

Molten Salt Reactors: Innovative Designs and Calculations of MSR Neutronics

joint IAEA-ICTP Workshop on

**Physics and Technology of Innovative Nuclear
Energy Systems for Sustainable Development**

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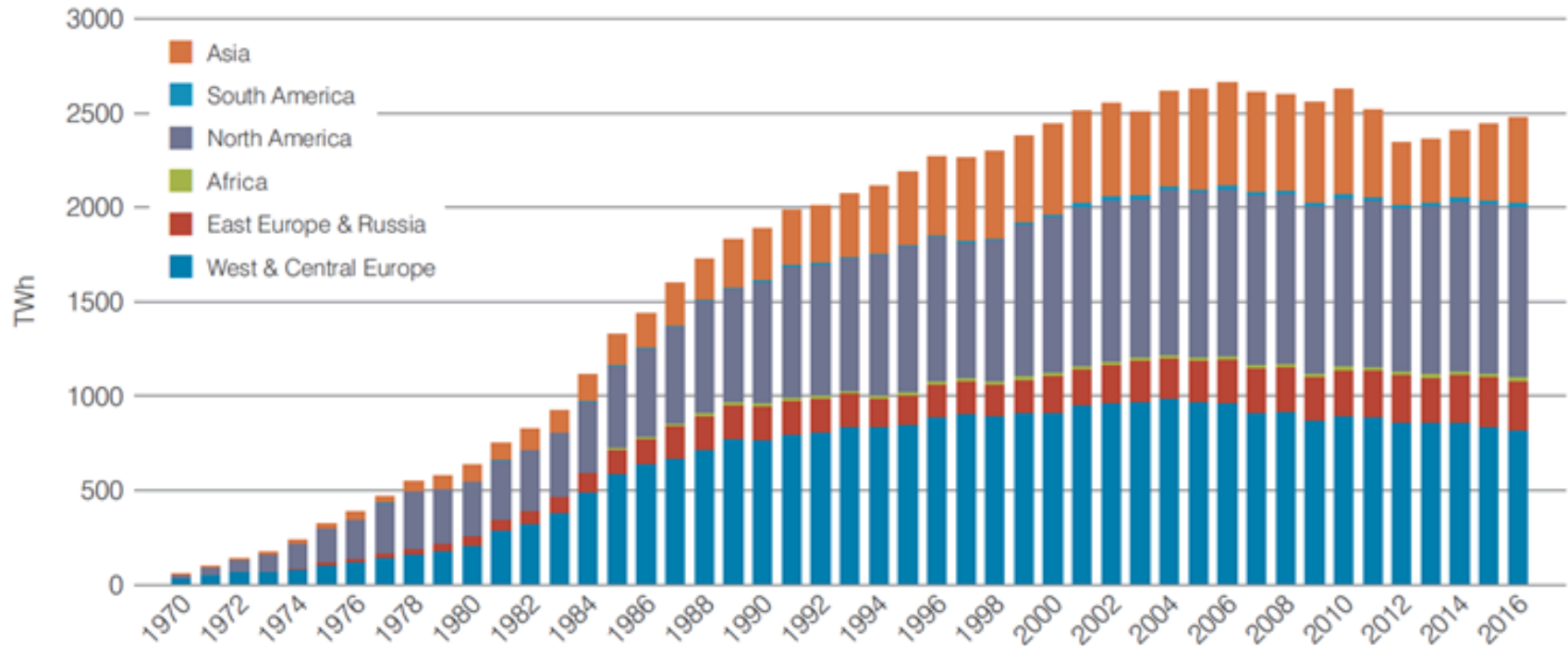
Outline

- The idea behind molten salt reactors
- History of molten salt reactors
- Introduction to (relevant) neutronics

Break?

- Neutronics of molten salt reactors
- Current designs of molten salt reactors

World Nuclear Energy Production

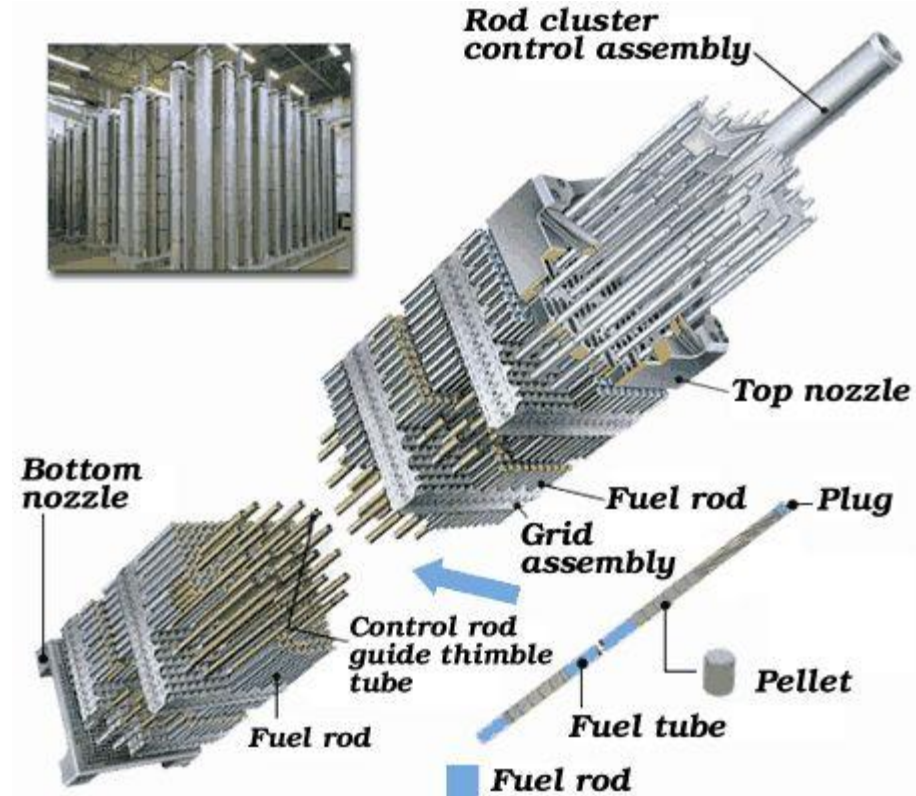


Source: World Nuclear Association, IAEA Power Reactor Information Service (PRIS)

- For the most part PWR, BWR
- Also new designs are mostly LWR with solid fuels.

Burnup distribution

- Fluxshape (power profile):
 - Axial ?
 - Radial ?
- Need to shape the flux
 - Graded enrichment
 - Control devices
 - (burnable absorbers)
 - Fuel shuffling between reloads:
 - Radially (PWR, BWR)
 - Axially (PHWR)
- Always uneven burn-up.



Liquid fuel

- Imagine you could use liquid fuel, flowing through the core:
 - Flux shape (power profile) would still be the same:
 - Axially: $\sim \sin\left(\frac{\pi}{H} z\right)$ H is height of cylinder
 - Radially: $\sim J_0\left(\frac{2.405 r}{R}\right)$ R is radius of cylinder
 - Burnup would be completely uniform!
(provided there is perfect mixing)
- Other immediate advantages:
 - No core-meltdown! (semantics, it's molten already...)
 - No fuel failure
 - Fission gases can be vented off.
 - Fuel is the coolant, no coolant needed (in primary circuit).



Choice of Liquid (Fluid) Fuel

- Salt



- Wikipedia: a salt is an ionic compound that can be formed by the neutralization reaction of an acid and a base. Salts are composed of related numbers of cations (positively charged ions) and anions (negative ions) so that the product is electrically neutral (without a net charge).
- Salts characteristically have **high melting points**.
- Long list of requirements for fuel:

Liquid Fuel Requirements

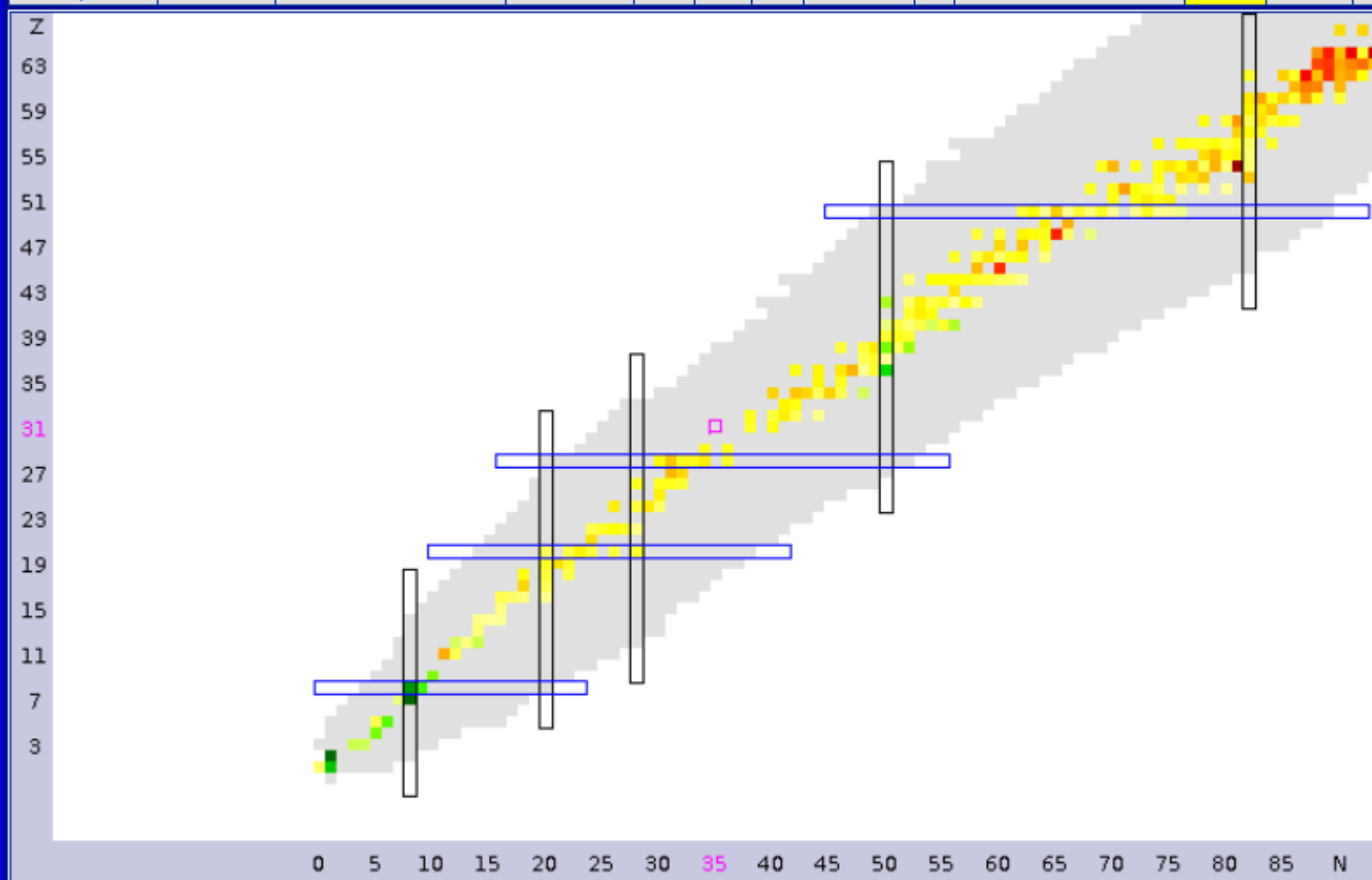
- Low capture x-sec for neutrons (*)
 - Stable against radiation (*)
 - Needs to be able to dissolve enough fissile/fertile material to achieve criticality (*)
 - Thermally stable (Eutectic)
 - Low vapor pressure
 - Good heat transfer
 - Non-aggressive to structural components
- (*) means relevant to neutronics


Choice of Liquid Fuel

- Only low-Z materials remain for neutronic reasons: Be, Bi, B-11, C, D, F, Li-7, N-15, O. (→ NNDC)
- Chemistry places additional requirements rejecting Bi, B-11, C, D, N-15, O;
- We are left with: F, Li-7, Be, commonly referred to as **Flibe**.
- Beryllium also acts as a neutron-doubler:

$${}^9_4\text{Be} + n \rightarrow 2{}^2_2\text{He} + 2n$$
- Also high elastic cross section → good moderator.
- But beryllium is poisonous.
- Other elements such as Zr, Na, K are sometimes added for different purposes.

Color code	Half-life	Decay Mode	Q_{β^-}	Q_{EC}	Q_{β^+}	S_n	S_p	Q_{α}	S_{2n}	S_{2p}	$Q_{2\beta^-}$	Q_{2EC}	Q_{ECp}
$Q_{\beta-n}$	BE/A	(BE-LDM Fit)/A	$E_{1st\ ex. st.}$	E_{2+}	E_{3-}	E_{4+}	E_{4+}/E_{2+}	β_2	$B(E2)_{42}/B(E2)_{20}$	$\sigma(n,\gamma)$	$\sigma(n,F)$	235U FY	239Pu FY





Zoom	Uncertainty
1	NDS
2	Standard
3	Screen Size
4	Narrow
5	Wide
6	Narrow
7	Wide

Nucleus

barns

1.01E+7	1.56E+1
2.66E+6	4.09
6.98E+5	1.07
1.83E+5	2.81E-1
4.80E+4	7.38E-2
1.26E+4	1.93E-2
3.30E+3	5.07E-3
8.66E+2	1.33E-3
2.27E+2	3.49E-4
5.95E+1	9.15E-5
1.56E+1	2.40E-5
unknown	

Ground and isomeric state information for $^{66}_{31}\text{Ga}$

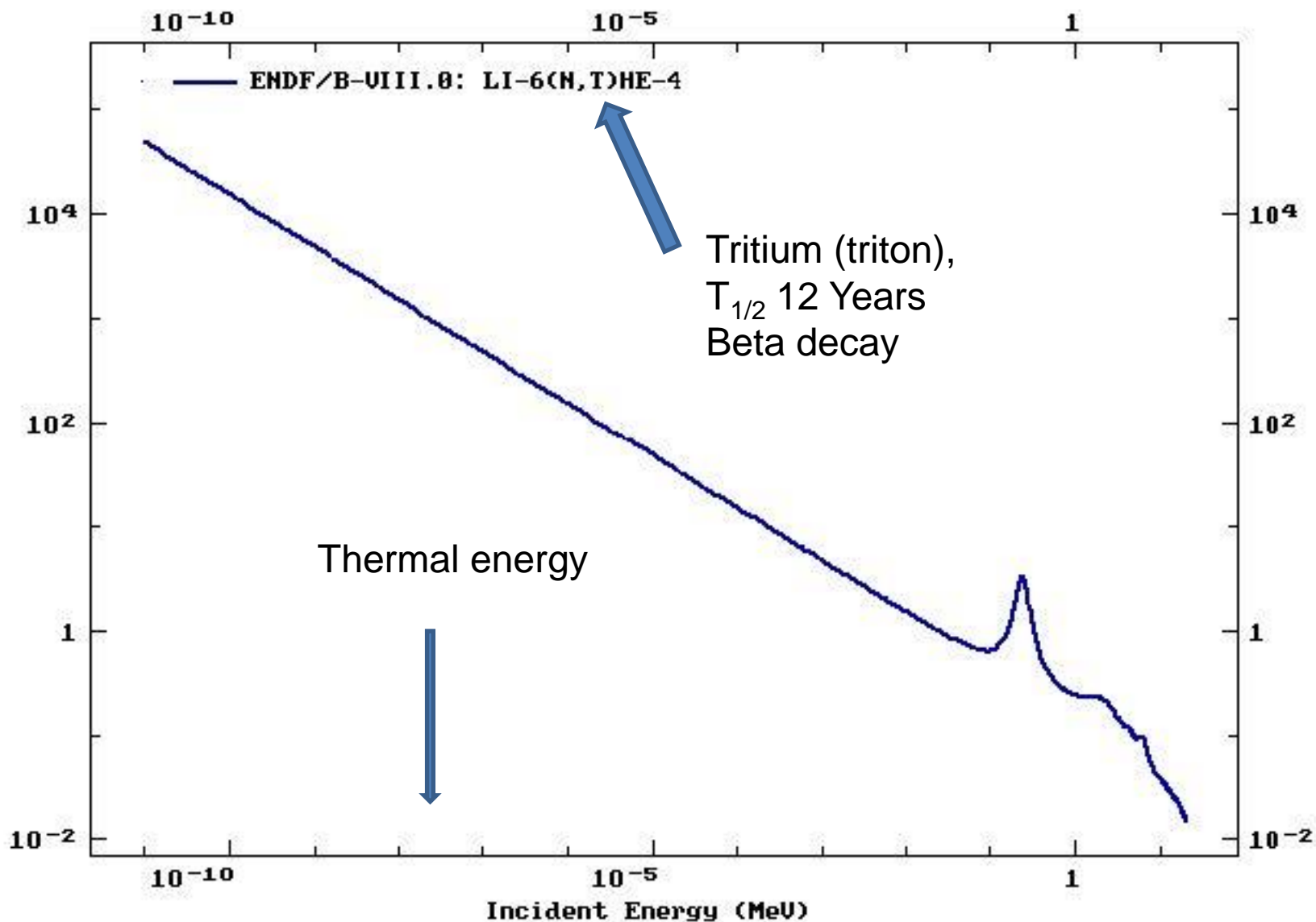
E(level) (MeV)	$J\pi$	$\Delta(\text{MeV})$	$T_{1/2}$	Decay Modes	$\sigma(n,\gamma)$ (b)
0.0	0+	-63.72366015625	9.49 h 3	ϵ : 100.00 %	

