Module 4: MSR Neutronics

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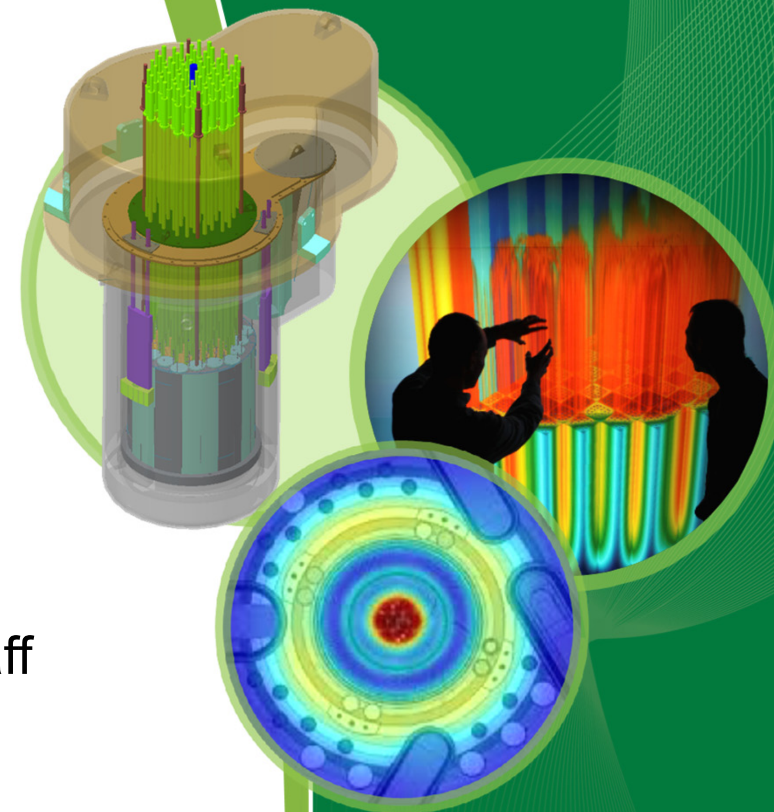

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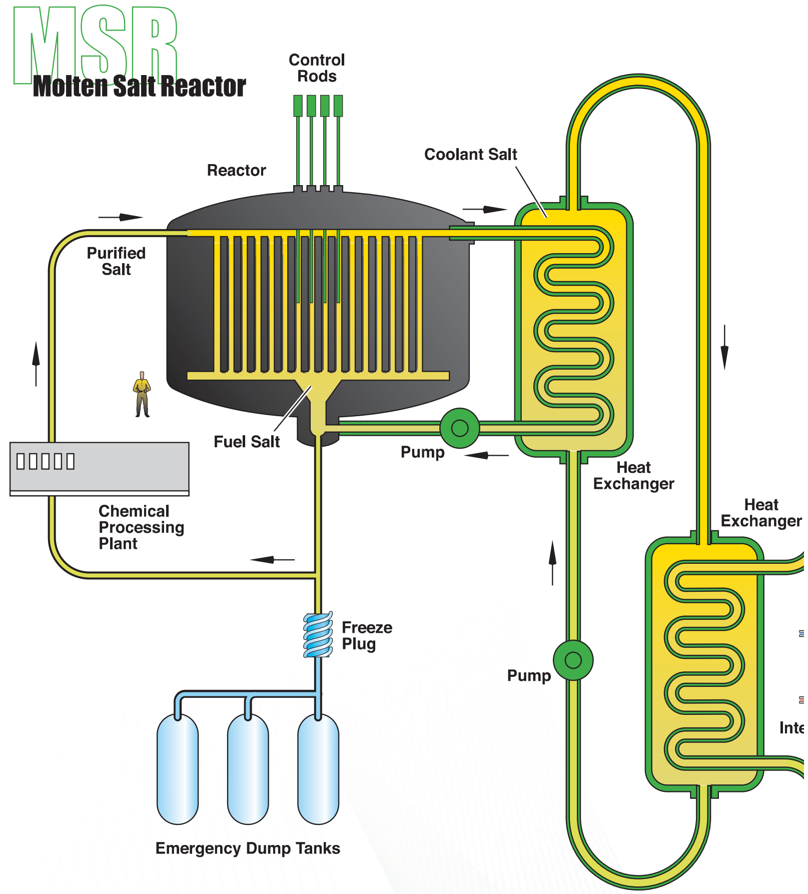
Overview

- Applications and advantages of MSRs
- Neutron flux spectrum characteristics
- Neutronic aspects of liquid fueled reactors that are different from solid fueled reactors
  - Delayed neutron precursor motion
  - Fission product removal
  - Fission gas bubble flow
- Reactivity feedback effects in MSRs
- Challenges
  - Nuclear data availability and uncertainty
  - Modeling tools, group structures, etc.

Liquid-fueled Molten Salt Reactors:
Unique Reactor Physics Characteristics

- Liquid fuel reactor as a chemical plant
  - Simplifying the handling and reprocessing of fuel
  - Fuel (and delayed neutrons) flows around primary loop
  - Continuous production of gaseous fission and transmutation products in the salt
- Complex chemical processes
  - Online removal of fission products (e.g., sparging)
  - Online or batch feed of fissile material
  - Batch discard of fuel material
- Thermal spectrum and fast spectrum MSRs are possible
  - Fluoride and chloride salts
  - FLiBe salt and graphite moderator are "classic" thermal MSR configuration



Source: *A Technology Roadmap for Generation IV Nuclear Energy Systems*. GIF-002-00.

Why Liquid Fuel Molten Salts?

- Enables high temperature at low pressure
- Online chemistry adjustment
  - Can include fuel processing
- Potential for inherent safety depending on design options
  - Fuel salt thermal expansion provides negative reactivity insertion
  - Fuel draining under thermal excursions
  - Low excess reactivity – fuel normally in most reactive configuration
- Potential to substantially reduce actinide waste production
  - Eliminates requirement for precision fuel fabrication
- MSRs can be refueled as “infinite batch” reactors
  - Results in <u>maximum possible burnup</u>

Neutronics advantages of MSRs

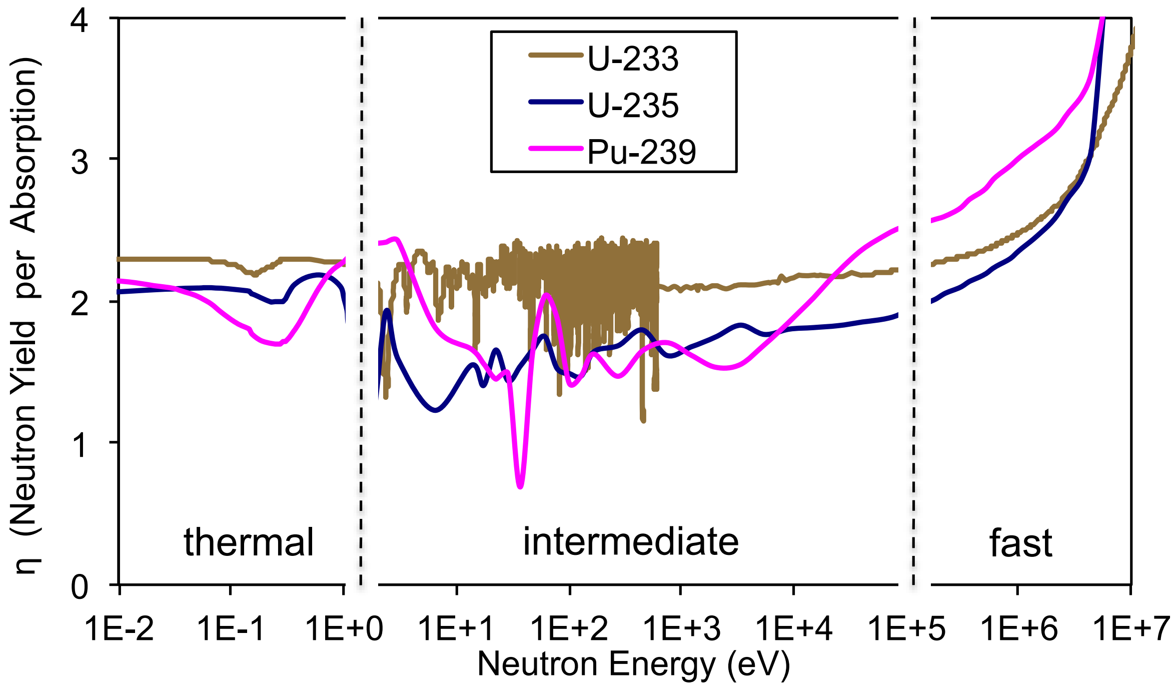
- Online refueling and reprocessing
- Excellent neutron economy
- Low absorption materials and no cladding
- Online criticality maintenance
  - High availability
- Flexible fuel composition
  - Without blending and fabrication
  - Enables actinide recycling
- Excess neutrons
  - Thorium breeding and/or actinide burning
  - Fixed fuel cost
- Fuel presence in salt
  - Negative thermal feedback coefficient
- Low source term
  - Low radiotoxic risk
- Low fuel load
  - Low excess reactivity

Safety, Economics, Sustainability

Source: J. Křepel et al. 2014. "Fuel cycle advantages and dynamics features of liquid fueled MSR," *Annals of Nuclear Energy* 64: 380-397.

