SCALE Users' Group Workshop

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Advanced BWR Criticality Safety with Quantified Uncertainty Using Various SCALE Modules

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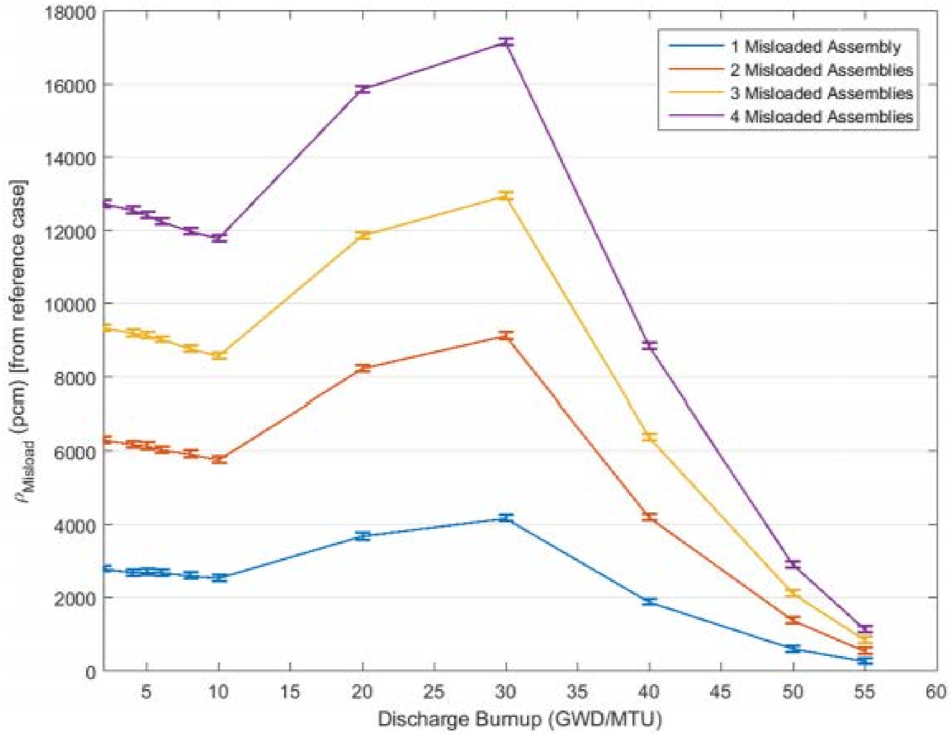
Contents

Objectives

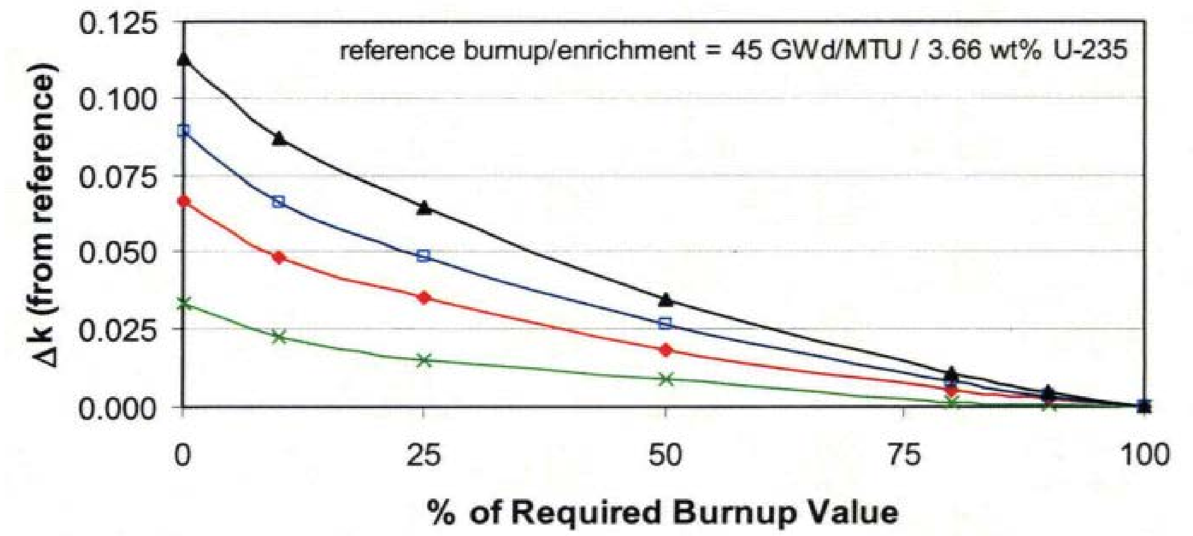
- Review our activities on BWR criticality safety based on NEUP funded project on “Cask Misload Evaluation Techniques”.
- Build a set of computational models in SCALE to capture BWR design complexities for a single lattice
  - Control rod insertion.
  - Gadolinium presence.
  - Heterogenous radial and axial enrichment.
  - Part-length rods.
  - Axial burnup and void fraction profiles.
  - Partial control rod insertion.
  - Control rod movement.
- Investigate the effect of such complexities on the burnup credit and cask k_{eff} uncertainty.
- Incorporate BWR spent fuel assay data to quantify the isotopic uncertainty in cask k_{eff}.