Search options:

Levels and Gammas
Nuclear Wallet Cards
Decay Data

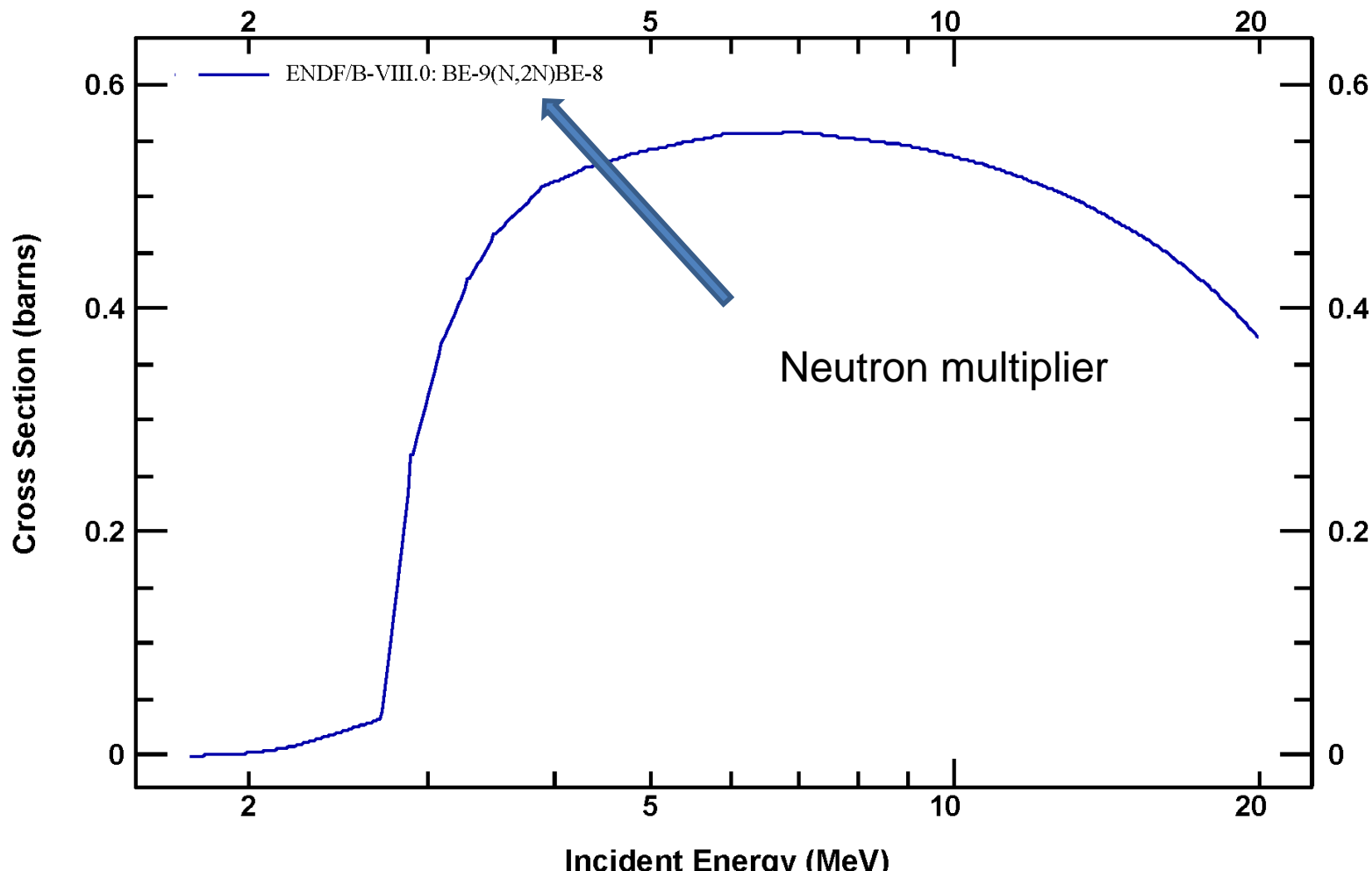
Li-6 Cross Section

ENDF Request 15670, 2018-Jul-31, 19:08:26



Be-9 Cross Section

ENDF Request 923, 2018-Aug-01, 13:41:55



Fuel Salt

- Nuclear fuel is U, Pu, Th.
(fissile, fissionable and fertile)
- Included in the salt as fluorides:
 - UF_4 , not to be confused with UF_6 , used in uranium enrichment process.
 - Uranium is enriched (typically 20%, LEU)
 - ThF_4 , breeding material,
 - either in fuel or blanket.
 - PuF_3
- Typical salt would be (MSRE):
 - 65% 7LiF – 29.1% BeF_2 – 5% ZrF_4 – 0.9% UF_4
 - With 35% enriched uranium

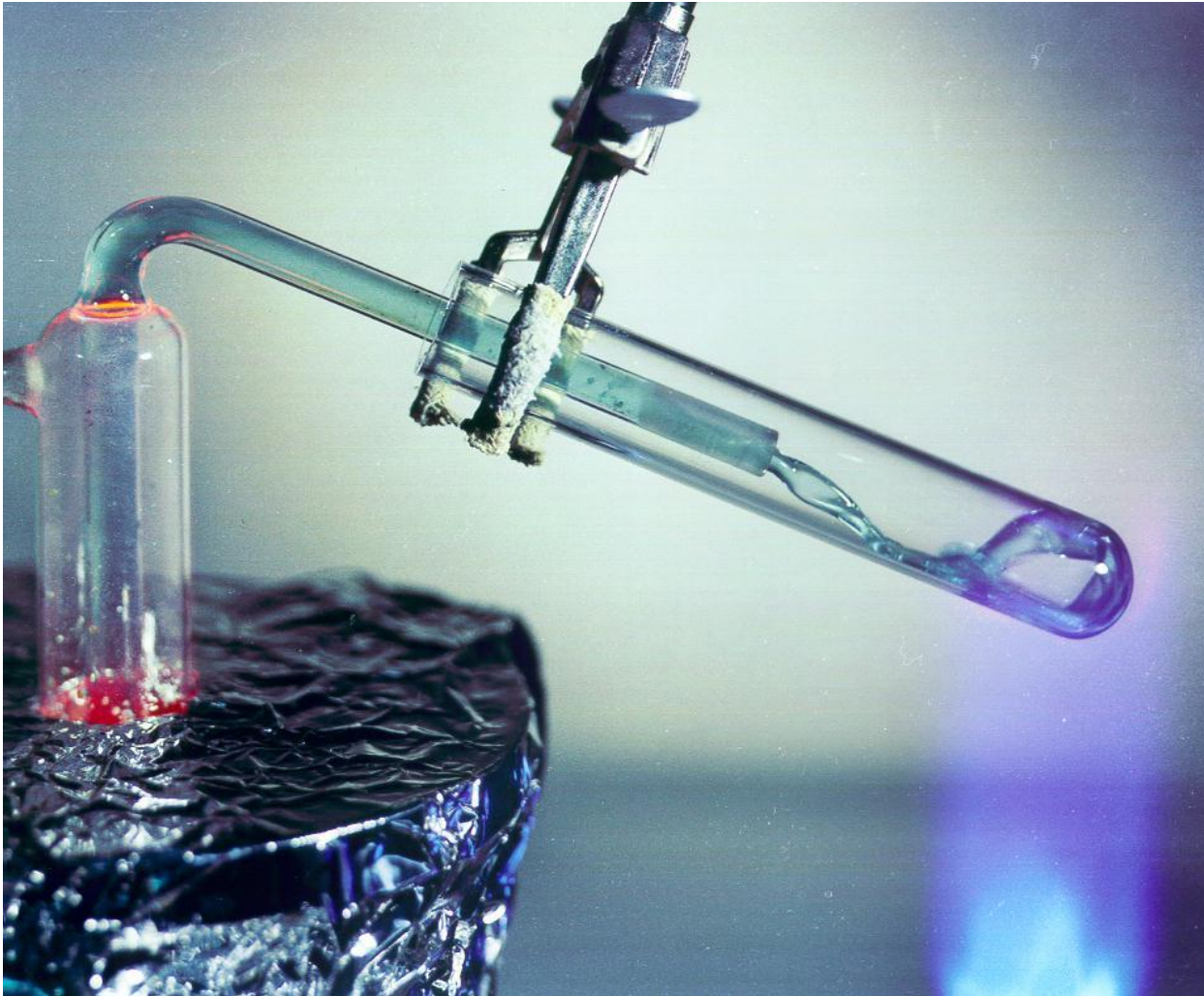
Fuel Salt Properties

MSRE Fuel

Property	H ₂ O	Na	Li	⁷ LiF-BeF ₂ -ZrF ₄ -UF ₄ 65-29.1-5.0-0.9
Melting point (°C)	0	98	181	434
Boiling point (°C)	100	880	1342	1435
Density (kg/m ³) (*)	712	830	483	2300
Thermal conductivity (W/K/m) (*)	0.54	67	53	1.43
Specific heat capacity (J/g/K) (*)	5.7	1.26	4.23	2.0
Viscosity (10 ⁻⁶ Pa s) (*)	89	250	360	8050

(*) typical reactor conditions

Flibe



Strong Point of MSR

- Inherent safety:
 - No meltdown;
 - Negative power coefficient (*);
 - Dump tank with freeze plug;
- Fission products can be removed easily.
- Fission products form stable fluorides.
- Operation is at low pressure.
- Xe can be skimmed off. (*)
- Fuel can be added at will. (*)
- No water or sodium present, less risk of steam explosions or hydrogen production.

History

- MSR's were pioneered at Oak Ridge National Labs, Tennessee in the 1940's
- First experiments were Aircraft Reactor Experiments:



Aircraft Reactor Experiment

- Operated for 9 days in 1954 (ORNL)
 - Salt: 53% NaF – 41% ZrF₄ – 6% UF₄ (HEU 93.4%)
 - Moderator: BeO, Temperature: 860 °C
 - Power: 2.5 MWth

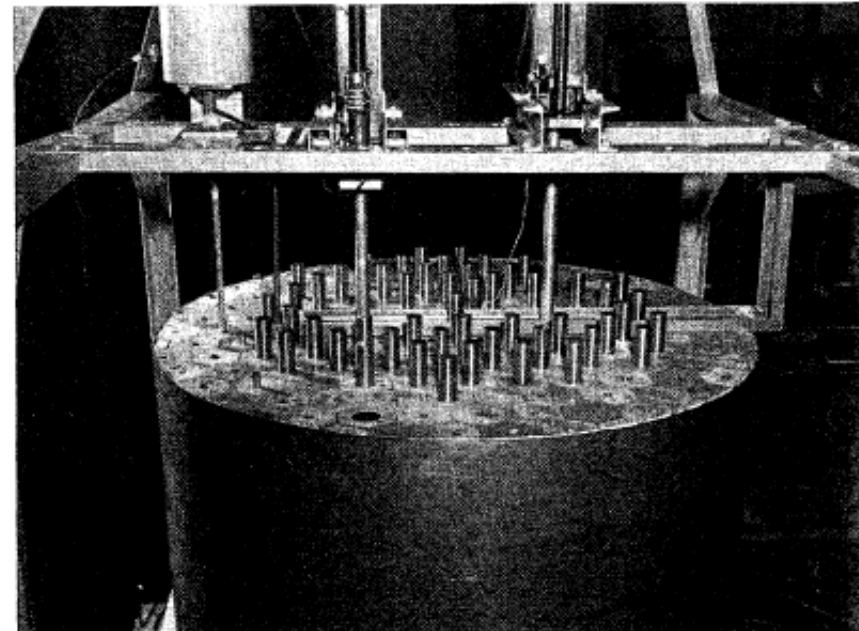
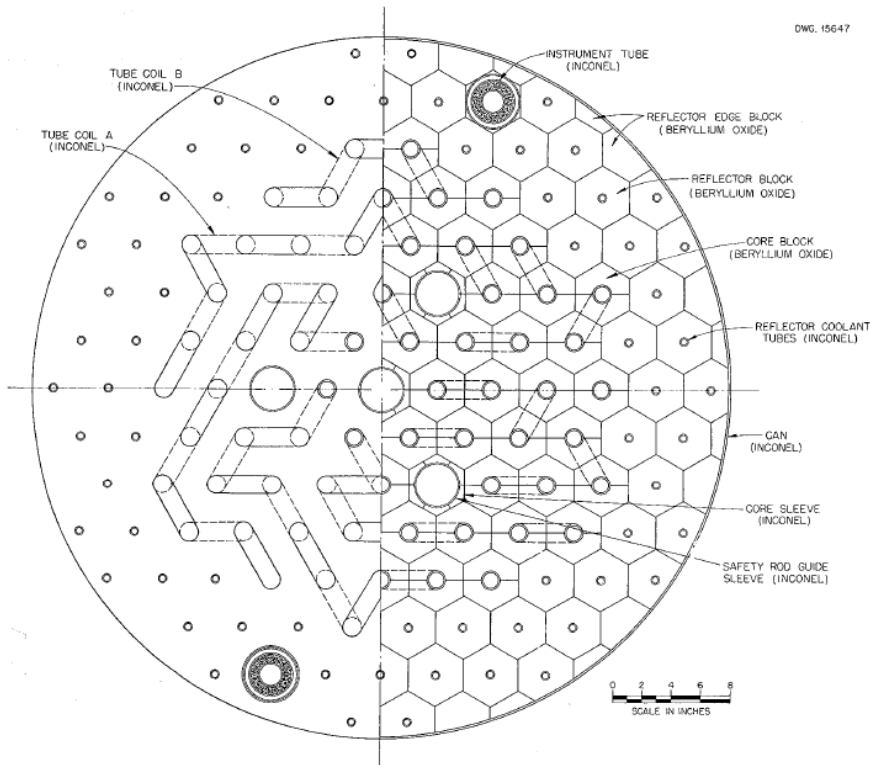


