MSRE Problem Description

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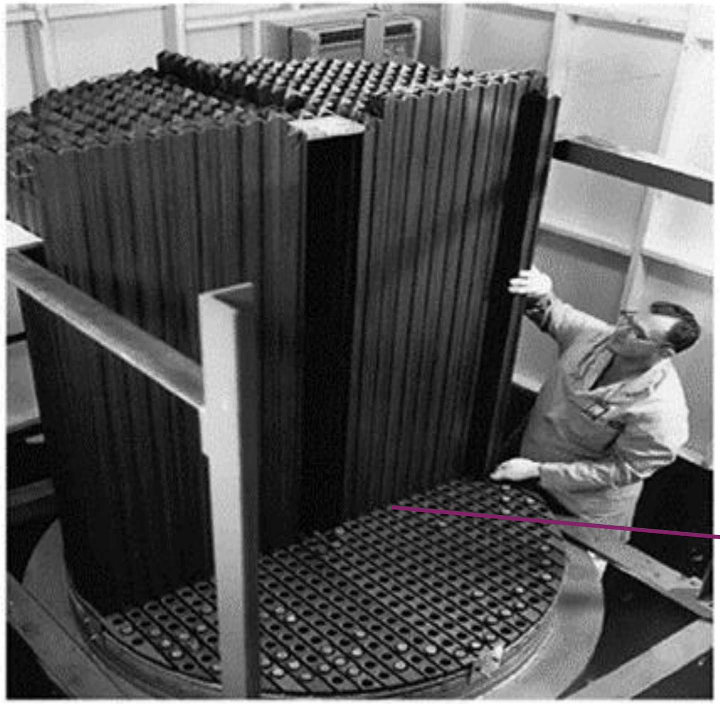
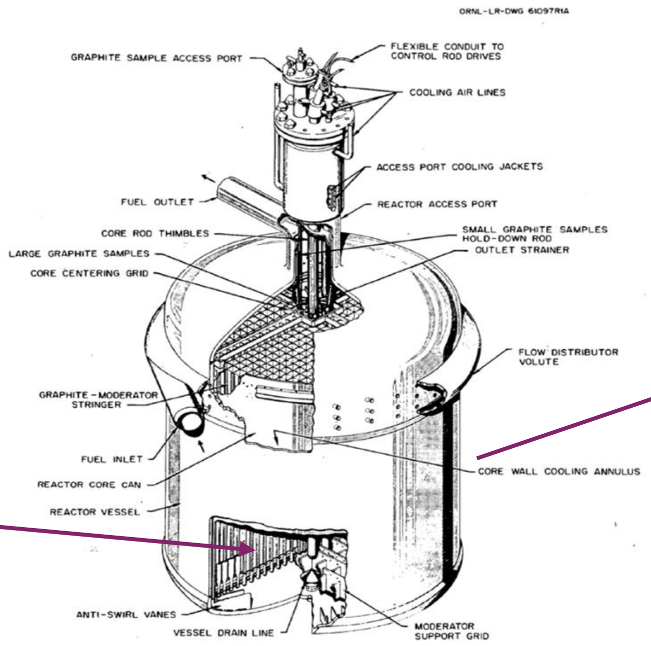
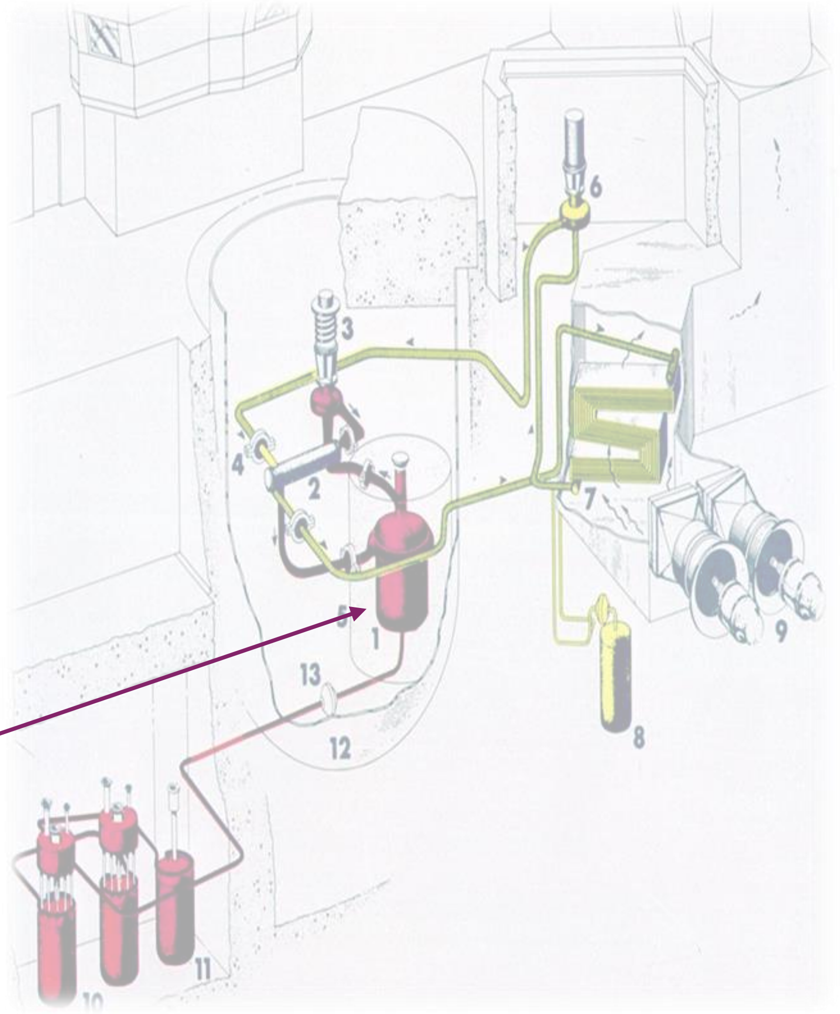
Molten Salt Reactor Experiment (MSRE)

- The Molten Salt Reactor Experiment (MSRE) was built in 1964 at ORNL
  - Thermal power of 10 MW.
  - Utilized a thermal neutron spectrum.
  - Liquid fuel salt flow into graphite moderator channels.
  - Operated with U-235 fuel then it was replaced with U-233 fuel.





