Introduction to fusion power plant fuel cycles and tritium breeding blankets

Sara Ferry
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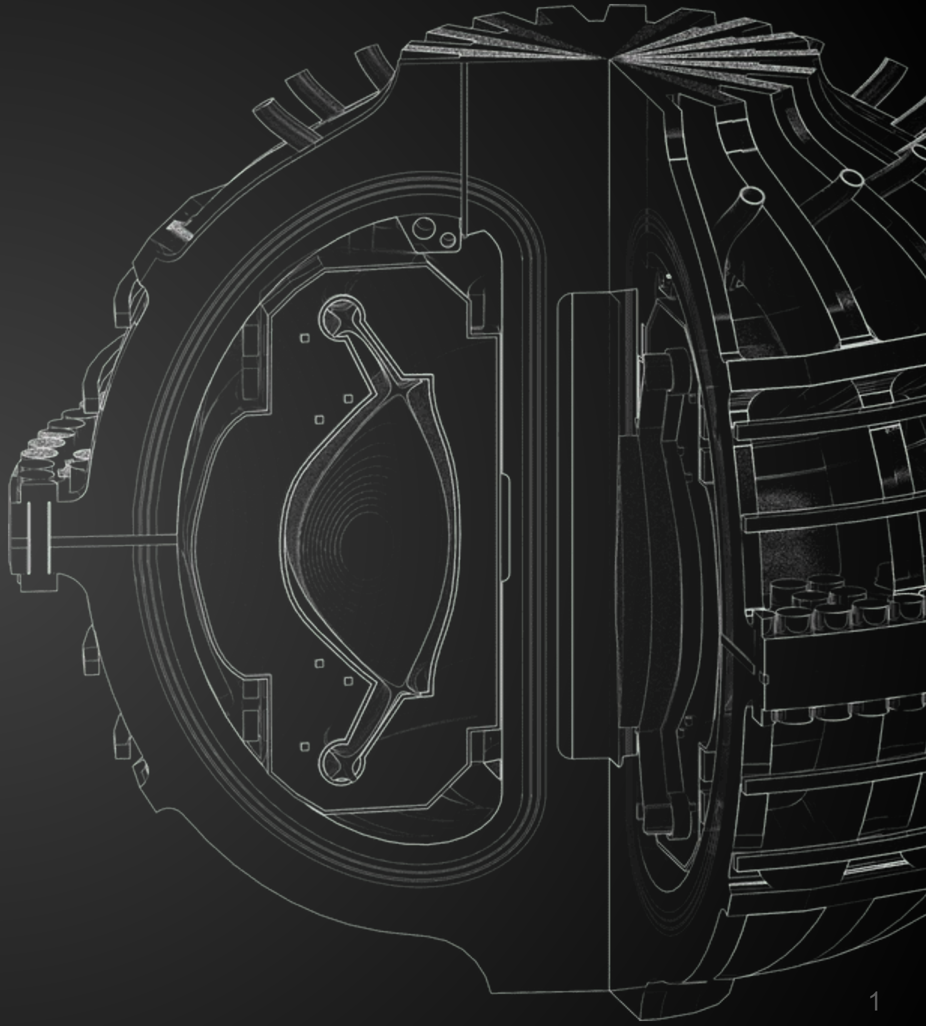
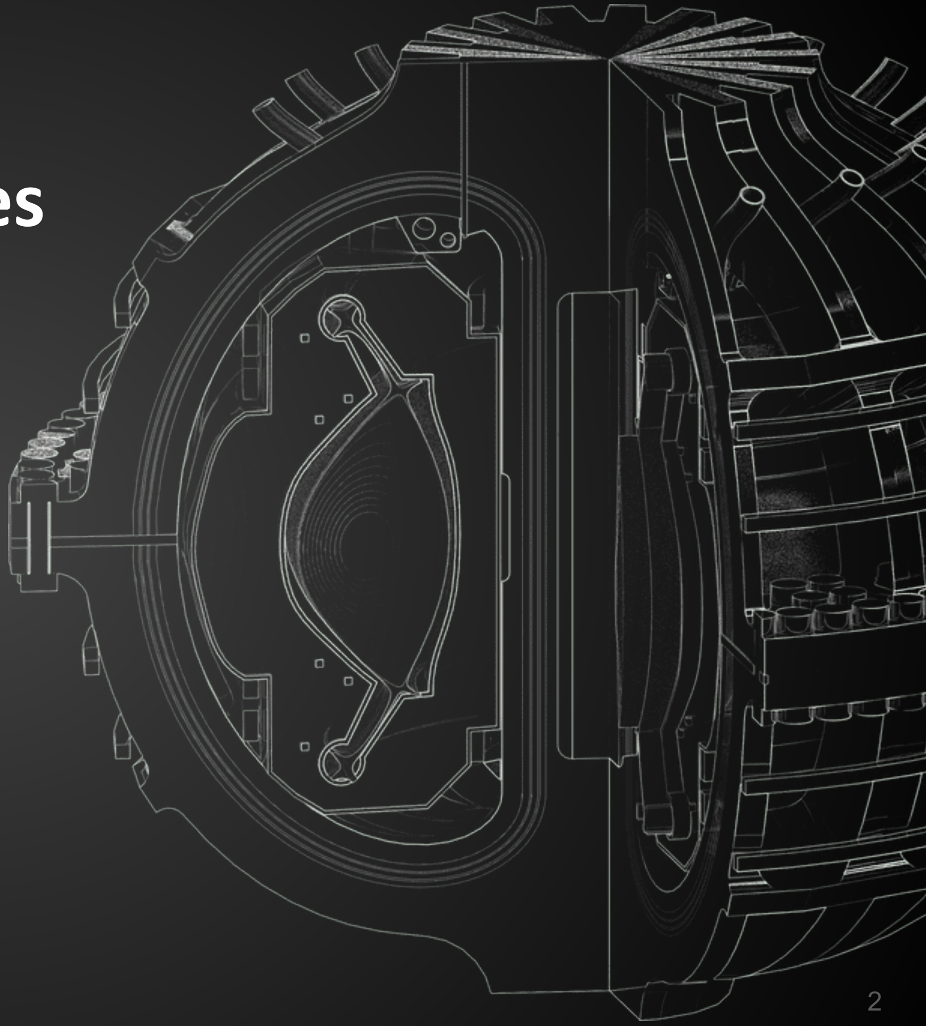
June 13, 2023
PPPL Introduction to Fusion Energy and Plasma Physics Course



Part I:
Fusion power plant fuel cycles



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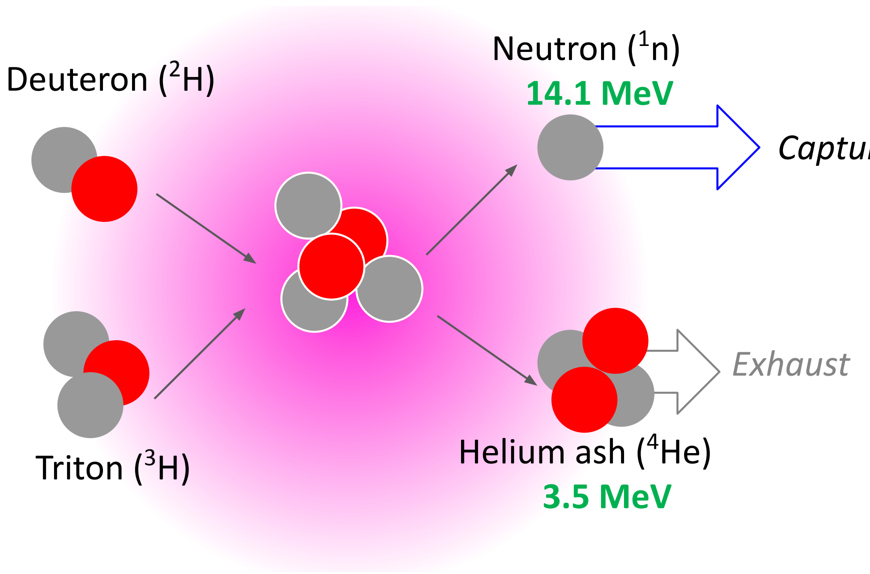
Dennis Whyte (MIT PSFC)
Remi Delaporte-Mathurin (MIT PSFC)
Samuele Meschini (MIT PSFC/Politecnico di Torino)

The basics: deuterium and tritium as fusion fuel

in the plasma:



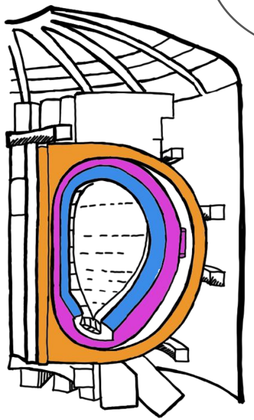
*Capture in the **blanket***

Shielding:
Moderate + absorb neutrons to shield the magnets

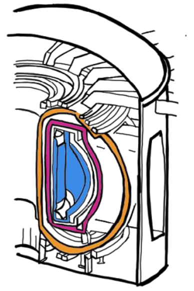
Heat capture: generate power

Tritium breeding:
Make more fuel (Li + n → T)

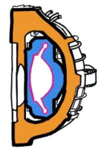
Key:
Orange = toroidal field magnet
Pink = vacuum vessel
Blue = blanket



EU-DEMO
R = 9 m, P = 3GW



ARIES-AT
R = 5.2 m, P = 1.7 GW



ARC
R = 3.5m, P = 0.5 GW

The (advanced) basics: tritium burn efficiency

We often characterize fuel utilization as the product of **fueling efficiency** and tritium **burn fraction**:

$$\eta_f f_b$$

- Previous results: need $\eta_f f_b$ > 2% for self-sufficiency
- EU-DEMO: $\eta_f f_b$ < 0.09%

Source: M. Abdou et al., "Physics and technology considerations for the deuterium-tritium fuel cycle and conditions for tritium fuel self sufficiency," Nuclear Fusion, 2021.

This formulation works well for discrete fueling events (e.g. ICF), but in a tokamak, a given triton will "recycle" many times before being burned or being exhausted ⟶ use **Tritium Burn Efficiency**

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By considering the balance of particles in the vacuum boundary, we can derive an expression for TBE:

$$\mathrm{TBE} = \left(\frac{1}{2 f_{\mathrm{He,div}} \Sigma} + 1 \right)^{-1}$$

Relative density of neutral helium ash to hydrogenic fuel in the divertor

$$f_{\mathrm{He,div}} \equiv \frac{n_{\mathrm{He,div}}}{n_{\mathrm{Q,div}}}$$

Ratio of pumping speed (at the divertor exhaust) of helium ash to pumping speed of unburned hydrogenic species

$$\Sigma \equiv \frac{S_{\mathrm{He}}}{S_{\mathrm{Q}}}$$

Source: D. Whyte, S. Ferry, R. Delaporte-Mathurin, and S. Meschini, "Tritium burn efficiency in magnetic fusion," upcoming, 2023.

The (advanced) basics: tritium burn efficiency

It's summer and I don't want to memorize equations. What should I remember from this equation?

- Density of He ash in the divertor is higher than density of D and T: TBE increases ($f_{He,div}$ goes up)
  - ***CAVEAT:*** *as you increase $f_{He,div}$, fusion power density goes down*
- He ash is pumped out of the divertor faster than D and T: TBE increases
  - ***CAVEAT:*** *if you only pump out ash, and leave D/T until they burn, you lose plasma control*

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The (advanced) basics: tritium burn efficiency

It's summer and I really don't want to memorize equations. What should I remember from this equation?

If you are designing the fusion power plant, there are **tradeoffs** to consider everywhere. "Bigger TBE" isn't automatically better - you might want to operate at a lower TBE to maintain higher power density, for example. (Plus you want to consider how your tritium burn in the plasma integrates with your overall fuel cycle, which we'll talk about next.)

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The basics: producing tritium in the blanket

we'll talk more about blankets in Part II

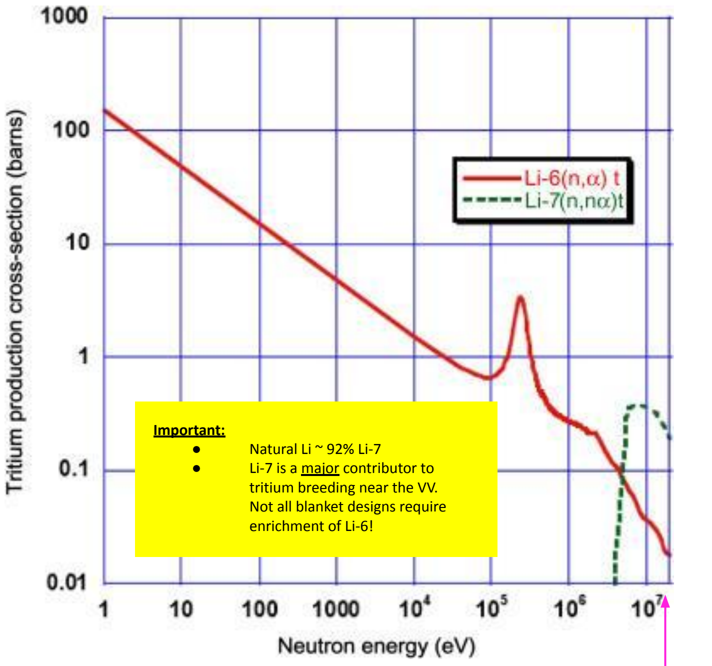
Li-6 + n $\longrightarrow \alpha$ + T + heat

Li-7 + n $\longrightarrow$ T + n' + α - heat

Li-7 + n $\longrightarrow$ 2T + 2n - heat

- Tritium is produced when the fusion neutrons interact with **lithium** in the blanket breeder material.
- Blanket **breeder** options include:
  - Lithium solids (e.g. lithium titanate)
  - Molten PbLi
  - Molten salts, like FLiBe or FLiNaK
  - Liquid lithium
- In the rest of the fuel cycle, tritium is extracted from the blanket and prepared for injection as fuel.

Tritium production cross-sections for Li-6 and Li-7



Important:
- Natural Li ~ 92% Li-7
- Li-7 is a major contributor to tritium breeding near the VV. Not all blanket designs require enrichment of Li-6!

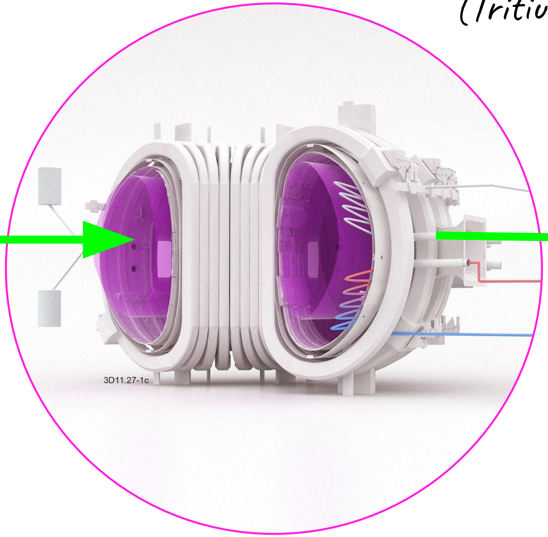
14 MeV = 1.4E7 eV

A successful fuel cycle achieves tritium self-sufficiency.

Misconception: all you need is **TBR = 1** to achieve self-sufficiency

(Tritium breeding ratio)

For every triton burnt in the plasma…



You produce one triton in the blanket!

A successful fuel cycle achieves tritium self-sufficiency.

