Safety Analysis of the Molten Salt Fast Reactor Fuel Composition with Moltres

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   - Moltres
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2. Method
   - Molten Salt Fast Reactor
   - Moltres Workflow
3. Results
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   - Steady State Results
   - Unprotected Loss of Heat Sink
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Molten Salt Reactors

- A class of advanced nuclear reactor concepts that contain nuclear fuel dissolved and circulating in a molten salt coolant loop
- May also include designs with solid fuel and molten salt coolant
- Can potentially run for extended periods with minimal shutdown time due to online fuel reprocessing capabilities

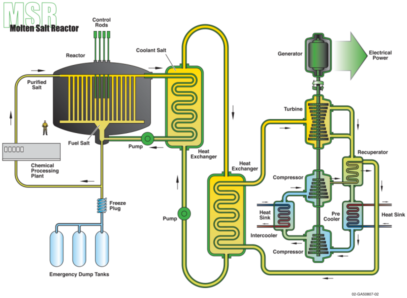

Figure 1: Schematic diagram of a general Molten Salt Reactor (MSR) concept.

Molten Salt Reactors

Characteristics and challenges

- Strong coupling between neutronics and thermal-hydraulics
  - Strong density feedback in the fuel salt
  - Stronger prompt response expected compared to existing LWRs
- Movement of Delayed Neutron Precursors (DNPs) in the molten salt loop
  - Conventional safety analysis codes do not account for the delayed neutron precursor movement
- Constantly evolving fuel composition across the lifetime of an MSR
  - Reactor safety parameters and transient response may change over time from start-up to equilibrium compositions

Moltres

What is Moltres?

- Moltres [5] is an application built on the Multiphysics Object-Oriented Simulation Environment (MOOSE) framework, for the simulation of MSRs
- MOOSE [3] is an open source finite element framework written in C++ that relies on Libmesh and PETSc for advanced meshing and PDE solving capabilities
- Moltres can run transient, implicitly coupled neutronics/thermal-hydraulics simulations
  - Multi-group neutron diffusion (arbitrary no. of groups)
  - DNP decay (with advection)
  - Incompressible Navier-Stokes for temperature advection-diffusion
  - 1D, 2D, and 3D modeling are supported

Objectives

Objectives

- Verify the neutronics results in Moltres against Serpent [4] with the Molten Salt Fast Reactor (MSFR) concept with six neutron energy groups
  - Previous validation study has been performed with an MSRE-like model with two (fast/thermal) neutron energy groups [5]
- Demonstrate coupled neutronics/thermal-hydraulics simulations of the MSFR concept using Moltres,
  - with start-up, early-life, and equilibrium fuel compositions
  - for steady state and transient cases: Unprotected Loss of Heat Sink (ULOHS) accident

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Molten Salt Fast Reactor

Features

- Fast-spectrum MSR concept
- Designed to run on a closed thorium fuel cycle
- Primary fuel salt flows upwards through the central core region and separates into 16 smaller external loops towards the heat exchangers and pumps
- Radially surrounded by a tank of blanket salt consisting of fertile isotopes such as ^{232}Th for breeding

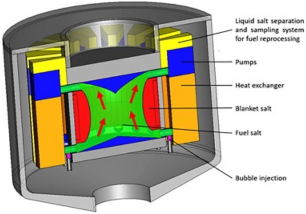

Figure 2: MSFR concept [7].

Table 1: Specifications of the MSFR design [7].

Parameter	Value
Thermal output [MW_{th}]	3000
Electric output [MW_e]	1500
Salt volume [m^3]	18
Nominal flow rate [$kg\ s^{-1}$]	18500
Nominal circulation time [s]	4.0
Inlet temperature [K]	923
Outlet temperature [K]	1023
Blanket volume [m^3]	7.3

Moltres Workflow

Moltres solves the implicitly coupled governing equations for neutrons flux ϕ, DNP concentration C, and temperature T.

Moltres requires neutron group constant data, generated from a neutron transport code (e.g. Serpent, SCALE), for the neutronics calculations.