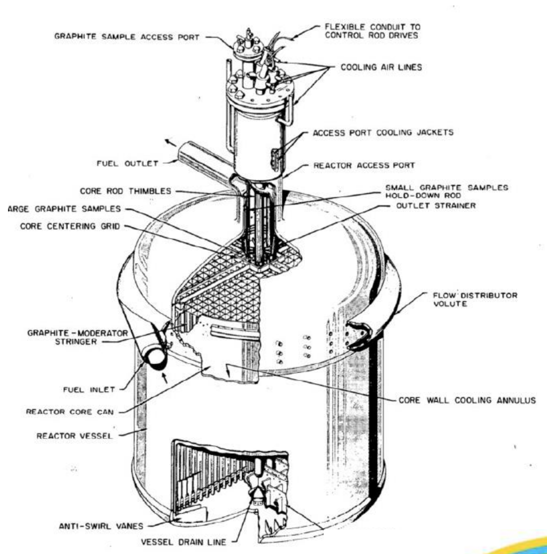


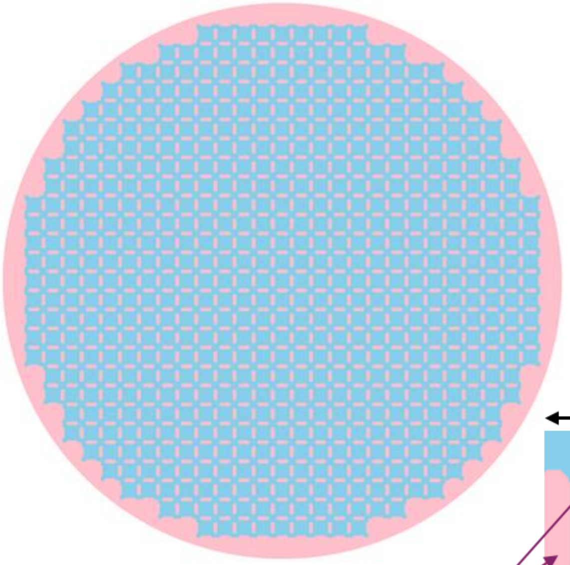
MSRE Modeling

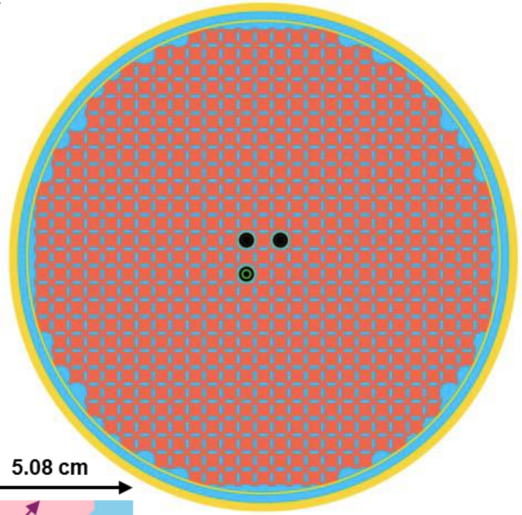
- **MSRE Specifications**
  - Core geometry and material prosperities.
  - Thermo-physical properties of fuel salt and graphite. (ORNL work on experience, database, correlations).

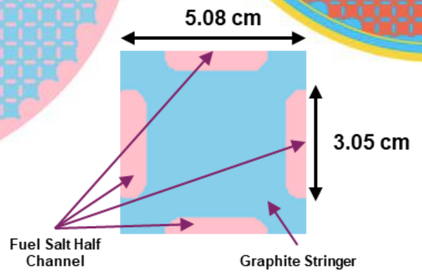
- **Multigroup Cross Sections**
  - Utilized OpenMC code to generate multigroup xs and delayed neutron data.
  - Converted OpenMC-Outputs into readable format by griffin (ISOXML).
  - OpenMC-XS-Griffin converter is available under Griffin code.

- **Simplified MSRE Model with Griffin/Pronghorn**
  - 2D model in RZ geometry.
  - Fuel and graphite in homogeneous mixture.
  - Steady-state and transient verification tests.









MSRE Specifications

MSRE Specifications & Thermal Properties

Thermal power	10 MWth
Fuel Salt	$LiF-BeF_2-ZrF_4-UF_4$
Molar composition	65.0%-29.1%-5.0%-0.9%
Enrichment	33.0%
Fuel inlet/ Outlet temperature	908 K / 936 K
Core height / Core diameter	1.63 m / 1.39 m
Fuel salt density [kg/m3] [@ 922K]	ρ(T)= 2263-0.4798x(T(K)-923.0) [2263.5]
Fuel salt dynamic viscosity [Pa·s]	μ=0.263371
Fuel salt thermal conductivity [W/m·K]	k=1.4
Fuel salt specific heat [J/kg·K]	C_p= 1868.0
Graphite density [kg/m3]	ρ_g=1860.0
Graphite thermal conductivity [W/m·K]	k_g=40.1
Graphite specific heat [J/kg·K]	$C_{p,g}$= 1757.3