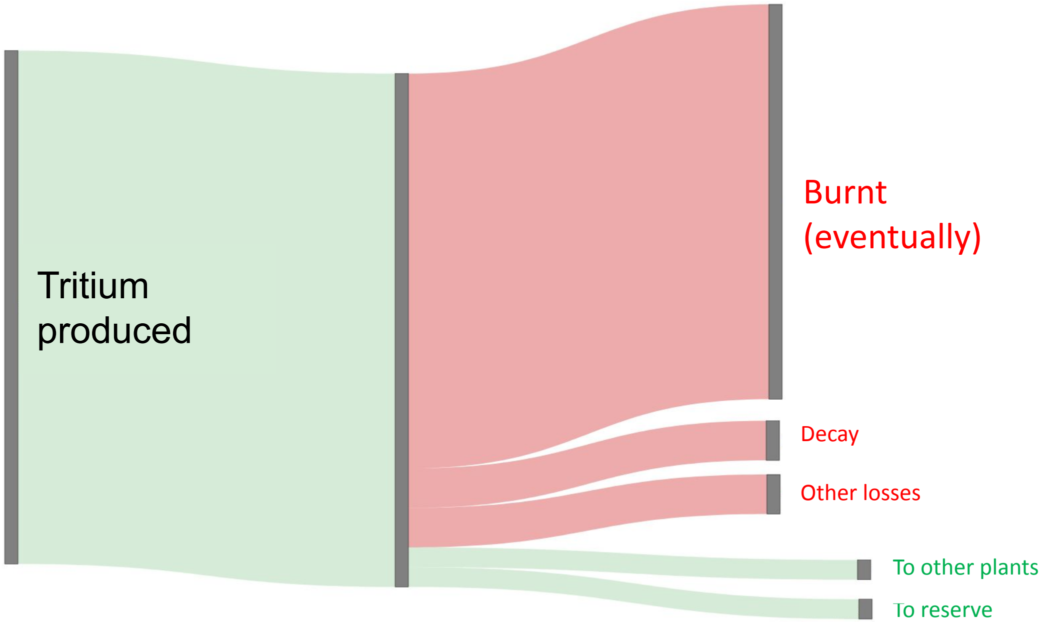
The required TBR will be > 1, but by how much?

We need to:
- Produce tritium for **self-fueling** of plant
- Account for **tritium losses** due to decay, tritium trapping, and component inefficiencies
- Maintain a certain tritium **reserve inventory** so that plant can be re-started and fueled after an outage, or fueled in the event that part of the fuel cycle fails
- Provide **start-up inventory** for additional FPPs within a certain timeframe (**doubling time**)

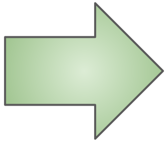
If you can live with a smaller $I_{reserve}$ and a longer $t_{doubling}$ it will be easier to design a fuel cycle that achieves self-sufficiency.

We want to achieve tritium self-sufficiency while still **minimizing overall tritium inventory** (ease regulatory/operating/safety burdens)

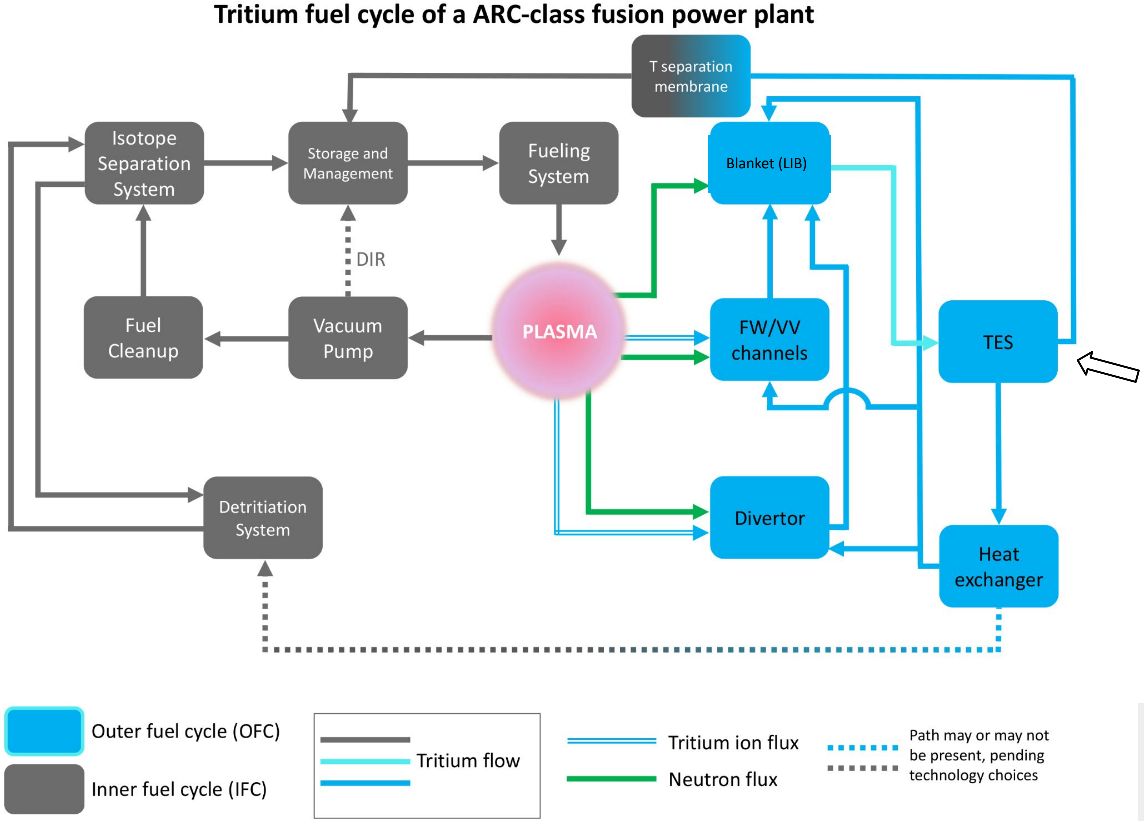
What happens to the tritium produced in the blanket?



Let's consider the fuel cycle in more detail:



Outer fuel cycle (OFC)



OFC key idea:
Tritium is bred in the outer fuel cycle and passed to the inner fuel cycle for storage and fueling.

Example - consider the **tritium extraction system**:

- ***Inflow:*** FLiBe from blanket carries bred tritium
- ***Outflow:*** Tritium is sent to the IFC
- ***Outflow***: The TES isn't perfect, so a small amount of tritium will remain in the FLiBe as it passes through to the HX

Figure: upcoming paper, S. Meschini, S. E. Ferry, R. Delaporte-Mathurin, D. G. Whyte. "Modeling and analysis of the tritium fuel cycle for ARC- and STEP-class D-T fusion power plants"

Inner fuel cycle (IFC)

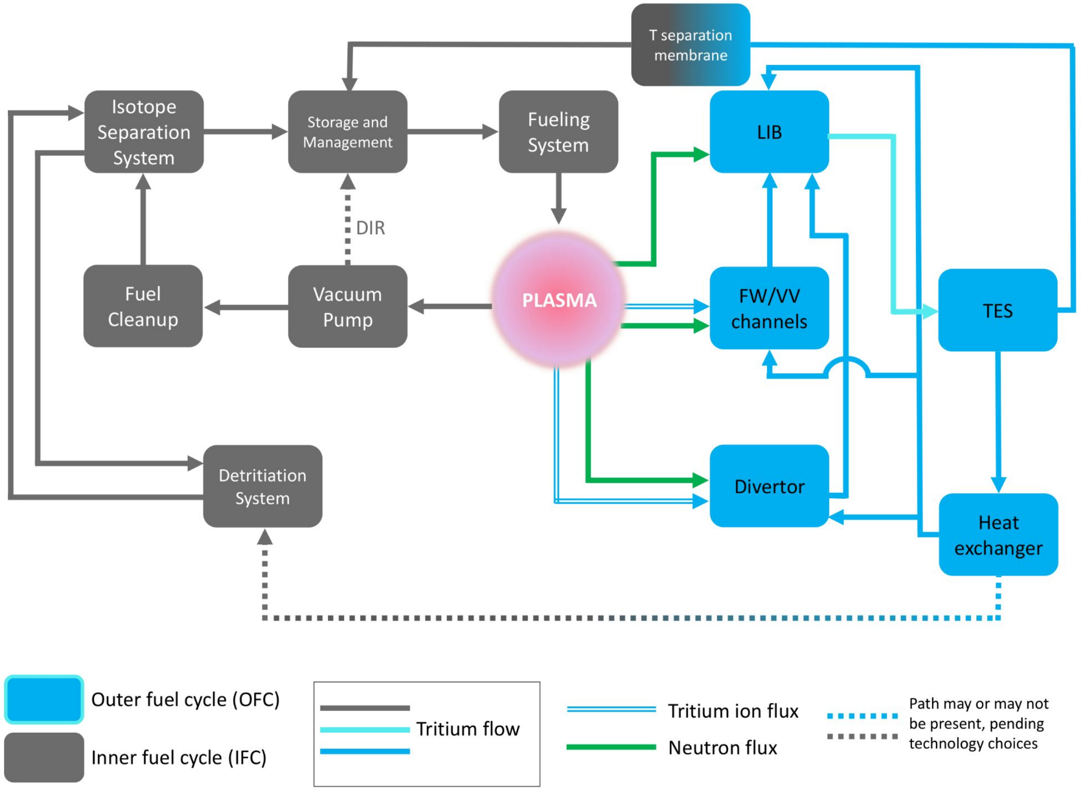
IFC key idea:
In the IFC, bred and recovered tritium is prepared for fuel injection.

These fuel cycle plots show possible particle pathways, but not flux magnitudes or system dynamics:

*Build **dynamic numerical models** to understand conditions that lead to tritium self-sufficiency!*

Figure: upcoming paper, S. Meschini, S. E. Ferry, R. Delaporte-Mathurin, D. G. Whyte. “Modeling and analysis of the tritium fuel cycle for ARC- and STEP-class D-T fusion power plants”

Tritium fuel cycle of a ARC-class fusion power plant



Modeling FPP fuel cycles: the basics

<u>Modeling a simplified fuel cycle</u>

$$\frac{dI_{\mathrm{OFC}}(t)}{dt} = \dot{N}_{\mathrm{T,burn}} \cdot \mathrm{TBR} - \frac{I_{\mathrm{OFC}}(t)}{\tau_{\mathrm{OFC}}}$$

$$\frac{dI_{\mathrm{IFC}}(t)}{dt} = \frac{1-\mathrm{TBE}}{\mathrm{TBE}}\dot{N}_{\mathrm{T,burn}} + \frac{I_{\mathrm{OFC}}(t)}{\tau_{\mathrm{OFC}}} - \frac{I_{\mathrm{IFC}}(t)}{\tau_{\mathrm{IFC}}}$$

$$\frac{dI_{\mathrm{storage}}(t)}{dt} = \frac{I_{\mathrm{IFC}}(t)}{\tau_{\mathrm{IFC}}} - \frac{\dot{N}_{\mathrm{T,burn}}}{\mathrm{TBE}}$$

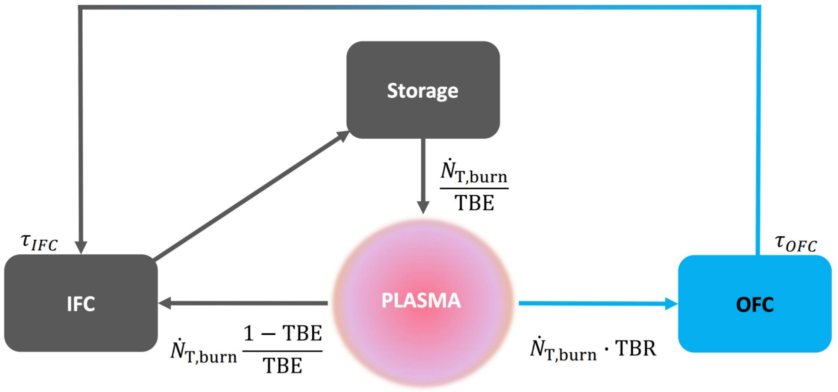
with initial conditions:

$$I_{\mathrm{IFC}}(0) = I_{\mathrm{OFC}}(0) = 0$$

$$I_{\mathrm{storage}}(0) = I_{\mathrm{startup}}$$

τ describes how long a triton stays in that component, on average. I indicates inventory.

1. Mathematically describe the inflow and outflow of tritium from each component (Plasma, IFC, OFC, Storage)
2. Model the system using a program like MATLAB Simulink



Applying this to more realistic fuel cycles:

- EU-DEMO : Abdou et al. 2021
- ARC-class tokamak: Whyte et al. 2023 (upcoming)

Note: we will publish the Simulink model and code on GitHub so others can modify for other fuel cycles of interest

Learning from fuel cycle models: TBE vs. $TBR_{required}$ and $I_{startup}$

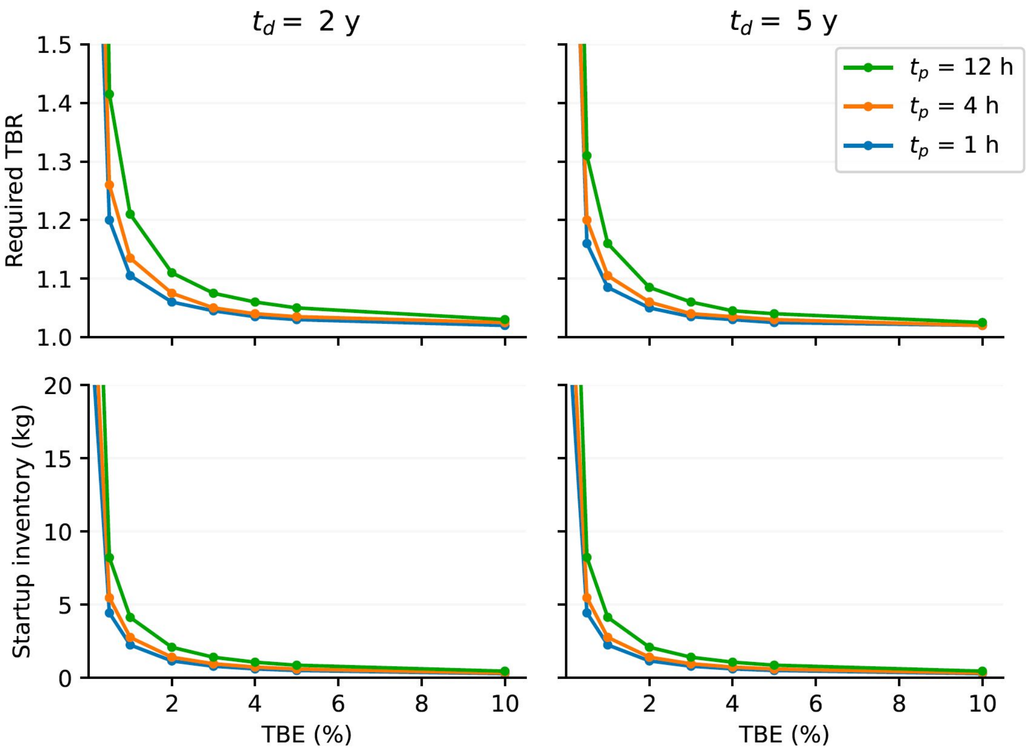
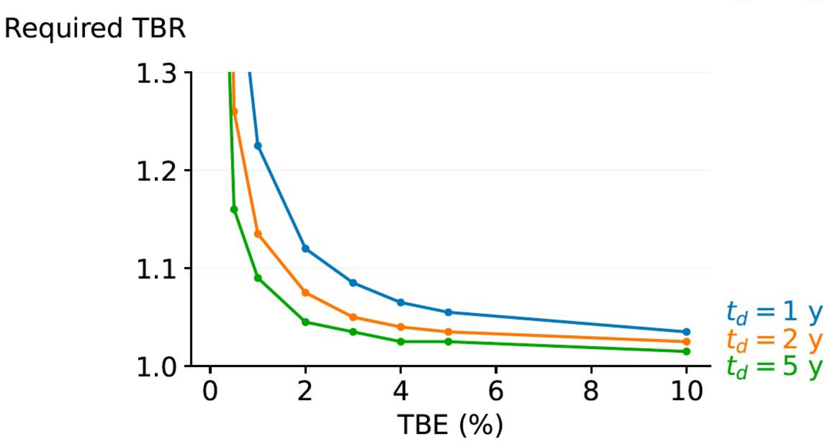


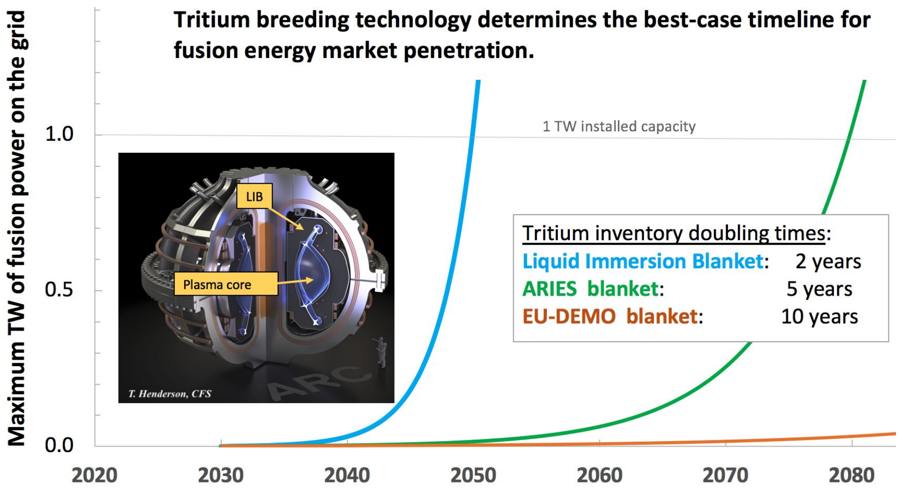
Figure: upcoming paper, S. Meschini, S. E. Ferry, R. Delaporte-Mathurin, D. G. Whyte. "Modeling and analysis of the tritium fuel cycle for ARC- and STEP-class D-T fusion power plants"

All following results are for an ARC-class FPP, with P_{fus} = 525 MWth and 30 minute pulses (1 min off). Unless otherwise specified, we have:
AF = 0.7, t_d = 2y , t_p = 4h, t_{res} = 24h, and TBE = 0.02.

