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Neutronics of MSR

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MSR Summer School

July 2-4, 2017, Lecco (Como Lake), Italy

SAMOFAR

Programme of the school

Monday July 3, 2017

09:00-09:45	MSR Concepts	Ondrej Benes (JRC Karlsruhe)
09:45-10:30	Neutronics of MSR	Sandra Dulla (POLITO)
10:30-11:00	*Coffee break*	
11:00-12:30	Integral Safety Analysis	Elsa Merle (CNRS)
12:30-14:00	*Lunch break*	
14:00-14:45	Fuel cycle aspects of MSR	Jiri Krepel (PSI)
14:45-15:30	Thermal-hydraulics and CFD	Pablo Rubiolo (CNRS)
15:30-16:00	*Coffee break*	
16:00-16:45	Multiphysics simulation of MSR	Danny Lathouwers (TU Delft)
16:45-17:30	Control Strategies of MSR	Stefano Lorenzi (POLIMI)

Tuesday July 4, 2017

09:00-10:30	Kinetics and dynamics (incl noise analysis) of MSR	Imre Pazsit (Chalmers Univ)
10:30-11:00	*Coffee break*	
11:00-12:30	Thermodynamics analysis of salts Physico-Chemical properties of salts	Ondrej Benes (ITU)
12:30-14:00	*Lunch break*	
14:00-15:30	Materials and metals in MSR	Victor Ignatiev (Kurchatov Inst)
15:30-16:00	*Coffee break*	
16:00-17:30	Reprocessing of salt	Sylvie Delpech (CNRS)

- Good mix of «basic» information and recent developments related to the progress of the SAMOFAR project (and other projects also …)
- Audience, coming from very different backgrounds, is likely to appreciate the mix
- My role: provide neutronics basics

Neutronics of MSR

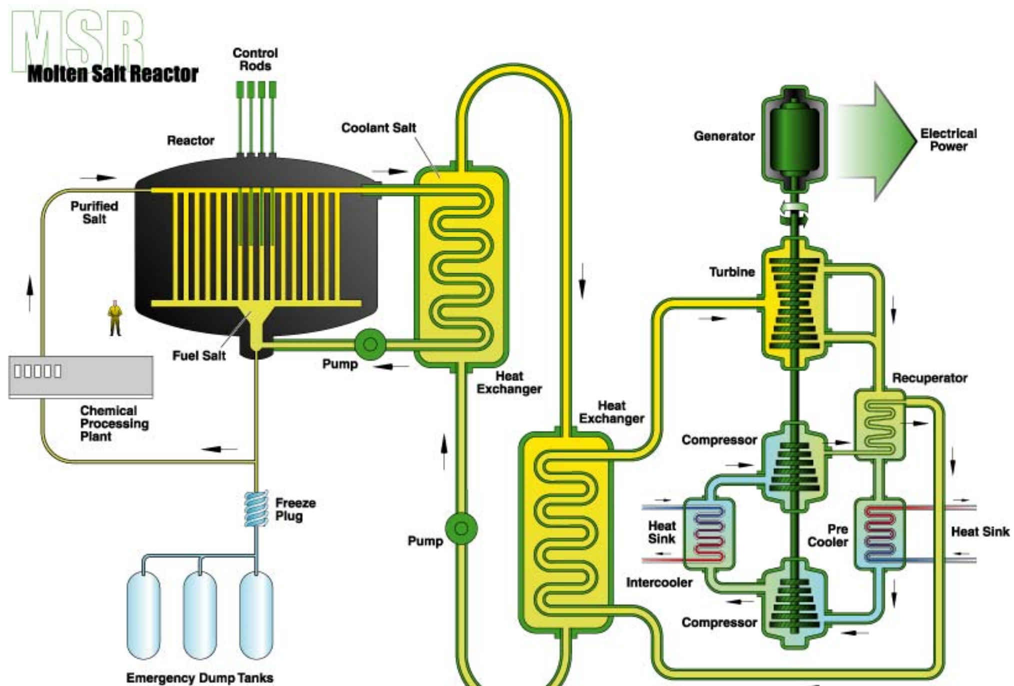
- This first lecture should serve as an introduction to the peculiarities of MSR for what concerns neutronics and how easily it leads to thermalhydraulic/chemistry/fuel cycle coupling → **multiphysics**
- Various aspects will be introduced, that will be surely treated in more detail in the following lectures, such as:
  - Thermal-hydraulics and CFD
  - Multiphysics simulation of MSR
  - Kinetics and dynamics (incl. noise analysis) of MSR
  - …

The MSR in Gen-IV

- *Sustainability* → minimal generation of nuclear waste
- *Safety and reliability* → enhanced nuclear safety
- *Economics* → reduced capital cost
- *Proliferation resistance* → further reduction of the risk of weapons materials proliferation
- *Physical protection*

- The Molten Salt Reactor (MSR)
  - fission power from fissile elements dissolved in a molten salt carrier
  - fast or epithermal-spectrum
  - on-line reprocessing

- *NOTE:*
  - *This picture is 17 years old (first GIF meeting in 2000)*
  - *Current proposals of MSR are quite different from this sketch*
  - *I started my PhD in 2002 on MSR neutronics, the current perception of MSR design is VERY different with respect to the past …*



https://www.gen-4.org/gif/jcms/c_42150/molten-salt-reactor-msr

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MSR neutronics – why so peculiar ?