FIGURE 1: Critical Assembly of ARE

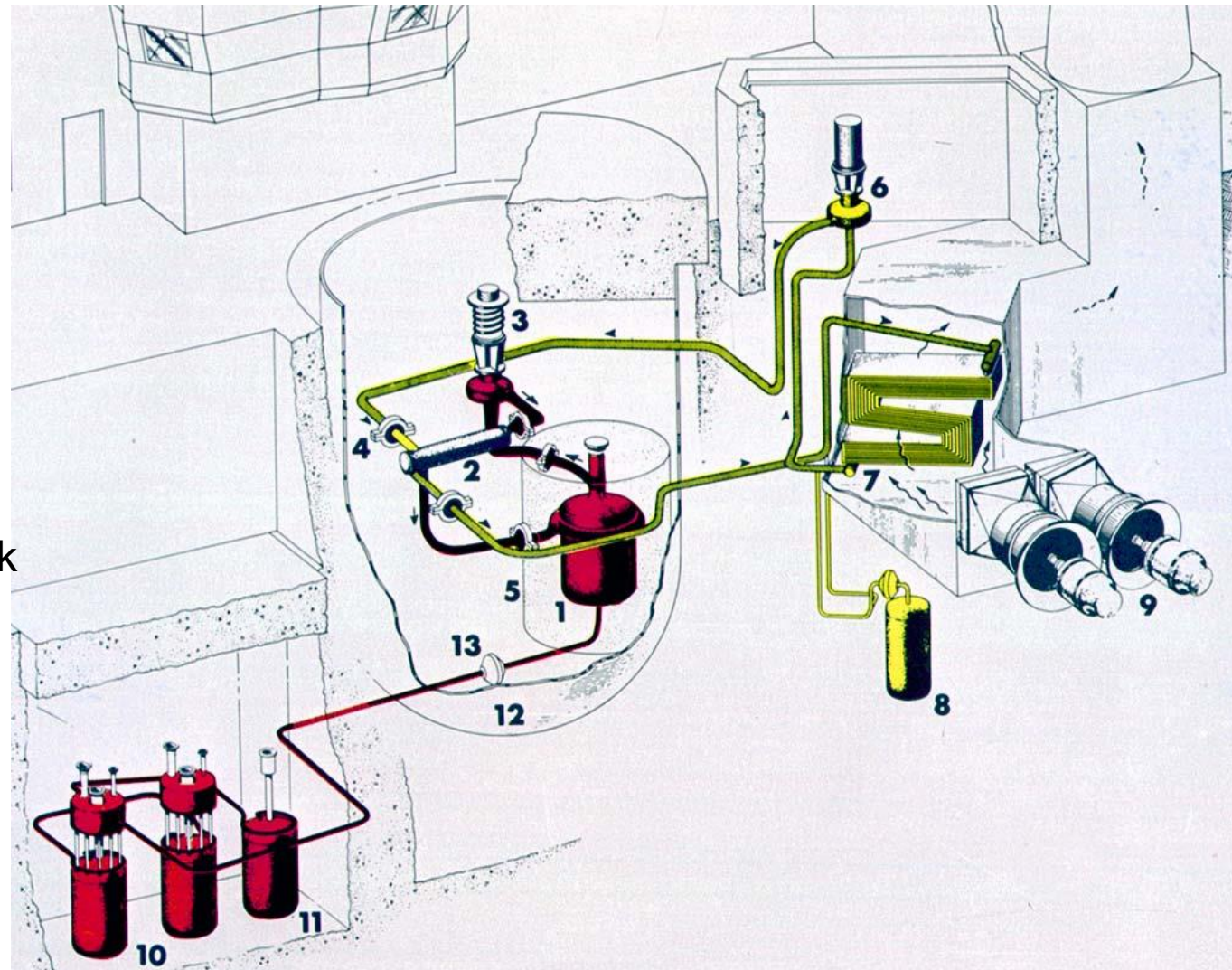
Molten Salt Reactor Experiment

- Operated from 1965 – 1969 (ORNL)
 - Salt: ${}^7\text{LiF}$ - BeF_2 - ZrF_4 - UF_4 (65- 29.1- 5 - 0.9)
 - 33% Enrichment. (${}^{233}\text{U}$ and ${}^{239}\text{Pu}$ also used)
 - Secondary circuit: LiF-BeF_2 (66–34 mole %)
 - Power 8 MWth, Temperature: 650 °C
 - Operated 9005 fph with U-235
 - Operated 4157 fph with U-233

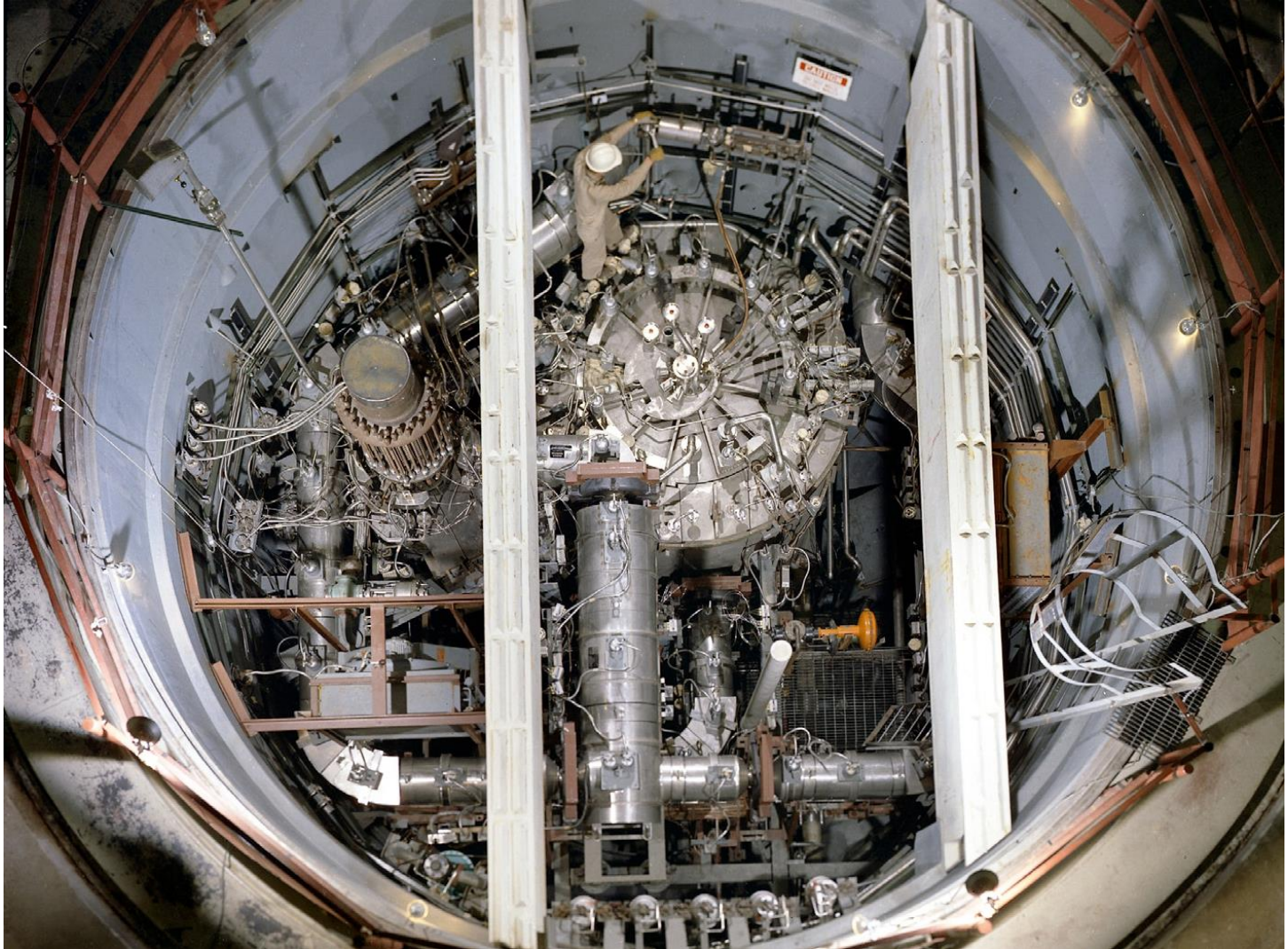
- It was a successful proof of concept

MSRE

- 1 Reactor vessel
- 2 Heat exchanger
- 3 Fuel pump
- 4 Freeze flange
- 5 Thermal shield
- 6 Coolant pump
- 7 Radiator
- 8 Coolant drain tank
- 9 Fans
- 10 Fuel drain tank
- 11 Flush tank
- 12 Containment
- 13 Freeze valve



MSRE



Summary of ORNL Experiments

Parameter	Aircraft Reactor Experiment (ARE)	Molten Salt Reactor Experiment (MSRE)
Date of operation	1954	1965-1970
Max. Power (MWth)	2.5	8.0
Max. Temperature (°C)	860	650
Moderator	BeO (solid)	Graphite (solid)
Fuel-Salt composition (%mol)	NaF-ZrF ₄ -UF ₄ (53-41-6)	⁷ LiF-BeF ₂ -ZrF ₄ -UF ₄ (65-29.1-5-0.9)
Secondary loop	Na	⁷ LiF-BeF ₂

Basic Equations of Neutronics

- Neutron balance equation: (one group)

$$\frac{\partial n(\vec{r}, t)}{\partial t} = D\nabla^2\Phi(\vec{r}, t) + \nu\Sigma_f\Phi(\vec{r}, t) - \Sigma_a\Phi(\vec{r}, t) \quad (1)$$

- In equilibrium:

$$-D\nabla^2\Phi(\vec{r}) + \Sigma_a\Phi(\vec{r}) = \nu\Sigma_f\Phi(\vec{r}) \quad (2)$$

- Introduce multiplication factor k_{eff}

$$-D\nabla^2\Phi(\vec{r}) + \Sigma_a\Phi(\vec{r}) = \frac{1}{k_{\text{eff}}}\nu\Sigma_f\Phi(\vec{r}) \quad (3)$$

Example of flux Distribution

ARE:
Axial

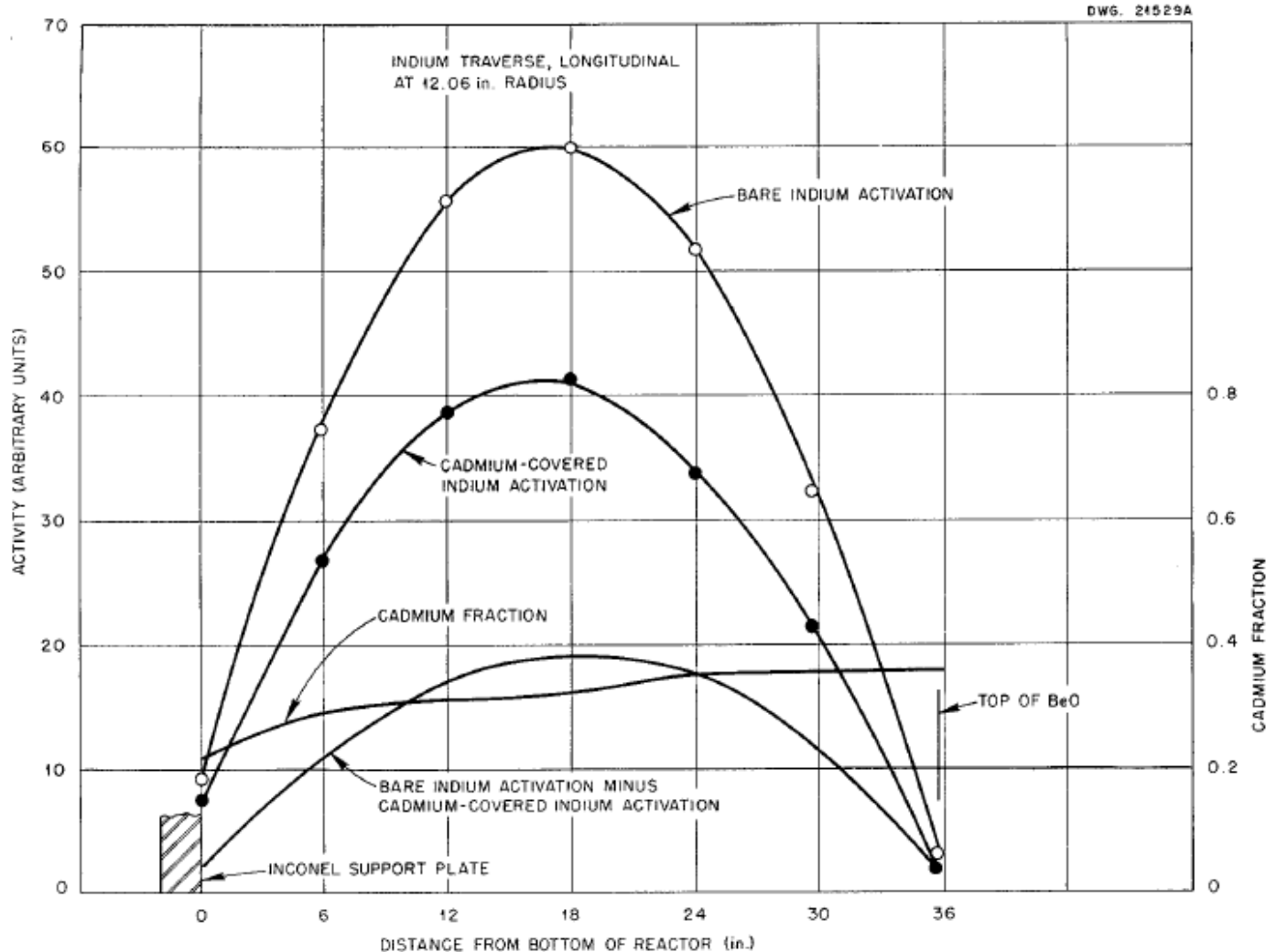


Fig. G.1. Longitudinal Neutron Flux Distribution.

