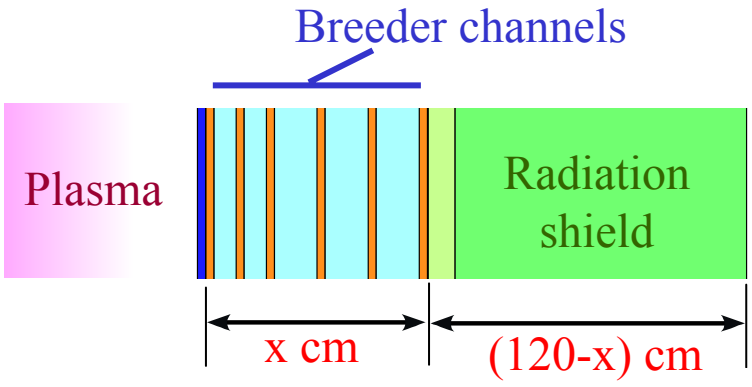
  Flibe/RAFM blanket
  ^{6}Li:55%, Local TBR: 1.22

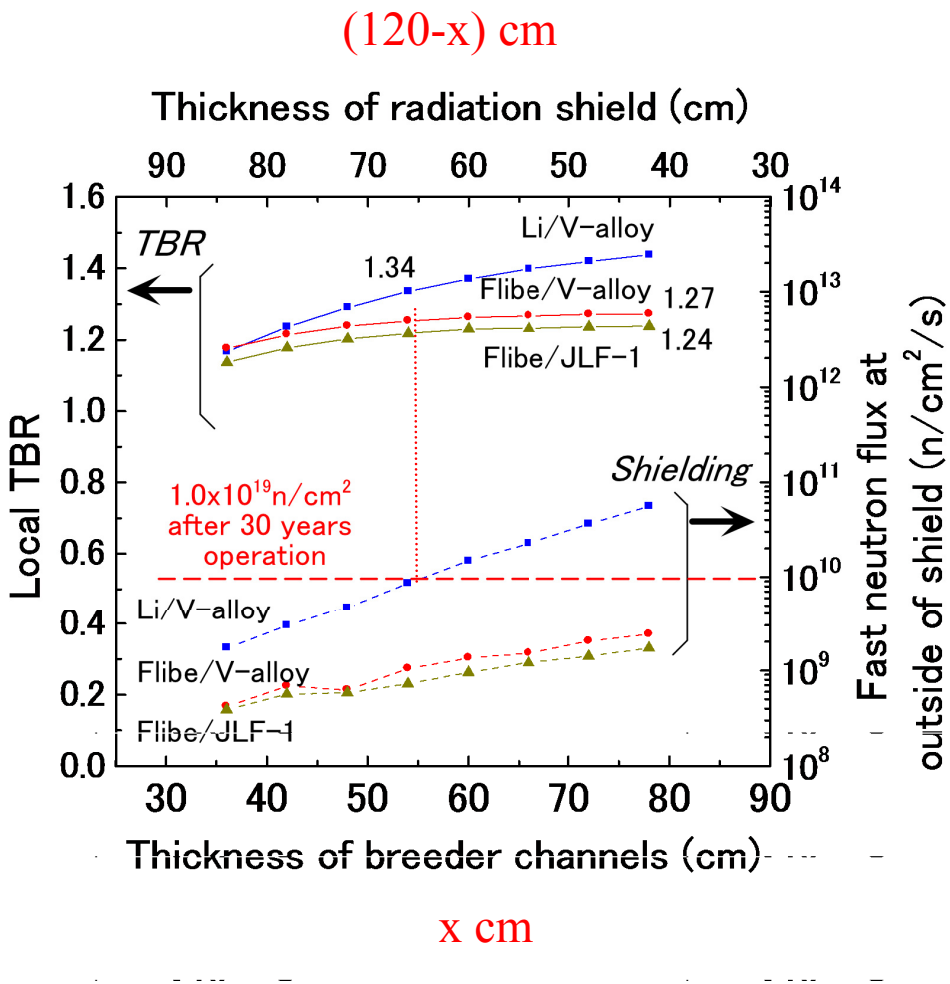
Impact of Tungsten Armor on Local TBR



- The effect of W armor on local TBR is in most cases non-monotonic
- The effect is either positive or negative depending on the system and Li isotopic control

Local TBR and Neutron Shielding (^{6}Li Enrichment to 35 %)





Li/V-alloy blanket

- Shielding: ~65 cm
- Breeder channels: ~55 cm
- Local TBR: ~1.34 (comparable to original FFHR2)