Moltres accepts all mesh file formats supported by MOOSE (e.g. exodus, gmsh, etc.)

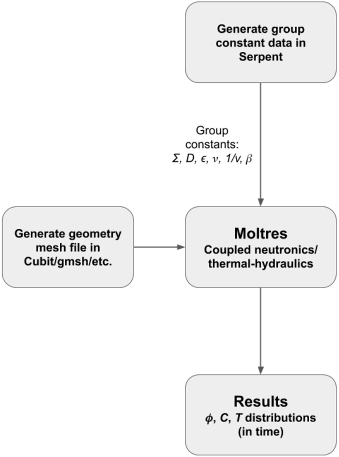

Figure 3: Flowchart for running a Moltres simulation.

Moltres Workflow

Multi-group neutron diffusion

$$\frac{1}{v_g}\frac{\partial \phi_g}{\partial t} - \nabla \cdot D_g \nabla \phi_g + \Sigma_{R,g}\phi_g = \sum_{g \neq g'}^{G} \Sigma_{s,g' \to g}\phi_{g'} + \chi_g^p \sum_{g'=1}^{G} (1-\beta)\nu\Sigma_{f,g'}\phi_{g'} + \chi_g^d \sum_{i}^{I} \lambda_i C_i \tag{1}$$

DNP concentration (with advection)

$$\frac{\partial C_i}{\partial t} = \sum_{g'=1}^{G} \beta_i \nu \Sigma_{f,g'}\phi_{g'} - \lambda_i C_i - \nabla \cdot \vec{u}\, C_i \tag{2}$$

Additional advection term in the DNP concentration equations to account for fuel advection effects.

Moltres Workflow

Temperature advection-diffusion

$$\rho c_p \frac{\partial T}{\partial t} + \nabla \cdot \left(\rho c_p \vec{u}\, T - k \nabla T\right) = Q_f - Q_{hx} \tag{3}$$

where

$$Q_f = \sum_{g=1}^{G} \epsilon_g \Sigma_g^f \phi_g \text{ and } Q_{hx}\text{: heat exchanger sink term}$$

- Moltres has access to the Navier-Stokes module on MOOSE for simulating flow
- For this work, we assumed uniformly vertical flow

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Neutronics Results

Serpent Input Parameters

- Neutron population
  - 200,000 neutrons per cycle
  - 50 inactive, 500 active cycles
  - 5 pcm statistical error in k_{eff}
- JEFF-3.1.2 nuclear data library
- Six neutron energy groups
- Eight delayed neutron precursor groups
- Temperatures defined from 900 K to 1200 K at 50 K intervals

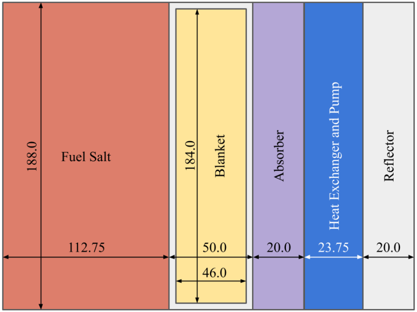


Figure 4: 2D axisymmetric model used in Serpent. Derived from the MSFR reference model [2]. (Figure not to scale)

Neutronics Results

Fuel Composition Data

- Depletion calculations performed in a previous paper by A. Rykhlevskii, B. Betzler, A. Worrall, and K. Huff [6]
- 60-year depletion calculation on SCALE/TRITON using a unit cell representation of the MSFR
- U/Th breeder reprocessing scheme
  - Th feed, with some ^{233}U from the blanket rerouted to the fuel salt to maintain criticality
- Compositions:
  - Start-up: LiF - ThF_4 - UF_4 (77.5% - 19.9% - 2.6%)
  - Early-life: 300 days after start-up
  - Equilibrium: 43 years after start-up ($<$3% change in TRU vector between depletion time-steps)