- At low TBE, $I_{startup}$ and TBR_{req} become unmanageably high
- Longer doubling time: lower TBR_{req} for the same TBE
- Faster processing time t_p in the IFC: lower TBR_{req}, lower $I_{startup}$

Learning from fuel cycles: importance of doubling time





Targeting faster doubling times (t_d) means we **need more efficient, advanced blankets** (especially if we struggle to achieve high TBE). Relaxing t_d targets means more achievable TBR for a given TBE.

However, if we relax t_d too much, once we have a working FPP, it will take **generations** to penetrate the electricity market and contribute to grid decarbonization. (This is motivation for the LIB concept - possibly much faster doubling times)

Learning from fuel cycle models: tritium residence time in blanket

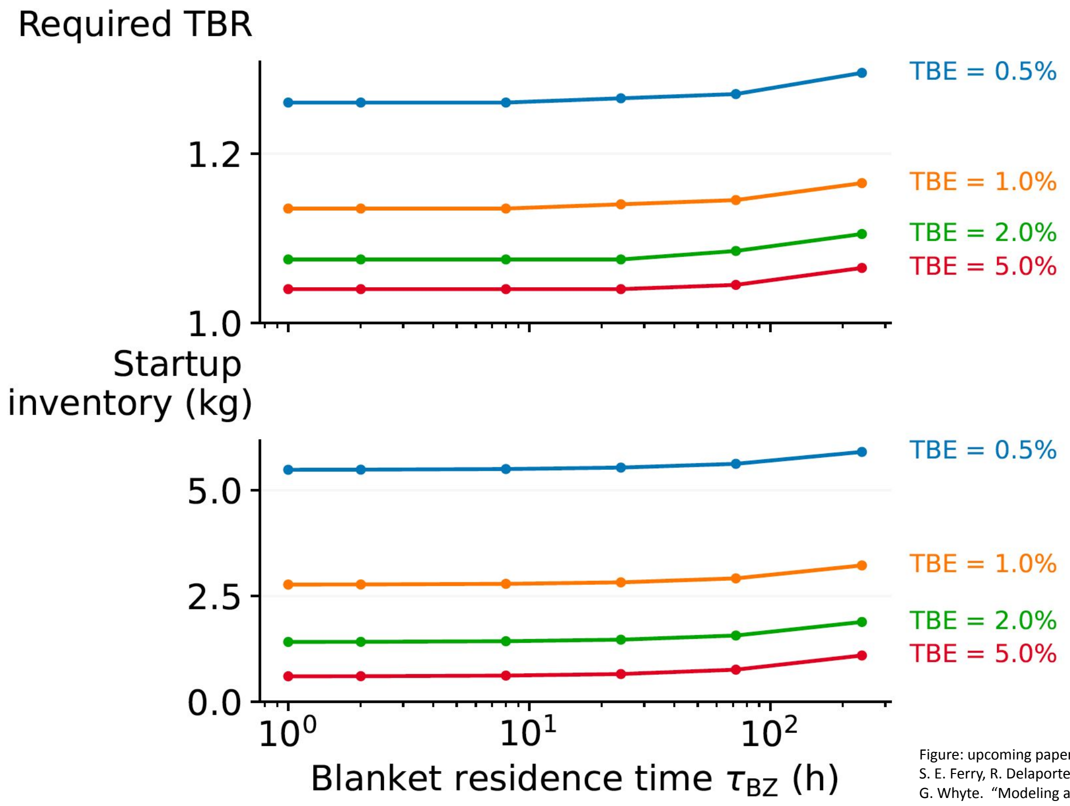
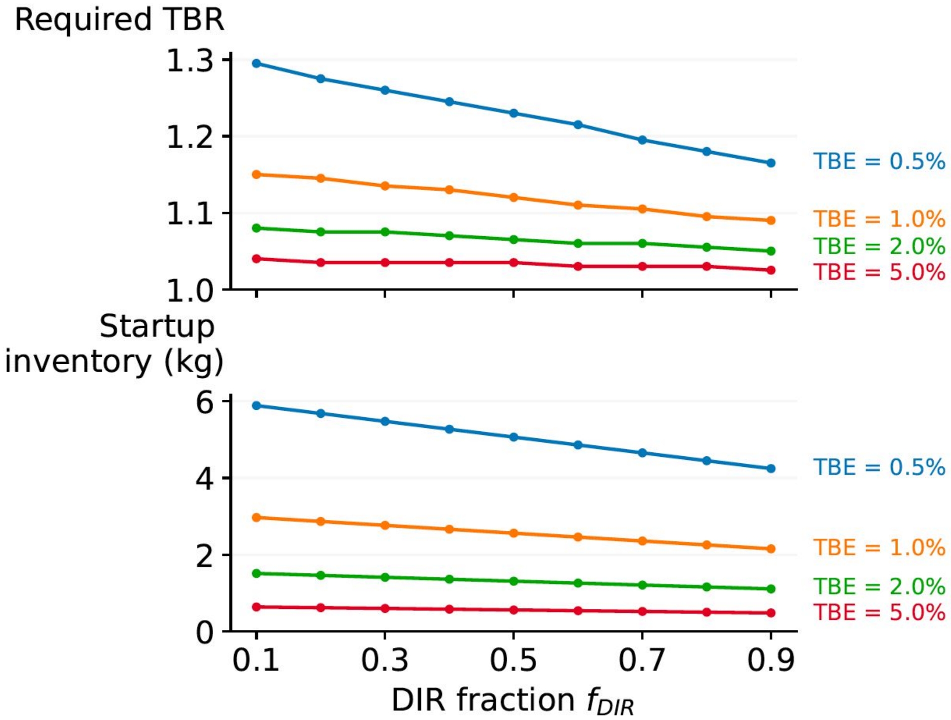


Figure: upcoming paper, S. Meschini, S. E. Ferry, R. Delaporte-Mathurin, D. G. Whyte. "Modeling and analysis of the tritium fuel cycle for ARC- and STEP-class D-T fusion power plants"

Unless the tritium residence time in the blanket is very high, we don't see much impact on $I_{startup}$ or TBR_{req}.

Learning from fuel cycle models: impact of DIR



Direct Internal Recycling may not be required (or worth the added system complexity) if we can achieve sufficiently high TBE, but will likely be necessary if TBE is low.

Figure: upcoming paper, S. Meschini, S. E. Ferry, R. Delaporte-Mathurin, D. G. Whyte. "Modeling and analysis of the tritium fuel cycle for ARC- and STEP-class D-T fusion power plants"

Learning from fuel cycle models: impact of availability factor (AF)

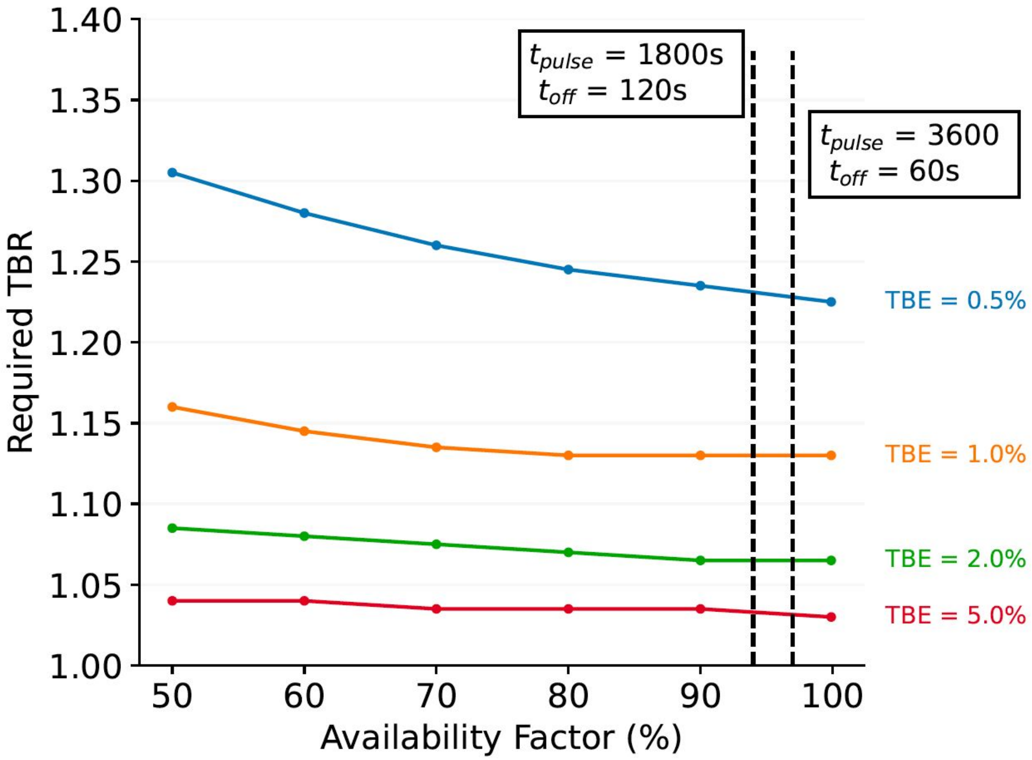
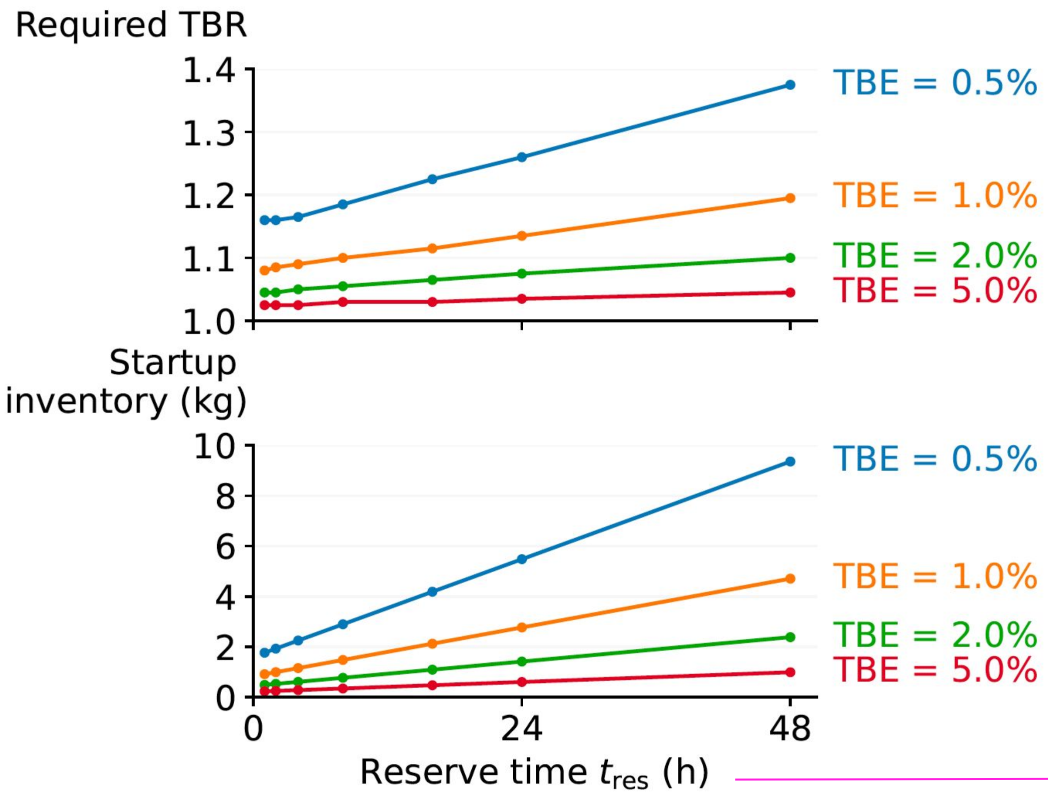


Figure: upcoming paper, S. Meschini, S. E. Ferry, R. Delaporte-Mathurin, D. G. Whyte. "Modeling and analysis of the tritium fuel cycle for ARC- and STEP-class D-T fusion power plants"

Low availability factors are associated with higher TBR_{req}. At low TBE, TBR_{req} is too high to be achievable even as AF approaches 100%.

Learning from fuel cycle models: impact of reserve time



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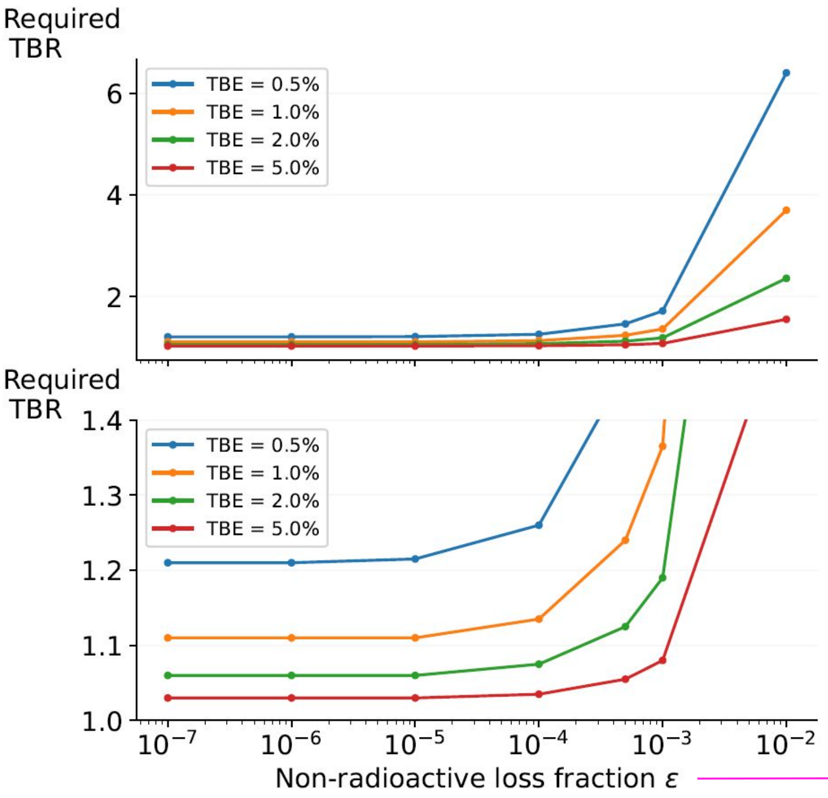
If we can tolerate a shorter reserve time, it is much easier to achieve tritium self sufficiency.

t_{res} = how long we can operate using only tritium reserves

Learning from fuel cycle models: impact of non-radioactive tritium losses



High non-radioactive tritium losses ($\varepsilon > 10^{-4}$) make it hard to achieve tritium self sufficiency.

ε = tritium loss fraction due to leakages, permeation through components, etc.

