Recent applications of Serpent related to nuclear technology research and development

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Nicholas R. Brown
Assistant Professor of Nuclear Engineering
Penn State University

nrb26@psu.edu

Thanks to collaborators on these projects:
B. Betzler, A. Wysocki, J. Gorton, C. Lu, K. A. Terrani,
J. W. McMurray, G. Singh, B. D. Wirth, and others

Outline

Discussion of recent applications of Serpent in support of nuclear technology research and development

- FHR DR design, including cross section generation with Serpent and core calculations with PARCS

- Informing thermodynamic simulation of TRISO particle constituents in HTGR with Serpent burnup calculations

- Impact of FCM fuel form on HTGR core design

- SiC-f/SiC-m channel box deformation analysis

ADTR Study: Background

- A 2017 U.S. DOE study of advanced reactor technology options, capabilities, and requirements within the context of national needs and public policy to support innovation in nuclear energy.

A Summary of the Department of Energy's Advanced Demonstration and Test Reactor Options Study

Nuclear Technology

D. Petti,[a]* R. Hill,[b] J. Gehin,[c] H. Gougar,[a] G. Strydom,[a] T. O'Connor,[d] F. Heidet,[b] J. Kinsey,[b] C. Grandy,[b] A. Qualls,[c] N. Brown,[c] J. Powers,[c] E. Hoffman,[b] and D. Croson[a]

Preconceptual design of a fluoride high temperature salt-cooled engineering demonstration reactor: Motivation and overview ☆

Annals of Nuclear Energy

A. Louis Qualls, Benjamin R. Betzler, Nicholas R. Brown *, Juan J. Carbajo, M. Scott Greenwood, Richard Hale, Thomas J. Harrison, Jeffrey J. Powers, Kevin R. Robb, Jerry Terrell, Aaron J. Wysocki, Jess C. Gehin, Andrew Worrall

Annals of Nuclear Energy

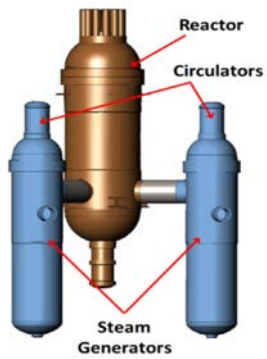
Preconceptual design of a fluoride high temperature salt-cooled engineering demonstration reactor: Core design and safety analysis ☆

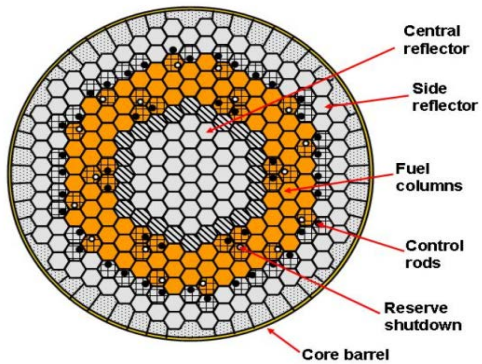
CrossMark

Nicholas R. Brown [a,b,*], Benjamin R. Betzler [a], Juan J. Carbajo [a], Aaron J. Wysocki [a], M. Scott Greenwood [a], Cole Gentry [a], A. Louis Qualls [a]

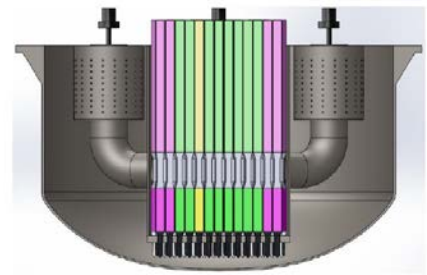
Resultant Top Options

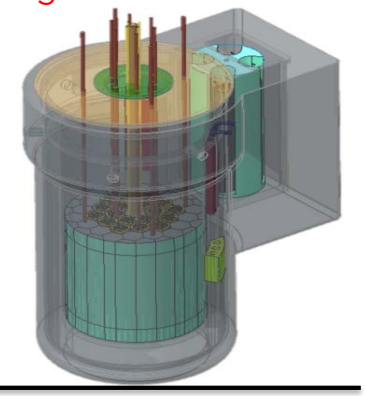
Strategic Objective 1: Process heat demonstration – modular HTGR commercial demonstration



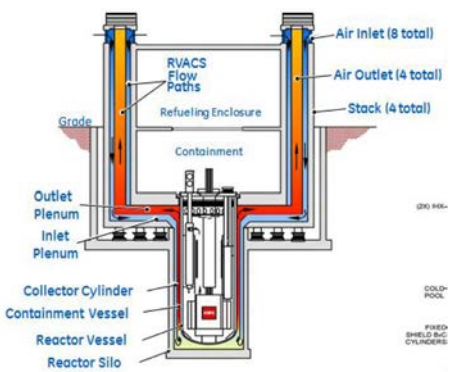


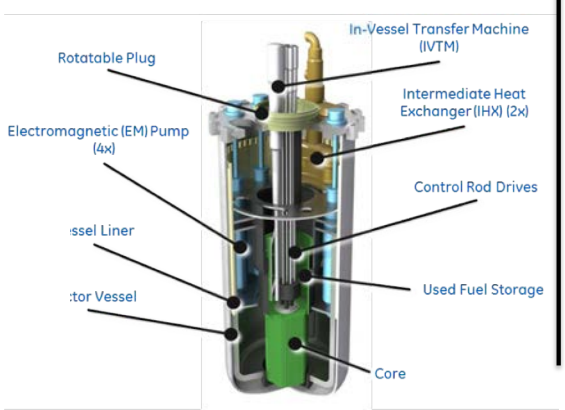
Strategic Objective 3: Demonstrating a Less Mature Technology – FHR or LFR engineering demonstration



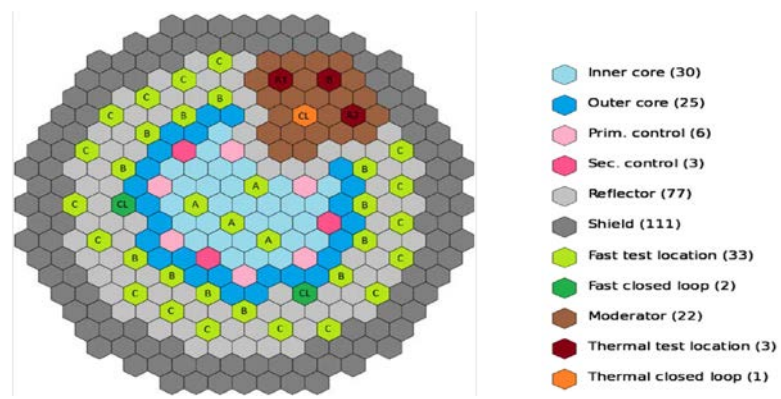


Strategic Objective 2: Resource Utilization and Waste Management – SFR commercial demonstration





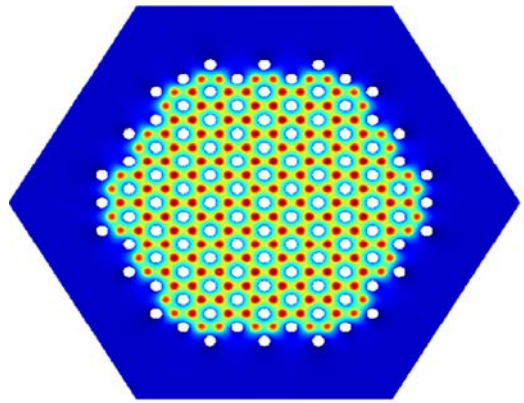
Strategic Objective 4: Test Reactor to Provide Fast Neutrons – Sodium–cooled Fast Test Reactor



Need: Design Toolkit for FHRs

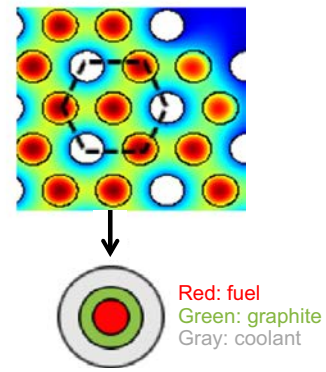
- One challenge the FHR DR team faced is the lack of a set of comprehensive core design and licensing tools for FHRs
- PARCS-FHR (with Serpent for few-group cross sections) was developed
  - Contributors: Aaron Wysocki, Ben Betzler, and Nicholas Brown
- FHR-relevant thermal feedback capability into PARCS
  - FLiBe hydraulic properties
  - Compact and graphite properties