- The multiplying medium is in a fluid phase, flowing through the reactor core

the fissile salt acts both as ***nuclear fuel*** and ***system coolant***

- The presence of a velocity field in the multiplying medium affects its neutronic behavior (due to the misplacement of delayed neutron precursors dragged by the fluid)

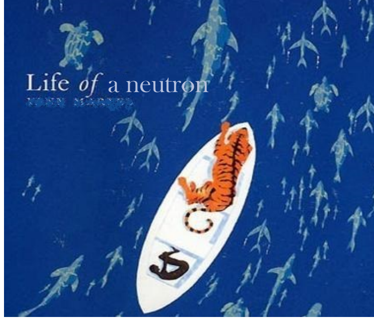
can be considered as a ***stronger*** neutronic / thermal-hydraulic coupling with respect to other reactor concepts

Multiphysics in MSR

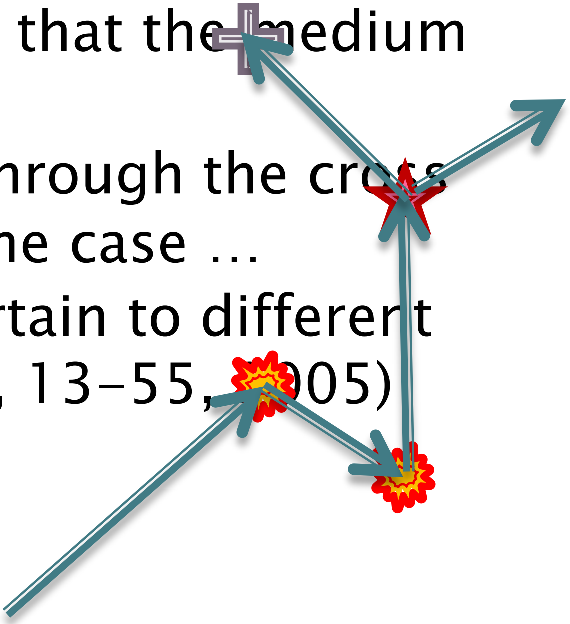
- Neutronic behavior is affected by the thermal-hydraulics of the molten salt
  - «Standard» feedback phenomena as in any reactor concept (Doppler, thermal expansion, …)
  - Effects associated to the motion of the fluid (*present also at zero power*)
- Consequences on both the steady-state and transient behavior
- (Interaction with the fuel chemistry, since on-line fuel modification is envisaged)

I will discuss this effect in particular, to introduce the relevance of the studies that will be described in the following lectures …

Life of a neutron in a MSR



- What happens when neutrons are propagating in a medium in motion ?
  - Collisions (localized phenomena) with the atoms of the medium
  - Free flights between collisions
  - Neutrons are not «affected» by the fact that the medium is moving (are not dragged in any way)
  - We may take into account the motion through the cross sections $\Sigma(x) \rightarrow \Sigma(x0-ut)$ … not really the case …
  - Effects related to “moving material” pertain to different situations (e.g., see *Prog. Nucl. En.*, **46**, 13-55, 2005)



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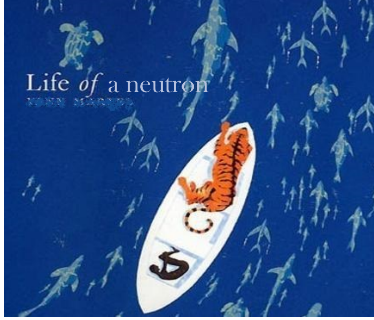
Life of a neutron in a MSR

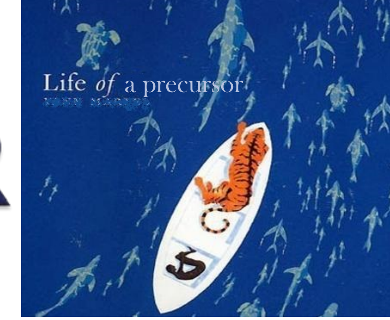


- Consequence ? The neutron balance equation is unchanged w.r.t. standard reactor concepts

$$\frac{1}{\mathrm{v}}\frac{\partial\varphi(\mathbf{r},E,\Omega,t)}{\partial t} = \Big[\hat{L}(t) + \underbrace{\hat{M}_p(t)\Big]\varphi}_{\textit{Prompt fission neutrons}}(\mathbf{r},E,\Omega,t) + \underbrace{\sum_{i=1}^{R}\frac{\chi_{d,i}}{4\pi}\lambda_i C_i(\mathbf{r},t)}_{\textit{Delayed neutrons emissions}}$$

- Instead, the fission products (atoms) that are precursors of delayed neutrons ***are dragged*** by the fluid medium, so the balance equation for them needs to be modified → introduction of a ***convective term***

Life of a precursor in a MSR



- Continuity equations for precursors **with a velocity field**

$$\frac{\partial}{\partial t}\frac{\chi_{d,i}C_i(\mathbf{r},t)}{4\pi} + \boxed{\nabla\cdot\left(\mathbf{u}(\mathbf{r},t)\frac{\chi_{d,i}}{4\pi}C_i(\mathbf{r},t)\right)} = -\lambda_i\frac{\chi_{d,i}}{4\pi}C_i(\mathbf{r},t) + \beta_i\hat{M}_{d,i}\varphi(\mathbf{r},E,\Omega,t)$$

$$i = 1, 2, \ldots R$$

- The velocity $\mathbf{u}(\mathbf{r},t)$ represents the macroscopic velocity field in the medium
- The mathematical nature of the problem is changed (boundary conditions required)
  - Solution in the full circuit may not be advisable
  - **Transit time** and BC on core inlet and outlet

$$C_i(\mathbf{r},t)\mathbf{u}(\mathbf{r},t)\cdot(-\mathbf{n})$$

$$= \int_{A_{out}} C_i(\mathbf{r}',t-\tau(\mathbf{r}'\to\mathbf{r},t))e^{-\lambda_i\tau(\mathbf{r}'\to\mathbf{r},t)}\mathbf{u}(\mathbf{r}',t-\tau(\mathbf{r}'\to\mathbf{r},t))\cdot\mathbf{n}'F(\mathbf{r}'\to\mathbf{r},t)dA, \qquad \mathbf{r}\in A_{in}$$

Full neutronic model of MSR

$$\begin{cases} \dfrac{1}{\mathrm{v}}\dfrac{\partial\varphi(\mathbf{r},E,\Omega,t)}{\partial t} = \left[\hat{L}(t)+\hat{M}_p(t)\right]\varphi(\mathbf{r},E,\Omega,t)+\displaystyle\sum_{i=1}^{R}\dfrac{\chi_{d,i}}{4\pi}\lambda_i C_i(\mathbf{r},t) \\ \dfrac{\partial}{\partial t}\dfrac{\chi_{d,i}C_i(\mathbf{r},t)}{4\pi}+\nabla\left(\mathbf{u}(\mathbf{r},t)\dfrac{\chi_{d,i}}{4\pi}C_i(\mathbf{r},t)\right) = -\lambda_i\dfrac{\chi_{d,i}}{4\pi}C_i(\mathbf{r},t)+\beta_i\hat{M}_{d,i}(t)\varphi(\mathbf{r},E,\Omega,t) \\ \qquad\qquad\qquad\qquad i=1,2\ldots R \qquad \text{plus IC and BC} \end{cases}$$