Design implications for an example ARC-class FPP

- **FC (fuel cycle) advances**: AF, f_{DIR}, and t_p are all improved relative to baseline
- **PO (plasma operations) advances**: TBE is improved relative to baseline
- "**Ambitious**" case: aim for doubling time of just one year

See how **required TBR** and $I_{startup}$ change

	Baseline case	Ambitious (t_d=1 y)			Moderate (t_d =2 y)		
Parameters	No FC or PC advances	FC advances only	PO advances only	Both	FC advances only	PO advances only	Both
AF (%)	70	90	70	90	90	70	90
TBE (-)	0.02	0.02	0.1	0.1	0.02	0.1	0.1
f_{DIR} (-)	0.3	0.7	0.3	0.7	0.7	0.3	0.7
t_p [h]	4	1	4	1	1	4	1
t_d [y]	1 or 2	1	1	1	2	2	2
TBR_r (-)	1.113 ($t_d = 1$) 1.067 ($t_d = 2$)	1.064	1.029	1.022	1.039	1.017	1.012
I_{startup} [kg]	1.14 ($t_d = 1$) 1.14 ($t_d = 2$)	1.12	0.33	0.30	1.12	0.33	0.30

Key takeaways: (1) Fuel cycle modeling helps us understand tradeoffs inherent in design decisions and choice of operational targets. (2) Improving TBE dramatically lowers $I_{startup}$. Overall, the more fuel cycle technology improves, the more achievable TBR_{req} gets.

Part II: Introduction to breeder blankets (with an emphasis on FLiBe)

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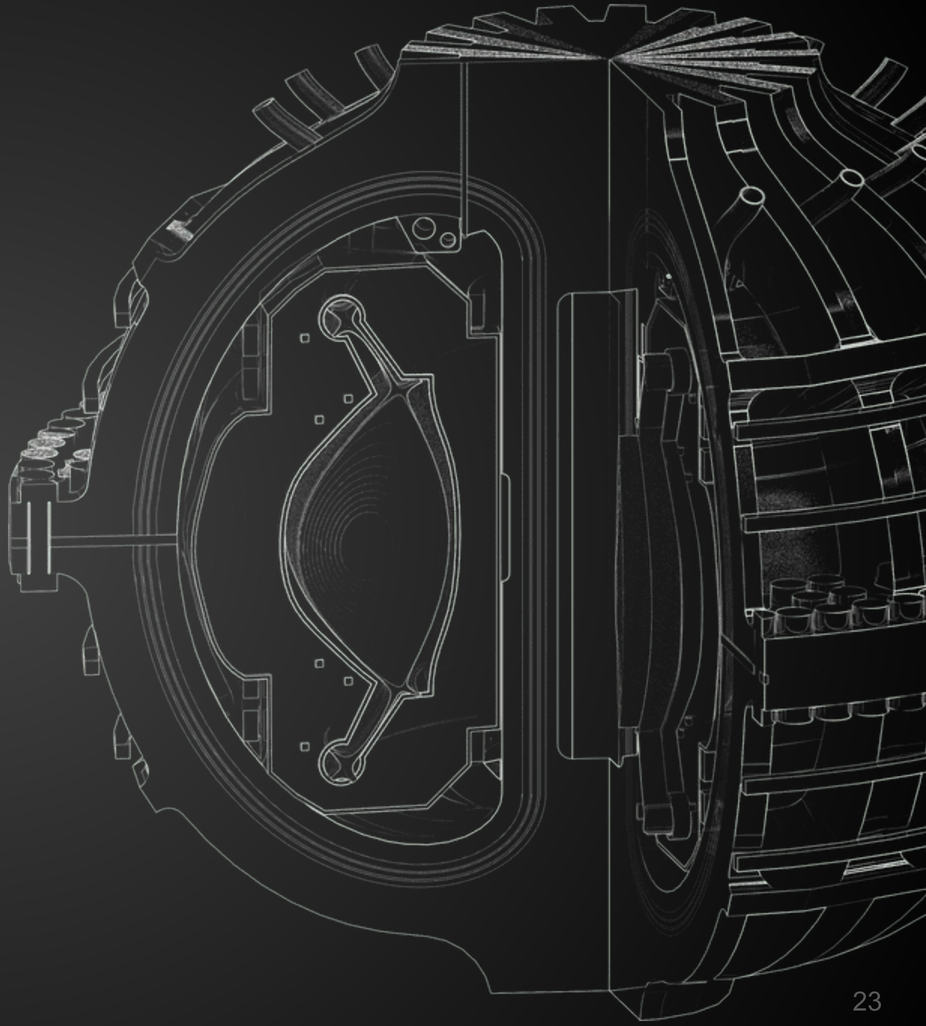
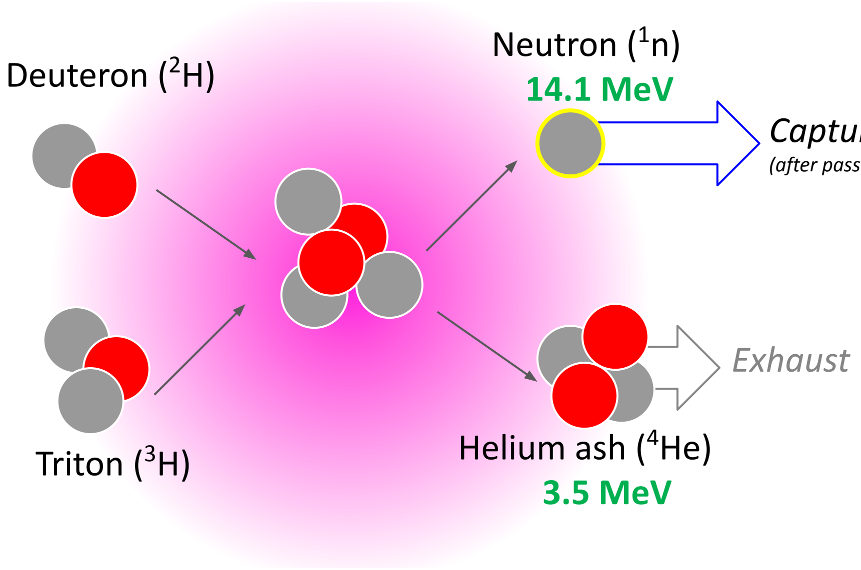
Tommy Fuerst (INL)
MIT PSFC LIBRA team



The basics: deuterium and tritium as fusion fuel

Fuse DT in the plasma:



*Capture in the **blanket***
(after passing through first wall structures)

Shielding:
Moderate + absorb neutrons to shield the magnets

Heat capture: generate power

Tritium breeding:
Make more fuel in the breeder, which contains **lithium**

$$\text{Li-6} + n \longrightarrow \alpha + T + 4.78\ \text{MeV}$$
$$\text{Li-7} + n \longrightarrow T + n' + \alpha - 2.47\ \text{MeV}$$

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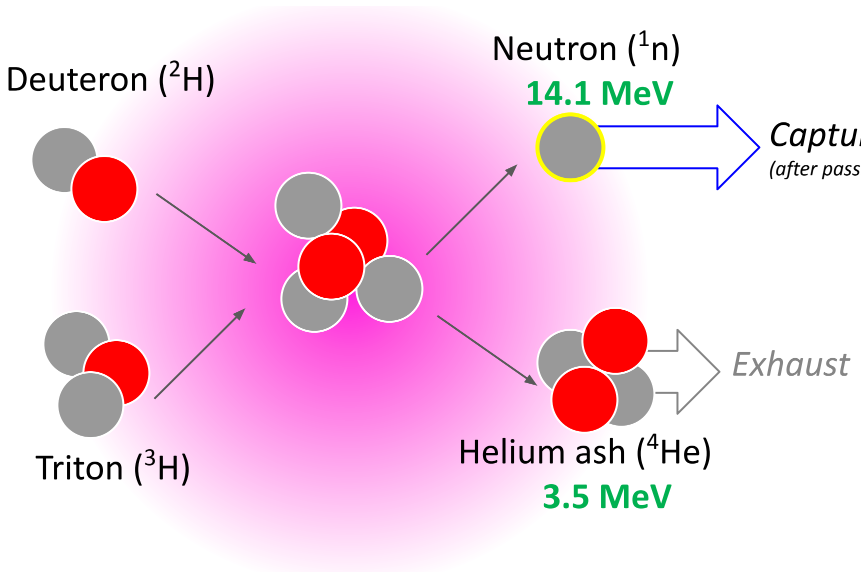

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Lithium solids
Li_2TiO_3, $LiAlO_2$
Li_2ZrO_3, Li_4SiO_4

Pure Li

PbLi

Lithium-fluoride salts
FLiBe ($LiF-BeF_2$)
FLiNaK (LiF-NaF-KF)
FLiNaBe($LiF-BeF_2-NaF$)

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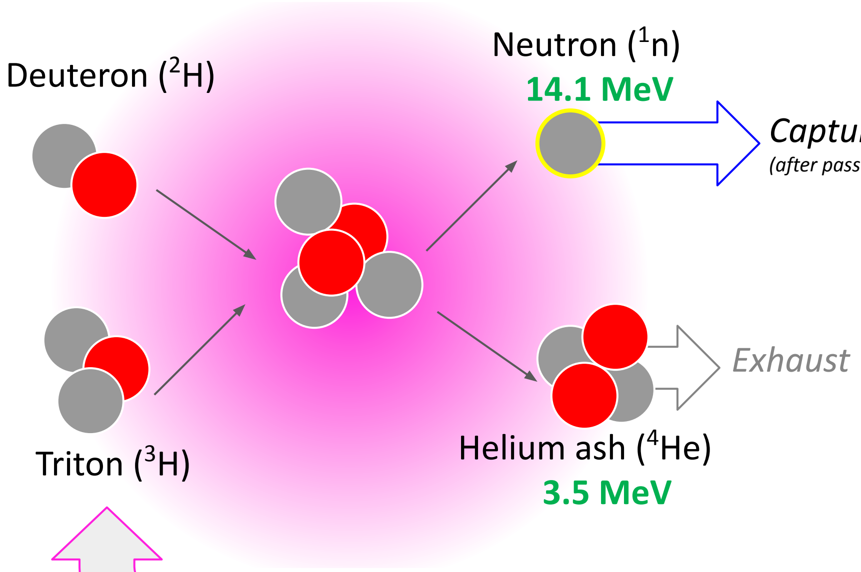

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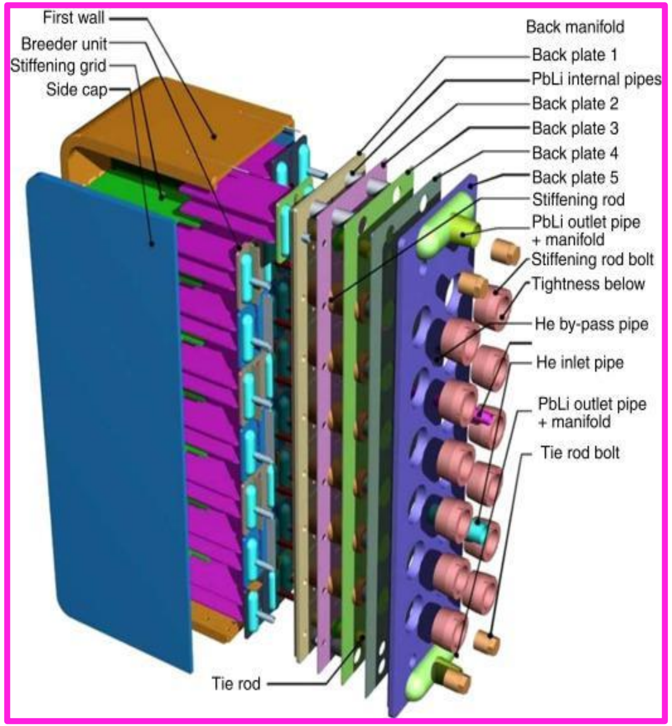
FLiBe/FLiNaBe/FLiNaK: not a priority in the ITER/DEMO pathway

TBM	Test blanket module for ITER (3 TBM ports, 2x TBM per port)	Developed by
DCLL	Dual-coolant lead-lithium	USA
HCCB*	Helium-cooled ceramic breeder	China
HCCR*	Helium-cooled ceramic reflector	Korea
HCLL*	Helium-cooled lithium lead	EU
HCPB*	Helium-cooled pebble bed	EU
HCSB	Helium-cooled solid breeder	India
LLCB*	Lead-lithium cooled ceramic breeder	India
WCCB*	Water-cooled ceramic breeder	Japan
WCLL	Water-cooled lithium-lead	EU

*selected for ITER TBM

Example of HCLL TBM design



Aiello G, de Dinechin G, Forest L, *et al.* HCLL TBM design status and development. Fusion Engineering and Design 2011; 86:2129–34.

Most past tritium breeding experiments focus on lithium solids

Yellow = used 14 MeV neutron source

Lithium solids
1983 TRIO (Argonne/ORNL)
1984 LILA (France - CEA)
1985 LISA (France - CEA)
1986 VOM (JAERI)
1986 EXOTIC 1 through 7 (European Fusion Technology Program)
1988 TTTEx (U Tokyo)
1988 MOZART (France – CEA)
1988 TEQUILA (France – CEA)
1991 OKTAVIAN (Osaka U) **pure lithium (solid)*
1991 BEATRIX-II (JAERI/Chalk River/PNNL)
1993 SIBELIUS (Argonne/CEA)
1994 ARIES Tokamak Reactor Study (Argonne, Chalk River)
1998 CRITIC I and II (Chalk River)
1998 JAERI Li2TiO3 experiment (JAERI)
1999 IVV-2M (Russian DEMO)
2001 EXOTIC 8+ (European Fusion Technology Program)
2007 Pebble Bed Assembly HCPB (EU-DEMO)
2007 Water-cooled mockup blanket (JAEA)
2011 HIDOBE HCPB (EU-DEMO)
2012 Batistoni HCPB mockup (EU-DEMO)
2015 LLCB mockup (BARC)
2015 CIPITISE HCCB (China DEMO)
2019 SAKURA (Kazakhstan)

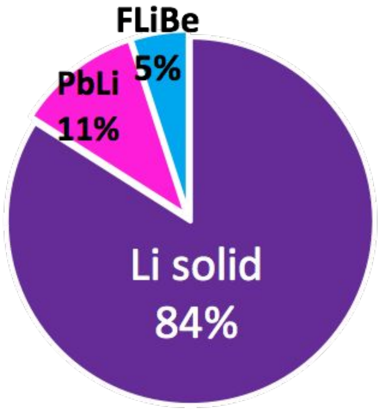
*** Also:**
PNNL: TMIST Li solid irradiations at ATR (NNSA)

Liquid PbLi
1990 LIBRETTO (Neth./Italy/France)
2012 Batistoni HCLL mockup (EU-DEMO)
2015 LLCB mockup (BARC)

FLiBe
1996 INTREXFLIBE (Japan)
2000 JUPITER-II (ORNL/Japan)
2009 Kyoto FLiBe irradiation (Japan)

FLiBe proposed for FFHR blanket

*** Also:**
2018 - Osaka University FLiNaK irradiations (10.7 MeV n)



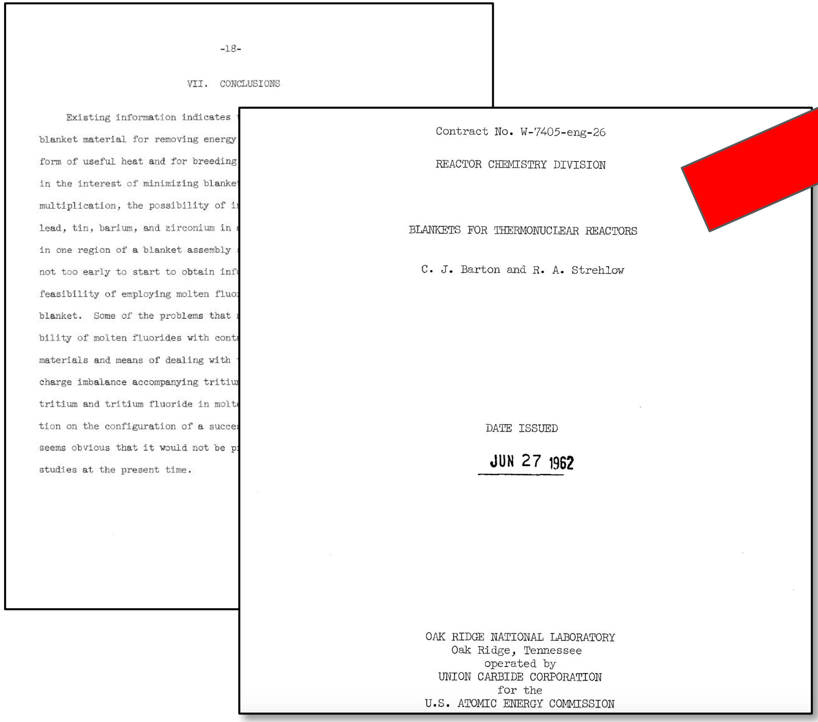
IMPORTANT:
None of the 8 milestone program awardees are considering Li-solid breeder blankets at this time.