Detailed Temperature Calculation (e.g. from COMSOL)



Simplified Temperature Calculation in PARCS (similar to RELAP5-3D)

→Reduce the detailed geometry to an equivalent sub-assembly (preserving total flow area and fuel volume)
→Scale the thermal conductivity to give the appropriate fuel temperatures

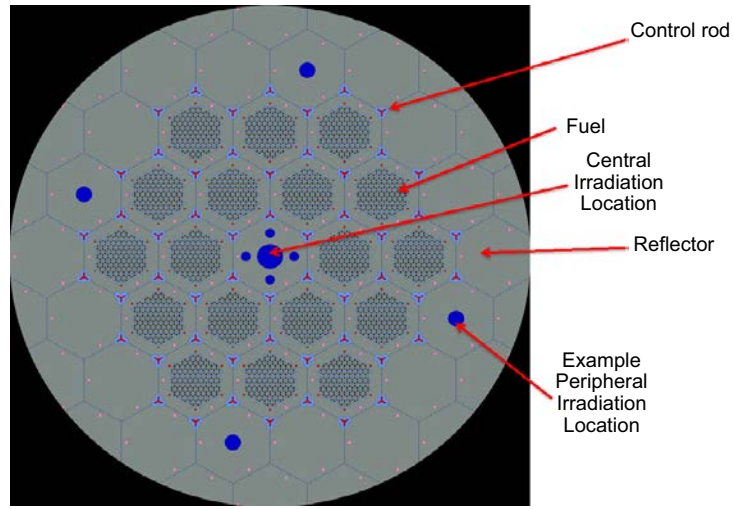


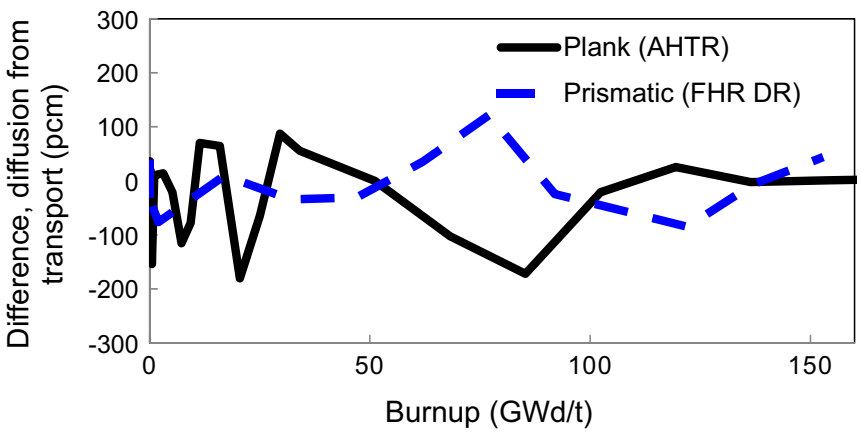
In a separate effort, Cole Gentry (UTK) demonstrated nodal core analysis for FHRs with Serpent-NESTLE and a 4-group structure

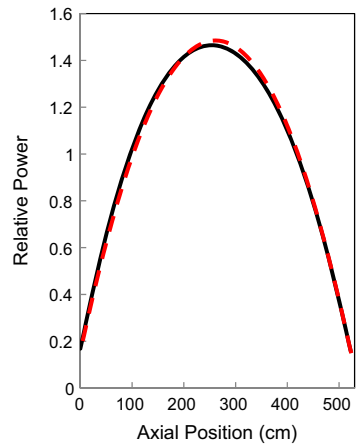
For the FHR DR, the 4-group structure worked as well as the more complicated structures

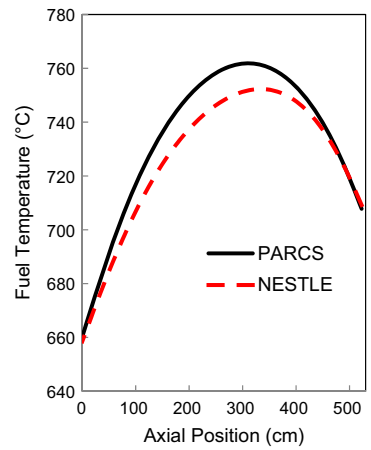
Serpent Cross Section Generation

- Full-core models in Serpent that account for region dependent variations in spectrum/leakage were needed to correctly collapse cross sections and discontinuity factors for the small FHR DR
- For the FHR-DR, region dependent cross sections are easy, because it is a prismatic, single-batch core

