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Innovative Nuclear Energy Systems: Core Design and Neutronics

Joint ICTP-IAEA Workshop on Physics and Technology of Innovative Nuclear Energy Systems

12-16 December 2022

ICTP, Miramare - Trieste, Italy

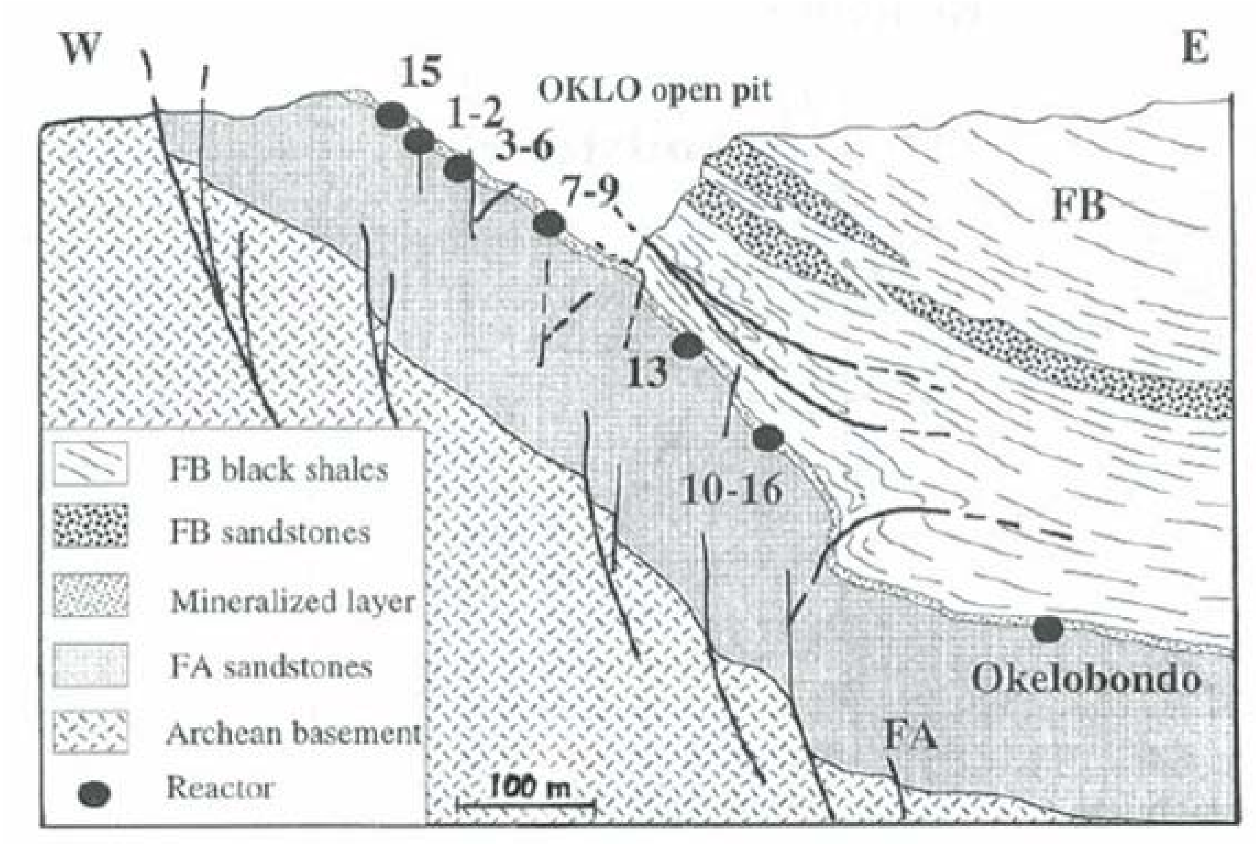
Adriaan Buijs

(McMaster University)

Outline and Purpose of Lecture

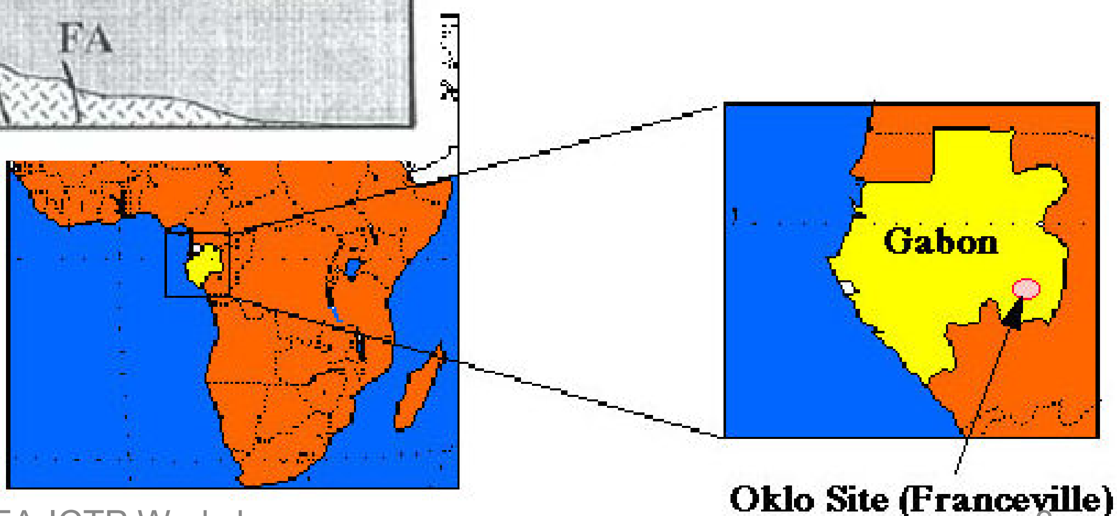
- Journey through reactor core design,
  - (Pre)-history
  - Inception
  - Evolution
  - Innovation
- Guided by personal experience
  - Heavy water
  - Molten salt
  - Solid state
- Concepts relevant to SMRs, based on the neutronics of the core design.

(Pre)-History



1.7 Billion years ago

U-235/U-238 ≈ 3%
Operated for several 100,000 years at 100kW



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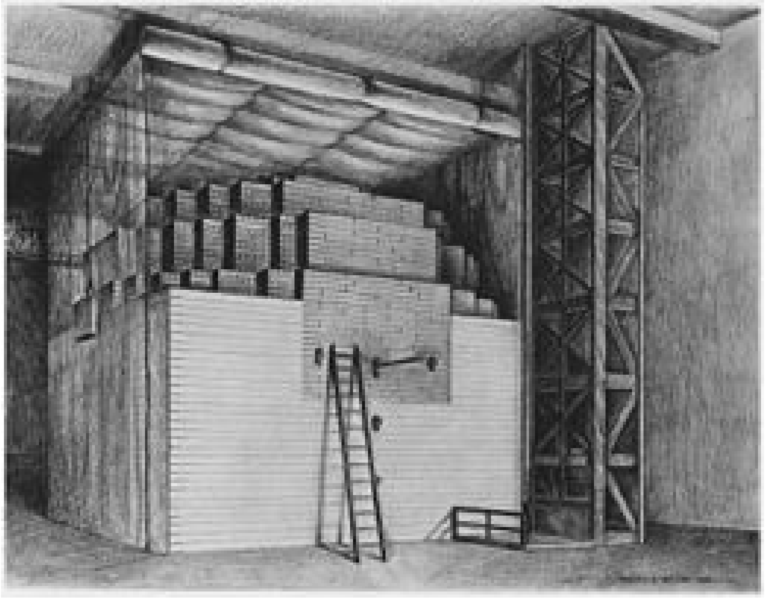
More Recent History

- 1939:
  - Otto Hahn, Fritz Strassmann, first observation of fission of uranium,
  - Otto Frisch, Lise Meitner, provided the right interpretation.
- Soon to be picked up in a race to develop the bomb as part of the WWII
- 1942:
  - Enrico Fermi, Chicago pile.



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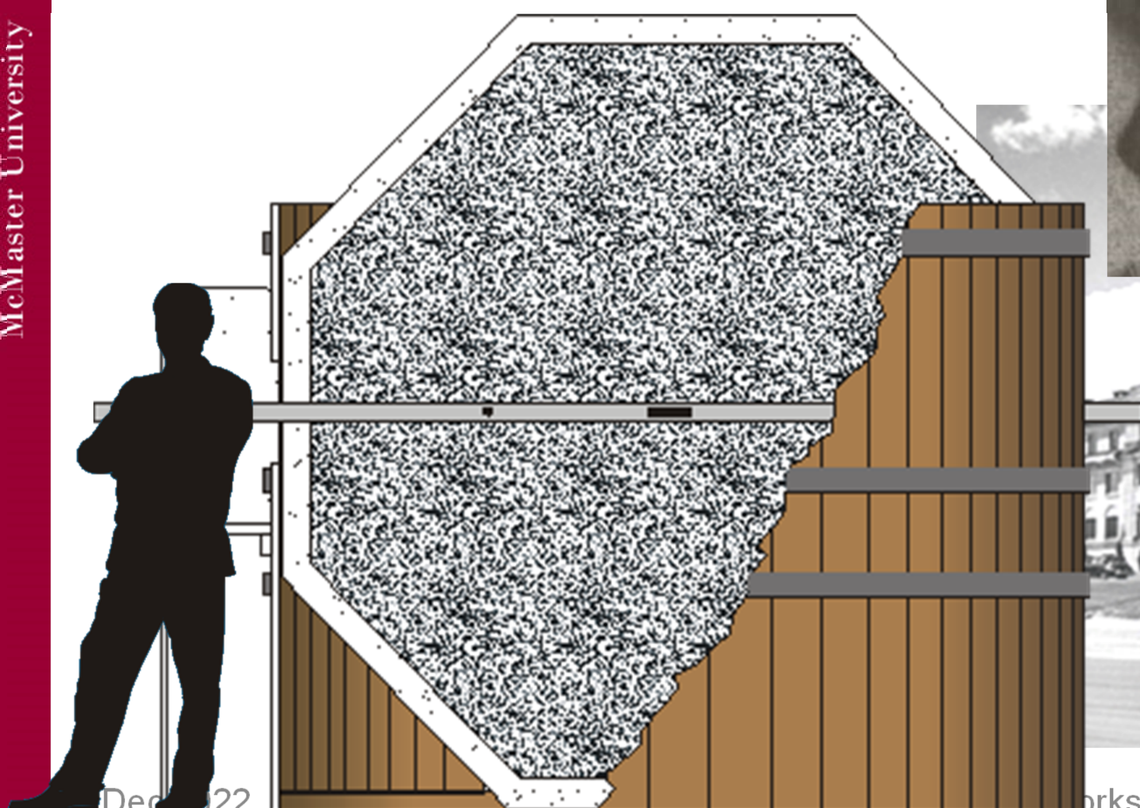
1941-42 …

George Laurence

(1905 - 1987)

World's first large-scale fission experiments in graphite

(National Research Council)

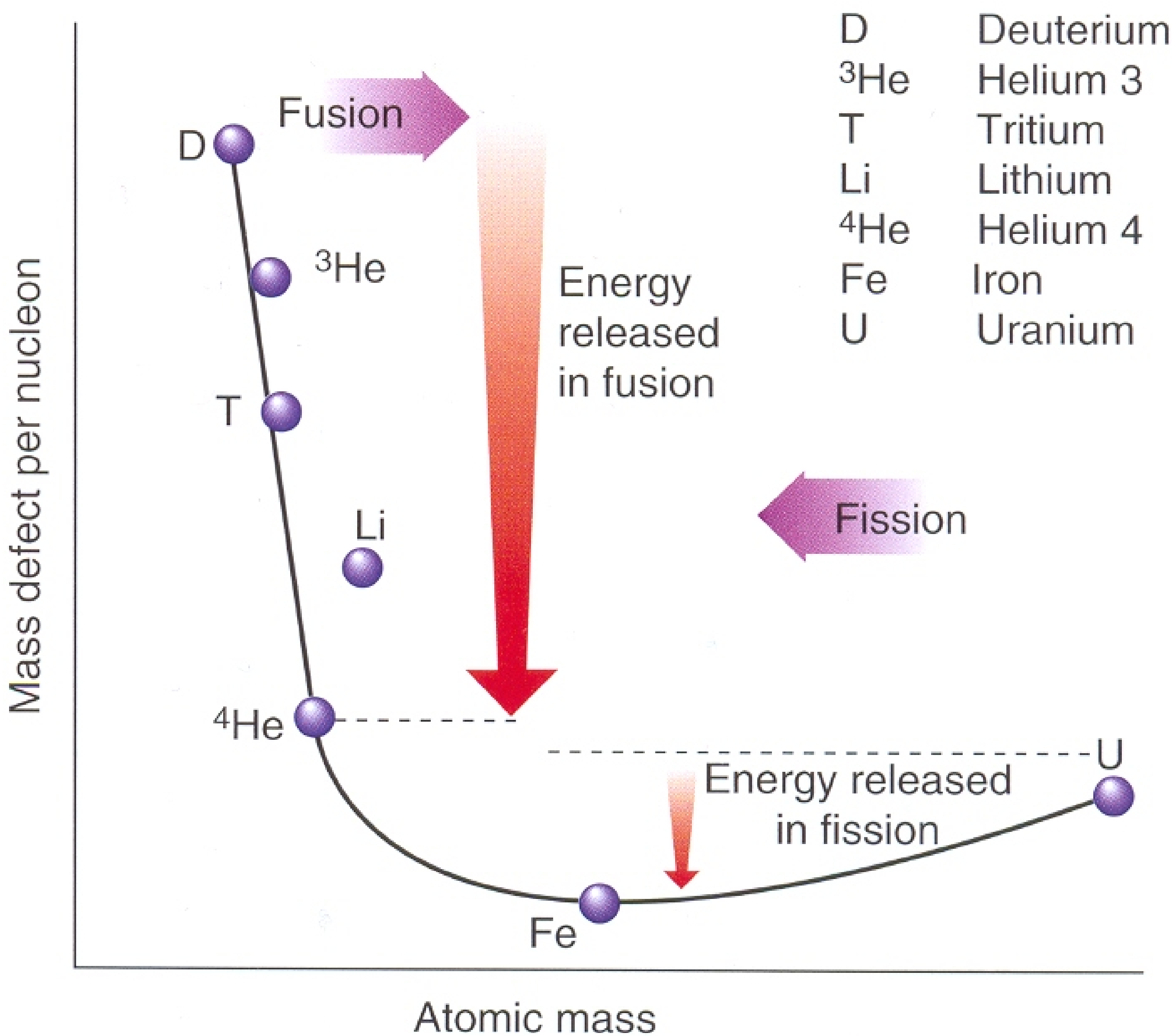


100 Sussex Drive, Ottawa

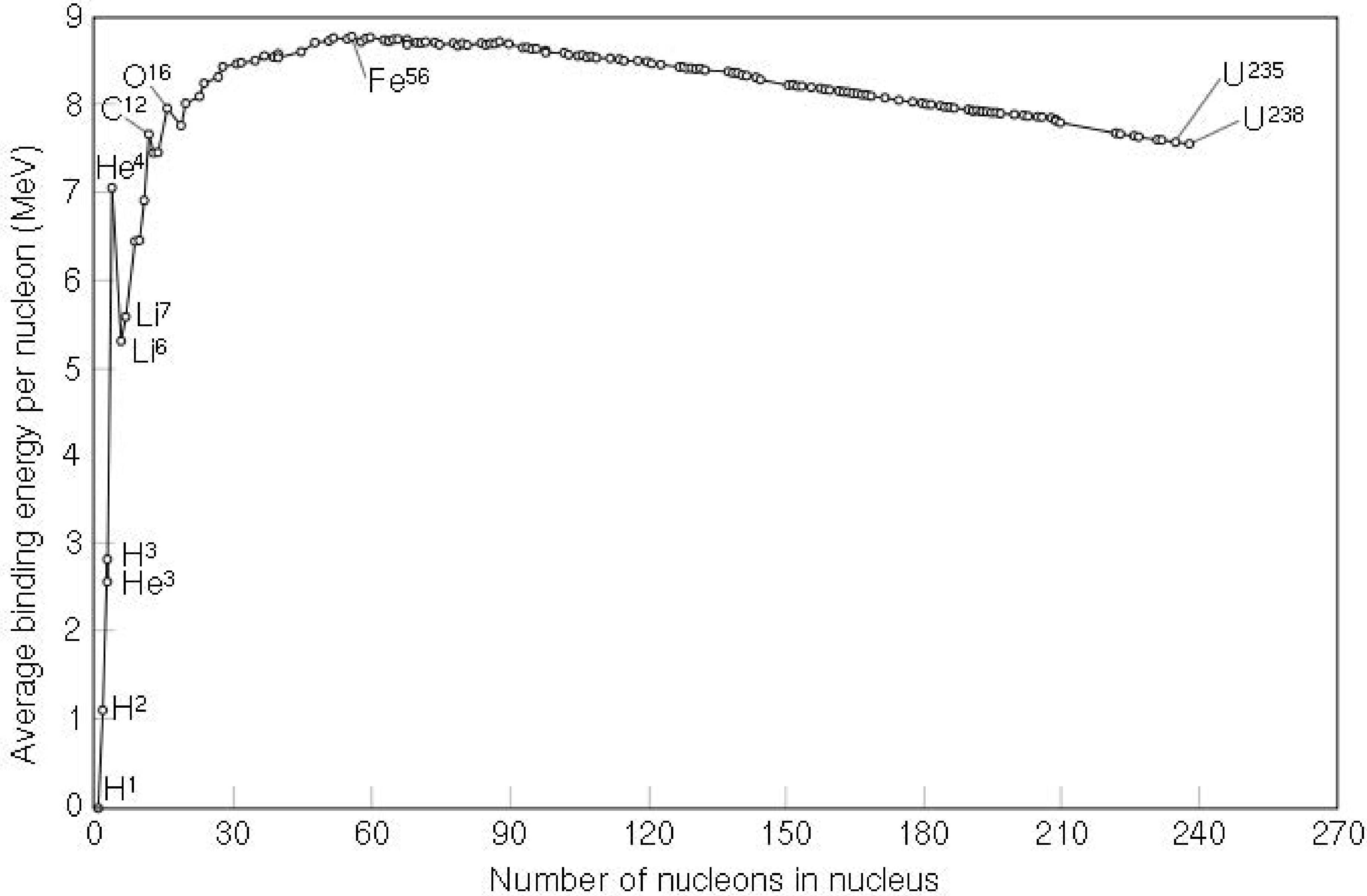
So how does it work?