Flibe/V-alloy, JLF-1 blanket

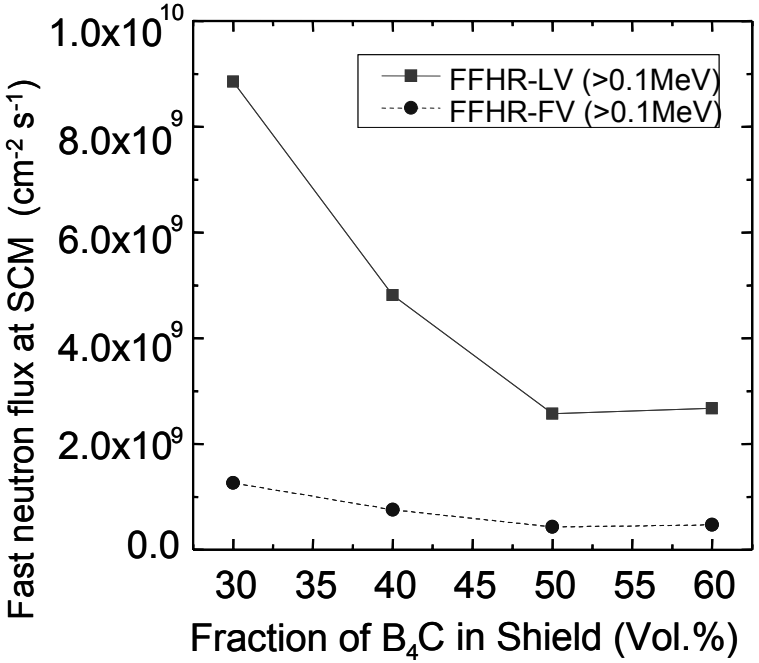
Local TBR: ~1.27 and ~1.24

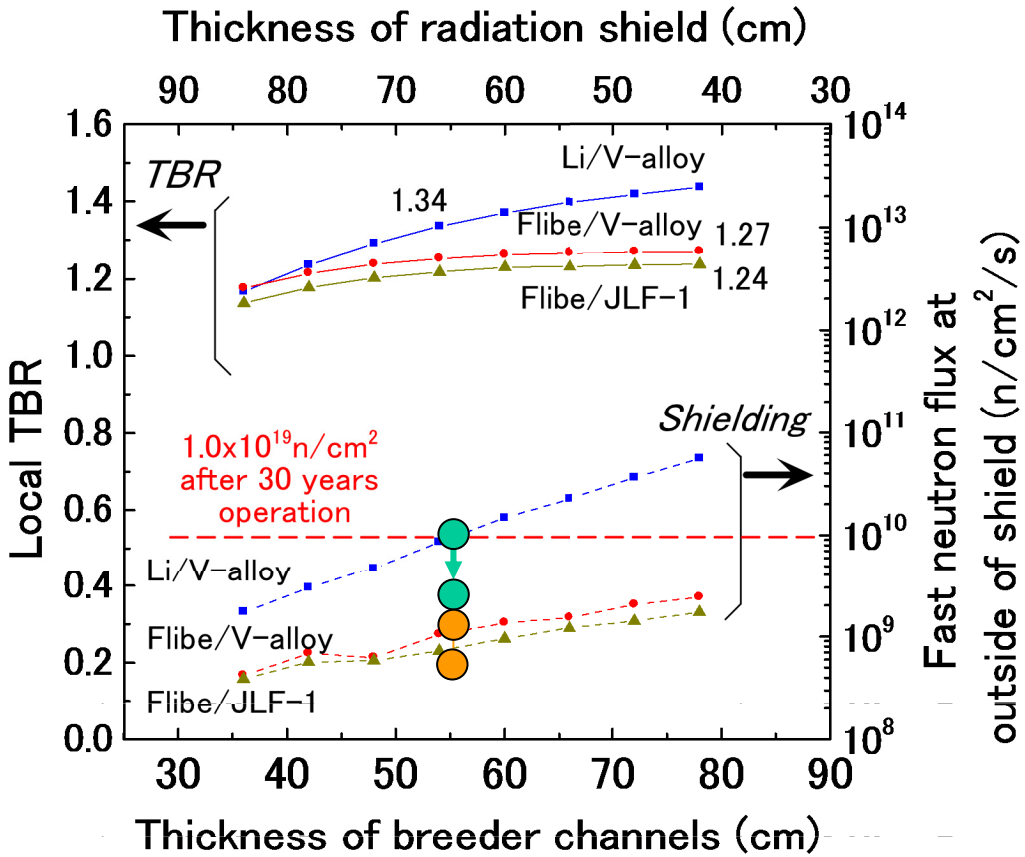
Superior shielding performance

- TBR can be enhanced by further design efforts

Marginal TBR for Flibe/V-alloy system
Marginal neutron shielding for Li/V-alloy system

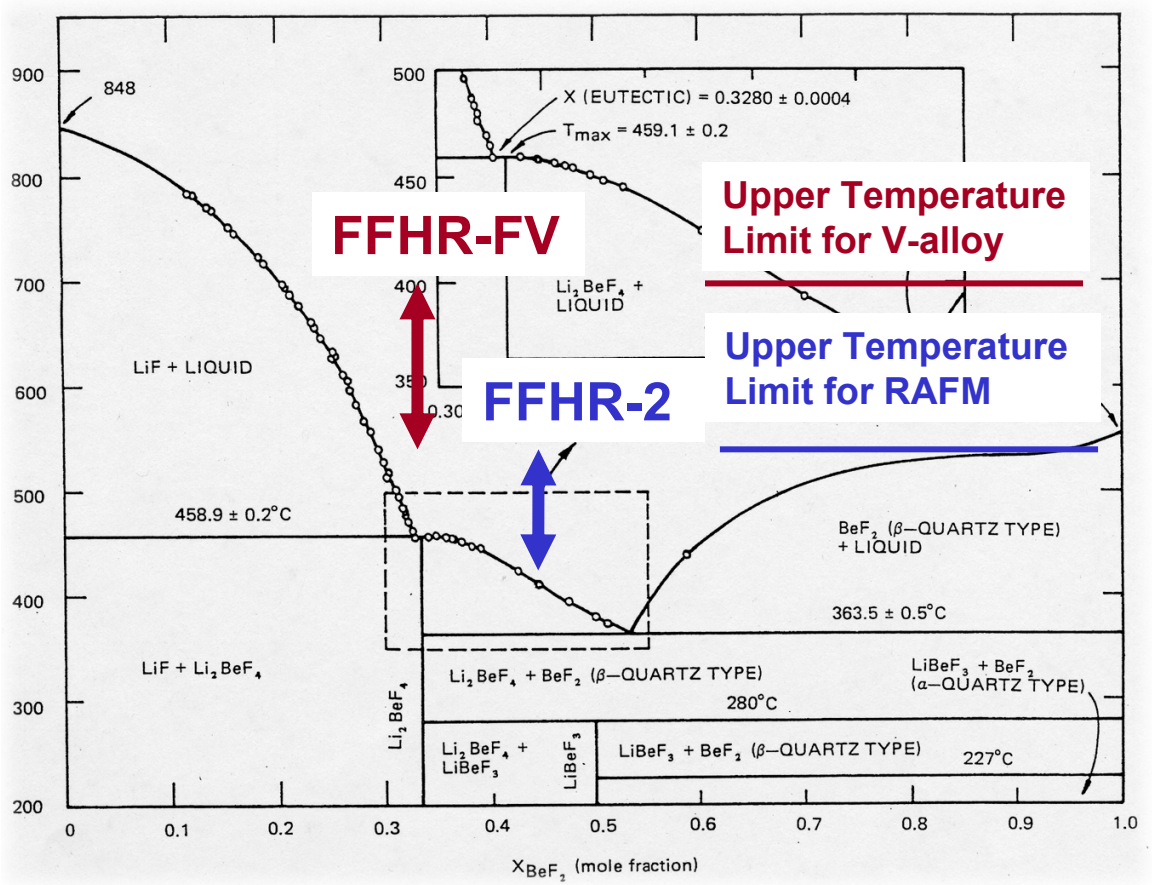
The Effects of B_4C Fraction in the Shield on Shielding Properties