- Additional coupling to the TH model
  - Temperature-dependence of cross sections (as usual)
  - Explicit presence of velocity field **u** in the neutronic model (also at zero-power)
- Necessity of multiphysics approach (coupled NE/TH code development)

This is so innovative …

- The word «multiphysics» is relatively new in the nuclear science and engineering world
  - Annals of Nuclear Energy: first appearance in 2010
  - Progress in Nuclear Energy: first appearance in 2012
  - ANS Journals: first appearance in 2009
- The MSR concept is Gen-IV → innovative, not just evolutionary
- Actually, this innovation has deep roots in the past …

The book to be referenced

- R. V. Meghreblian, D. K. Holmes, *Reactor Analysis*, McGraw-Hill, New York, 1960

as before by (9.167). The solutions for this case are

$$w(t) = \left(\frac{\delta k_0}{l_{th}}\right)\frac{e^{-\chi t}\sinh St}{S}$$
$$\Theta(t) = \frac{\delta k_0}{\gamma}\left[1 - e^{-\chi t}\left(\frac{\chi}{S}\sinh St + \cosh St\right)\right] \quad (9.173)$$

As in the preceding case, these solutions are nonoscillatory and the system is "overdamped." This corresponds to very large values of the heat capacity. In these cases the temperature and power are relatively loosely coupled functions and respond very slowly to reactivity changes (compare curves labeled $p = 50$ to those labeled $p = 142.3$ in Figs. 9.10 and 9.11).

(4) $p = 0$

This case is primarily of academic interest. On the basis of the present model, the solutions are found to be

$$w(t) = \frac{\delta k_0}{\beta}\left(1 - e^{-\beta t/l_{th}}\right) \quad (9.174)$$
$$\Theta(t) = 0$$

Since this case corresponds to a system with infinite heat capacity, it is to be expected that the temperature will be unaffected by changes in reactivity (i.e., power level). However, under these circumstances the negative temperature coefficient cannot be of aid in stabilizing the system and one would expect the power level to rise without limit. Since the result given above reaches a finite value it is clear that this elementary model breaks down when applied to this case; however, note that for short times after the addition of δk_0 (i.e., $\beta t/l_{th} \ll 1$) the power function has the correct form. In this case

$$w(t) \simeq \left(\frac{\delta k_0}{l_{th}}\right)t \quad \text{and} \quad P(t) \simeq P_0 \exp\left(\frac{\delta k_0 t}{l_{th}}\right) \quad (9.175)$$

Thus the power level rises exponentially.

The various cases discussed above are shown graphically in Figs. 9.10

9.5 Kinetics of Circulating-fuel Reactors

was demonstrated in the preceding sections. These two factors are also essential to the dynamical behavior of circulating-fuel systems. In the present section we treat several aspects of the general problem and in particular show how the influence of the delayed neutrons and of the temperature coupling is altered by the circulation of the fuel. Since the

general problem is very complex, no attempt is made here to provide a thorough study, and most of the discussion is based on relatively crude models of both the nuclear and fluid dynamical features of the system. Although this approach reduces considerably the analytical difficulties in the calculations, it also tends to render the final results less useful for computational purposes. However, these models do allow a clear exhibition of the essential features of the kinetics of circulating-fuel reactors; moreover, some of the analytical techniques developed in this treatment are sufficiently general that they may be applied to the analysis of more complicated problems. In any case the general conclusions and results obtained with the simplified models provide a correct qualitative description of these systems.

All the analyses which follow are based on two fundamental assumptions, namely, that the neutron population can be well described by means of the one-velocity model and that the circulation of the fuel can be represented by a slug flow.[1] The nuclear features of these systems, then, may be developed from the general results obtained in Sec. 9.4. The assumption of slug flow consists, essentially, of limiting the transport term in the fluid-flow relations to variations in the direction of flow only. Thus the flow of mass (and energy) around the fuel circuit occurs along parallel channels or tubes, and no communication (cross-flow) is allowed between adjacent channels. This assumption greatly simplifies the fluid dynamical aspects of the problem and thereby permits, with a minimum of effort, a clear display of the coupling between the fluid-flow and nuclear characteristics of the system.

In the present treatment the effect of the delayed neutrons and the effect of temperature-flux coupling on the reactor kinetics are handled separately along the general lines previously developed for the stationary-fuel systems. Although in general these two factors must be considered jointly, there are many practical situations wherein the two effects can be separated. The intent here is primarily to simplify the analysis so as to focus attention on one feature at a time.

Delayed Neutrons. The effect of delayed
ior of circulating-fuel reactors is studied
nodel which describes a *low-power* thermal
ssion the expression low power is used to
ower level is sufficiently low that small
gible changes in the temperature of the
actor. Thus the energy balance of the

system is not considered, and no coupling is allowed between the temperature and the neutron flux. For the present, then, we ignore entirely temperature coefficients of reactivity and focus attention on the role of the delayed neutrons. The specific objective is to derive the inhour

[1] The expression "slug flow" is used to denote a one-dimensional flow field.