MSRs Are Flexible Fuel Cycle Machines

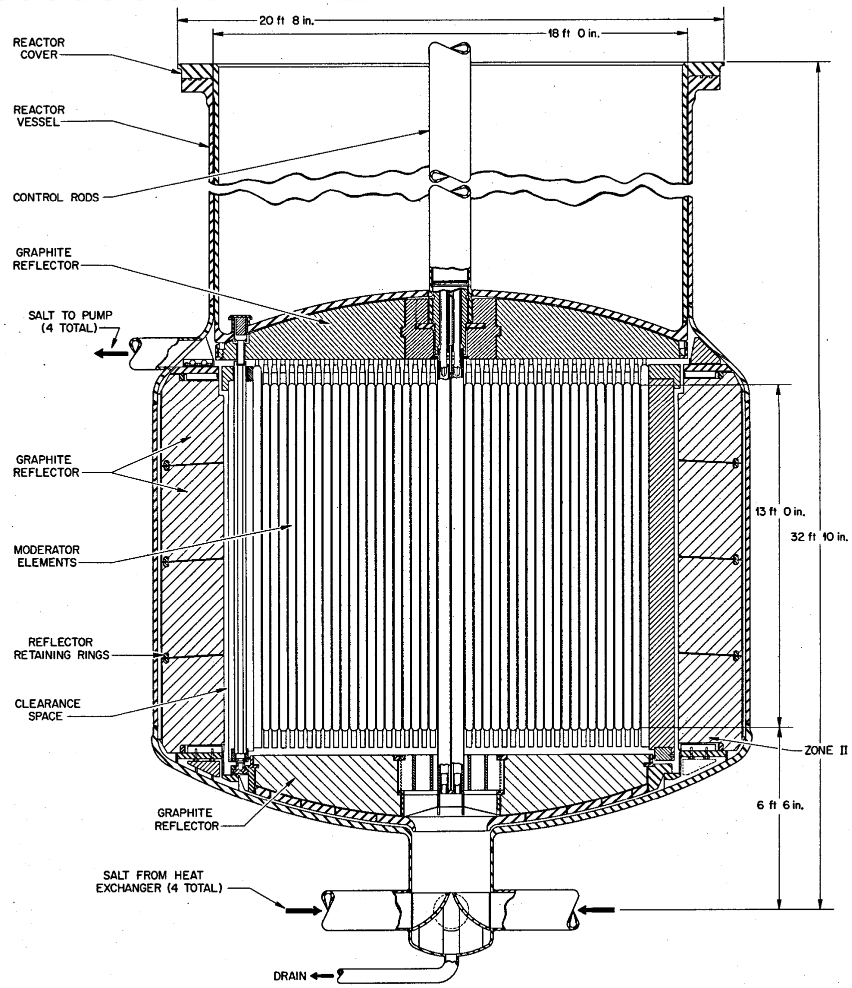
- MSRs may be operated with a variety of fissile feed materials, as burner, breeder, or self-sustaining reactors
- LEU, Th/^{233}U, U/Pu, U/TRU, etc.
- MSRs can breed ^{233}U from ^{232}Th in any spectrum: thermal, intermediate or fast



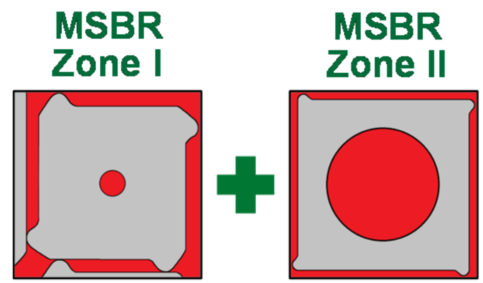
Source: N. R. Brown et al. 2015. "Sustainable thorium nuclear fuel cycles: A comparison of intermediate and fast neutron spectrum systems." *Nuclear Engineering and Design* 289: 252-265.

Two-zone MSBR Geometry Design Example

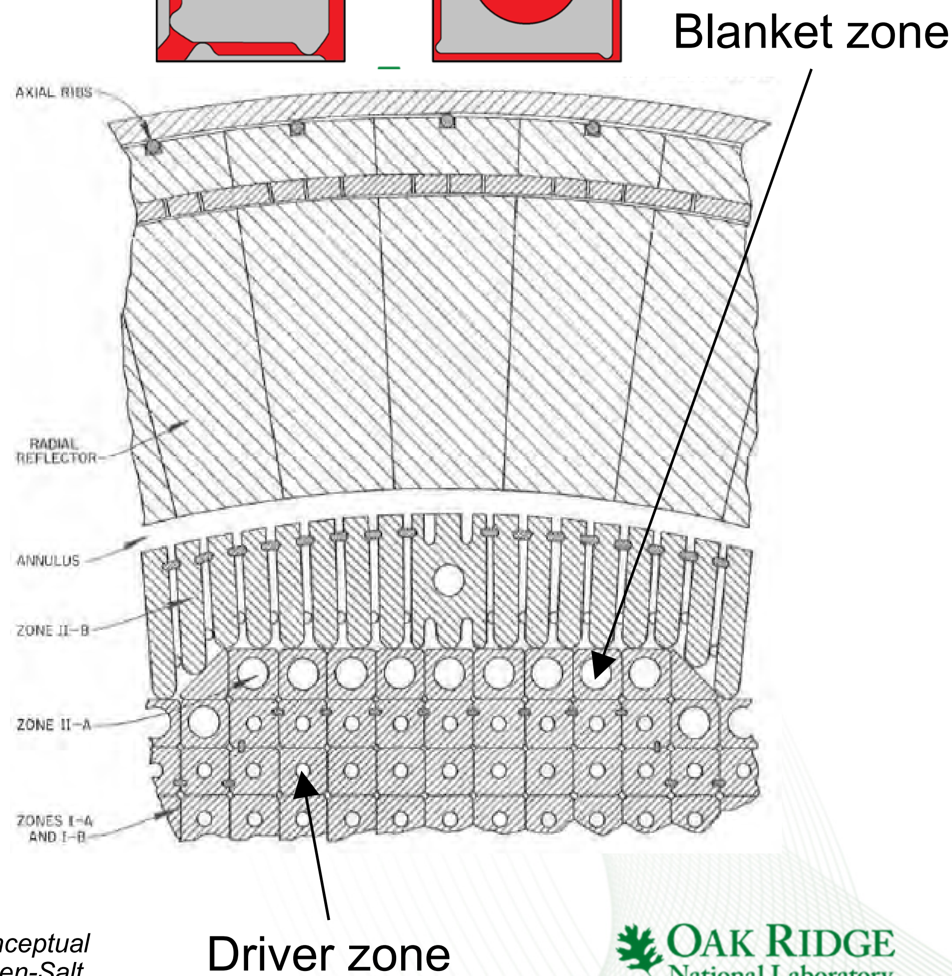
Fissile fuel is “bred” in the blanket channels



Source: R. C. Robertson. 1971. *Conceptual Design Study of a Single-Fluid Molten-Salt Breeder Reactor*. ORNL-4541