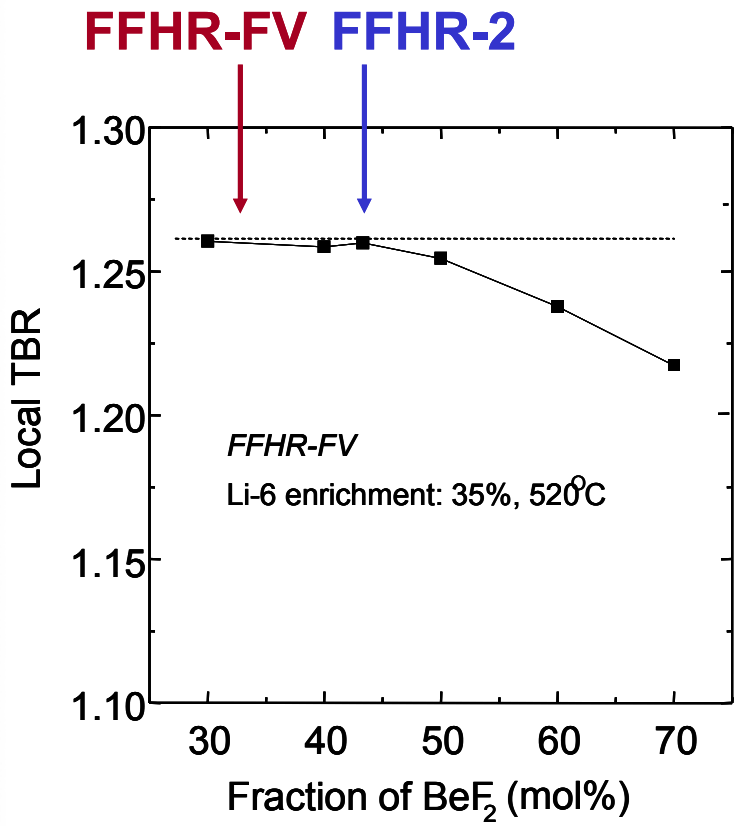




- With the increase in B_4C fraction, the fast neutron flux at SCM of FFHR-LV becomes less than 1/3.

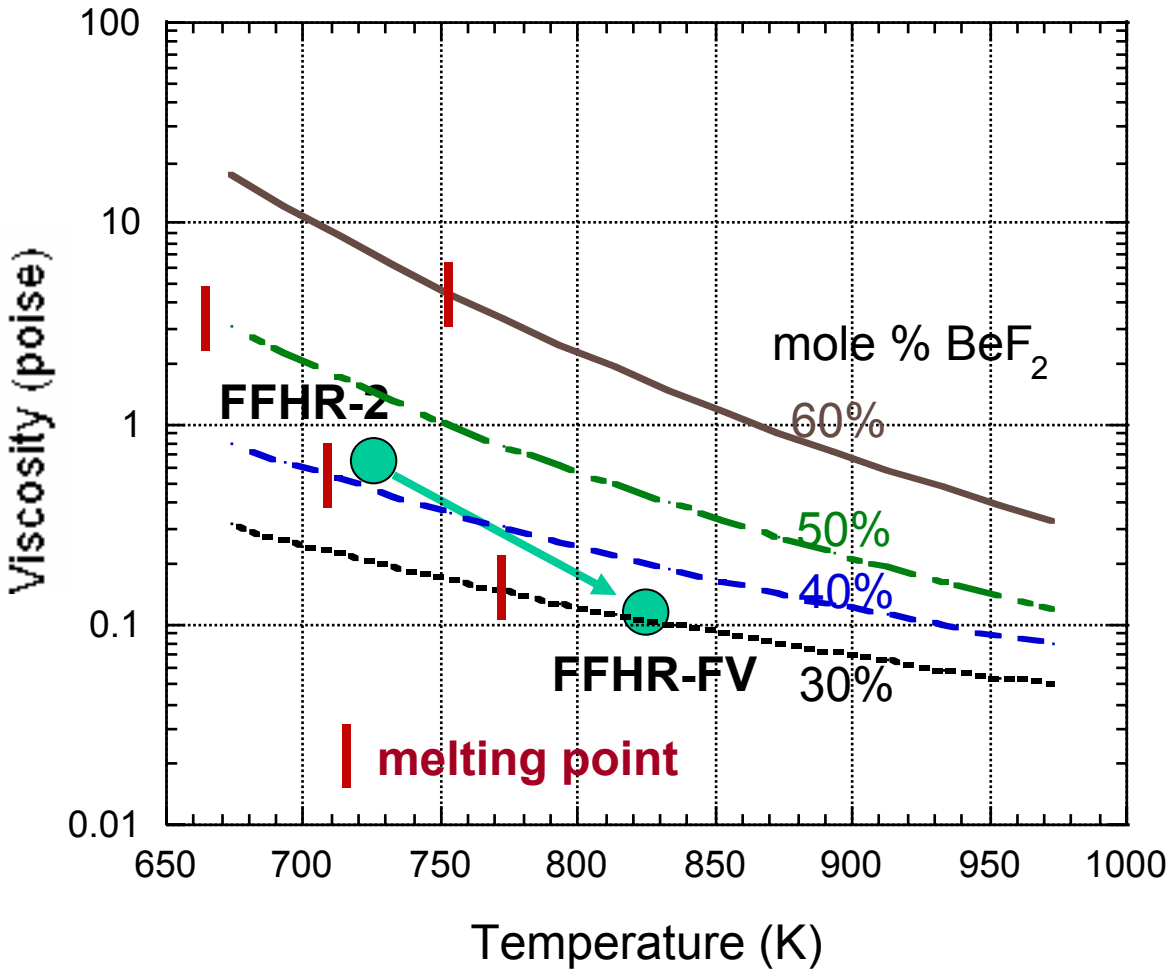
Part 2 : Operation Temperature and BeF_2 Mole Fraction for FFHR-FV





- Relative to FFHR-2, the operation temperature range increased from 450C/550C to 550/700C
- With the increased operation range, the BeF_2 mole fraction can be decreased from 43% to 32%