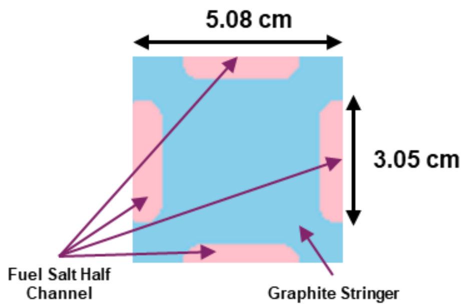
Fuel Salt composition

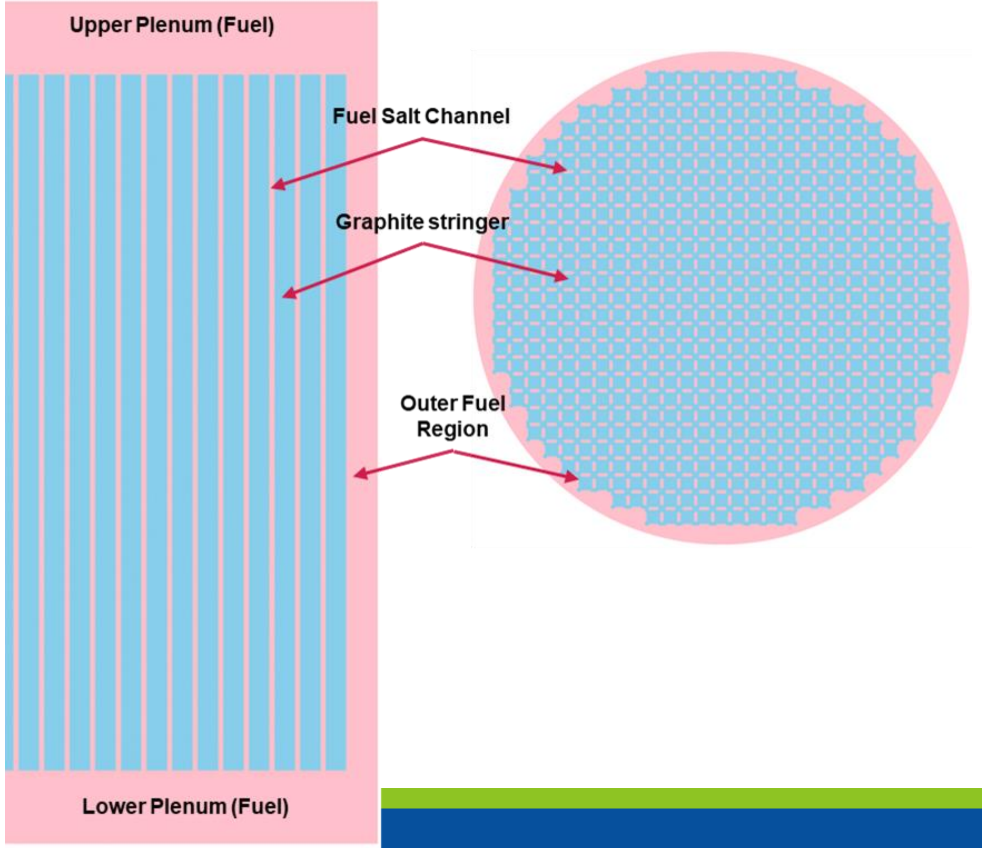
Isotope	**Atom Fraction**
Li-7	2.63371E-01
F-19	5.94814E-01
Be-9	1.17909E-01
Zr-90	1.04234E-02
Zr-91	2.27310E-03
Zr-92	3.47447E-03
Zr-94	3.52107E-03
Zr-96	5.67260E-04
U-235	1.20340E-03
U-238	2.44327E-03

MSRE Cross Sections Generation Model

- The MSRE lattice is made of vertical graphite stringers
- The fuel salt flows through a rectangular channel (3.05 cm x1.016 cm with round corners of radius 0.508 cm) in the sides of the stringers.
- 3D core model for MSRE was developed with OpenMC code:
  - MG XS generation.
  - Model verification.
- XS were generated in 16 groups energy structure and 6 delayed precursor groups.

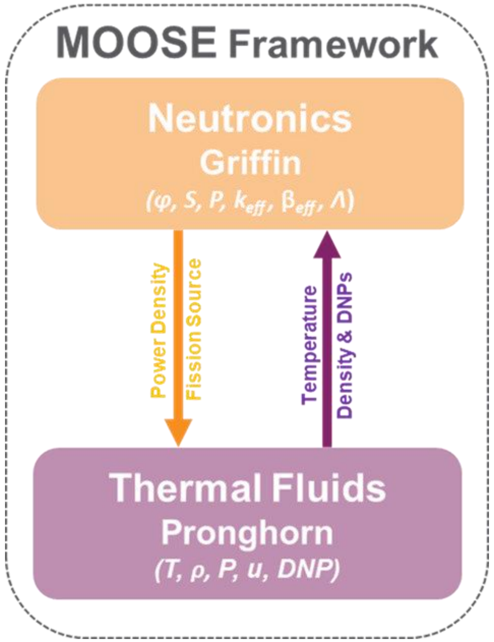
Core Height	197.3
Outer Core Diameter (Stringers +Outer Fuel)	148.7
Inner Core Diameter (Stringers)	139.58
No. Graphite Stringers	593
Size Stringers	5.08 cm x 5.08 cm x 163.0 cm