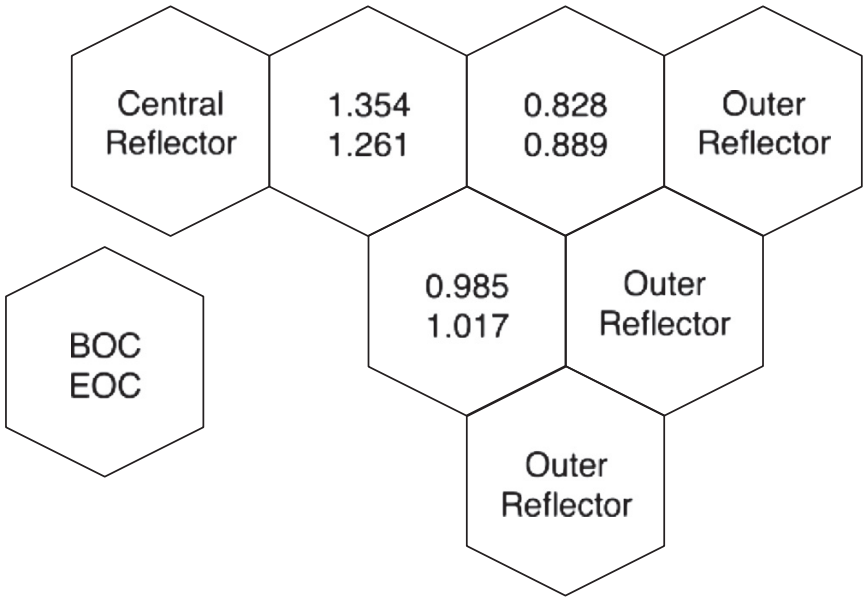




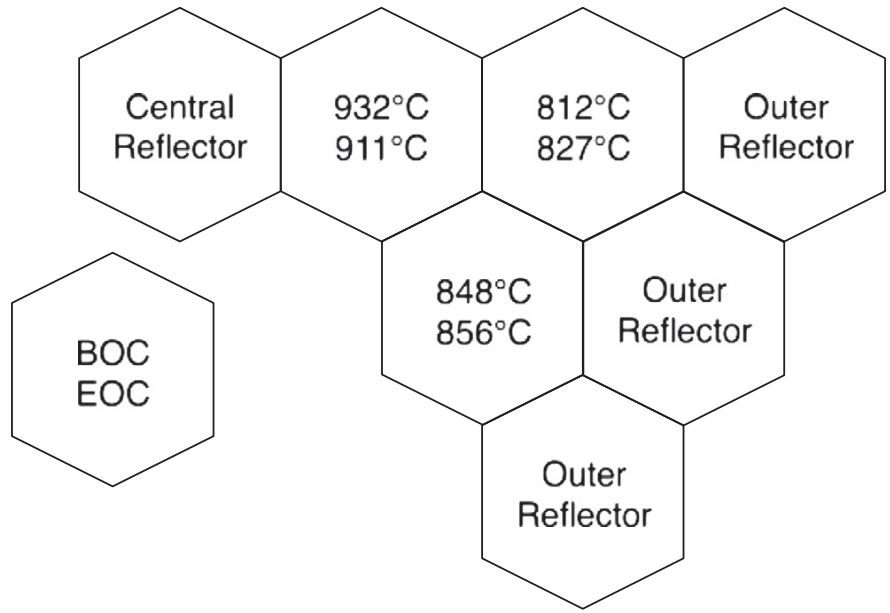




PARCS-FHR: Example Results

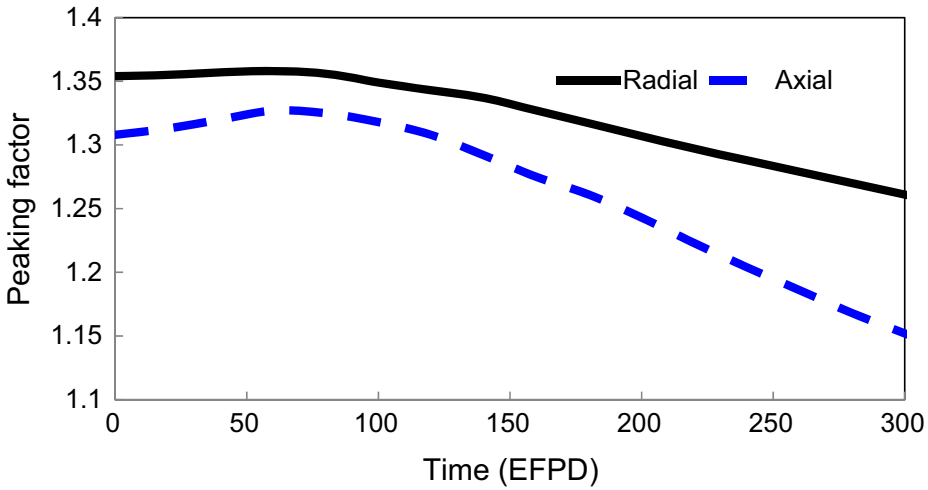


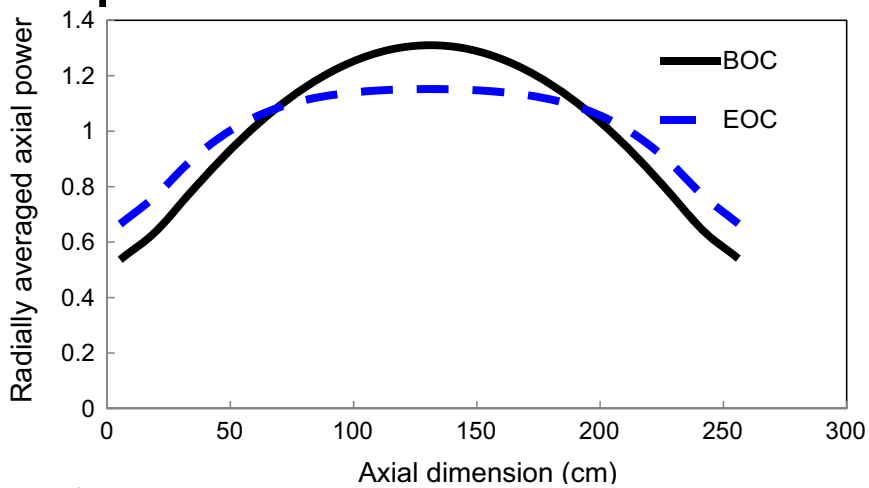
1/6-th Core Power Distribution

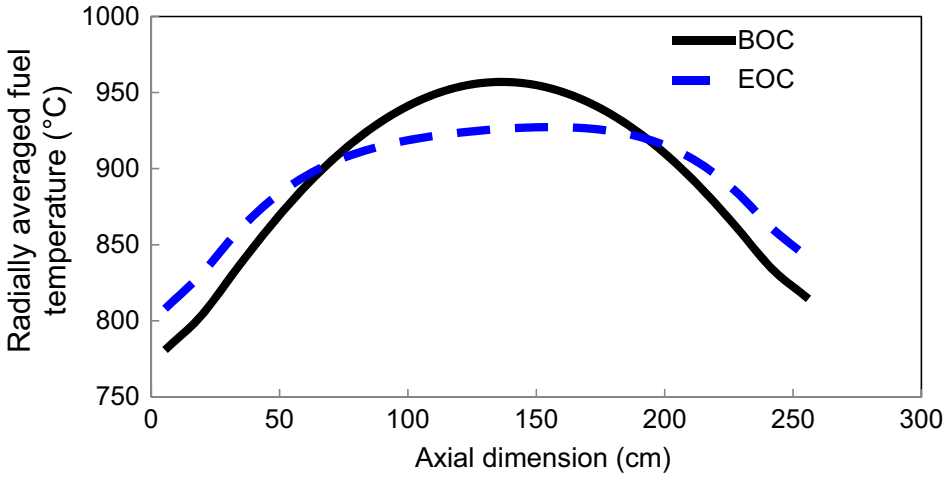


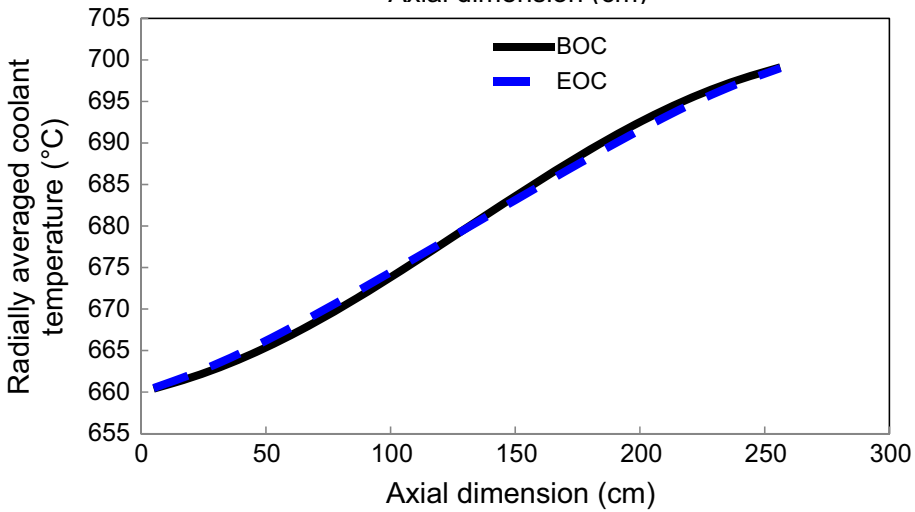
1/6-th Core Temperature Distribution

PARCS-FHR: More Example Results









Serpent and PARCS-FHR: Conclusions

- Conventional two-step method applied to FHRs
- We found the four-group structure Cole Gentry developed for plank-based FHRs is also relevant for prismatic FHRs (actually, it works even better for these systems)
- Capability was quickly ramped up and used for scoping calculations for this point design
- Additional work is needed to further develop and refine the two-step core analysis capability for other FHR concepts, such as the Kairos PB FHR

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- Informing thermodynamic simulation of TRISO particle constituents in HTGR with Serpent burnup calculations

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- SiC-f/SiC-m channel box deformation analysis

Informing thermodynamic simulations of HTGR TRISO fuel behavior with Serpent

- Three important failure mechanisms in HTGR fuel for high burnup are:
  - SiC layer rupture
  - SiC corrosion by CO
  - coating compromise from kernel migration
- All are related to high CO pressures from O release when uranium present as UO_2 fissions and the O is not bound by other elements
- CO buildup from excess O is controlled with additional uranium in the form of a carbide, UCx, and this fuel form is designated UCO

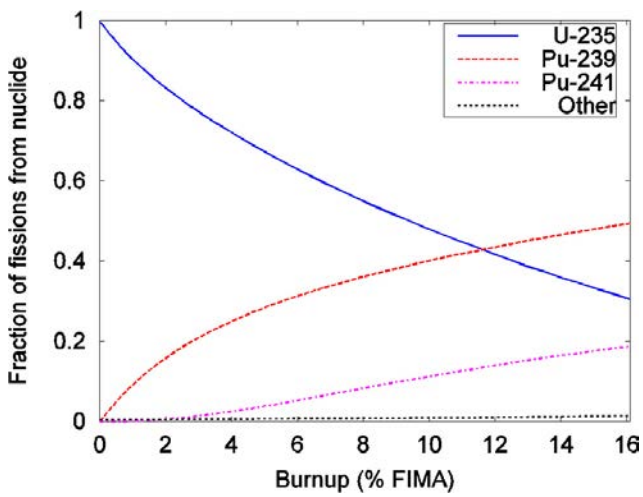
Determining the minimum required uranium carbide content for HTGR UCO fuel kernels ☆