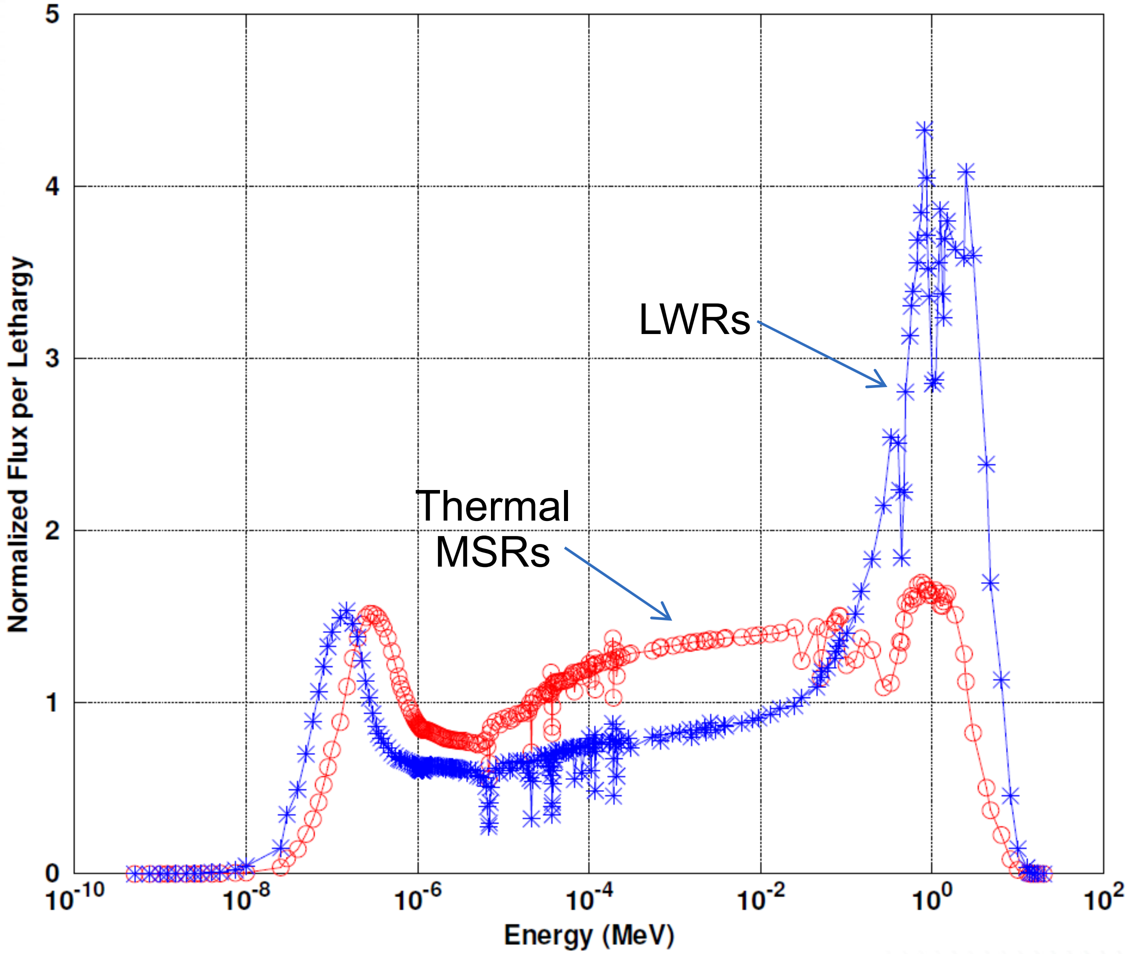


Source: J.J. Powers et al. “An Inventory Analysis of Thermal-Spectrum Thorium-Fueled Molten Salt Reactor Concepts.” *PHYSOR 2014*



Key Differences in LWR and MSR Flux Spectrum

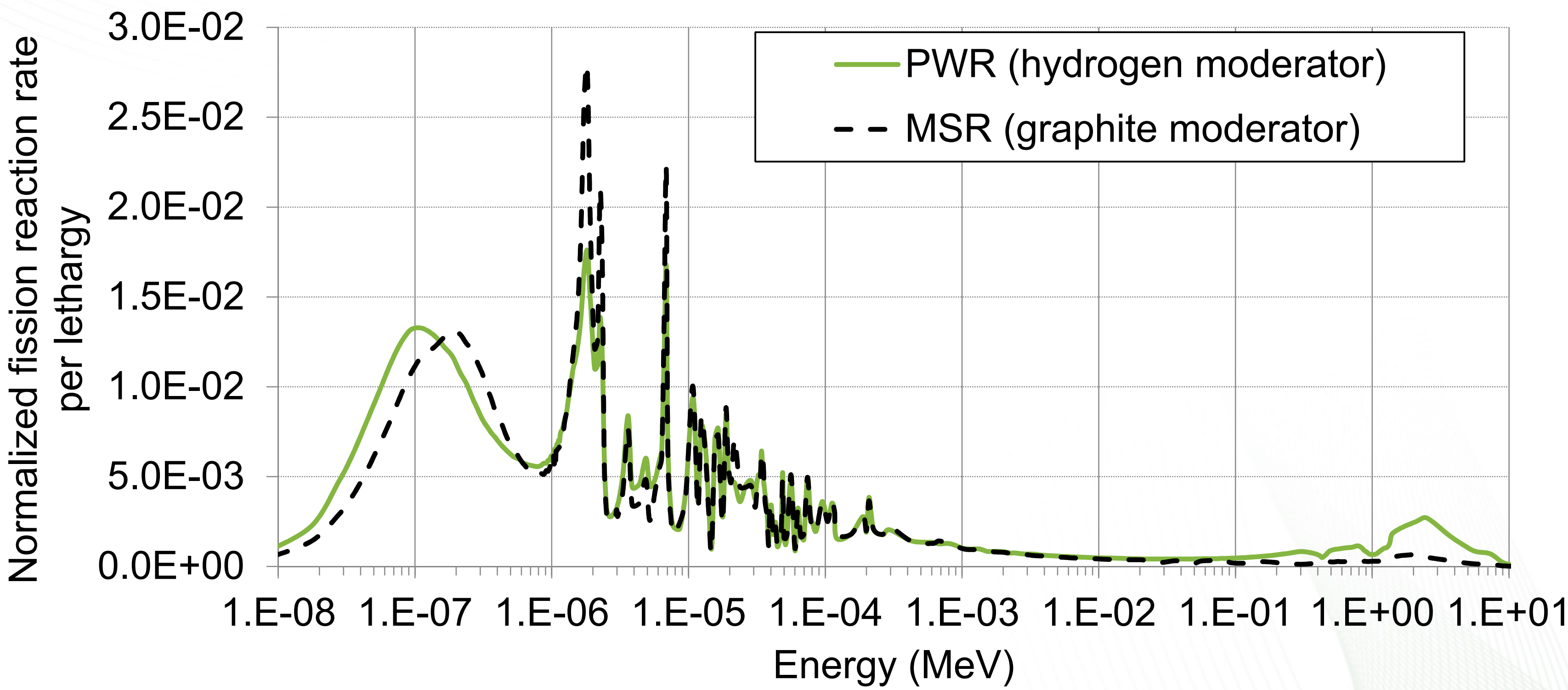
- Typical LWR diffusion length (6 cm) vs. typical fluoride salt MSR diffusion length (16 cm)



C.A. Gentry. 2016. *Development of a Reactor Physics Analysis Procedure for the Plank-Based and Liquid Salt-Cooled Advanced High Temperature Reactor*. Dissertation, University of Tennessee Knoxville.

Fission Reaction Rate Spectrum of MSR versus Typical PWR

- Graphite moderator hardens fission reaction spectrum
- Graphite lifetime is an important consideration in thermal spectrum MSRs