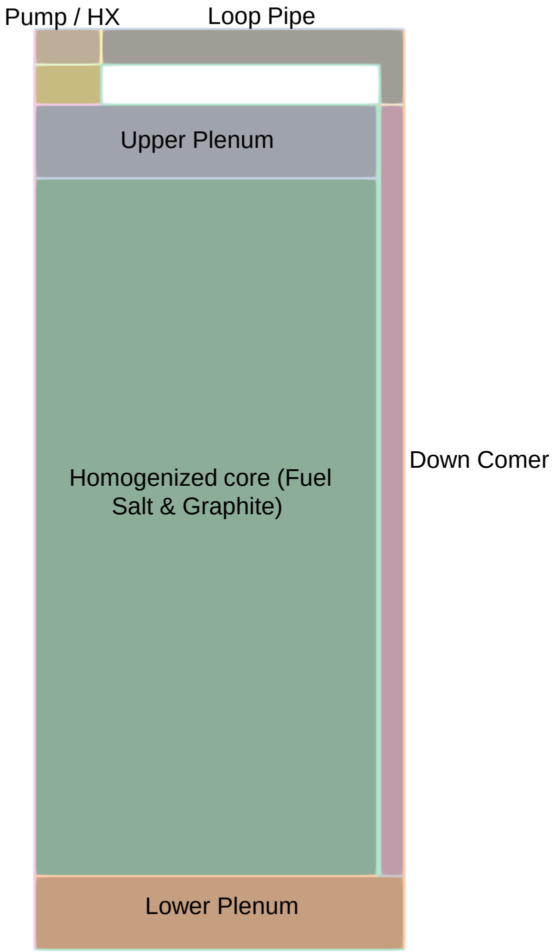


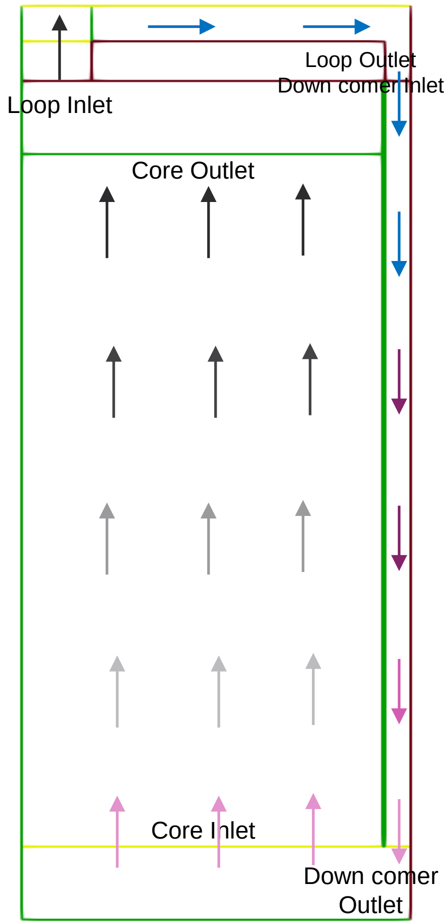


MSRE Multiphysics Model

- **A Multiphysics** model of the **MSRE** was developed in **RZ** geometry including the following components:
  - Neutronics core model of core: Griffin
  - Thermal hydraulics core & loop models: Pronghorn







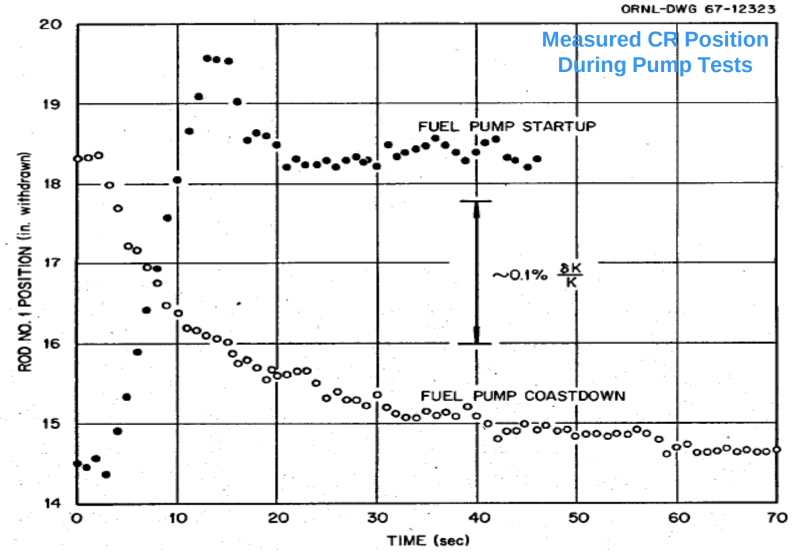
- **Three feedback mechanisms**:
  - **Temperature**: Fuel Salt and Graphite.
  - **Density**: concentration of the salt nuclides due to salt expansion.
  - **Velocity**: delayed neutron precursors distributions in core & outer loop.

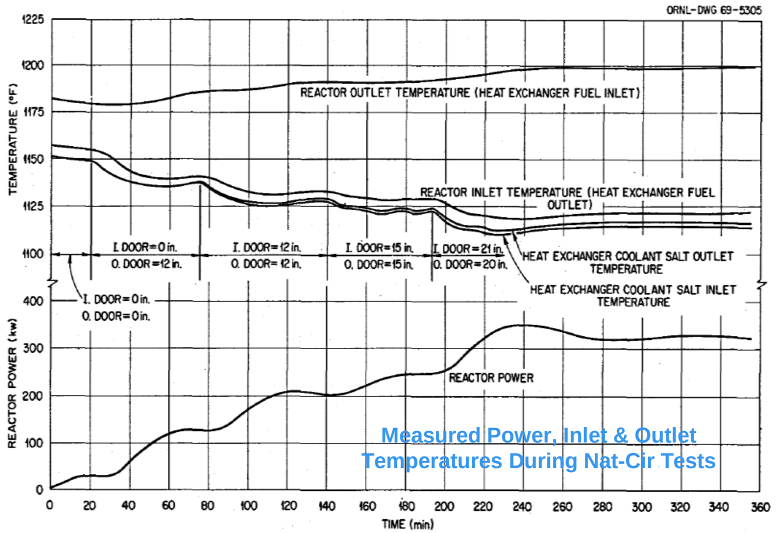
Test Results

Transient Tests

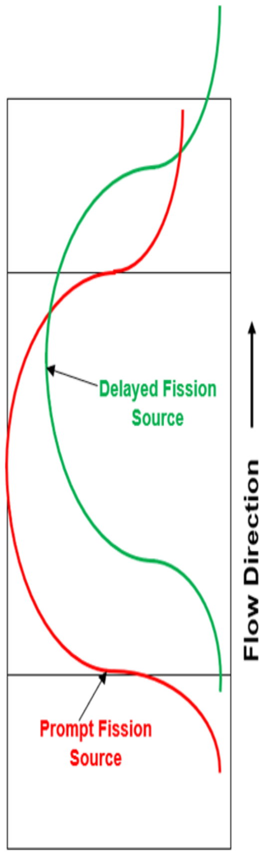
- **Validation Tests**
  - Pump Startup Test.
  - Pump Coast-Down Test.
  - Natural Circulation Test.