The book to be referenced

- R. V. Meghreblian, D. K. Holmes, *Reactor Analysis*, McGraw-Hill, New York, 1960
- At page 592, the equations for the neutronics are given:

$$-D\nabla^2\phi(\mathbf{r},t) + \Sigma_a\phi(\mathbf{r},t) = -\frac{1}{v}\frac{\partial}{\partial t}\phi(\mathbf{r},t) + (1-\beta)\nu\Sigma_f p_{\mathrm{th}} g_{\mathrm{th}}\phi(\mathbf{r},t) + p_{\mathrm{th}} g_{\mathrm{th}} \sum_i \lambda_i C_i(\mathbf{r},t) \quad (9.84)$$

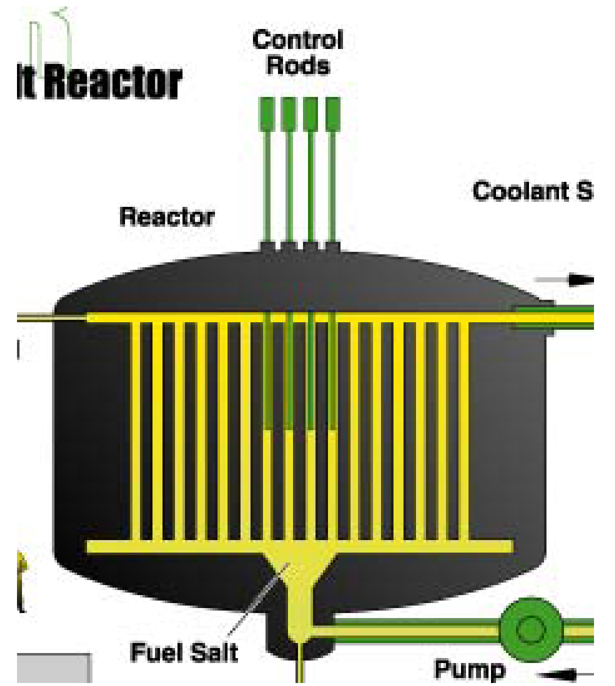
$$\frac{\partial}{\partial t} C_i(\mathbf{r},t) + V\frac{\partial}{\partial z} C_i(\mathbf{r},t) = \nu\beta_i\Sigma_f\phi(\mathbf{r},t) - \lambda_i C_i(\mathbf{r},t) \qquad 0 \le z \le h \quad (9.176)$$

$$\frac{\partial}{\partial t} C_i(\mathbf{r},t) + V\frac{\partial}{\partial z} C_i(\mathbf{r},t) = -\lambda_i C_i(\mathbf{r},t) \qquad h \le z \le h_\star \quad (9.177)$$

- The fundamental aspects are all here:
  - Balance equation for neutrons unchanged (ok, here is one-group diffusion, but the physics is there)
  - Convective term in the precursor equation to account for dragging (velocity V supposed constant and along z, called slug flow)
  - Necessity of proper BC for this problem (ok, here the balance is made on the whole primary circuit, thus the transit time is $\tau = \dfrac{h_\star - h}{V}$

Solution to the neutronics eqs

- The simplified model just introduced has the nice characteristics of allowing almost analytical treatment
- The effect of fluid motion on steady-state (and time-dependent) configurations is clearly visible and allows immediate physical comprehension
- The hypothesis of slug flow seemed to be rather appropriate for a MSR as in the GEN-IV sketch (parallel pipes with graphite matrix)
- Lot of work in this respect in the years 2002-2006 (e.g. the MOST project)



Effect of fuel motion – steady state

- 1D diffusion, slab geometry, zero-flux BC → analytical solution with Helmholtz eigenfunctions
- Dependence of the multiplication eigenvalue k on the velocity field and delayed neutron characteristics (β and λ)

$$
\begin{cases}
D\dfrac{d^2\Phi}{dx^2} - \Sigma_a\Phi + \dfrac{1}{k}(1-\beta)\nu\Sigma_f\Phi + \lambda C = 0 \\
u\dfrac{dC}{dx} + \lambda C = \dfrac{1}{k}\beta\nu\Sigma_f\Phi \\
\Phi(x=0) = \Phi(x=H) = 0 \\
C(x=0) = C(x=H)e^{\lambda\tau_\ell}
\end{cases}
$$

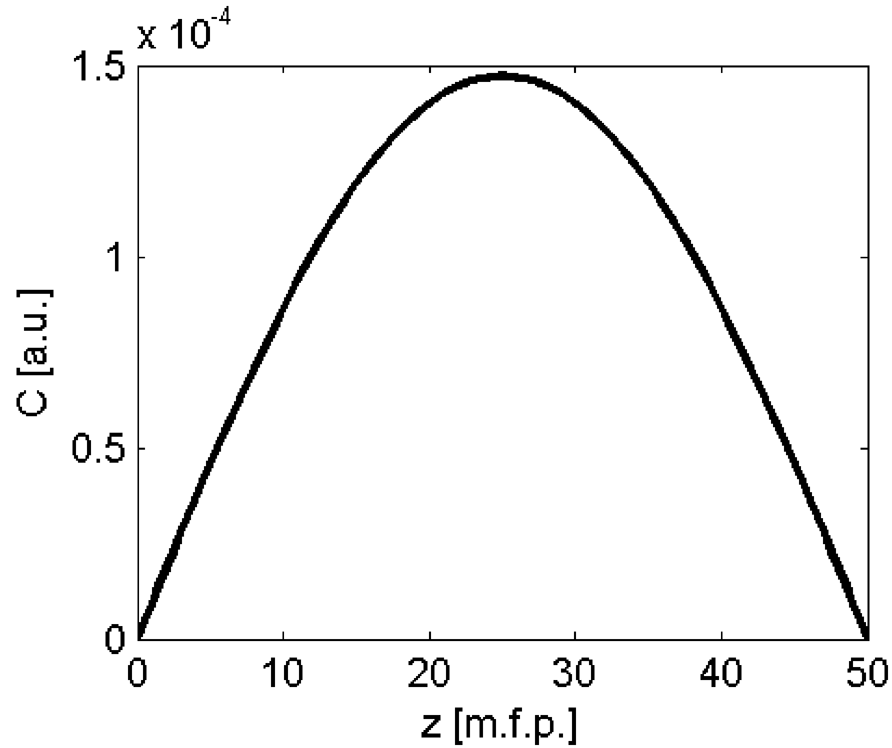
Transit time

- Spatial redistribution of the delayed neutron precursors density and reduction of their ***importance***

Results (u=0 mfp/s)