Previous Cask Misloading Efforts

BWR Misloading (Radaideh et al.)



PWR Misloading (Wagner)

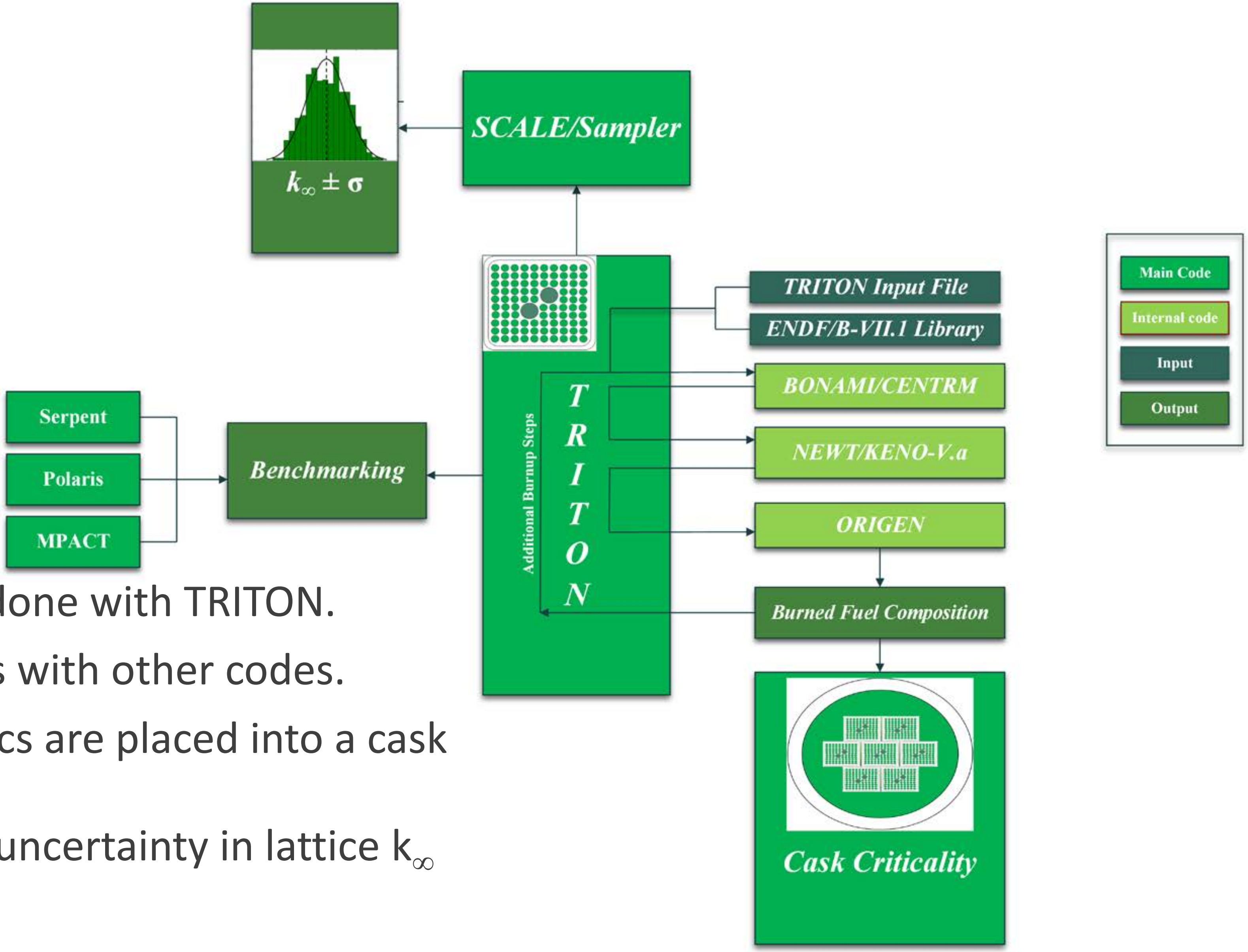


Radaideh, M.I., 2018. Criticality and Uncertainty Assessment of Assembly Misloading in BWR Transportation Cask, Annals of Nuclear Energy, 113, pp. 1-14.