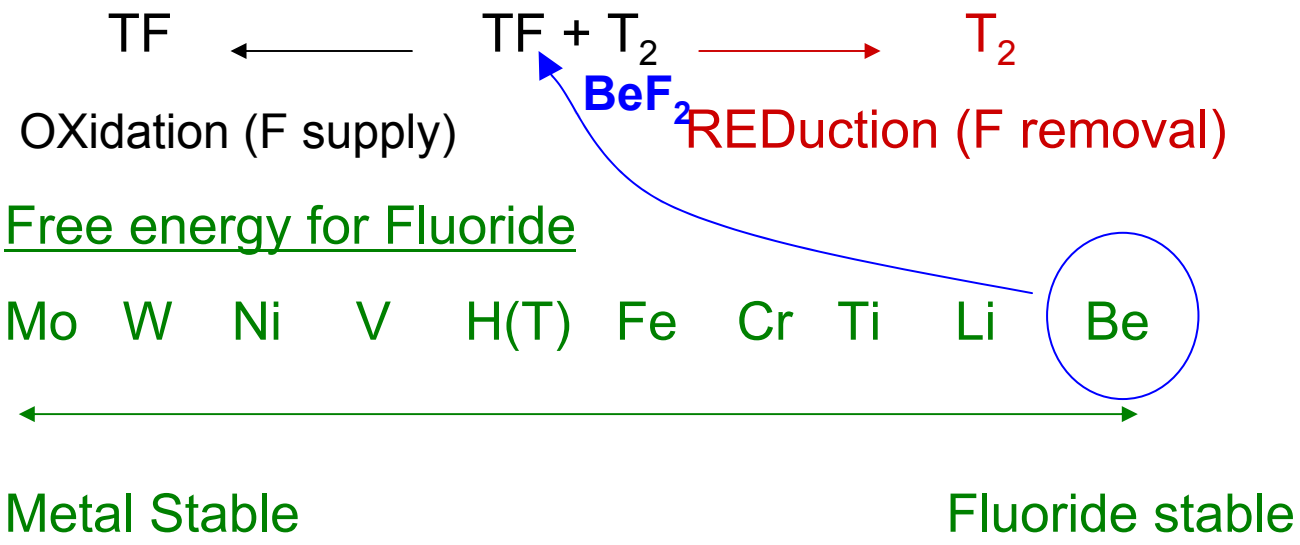
The Effects of Composition and Temperature on Viscosity of Flibe



- Viscosity of Flibe deceased to ~1/8 by increasing temperature and decreasing BeF_2 fraction
- The decrease in the viscosity results in reducing Prandtl number, and could improve the heat transfer characteristics of Flibe

Part 3 : Proposal for an alternative REDOX control

REDOX control with Be (conventional method)



REDOX control in FFHR

TF + Be (also as multiplier) → T_2 + BeF_2 (T_2control)

Suppression of corrosion by reducing TF level

Tritium transfer in the form of T_2 to out of blanket

Tritium level in Flibe should be <0.01 ppm for reducing T leakage (Fukada et al. 2003)

Estimation of T inventory in the case of the conventional REDOX control

1000K 1atm T_2 solubility in Flibe ~ 3 mol-ppm (Fukada et al. 2004)

assuming 0.01 mol-ppm : $P(T_2)$ --0.003 atm (Henry's low)

1000K V-4Cr-4Ti Sievert's const. ~0.05 fr/atm$^{0.5}$ (Baxbaum 2002)

Tritium equilibrium level in V-alloy structure = $0.05 \times (0.003)^{0.5} = 0.003$ molfr

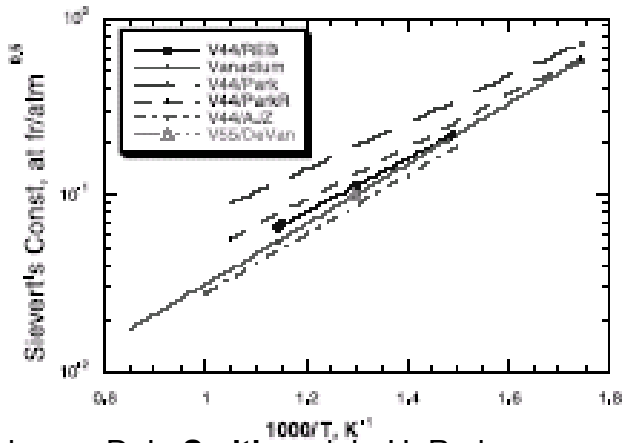
FFHR-FV 700 ton V-alloy T inventory :

$7 \times 10^5 \times 3/50 \times 0.003$ ~ 130 kg !!

Conventional REDOX control cannot be applied to V-alloy structure

Li/V blanket

0.015 kg (manageable)



R. E. Buxbaum, D. L. **Smith** and J. -H. Park
Journal of Nuclear Materials, Volumes 307-311, p576-579

Sievert's const. for H with V-4Cr-4Ti

Alternative REDOX control (after Sze, 1986)

$TF \leftarrow TF + T_2 \rightarrow T_2$

Mo or W

OXidation (F supply) REDuction (F removal)