- **Variety of Verification Tests**
  - Pump Driven Transients
    - Loss of flow / coast down
    - Pump over speed / startup
  - Temperature Driven Transients
    - Loss of heat sink
    - Fuel salt over cooling
  - Reactivity Driven Transients
    - Control rod withdrawal
    - Fissile nuclide injection
    - Fertile nuclide injection
  - Localized Transients
    - Channel blockage/unblockage
    - Control rod withdrawal



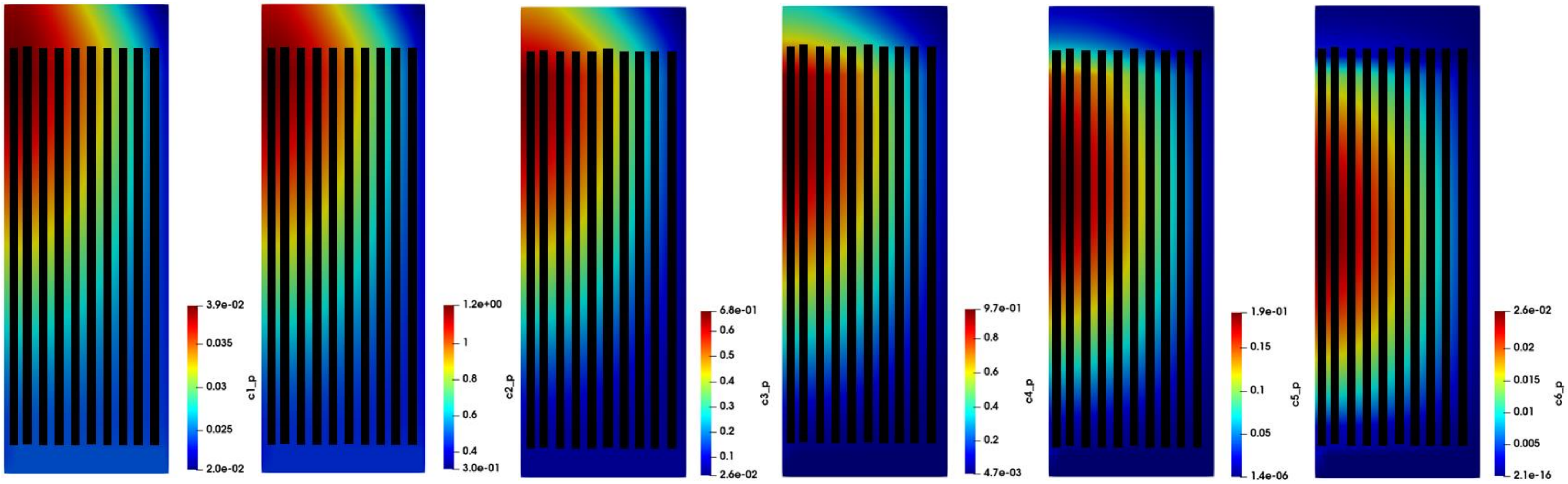


Delayed Neutron Precursors Steady State Solution



Long Lived → Short Lived

$\lambda_1 = 0.013\ s^{-1}$	$\lambda_2 = 0.033\ s^{-1}$	$\lambda_3 = 0.121\ s^{-1}$	$\lambda_4 = 0.303\ s^{-1}$	$\lambda_5 = 0.849\ s^{-1}$	$\lambda_6 = 2.853\ s^{-1}$
$T_{1/2\_1}$ = 51.976 s	$T_{1/2\_2}$ = 21.172 s	$T_{1/2\_3}$ = 5.739 s	$T_{1/2\_4}$ = 2.289 s	$T_{1/2\_5}$ = 0.816 s	$T_{1/2\_6}$ = 0.243 s



Flowing	$\beta_1 = 0.000006$	$\beta_2 = 0.000374$	$\beta_3 = 0.000535$	$\beta_4 = 0.001846$	$\beta_5 = 0.001048$	$\beta_6 = 0.000481$
Stationary	$\beta_1 = 0.000237$	$\beta_2 = 0.001218$	$\beta_3 = 0.001163$	$\beta_4 = 0.002592$	$\beta_5 = 0.001068$	$\beta_6 = 0.000447$
Losses	$\beta_1 = 0.000231$	$\beta_2 = 0.000844$	$\beta_3 = 0.000628$	$\beta_4 = 0.000746$	$\beta_5 = 0.000020$	$\beta_6 = 0.000034$

Calculated total reactivity losses due to fuel salt flow is 240 pcm.