Step 1:

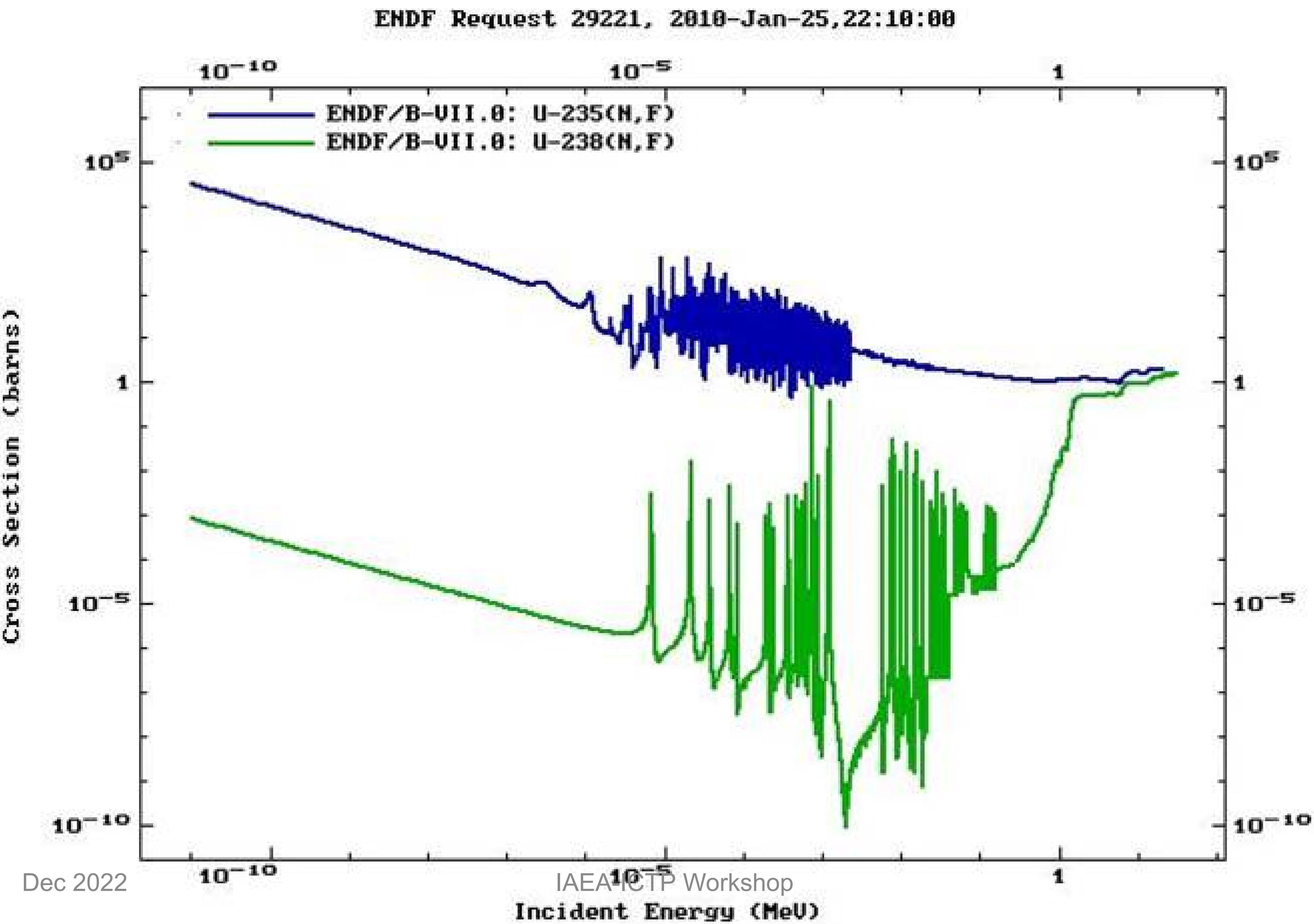
$E=mc^2$



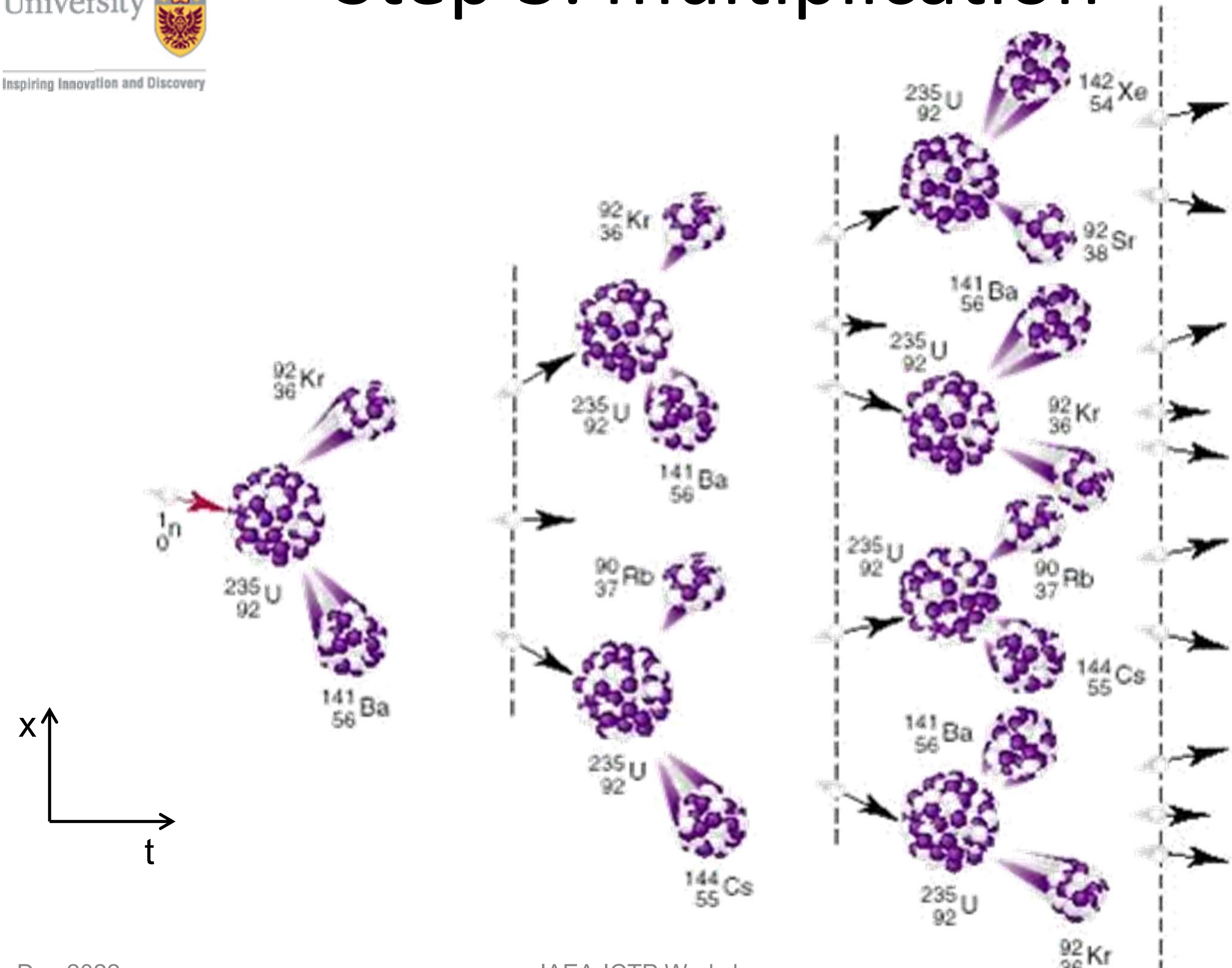
Same thing, binding energy



Step 2: Quantum Mechanics



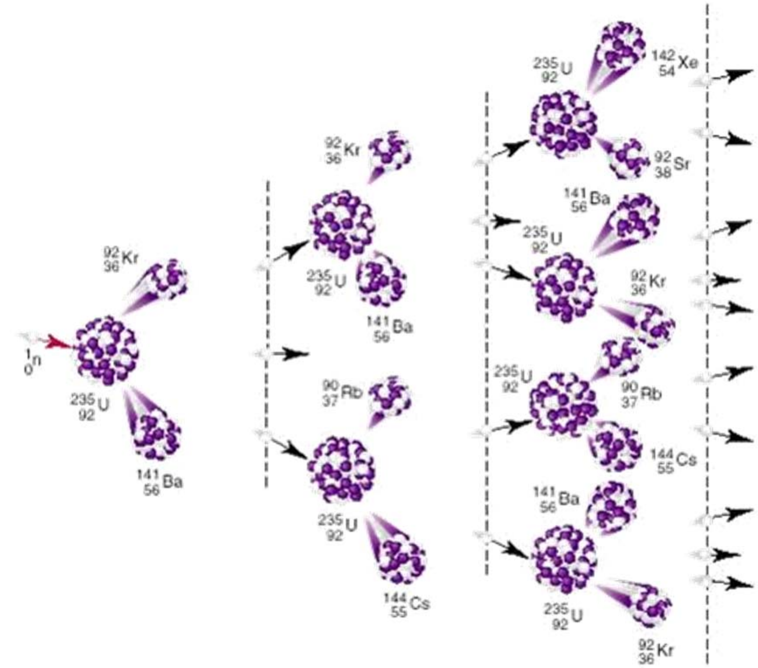
Step 3: multiplication



Let's do the Math

- Seems to double every generation:

$$\frac{dn}{dt} = \frac{1 \times n}{T}$$



- Solution:

$$n = n_0 \times e^{t/T}$$

- What a bomb!

Don't overreact

- In a real reactor, we would rather

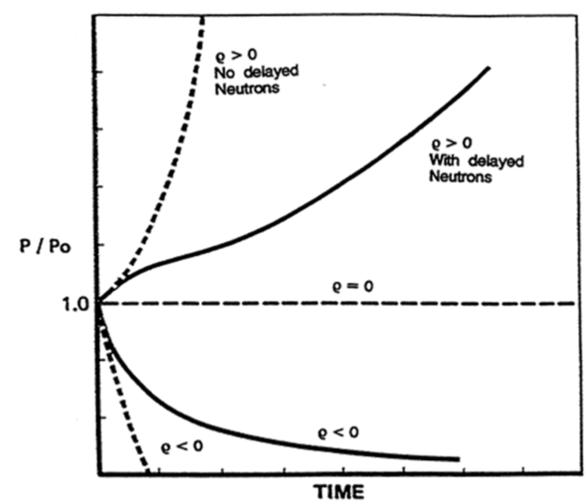
$$\frac{dn}{dt} = \frac{\rho}{T} n \cong 0$$

- Rho (ρ) is called **reactivity:**

$$\boxed{\begin{array}{ll} \rho = 0 & : \text{critical} \\ \rho > 0 & : \text{super critical} \\ \rho < 0 & : \text{subcritical} \end{array}}$$

- Solution:

$$n = n_0 \times e^{\frac{\rho}{T} t}$$



Choices

- Spectrum:
  - Fast neutrons
    - No moderation
  - Slow (thermal) neutrons?
    - Moderator
      - Graphite
      - Light water
      - Heavy water

- Fuel:
  - Enriched uranium
  - Natural uranium
    - Need heavy water or graphite

Natural Selection and Evolution

- Canada:
  - Heavy-water moderated, natural U fueled
- US and others:
  - Light-water moderated, enriched U fueled
- Soviet Union:
  - Graphite moderated, low enriched U fueled
- (Other designs)
- All solid fuel, liquid (gas) cooled.

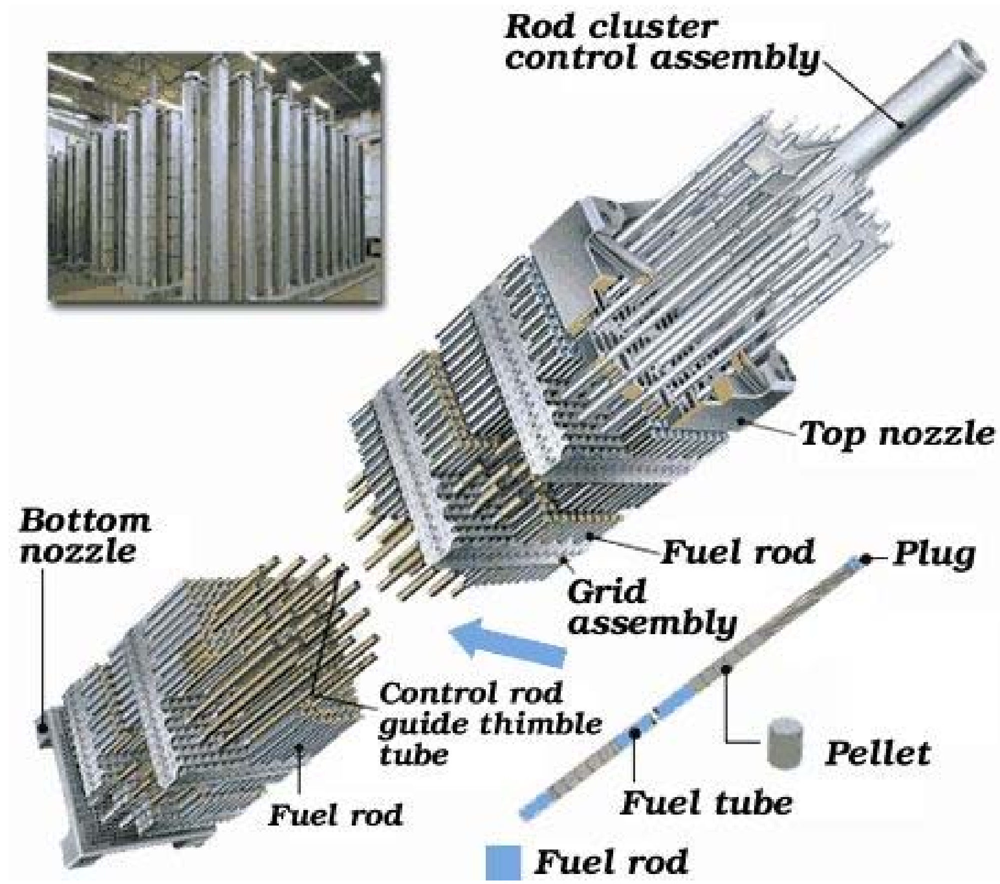
Mutations

- What if we think out of the box?
- What if the fuel is liquid, the moderator (if present) is solid?
- The fuel is the coolant.

- Consider a traditional design:

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
- Inefficient



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:

$$^{9}_{4}\mathrm{Be} + n \rightarrow 2\,^{2}_{2}\mathrm{He} + 2n$$

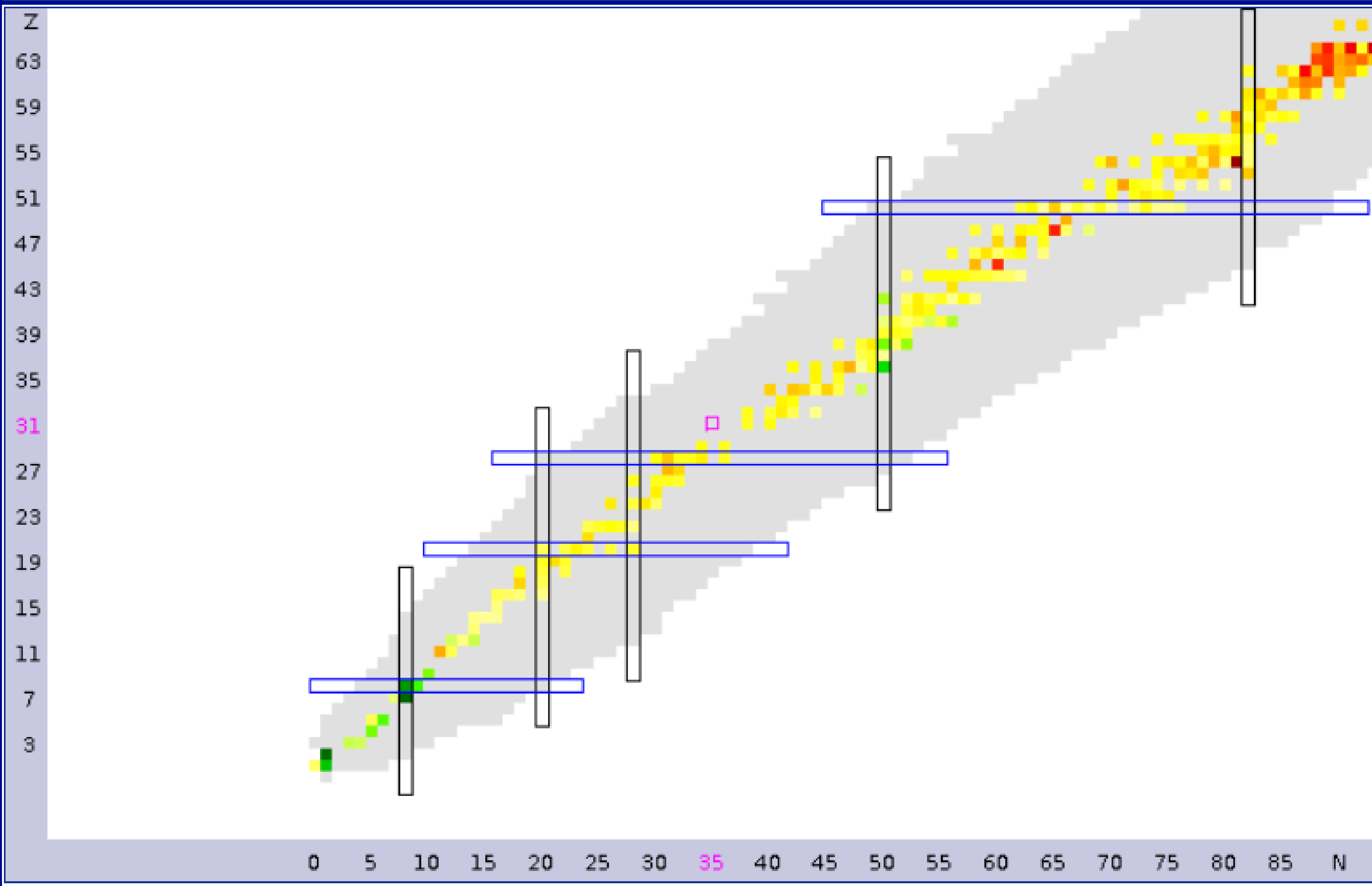
- Also high elastic cross section → good moderator.
- But beryllium is poisonous.
- Other elements such as Zr, Na, K are sometimes added for different purposes.