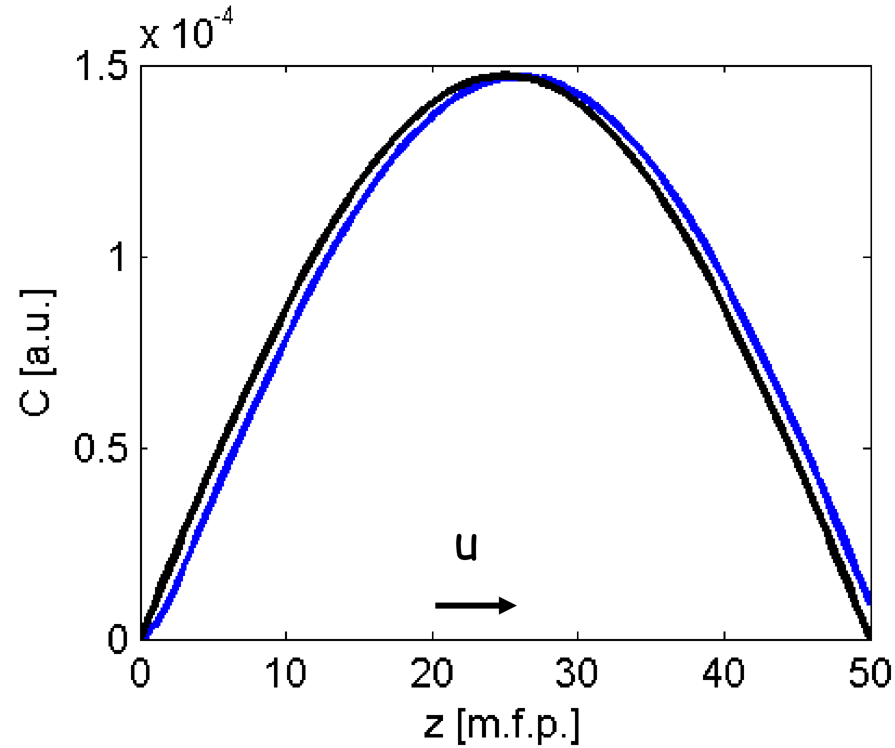
- $u = 0$ mfp/s
- $\tau_{core} \to \infty$
- $\tau_{\ell} \to \infty$

- $k_{eff} = 1.00000$
- $\Delta\rho = 0$ pcm
- $\beta_{eff} = 650$ pcm
- $\Delta\beta_{eff} = 0$ pcm



Results (u=0.1 mfp/s)

- $u = 0.1$ mfp/s
- $\tau_{core} = 500$ s
- $\tau_\ell = 1000$ s



- $k_{eff} = 0.99997$
- $\Delta\rho = -3$ pcm
- $\beta_{eff} = 647$ pcm
- $\Delta\beta_{eff} = -3$ pcm

How is this quantity defined ?
Some discussion coming next

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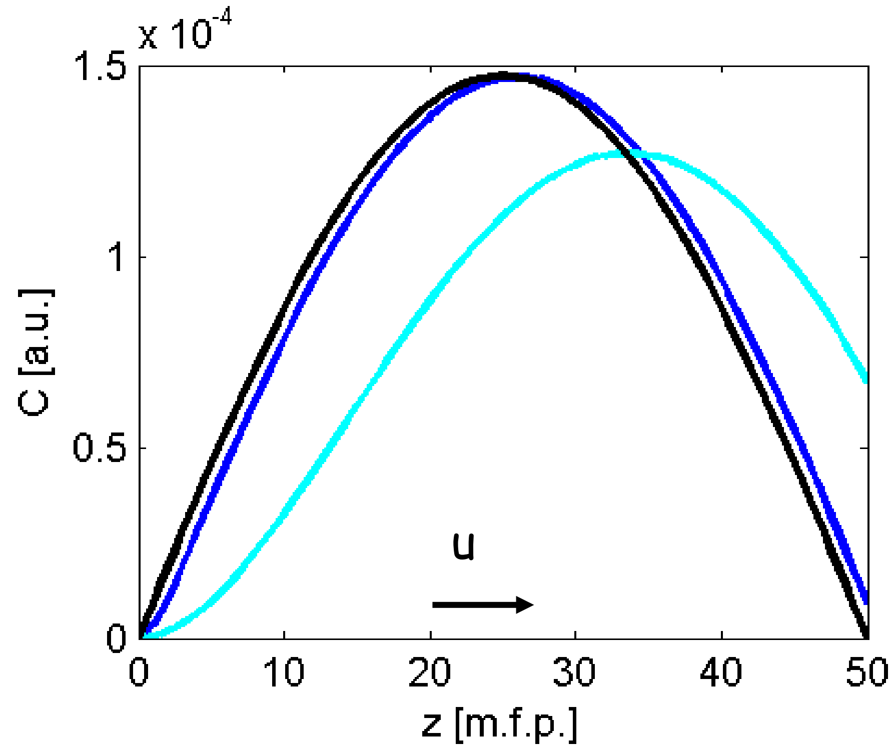
Results (u=1 mfp/s)

- $u = 1$ mfp/s
- $\tau_{core} = 50$ s
- $\tau_{\ell} = 100$ s

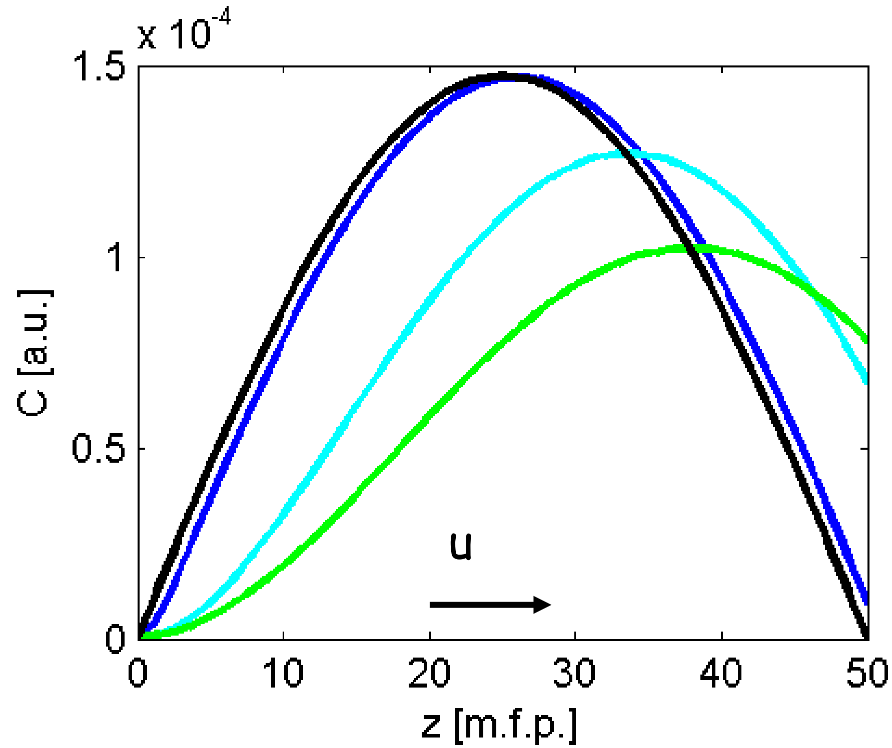
- $k_{eff} = 0.99868$
- $\Delta\rho = -132$ pcm
- $\beta_{eff} = 518$ pcm
- $\Delta\beta_{eff} = -132$ pcm