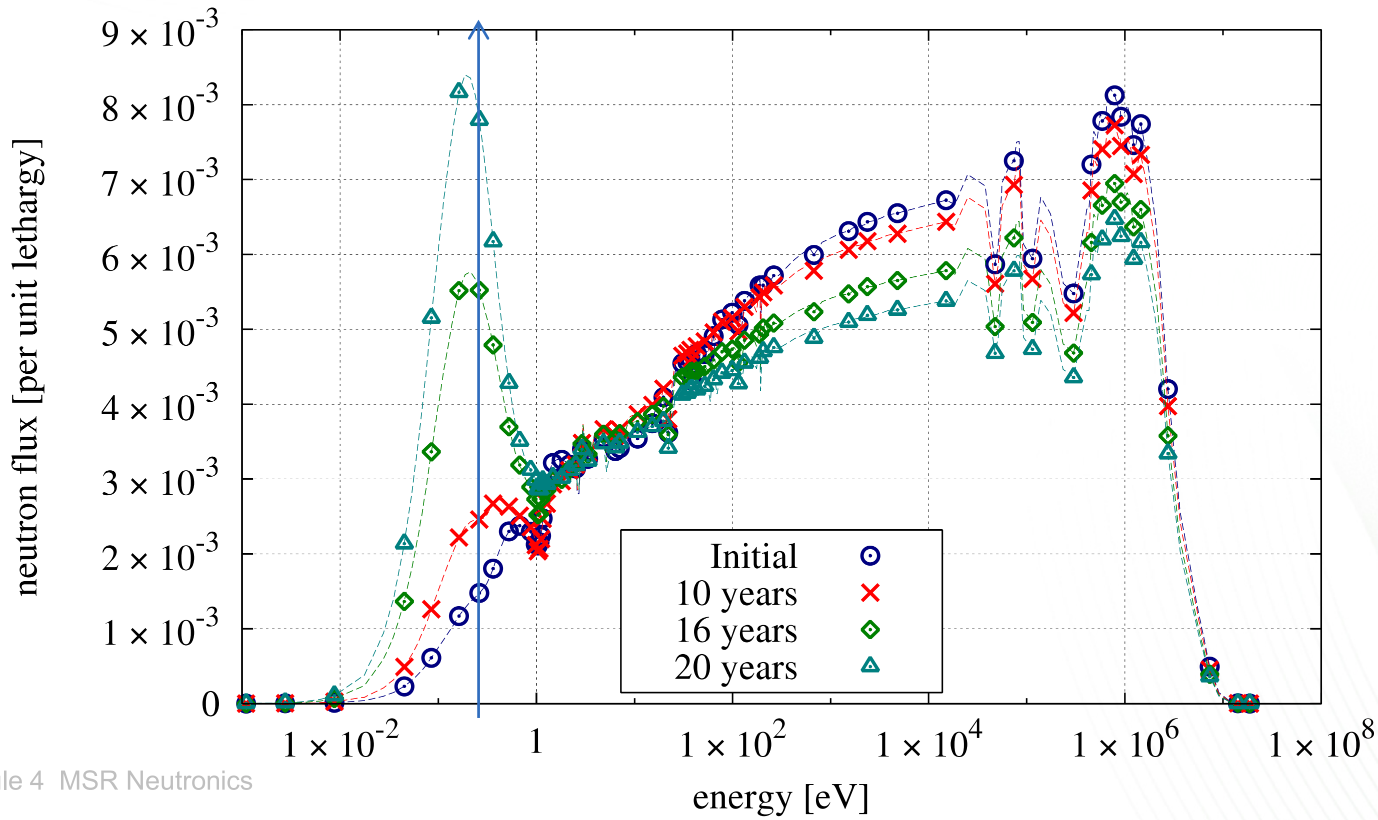


Source: N. R. Brown et al. 2015. "Sustainable thorium nuclear fuel cycles: A comparison of intermediate and fast neutron spectrum systems." *Nuclear Engineering and Design* 289: 252-265.

Neutron Flux Spectrum of MSRs (cont.)

- The neutron flux spectrum of MSRs can vary significantly as a function of energy, even for the same design
- Example is the startup of a thorium fuel cycle using U/Pu from spent nuclear fuel

Spectrum softens during transition from U/Pu to Th/^{233}U fuel



Source: B. Betzler et al. 2016. “Modeling and Simulation of the Start-up of a Thorium-Based Molten Salt Reactor,” in *Proceedings of PHYSOR 2016*

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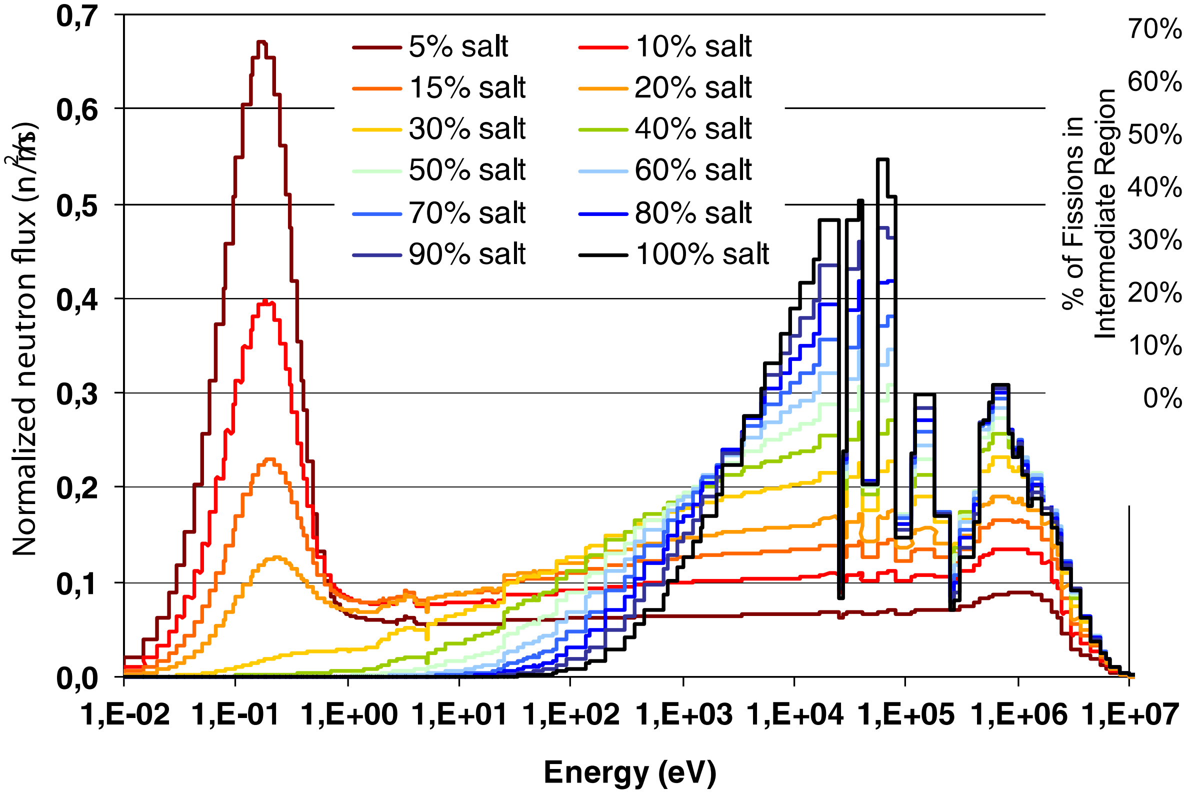
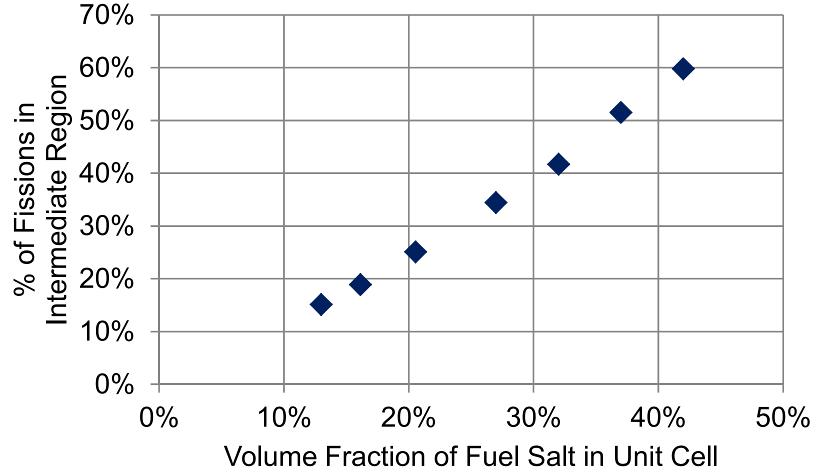
Fuel Salt versus Moderator Ratio

- Neutron flux spectrum shifts as fuel salt is added to the system and moderator is removed
- Enrichment is adjusted to maintain criticality in these examples



Source: J. Křepel et al. 2014. “Fuel cycle advantages and dynamics features of liquid fueled MSR,” *Annals of Nuclear Energy* 64: 380-397. (Used with permission from Elsevier)



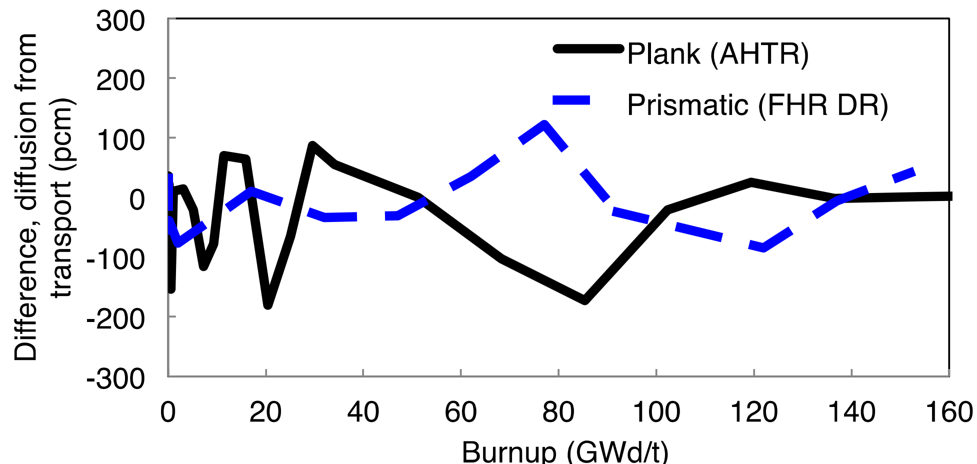
Source: N. R. Brown et al. 2015. "Sustainable thorium nuclear fuel cycles: A comparison of intermediate and fast neutron spectrum systems." *Nuclear Engineering and Design* 289: 252-265.