NNDC

Chart of Nuclides

Click on a nucleus for information

Color code	Half-life	Decay Mode	$Q_{\beta-}$	Q_{EC}	$Q_{\beta+}$	S_n	S_p	Q_α	S_{2n}	S_{2p}	$Q_{2\beta-}$	Q_{2EC}	Q_{ECp}
$Q_{\beta\text{-}n}$	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



Tooltips
On
Off

Zoom
1
2
3
4
5
6
7

Uncertainty
NDS
Standard

Screen Size
Narrow
Wide

Nucleus

go

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

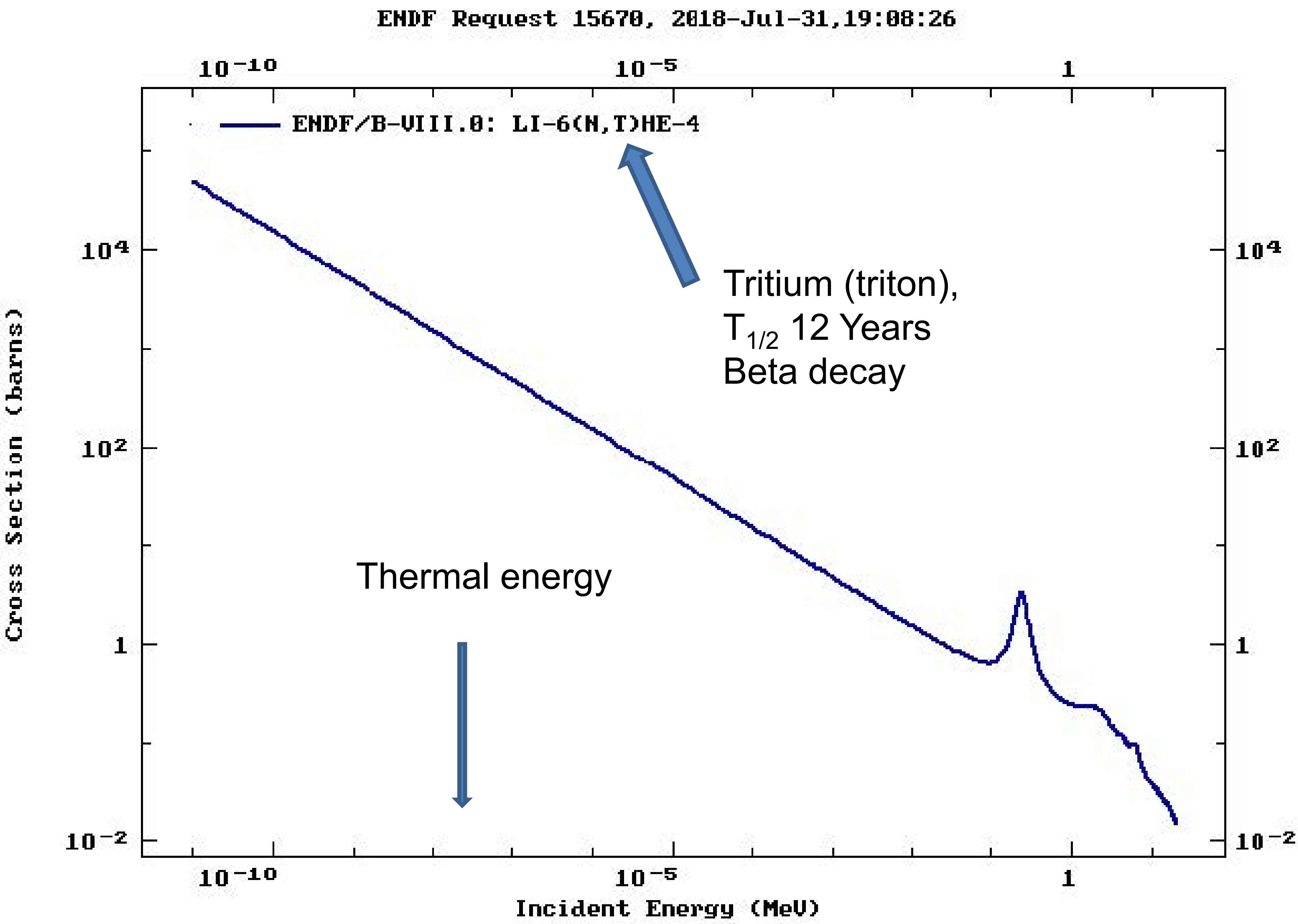
Search options:

Levels and Gammas
Nuclear Wallet Cards

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

E(level) (MeV)	Jπ	Δ(MeV)	$T_{1/2}$	Decay Modes	σ(n,γ) (b)
0.0	0+	-63.72366015625	9.49 h *3*	ε : 100.00 %	

Li-6 Cross Section



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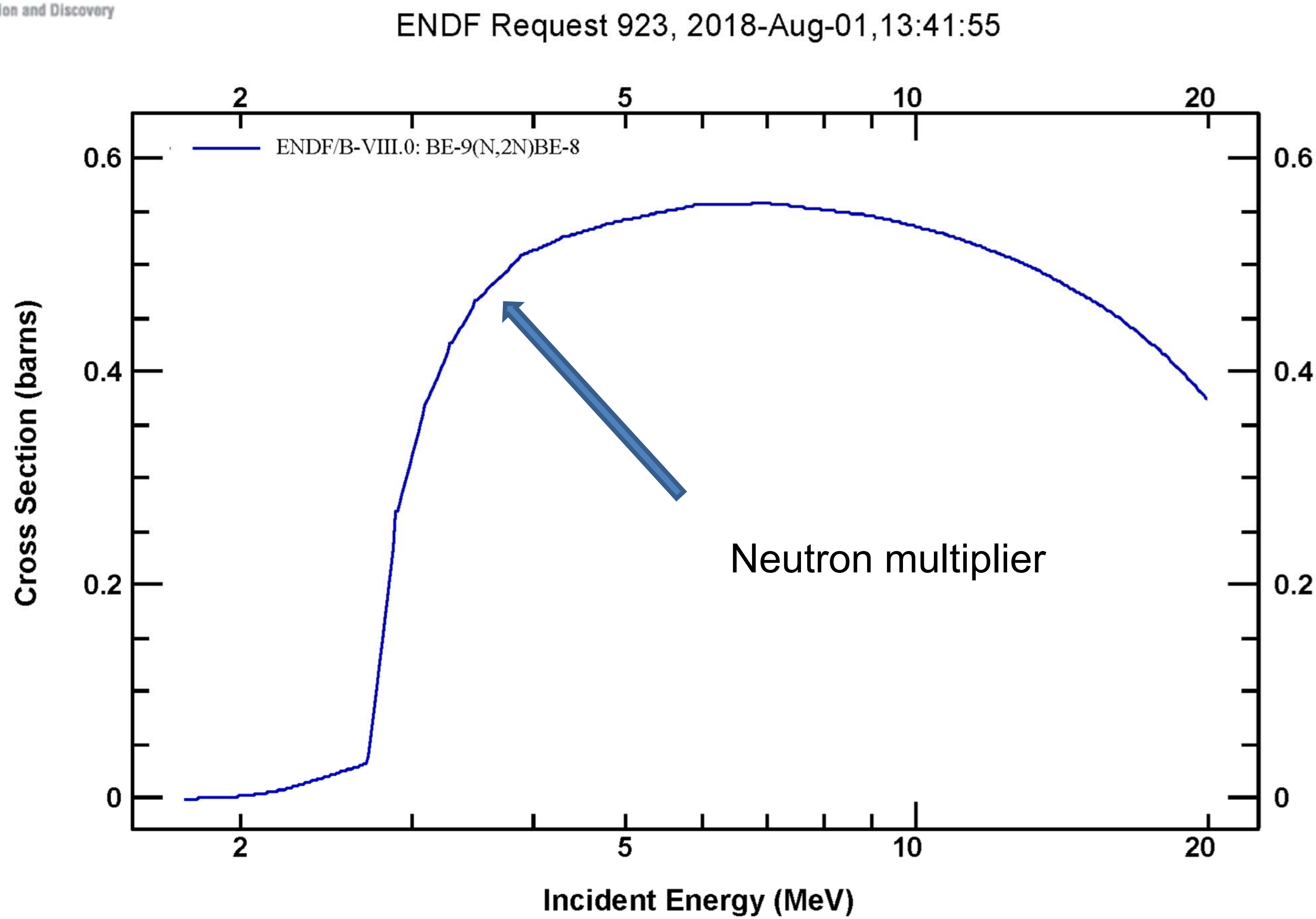
Be-9 Cross Section



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% ^{7}LiF – 29.1% BeF_2 – 5% ZrF_4 – 0.9% UF_4
  - With 35% enriched uranium

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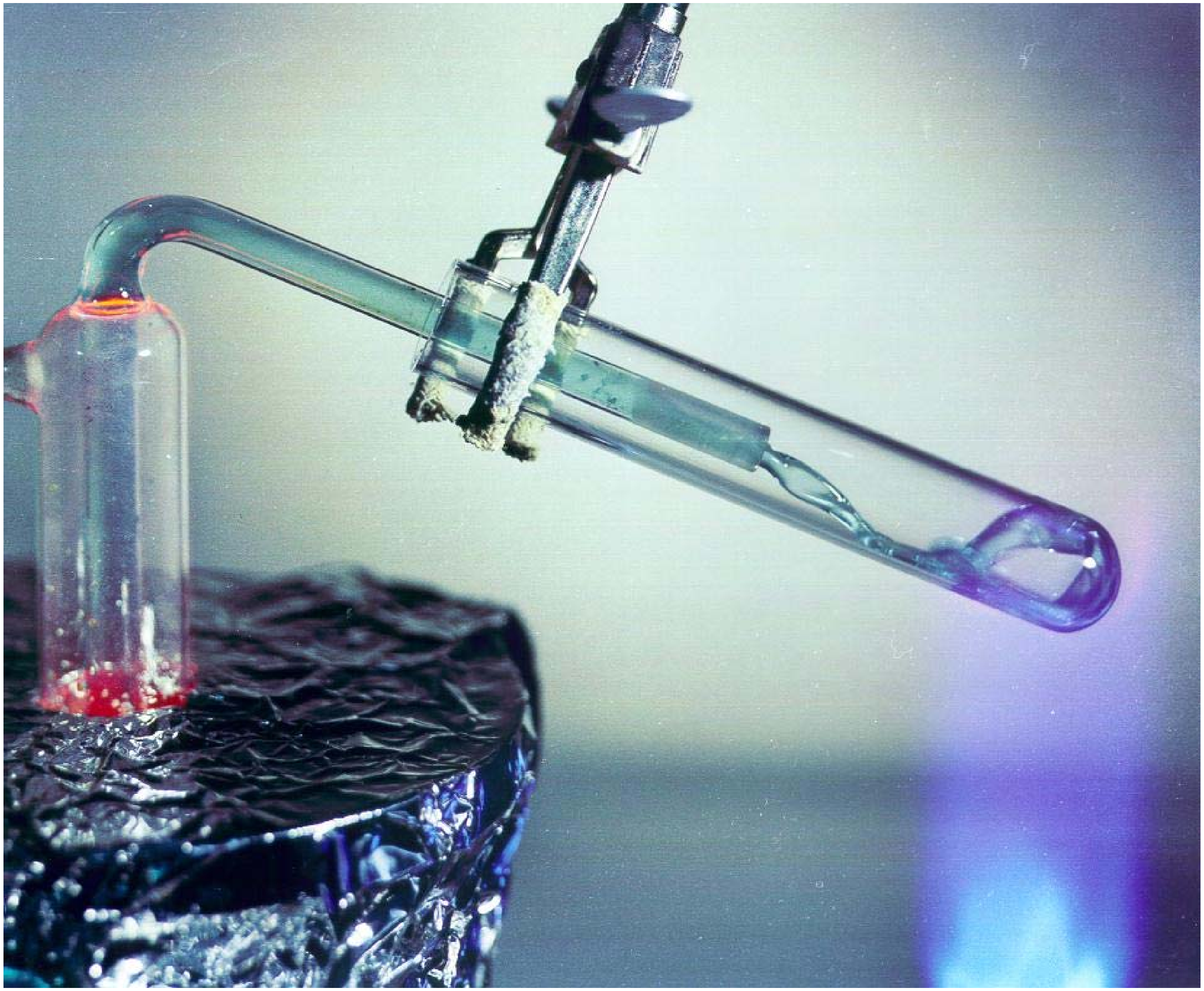
Fuel Salt Properties

MSRE Fuel

Property	H_2O	Na	Li	$^7LiF-BeF_2-ZrF_4-UF_4$ 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.

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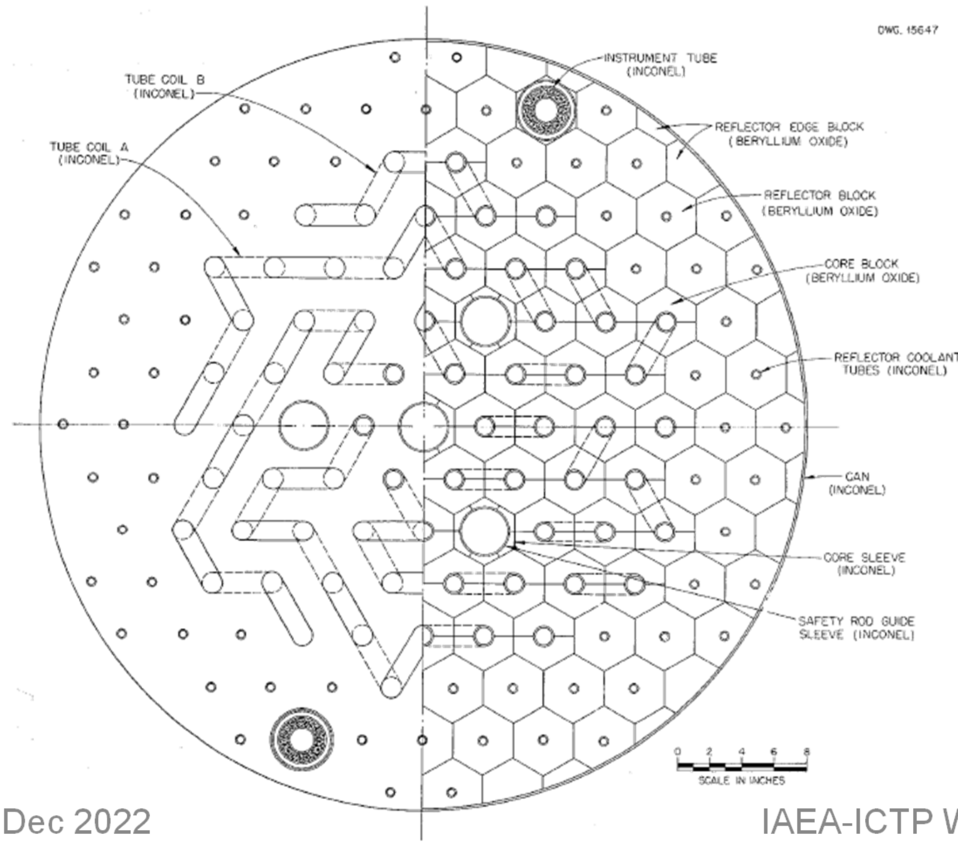
History

- MSRs 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% ZrF_4 – 6% UF_4 (HEU 93.4%)
  - Moderator: BeO, Temperature: 860 °C
  - Power: 2.5 MWth



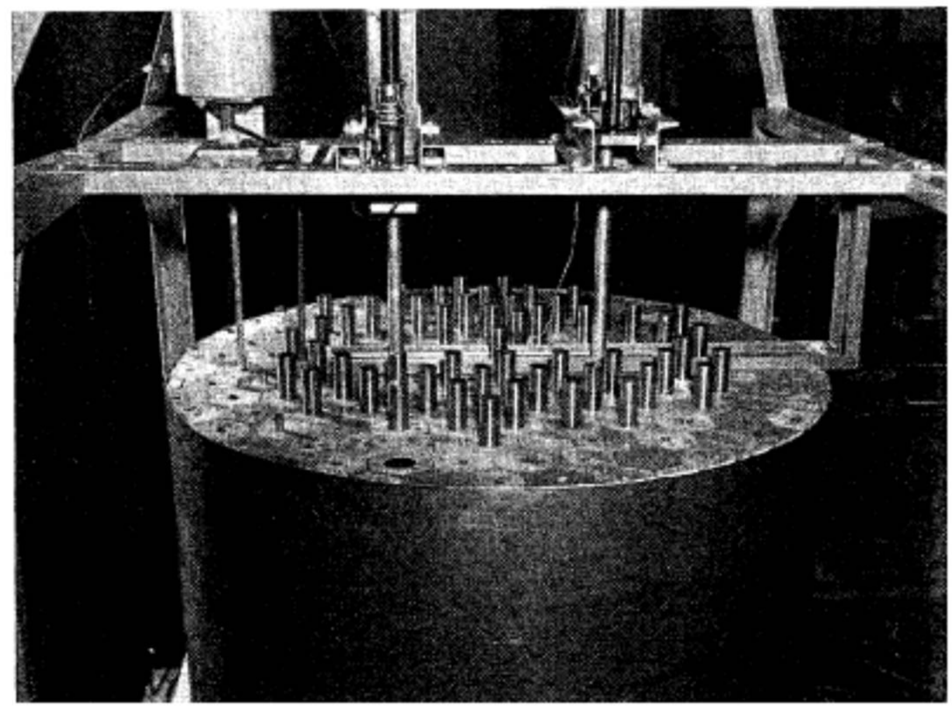


FIGURE 1: Critical Assembly of ARE

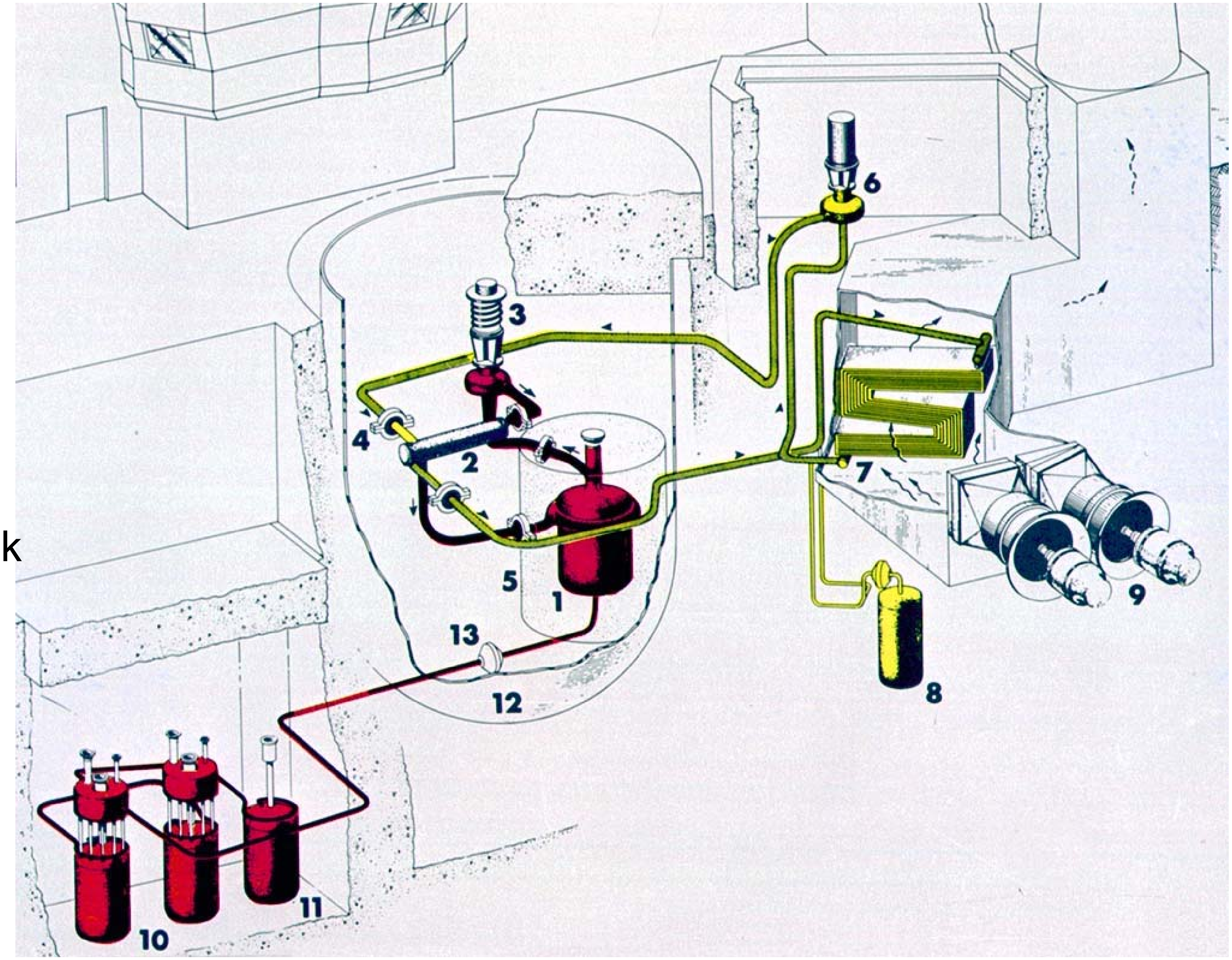
Molten Salt Reactor Experiment

- Operated from 1965 – 1969 (ORNL)
  - Salt: ^{7}LiF - BeF_2 - ZrF_4 - UF_4 (65- 29.1- 5 - 0.9)
  - 33% Enrichment. (^{233}U and ^{239}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