ARE Radial

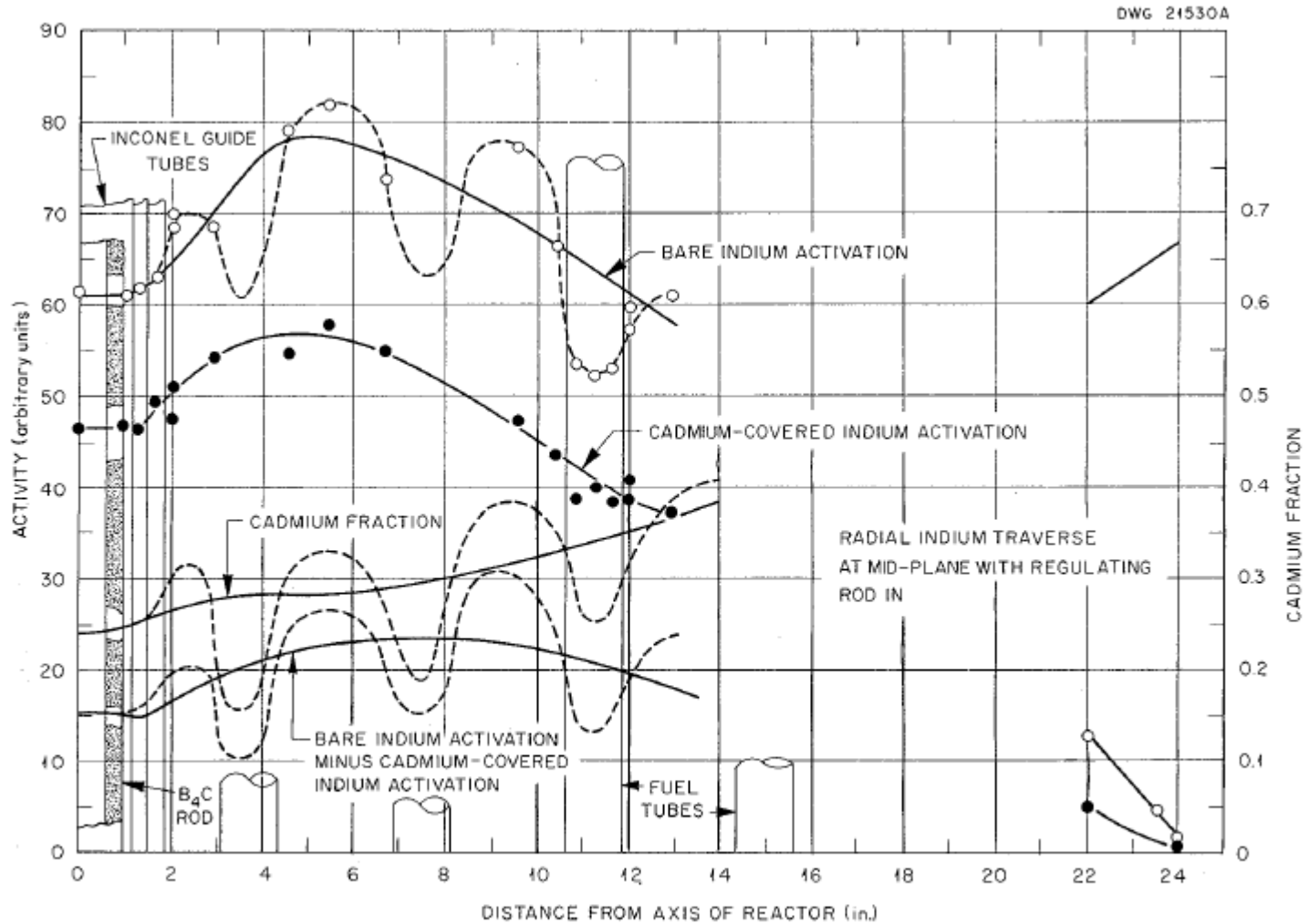
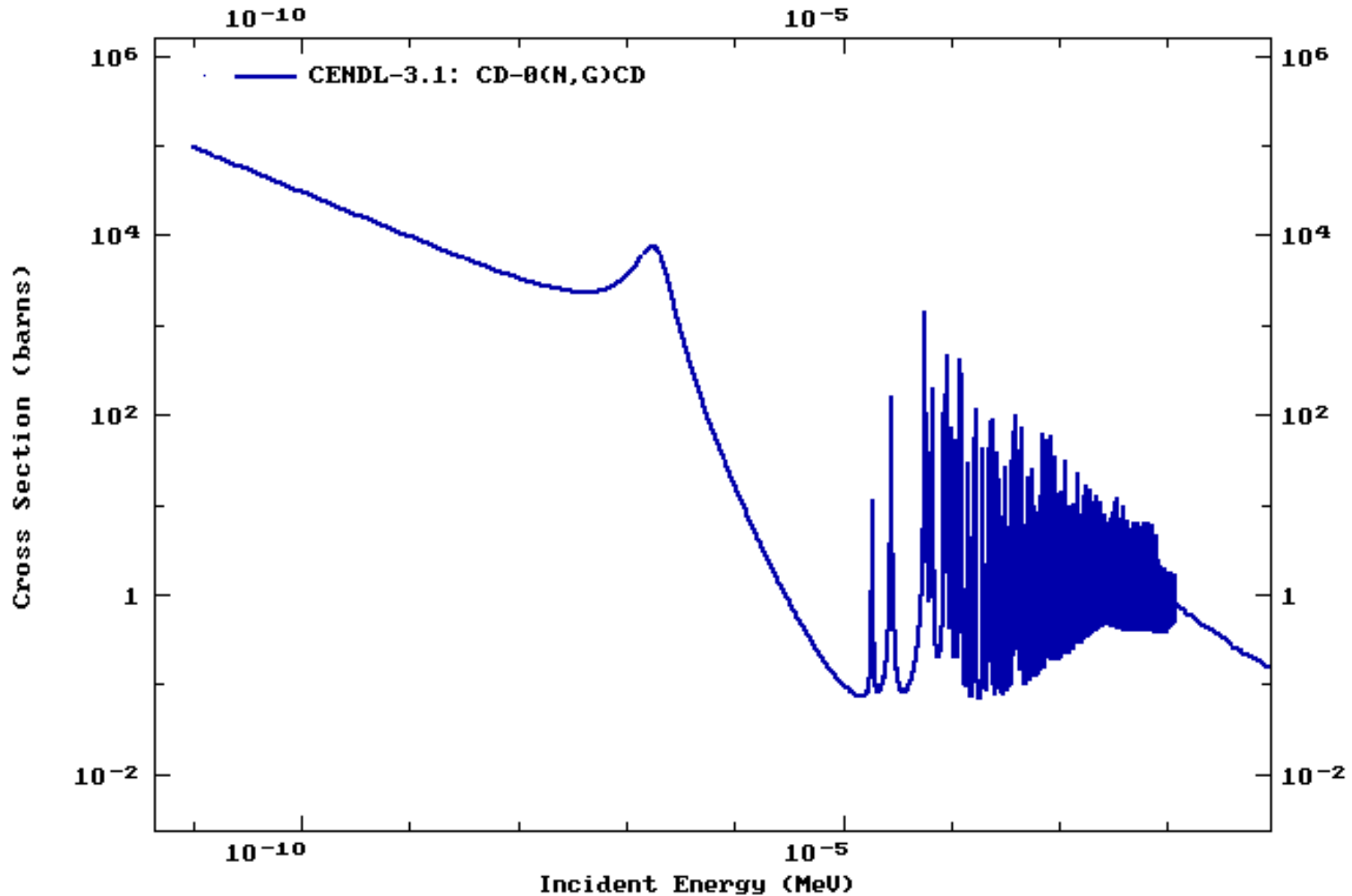


Fig. G.2. Radial Neutron Flux Distribution.

Cd Absorption X-sec

ENDF Request 16725, 2018-Aug-22, 16:44:58



Point kinetics

Assume the flux distribution does not change, only the amplitude: **point kinetics**

Define average neutron **generation time:**

$$\Lambda = \frac{\text{neutron population}}{\text{production rate}}$$

And **reactivity**

$$\rho = \frac{\text{production rate} - \text{loss rate}}{\text{production rate}} = 1 - \frac{1}{k_{\text{eff}}}$$

Point Kinetics

Now
$$\frac{dn(t)}{dt} = \frac{\rho}{\Lambda} n(t)$$

With obvious solution

$$n(t) = n(0)e^{\frac{\rho}{\Lambda}t}$$

All of this only considers neutrons from fission.

Fortunately, there are **delayed neutrons**.

Delayed Neutrons

- Fission products are always
 - Radioactive
 - South of the line of stability (too many neutrons)
- Decay towards line of stability by β -decay (electron), followed possibly by emission of a neutron.
- β -decay is slow: ms, s, min, \rightarrow ...
- Emitters are called **precursors**
- Emitted neutrons are **delayed neutrons.**

DN distribution

Table 1 Delayed-neutron data for thermal fission in ^{235}U ([Rose1991])

Group	Decay Constant, λ_k (s^{-1})	Delayed Yield, ν_{dk} (n/fiss.)	Delayed Fraction, β_k
1	0.01334	0.000585	0.000240
2	0.03274	0.003018	0.001238
3	0.1208	0.002881	0.001182
4	0.3028	0.006459	0.002651
5	0.8495	0.002648	0.001087
6	2.853	0.001109	0.000455
Total	-	0.016700	0.006854

$\beta = \sum_{k=1}^6 \beta_k$ is a crucial parameter in a reactor

Point Kinetics with DN

- Interesting thought: every neutron in a reactor is in a chain that originated in a delayed neutron precursor.
- With DN, the point kinetics equation becomes

$$\frac{dn(t)}{dt} = \frac{\rho - \beta}{\Lambda} n(t) + \lambda C(t)$$

with $C(t)$ the average precursor concentration.

Precursor Concentration

- Precursors originate in fission, then decay:

$$\frac{dC(t)}{dt} = \frac{\beta}{\Lambda} n(t) - \lambda C(t)$$

- Taking the six precursor groups:

$$\frac{dC_1(t)}{dt} = \frac{\beta_1}{\Lambda} n(t) - \lambda_1 C_1(t)$$

⋮

$$\frac{dC_6(t)}{dt} = \frac{\beta_6}{\Lambda} n(t) - \lambda_6 C_6(t)$$

$$\frac{dn(t)}{dt} = \frac{\rho - \beta}{\Lambda} n(t) + \sum_{k=1}^6 \lambda_k C_k(t)$$

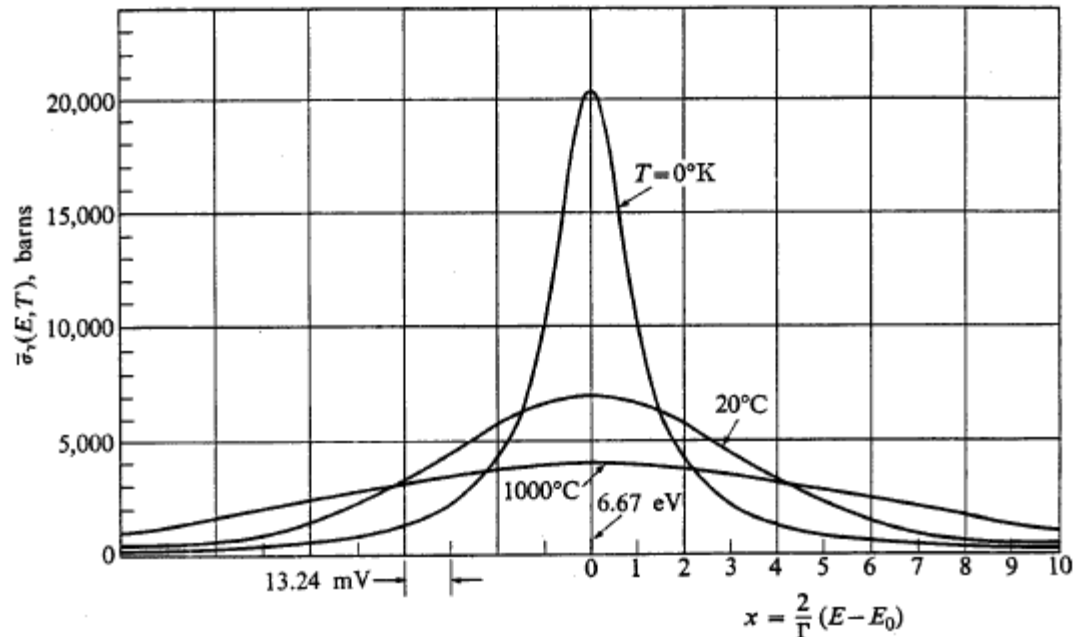
Doppler Broadening

- Remember definition of reactivity:

$$\rho = \frac{\text{production rate} - \text{loss rate}}{\text{production rate}}$$

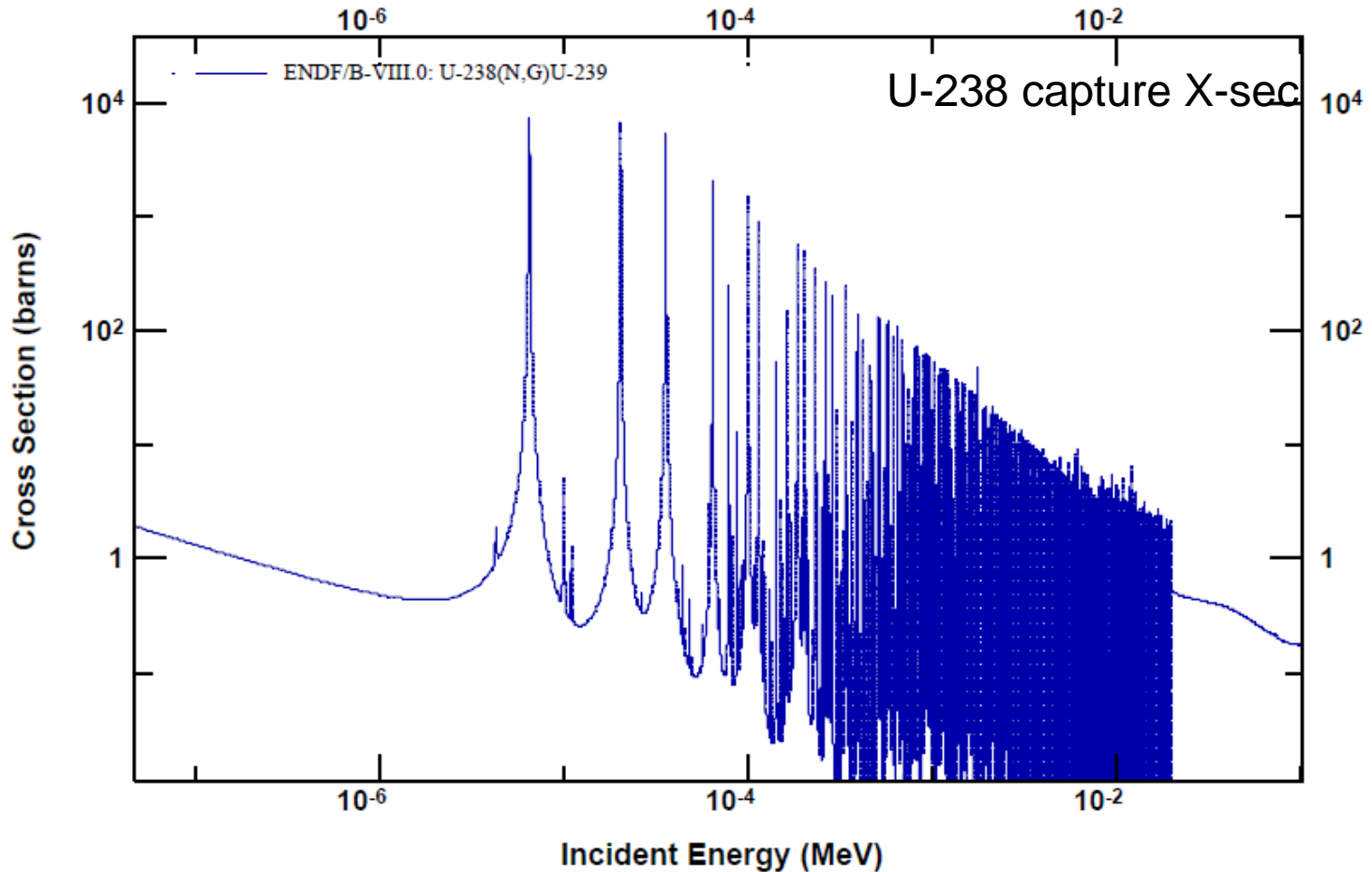
If loss rate increases, reactivity decreases.

- Doppler broadening:
- Leads to higher absorption.