MSR Spectrum: Challenges

- Although diffusion calculations have been shown to work well for MSRs, fine energy group and few energy group structures are not well defined
- These group structures would need to be developed for each MSR type
- For thermal spectrum (graphite moderated, fluoride salt) MSRs with LEU fuel, 4-group structure developed for FHRs may be a good starting point

Group #	Upper Bound	Lower Bound
1	2.0000E+01	9.1188E-03
2	9.1188E-03	2.9023E-05
3	2.9023E-05	7.3000E-07
4	7.3000E-07	1.0000E-12

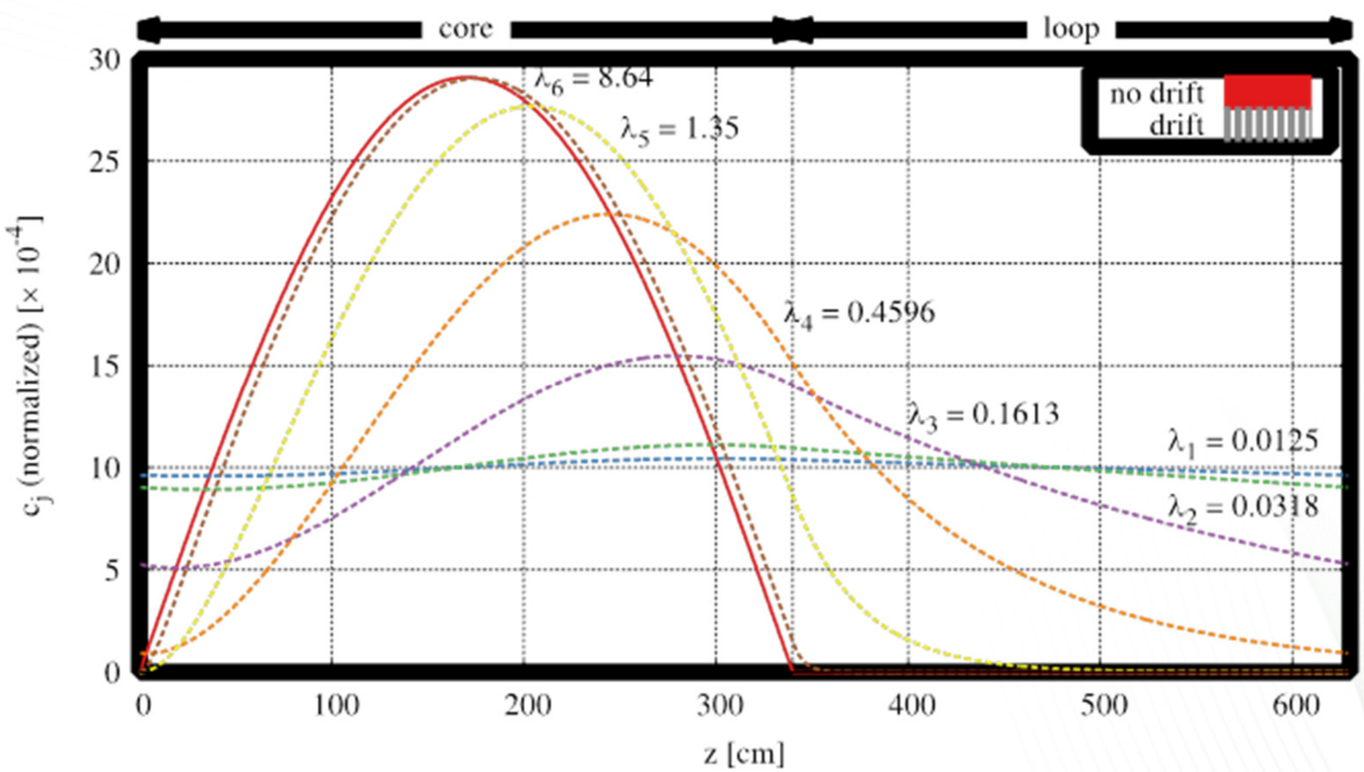
Source: C. Gentry, G. I. Maldonado, and K. S. Kim. 2016. “Development of a Two-Step Reactor Physics Analysis Procedure for Advanced High Temperature Reactors,” in *Proceedings of PHYSOR 2016: Unifying Theory and Experiments in the 21st Century.*



Source: N.R. Brown et al. 2016. “Preconceptual design of a fluoride high temperature salt-cooled engineering demonstration reactor: core design and safety analysis.” *Annals of Nuclear Energy* 103: 49-59.

Delayed Neutron Precursor Drift

- Because the fuel is flowing, approximately 50% of delayed neutrons are generated outside of the core region
- This impacts the value of β and the controllability of the reactor



Source: B. Ade, B. Betzler, et. al., *MSR Modeling Tools: Past, Present and Future*, presented at the Advanced Reactor Working Group Modeling & Simulation Workshop, EPRI, Charlotte, NC, January 24-25, 2017.

Consequences of Moving Fuel in MSRs

- Fuel carries delayed neutron precursors out of the core
  - Solid fuel reactors are critical due to delayed neutrons emitted from precursor decay (fundamental *α* eigenvalue is limited by the precursor decay constants and is on the order of s^{-1})
  - Without delayed neutron precursors, the reactor is uncontrollable (prompt *α* eigenvalues are much greater in magnitude than precursor decay constants)
- Fission source calculated by standard lattice physics codes is <u>biased</u>
  - Prompt neutrons and some delayed neutrons are emitted in the liquid fuel while it is in the core
  - Some delayed neutrons are emitted after the liquid fuel leaves the core (coolant loop, chemical processing, etc.)
  - Neutronics tools need delayed neutron convection term to model fission source for MSRs

Fission Product Removal

- Some MSR designs are intended to actively separate fission and/or transmutation products
- Even if there is no active separation, there will be passive separation, e.g., noble gas fission products
- Fission product gas bubbles may impact reactor stability
  - Although MSRE was shown to be stable during operation

Modeling and Simulation of MSRs:
Depletion (Bateman) Equations

- ORIGEN solves a set of depletion equations using fluxes provided from a transport calculation
- These equations describe the rate of change of the nuclides in the problem

$$\frac{dN_i}{dt} = \sum_{j=1}^{m} l_{ij}\lambda_j N_j + \bar{\Phi}\sum_{k=1}^{m} f_{ik}\sigma_k N_k - (\lambda_i + \bar{\Phi}\sigma_i + \cancel{r_i})N_i$$

($r_i \rightarrow 0$)

Decay rate of nuclide *j* into nuclide *i*	Production rate of nuclide *i* from irradiation	Loss rate of nuclide *i* due to decay, irradiation, or other means

- For a solid fuel reactor, the fuel is stationary; there is no additional removal or feed term

Source: B. Betzler et al. 2016. “Modeling and Simulation of the Start-up of a Thorium-Based Molten Salt Reactor,” in *Proceedings of PHYSOR 2016.*

Modeling and Simulation of MSRs:
Depletion (Bateman) Equations

- For a liquid fuel reactor, the additional removal/feed term is likely nonzero
  - Represents removal of fission products, addition of fertile and fissile material, etc.
  - Must be expressed in terms of a decay constant
  - An accurate removal/feed rate must take into account liquid fuel flow rates and reactor design

$$\frac{dN_i}{dt} = \sum_{j=1}^{m} l_{ij}\lambda_j N_j + \bar{\Phi}\sum_{k=1}^{m} f_{ik}\sigma_k N_k - (\lambda_i + \bar{\Phi}\sigma_i + r_i)N_i$$