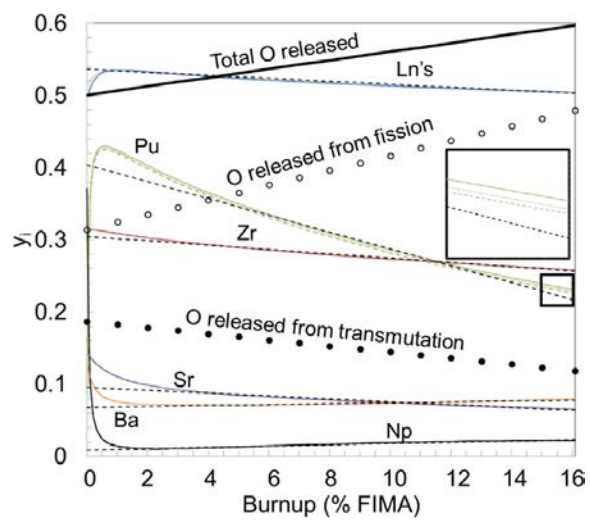
Jacob W. McMurray [a,*], Terrence B. Lindemer [b], Nicholas R. Brown [c,1], Tyler J. Reif [d], Robert N. Morris [e], John D. Hunn [e]

[a] Materials Science and Technology Division, Oak Ridge National Laboratory, Oak Ridge, TN 37831-6063, United States
[b] MPi Business Solutions, Inc., Knoxville TN 37915, United States
[c] Reactor and Nuclear Systems Division, Oak Ridge National Laboratory, Oak Ridge, TN 37831-6165, United States
[d] X-Energy, LLC, 7701 Greenbelt, MD 20770, United States
[e] Fusion and Materials for Nuclear Systems Division, Oak Ridge National Laboratory, Oak Ridge, TN 37831-6093, United States

Feeding CALPHAD with Serpent: Conclusions

- CALPHAD calculates thermodynamic composition of a chemical mixture
- Serpent depletion results were fed to CALPHAD to determine the optimal uranium carbide content in UCO TRISO fuel
- This was compared with previous reduced order models
- Optimal UCx content for example case is about ~5%





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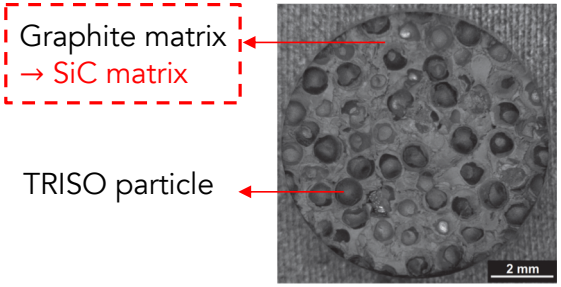
HTGR Core Design with FCM Fuel

- Fully Ceramic Microencapsulated (FCM) fuel consists of TRISO fuel particles embedded in a matrix of silicon carbide (SiC).
  - Higher stability under irradiation
  - Better fission product retention ability
  - Less sensitive to physical disturbances
  - Higher oxidation resistance



Fully ceramic microencapsulated fuel in prismatic high temperature gas-cooled reactors: Analysis of reactor performance and safety characteristics

Cihang Lu [a], Briana D. Hiscox [b,1], Kurt A. Terrani [c], Nicholas R. Brown [a,*]

[a] *Pennsylvania State University, University Park, PA 16802, USA*
[b] *Massachusetts Institute of Technology, Cambridge, MA 02139, USA*
[c] *Oak Ridge National Laboratory, Oak Ridge, TN 37831, USA*

What are the key changes in fuel cycle performance of an mHTGR with the FCM fuel form?

- Fuel cycle length
- Natural resource requirement

2-D neutronics model

What is the potential impact of the FCM fuel form on reactor performance and safety characteristics of mHTGRs?

(10 CFR 50 Appendix A)

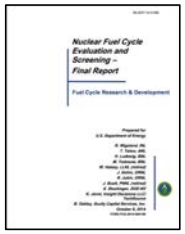
- Reactivity temperature coefficients (RTC)
- Decay power

- Control rod worth
- Steady-state power distribution

3-D neutronics model

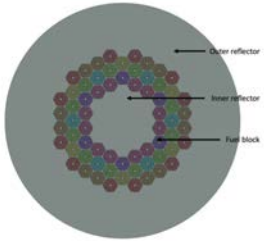
- Steady-state temperature distribution

Thermal-Hydraulics ring model

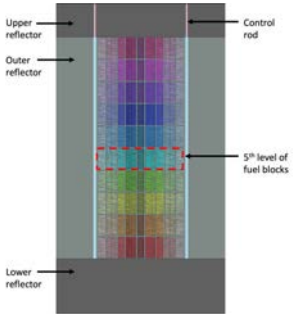


The U. S. Department of Energy, Office of Nuclear Energy (DOE-NE) chartered Nuclear Fuel Cycle Evaluation and Screening Study

(Pope, 2012)

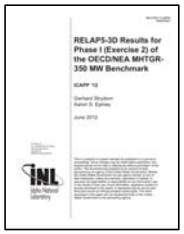


2-D neutronics model

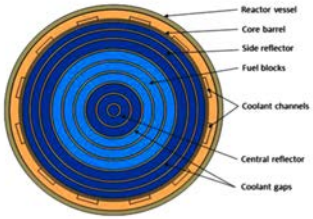


3-D neutronics model

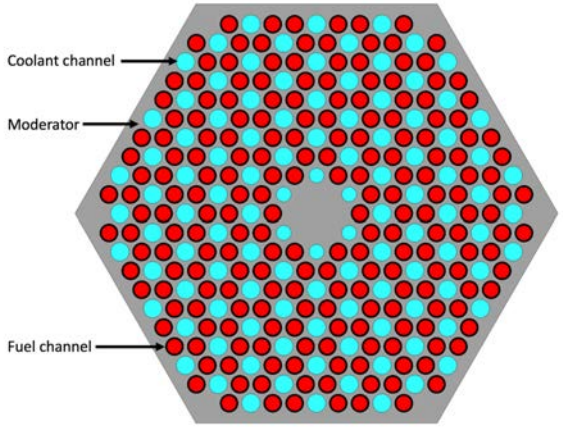
International Benchmark Definition of the MHTGR-350 MW core design



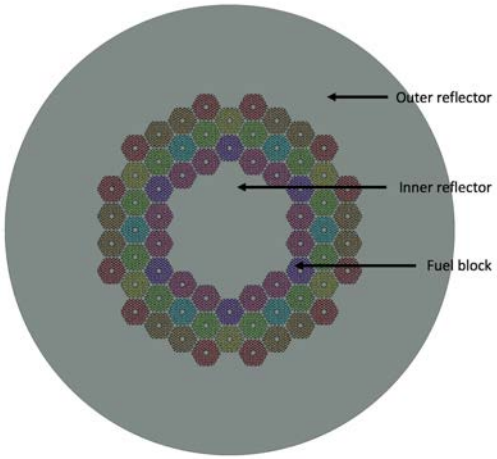
(Strydom and Epiney, 2012)