A quick reference guide comparing the main breeder options being considered

Concept	Tritium	Safety	Chemical	Thermomechanical
Liquid Lithium	• High T affinity+ solubility • **Extraction is difficult** • **Excellent TBR: 100% of breeder can make T** • Good for concepts with radial space constraints • Doesn't need enrichment	• Flammability • **Extreme reactivity with air and water** • Gamma attenuation • Need to model failure modes • Need monitoring techniques	• Purification challenges • Corrosion barriers • Impurity-driven corrosion • Generally good compatibility • Va alloys, RAFM good choices • 16-17% Li	• Pressure wave dampening • Low melting point (180 C) • **Need to insulate electrically from the B field** • Difficult to make self-cooled design until we know more about MHD
Liquid PbLi	• Low T solubility • **Easier to extract tritium** • May permeate through materials • Likely to require Li-6 enrichment • Lead is a neutron multiplier (TBR boost)	• Not chemically reactive with air • Chemically reactive with water at high temperatures • Hot liquid in VV in accident scenario • Alumina TPBs not stable • Higher activation • Compatible with low-activation materials	• **Corrosive** at high temperatures • Often paired with RAFM or SiC • May require coatings to ensure compatibility	• Low melting point, easier to keep molten • Can operate at low or high temp • **B field effects, requires** insulation dissolution? • Low viscosity
Molten FLiBe	• Low T solubility • **Easier to extract tritium** • May permeate through materials • Beryllium is a neutron multiplier (TBR boost)	• **Beryllium toxicity** • T permeation through materials • Low activation material solution needed for high-field approach	• Redox control • No standard commercially available reference electrode • T kinetics not well characterized • **Purification for supply chain** • Compatibility depends on salt purity/chemistry	• **Low electrical conductivity, minimal MHD pressure drop** • High heat capacity • High melting point (harder to keep molten if using lower operating temps) • High viscosity can add to pumping challenge
Solid breeder (lithium ceramic)	• **Most technologically mature tritium extraction** • Requires Be multiplier addition • Pebble bed form is common • **Generally a lower TBR than can be achieved with the other concepts**	• Stable behavior in off-normal scenarios; quantity of activated material may be high • Beryllium toxicity from multiplier	• Fewer compatibility issues than liquid coolants • Pair with RAFM steel, He or H_2O coolant	• Solid property changes under irradiation add complexity • Need to understand how behavior changes under irradiation • Breeder regions require external coolant

FLiBe blankets: not a new idea… but lots of work left to do

1962 ORNL report by Barton and Strehlow
"Blankets for thermonuclear reactors"


-18-

VII. CONCLUSIONS

Existing information indicates
blanket material for removing energy
form of useful heat and for breeding
in the interest of minimizing blanke
multiplication, the possibility of i
lead, tin, barium, and zirconium in
in one region of a blanket assembly
not too early to start to obtain inf
feasibility of employing molten fluo
blanket. Some of the problems that
bility of molten fluorides with cont
materials and means of dealing with
charge imbalance accompanying tritiu
tritium and tritium fluoride in molt
tion on the configuration of a succe
seems obvious that it would not be p
studies at the present time.

Contract No. W-7405-eng-26

REACTOR CHEMISTRY DIVISION

BLANKETS FOR THERMONUCLEAR REACTORS

C. J. Barton and R. A. Strehlow

DATE ISSUED

JUN 27 1962

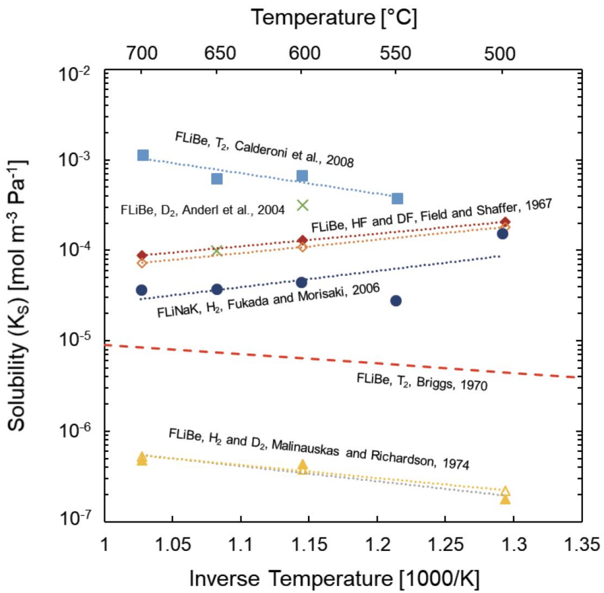
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

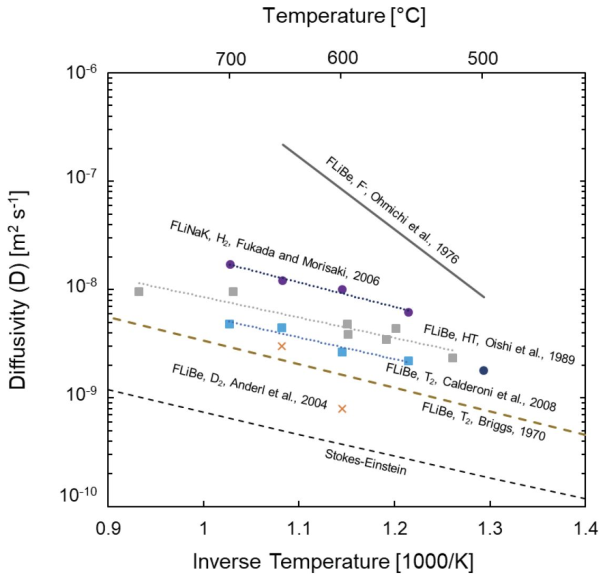
"…molten LiF-BeF$_2$ is a promising blanket material for removing energy from a thermonuclear reactor in the form of useful heat and for breeding tritium…"

*"Some of the problems that need to be examined are: **compatibility** of molten fluorides with container and neutron multiplying materials and means of dealing with the corrosion problem … and **solubility** of tritium and tritium fluoride in molten LiF-BeF$_2$."*

60 years later, we are still investigating the same topics!

We need to close big gaps in our understanding of how tritium behaves in molten fluoride salt.





Key takeaways:

1. Orders-of-magnitude spread in experimental data for solubility and diffusivity of hydrogenic species in fluoride salts
2. There's not that much experimental data, period.
   - Even less if you look at just FLiBe and tritium

Humrickhouse, Paul W., and Thomas F. Fuerst. *Tritium Transport Phenomena in Molten-Salt Reactors*. No. INL/EXT-20-59927-Rev000. Idaho National Lab.(INL), Idaho Falls, ID (United States), 2020.

Also see R. Delaporte-Mathurin's hydrogen property dashboard: https://htm-dashboard-uan5l4xr6a-od.a.run.app/

FLiBe ecosystem at a glance

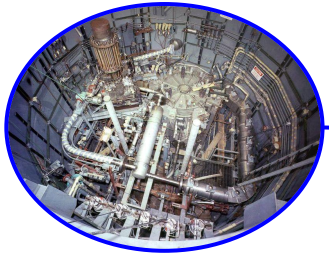
Fission research

Concepts

- LFTR (liquid fluoride thorium reactor)
- AHTR (advanced high temperature reactor - ORNL)
- TMSR - thorium MSR (China); type of FHR, FLiBe as primary coolant

Active tests

- SALIENT-02 loop (planned; Pettens)
- ORNL liquid salt test loop
- Hot FLiBe zone at LR-0 (Czech Republic + ORNL)
- FLiBe research at MIT Nuclear Reactor Laboratory
- FLiBe research at INL STAR facility



ORNL MSRE:

- Operated 1965-1969
- FLiBe carrying enriched U
- Graphite moderated
- 650C operating temperature

Suppliers

Old salt from MSRE

MATERION

AlphaTech RESEARCH CORP

Fusion research

Past

- Japan: Force-free helical reactor (FLiBe breeding)
- 1996 INTREXFLIBE breeding campaign (Japan)
- 2000s JUPITER-II breeding campaign (ORNL/Japan)
- 2009: Kyoto University FLiBe irradiation tests

Present

- 2020+: MIT PSFC LIBRA and BABY experiments
- ORNL FLiBe materials compatibility testing

Fission industry

Kairos Power

FLiBe-cooled KP-FHR (TRISO fuel)
Currently operating: Engineering Testing Unit (non-nuclear)
Next: HERMES (reduced-scale NPP)

Flibe ENERGY

FLiBe-cooled, graphite-moderated thermal reactor concept; FLiBe carrier for nuclear fuel

Note: future iterations of Terrestrial Energy's IMSR may use FLiBe

Fusion industry

Commonwealth Fusion Systems

ARC FPP concept:
FLiBe liquid immersion blanket for tritium breeding and heat transfer

Xcimer Energy Company

FLiBe-protected first wall (HYLIFE design)

KYOTO FUSIONEERING

Supporting FLiBe breeding research (UNITY facility)

FLiBe presents a lot of R&D challenges.

Hazards make handling complicated and expensive:
- Beryllium exposure
- HF
- Difficult to find space rated for beryllium handling

Supply chain issues:
- Not commercially available
- Difficult to synthesize and purify in large quantities
- Limited operational experience

Limited resources:
- Past research funding has prioritized PbLi and solid breeder blankets

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Kyoto Fusioneering

Supporting FLiBe breeding research (UNITY facility)

Near-term R&D strategies: combine resources to solve common challenges

Minimize tritium production

- Understand T solubility and diffusivity in FLiBe
- T permeation barriers
- FLiBe radiochemistry
- Effective T extraction

Maximize tritium production

Fission industry

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Currently operating: Engineering Testing Unit (non-nuclear)
Next: HERMES (reduced-scale NPP)

Flibe ENERGY

FLiBe-cooled, graphite-moderated thermal reactor concept; FLiBe carrier for nuclear fuel

Fission also helpful for expertise on tritium management (especially Canadian CANDU operators/associated vendors)

Note: future iterations of Terrestrial Energy's IMSR may use FLiBe

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KYOTO FUSIONEERING

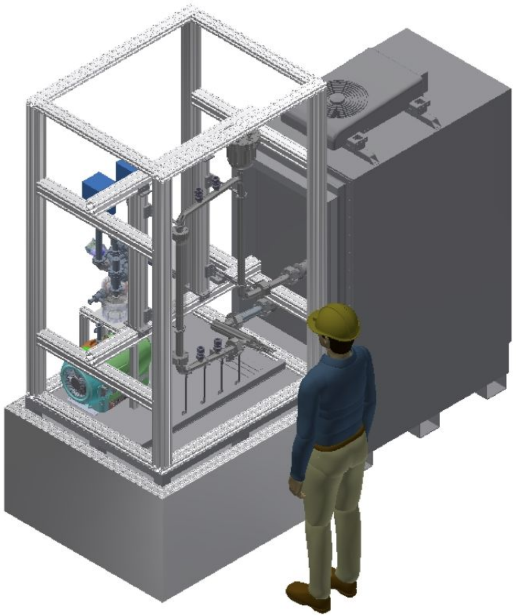
Supporting FLiBe breeding research (UNITY facility)

Example 1 : Molten Salt Tritium Transport Experiment at INL

- ***MSTTE** is a semi-integral tritium transport experiment for flowing fluoride salt systems.*
- Location: Safety and Tritium Applied Research (STAR) facility
- **Objectives:**
  - **(1) Safety code validation data.**
  - **(2) Test stand for tritium mitigation technology.**
- *Major Equipment:*
  - **Copenhagen Atomics Salt Loop**: salt tank, pump, & flow meter
  - **External Test Section:** hydrogen injection, permeation, & plenum
- *Phased approach*
  - Phase I: FLiNaK and D_2
  - Phase II: **FLiBe** and D_2
  - Phase III: **FLiBe** and T_2

Molten Salt Reactor PROGRAM

Idaho National Laboratory

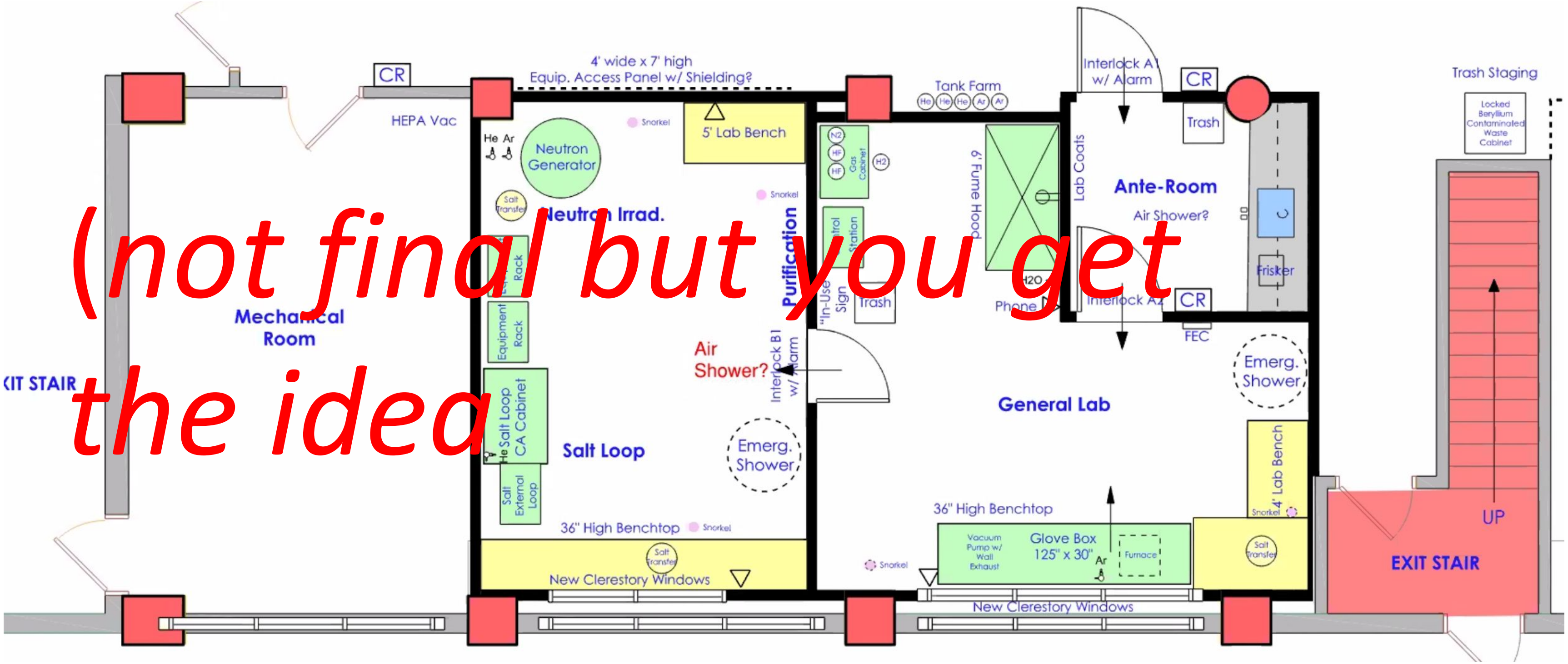


Example 2 : Molten Salt Lab at MIT (plans currently being finalized)

MIT NRL: working with FLiBe for fission research
MIT PSFC: working with FLiBe for fusion research

→

Share resources/funding to build a lab capable of handling FLiBe & carrying out neutron irradiations

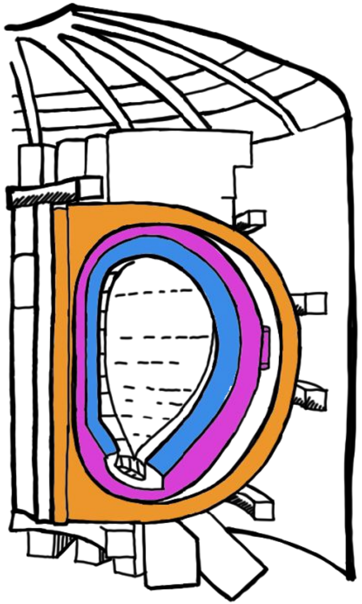


(not final but you get the idea

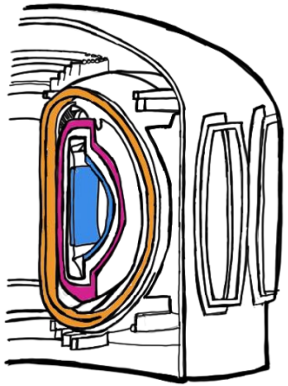
Why does FLiBe make sense now?