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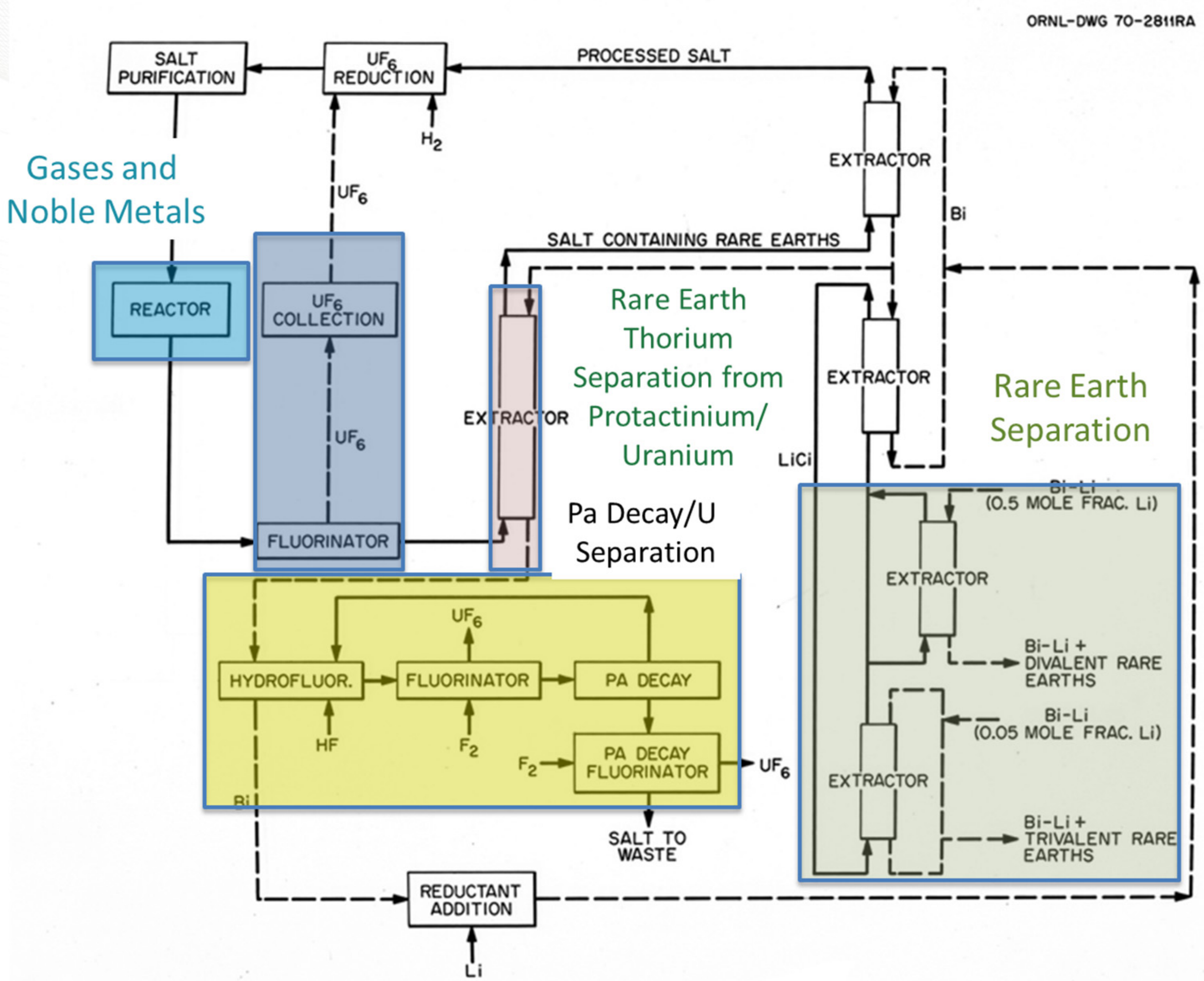
Example MSR Separation Processes



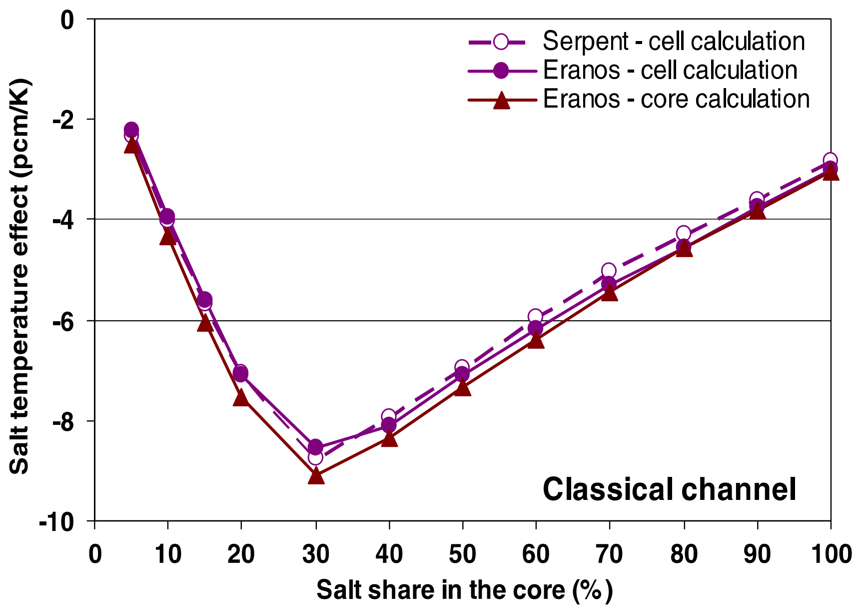
Source: R. C. Robertson. 1971. *Conceptual Design Study of a Single-Fluid Molten-Salt Breeder Reactor*. ORNL-4541

Reactivity Feedback Effects

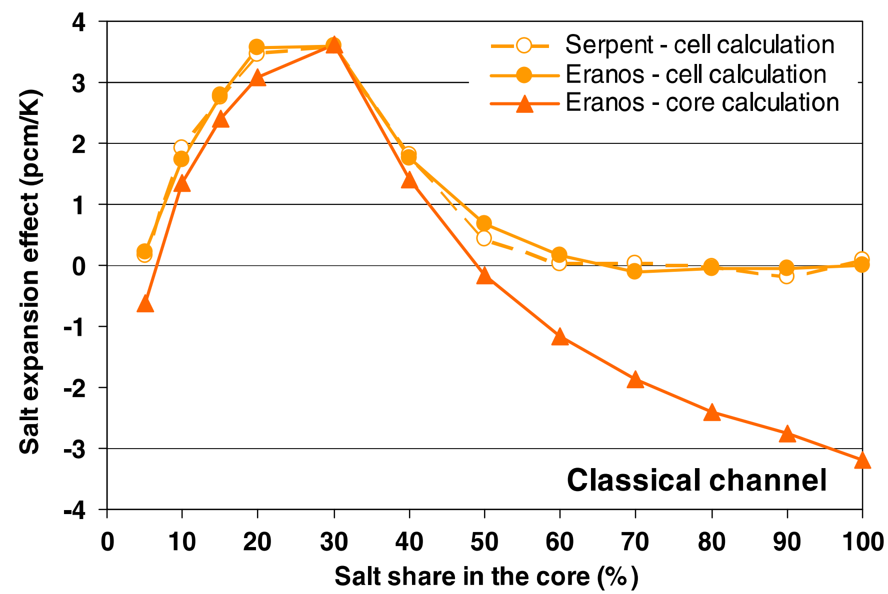
- Fuel salt temperature (spectral) and density
  - Net negative (density component may be positive or negative)
- Moderator temperature
  - May be negative or positive
- Moderator thermal expansion
  - Negative, but longer time scale
- Changes in flow rate
  - Stable, depending on design

Example Fuel Salt Temperature and Density Reactivity Feedback Effects

- Net effect is negative, driven by strongly negative fuel temperature spectral effect
- Density component can sometimes be positive