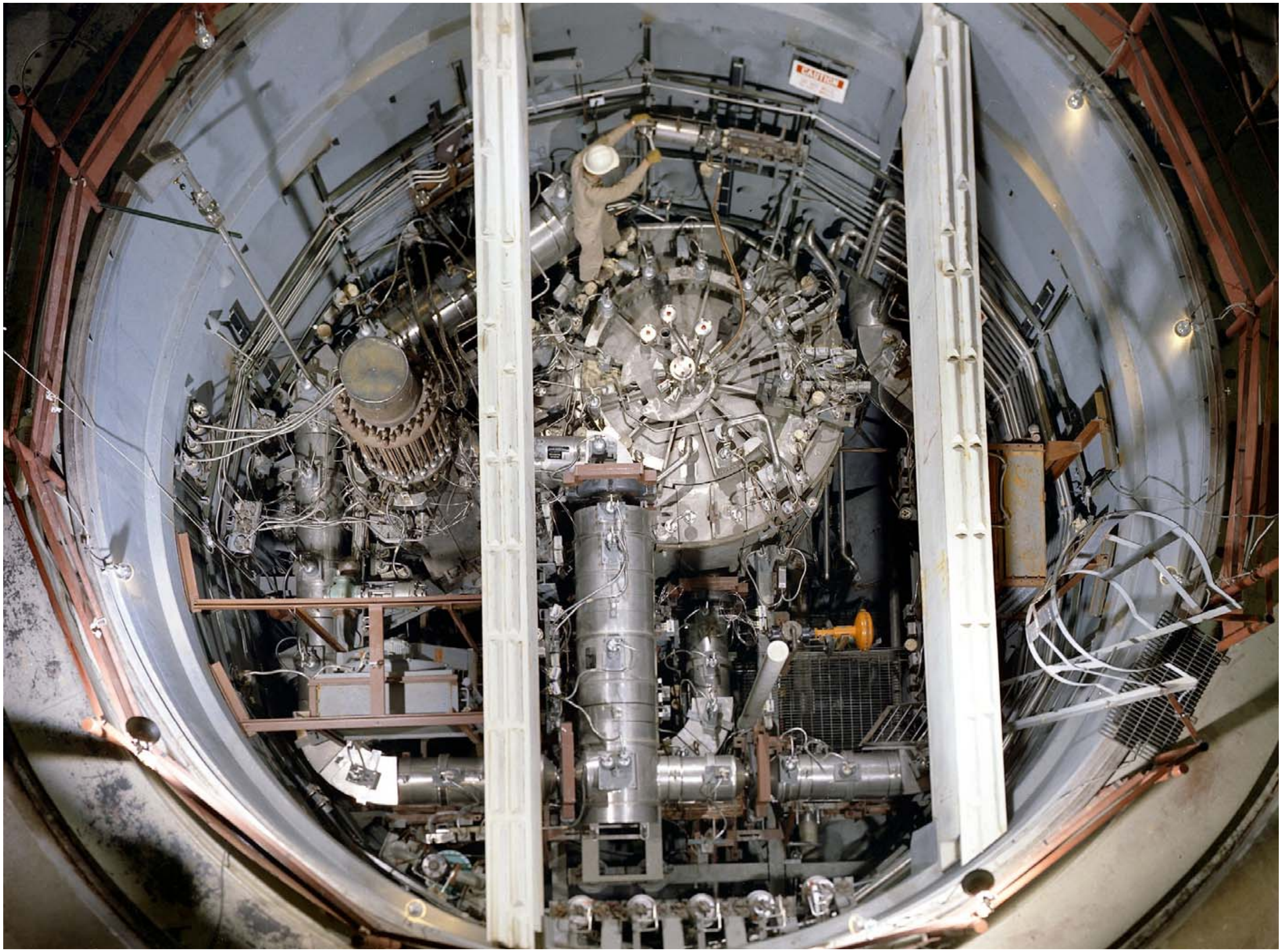
- It was a successful proof of concept

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)	$NaF\text{-}ZrF_4\text{-}UF_4$ (53-41-6)	$^7LiF\text{-}BeF_2\text{-}ZrF_4\text{-}UF_4$ (65-29.1-5-0.9)
Secondary loop	Na	$^7LiF\text{-}BeF_2$

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Basic Equations of Neutronics

- Neutron balance equation: (one group)

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

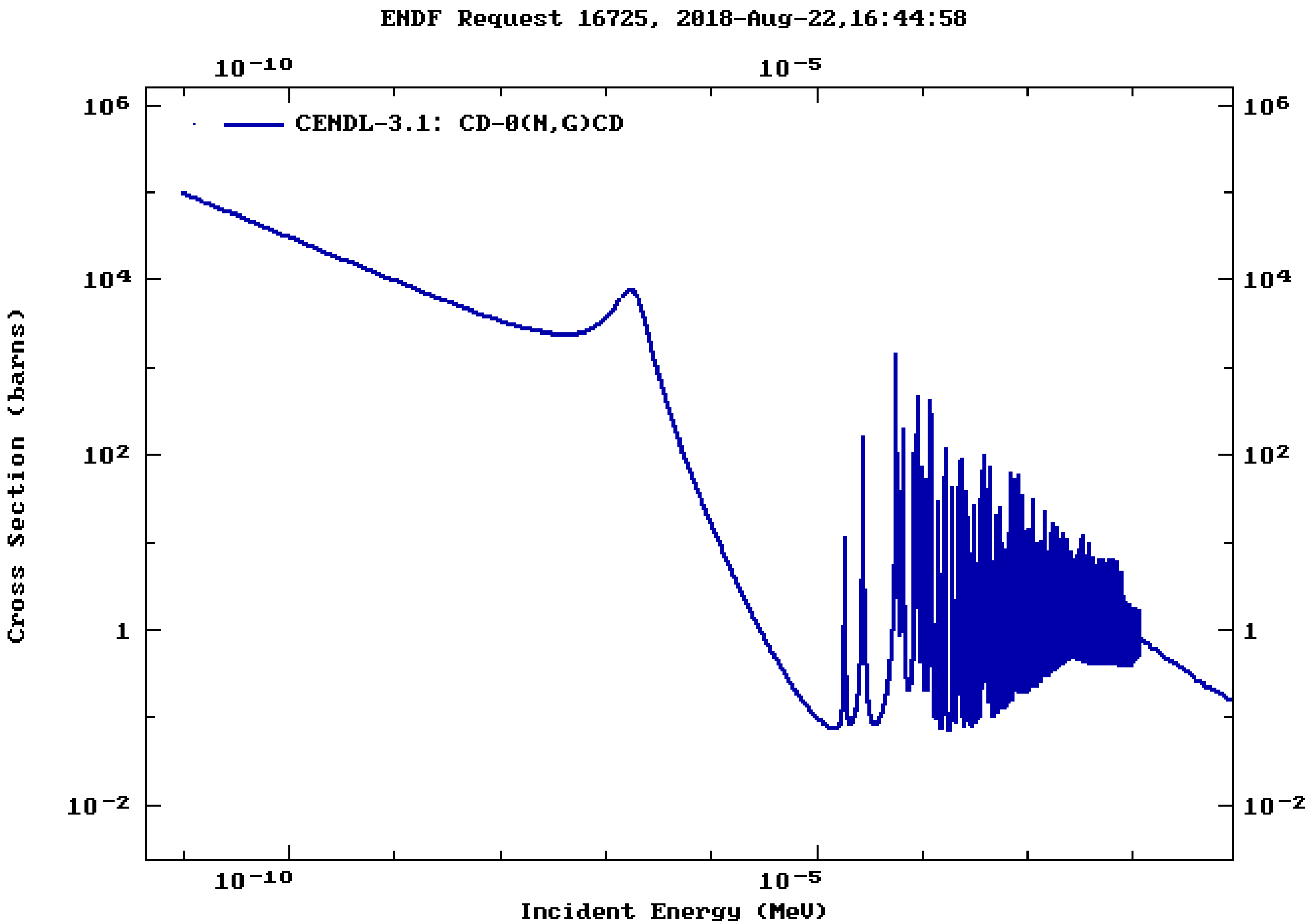
- In equilibrium:

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

- Introduce multiplication factor k_{eff}

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

Cd Absorption X-sec



Example of flux Distribution

ARE:

Axial

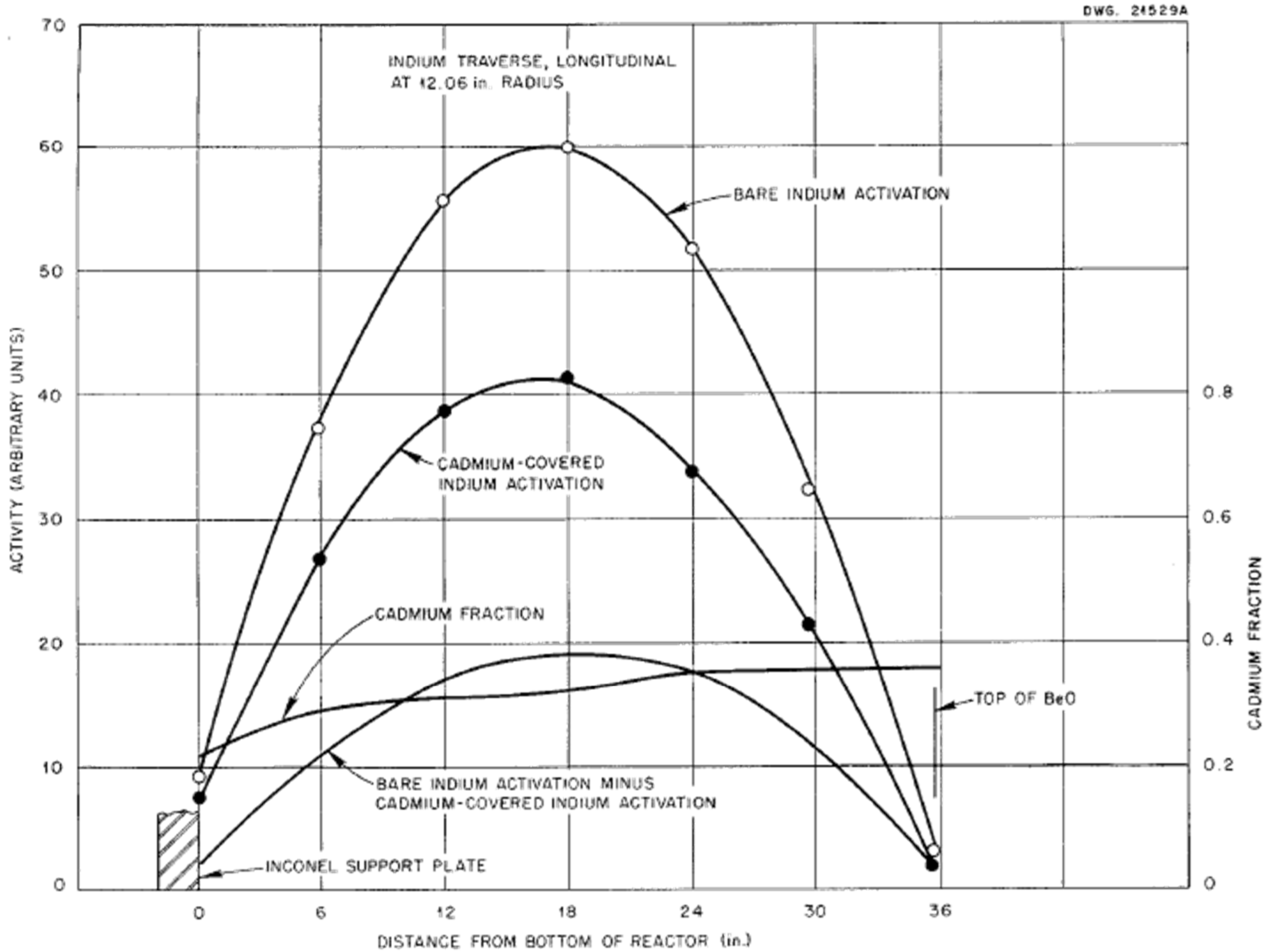


Fig. G.1. Longitudinal Neutron Flux Distribution.

ARE Radial

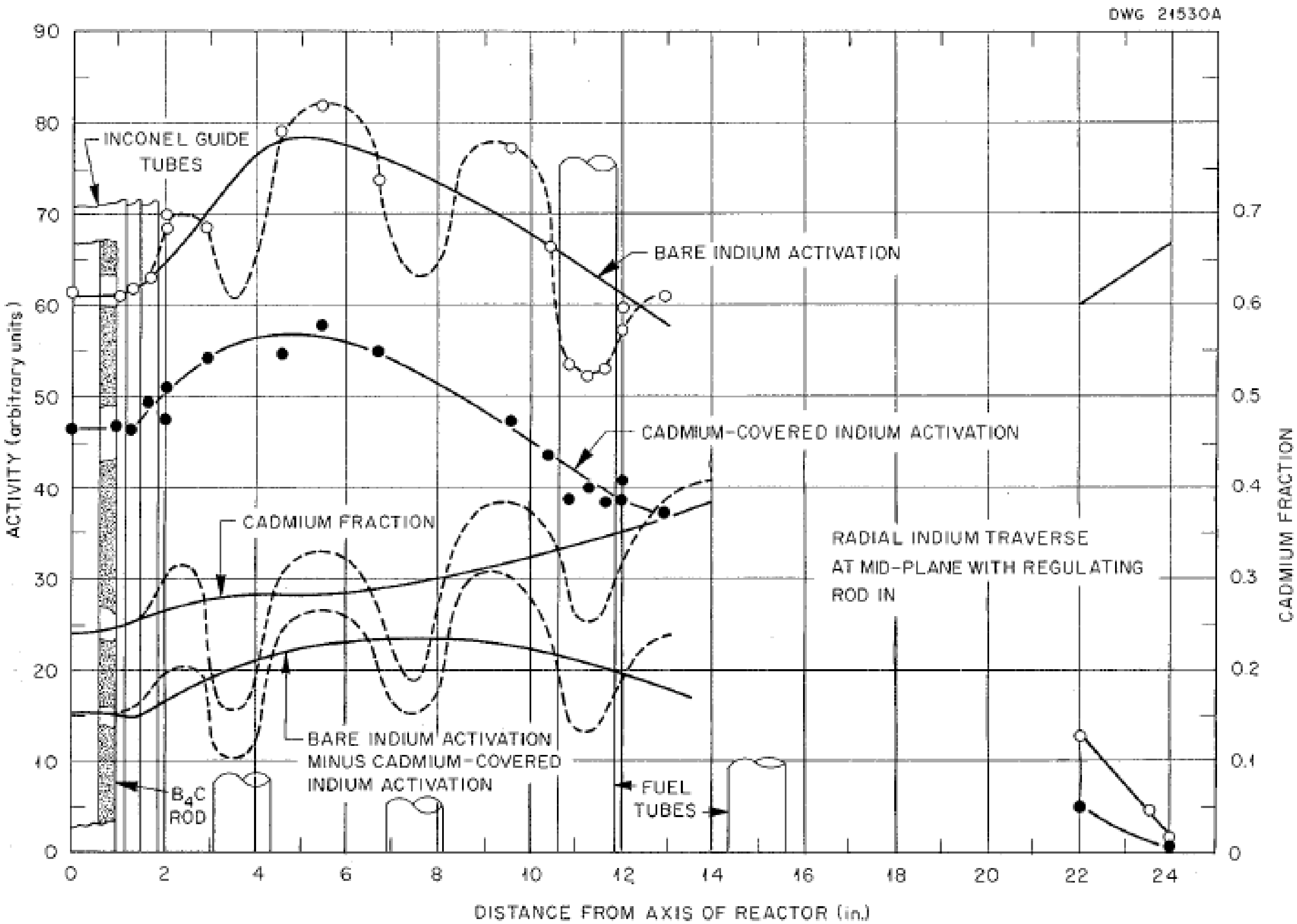


Fig. G.2. Radial Neutron Flux Distribution.

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

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

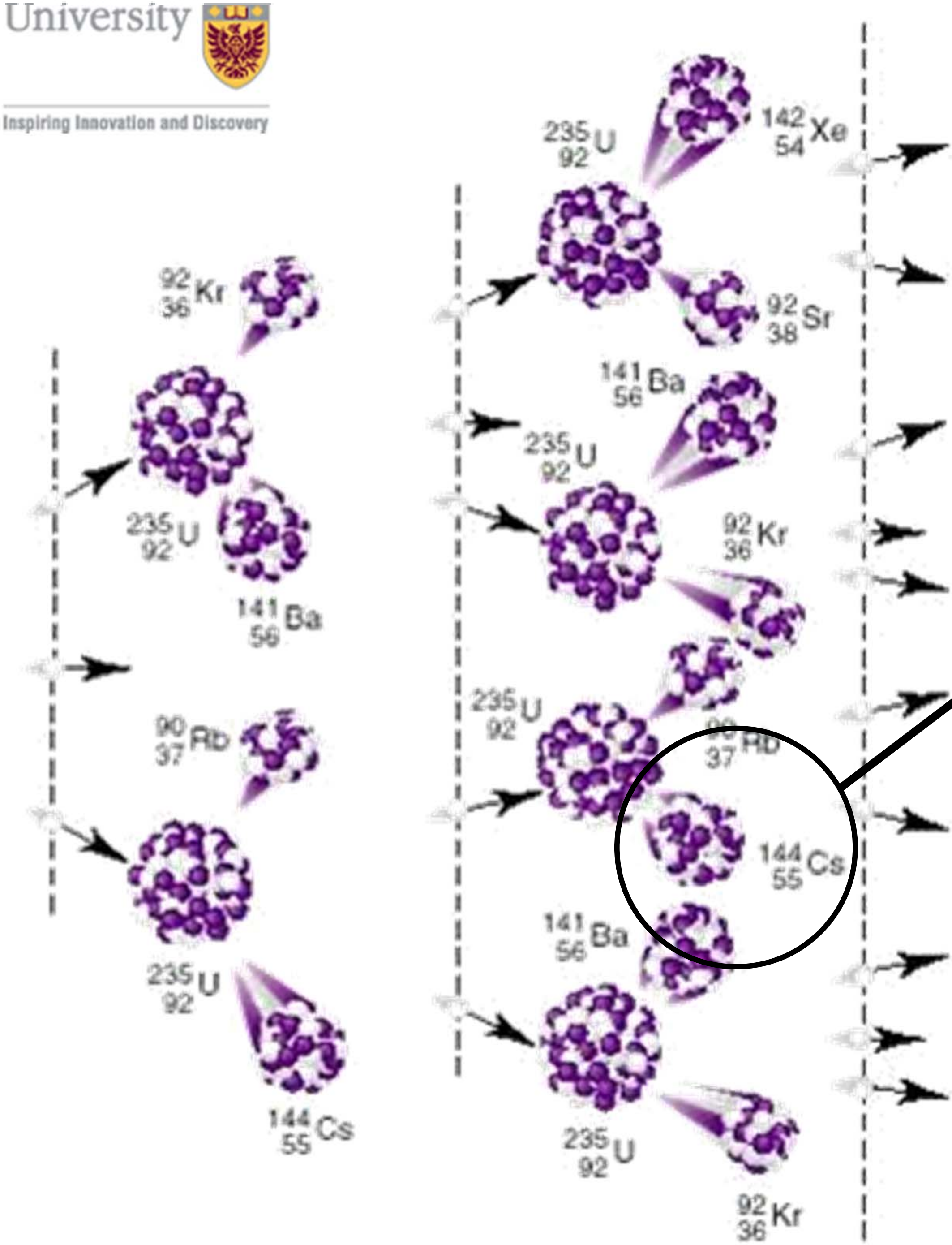
All of this only considers neutrons from fission.