Results (u=2 mfp/s)

- $u = 2$ mfp/s
- $\tau_{core} = 25$ s
- $\tau_\ell = 50$ s

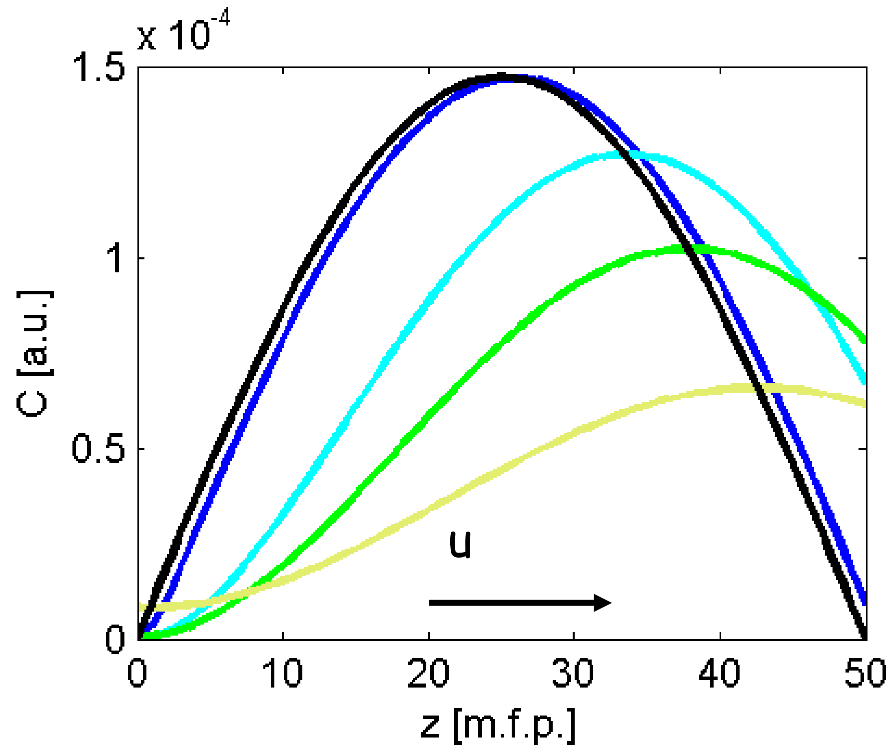
- $k_{eff} = 0.99736$
- $\Delta\rho = -265$ pcm
- $\beta_{eff} = 386$ pcm
- $\Delta\beta_{eff} = -264$ pcm



Results (u=5 mfp/s)

- $u = 5$ mfp/s
- $\tau_{core} = 10$ s
- $\tau_{\ell} = 20$ s

- $k_{eff} = 0.99587$
- $\Delta\rho = -415$ pcm
- $\beta_{eff} = 236$ pcm
- $\Delta\beta_{eff} = -414$ pcm



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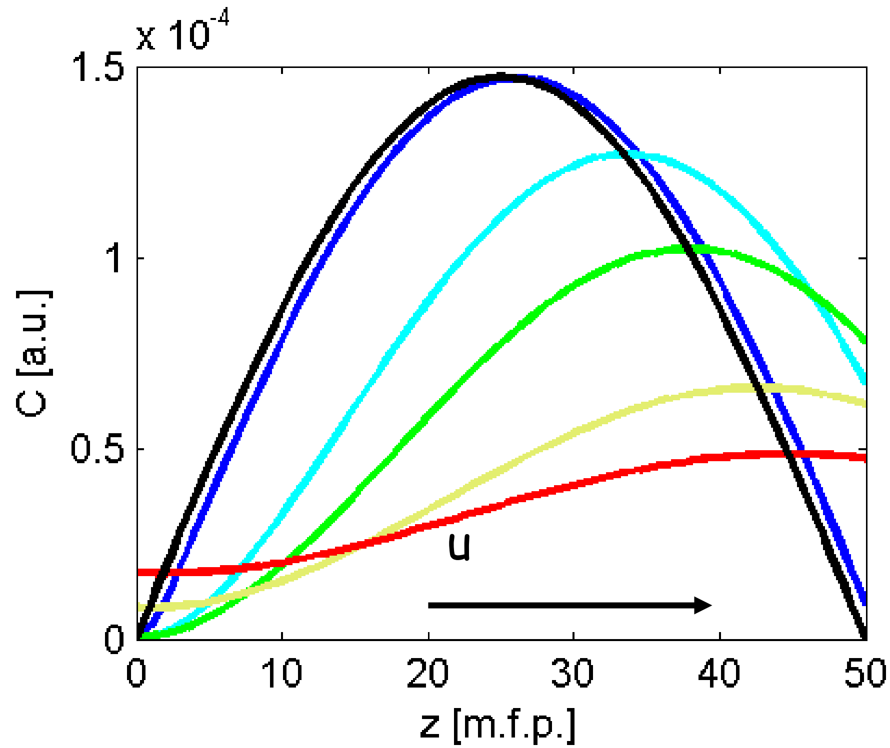
Results (u=10 mfp/s)