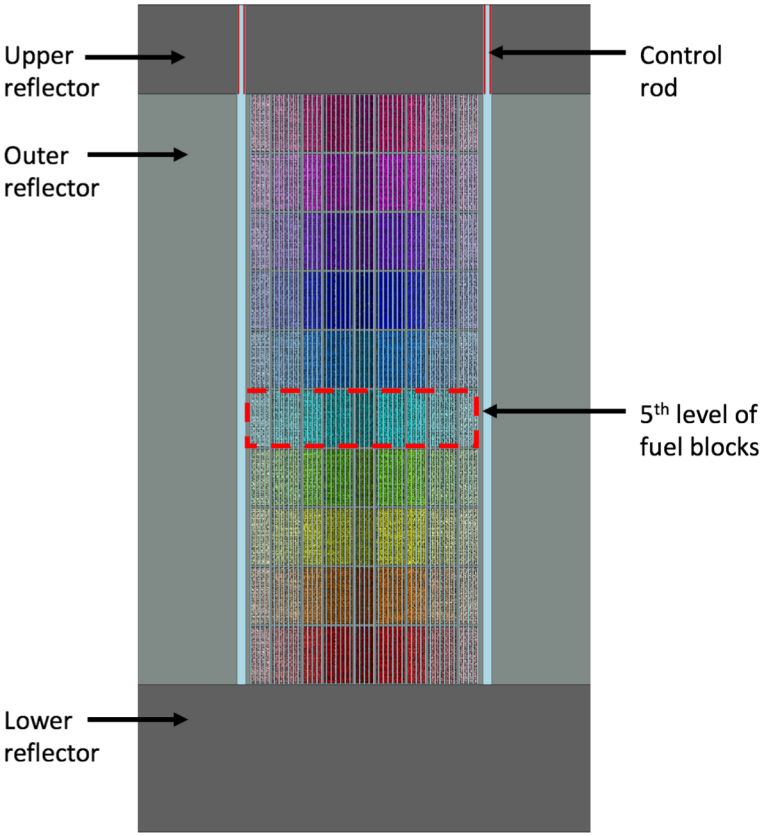
Thermal-Hydraulics ring model

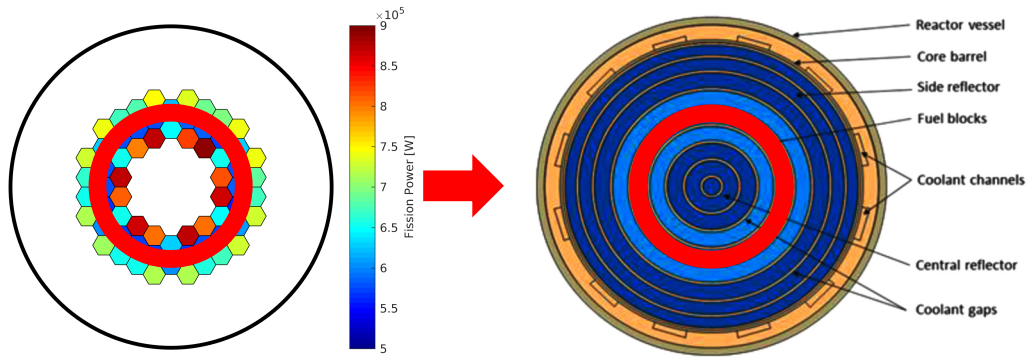


Fuel cycle length	**Maintained**
Natural resource requirement	**75%↑**
Reactivity temperature coefficients	**Similar**
Decay power	**3%↑**



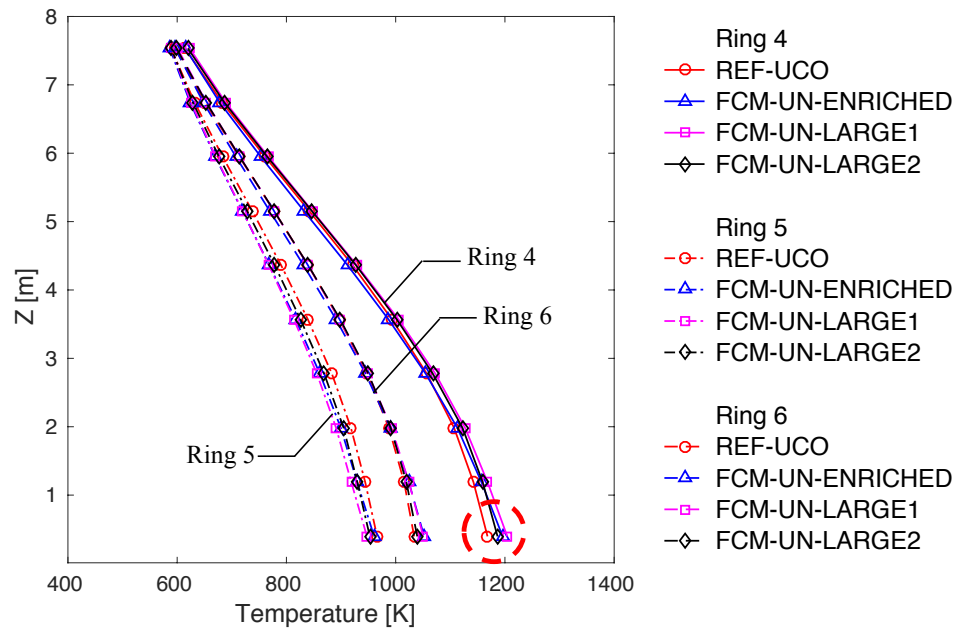
Case	^{235}U Mass [kg]	Enrichment [%]	Kernel Volume [cm^3]
REF-UCO	-	-	-
FCM-UN-ENRICHED	**48%↑**	**9%↑**	-
FCM-UN-LARGE1	**75%↑**	-	**30%↑**
FCM-UN-LARGE2	**44%↑**	-	**6%↑**





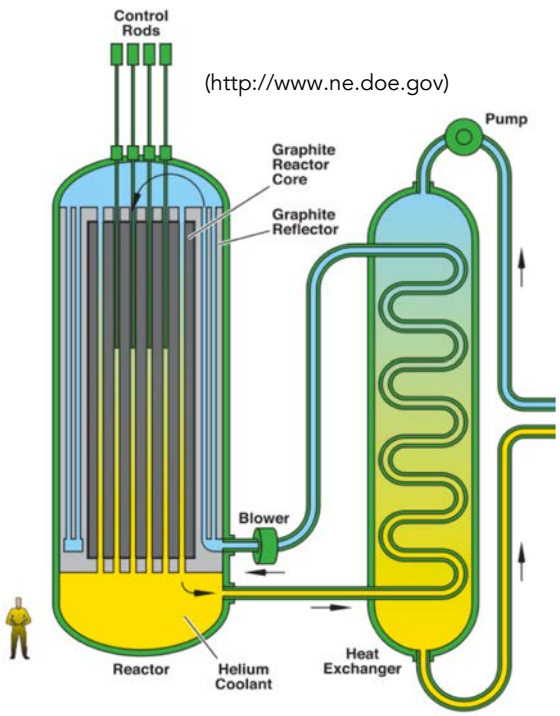
(Strydom and Epiney, 2012)

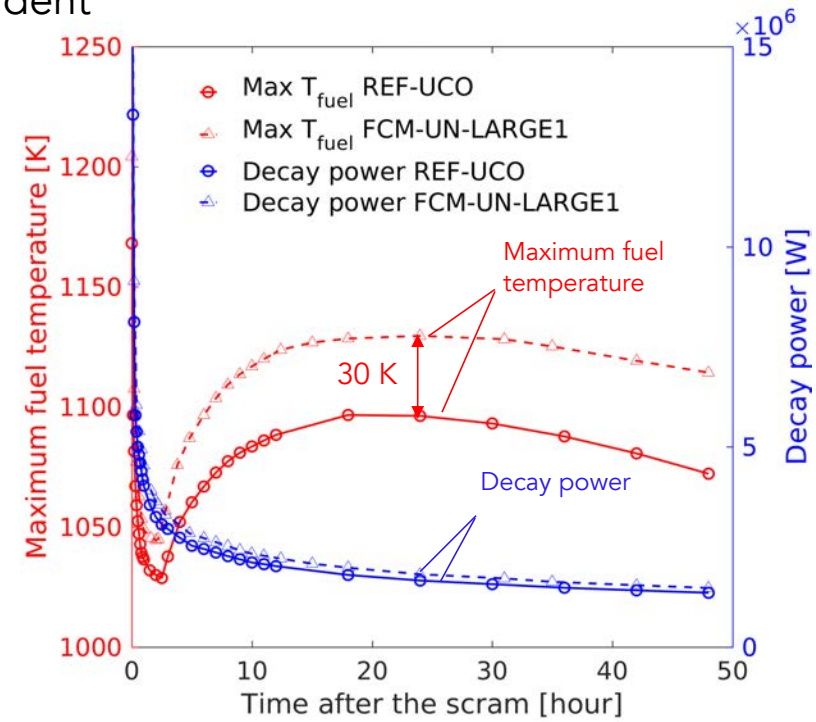
Control rod worth	**15.5%↓**
Power distribution	**Similar**



How does the FCM fuel form impact infrequent and limiting design basis fault conditions?

- Pressurized and Depressurized Loss Of Forced Circulation Accidents (P- and D-LOFC)
- Control Rod Withdrawal Accident





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Analysis of SiC-f/SiC-m BWR Channel Box with CASL CTF and Serpent

The 2011 accident at the Fukushima-Daiichi nuclear power plant led to an increased focus on Accident Tolerant Fuel (ATF) materials.