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, → …
- Emitters are called **precursors**
- Emitted neutrons are **delayed neutrons.**

Delayed neutrons



144Ba 11.5 s β-: 100.00%	145Ba 4.31 s β-: 100.00%	146Ba 2.22 s β-: 100.00%
143Cs 1.791 s β-: 100.00% β-n: 1.64%	144Cs 0.994 s β-: 100.00% β-n: 3.03%	145Cs 0.587 s β-: 100.00% β-n: 14.70%
142Xe 1.23 s β-: 100.00% β-n: 0.21%	143Xe 0.511 s β-: 100.00% β-n: 1.00%	144Xe 0.388 s β-: 100.00% β-n: 3.00%
141I	142I	143I

Cs-144 has ~1s half life and emits a neutron in 3% of cases.
There are others.

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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

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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:

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

$$\vdots$$

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

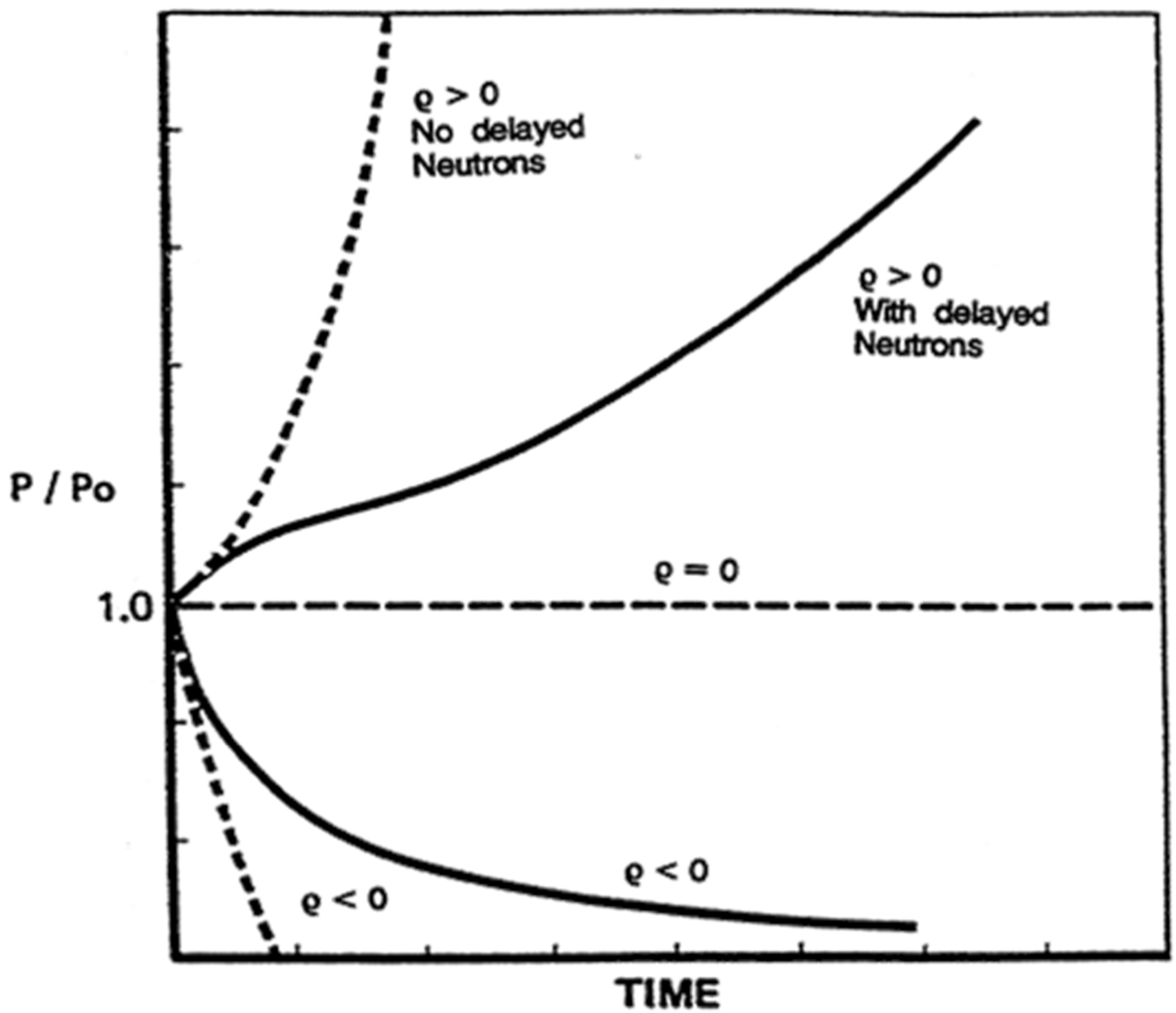
$$\frac{dn(t)}{dt} = \frac{\rho - \beta}{\Lambda} n(t) + \sum_{k=1}^{6} \lambda_k C_k(t)$$

Transient with DN

Reactor becomes more controllable!

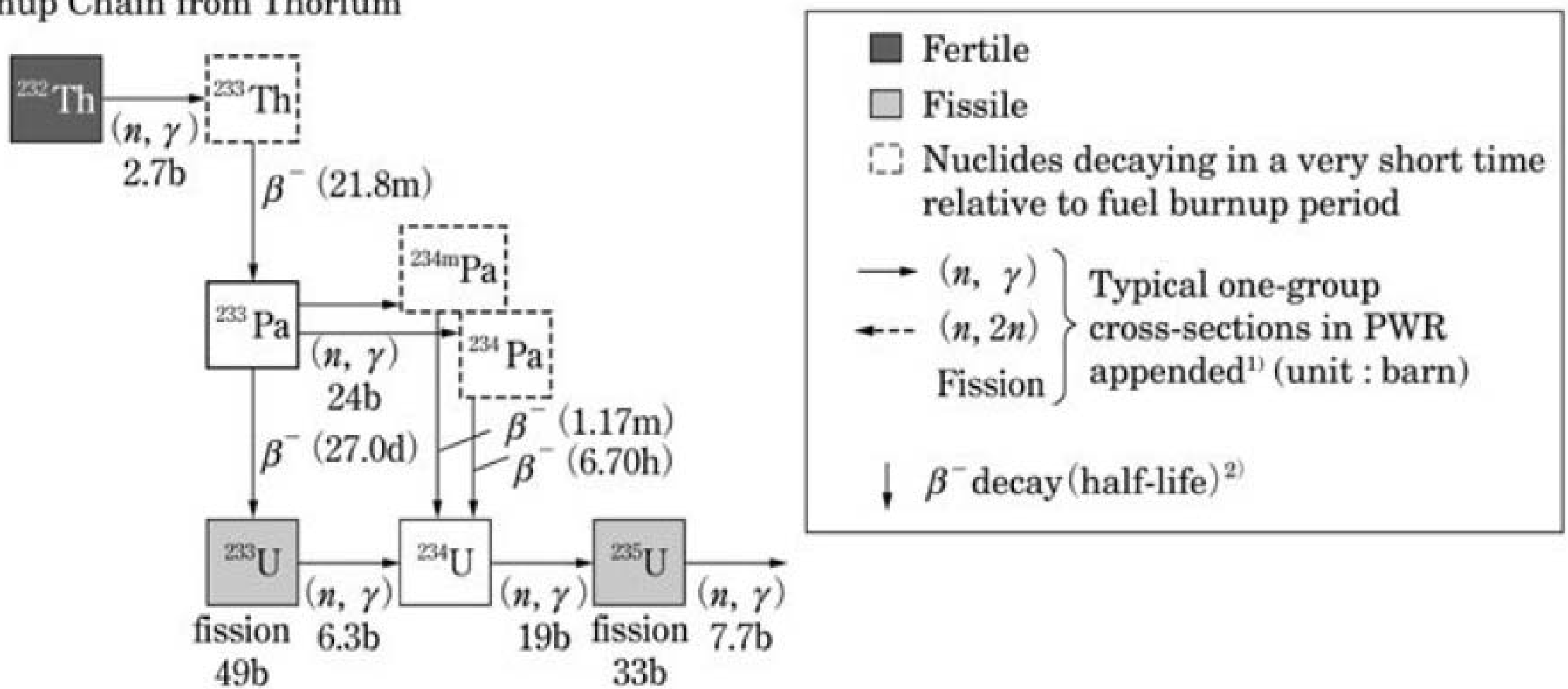
(Daniel Rozon, Introduction to reactor kinetics)

"We can consider a critical reactor to be a subcritical reactor in which the flux ...
is supported by the delayed neutron source."



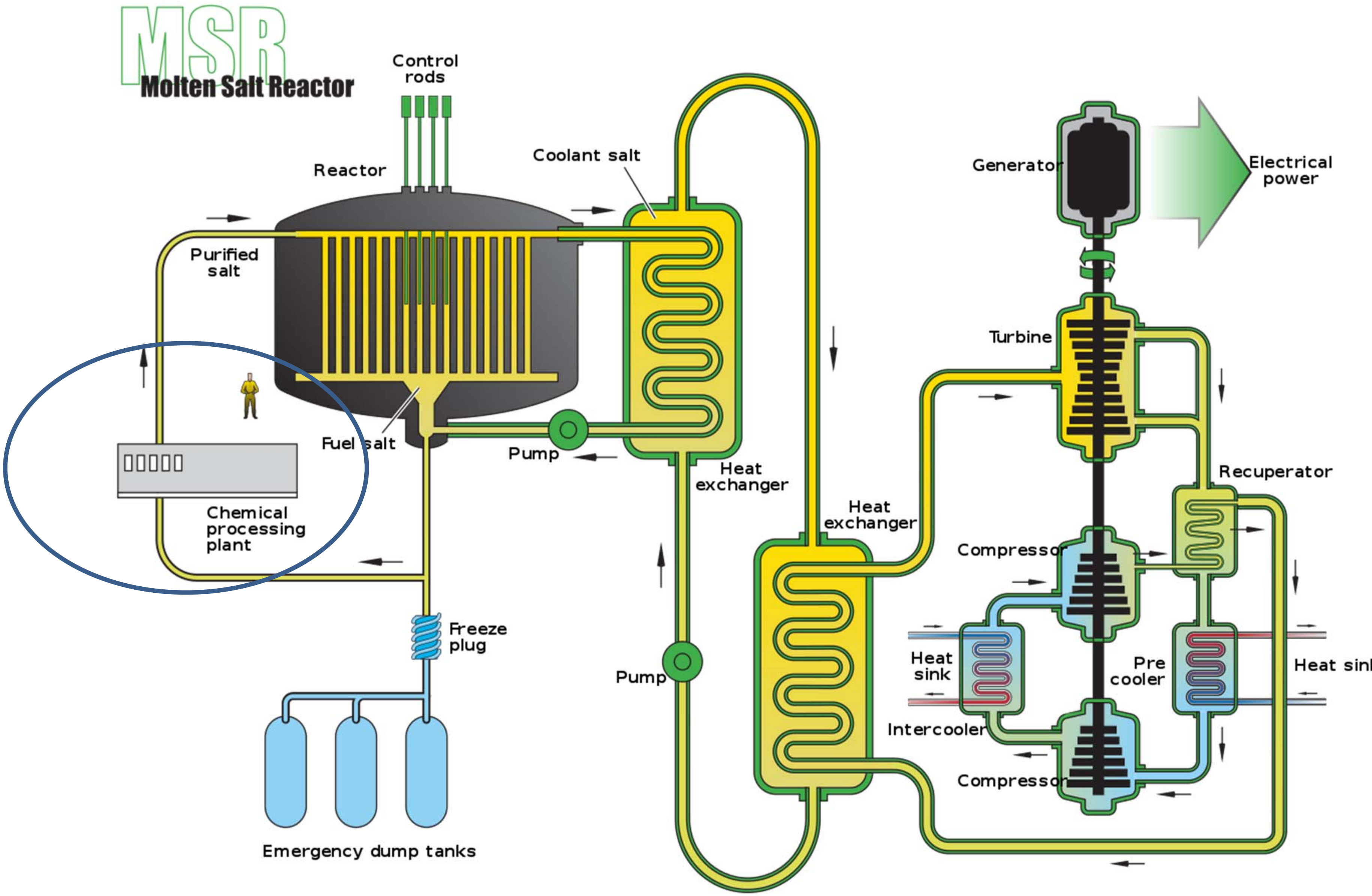
A Word about Breeding

Burnup Chain from Thorium



Source: "Nuclear Reactor Design", Yoshiaki Oka, Editor, Springer Verlag, 2014

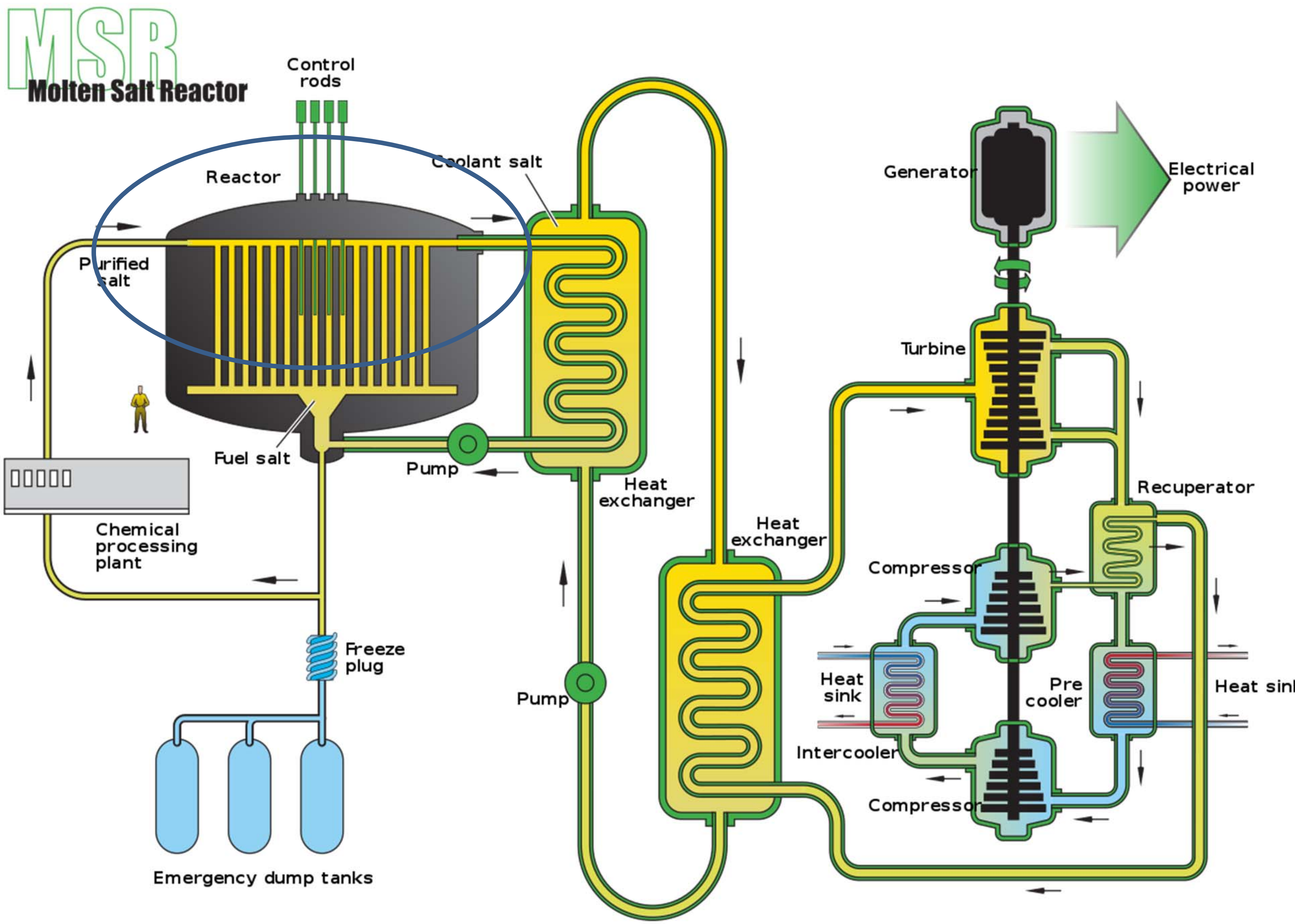
Apply to MSRs



Chemical Processing Plant

- Remove fission products
  - One of the main design features of original ORNL design.
  - Not considered in modern designs.
- 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.