Absorption

ENDF Request 1046, 2018-Aug-05,12:15:09



Reactivity Coefficient

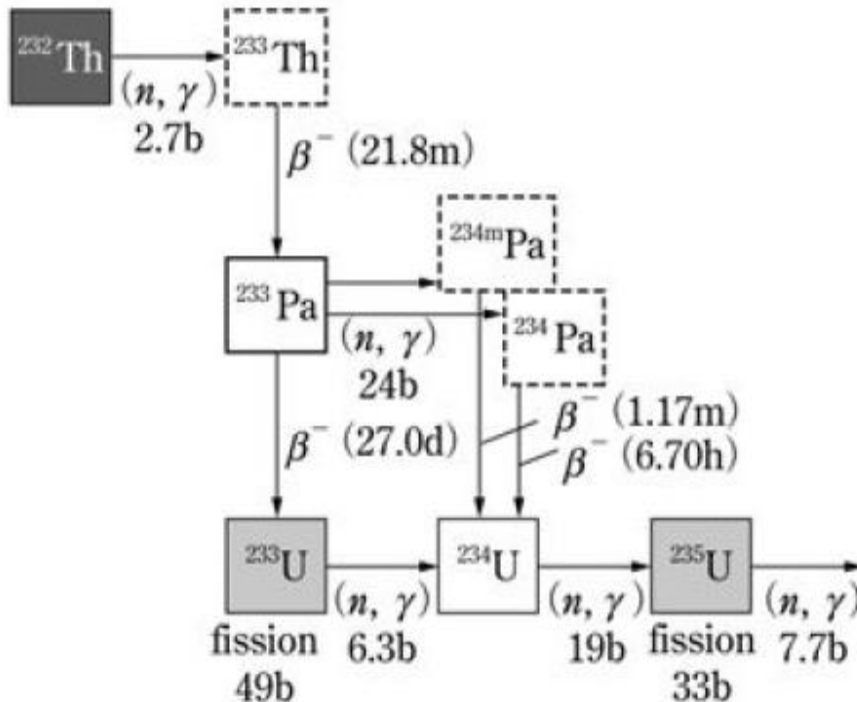
- Reactivity coefficient is defined as

$$\alpha_x = \frac{\partial \rho}{\partial x}$$

- So $[\alpha_x] = 1/[x]$, e.g. $\alpha_T = \frac{\partial \rho}{\partial T} = \dots \text{mk/K}$
- Lots of coefficients:
 - Temperature (fuel, coolant, moderator)
 - Density (coolant, moderator)
 - Poison
- Some provide feedback (positive, negative)

A Word about Breeding

Burnup Chain from Thorium



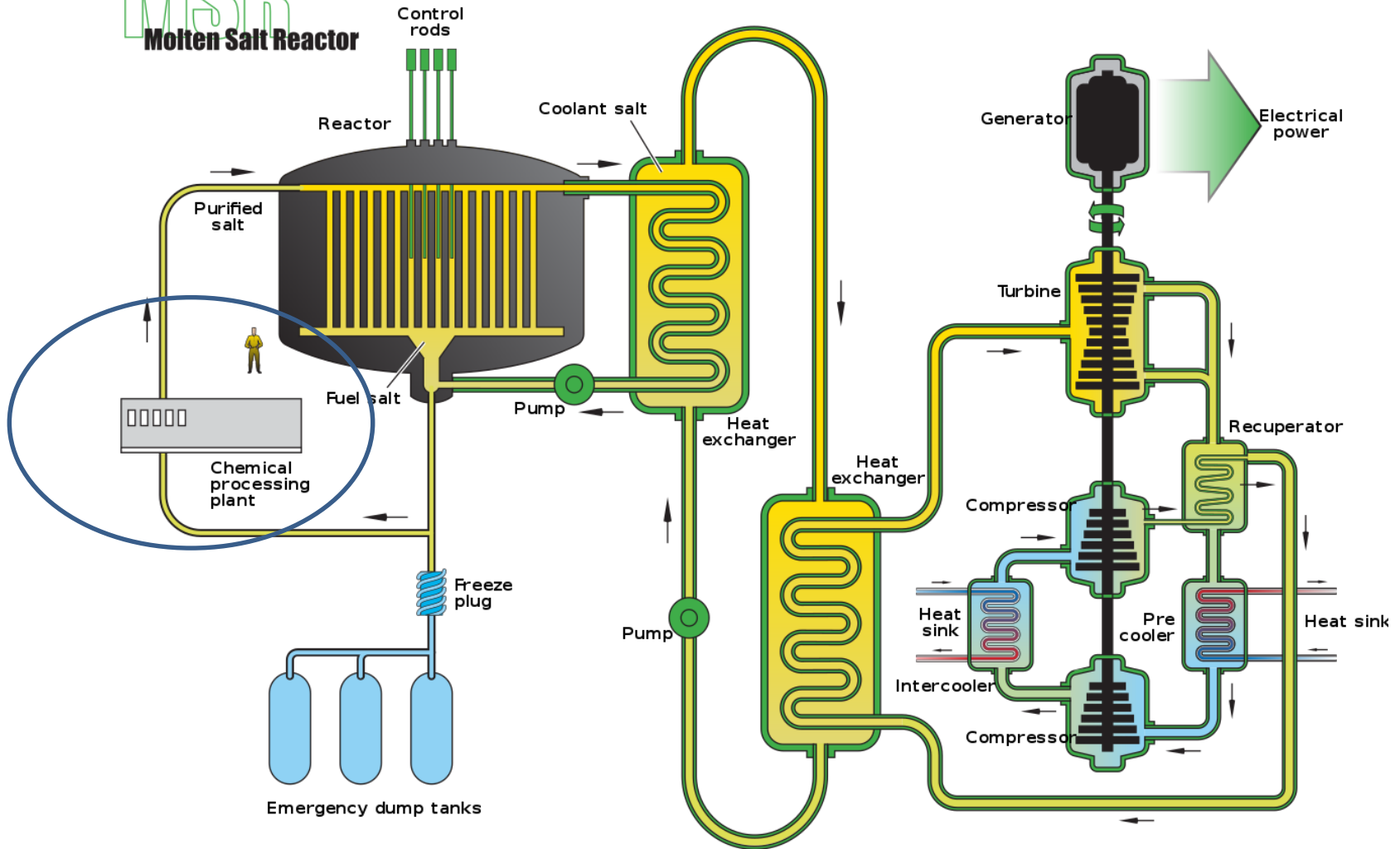
- Fertile
- Fissile
- Nuclides decaying in a very short time relative to fuel burnup period

$\longrightarrow (n, \gamma)$ } Typical one-group
 $\longleftarrow (n, 2n)$ } cross-sections in PWR
 Fission } appended¹⁾ (unit : barn)

$\downarrow \beta^-$ decay (half-life)²⁾

Apply to MSR

MSR
Molten Salt Reactor



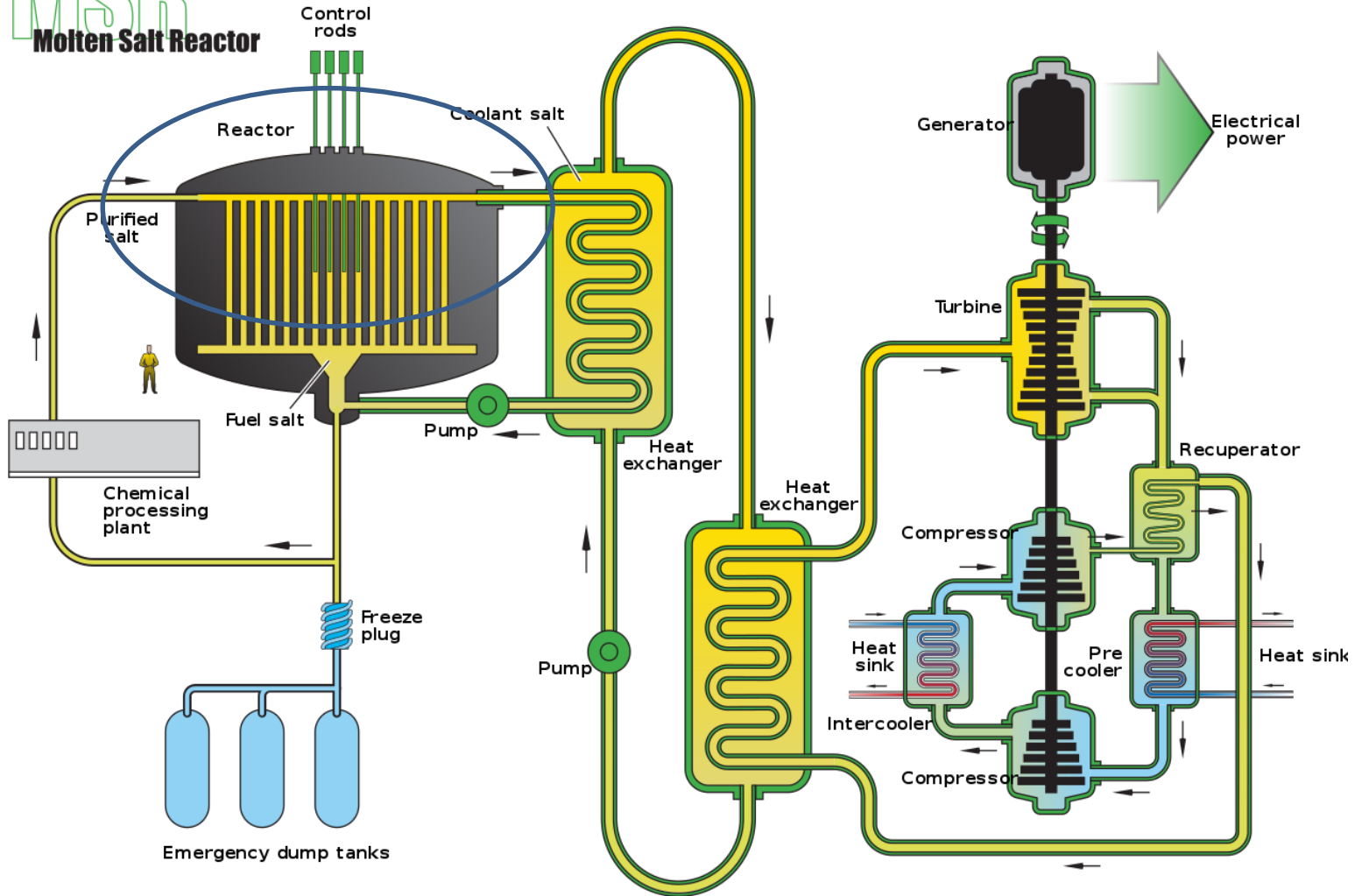
Chemical Processing Plant

- Remove fission products
 - One of the main design features of original ORNL design.
 - In thorium operation, remove protactinium-233 to let it decay to U-233, avoiding the n-capture.
 - Topping up the fuel, to compensate for burnup.

Apply to MSR

MSR

Molten Salt Reactor



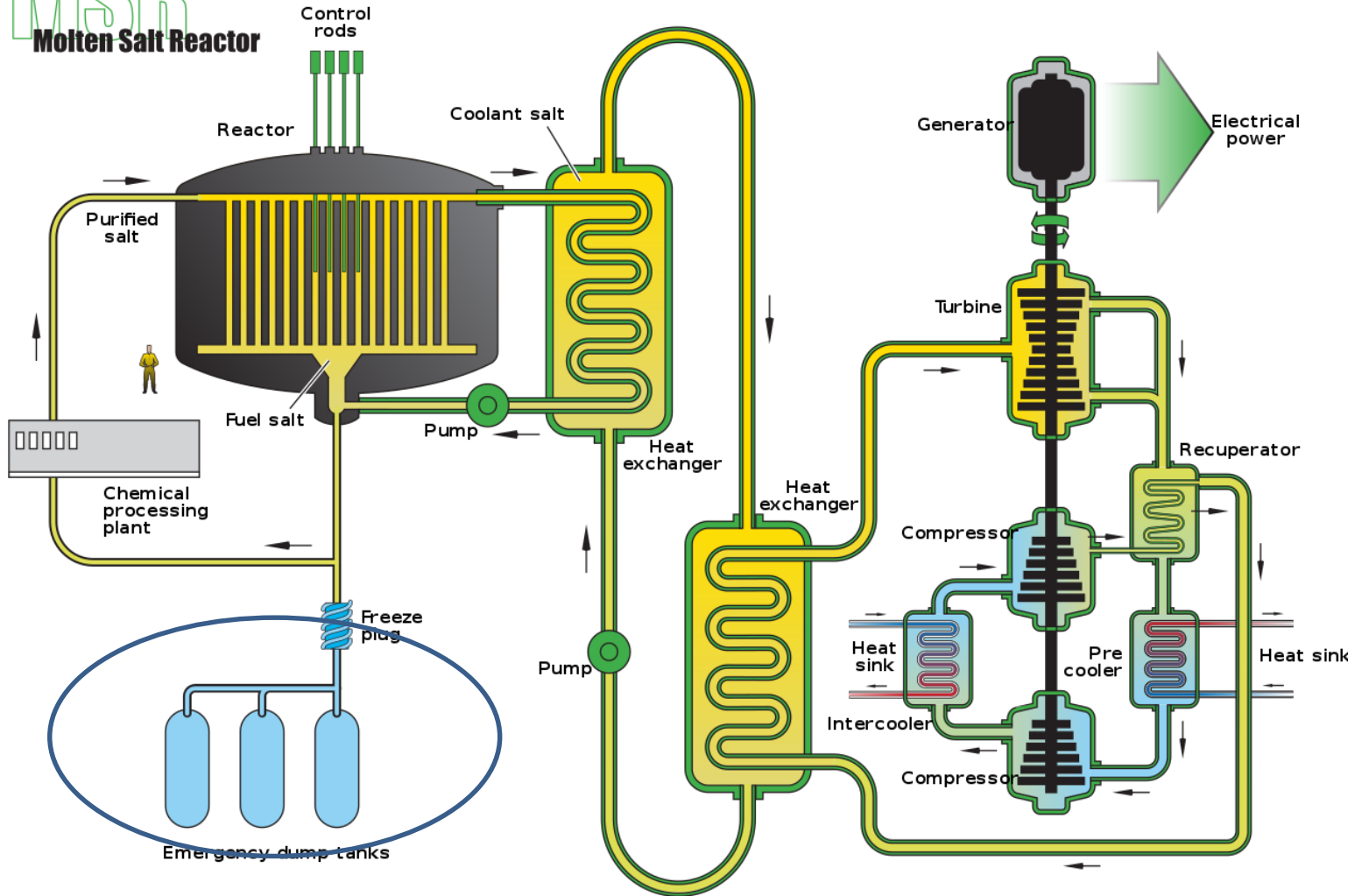
Vessel Head

- Low pressure operation
- “Vent off”, extract fission gases
 - Krypton
 - Xenon, strong n-absorber: no more poisoning out after shutdown, can restart immediately.