Free energy for Fluoride

Mo W Ni V H(T) Fe Cr Ti Li Be

MoF_6 or WF_6

Metal Stable ⟷ Fluoride stable

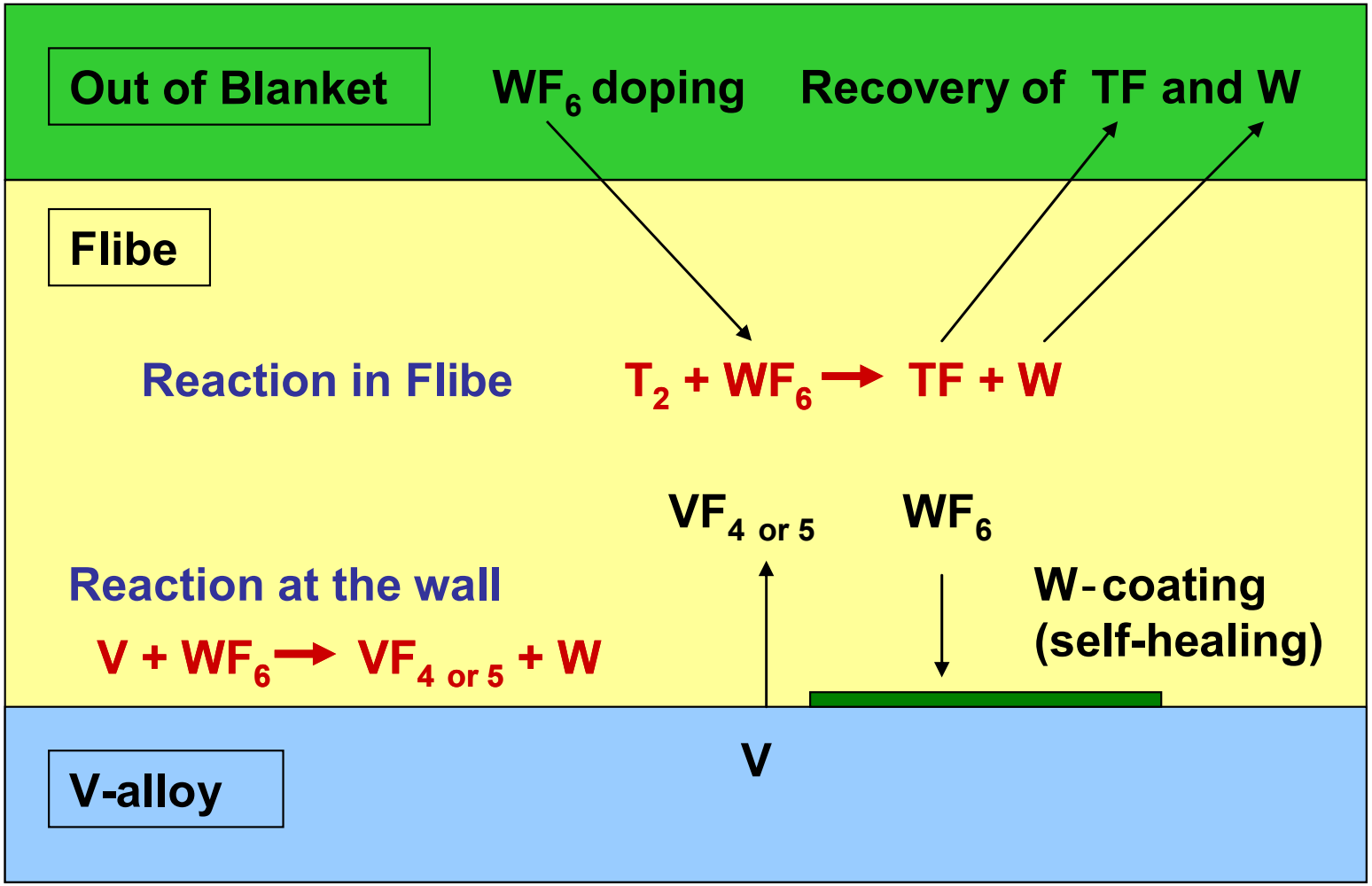
Addition of WF_6 (or MoF_6)

$T_2 + WF_6 \rightarrow TF + W$ (in Flibe : TF Control)

$V + WF_6 \rightarrow VF_{4\ or\ 5} + W$ (at wall surface)

(Wall corrosion will be suppressed by W in-situ coating with self-healing capability)

Tritium extraction out of the blanket in the form of TF



Concept of TF control Flibe blanket

(WF_6 doping)

Thermodynamics Investigation

How low the T inventory can be reduced by TF control?

Is use of W instead of Mo feasible? (W is better for low activation)

	1/6 WF_6	**+**	**½ T_2**	**→**	**1/6 W** **+**	**TF**
					↓ (precipitation)	
ΔG_f^0 (1000K) (kcal/mol)	**-56.8***					**-66.2***

$$\Delta\Delta G_f^{\ 0} = -RT \ln P_{TF} / P_{WF6}^{\ 1/6} P_{T2}^{\ 1/2}$$

Assuming 1ppm WF_6 in solution into Flibe

$P(WF_6)$ ~ 10^{-6} (assuming high solubility of WF_6) **$P(TF) = \sim 11 \times P(T_2)^{1/2}$**

In the case of MoF_6 ΔG_f^0 (1000K) = -50.2* **$P(TF) = \sim 300 \times P(T_2)^{1/2}$**

* Grimes and Cantor 1971

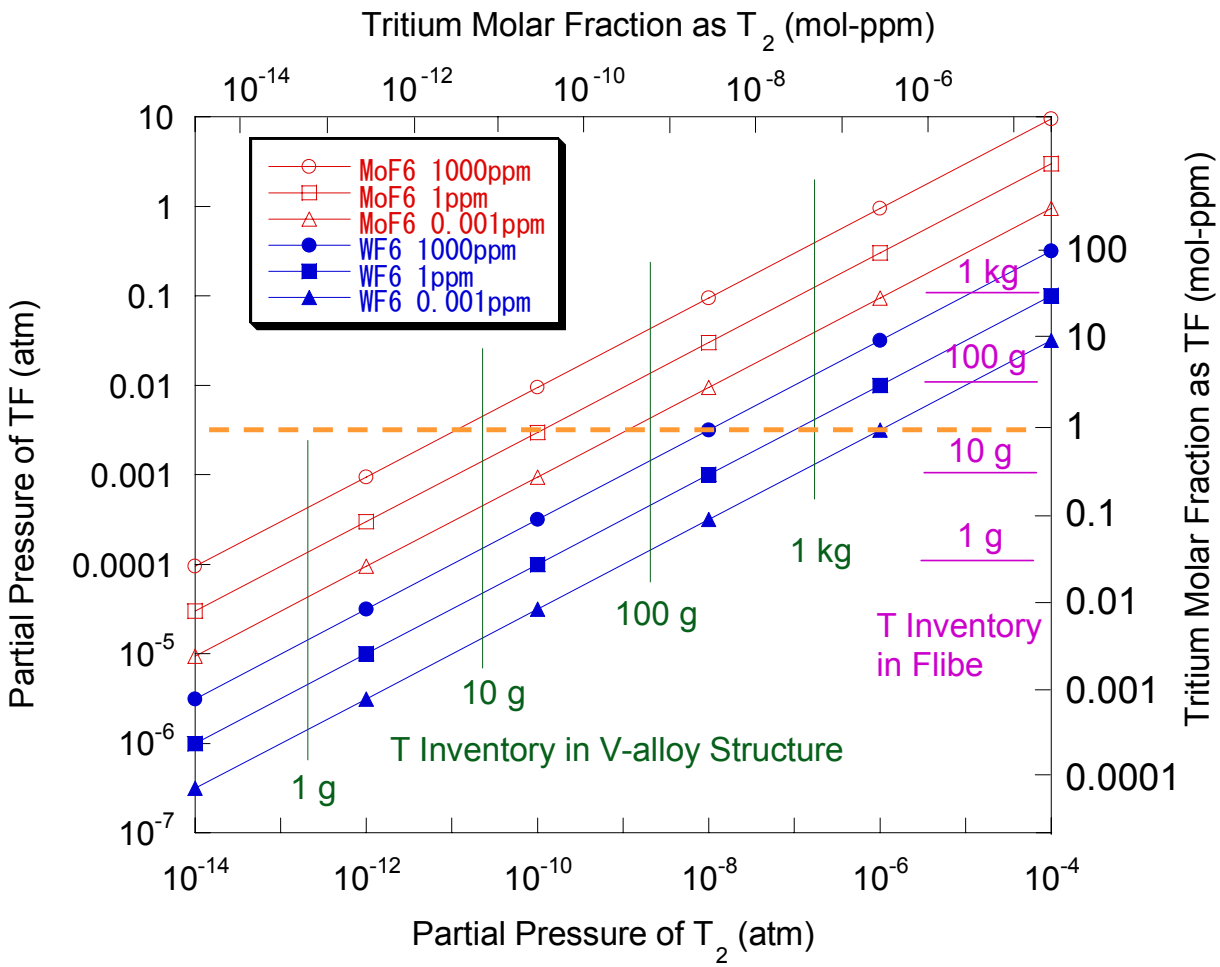
Estimation of Tritium Inventory (FFHR-FV simple torus)