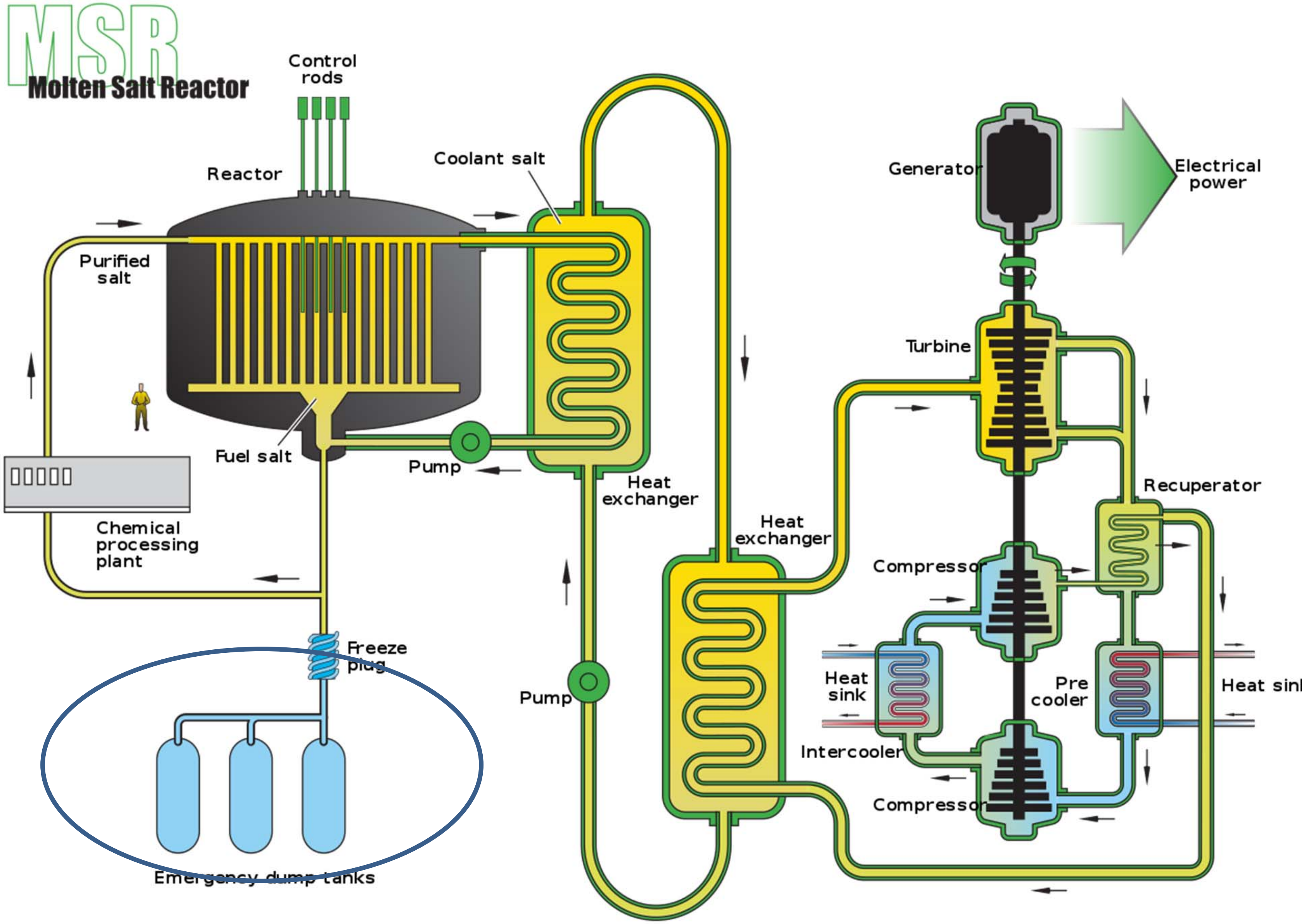
Apply to MSRs



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 MSRs



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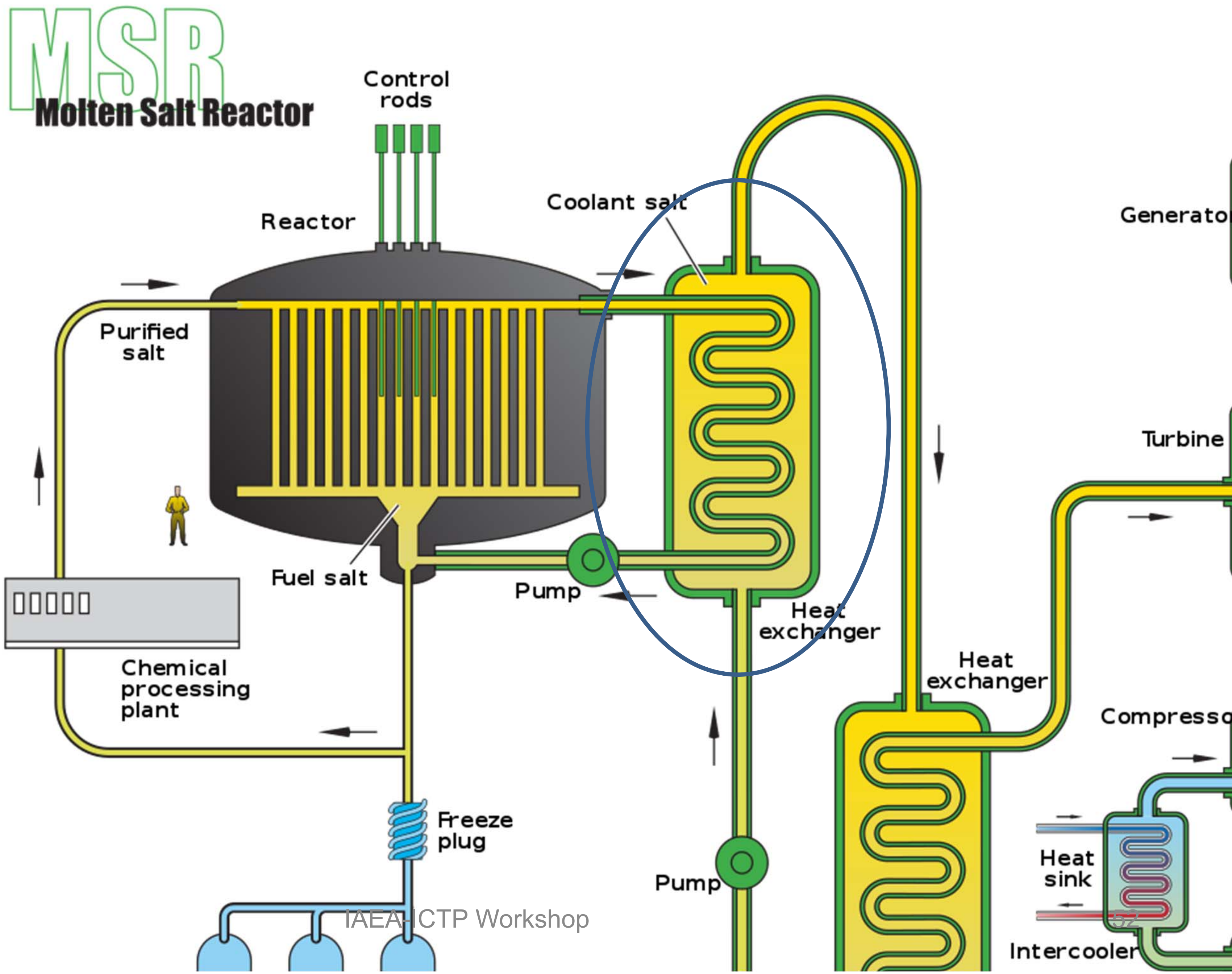
Dump Tanks

- 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 MSRs



H

H

Primary Circuit Outside Core

- Good for letting Pa decay
- Ratio: $\mathrm{R} = \dfrac{\text{time in core}}{\text{time out of core}}$ 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:

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

$$\vdots$$

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

Now (group k=1 only):

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

$$\frac{dC_1(z,t)}{dt} = \frac{\partial C_1(z,t)}{\partial t} + u_z \frac{\partial C_1(z,t)}{\partial z}$$

$$= \frac{\beta_1}{\Lambda} n(z,t) - \lambda_1 C_1(z,t)$$

Point Kinetics cont'ed

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

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

Bad news:

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

MSRE: Zero-Power Exp.

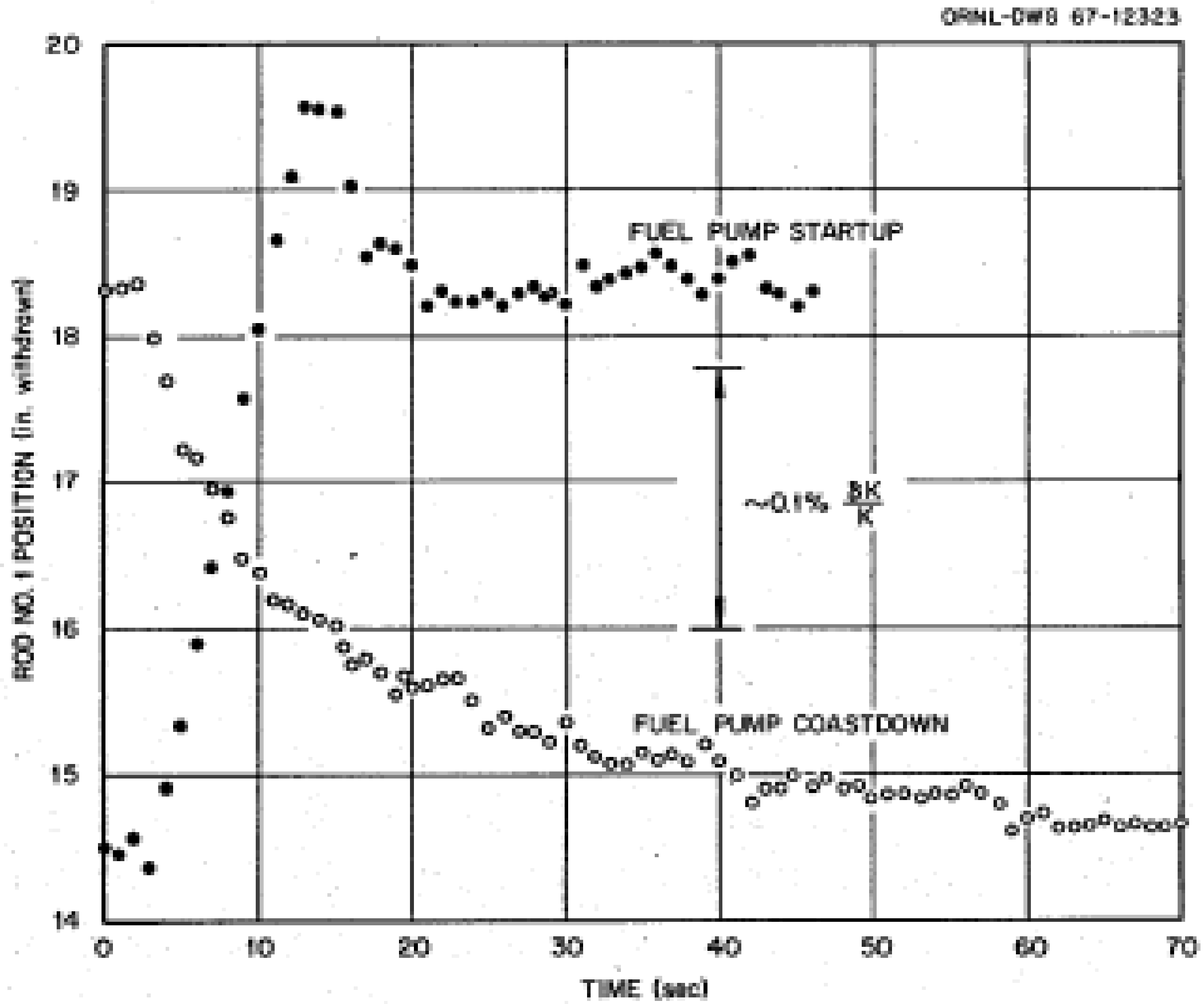
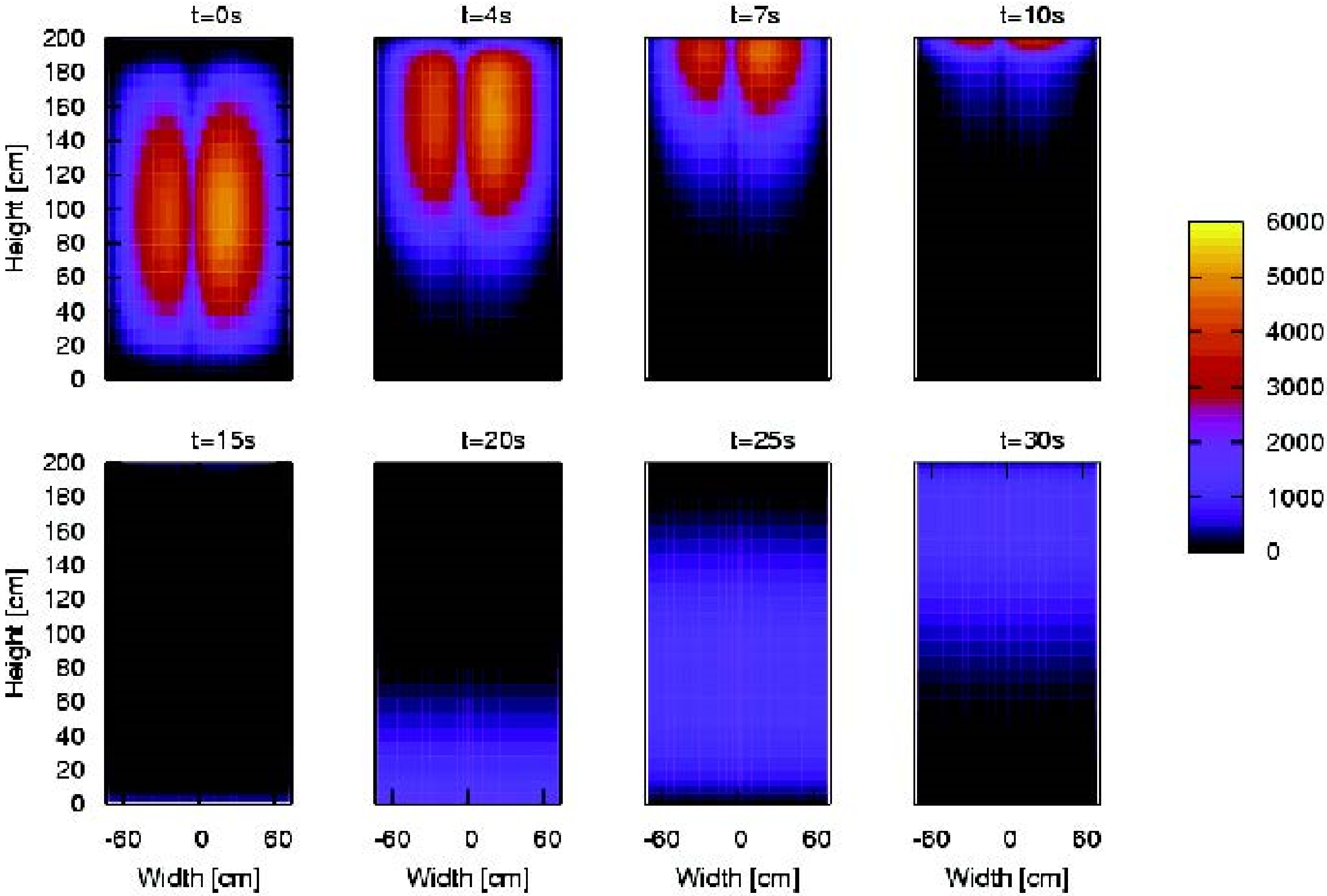


Fig. 24. Control-Rod Response to Fuel-Pump Startup and Coastdown.

MSRE Calculation



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

Doppler Broadening

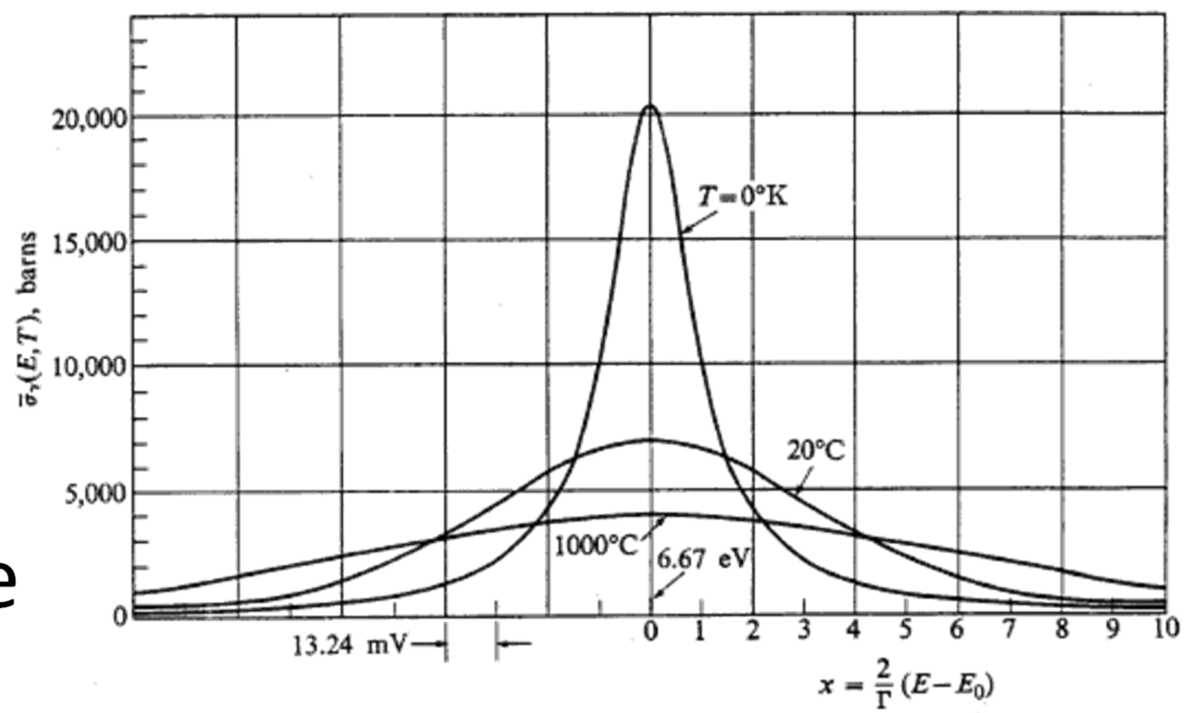
- Remember definition of reactivity:

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

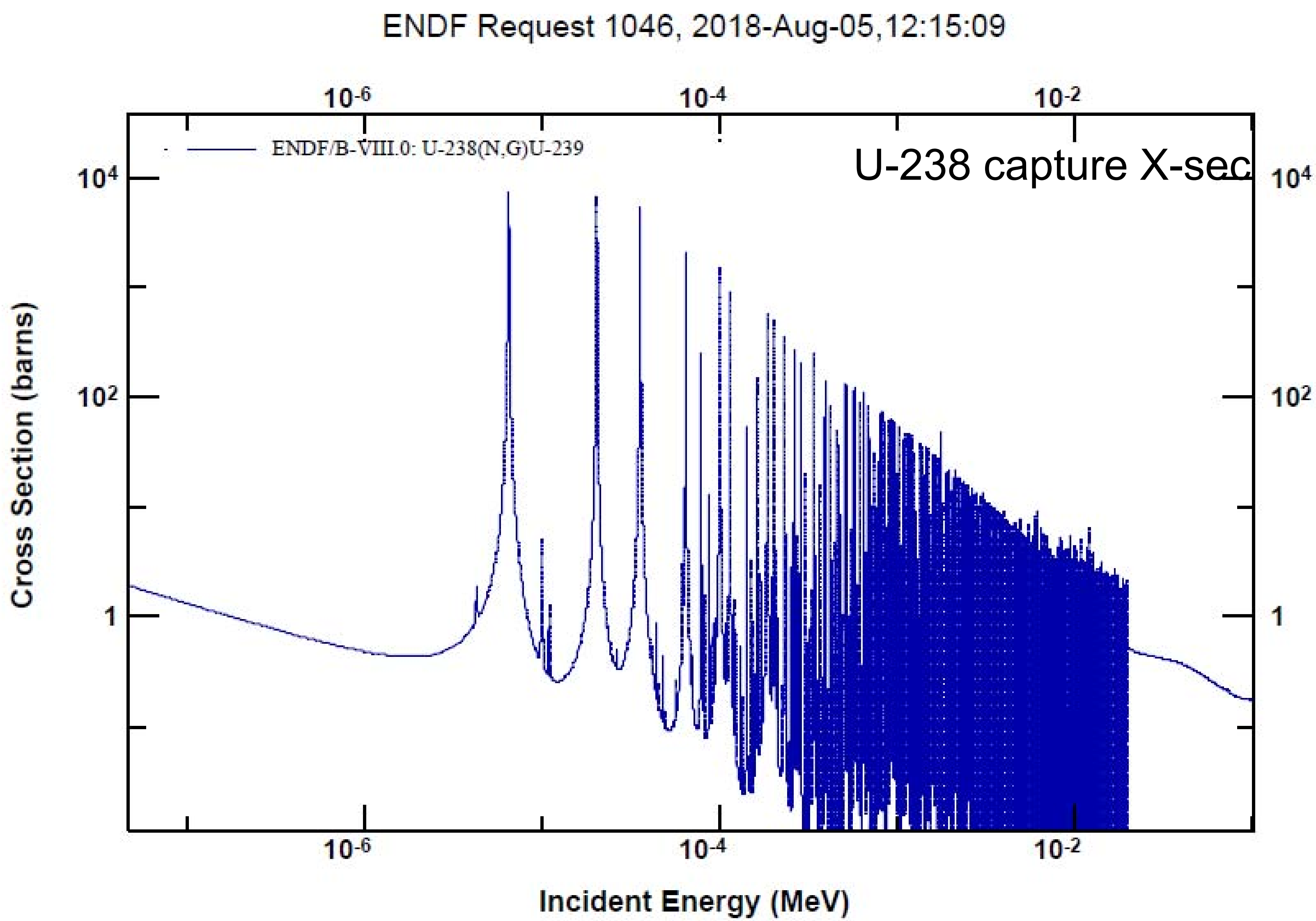
If loss rate increases, reactivity decreases.