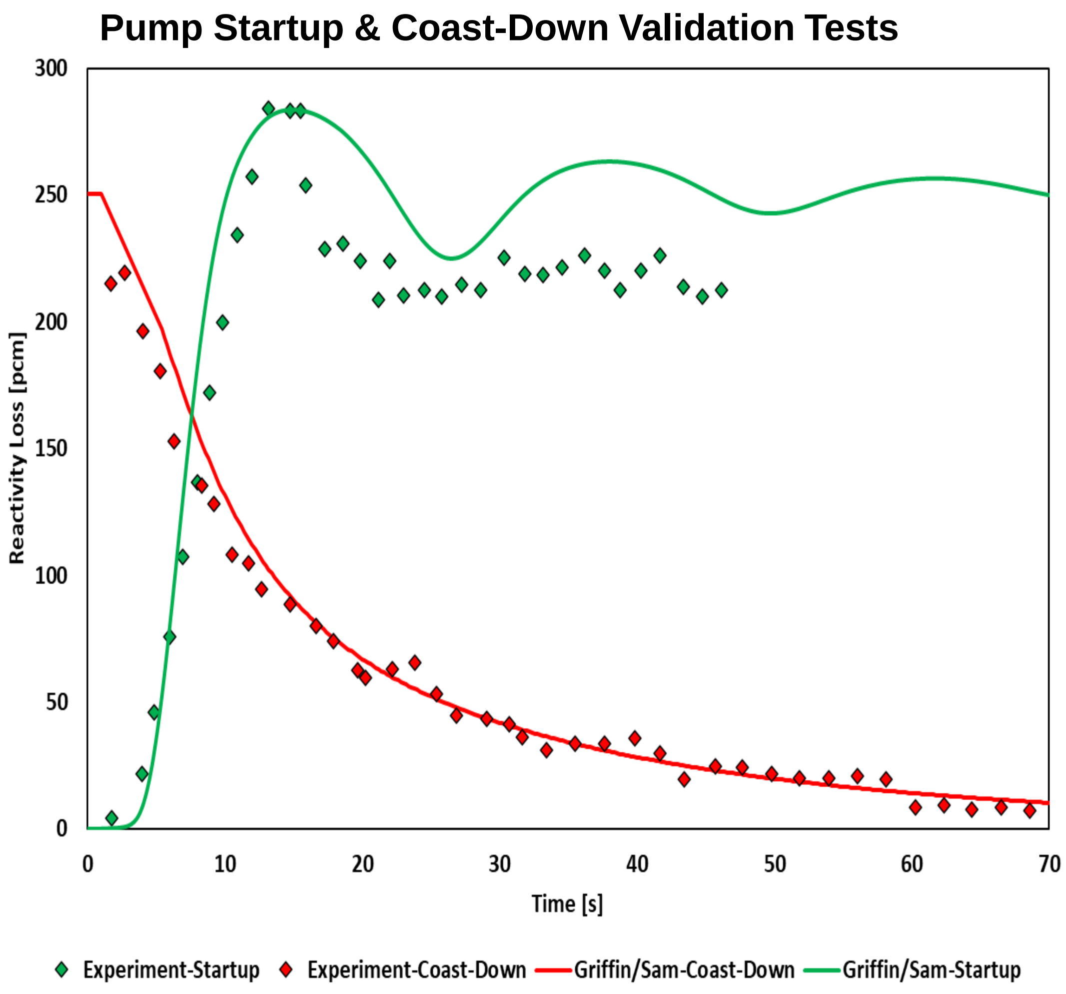
M. Jaradat, J. Ortensi, Thermal spectrum molten salt-fueled reactor reference plant model, Idaho National Laboratory, INL/RPT-23-72875, 2023.

Transient Validation Tests / Pump Transients

- **Pump Startup** & **Coast-Down** transient experiments were performed to demonstrate the impact of the delayed neutrons flow on reactivity.

- During the zero-power experiment:
  - Initial equilibrium conditions were established with minimal flow rate.
  - Reactor was at low or zero power level.
  - No temperature changes of the fuel salt and graphite.
  - **DNP losses (Fuel Salt Velocity)** is only feedback mechanism.
  - Criticality was maintained at zero power by adjusting control rod positions.
  - Reactivity inserted by control rods compensates reactivity changes due to redistribution of the DNPs in the core region.

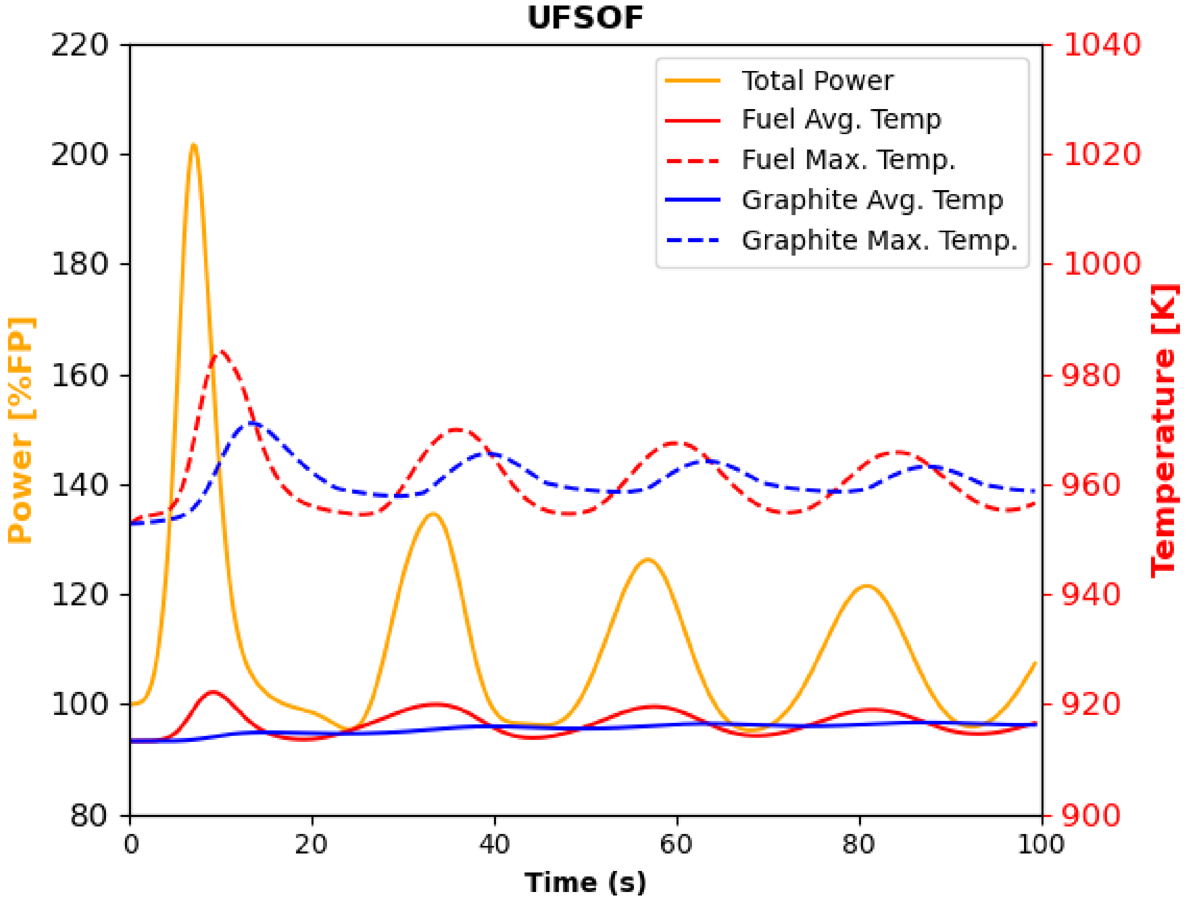
- Multiphysics transient simulations were performed to validate the developed model of MSRE against experimental data.
  - Initial equilibrium conditions were established.
  - Minimal flow **(0.01% Nominal Value**).
  - Zero power state (**0.01% Full Power**).
  - Both pump transient scenarios were performed with the same simulation.
  - Mass flow rate was adjusted similar to the measured values.