Neutronics Results

Moltres Simulation Details

- Adaptive backward Euler time-stepper
- Six neutron groups
- Eight DNP groups
- Vacuum neutron boundary condition on outer edges
- Uniform upward flow of 1.125 m s^{-1}
- Flow/decay of DNPs and heat removal in the outer loop is simulated on a simplified 1D geometry separate from active core region
- Fixed heat removal rate, Q_{hx}, to secondary loop system

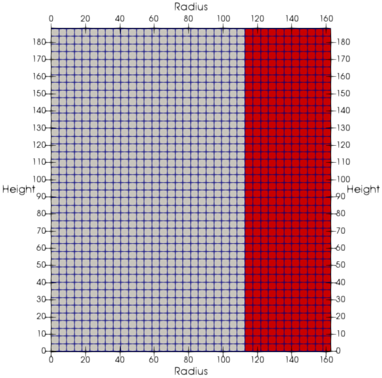


Figure 5: Mesh of the 2D axisymmetric model used in Moltres. The grey and red regions represent the fuel and blanket salt respectively.

Neutronics Results

Results

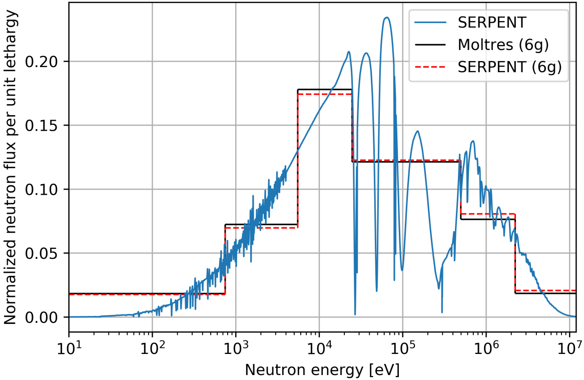


Figure 6: Continuous and six energy group neutron flux distributions from Serpent and Moltres (start-up composition without DNP drift at 973 K).

Neutronics Results

Fuel Temperature Reactivity Feedback

Table 2: Fuel temperature reactivity feedback coefficients with the start-up fuel composition.

Composition	α_T [pcm K^{-1}] Serpent	α_T [pcm K^{-1}] Moltres	Difference [%]
Start-up	-7.39 ± 0.03	-7.46	-0.94
Early-life	-7.25 ± 0.03	-7.33	-1.1
Equilibrium	-6.24 ± 0.03	-6.34	-1.6

Steady State Results

Steady state neutron flux (with DNP drift)

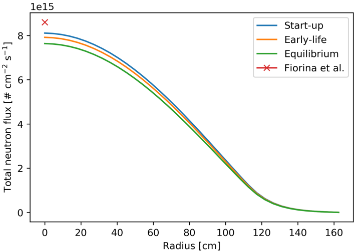


Figure 7: Total radial neutron flux at reactor half-height, for start-up, early-life, and equilibrium fuel compositions.

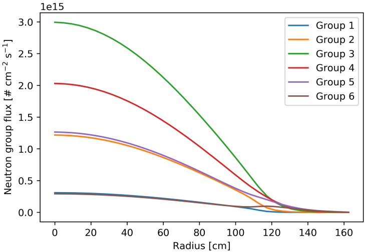
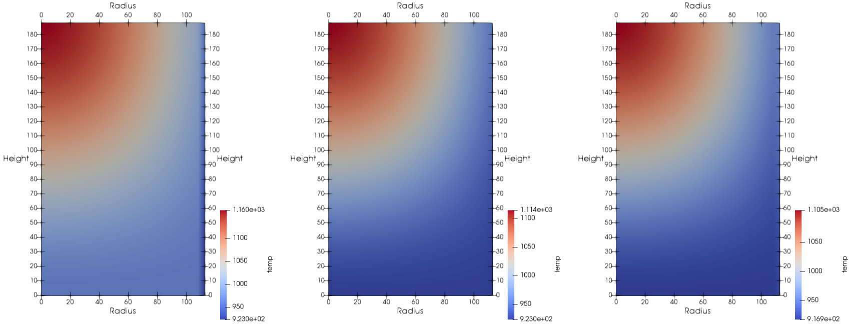


Figure 8: Neutron group fluxes at reactor half height, for start-up fuel compositions.