Apply to MSR's

MSR

Molten Salt Reactor



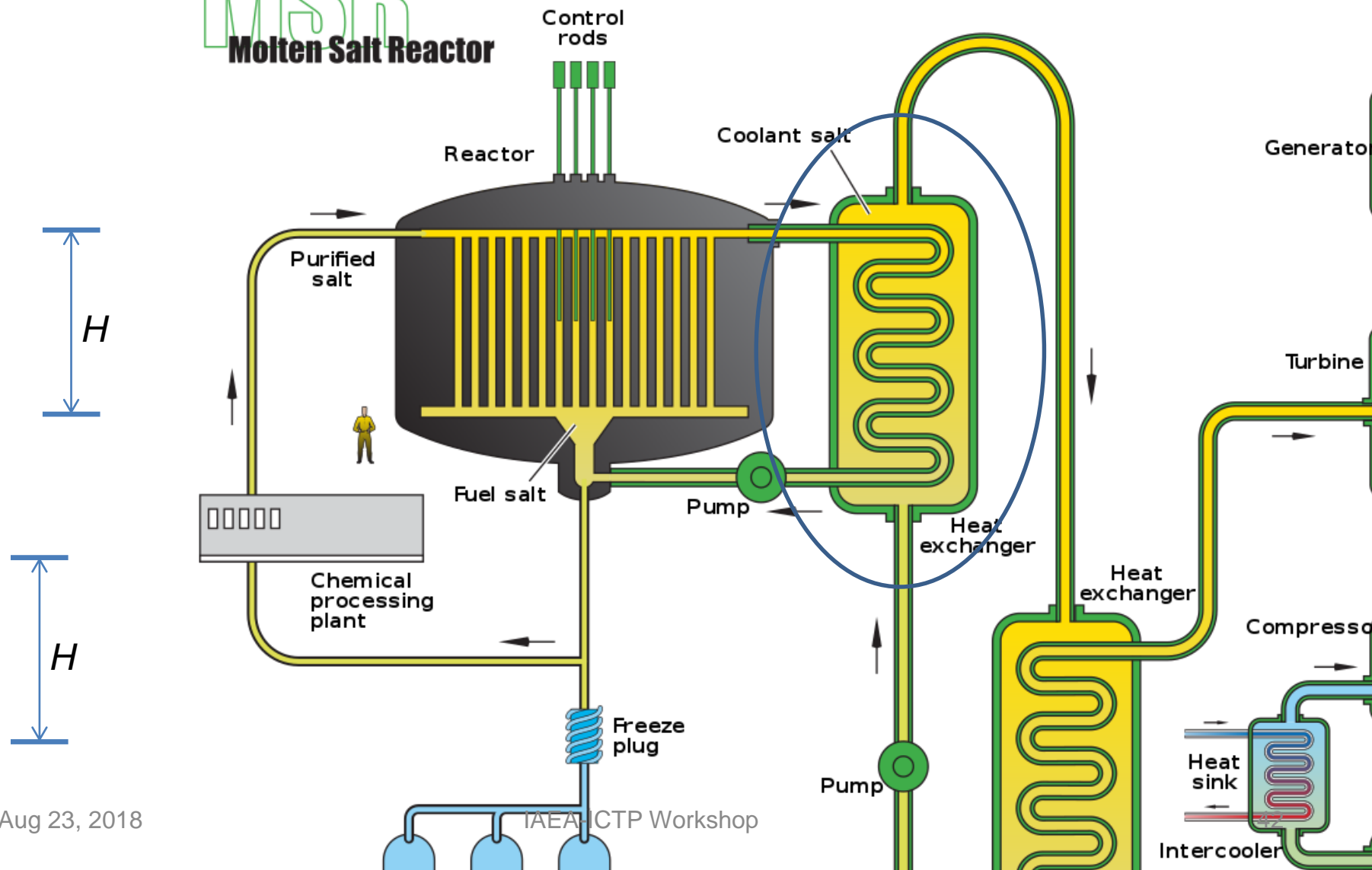
Dump Tanks

- Freeze plug: melts when temperature gets too high, fuel is dumped in tanks.
- Still need cooling from decay heat, passive cooling system.
- Worry about flooding.

Apply to MSR

MSR

Molten Salt Reactor



Aug 23, 2018

IAEA-NCTP Workshop

Primary Circuit Outside Core

- Good for letting Pa decay
- Ratio: $R = \frac{\text{time in core}}{\text{time out of core}}$ for a given sample of fuel salt.
- Equal to the ratio of volumes: $V_{\text{in}}/V_{\text{out}}$.
- Small R = good for **Pa decay**.
- Small R = bad for **delayed neutrons**.

$$\text{time in core}(\tau_{\text{in}}) = \frac{\text{height of core } (H)}{\text{liquid velocity } (u)}$$

Point Kinetics in MSR

Recall:

$$\begin{aligned}\frac{dC_1(t)}{dt} &= \frac{\beta_1}{\Lambda} n(t) - \lambda_1 C_1(t) \\ &\vdots \\ \frac{dC_6(t)}{dt} &= \frac{\beta_6}{\Lambda} n(t) - \lambda_6 C_6(t)\end{aligned}$$

Now (group k=1 only):

u_z is the velocity of the salt flowing in the z direction.

$$\begin{aligned}\frac{dC_1(z, t)}{dt} &= \frac{\partial C_1(z, t)}{\partial t} + u_z \frac{\partial C_1(z, t)}{\partial z} \\ &= \frac{\beta_1}{\Lambda} n(z, t) - \lambda_1 C_1(z, t)\end{aligned}$$

Point Kinetics cont'd

$$C_k(0, t) = C_k(H, t - \tau_{\text{out}})e^{-\lambda_k \tau_{\text{out}}}$$

$$\tau_{\text{out}} = \tau_{\text{in}} \frac{V_{\text{out}}}{V_{\text{in}}}$$

Bad news:

- Delayed neutron precursors decay outside of core.
 - Reduces beta (β)
 - Affects the controllability of the reactor
 - Activates the outer circuit

MSRE Experience (1969)

Summary of MSRE Nuclear Parameters with ^{235}U and ^{233}U Fuels

Parameter	Units	^{235}U Fuel		^{233}U Fuel	
		Calculated	Measured	Calculated	Measured
Initial critical concentration in salt	g U/liter	33.06 ^a	32.85 ± 0.25 ^a	15.30 ^b	15.15 ± 0.1 ^b
Reactivity loss due to circulation of delayed-neutron precursors	% $\delta k/k$	0.222	0.212 ± 0.004	0.093	^c
Control-rod worth at initial critical loading ^d	% $\delta k/k$				
1 Rod		2.11	2.26	2.75	2.58
3 Rods, banked		5.46	5.59	7.01	6.9
Temperature coefficient of reactivity at operating loading	$\frac{\delta k/k}{^\circ\text{F}}$ ($\times 10^5$)				
Total		-8.1	-7.3 ± 0.2	-8.8	-8.5
Fuel		-4.1	-4.9 ± 2.3	-5.7	^e
Concentration coefficient of reactivity	$\frac{\% \delta k/k}{\% \delta c/c}$	0.234	0.223	0.389	0.369

^a ^{235}U only.

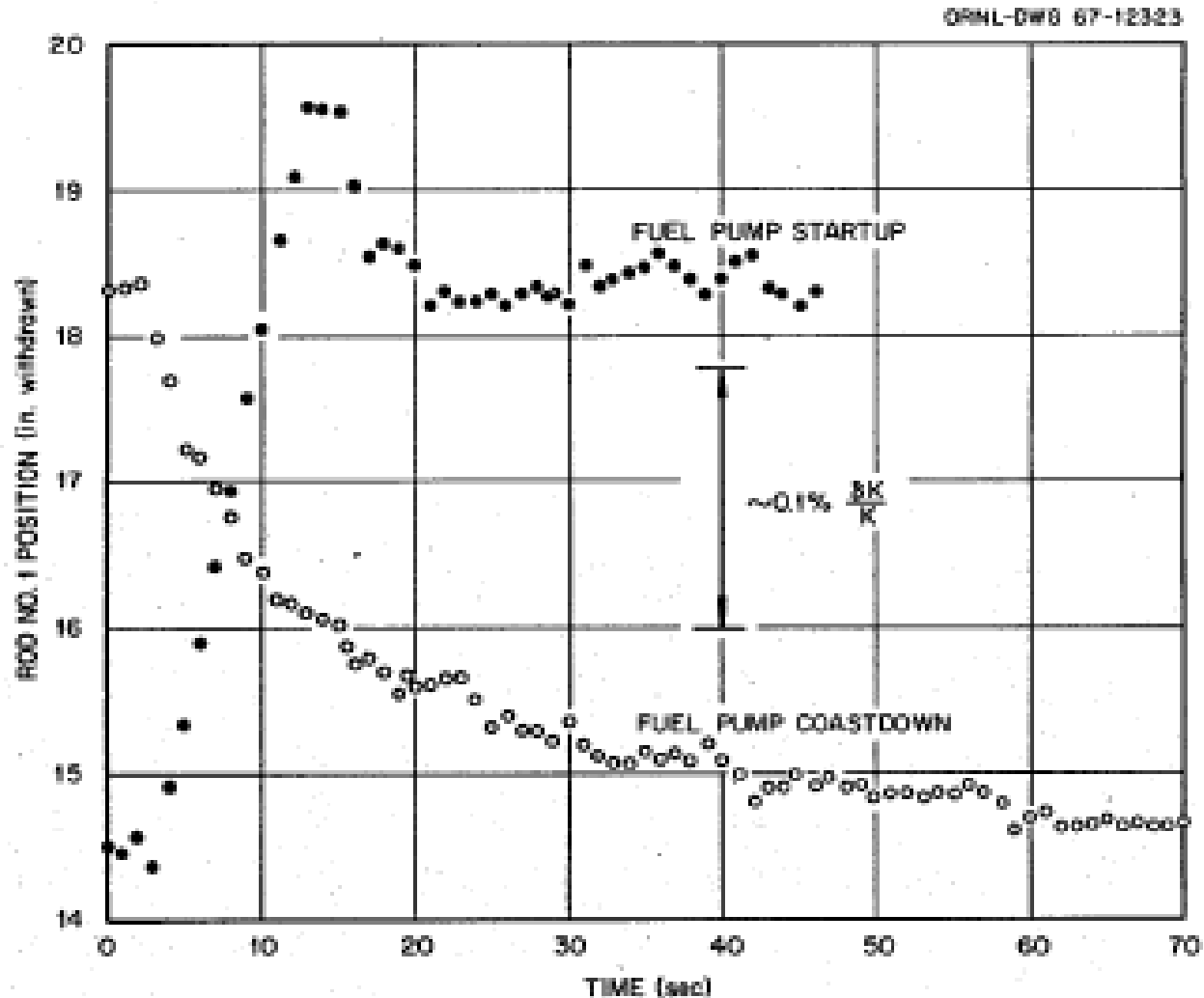
^bUranium of the isotopic composition of the material added during the critical experiment (91% ^{233}U).

^cMeasurement obscured by effect of circulating voids.

^dNormal full travel of rod(s).

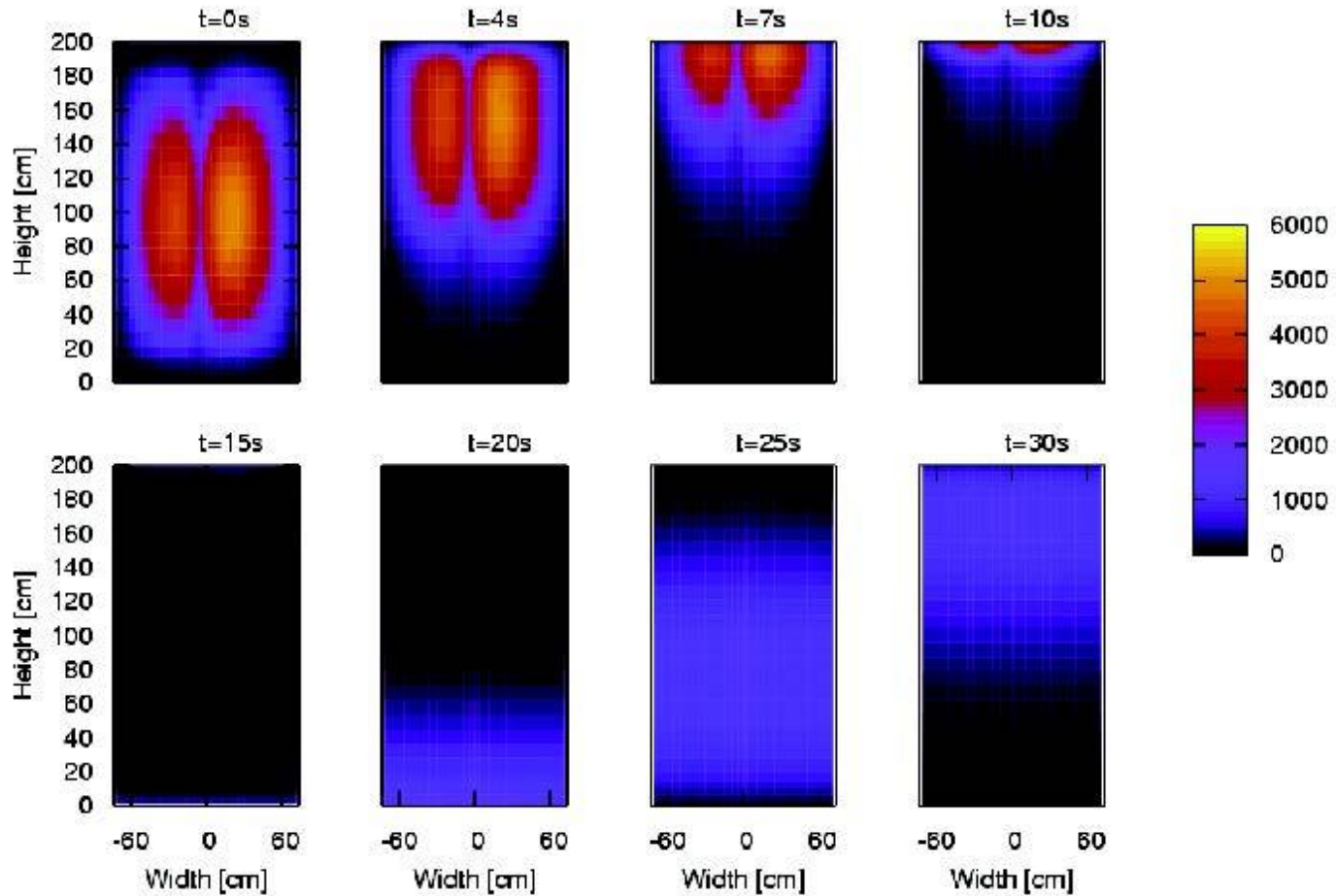
^eNot separately evaluated.

MSRE: Zero-Power Exp.



Aug: Fig. 24. Control-Rod Response to Fuel-Pump Startup and Coastdown.

MSRE Calculation



Multiphysics analysis by Danny Lathouwer (TU Delft)
Longest-living precursor group only.

Simulating MSR

- Static (design calculations):
 - Neutronics code; most are satisfactory:
 - MCNP
 - SCALE suite
 - Serpent
 - DRAGON/DONJON
 -
 - Depletion code:
 - Serpent
 - TRITON (SCALE)
 - DRAGON
 -

Simulating MSR

- Difficulties:
 - Very strong feedback with T/H.
 - Need iteration to get static solution, e.g. with a code such as RELAP.
 - May need CFD code.
 - Fortunately, only single phase flow.
 - Simulation of delayed neutrons.
 - Effect of Xe removal.
 - Simulation of abnormal conditions
 - Flow blockage
 - Travelling “slugs”, higher/lower density

Simulating MSR

- Much development is being done in this area, notably the Chinese **COUPLE** code: a time-space-dependent coupled neutronic and thermalhydraulics code.
- An important aspect of all these calculations is the determination of sensitivities and uncertainties:
 - E.g. the fuel temperature is negative, but what is the uncertainty? (in other words, how sure are we that it is negative?)
 - Focus has been on S/U due to nuclear data.
 - TSUNAMI, part of SCALE was developed for S/U studies.