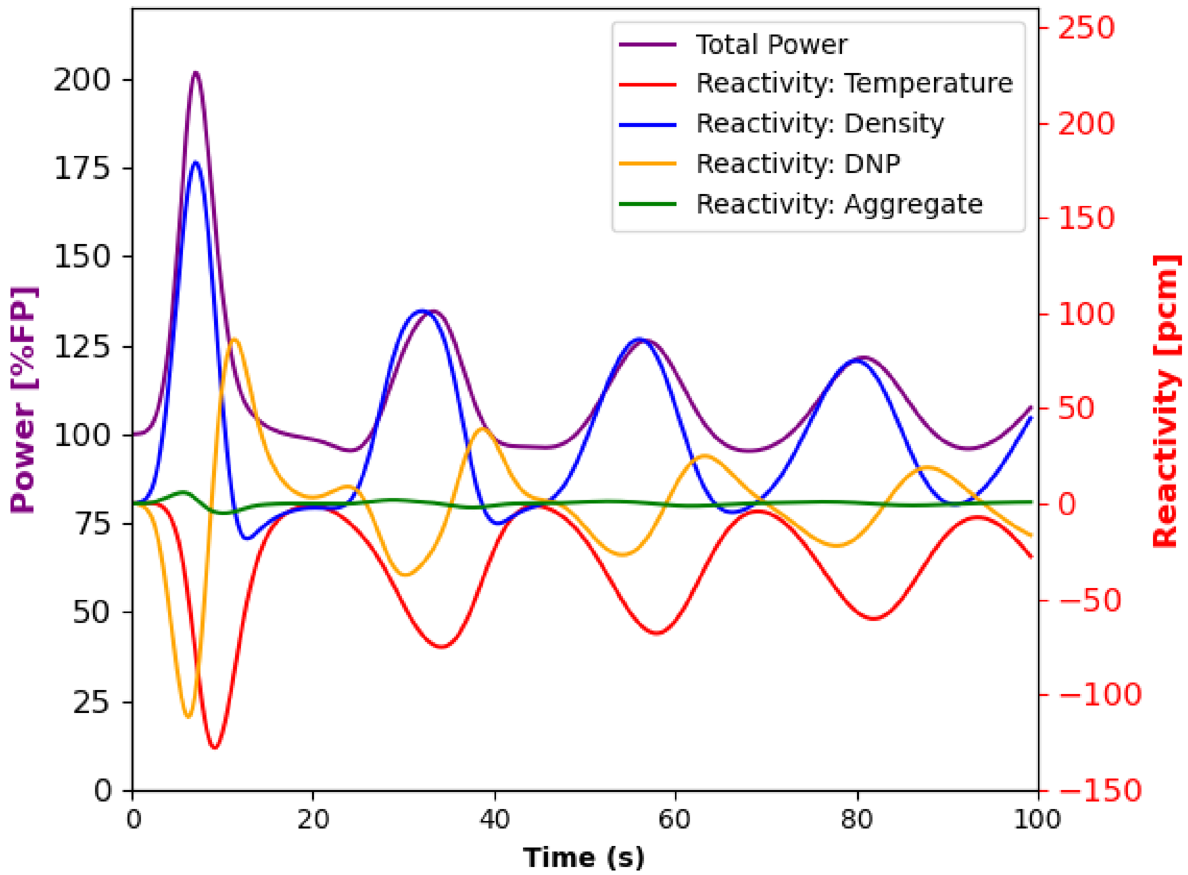


M. Jaradat, J. Ortensi, Thermal spectrum molten salt-fueled reactor reference plant model, Idaho National Laboratory, INL/RPT-23-72875, 2023.

Unprotected Fuel Salt Over Fueling

0.2 kg of U-235 in 4.0 s

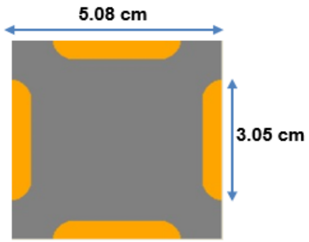




Cross Section Generation

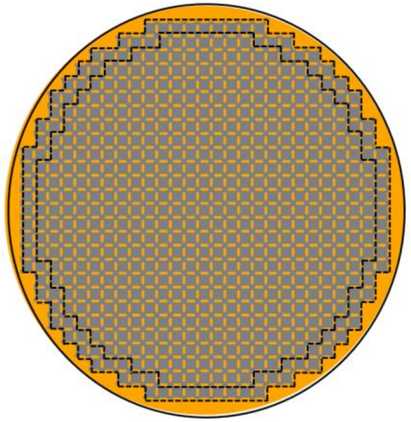
Selection of XS Generation Model

OpenMC Models for Multigroup Cross Sections Generation.

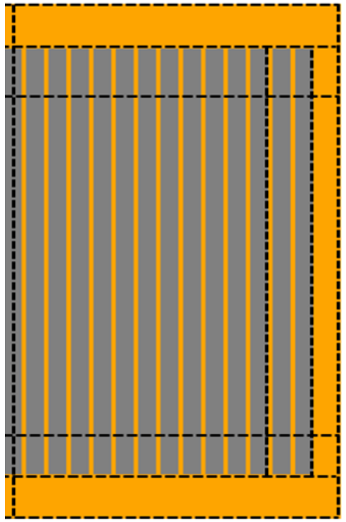


Inner Region Unit Cell

Outer Region Unit Cell



Radial Layout of 2-D and 3-D Models



Axial layout of 3-D Model

Comparison of Eigenvalues and Leakage Fractions Obtained with Different Cross Section Models