- Peak neutron flux is close to 8.6×10^{15} cm^{-2} s^{-1} reported by Fiorina et al. [1]

Steady State Results

Steady state temperature distribution



(a) Start-up (b) Early-life (c) Equilibrium

Figure 9: Temperature distribution in fuel salt region for start-up, early-life, and equilibrium fuel compositions.

Steady State Results

Table 3: Average fuel inlet and outlet temperatures

Composition	Inlet temperature [K]	Outlet temperature [K]
Start-up	964.85	1067.40
Early-life	925.11	1025.39
Equilibrium	916.85	1016.67

Observation

Highest average inlet and outlet temperatures observed for the start-up composition, followed by early-life, and equilibrium compositions.

Steady State Results

DNP Distribution

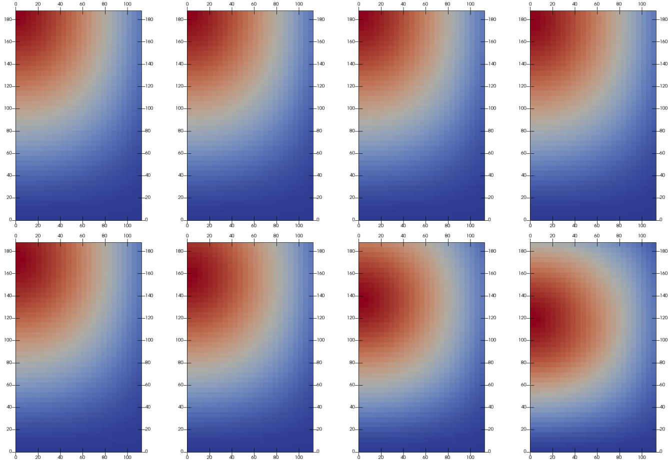


Figure 10: DNP distribution from group 1 to group 8 (left to right, top to bottom). $t_{1/2} = 55.60$ s, 24.50 s, 16.30 s, 5.21 s, 2.37 s, 1.04 s, 0.42 s, and 0.20 s.

Unprotected Loss of Heat Sink

Problem Description

- ULOHS accidents are imagined to occur due to various causes of failure in the heat exchanger or secondary coolant loop with no safety response (e.g. scram)
- In this work, ULOHS is simulated by an exponential decrease in the heat loss rate from the heat exchanger with a time constant of 5 s

$$Q_{hx} = Q_{hx,0} e^{-t/5}$$

- Expect to observe a rise in temperature, accompanied by a decrease in neutron flux and heat generation

Unprotected Loss of Heat Sink

Results

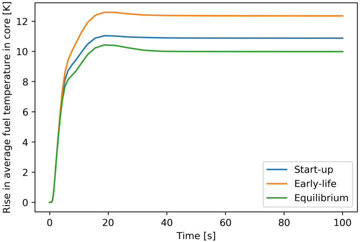


Figure 11: Rise in average core temperature during ULOHS

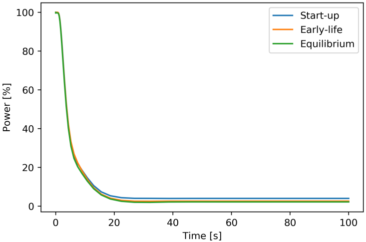


Figure 12: Power generation during ULOHS

- ULOHS starts at t = 0
- Average fuel temperature in core stabilizes at 10 K higher than the initial temperature
- Lower than expected temperature increase compared to Fiorina et al. [1]

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Conclusion

The neutron group fluxes and reactivity coefficients from Moltres (without DNP drift) are in good agreement with Serpent results.

The steady state neutron flux, temperature, and DNP distributions observed are expected for uniform flow.

Discrepancies present in peak neutron flux and temperature distributions relative to other work

Higher core temperatures observed for start-up fuel composition than early-life and equilibrium compositions.