- Doppler broadening:
- Leads to higher absorption.

(Story is a bit more complicated)



Absorption


ENDF Request 1046, 2018-Aug-05,12:15:09
ENDF/B-VIII.0: U-238(N,G)U-239
U-238 capture X-sec
Cross Section (barns)
Incident Energy (MeV)
10^{-6}
10^{-4}
10^{-2}
10^{4}
10^{2}
1

Reactivity Coefficient

- Reactivity coefficient is defined as

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

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

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	$\frac{\delta k/k}{°F}$ $(\times 10^5)$				
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.
[b] Uranium of the isotopic composition of the material added during the critical experiment (91% ^{233}U).
[c] Measurement obscured by effect of circulating voids.
[d] Normal full travel of rod(s).
[e] Not separately evaluated.

Simulating MSRs

- Static (design calculations):
  - Neutronics code; most are satisfactory:
    - MCNP
    - SCALE suite
    - Serpent
    - OpenMC
    - ….
  - Depletion code:
    - Serpent
    - TRITON (SCALE)
    - OpenMC
    - ….

Simulating MSRs

- 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

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Simulating MSRs

- 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)
  - Stable Salt Reactors by Moltex Energy (UK, CA)

Summary of Commercial Ideas

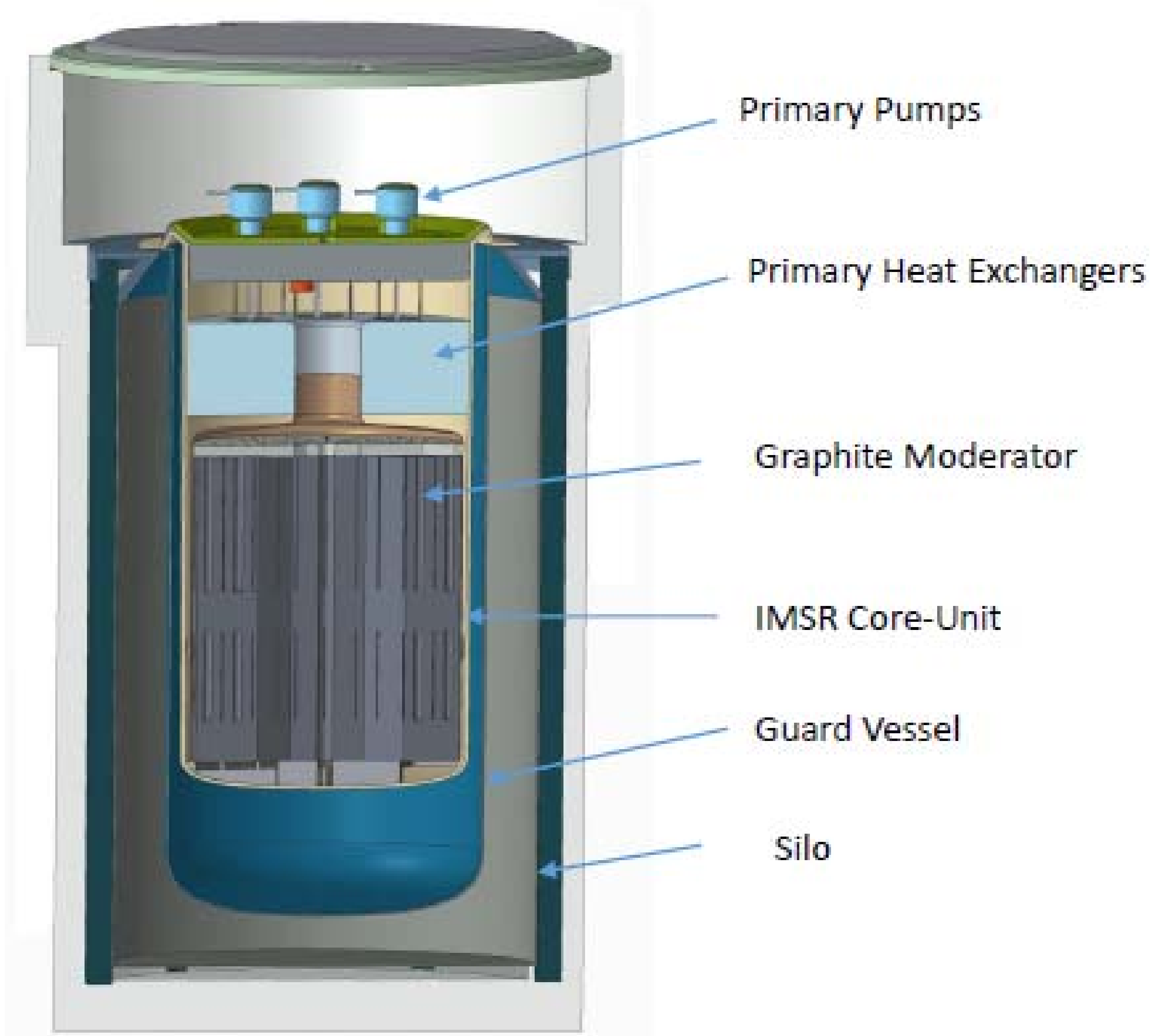
Vendor	Name	MWth	MWe	Mod	2nd Circ	Salt	Fuel	Breeding
Terrestrial Energy	IMSR-400	400	192	Graphite	Salt	FLi (no Be)	U/Th	possible
Flibe Energy	LFTR	600	250	Graphite	CO2	FLiBe	U/Th	Blanket
Terrapower	MCFR							
Transatomic Power		1250	520	ZrH		FLi (no Be)	U	
ThorCon Power			250	Graphite		NaBe/FLiBe	U/Th	in core
Copenhagen Atomics	Thorium MSR	Peripheral research						
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)

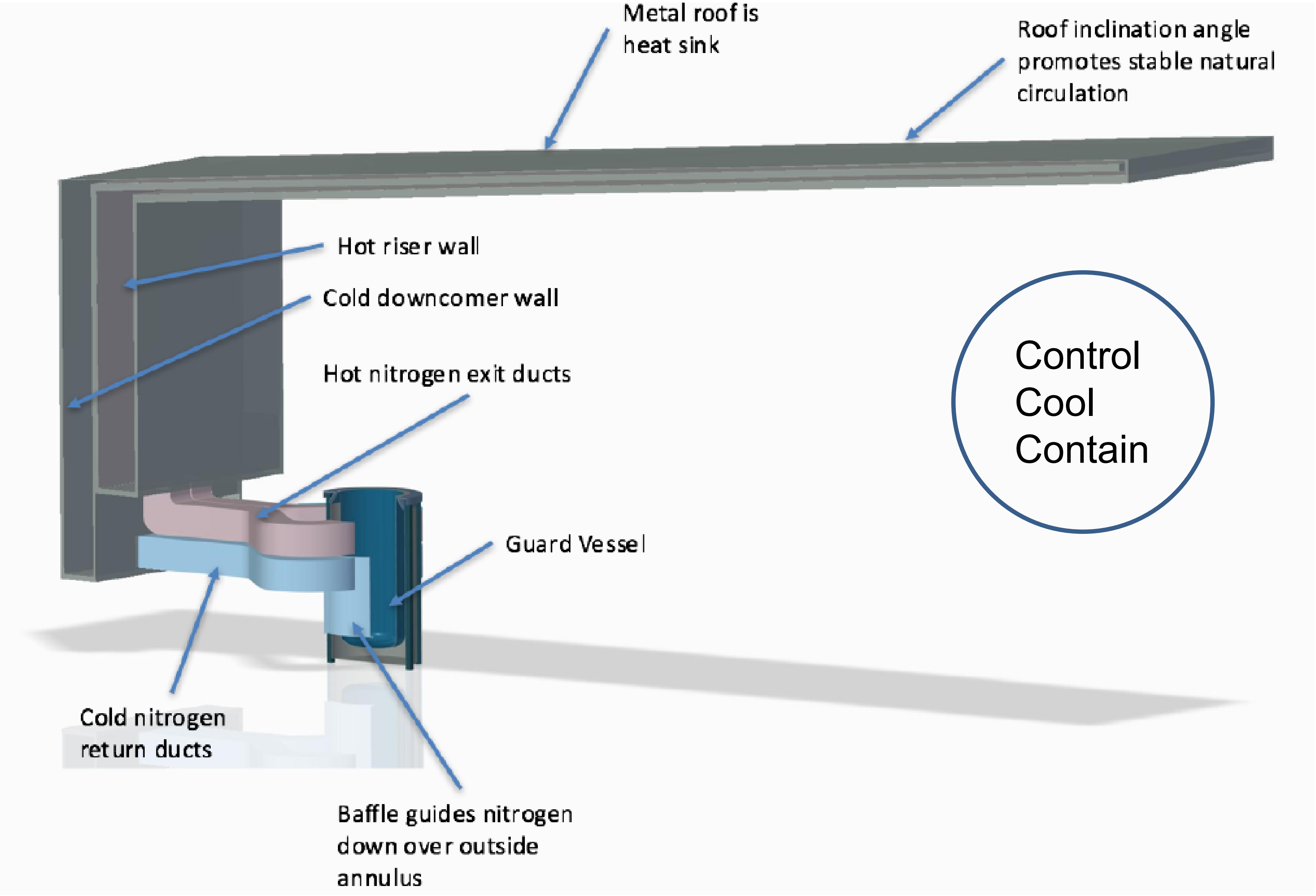
IMSR-400 Lay-out

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



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

IMSR-400 Passive cooling



Control
Cool
Contain

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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-BeF2-UF4-ThF4(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

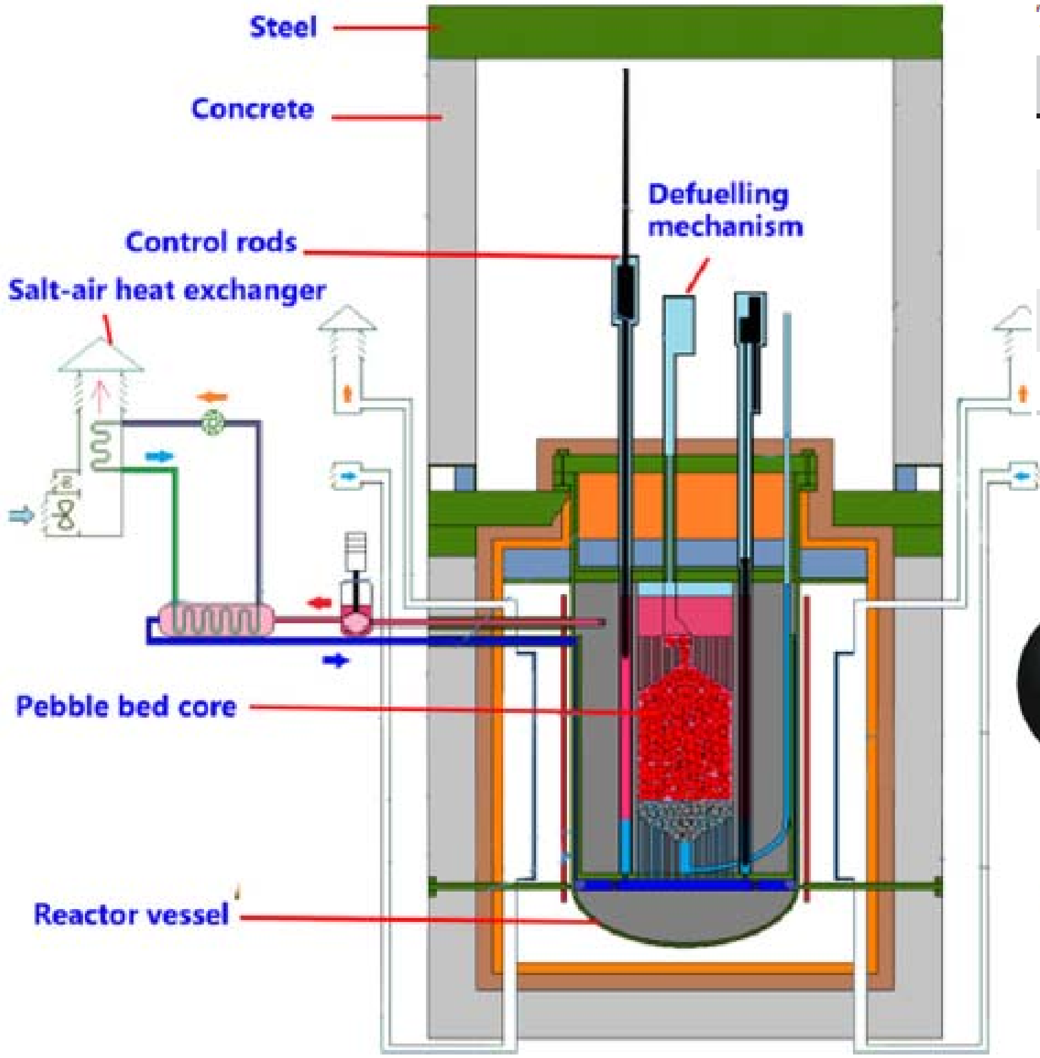
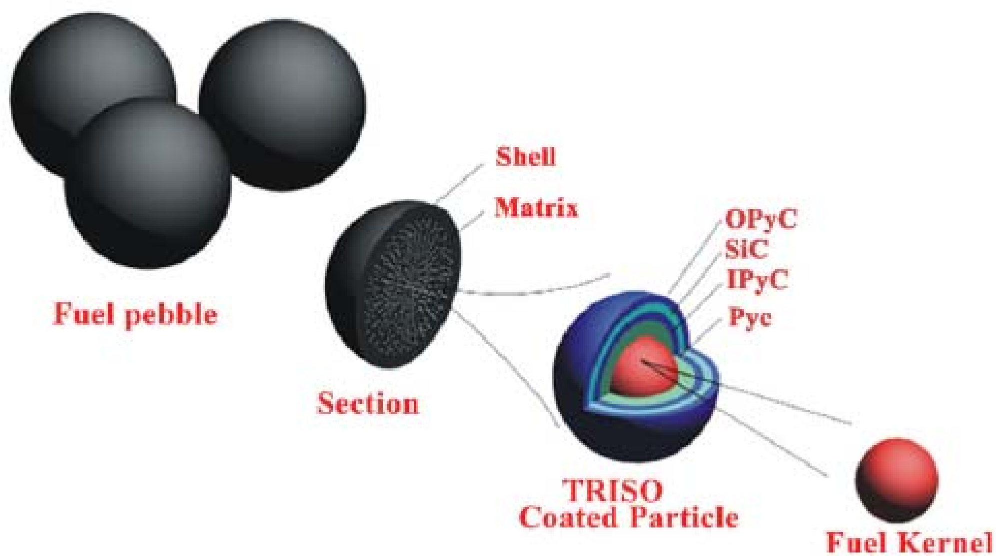


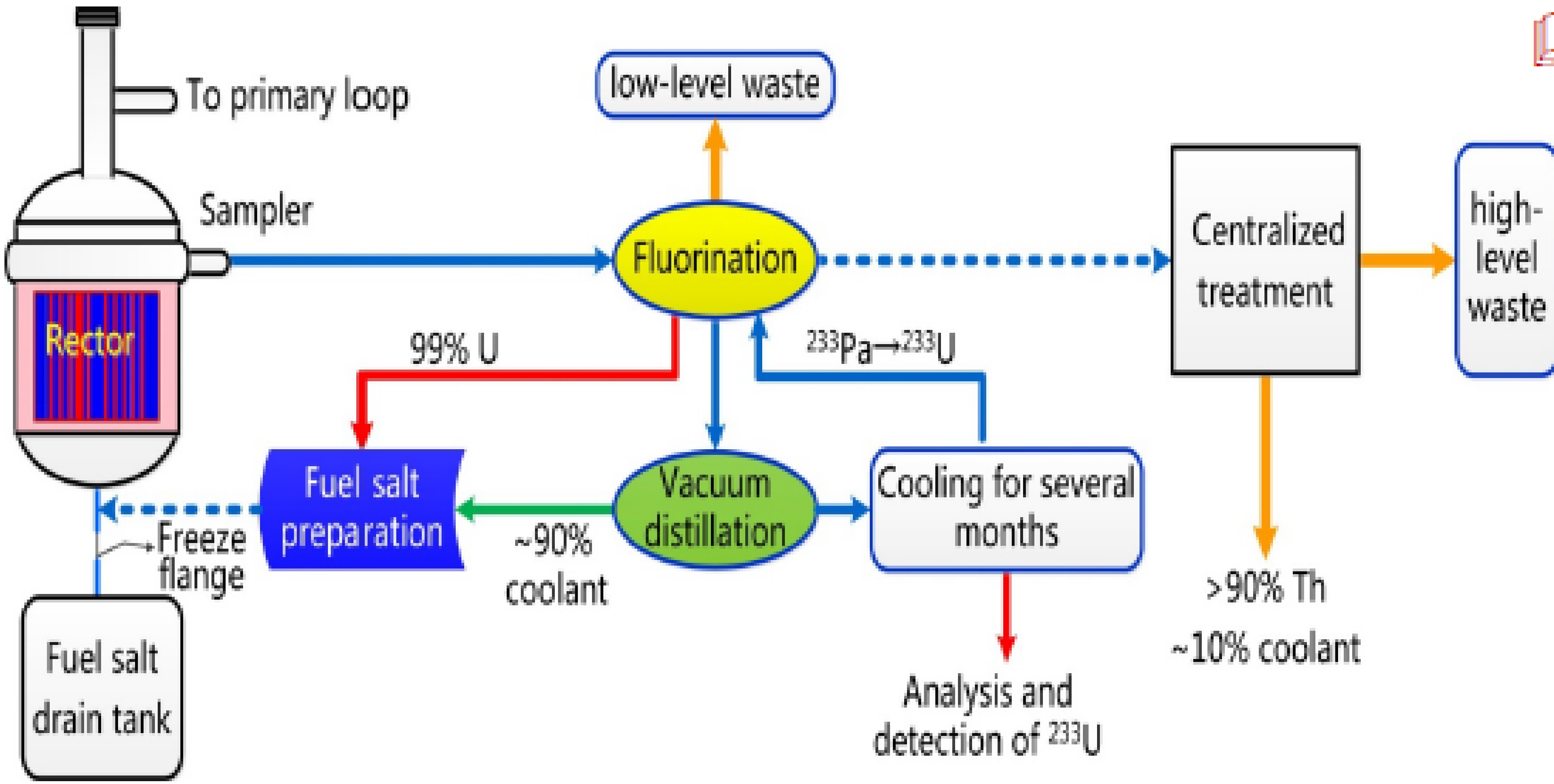
TABLE 2 Operation parameters of 2 MW TMSR-SF and 10 MW TMSR-SF