Inventory in V-alloy structure (700ton)

Determined by Partial pressure of T_2

Inventory in Flibe in blanket (1140ton)

Determined by molar fraction of TF in Flibe



Partial pressure and tritium molar fraction of TF and T_2 in various levels of MoF_6 or WF_6 doping into Flibe at 1000K.

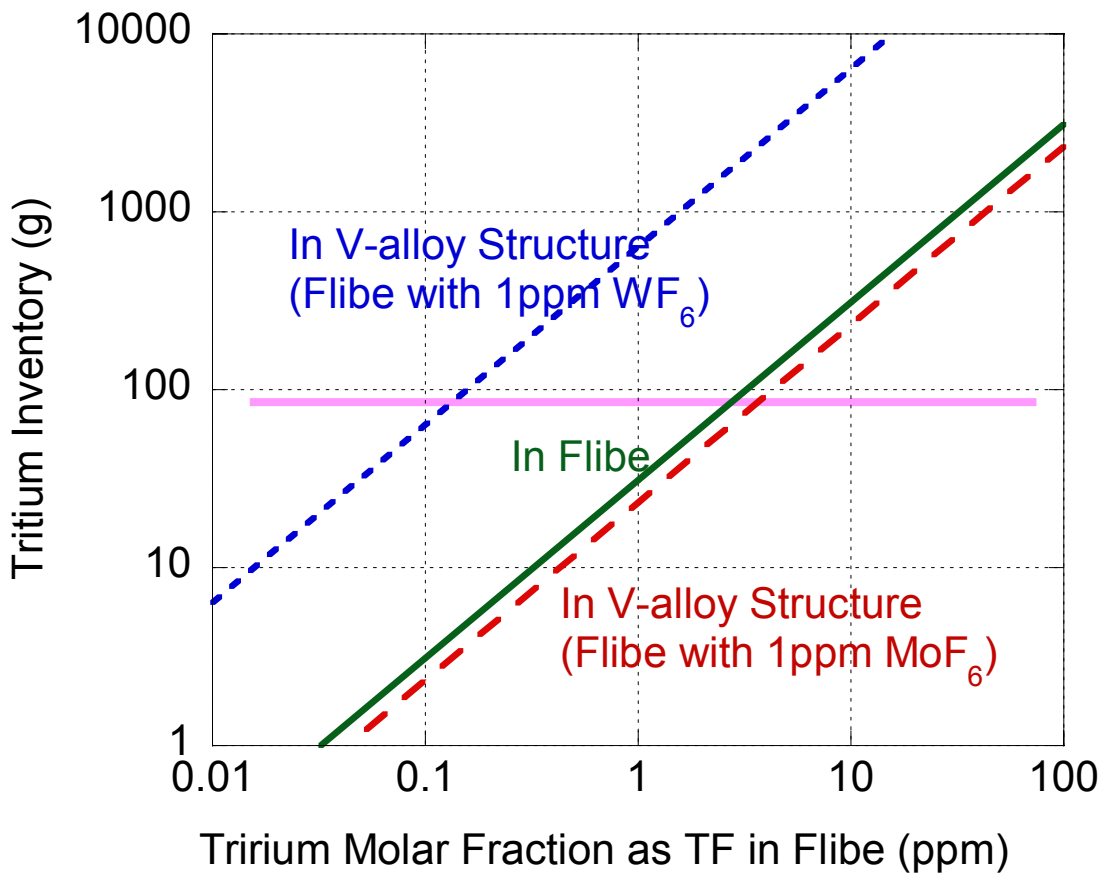
Tritium inventory in V-4Cr-4Ti structure and in Flibe are also indicated assuming FFHR-FV blanket structure.

Tritium Inventory as a Function of TF Level in Flibe

Similar T inventory in V-alloy structure and Flibe in the case of MoF_6 doping

~30 times larger T inventory in V-alloy than in Flibe in the case of WF_6 doping

T inventory of <100g is feasible by keeping the TF level in Flibe ~0.1 ppm for WF_6 doping



The tritium inventories in V-4Cr-4Ti structure and Flibe in the blanket area as a function of the tritium molar fraction as TF in Flibe in two cases of MoF_6 and WF_6 doping at 1000K.

The blanket structure of FFHR-FV was assumed, which is composed of 700 tons of V-4Cr-4Ti structure and 1140 tons of Flibe.