Silicon carbide fiber-reinforced, silicon carbide ceramic matrix composites (SiC/SiC) are a top ATF candidate.

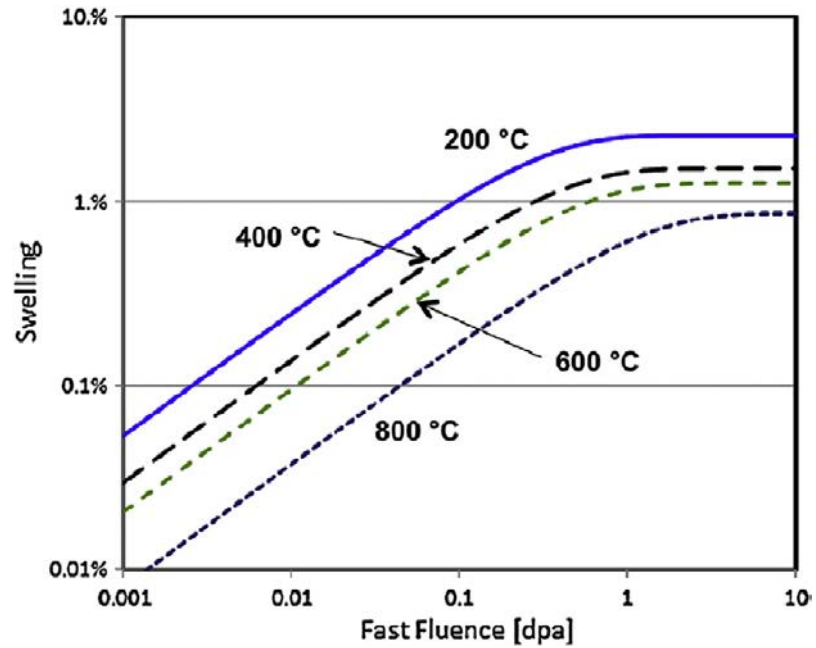
This presentation focuses on the predicted thermal hydraulic and neutronic characteristics of a SiC/SiC channel box in a boiling water reactor (BWR) fuel assembly.

These characteristics impact the viability of deploying SiC/SiC channel boxes.

Goals of the Current Study

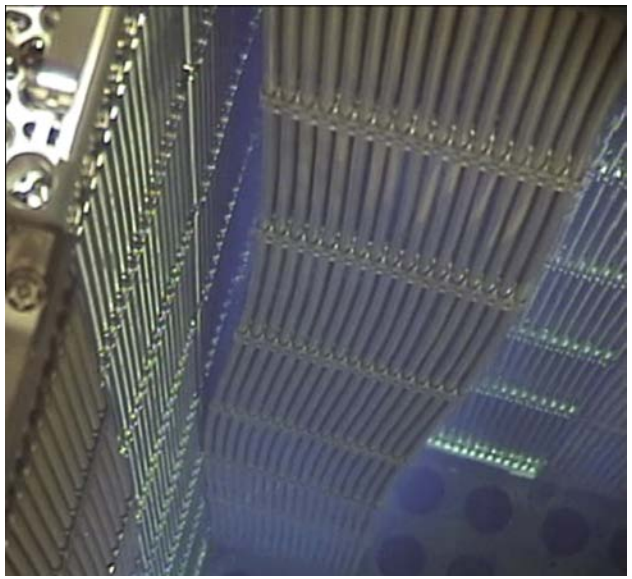
Main Objective: Provide thermal and neutronic boundary conditions for a study being to determine if the induced stress and deflection of a SiC/SiC channel box is acceptable

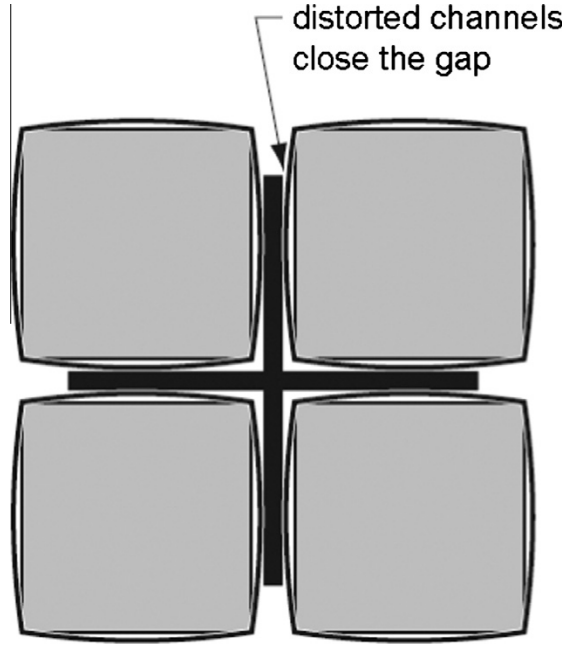
- Thermal boundary conditions needed because temperature gradient will cause thermal stress and bowing
- Neutron fast flux boundary conditions needed because of swelling of SiC-f/SiC-m when irradiated

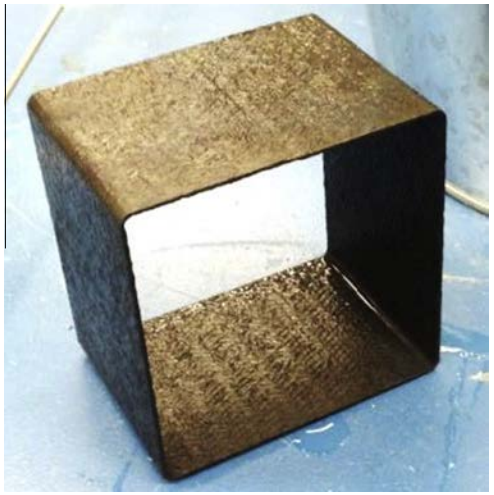


Fuel Assembly and Channel Distortion

- Examples from Yueh and Terrani (2014)