Modern Designs

- The MSRE was an all-round good experience
- Renewed interest by private enterprises (start-ups); all have nice websites:
 - [IMSR by Terrestrial Energy](#) (*) (CA, US, UK)
 - [LFTR by Flibe Energy](#) (US)
 - [MCFR by Terrapower](#) (US)
 - [Transatomic Power](#) (US)
 - [ThorCon Power](#) (US, Indonesia)
 - [Thorium MSR by Copenhagen Atomics](#) (DK)
 - [Cube Reactor by Seaborg Technologies](#) (DK)
 - [Stable Salt Reactors by Moltex Energy](#) (UK, CA)

Summary of Commercial Ideas

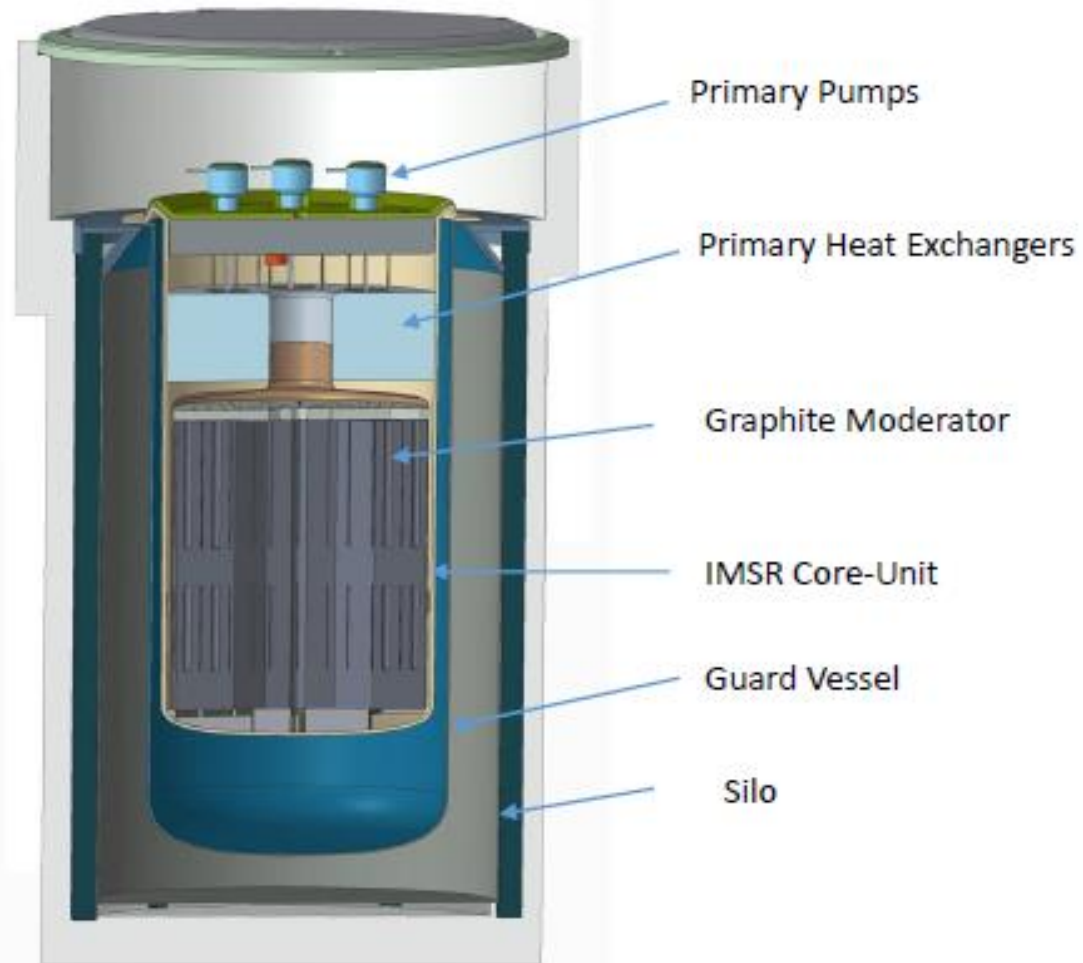
Vendor	Name	MWth	MWe	Mod	2nd Circ	Salt	Fuel	Breeding
Terrestrial Energy	IMSR-400	400	192	Graphite	Salt	FLi (no Be)	U/Th	possible
Flibe Energy	LFTR	600	250	Graphite	CO2	FLiBe	U/Th	Blanket
Terrapower	MCFR							
Transatomic Power		1250	520	ZrH		FLi (no Be)	U	
ThorCon Power			250	Graphite		NaBe/FLiBe	U/Th	in core
Copenhagen Atomics	Thorium MSR	Peripheral research						
Seaborg Technologies	Cube Reactor	50 (pilot)		Graphite	FLiNaK	FLi (no Be)	U/Th	Waste burner
Moltex Energy	Stable Salt Reactors	375	150		Na+K+Zr &F	Na+U+Pu&Cl	U/Pu	(fuel tubes)

IMSR-400 by Terrestrial Energy

- Based on MSRE experience;
- Modular design (SMR):
 - Two units, one operational, one cooling down
 - Containment is never opened
 - Seven year life-cycle
- Fission gas venting, but
 - No fission product removal
 - No online reprocessing
 - Top up with 20% LEU
- Fuel salt composition proprietary (no Be)

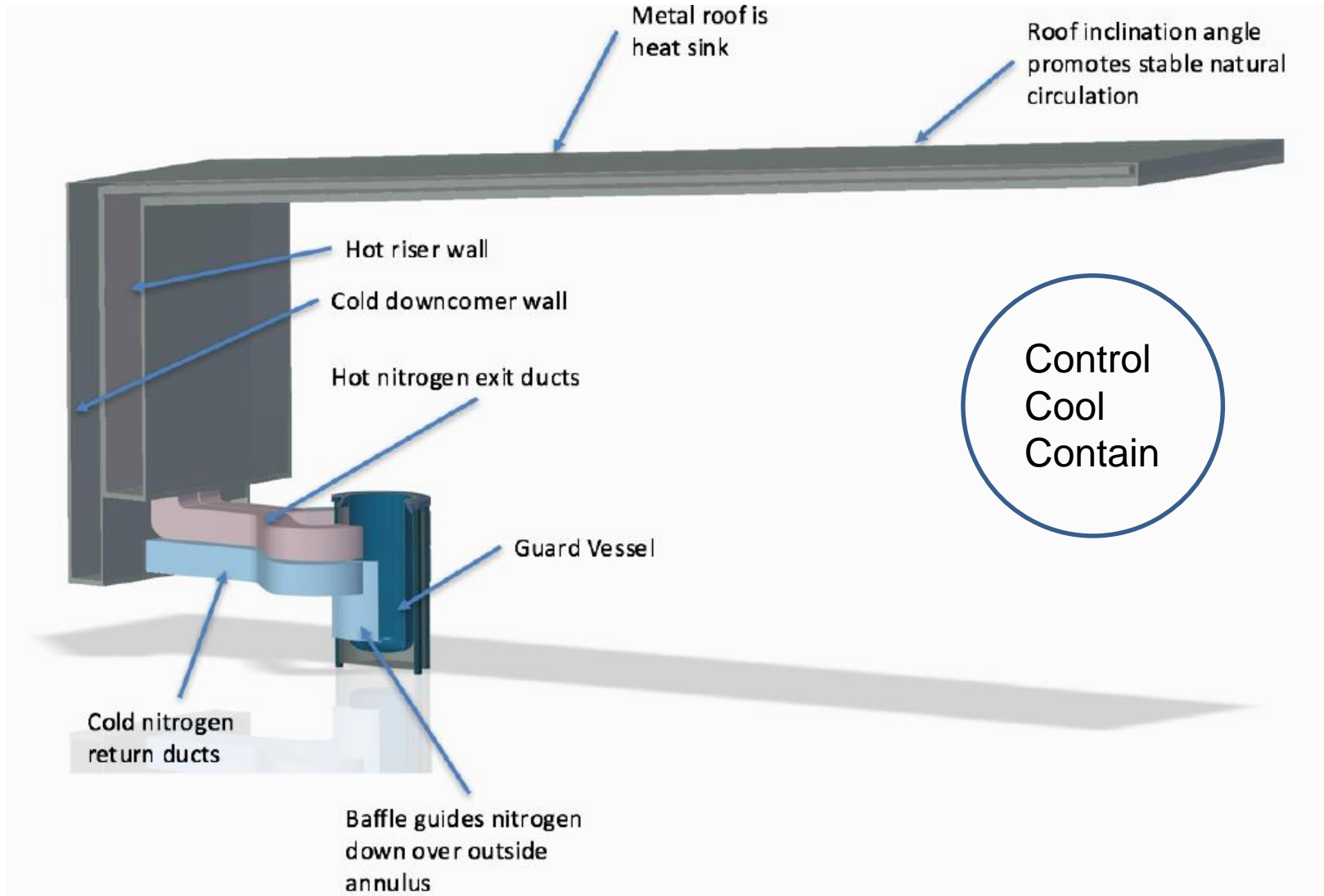
IMSR-400 Lay-out

- No dunk-tank!
- Instead always-on passive cooling



Core-unit and guard vessel in one of the two reactor silos

IMSR-400 Passive cooling



Big Players: China

- Part of Gen IV program.
- Thorium Molten Salt Reactor developed by SINAP, two phases:

TABLE 1 Comparison of TMSR-SF and TMSR-LF

	TMSR-SF	TMSR-LF
Fuel	Solid pebble fuel, with TRISO particles.	Liquid fuel, no solid fuel fabrication (LiF-BeF ₂ -UF ₄ -ThF ₄ (19.75% U-235))
Fuel cycle	Once-through	Reprocessing and recycle (online degassing (Xe, ke, T), off-line remove solid fission products)
Neutron spectrum	Thermal spectrum	Thermal spectrum Fast spectrum
Application	High-temperature application: Electricity production; Seawater desalination Hydrogen or methyl alcohol production; Steam supply; Utilization of thorium fuel	Near term: High-temperature application; Utilization of thorium fuel; Burning minor actinides

TMSR-SF Concept

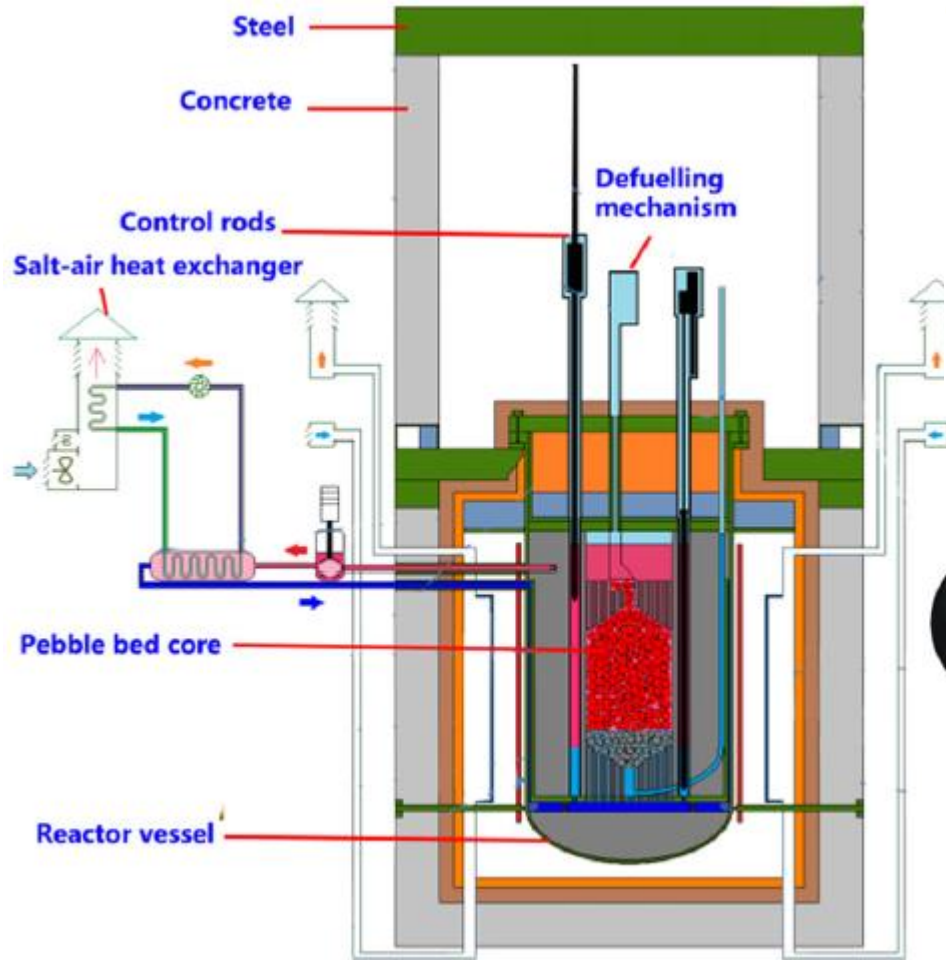
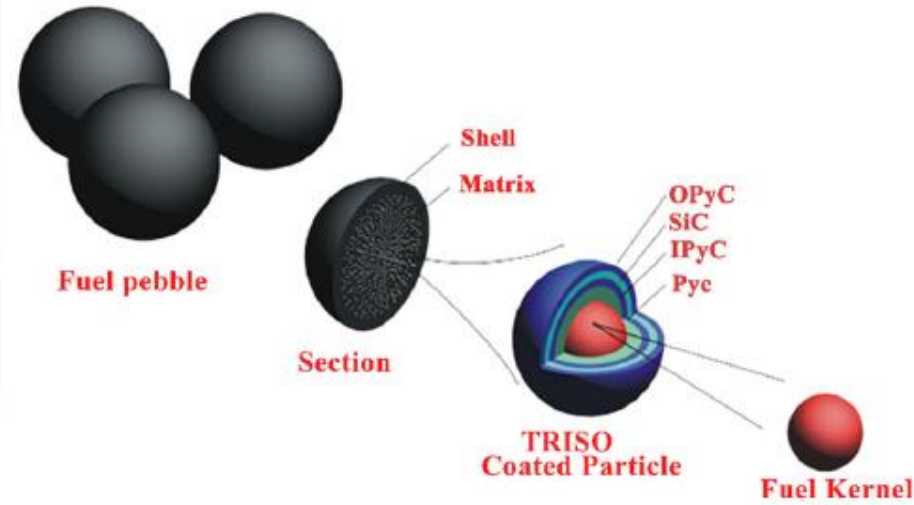


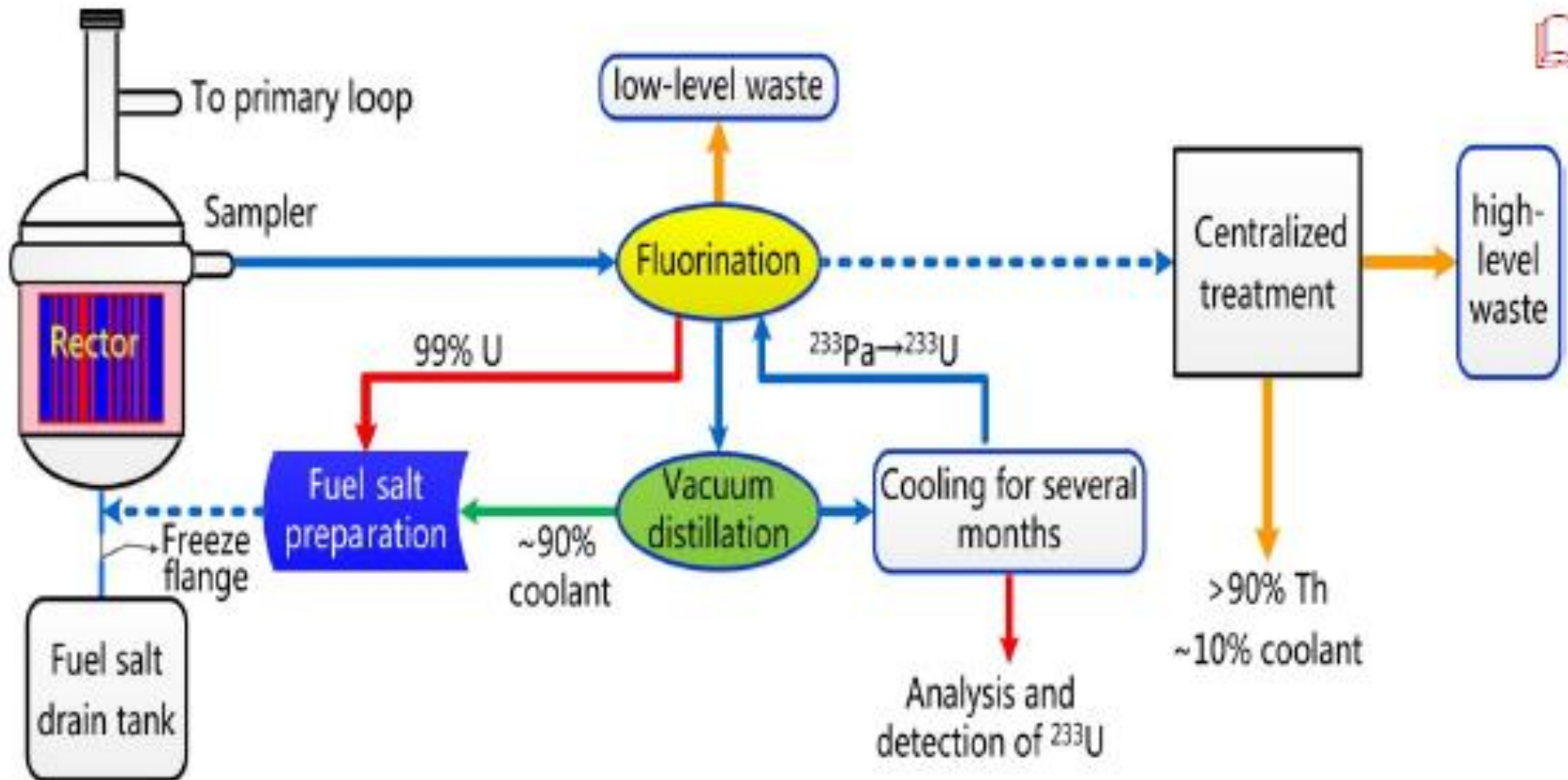
TABLE 2 Operation parameters of 2 MW TMSR-SF and 10 MW TMSR-SF