Future Work

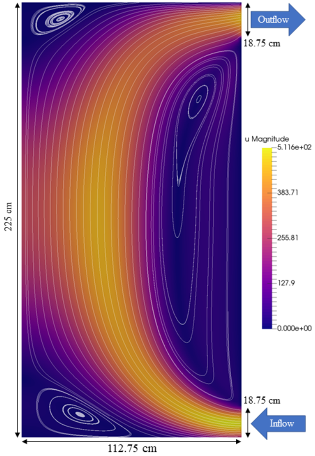
Improvements in thermal-hydraulics model

- Decay heat model
- Heat exchanger model

Turbulence model

- Implement a Reynolds-averaged Navier-Stokes (RANS) -based turbulence model in Moltres

Future Work



MSFR core fuel salt region

- Fuel salt composition: LiF-UF_4-ThF_4
- Incompressible Navier-Stokes with SUPG-PSPG stabilization
- Axisymmetric about the left boundary
- Inlet on the lower right corner (0.1875m wide)
- Outlet on the upper right corner (0.1875m wide)
- Re = 450 in this simulation (actual Re is higher)
- Boundary conditions
  - u_x
    - Top, bottom, left, right: $u_x = 0$
    - Inlet: $u_x = -5.085 \times 4 \times \left[\frac{y}{0.1875} - \left(\frac{y}{0.1875}\right)^2\right]$
    - Outlet: $\frac{du_x}{dx} = 0$
  - u_y
    - Top, bottom, right, inlet: $u_y = 0$
    - Left, outlet: $\frac{du_y}{dx} = 0$

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References I

[1] Carlo Fiorina, Manuele Aufiero, Antonio Cammi, Fausto Franceschini, Jiri Krepel, Lelio Luzzi, Konstantin Mikityuk, and Marco Enrico Ricotti.
Investigation of the MSFR core physics and fuel cycle characteristics.
Progress in Nuclear Energy, 68:153–168, September 2013.

[2] Carlo Fiorina, Danny Lathouwers, Manuele Aufiero, Antonio Cammi, Claudia Guerrieri, Jan Leen Kloosterman, Lelio Luzzi, and Marco Enrico Ricotti.
Modelling and analysis of the MSFR transient behaviour.
Annals of Nuclear Energy, 64(Supplement C):485–498, February 2014.

[3] Derek Gaston, Chris Newman, Glen Hansen, and Damien Lebrun-Grandie.
MOOSE: A parallel computational framework for coupled systems of nonlinear equations.
Nuclear Engineering and Design, 239(10):1768–1778, October 2009.

[4] Jaakko Leppanen, Maria Pusa, Tuomas Viitanen, Ville Valtavirta, and Toni Kaltiaisenaho.
The Serpent Monte Carlo code: Status, development and applications in 2013.
Annals of Nuclear Energy, 82:142–150, August 2015.

[5] Alexander Lindsay, Gavin Ridley, Andrei Rykhlevskii, and Kathryn Huff.
Introduction to Moltres: An application for simulation of Molten Salt Reactors.
Annals of Nuclear Energy, 114:530–540, April 2018.

References II

[6] Andrei Rykhlevskii, Benjamin Betzler, Andrew Worrall, and Kathryn Huff.
Fuel cycle performance of fast spectrum molten salt reactor designs.
In *Int. Conf. Mathematics and Computational Methods Applied to Nuclear Science and Engineering*, 2019.

[7] Jerome Serp, Michel Allibert, Ondrej Benes, Sylvie Delpech, Olga Feynberg, Vronique Ghetta, Daniel Heuer, David Holcomb, Victor Ignatiev, Jan Leen Kloosterman, Lelio Luzzi, Elsa Merle-Lucotte, Jan Uhl, Ritsuo Yoshioka, and Dai Zhimin.
The molten salt reactor (MSR) in generation IV: Overview and perspectives.
Progress in Nuclear Energy, 77(Supplement C):308–319, November 2014.