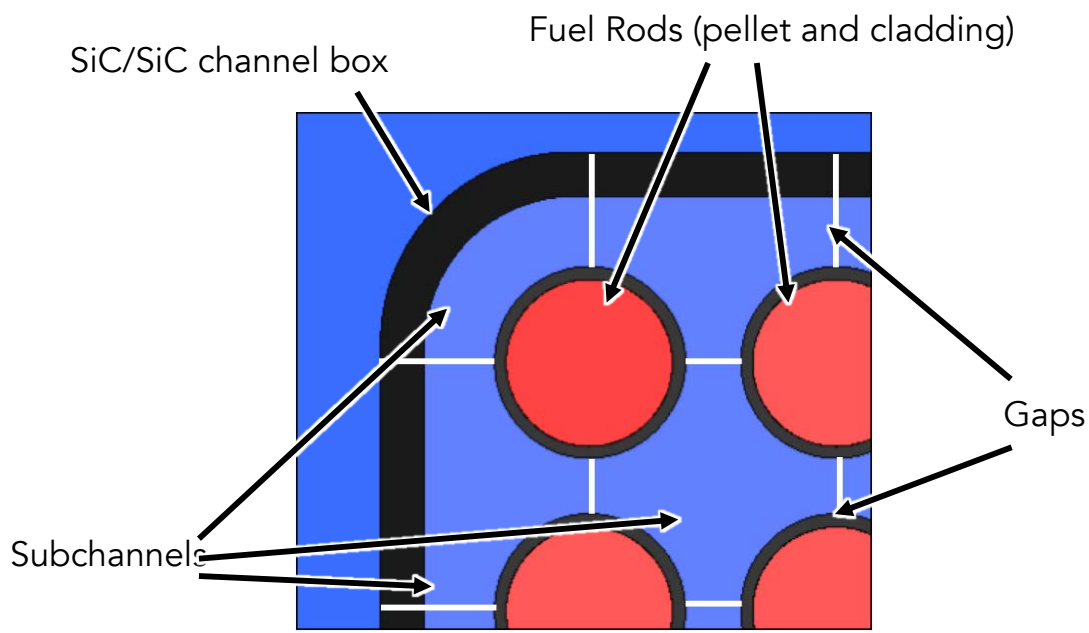






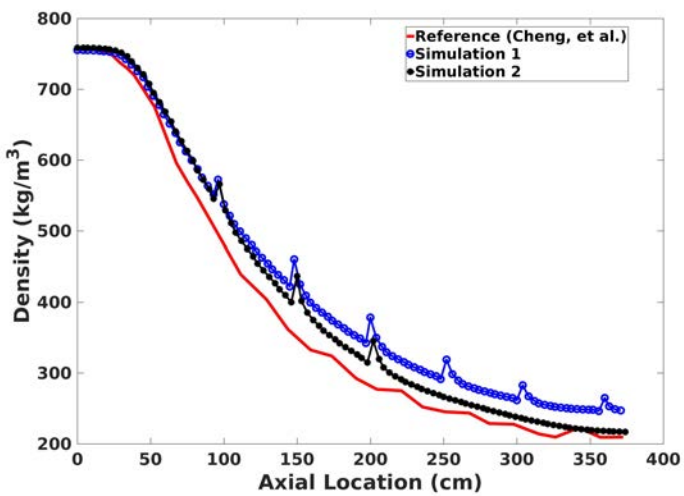
Methodology

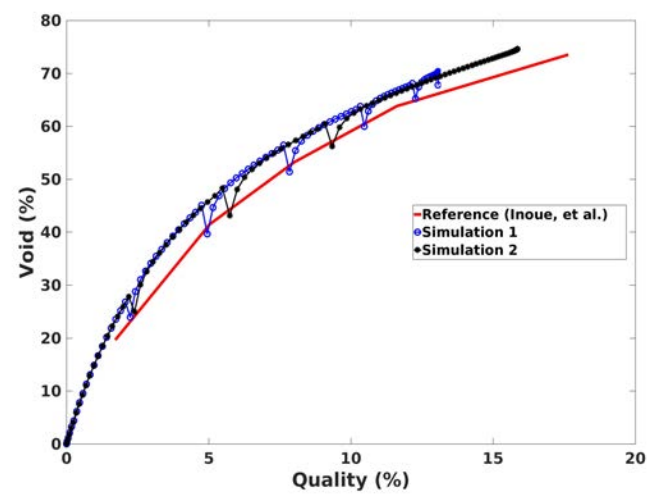
- Thermal hydraulic analysis performed using the Consortium for Advanced Simulation of Light Water Reactors (CASL) subchannel code CTF
  - Fuel assembly model created using a variety of publicly accessible open sources due to the fact that GE designs are proprietary
- Neutronic analysis performed using Serpent



CTF Models and Validation

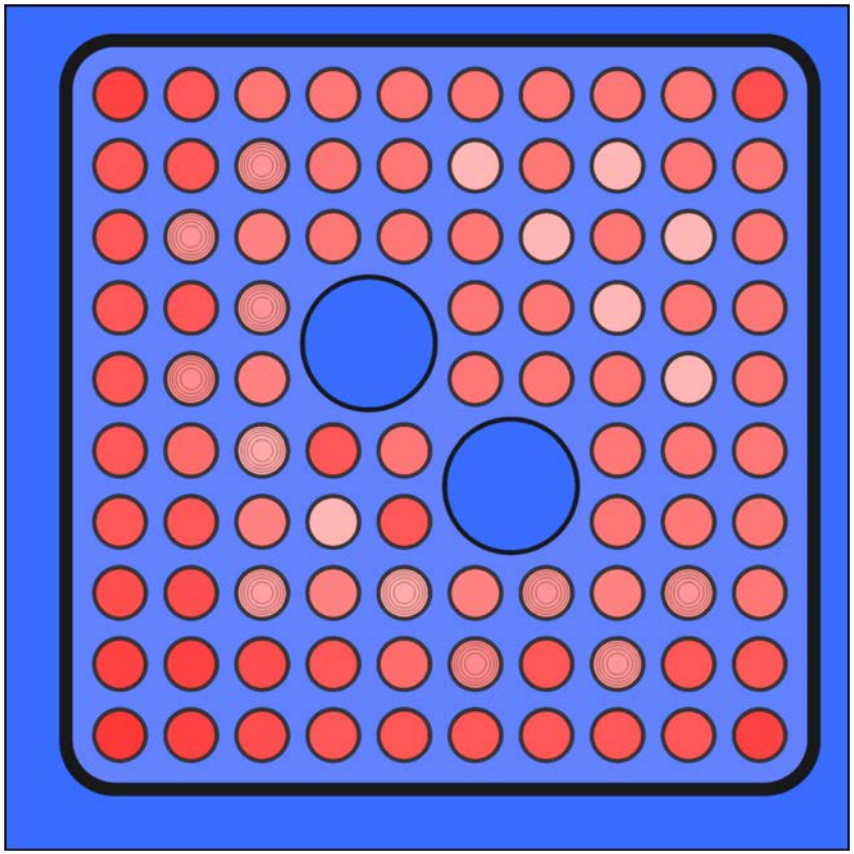
- Two models developed for the study referred to as Simulation 1 and Simulation 2
- Both models have the same geometric inputs, except for assembly length
- Sources are P. Ferroni (MIT) and M. Fensin (Florida) theses





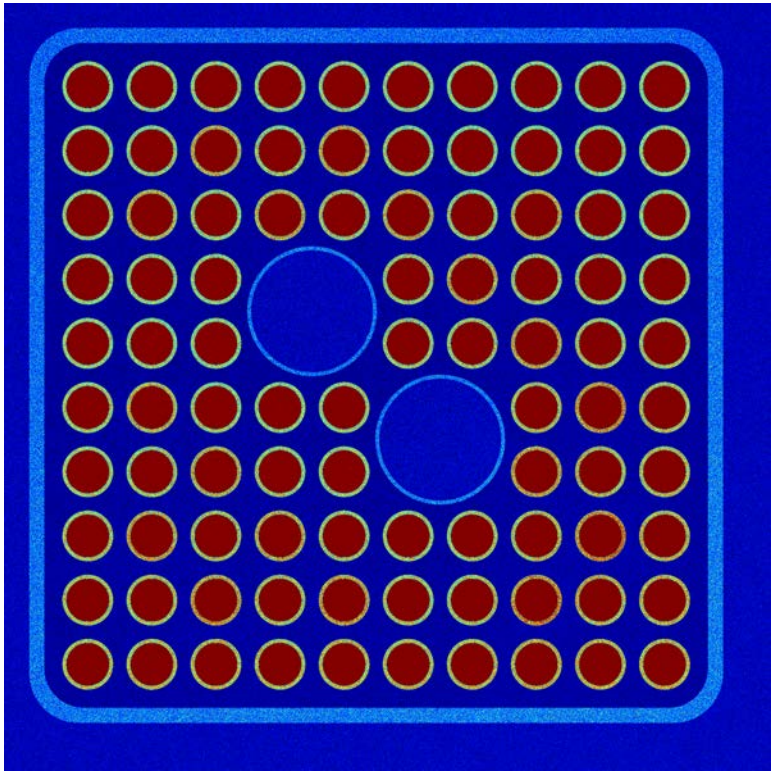
Neutron Flux Analysis

- Goal of the neutronics analysis: determine spatial distribution of neutron flux in the SiC/SiC channel box for neutron energies greater than 0.1 MeV
- Models developed by defining fuel rod lattice geometry and material properties
- A layout of the lattice geometry is shown here, with different colors corresponding to different materials (including fuels with different enrichments and gadolinium contents)



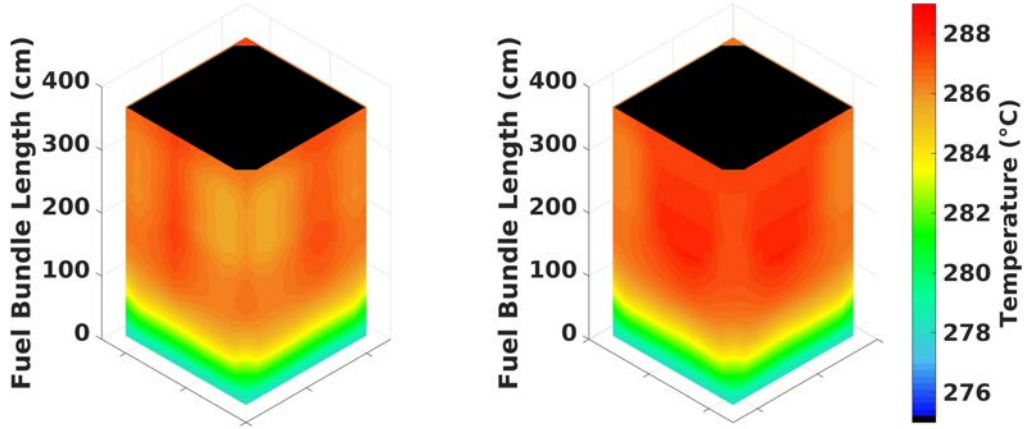
Direct Energy Deposition in SiC/SiC Channel Box