	2 MW	10 MW
Power	2 MWt	10 MWt
Inlet temperature	600°C	600°C
Outlet temperature	620°C	650°C
Massflow rate	42.3 kg/s	84 kg/s
Primary coolant	FLiBe	FLiBe

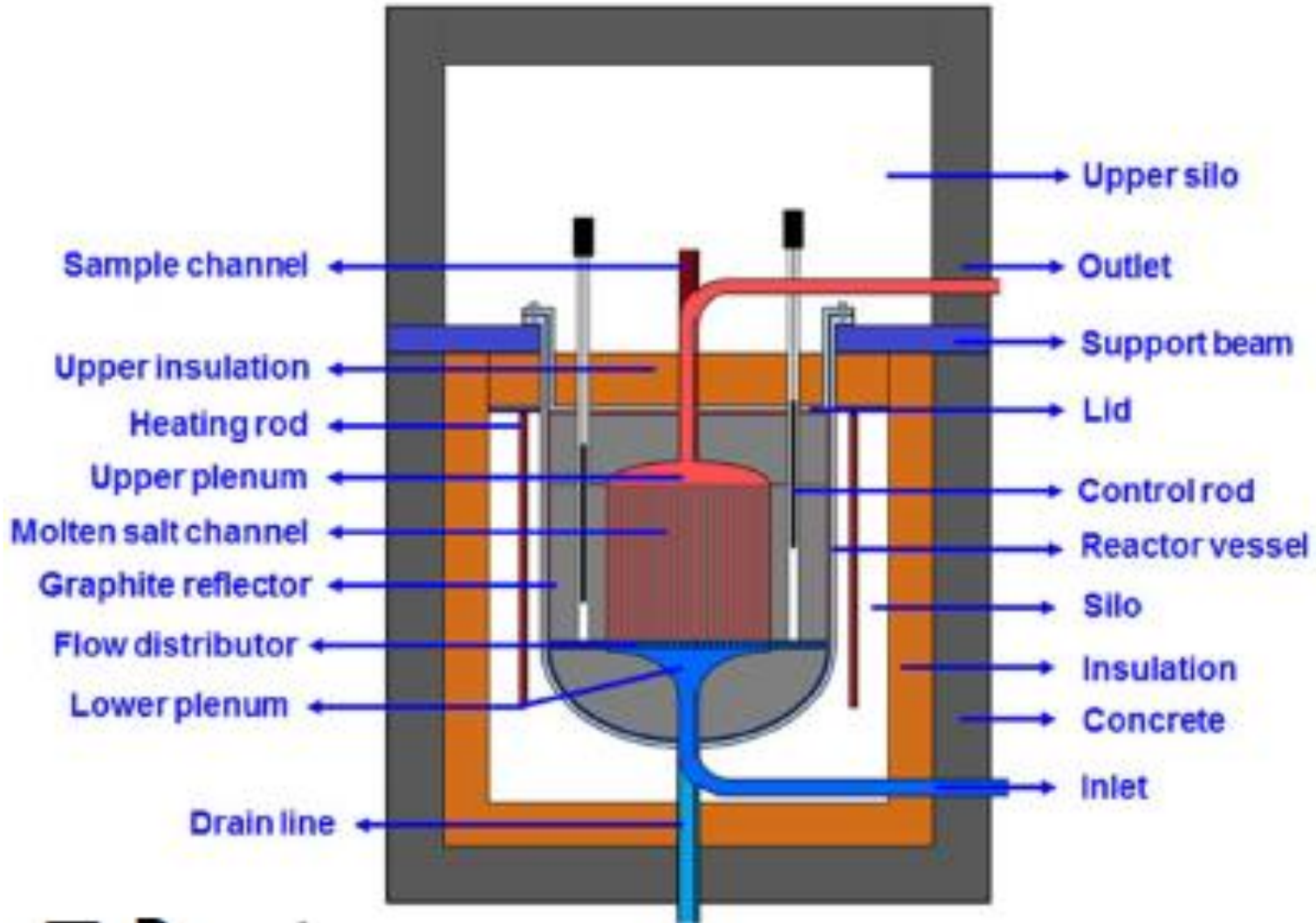
Uses TRISO particles.



TMSR-LF Concept



TMSR-LF1 Layout



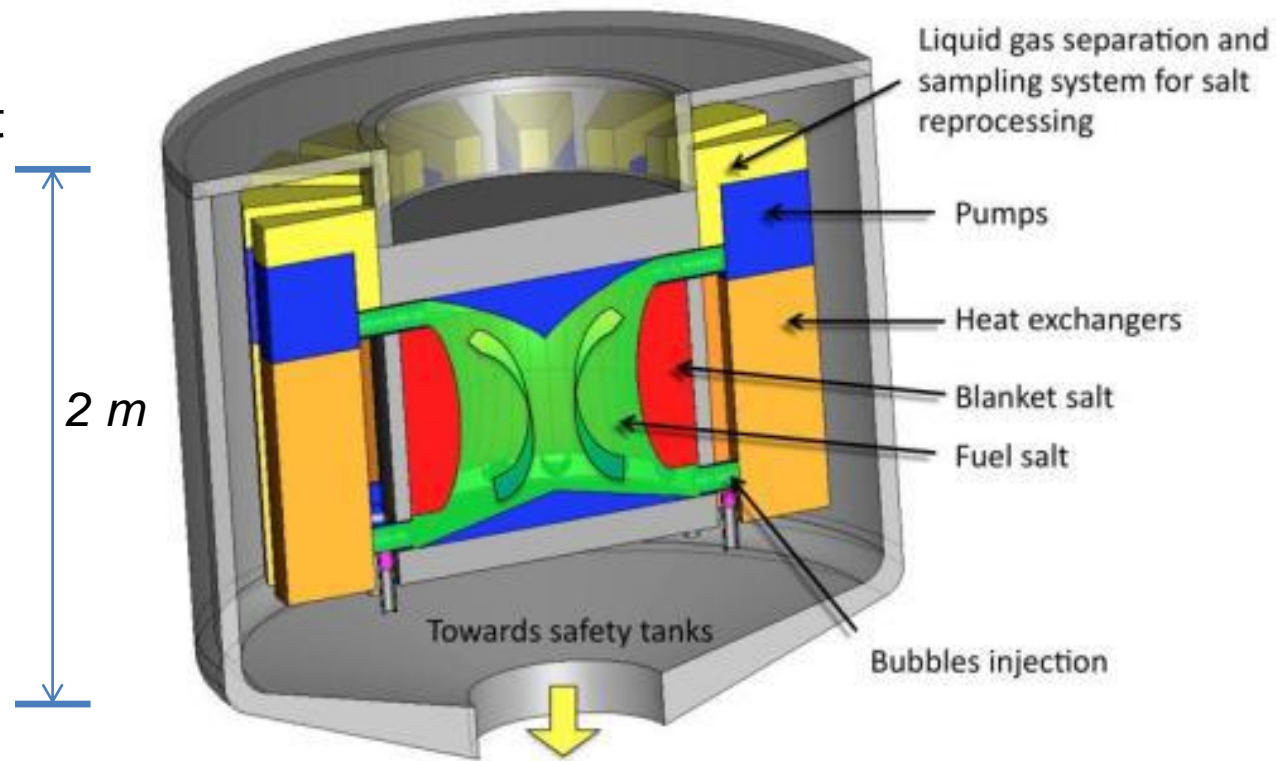
TMSR-LF Design Parameters

Parameter	Value
Thermal Power	395 MW
Electrical Power	168 MW
Inlet Temperature	600 °C
Outlet Temperature	700 °C
Fuel Salt (primary loop)	LiF-BeF ₂ -UF ₄ -ThF ₄
Cooling Salt (secondary loop)	FNaBe
Moderator	Graphite
Reactor vessel size (D × H)	5.3 m × 6.0 m

- International Collaborations with ORNL and MIT.
- Existing test facilities for studying salt properties.

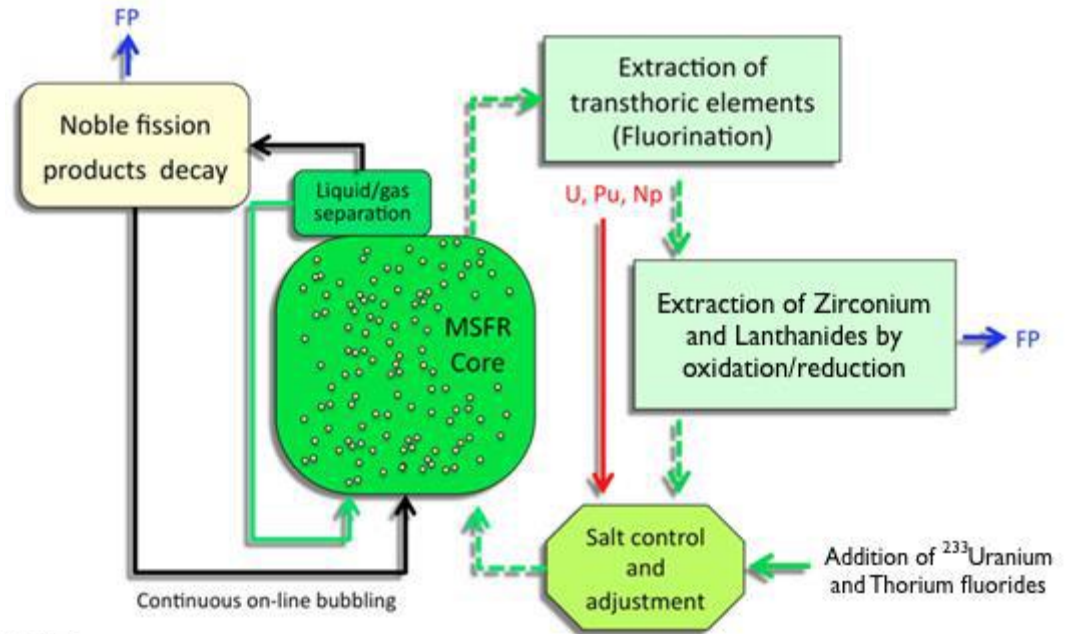
MSFR (CNRS, EVOL, SAMOFAR)

- GEN IV reference reactor
- Fast spectrum, no moderator.
- Large:
 - 3000 MWth
 - 18 m³ fuel salt
 - 50/50, 3.9 s
- Fuel:
 - LiF-HMF_x
(77.5-22.5)
 - 750 °C
- Blanket:
 - LiF-ThF₄
(77.5-22.5)
- No control rods
 - Temp FB
-8 pcm/K

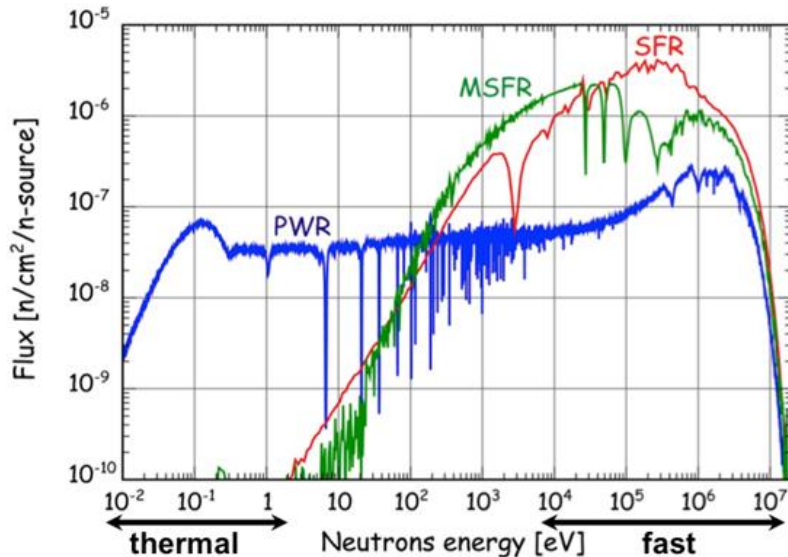


MSFR

- Physical separation in core:
 - Gas bubbling
 - Extract Kr, Xe, He and F.P. in suspension



From E. Merle, 2017

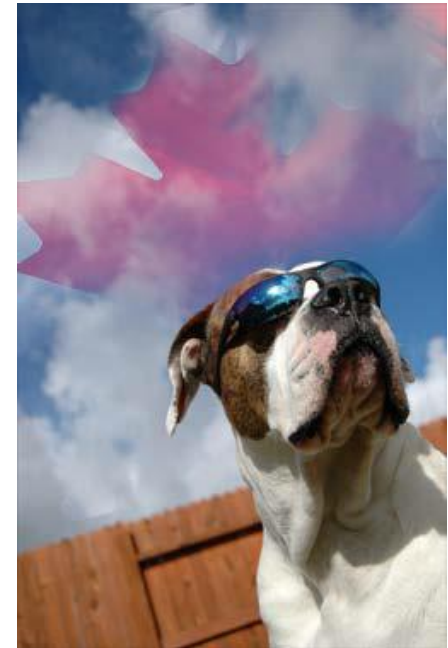


Fast spectrum
Breeding ratio 1.1

Last Word: the Regulator

Inspiring Innovation and Discovery

- Each country has its own regulator. Often working with and/or supported by IAEA.
- E.g. Canadian Nuclear Safety Commission
 - Not prescriptive, onus is on vendor
 - Need to prove design is safe
 - Diverse (support) staff, e.g.
 - Rumina Velshi (President to be)
 - Dumitru Serghiuta
 - Ramzi Jammal
 - Parvaiz Akhtar
 - Nana-Owusua Kwamena
 - Wei Shen
 - Mok Cher Fong



Conclusion

- MSR's have a long history.
- Early designs seem to have been successful.
- Renewed interest in the technology:
 - Private industry
 - Gen IV
 - International collaborations
 - Conservative designs likely to succeed
- MSR's are a safe, reliable and sustainable source of low-carbon electricity.