(disclaimer: different fusion power plant concepts will have different design constraints and tradeoffs to consider -- FLiBe is not a one-size-fits-all breeder choice for every concept)

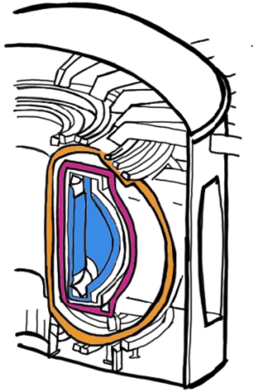
Orange = toroidal field magnet
Pink = vacuum vessel
Blue = blanket



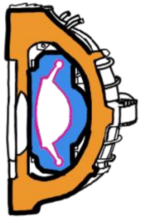
EU-DEMO
R = 9 m
P = 3 GW



ARIES-RS
R = 5.5 m
P = 2.2 GW



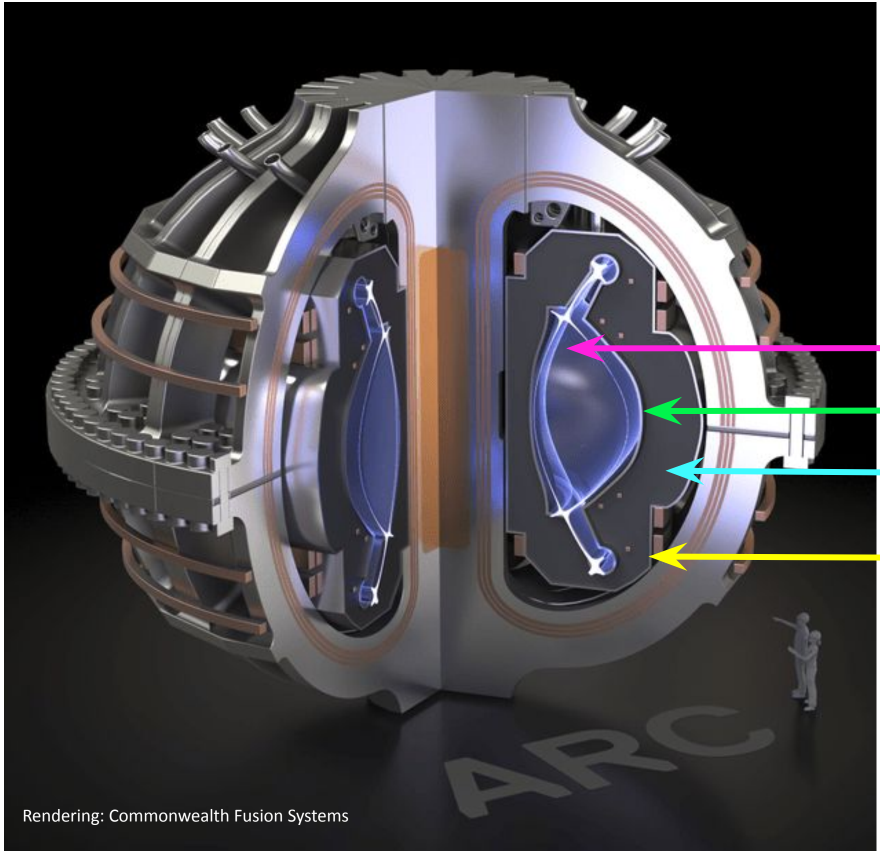
ARIES-AT
R = 5.2 m
P = 1.7 GW



ARC
R = 3.5 m
P = 0.5 GW

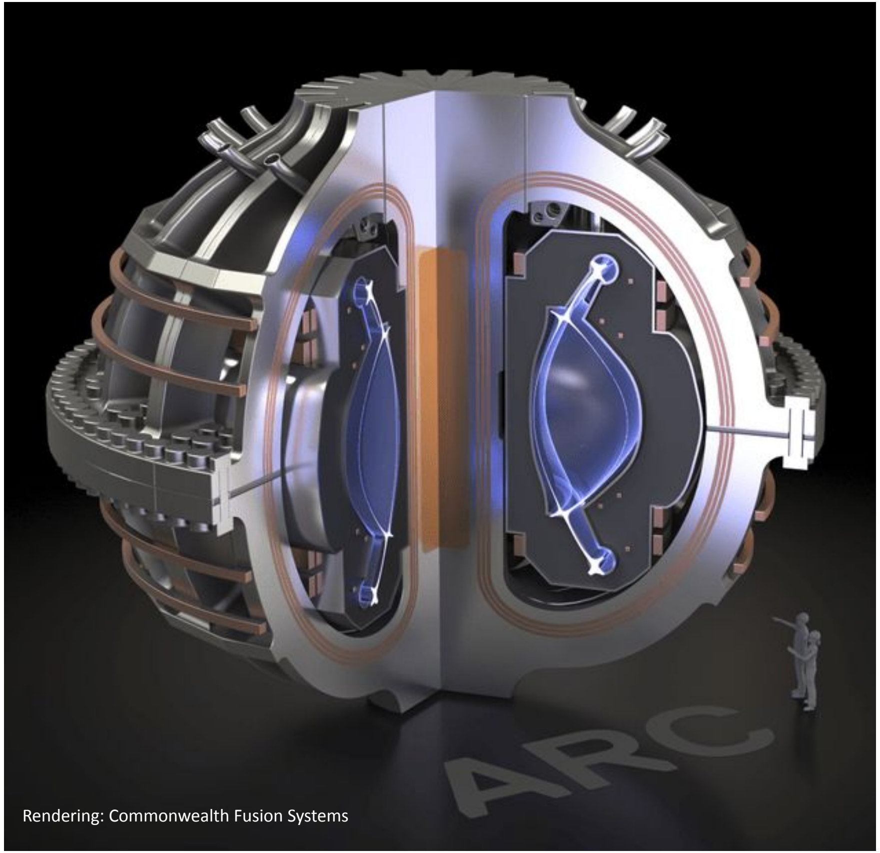
The Liquid Immersion Blanket (LIB) for ARC-class fusion power plants

NOTE: the FLiBe LIB can be adapted to non-ARC concepts.



Rendering: Commonwealth Fusion Systems

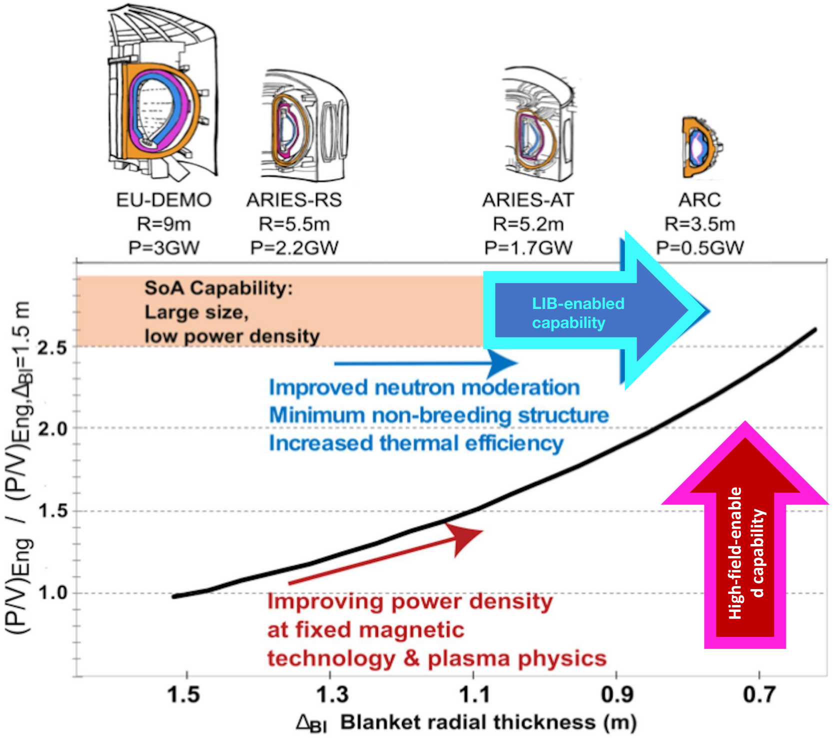
Advantages of the FLiBe LIB approach



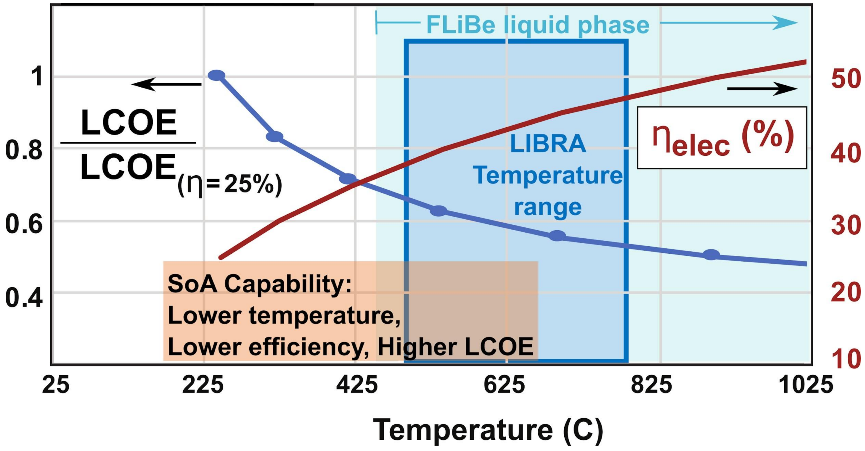
Rendering: Commonwealth Fusion Systems

- **Maximize solid coverage of neutron source to maximize TBR and enhance magnet shielding**
- FLiBe interaction with magnetic fields is **minimal** compared to PbLi liquid breeders
- No solid sector handling or changeout
- FLiBe **can be pumped out** of tank to do maintenance on VV components
- **Online chemistry control** to mitigate corrosion and keep T in preferred form (T_2, HT over TF)
- T has **low solubility** in FLiBe; easier extraction
- MIT neutronics results indicate that FLiBe is a lifetime component: you don't need to add "fresh" Li-6
  - May not need to enrich Li-6, either
- Minimized complexity; blanket tank is the main fabricated component
- Excellent **thermal properties** - good for high power density concepts like ARC

FLiBe LIB handles higher temperature, higher neutron fluence without sacrificing T breeding



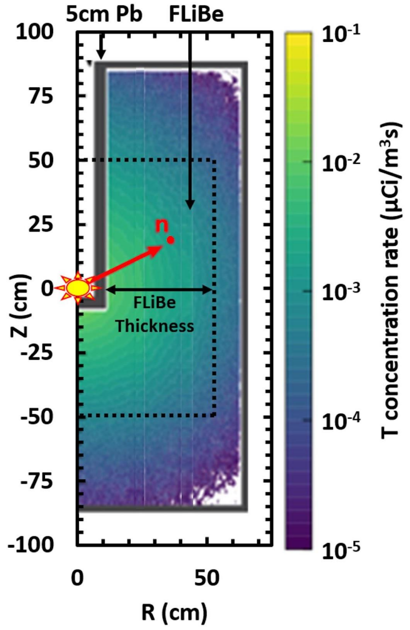
- HTS approach enables high power density in a smaller package
    - Comes with higher temperatures and higher neutron flux
- FLiBe LIB capable of dealing with the higher neutron & heating load
- Higher temperatures → higher efficiency power plant



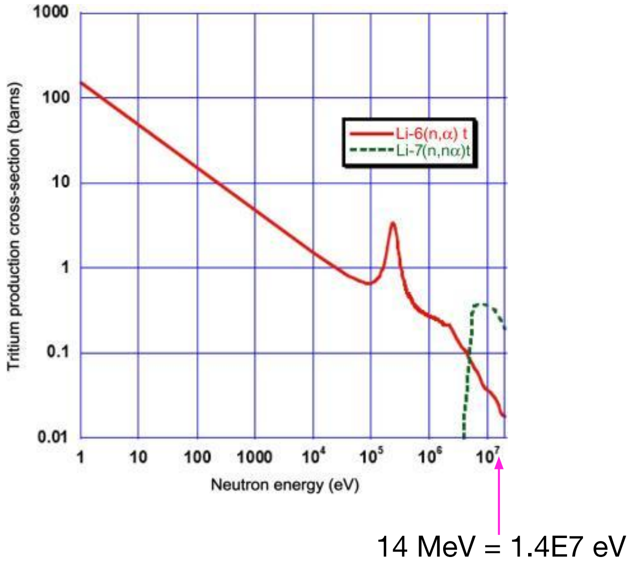
Ferry, Sara E., Kevin B. Woller, Ethan E. Peterson, Caroline Sorensen, and Dennis G. Whyte. "The LIBRA Experiment: Investigating Robust Tritium Accountancy in Molten FLiBe Exposed to a DT Fusion Neutron Spectrum." *Fusion Science and Technology* 79, no. 1 (2023): 13-35.

Quasi-detour 1: *a quick note on Li-7 vs Li-6*

Neutronics simulations of the LIBRA tank show that the **majority of tritium breeding** happens next to the neutron source



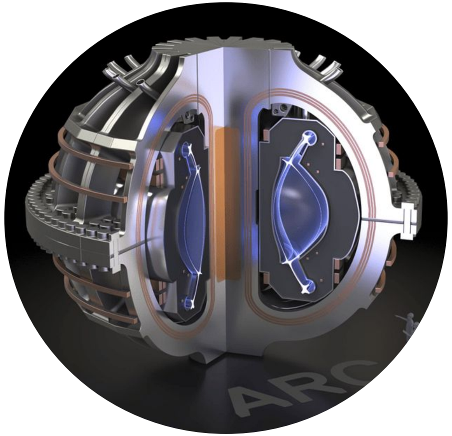
Tritium production cross-sections for Li-6 and Li-7



Li-7 contributions are very important to T breeding, especially near the neutron source

- Most neutrons there aren't thermalized
- High density of neutrons ($1/r^3$ dependence)
- Lots of Li-7 (92%)

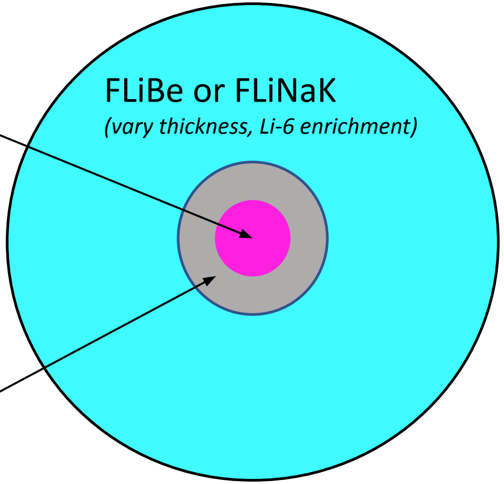
The FLiBe LIB geometry should ease requirements on Li-6 enrichment



The ARC-class tokamak's LIB **completely surrounds** the neutron source (the toroidal plasma in the VV) with breeder material

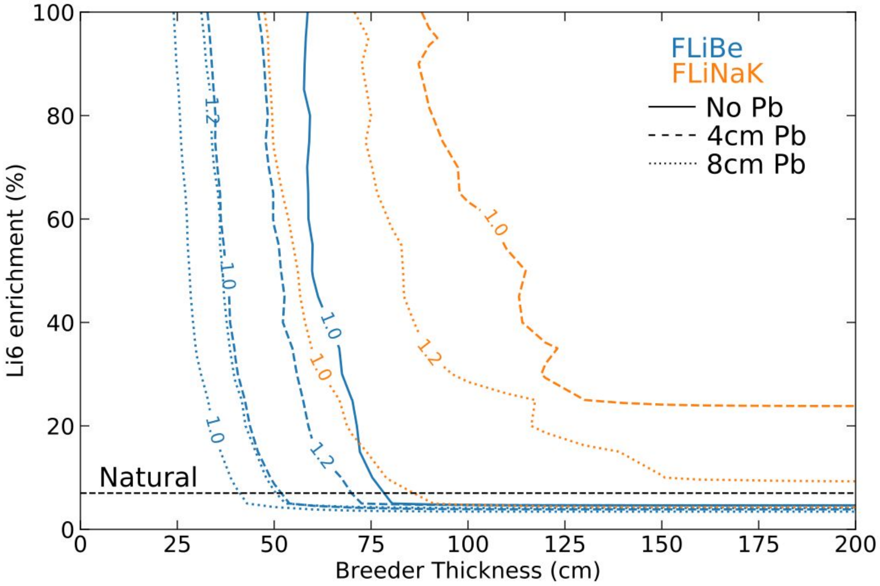
Quasi-detour 2: *why not avoid beryllium and use FLiNaK?*

Set up a simple illustrative case in OpenMC:



As we vary these basic parameters, what's the effect on tritium breeding ratio (TBR)?