- Used new photon transport features to enable estimate of direct energy deposition due to neutrons/photons in channel box
- Both total deposition and radial variation is important
- Channel box walls account for a very small fraction of direct deposition, and it is fairly uniform

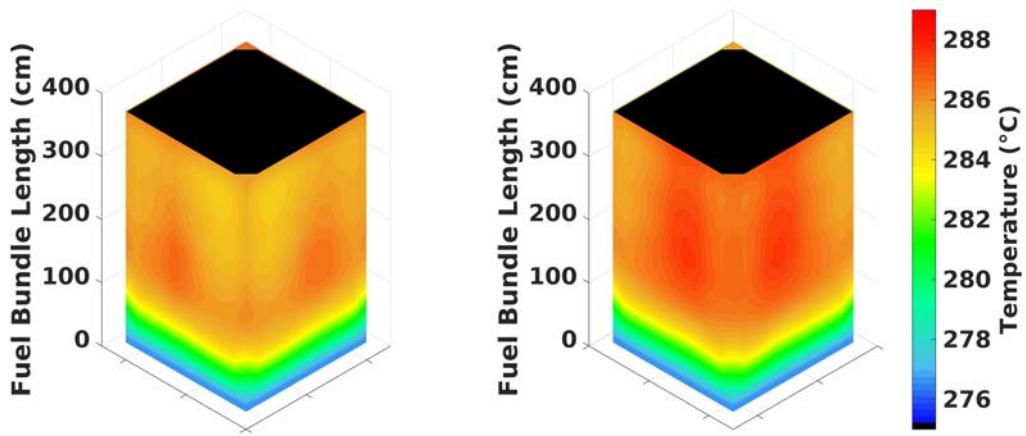


CTF Results

- The temperature distributions in the channel box are shown for both simulations
  - The box is rotated so that all four sides can be seen
- The maximum temperature in each channel box is approximately equal to the coolant saturation temperature at the system pressure



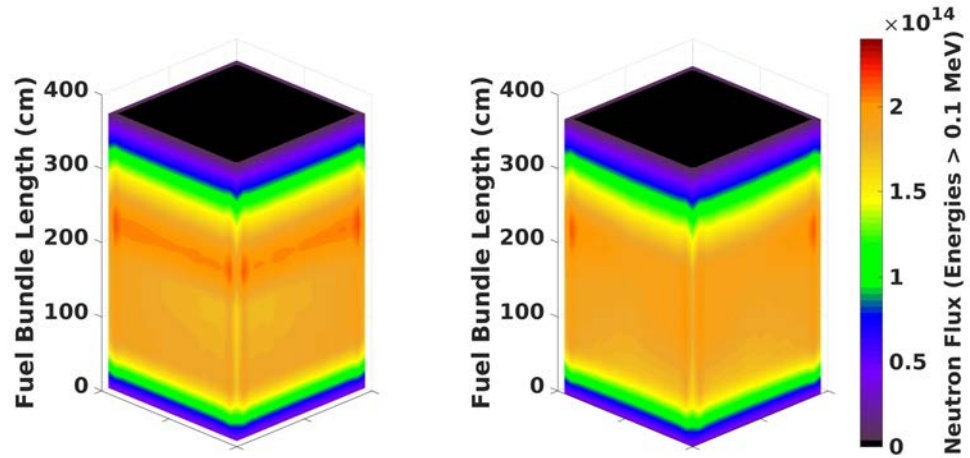
Simulation 1



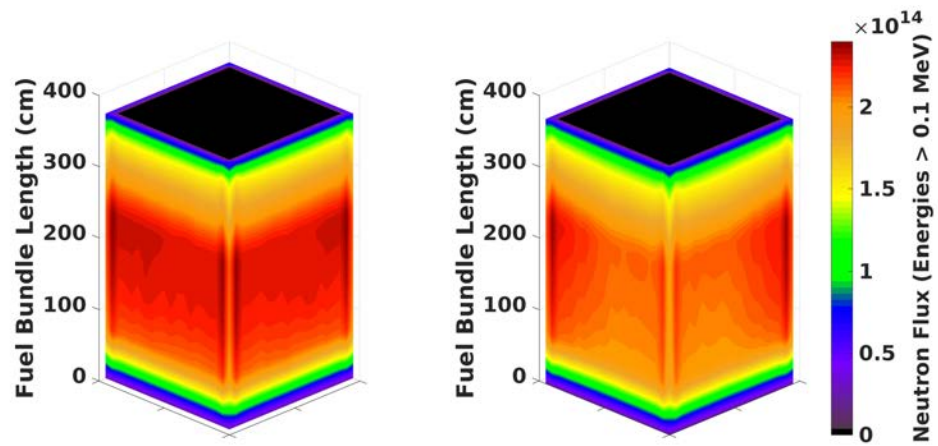
Simulation 2

Neutron Flux

- Neutron flux only tallied for neutron energies greater than 0.1 MeV
- Significant axial variation of neutron flux due to different fuel enrichments and power in each axial zone
- Comparatively small radial neutron flux variation



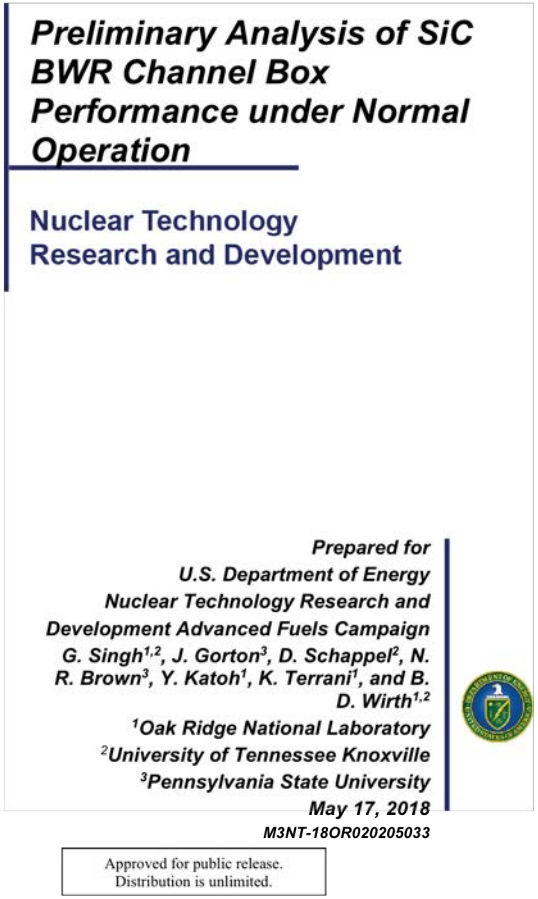
Simulation 1



Simulation 2

Summary of Channel Box Study

- Thermal and neutronic boundary conditions were generated for a SiC/SiC channel box in a BWR using CTF and Serpent
- Thermal hydraulic results showed that the temperature of the channel box varies significantly both axially and radially
- Neutronic results show that fast neutron flux also varies significantly radially and between each axial zone
- The conditions will induce swelling, stress and deflection in the channel box



Preliminary Analysis of SiC BWR Channel Box Performance under Normal Operation

Nuclear Technology Research and Development

Prepared for
U.S. Department of Energy
Nuclear Technology Research and Development Advanced Fuels Campaign
G. Singh[1,2], J. Gorton[3], D. Schappel[2], N. R. Brown[3], Y. Katoh[1], K. Terrani[1], and B. D. Wirth[1,2]
[1]Oak Ridge National Laboratory
[2]University of Tennessee Knoxville
[3]Pennsylvania State University

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