- $u = 10$ mfp/s
- $\tau_{core} = 5$ s
- $\tau_{\ell} = 10$ s

- $k_{eff} = 0.99543$
- $\Delta\rho = -459$ pcm
- $\beta_{eff} = 194$ pcm
- $\Delta\beta_{eff} = -456$ pcm



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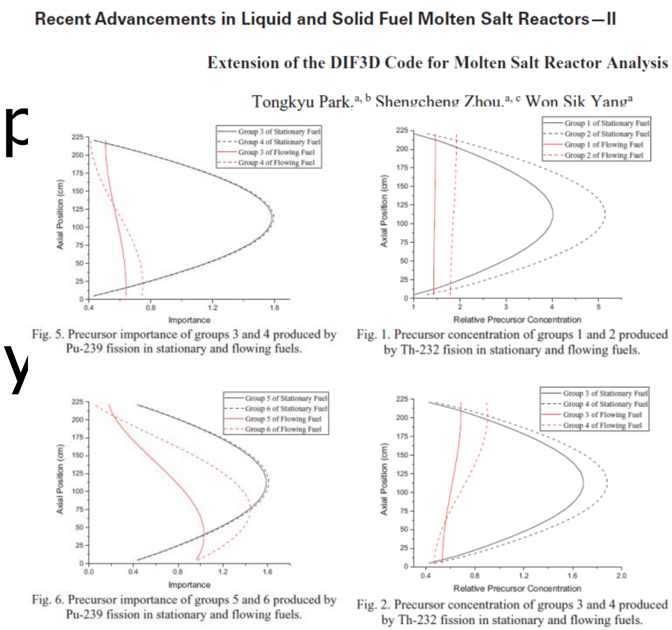
Comment on these results

- You may think this is just a simp
  but…


Recent Advancements in Liquid and Solid Fuel Molten Salt Reactors—II

Extension of the DIF3D Code for Molten Salt Reactor Analysis

Tongkyu Park,[a, b] Shengcheng Zhou,[a, c] Won Sik Yang[a]

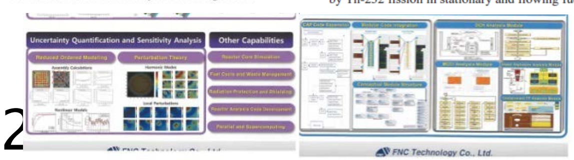
Fig. 5. Precursor importance of groups 3 and 4 produced by Pu-239 fission in stationary and flowing fuels.

Fig. 1. Precursor concentration of groups 1 and 2 produced by Th-232 fission in stationary and flowing fuels.

Fig. 6. Precursor importance of groups 5 and 6 produced by Pu-239 fission in stationary and flowing fuels.

Fig. 2. Precursor concentration of groups 3 and 4 produced by Th-232 fission in stationary and flowing fuels.

- I have seen these graphs already
  different situations
  - My PhD thesis
  - Engineering companies's leaflets
  - Very recent papers (*ANS Trans.*, June 2



- This may mean that the subject of MSR neutronics still is a fascinating subject …

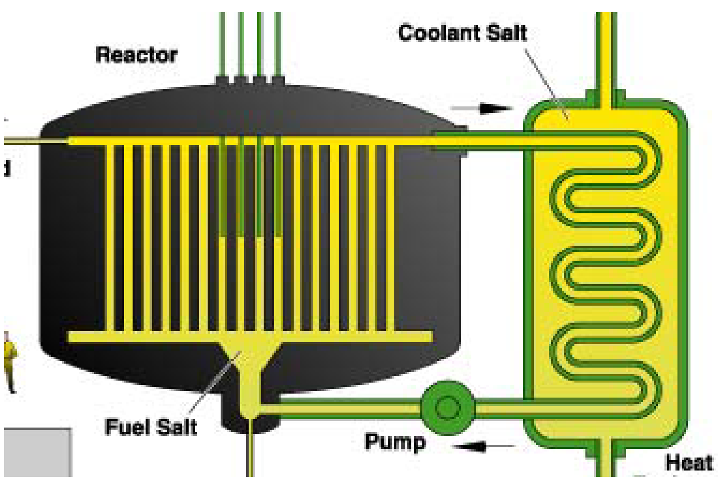
Effect of fuel motion – steady state

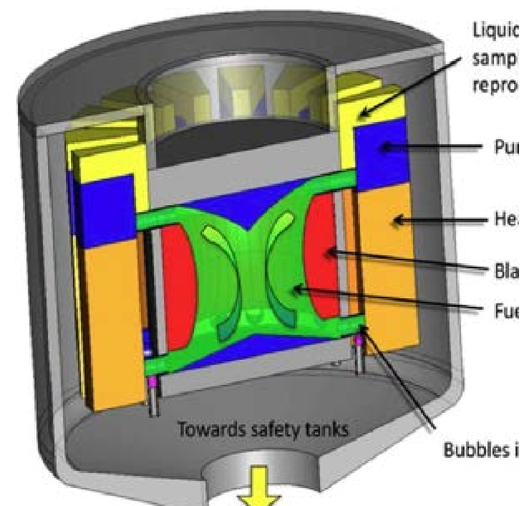
- Summary
  - Fluid-dynamics affects the criticality state of the reactor
  - The distribution of precursors is distorted (depending on decay constants)
  - The flux is usually almost unperturbed (due to small value of the delayed neutron fraction β)
  - In real cases, a more complex fluid velocity field may induce other physical effects
    - Recirculation spots → localized delayed neutron sources
    - Bypasses → transit times affected
    - …

may be relevant depending on core geometry → CFD calculations required

Relevance to current MSR design

- MSRE (MOST)
  - Bank of pipes surrounded by graphite → thermal spectrum
  - Neutronics could be treated with imposed fuel velocity
- MSFR (SAMOFAR)
  - Homogeneous core → fast spectrum
  - (possible) Complex velocity field
  - Necessity of tight coupling of NE and TH





J. Serp et al., Progr. Nucl. En., **77**, 308–319, 2014

Effect of fuel motion – dynamics

- It is clear from the previous comments that the «pure» neutronic approach is not sufficient
- When moving to dynamics, it is then
  - Important to identify the peculiarities of the coupled physics of MSR w.r.t. solid fuel systems
  - Fundamental to prove the stability and safety of the concept in light of these differences