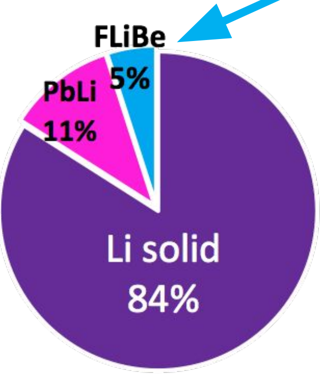
Results: we need a much bigger radial build to achieve a given TBR with FLiNaK instead of FLiBe



Pb and Be are both neutron multipliers, so their presence boosts TBR.

Ethan Peterson, MIT LIBRA team

What can we learn from prior experiments looking at tritium breeding in FLiBe?



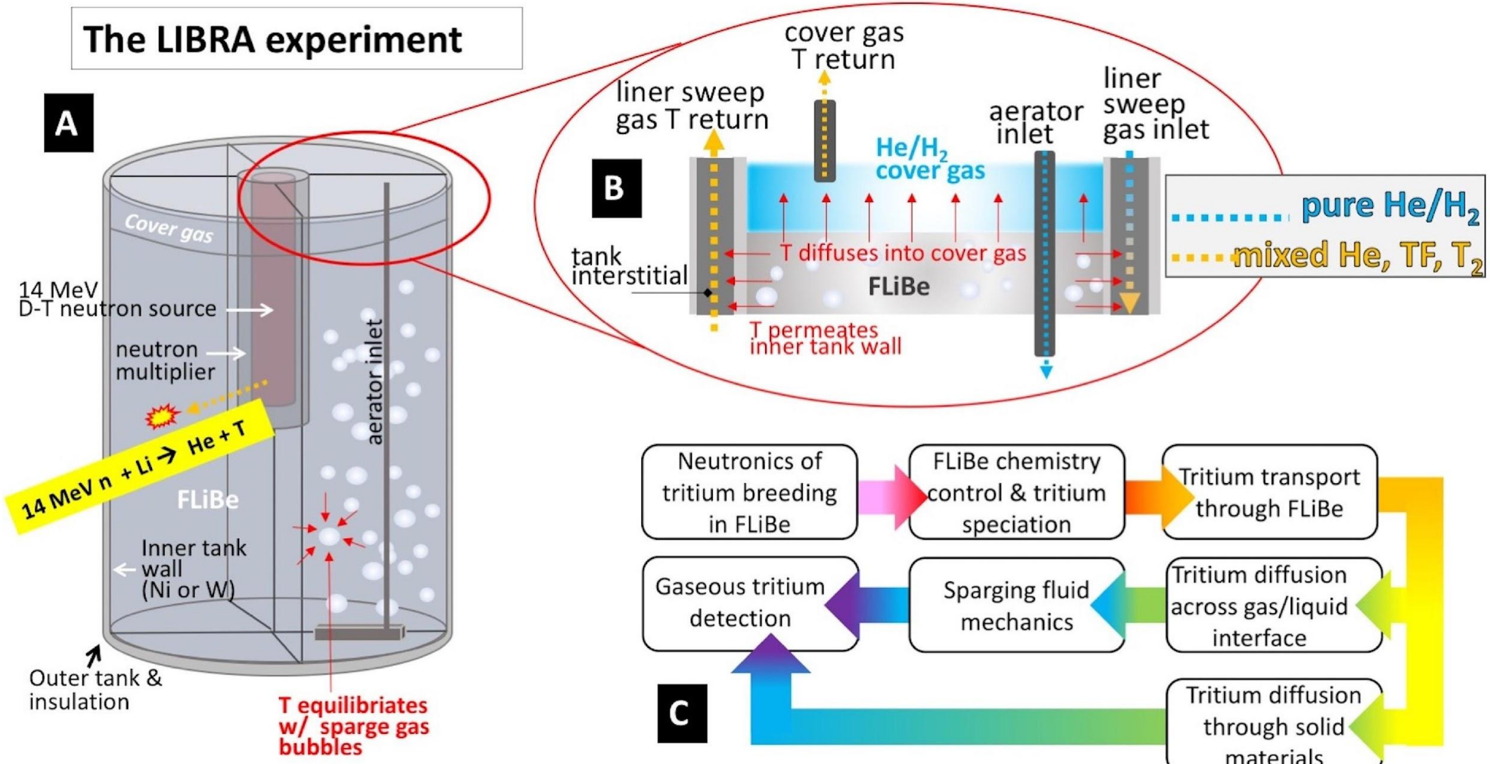
		Radiation		T accountancy				T kinetics			FLiBe chemistry			
		Neutrons	14 MeV fusion spectrum	Tritium	Measure TBR	Complementary neutronics	T extraction, detection, & speciation	T transport in Flibe	T diffusion at gas/liquid IF	T diffusion through solids	Redox control	Material compatibility	Sparging fluid effects	> 1 L flibe
LIBRA		Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y
INTREXFLIBE [68]	[99]	Y	Y	Y	N	N	Y	N	Y	N	N	Y	Y	N
	[94]	Y	N	Y	N	N	Y	N	N	N	N	N	N	N
	[101]	Y	N	Y	N	N	Y	Y	Y	N	N	N	Y	N
	[102, 103]	Y	N	Y	N	N	Y	N	Y	N	N	N	Y	N
	[105]	Y	N	Y	N	N	Y	N	Y	Y	N	N	N	N
	[118]	N	N	N	N	N	N	N	N	N	N	Y	N	N
JUPITER II [97]	[107, 108]	N	N	N	N	N	N	D, H used as sub. for T			N	N	N	N
	[112]	N	N	Y	N	N	Y	Y	N	N	N	N	N	N
	[89]	Y	N	Y	N	Y	Y	Y	N	N	N	N	N	N
Redox experiments	[120-122]	N	N	N	N	N	N	N	N	N	Y	N	N	N
	[123]	N	N	N	N	N	N	N	N	N	Y	Y	N	N
FLiBe corrosion studies	[125, 126]	N	N	N	N	N	N	N	N	N	N	Y	N	~
	[133]	Y	N	Y*	N	N	N	N	N	N	N	Y	N	N
	[104, 129-132, 134, 136-138]	N	N	N	N	N	N	N	N	N	N	Y	N	N

Non-experimental papers or papers using FLiNaK were not included.

* FLiBe was Li-7 enriched to reduce T production

~ Loop tests with flowing FLiBe.

LIBRA: Liquid Immersion Blanket - Robust Accountancy



MIT PSFC
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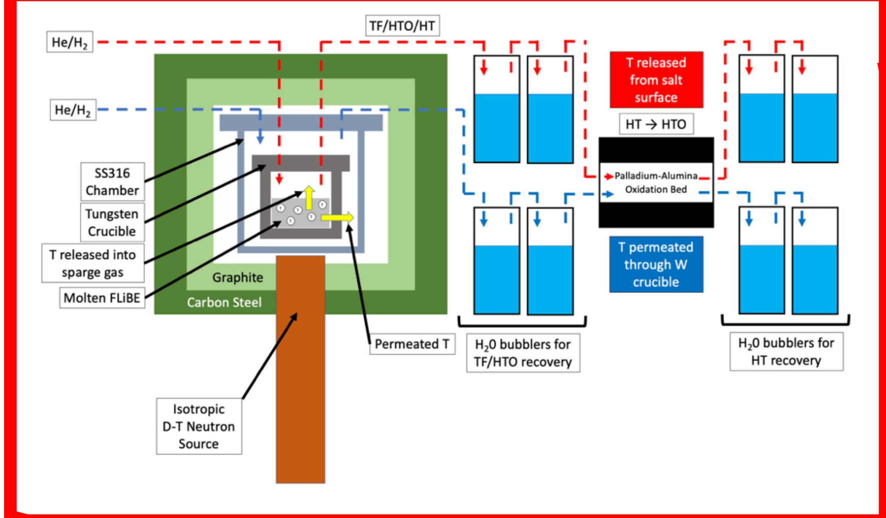
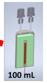
Raluca Scarlat (UC Berkeley)

MIT PSFC: intermediate tritium breeding tests before LIBRA

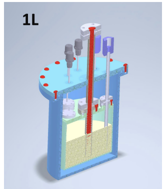


“Dress rehearsals” using existing neutron generator + facilities while molten salt lab under construction

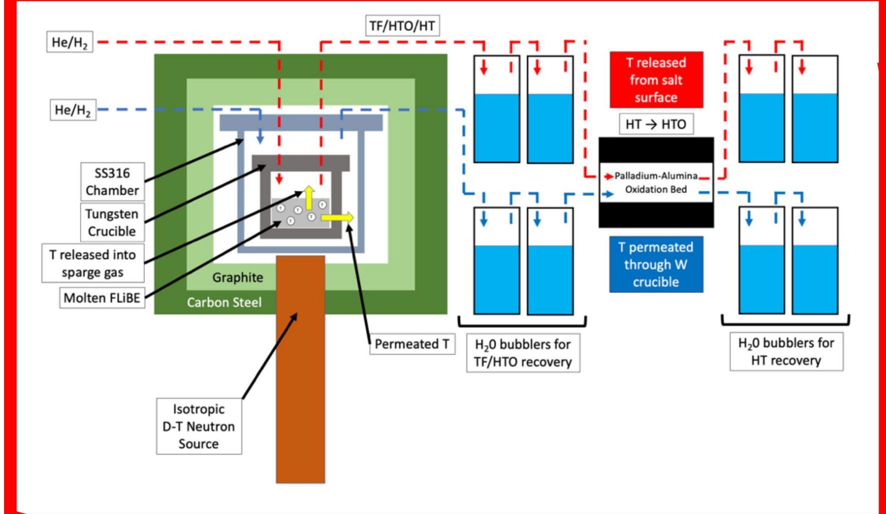
100 mL FLiBe tests



1 L FLiBe test



MIT PSFC: intermediate tritium breeding tests before LIBRA



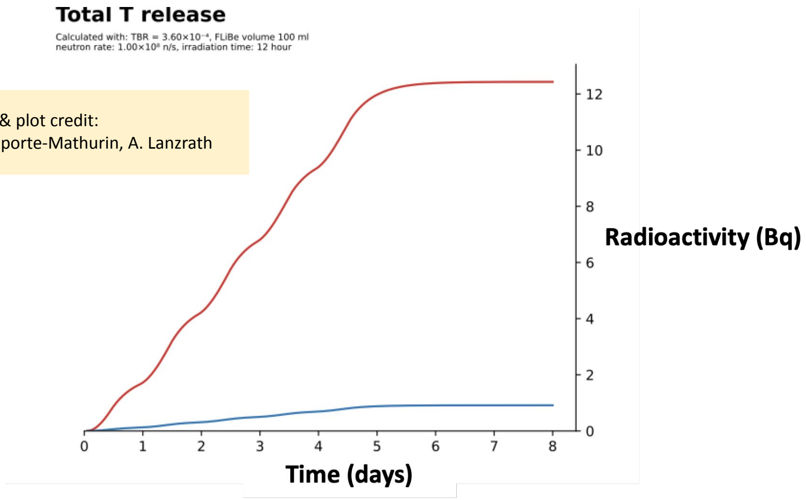
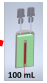


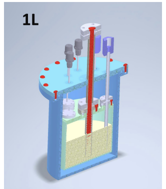
Figure & plot credit:
R. Delaporte-Mathurin, A. Lanzrath

"Dress rehearsals" using existing neutron generator + facilities while molten salt lab under construction

100 mL FLiBe tests



1 L FLiBe test



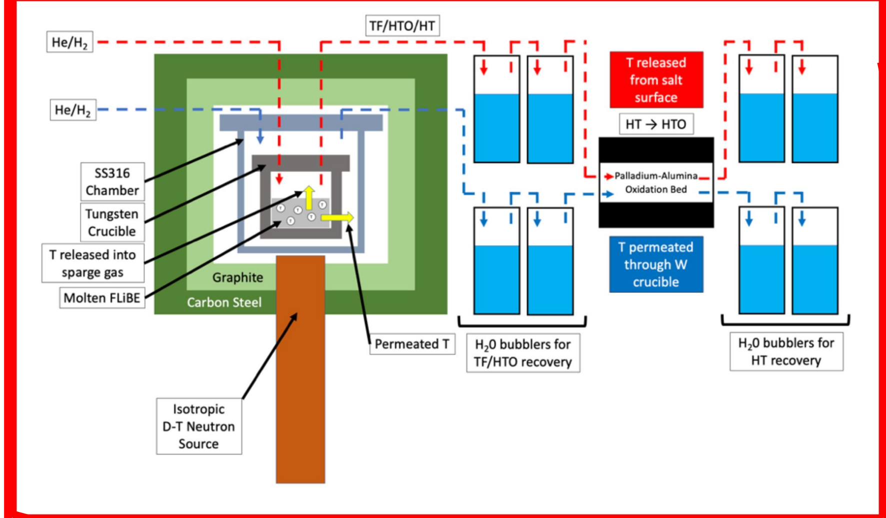
- Expectation: majority of T is picked up by **sparging** through the salt
- Sweep gas stream between crucible + outer wall picks up tritium that **permeates** through crucible

0D tritium transport model:

$$V\frac{\partial c}{\partial t} = S - A_p\ k_p\ c - A_r\ k_r\ c$$

V: salt volume
c: salt tritium concentration
S: tritium source
A_i: surface area
k_i: mass transport coefficient

MIT PSFC: intermediate tritium breeding tests before LIBRA



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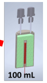
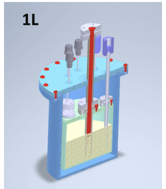
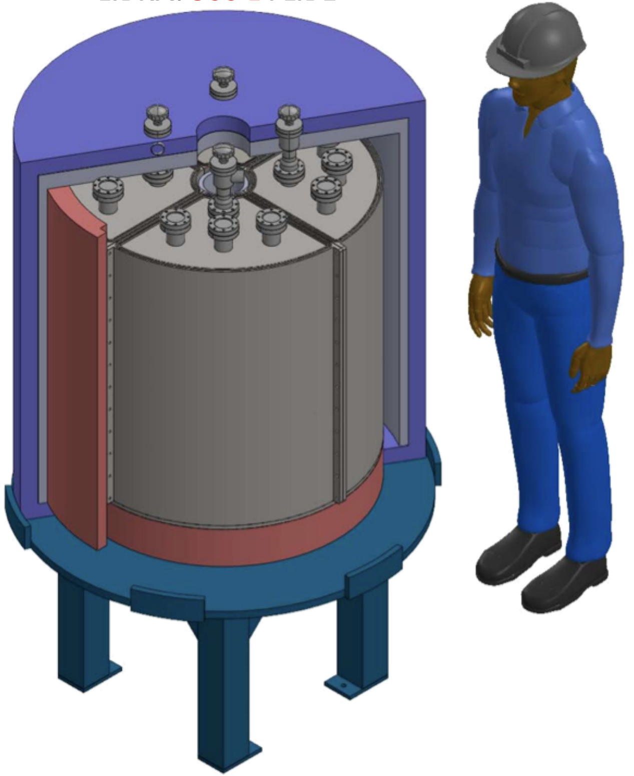
100 mL FLiBe tests