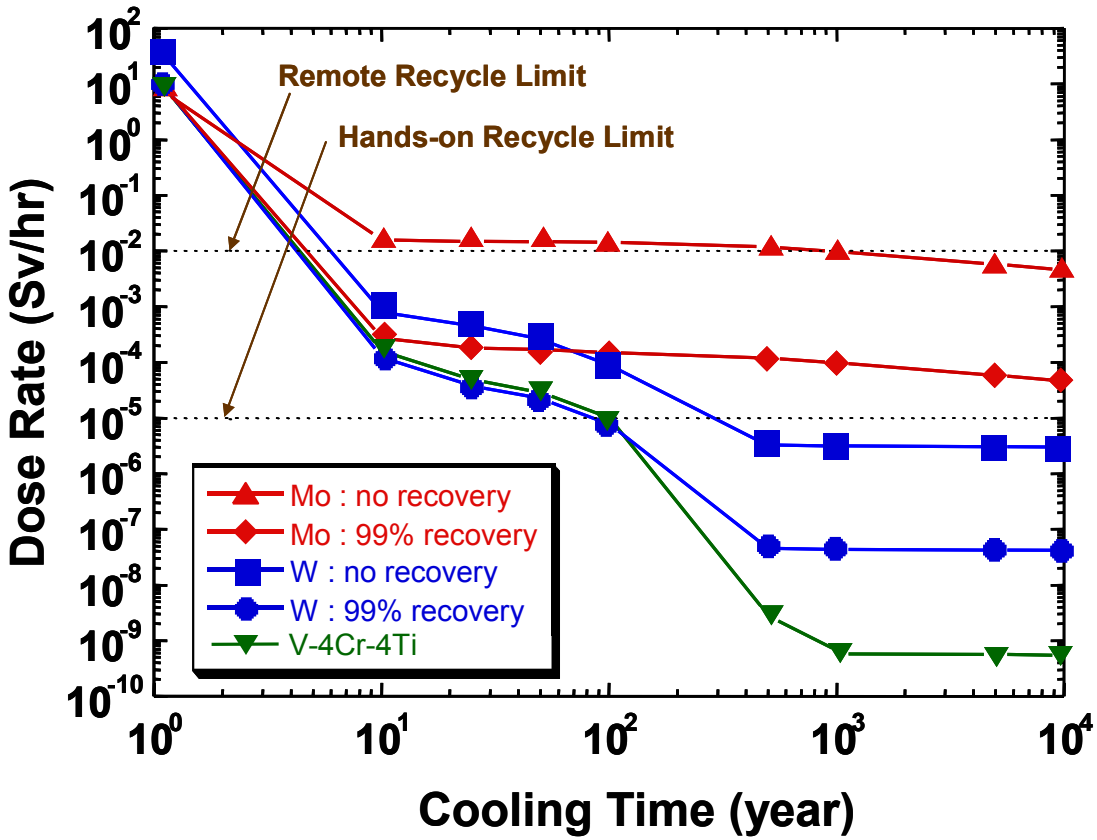
Activation as a Function of the Cooling Time

Recycling is not feasible in the case of no recovery of Mo

Remote-recycling is feasible in the case of 99% recovery of Mo

Hands-on recycling is still feasible in the case of no recovery of W but need longer cooling time

Hans-on recycling is feasible with 100 years' cooling in the case of 99% recovery of W



Dose rate of the blanket as a function of the cooling time after 30 years' operation.

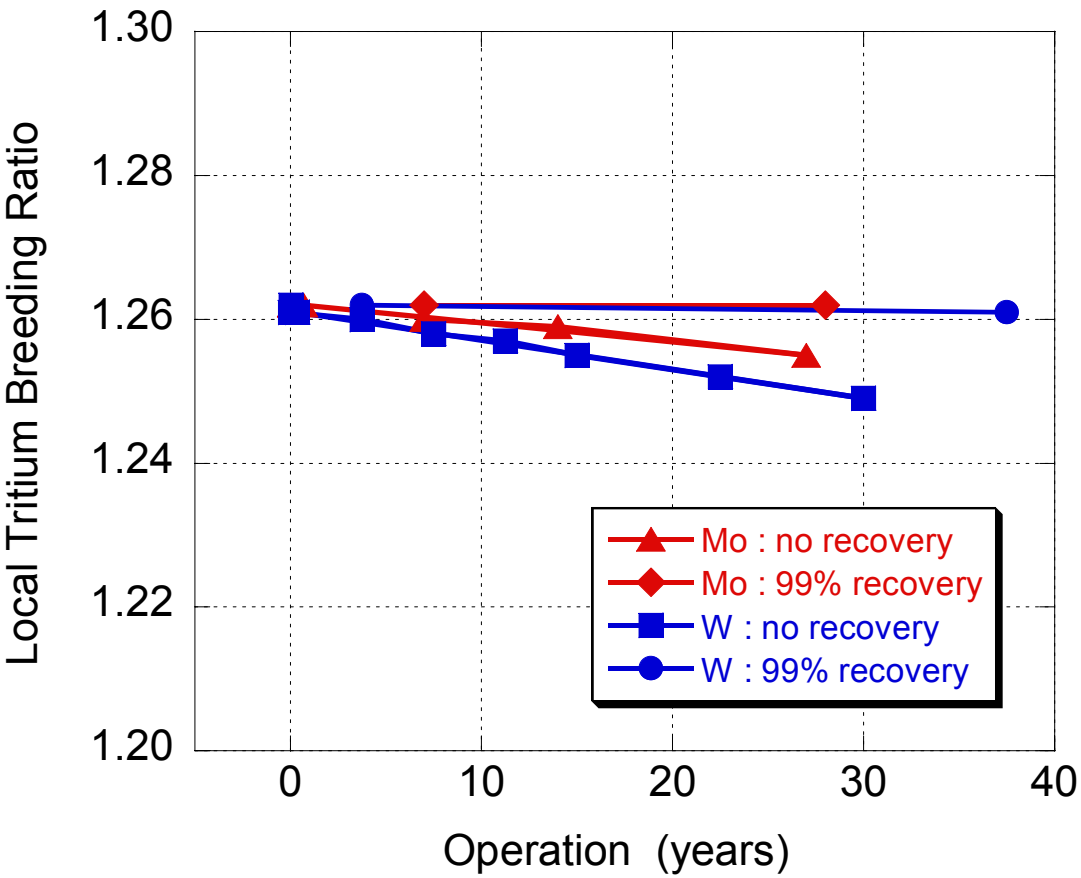
The blanket parameter of FFHR-FV with neutron loading of 1.5 MW/m^2 was used.

Two cases of no recovery and 99% recovery of Mo or W precipitates were compared.

Local Tritium Breeding Ratio (TBR)