- Sticking to neutronics, there is one parameter of paramount importance in dynamics that is
  - «Neutronic based»
  - Affected strongly by fluid-dynamics in MSR

The effective delayed neutron fraction β_{eff}

The role of delayed neutrons

- Neutron kinetics can be characterized by the value of the delayed neutron fraction β
  - Dependent on the fuel adopted
  - In the range 300 – 700 pcm
  - A larger value of β implies a less prompt response → safer and easier control

- The dynamic role of the delayed neutrons depends also on the energy spectrum of the system
  - Delayed neutrons have a softer spectrum
  - In a thermal system, this implies a larger role in fission production
  - The opposite may occur when dealing with a fast system

- To evaluate this role → β_{eff} **(beta effective)**

Beta effective – β_{eff}

- The effective delayed neutron fraction allows to quantify the effective role of delayed neutrons in dynamics
  - Evaluated as integral of the delayed production weighted on the neutron importance (deterministic approach)

$$\beta_{eff} = \frac{\left\langle \varphi^{\dagger} \middle| \beta_i \hat{M}_{d,i} \varphi \right\rangle}{\left\langle \varphi^{\dagger} \middle| \hat{M}_t \varphi \right\rangle}$$

Z. Akcasu, G.S. Lellouche, L.M. Shotkin, Mathematical Methods in *Nuclear Reactor Dynamics*, Academic Press, New York, 1971.
A.F. Henry, *Nuclear-reactor Analysis*, MIT Press, Cambridge, MA, 1975.

  - Evaluated as difference in multiplication constant when delayed fissions are neglected (Monte Carlo approach)

$$\beta_{eff} = \frac{k - k_p}{k}$$

R. K. Meulekamp, S. C. van der Marck, Calculating the effective delayed neutron fraction with Monte Carlo, *Nucl. Sc. Eng.*, **152**, 142-148, 2006.
Y. Nagaya et al., Comparison of Monte Carlo calculation methods for effective delayed neutron fraction," *Ann. Nucl. En.*, **37**, 1308-1315, 2010.

Beta effective – β_{eff}

- Both definitions (equivalent at the first order) evaluate the different role of delayed neutrons in kinetics, associated to their softer spectrum

M. Carta et al., Calculation of the Effective Delayed Neutron Fraction by Deterministic and Monte Carlo Methods, *Science and Technology of Nuclear Installations*, **2011**, Article ID 584256, 8 pages, 2011

- As a general statement:
  - $\beta_{eff} > \beta_{phys}$ for thermal systems
  - $\beta_{eff} < \beta_{phys}$ for fast systems

- These are the kind of results you may get when running ERANOS, MCNP, SERPENT

(*unless some smart guys made some modifications to study MSR …*)

Aufiero et al., Calculating the effective delayed neutron fraction in the Molten Salt Fast Reactor: Analytical, deterministic and Monte Carlo approaches, *Ann. Nucl. En.*, **65**, 78–90, 2014

Beta effective in MSR – I

- In MSR, the role of delayed neutrons is affected also by the fuel motion
- In principle, the role will be reduced, due to
  - *displacement in the reactor core* → neutrons emitted in regions with reduced importance (and Prof. Pazsit will discuss importance in MSR tomorrow)
  - *production in the primary circuit* → zero contribution to the fission chain

- To evaluate the amount of this reduction, a **different calculation approach** needs to be adopted

Beta effective in MSR – II

- How to evaluate the β_{eff} in MSR ?
- Some examples of approaches adopted:
  - Reduce β_{eff} by a fraction equal to the fraction of precursors still decaying in the core (both in deterministic and stochastic approaches)
    - Purely geometrical, no info on neutron importance

M. Delpech et al., Benchmark of Dynamic Simulation Tools for Molten Salt Reactors, *Int. Conf. GLOBAL 2003*, New Orleans, 2182-2187, 2003.

  - Set the variation of β_{eff} equal to the reactivity loss
  - Use a different definition of the weighted integral (e.g., using the precursor importance instead of the flux)
    - Necessity of adjoint neutronic model for MSR
    - Consistent to standard formulation of point kinetics

S. Dulla, P. Ravetto, M.M. Rostagno, Neutron kinetics of fluid-fuel systems by the quasi-static method, *Ann. Nucl. En.*, **31**, 1709-1733, 2004.

  - …
- All these definitions will produce a smaller β_{eff} (*and surely not the same value*)

Example of results (from EVOL WP2)

^{233}U-started composition	LPSC ENDF/B6	POLITO JEFF-311 (see section 3.2.5)	POLIMI SERPENT			POLIMI ERANOS JEFF-3.1	TU Delft ENDF-B7
			JEFF-3.1		ENDF-B7		
			Nominal flow rate	Uniform sampling			
β_0 [pcm]	330	315.00 ± 0.04	310		325	-	310
β_{eff} [pcm]	320	305.00 ± 0.76	305		317.8	318.1	290
β_{circ}/β_{eff}	0.529	0.3837	0.479	0.407	-	0.540[b]	0.430
β_{circ} [pcm]	169.46	117.3	146	124	-	171.9[b]	124.6

Why interested in β_{eff}?

- All the integral parameters (effective delayed neutron fractions, effective lifetime, …) have a role in:
  - Quick characterization of the system in view of its time-dependent behavior
  - Input data for dynamic simulations based on point-like models

- The analysis of these parameters can allow to identify some potential issues in the dynamic behavior

Relevance to MSR

- β_{eff} characterizes the dynamic behavior
- It would be advisable to have a «large» β_{eff}

BUT

- Innovative reactors are also interesting in the perspective of transmutation (Pu-39 and TRU) and breeding (U-233)

nuclear fuels with lower β_{phys} values

Safety assessments must include transient scenarios involving the peculiar neutronic aspects of MSR

Summary

- On the basis of the discussion we just had we can say that
  - MSRs require accurate coupled NE/TH analysis to understand their peculiar behavior
    - LECTURES TODAY IN THE MORNING AND AFTERNOON
  - The definition of the composition of the fuel, its evolution and the techniques to control it play a significant role in this perspective
    - LECTURE TODAY IN THE MORNING AND TOMORROW

Ok guys,
Have fun !