Spectral Effect

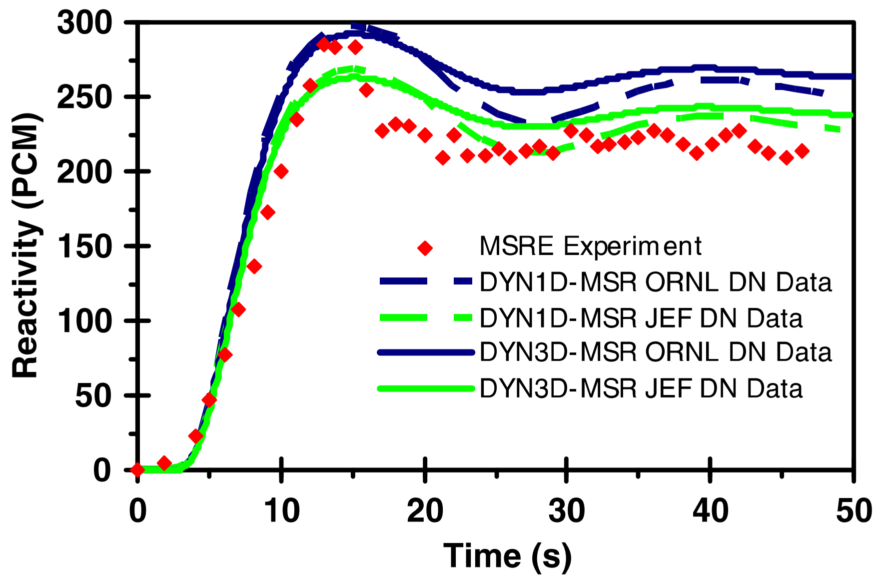


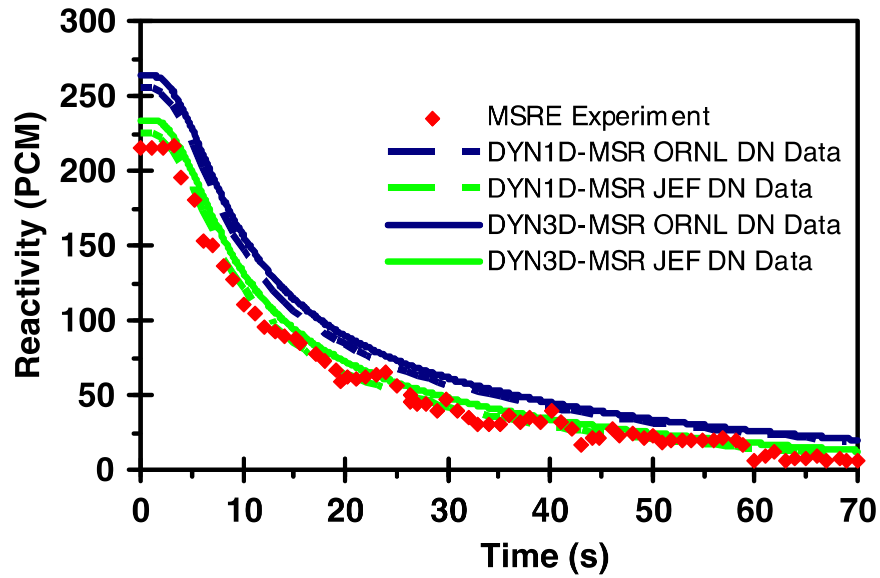
Density Effect

Source: J. Křepel et al. 2014. “Fuel cycle advantages and dynamics features of liquid fueled MSR,” *Annals of Nuclear Energy* 64: 380-397. (Used with permission)

Reactivity Effects of Delayed Neutron Precursor Drift (1/2)

- Experimental observations from MSRE and model predictions for fuel pump start-up and coast-down transients
- Results from DYN3D German nodal kinetics code in two groups, similar to US NRC code PARCS
- US NRC code PARCS needs modification for delayed neutron precursor motion

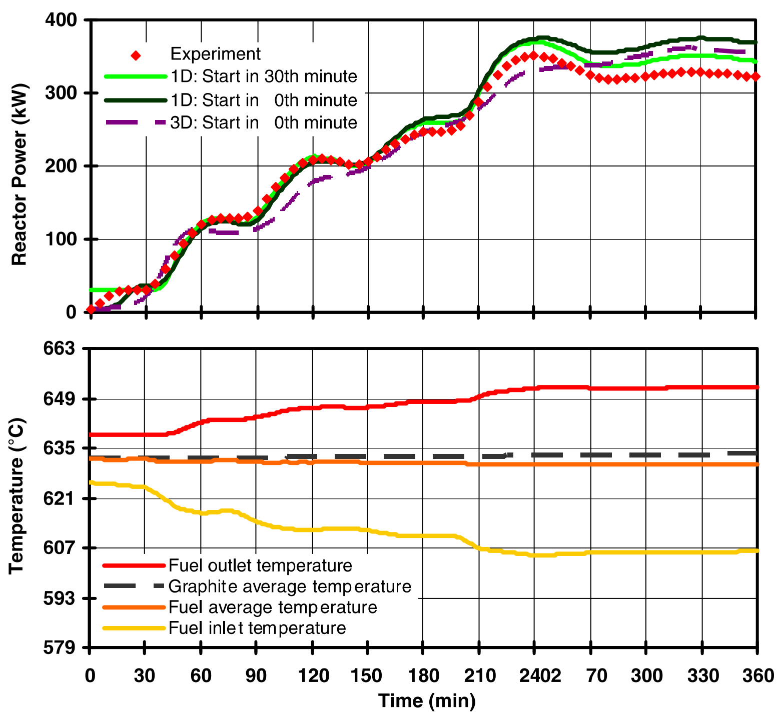




Source: J. Křepel et al. 2007. "DYN3D-MSR spatial dynamics code for molten salt reactors." *Annals of Nuclear Energy* 34: 449-462.

Reactivity Effects of Delayed Neutron Precursor Drift (2/2)

- Experimental observations from MSRE and model predictions for natural circulation transient
- This example shows that neutronics codes (DYN3D) with the fidelity of the US NRC code PARCS can accurately predict passive safety performance of MSRs (if modified for precursor drift)



Source: J. Křepel et al. 2007. "DYN3D-MSR spatial dynamics code for molten salt reactors." *Annals of Nuclear Energy* 34: 449-462. (Used with permission)

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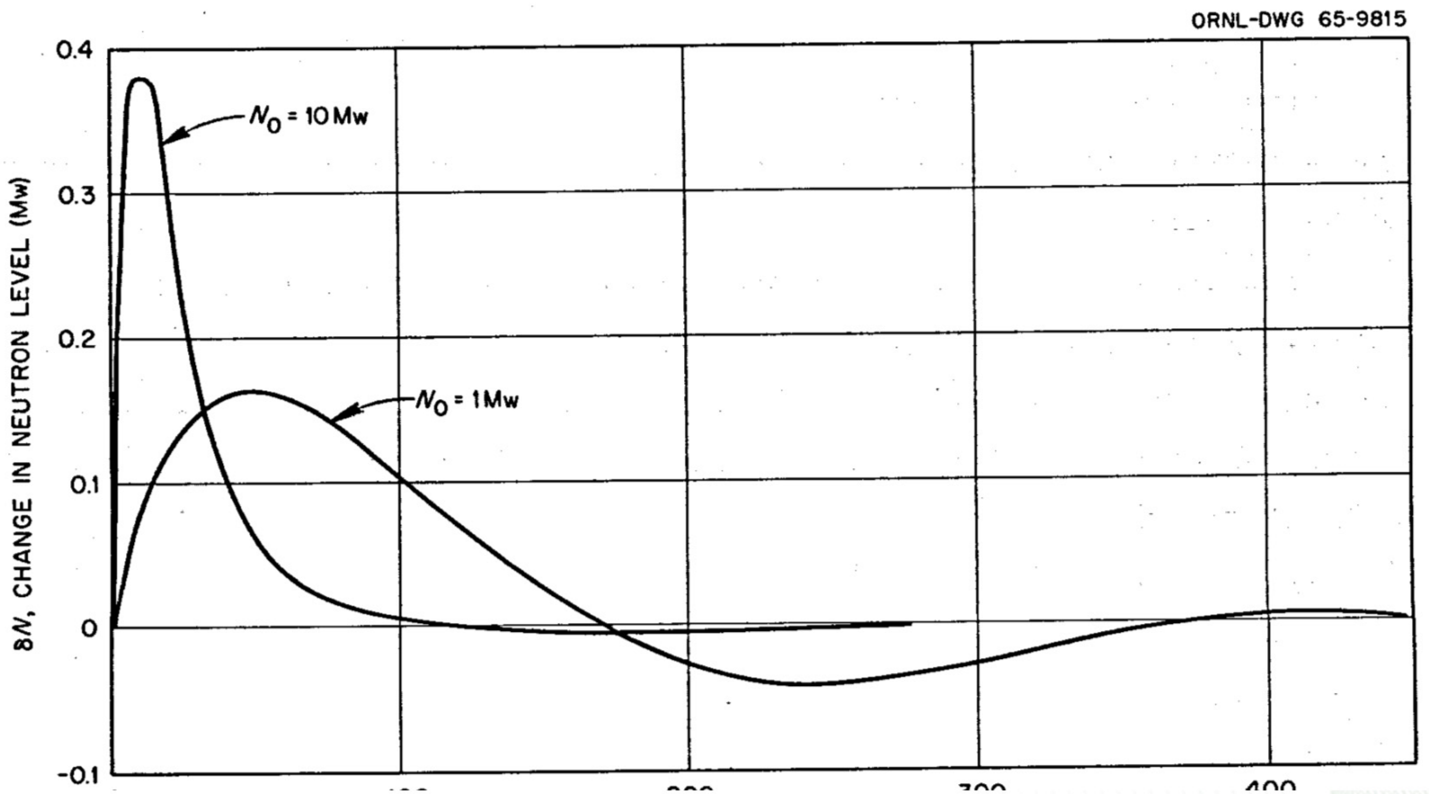
Stability of MSRE and Reactivity Feedback

- MSRE was determined analytically to be inherently stable
- Predictions were confirmed experimentally
- Example: reactivity insertion behavior