1 L FLiBe test



LIBRA: 500 L FLiBE

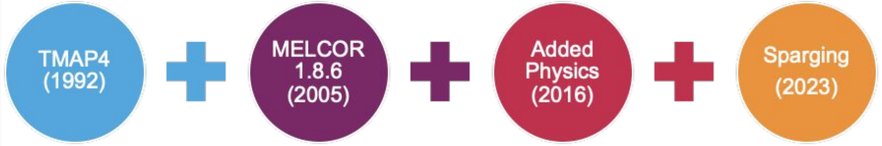


LIBRA contributions

Team: Tommy Fuerst, Adriaan Riet, Matt Eklund, Chase Taylor, Hanns Gietl

Sparging in MELCOR-TMAP

- MELCOR-TMAP is a thermal-hydraulic system level fusion safety code with tritium transport and FLiBe properties implemented.
- Current work: add sparging physics with SPARC-90.
- Goal: complete system-level analysis of LIBRA



TMAP4 (1992)
- Tritium migration
- Trapping, diffusion, solubility
- Surface rxns
- Multiple species tracking

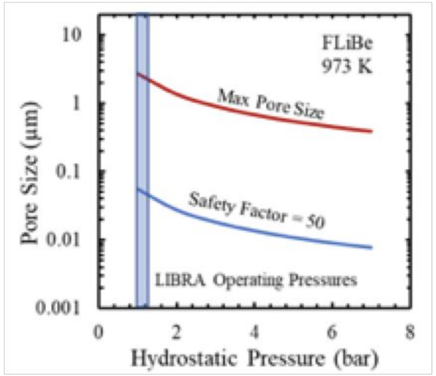
MELCOR 1.8.6 (2005)
- Heat transfer
- Thermal hydraulics
- Vapor & aerosol tracking
- Reactor accidents

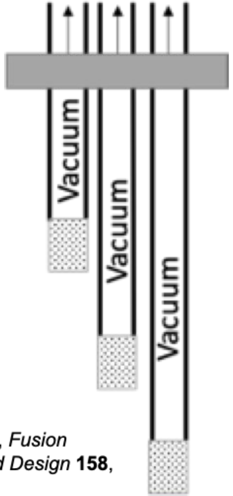
Added Physics (2016)
- HTO modeling
- Lithium fire accidents
- Be dust-steam oxidation
- Multiple working fluids (FLiBe)

Sparging (2023)
- SPARC-90 code
- Two-phase bubble flow
- Model tritium species and multiple working fluids.

In-Situ Gas Sensors

- Macroporous membrane gas-liquid contactors enable the measurement of gas evolution (including tritium species) from molten salts without wetting.[1]
- Formed out of low H permeability, reduced activation, and FLiBe compatible materials.
- Axial T detection in LIBRA salt



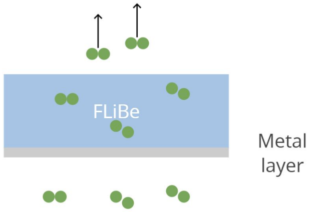


1.Tosti, S. *et al.*, *Fusion Engineering and Design* **158**, 111737 (2020).

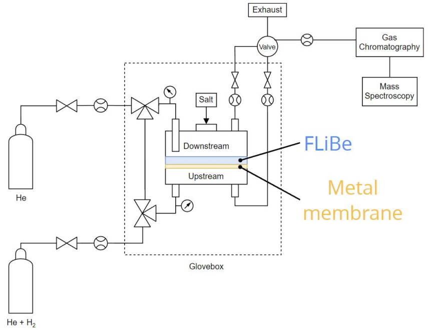
Commonwealth Fusion Systems: sponsoring salt chemistry research at PSFC

- Salt characterization important for accurately measuring and understanding hydrogen transport and corrosion
- Lots of variability in as-received salt; hard to compare results from different experiments in literature
- Electrochemical analysis - measuring impurities and seeing their effects
    - Supported by ICP-OES, ICP-MS, combustion/inert gas fusion analysis
- Hydrogen transport measurements - solubility and diffusivity of hydrogen in FLiBe
    - Done with a permeation rig
    - Understand how H/D will move through FLiBe in the blanket
- Corrosion testing - understand how the well-characterized salt corrodes fusion materials

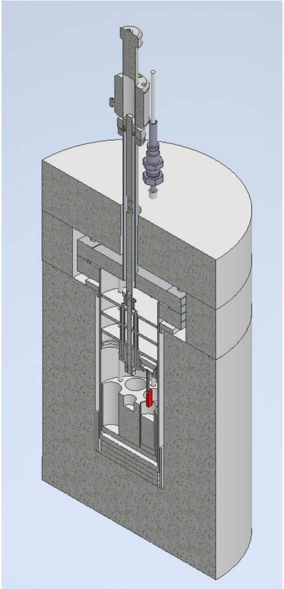
Permeation through metal membrane into FLiBe



Draft rig and gas system design



Planned crucible and furnace design



W. Zhou, J. Cota

Combine resources to tackle key engineering & scientific challenges

LIB example:
CFS: commercial developer of the LIB, sponsoring FLiBe research, carrying out own FLiBE research
Eni: funding support for postdocs
Kyoto Fusioneering: collaborations with PSFC on FLiBe compatibility
Kairos Power: fission company using FLiBe, collaborating with PSFC and MIT NRL

Private fusion startups
(+ private fission startups!)

Vendors

National labs
(US & international)

Universities
(US & international)

LIB example:
DOE: Sponsors of LIBRA through ARPA-E
INL: LIBRA partners
ORNL: FLiBe expertise + participation on LIBRA advisory board
SRNL: tritium expertise + participation on LIBRA advisory board
UKAEA: collaboration on liquid breeder research

LIB example:
AlphaTech, Materion: suppliers of FLiBe, BeF_2
+ **architects/contractors** who specialize in scientific laboratory construction
+ **general lab equipment suppliers**

- Neutron generators
- Glove boxes
- Ventilation systems
- Detectors

LIB example:
MIT PSFC: LIBRA lead
MIT NRL: FLiBe research for fission, collaborators with PSFC
Berkeley: salt chemistry research, participation on LIBRA advisory board
Clemson: T2M expertise, participation on LIBRA advisory board
Kyoto University (Japan): visiting PhD student exchange
Politecnico di Milano (Italy): visiting PhD student exchange

Remember: We've been using the FLiBe LIB as an example, but this same collaborative approach is going to apply to the development of all breeder concepts being explored for pilot plants.

Backup slides

Japan: FFHR2 STB blanket (Spectral-shifter and Tritium Breeder)

Sagara, Akio, et al. "Innovative liquid breeder blanket design activities in Japan." *Fusion science and technology* 47.3 (2005): 524-529.

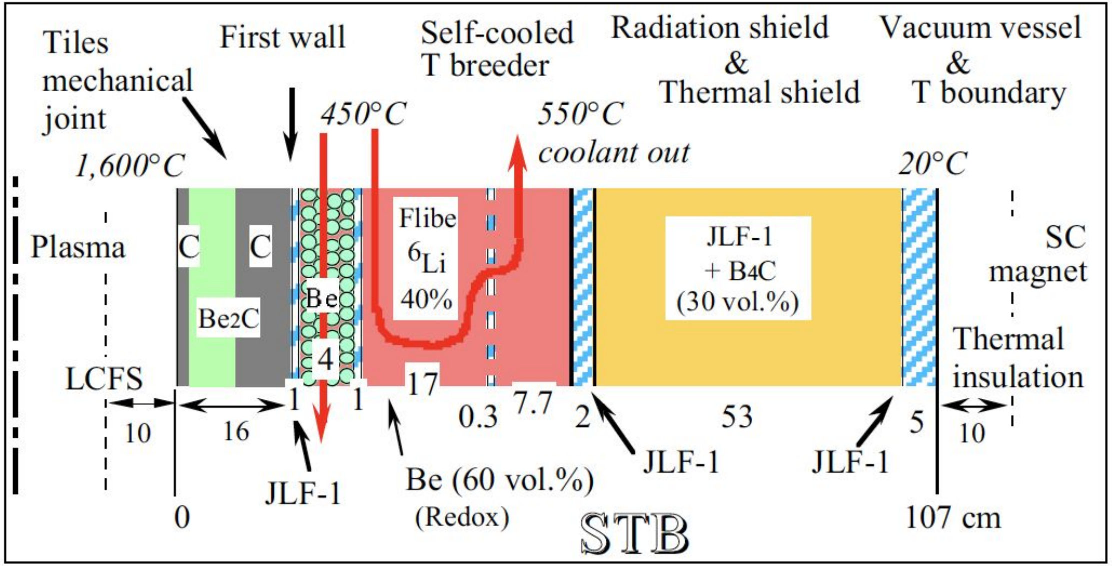


Fig.2 Radial build of STB Flibe blanket for FFHR2.

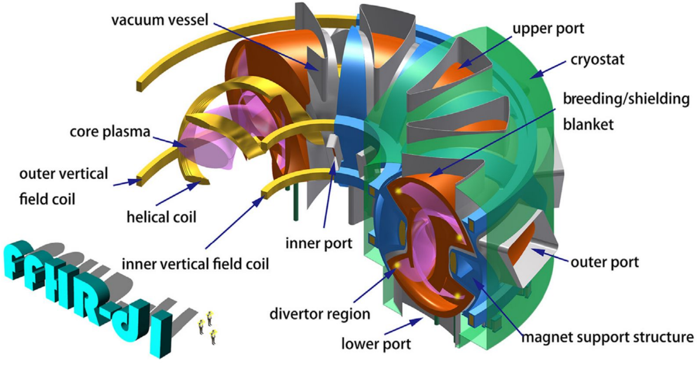
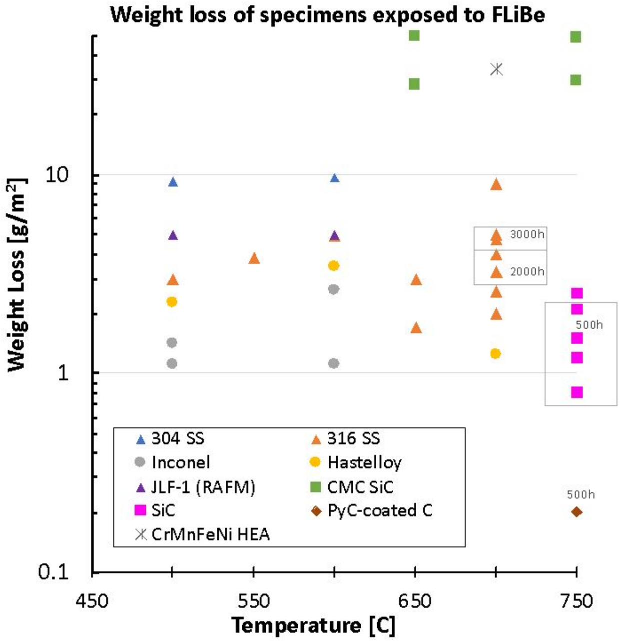


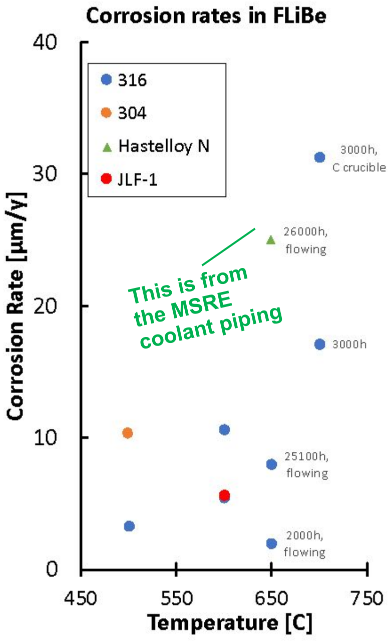
Fig. 2. 3D design of FFHR-d1, illustrating the whole size with human as a measure.

Sagara, Akio, et al. "Helical reactor design FFHR-d1 and c1 for steady-state DEMO." *Fusion Engineering and Design* 89.9-10 (2014): 2114-2120

Materials compatibility in FLiBe

These plots show the total of non-fueled FLiBe corrosion data available in the literature. (1000h static tests unless otherwise noted)





S.E.Ferry, K.B. Woller, et al., ""The LIBRA Experiment: Investigating robust tritium accountancy in molten FLiBe exposed to a D-T fusion neutron spectrum," *Fusion Science and Technology*, accepted 2022

New model for classifying molten salt corrosion, proposed by Weiyue Zhou

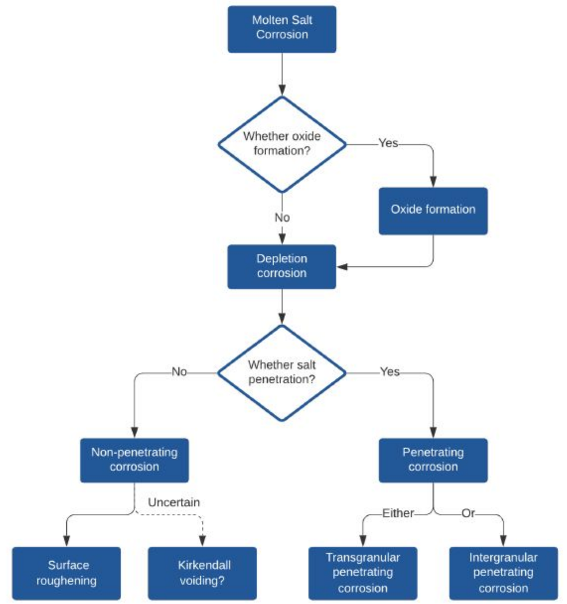


Figure 2-13: Proposed categorization of molten salt corrosion mechanisms

Zhou, Weiyue. *Influence of environmental conditions and proton irradiation on molten salt corrosion of metals*. Diss. Massachusetts Institute of Technology, 2021.

ed in these models

Parameter	Symbol	Value	Range	Units
Fusion power	P_{fus}	525	-	MW_{th}
Tritium burn rate	$\dot{N}_{\text{T,burn}}$	$8.99{\cdot}10^{-7}$	-	kg/s
Pulse duration	t_{pulse}	1800	1800 - 3600	s
Time between pulses	t_{off}	60	60-120	s
Tritium burn efficiency	TBE	0.02	0.005 – 0.1	-
Non-radioactive loss fraction	ϵ_i	10^{-4}	-	-
Availability factor	AF	0.7	0.5 - 1	-
Tritium processing time	t_p	4	1 – 12	h
Doubling time	t_d	2	-	y
Fraction of the system failing	q	0.25	-	-
Reserve time	t_{res}	24	0 - 48	h
Direct Internal Recycling fraction	f_{DIR}	0.3	0.1 – 0.9	-

Component	Symbol	Value	Range	Units
Breeding zone	τ_{BZ}	1.25	1 – 240	h
First wall	τ_{FW}	1000	-	s
Divertor	τ_{div}	1000	-	s
Tritium extraction system	τ_{TES}	24	1 - 240	h
Heat exchanger	τ_{HX}	1000	-	s
Vacuum pump	τ_{vp}	600	-	s
Fuel clean-up	τ_{fc}	0.3	0.1 - 1	h
Isotope separation system	τ_{ISS}	3.7	0.9 - 11	h
Detritiation system	τ_{det}	1	-	h

Parameter	Symbol	Mean value	Range	Units
Availability factor	AF	0.75	±0.25	-
Tritium burn efficiency	TBE	2.5	±2.5	%
DIR fraction	f_{DIR}	0.5	±0.5	-
Breeding zone residence time	τ_{BZ}	13	±11	h
TES efficiency	η_{TES}	0.7	±0.3	-
Tritium processing time	t_p	6.5	±5.5	h
Reserve time	t_{res}	24	±24	h
Doubling time	t_d	6	±5	y