W influences TBR more than Mo

However, decrease by 0.01 after 30 years would be negligible



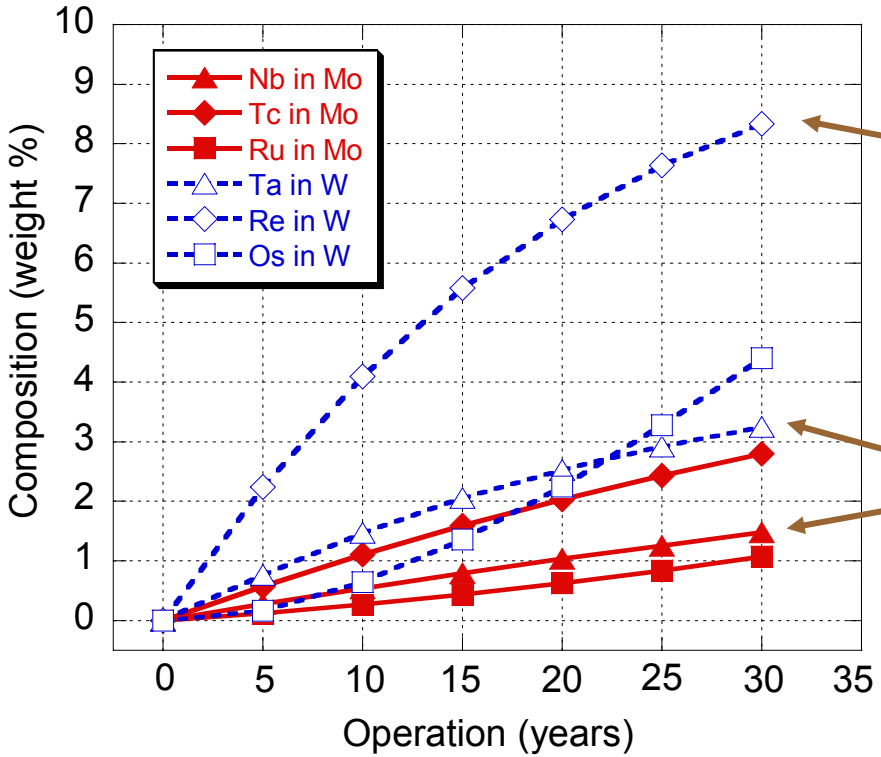
Local Tritium Breeding Ratio (TBR) as a function of the operation time. Two cases of no recovery and 99% recovery of Mo or W precipitates were assumed.

Potential Compatibility Change of Mo and W coatings by Transmutation

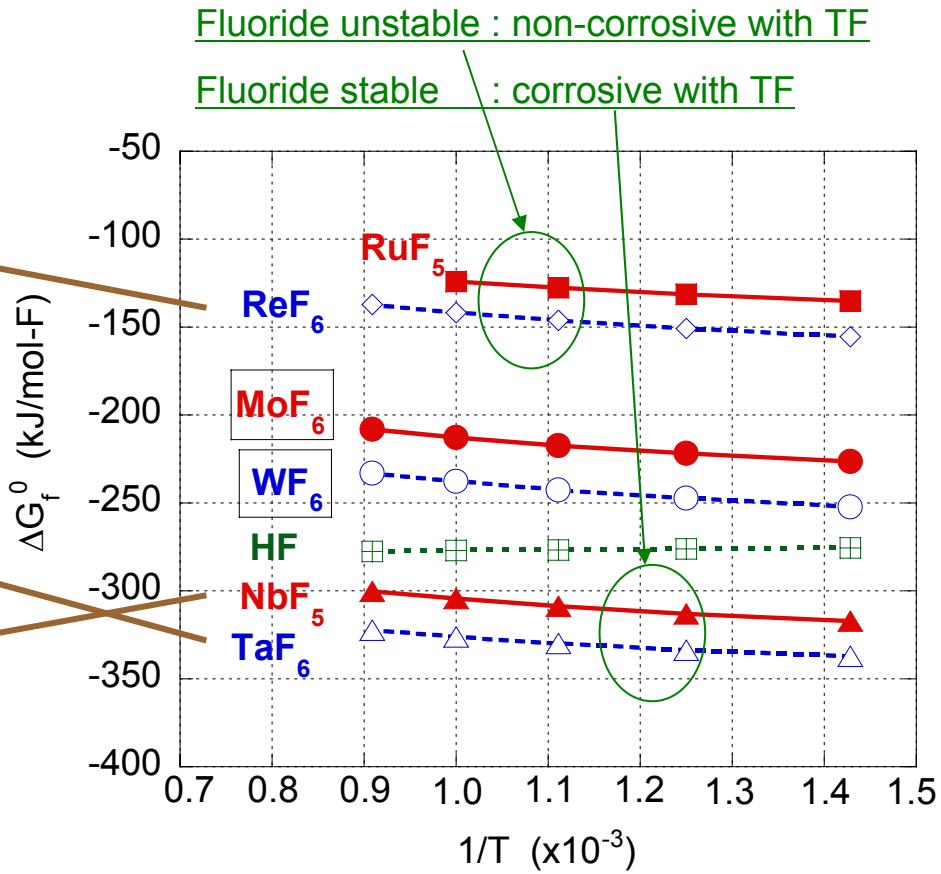
Nb, Tc and Ru are produced from Mo

Re, Ta and Os are produced fromW

Thermodynamically, production of Nb and Ta could be issues



Compositional change of Mo and W



Free energy for fluoride formation (MALT-2)

Advantages of TF control, Flibe/V-alloy system

Reduction of T-inventory

Reduction of T leakage

No necessity of coatings for MHD or T-barrier

No necessity of Be, only necessary for n-multiplying purposes if TBR is insufficient

Issues

Recovery of W (or Mo) precipitates

3 GWth -- 1.2×10^{26} T/day -- 5 kg W/day

Corrosion by TF out of blanket

Conclusions

1. Use of V-alloys instead of RAFM increases upper operation temperature and ΔT for Flibe self-cooled Blanket, potentially enhancing plant efficiency.
2. With the use of V-alloy, viscosity of Flibe can be reduced significantly by increased temperature and decreased molar fraction of BeF_2. This is a merit for controlling flow pressure and heat.
3. TBR of the V-alloy/Flibe blanket is on marginal level without using neutron multiplier Be.
4. Because of extremely large tritium inventory in structural materials, REDOX control using Be cannot be applied to V-alloy/Flibe blanket system.
5. REDOX control to enhance TF production by doping MoF_6 or WF_6 can reduce the tritium inventory, to acceptable level, and also reduce tritium permeation by large decrease in partial pressure of T_2.
6. The reaction of MoF_6 or WF_6 with wall material (V-alloy) would form corrosion-protective Mo or W coating on the wall in the initial operation period. The coating could have self-healing capability. This need experimental verification.
7. The tritium inventory in the blanket area would be less than 100g at the TF level of 1 and 0.1 ppm in Flibe, with addition of MoF_6 and WF_6, respectively.

Conclusions (continued)

8. WF_6 doping is more attractive than MoF_6 doping for low activation purposes.
9. Impact of precipitation of Mo or W in blanket on TBR will be negligible.
10. Transmutation products of Ta in W and Nb in Mo may degrade the stability of W and Mo coating in Flibe, respectively.
11. Neither MHD nor T-barrier coating would be necessary for the Flibe blanket system with TF-control REDOX.
12. Among the issues of the concept are recovery of tritium in the form of TF and recovery of Mo or W out of the blanket.