Nuclear Data Availability and Uncertainty

- Nuclear data uncertainties impact the ability to predict MSR neutronics
  - Absorption reactions
    - in lithium are important for thermal spectrum fluoride salt MSRs
    - in chlorine are important for fast spectrum chloride salt MSRs
  - Thermal neutron scattering
    - S(α,β) libraries are needed, especially for Li and Be in FLiBe
- Some examples follow for thermal spectrum and fast spectrum MSRs

Example: Sensitivity and Uncertainty (S/U) Analysis

- Identify potential sources of bias due to neutron cross-sections through uncertainty analysis
- Use sensitivity profiles as a function of energy as a tool to design informed experiments that can address those potential sources of bias

$$S_{k,\Sigma} = \frac{\delta k / k}{\delta \Sigma / \Sigma}$$

- At the high level, the goal of S/U analysis is to:
  – Have high quality critical experiments for validation of reactor physics calculations for fluoride salt reactor concepts: operations and design
  – Assess adequacy of ENDF cross-sections

Source: ORNL/TM-2016/729

S/U Analysis

Potential bias

- Use uncertainty analysis to identify potential sources of bias due to cross-section uncertainties

Validation need

- If there are significant contributors to uncertainty, identify specific target validation needs through sensitivity analysis

Experiments that capture sensitivities

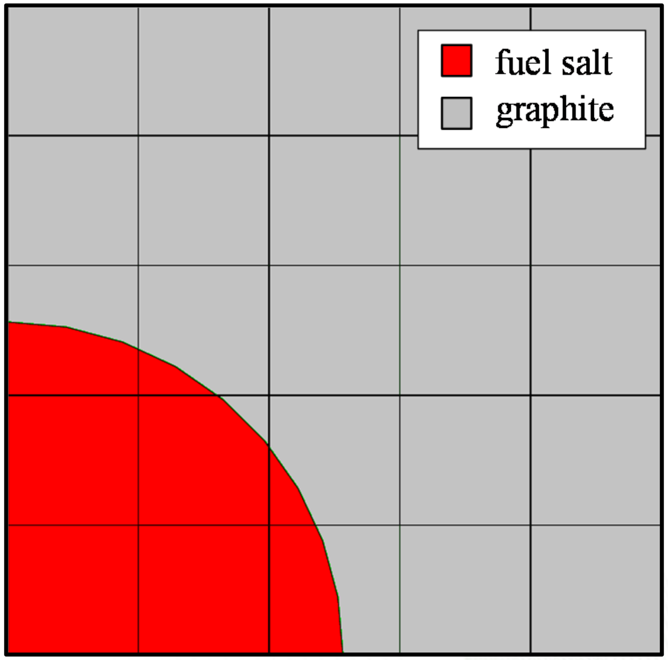
- Design experiments that capture the appropriate energy dependence of the sensitivities to meet the validation need

Source: ORNL/TM-2016/729

Sensitivity and Uncertainty (S/U) Analysis of MSR Application Models

- Model of a typical liquid fueled MSR unit cell geometry were adapted for S/U analysis
- Scoping S/U analysis was completed for MSR models
  - Both Th/^{233}U and LEU fueled MSR

S/U analysis of MSR LEU model shows uncertainty contributions from ^{7}Li, C, ^{19}F



MSR Model

Source: J. J. Powers, T. J. Harrison, and J. C. Gehin. 2013. "A New Approach for Modeling and Analysis of Molten Salt Reactors Using Scale." *Proceedings of the 2013 International Conference on Mathematics and Computational Methods Applied to Nuclear Science and Engineering (M&C 2013).*

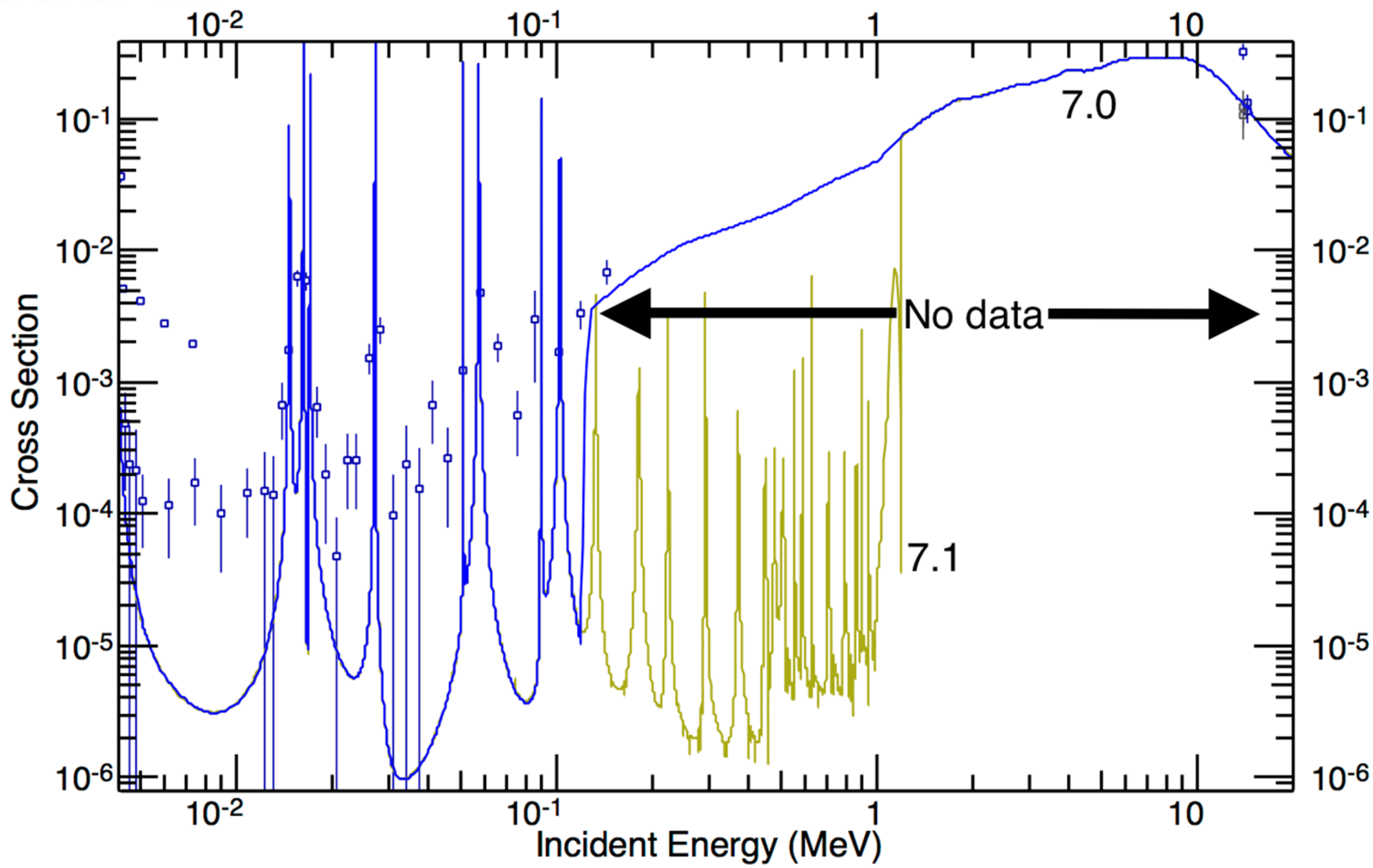
Observation from S/U Analysis

- For liquid fueled thermal spectrum fluoride salt reactors ^{7}Li seems to be the most significant contributor to potential bias in the FLiBe salt
  - For the range of ^{7}Li enrichments considered and the limited set of application models
- Unlike LWRs, SFRs, and HTGRs, there is an almost total lack of available benchmarks for MSRs
  - Integral critical experiments would support salt reactor development

Source: ORNL/TM-2016/729

Example: ^{35}Cl (n,p) for Chloride Salt Reactors

- Discrepancies in libraries (e.g., ENDF/B VII.0 vs. ENDF/B VII.1) and lack of data in the fast energy range significantly impacts criticality predictions (1000s of pcm)



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Conclusions

- MSRs present potential neutronics advantages
  - "Infinite batch" refueling (low excess reactivity)
  - Possibility for online removal of fission products
  - Strong potential for inherent safety and stability
- MSRs are very different from traditional solid fueled systems due to fuel cycle flexibility and delayed neutron precursor drift
- There is a wide variety of different MSR concepts with many different salts, potential missions, and neutronic characteristics
- US NRC tools such as PARCS need modification to account for reactor physics of MSRs
- Very strong need for benchmark experiments and validation data to benchmark simulation tools