	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



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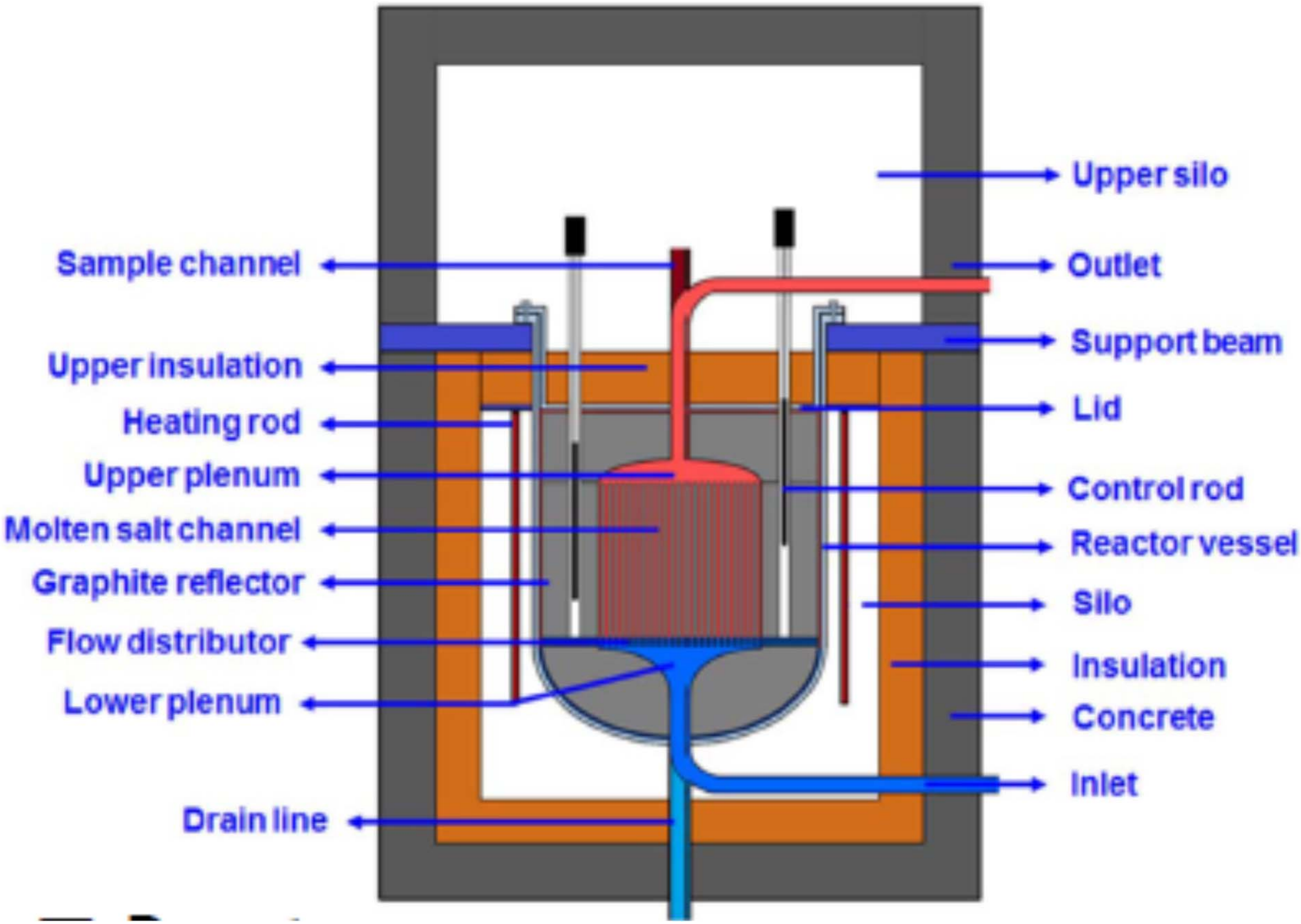
TMSR-LF1 Layout



TMSR-LF Design Parameters

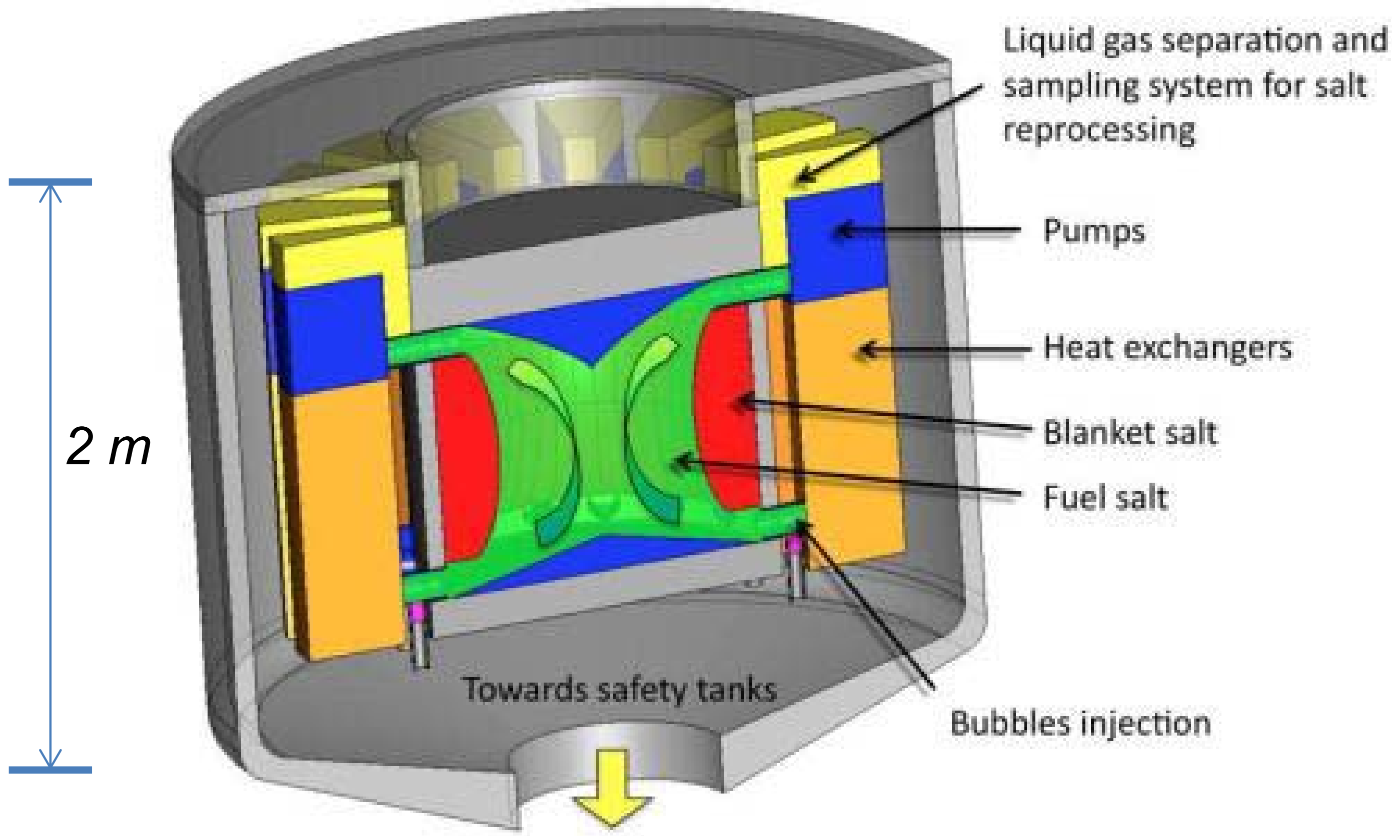
Parameter	Value
Thermal Power	395 MW
Electrical Power	168 MW
Inlet Temperature	600 °C
Outlet Temperature	700 °C
Fuel Salt (primary loop)	$LiF-BeF_2-UF_4-ThF_4$
Cooling Salt (secondary loop)	FNaBe
Moderator	Graphite
Reactor vessel size (D × H)	5.3 m × 6.0 m

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

MSFR (CNRS, EVOL, SAMOFAR)

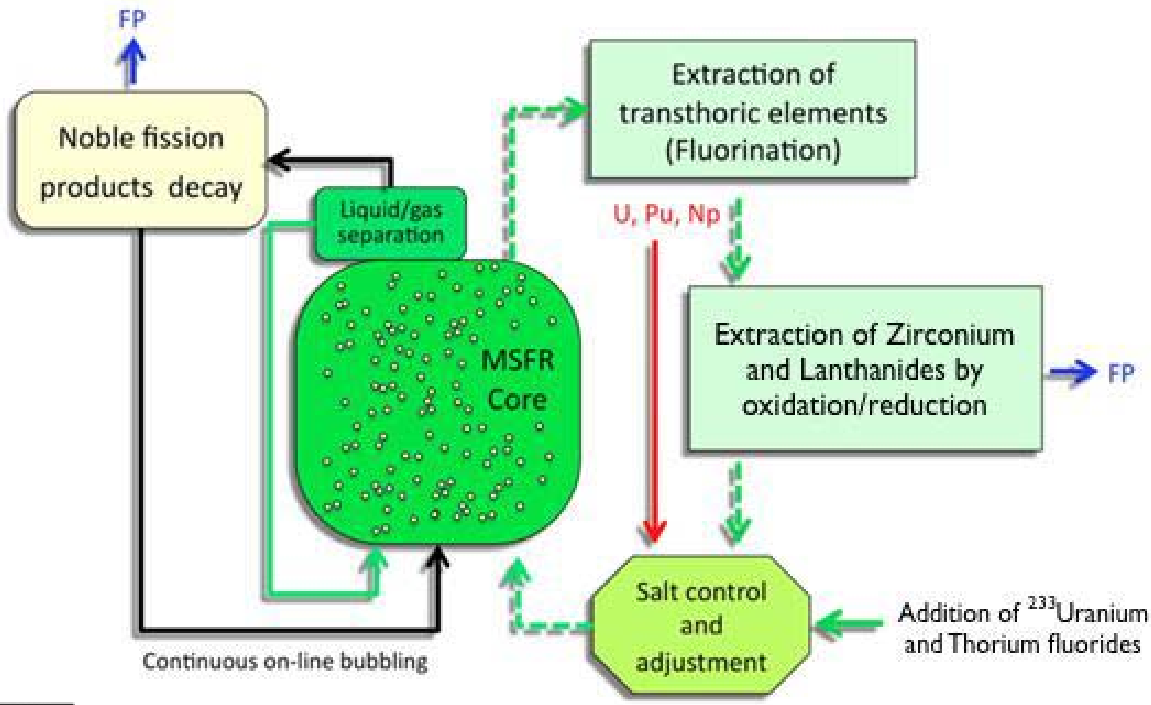
- GEN IV reference reactor
- Fast spectrum, no moderator.

- Large:
  - 3000 MWth
  - 18 m^3 fuel salt
  - 50/50, 3.9 s
- Fuel:
  - LiF-HMF_x (77.5-22.5)
  - 750 °C
- Blanket:
  - LiF-ThF_4 (77.5-22.5)
- No control rods
  - Temp FB -8 pcm/K

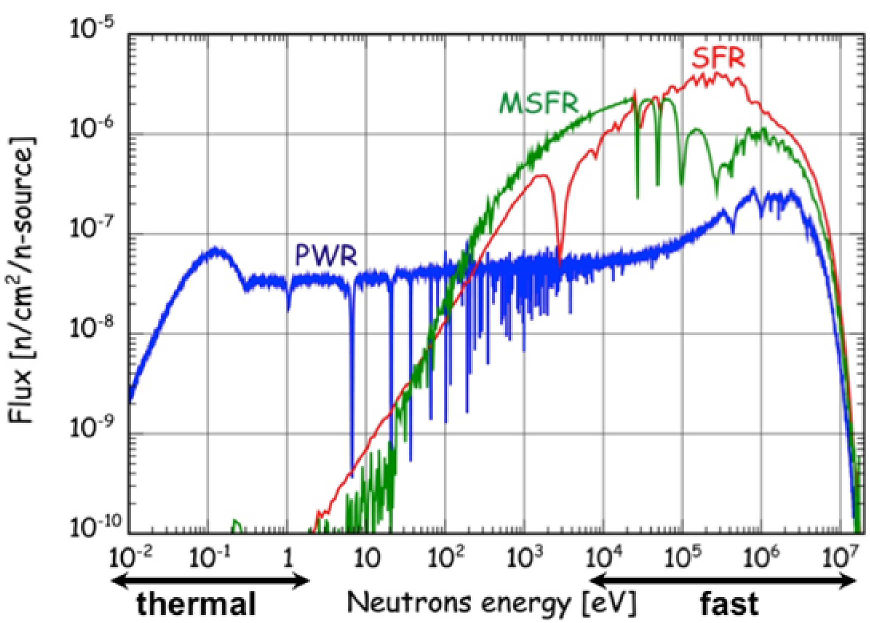


MSFR

- 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

From Kloosterman 2017

Conclusion

- MSRs 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
- MSRs are a safe, reliable and sustainable source of low-carbon electricity.

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More about Canada

- Molten Salt Reactor is indigenous design.
- Many foreign designers apply for licences in Canada.
- Canadian licensing regime is somewhat different: onus is on vendor to prove that reactor is safe.
- More about SMRs in Canada:

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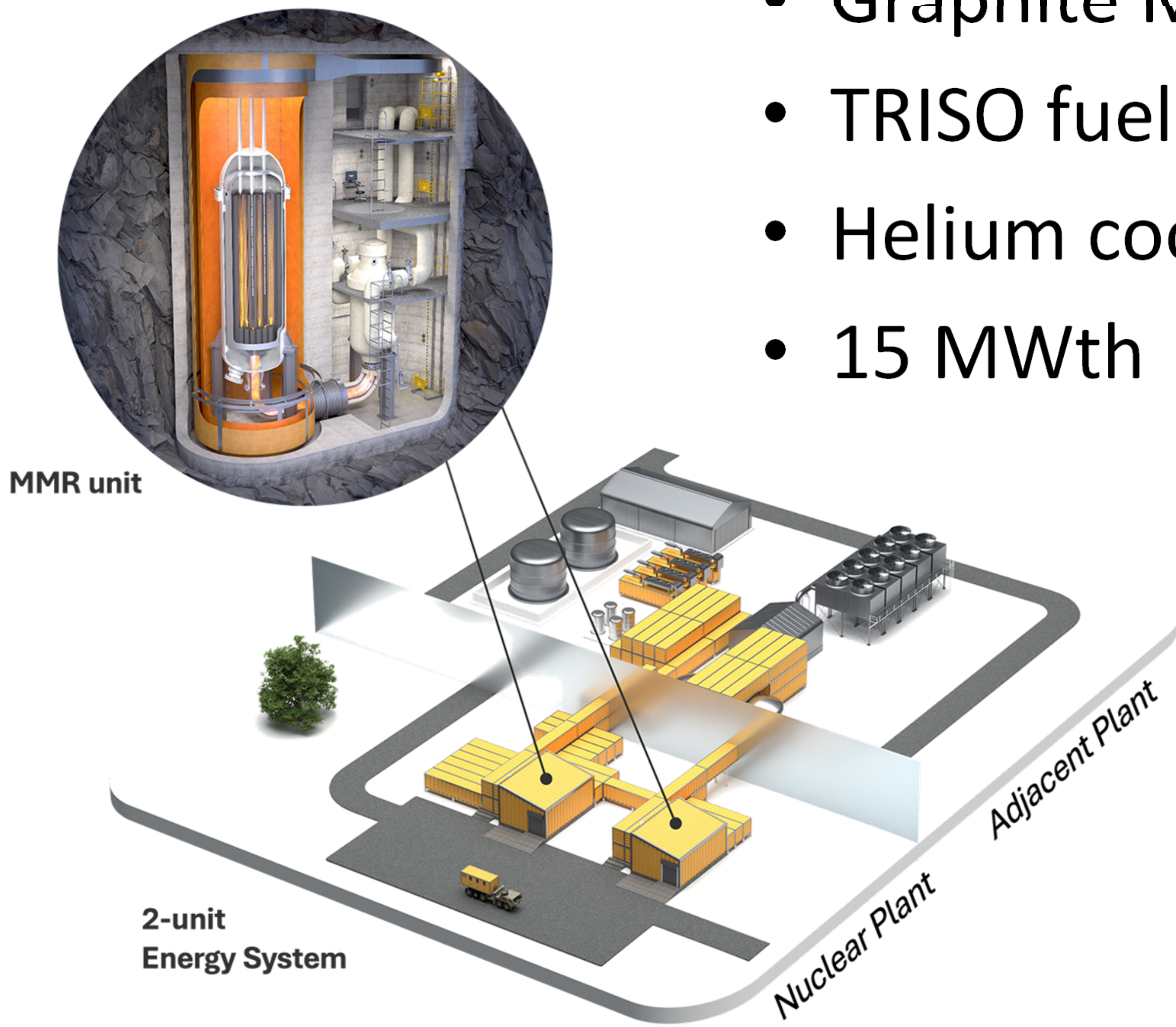
wnn daily
world nuclear news
Celebrating 15 years

Dec 12, 2022

McMaster University, Ultra Safe Nuclear Corporation and Global First Power have signed a Memorandum of Understanding to further examine the feasibility of deploying a Micro Modular Reactor at McMaster University or an affiliated site, following on from their announcement of their partnership in May this year. The university is to conduct an 18-month SMR feasibility study before deciding whether to pursue SMR deployment and beginning the process to obtain the necessary licences from the Canadian Nuclear Safety Commission

Micro Modular Reactor

- Graphite Moderated
- TRISO fuel in SiC matrix
- Helium cooled
- 15 MWth



Proven technology

Also

- December 2, 2022
- https://toronto.ctvnews.ca/ford-marks-start-of-construction-on-canada-s-first-grid-scale-nuclear-reactor-1.6177973
- Ontario Premier Doug Ford was in Clarington, Ont., Friday to mark the beginning of site preparation for Canada's first grid-scale small modular reactor (SMR) at the Darlington nuclear site.
- Chosen design: GE-Hitachi BWRX-300

GEH BWRX-300

- MOU signed with SaskPower as well.
- In addition to Canada, memoranda of understanding or contracts have been signed with companies in the United States, Poland, Sweden, Estonia and the Czech Republic to explore deployment of the technology.

Message:
proven technology