XS Model	Multiplication Factor	Diff. (pcm)	Leakage Fraction	Diff. (%)
OpenMC	1.06252 ± 0.00011	-	0.3362	-
3-D	1.06232	-19.6	0.3362	0.007
2-D	1.06487	235.9	0.3352	-0.294
Unit Cell	1.06993	741.4	0.3327	-1.035

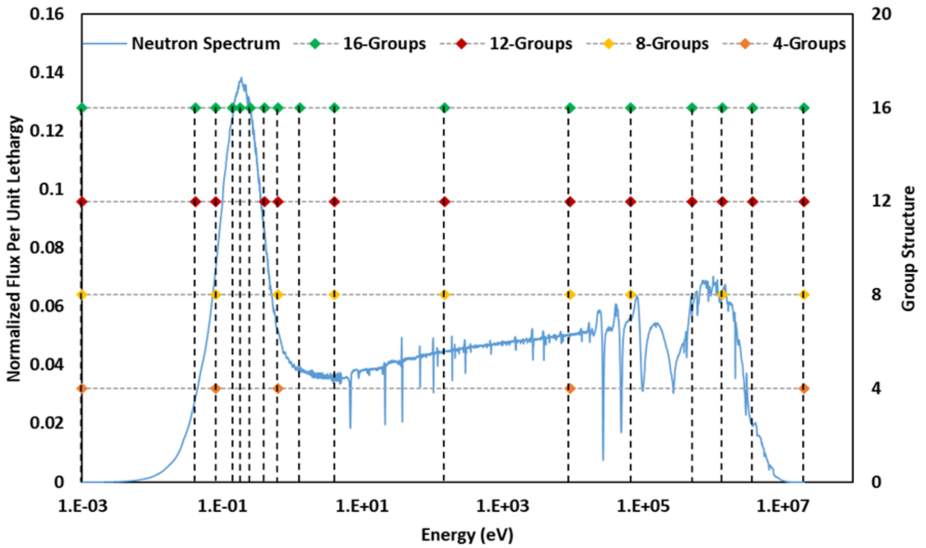
Reactivity Errors Analysis (pcm)

XS Model	Energy Range	Reaction/Leakage				
		Fission	Absorption	Leakage	Scattering	Sum
3-D	**Fast**	-0.1	0.0	-132.7	6.0	-126.8
	Resonance	-36.3	-34.0	67.3	0.0	-3.0
	Thermal	36.4	24.4	49.4	0.0	110.2
	Total	0.0	-9.6	-16.0	6.0	-19.6
2-D	**Fast**	1.2	2.0	-187.7	-2.1	-186.6
	Resonance	177.2	142.3	388.6	0.0	708.1
	Thermal	-178.3	-87.3	-19.9	0.0	-285.5
	Total	0.0	57.0	181.0	-2.1	235.9
Unit Cell	**Fast**	7.1	24.3	-9.3	-26.1	-3.9
	Resonance	233.0	245.7	601.3	0.0	1079.9
	Thermal	-240.1	-117.9	23.5	0.0	-334.6
	Total	0.0	152.0	615.4	-26.1	741.4

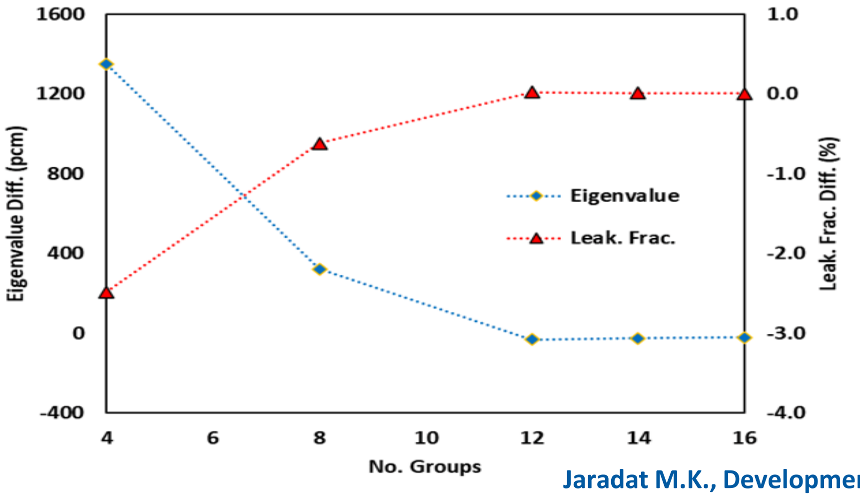
Jaradat M.K., Development of Neutronics Analysis Capabilities for Application to Flowing Fuel Molten Salt Reactors, Ph.D. thesis, University of Michigan (2021)

Selection of Energy Group Structure

Neutron spectrum in fuel salt and selected energy group structures.



Trends of eigenvalue and leakage fraction errors



Selected Energy Group Structures for MSRE Analysis

16-Groups	14-Groups	12-Groups	8-Groups	4-Groups
2.000E+07	2.000E+07	2.000E+07	2.000E+07	2.000E+07
3.679E+06	3.679E+06	3.679E+06		
1.353E+06	1.353E+06	1.353E+06	1.353E+06	
5.000E+05	5.000E+05	5.000E+05		
6.734E+04	6.734E+04	6.734E+04	6.734E+04	
9.118E+03	9.118E+03	9.118E+03	9.118E+03	9.118E+03
1.487E+02	1.487E+02	1.487E+02	1.487E+02	
4.000E+00	4.000E+00	4.000E+00	4.000E+00	
1.300E+00	1.300E+00			
6.250E-01		6.250E-01	6.250E-01	6.250E-01
4.000E-01	4.000E-01	4.000E-01		
2.500E-01	2.500E-01			
1.800E-01				
1.400E-01	1.400E-01			
8.000E-02	8.000E-02	8.000E-02	8.000E-02	8.000E-02
4.200E-02	4.200E-02	4.200E-02		
1.000E-03	1.000E-03	1.000E-03	1.000E-03	1.000E-03

Jaradat M.K., Development of Neutronics Analysis Capabilities for Application to Flowing Fuel Molten Salt Reactors, Ph.D. thesis, University of Michigan (2021)

Idaho National Laboratory

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