Wagner, J.C., 2008. Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask NUREG/CR-6955 ORNL/TM-2004/52. Oak Ridge National Laboratory, Oak Ridge, TN, USA.

Contents

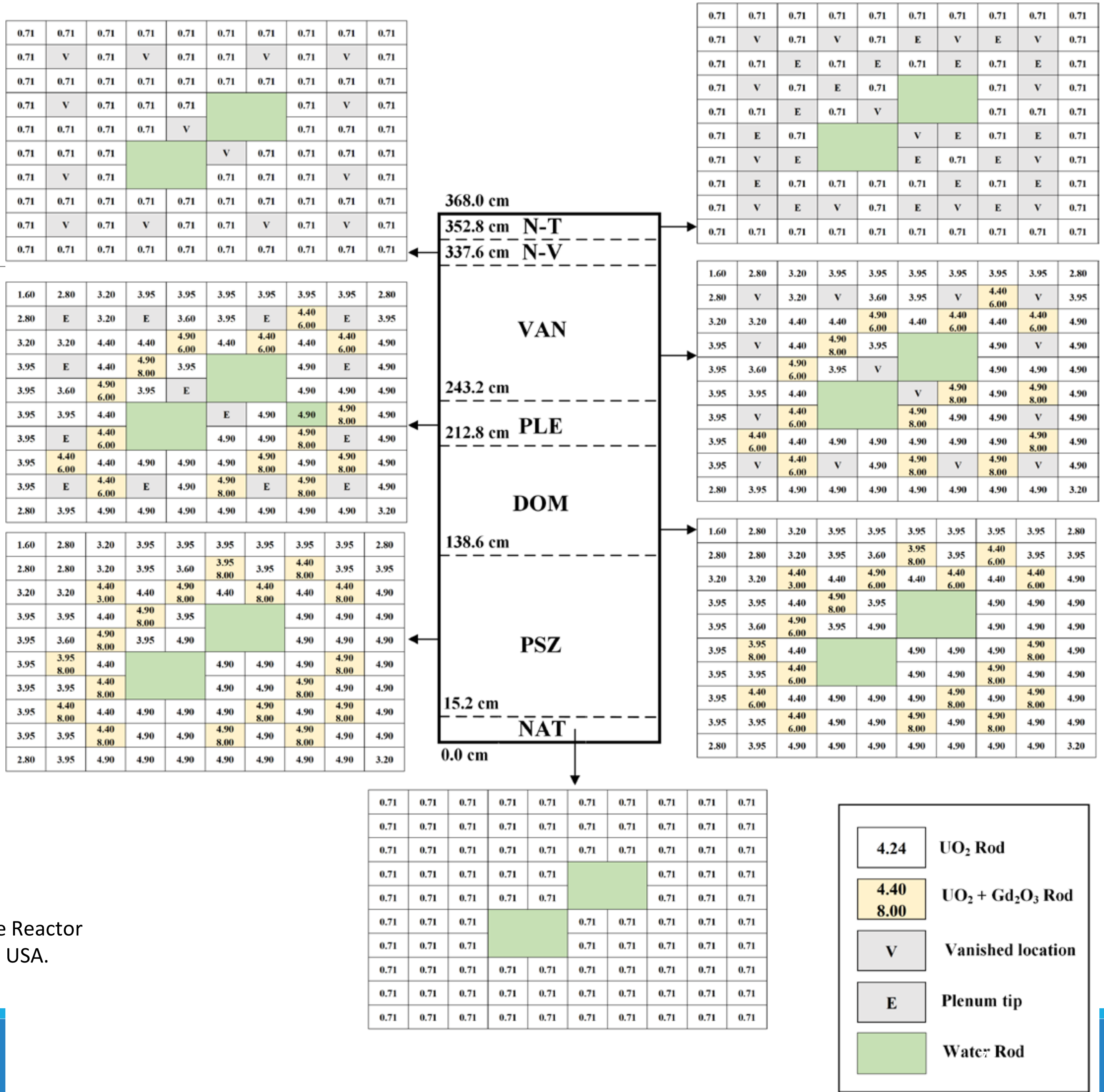
Methodology



- Depletion on verified models is done with TRITON.
- Benchmarking the lattice models with other codes.
- The calculated spent fuel isotopics are placed into a cask modelled in KENO-V.a.
- Sampler is used to calculate the uncertainty in lattice k_∞

BWR Data and Resources (Fensin)

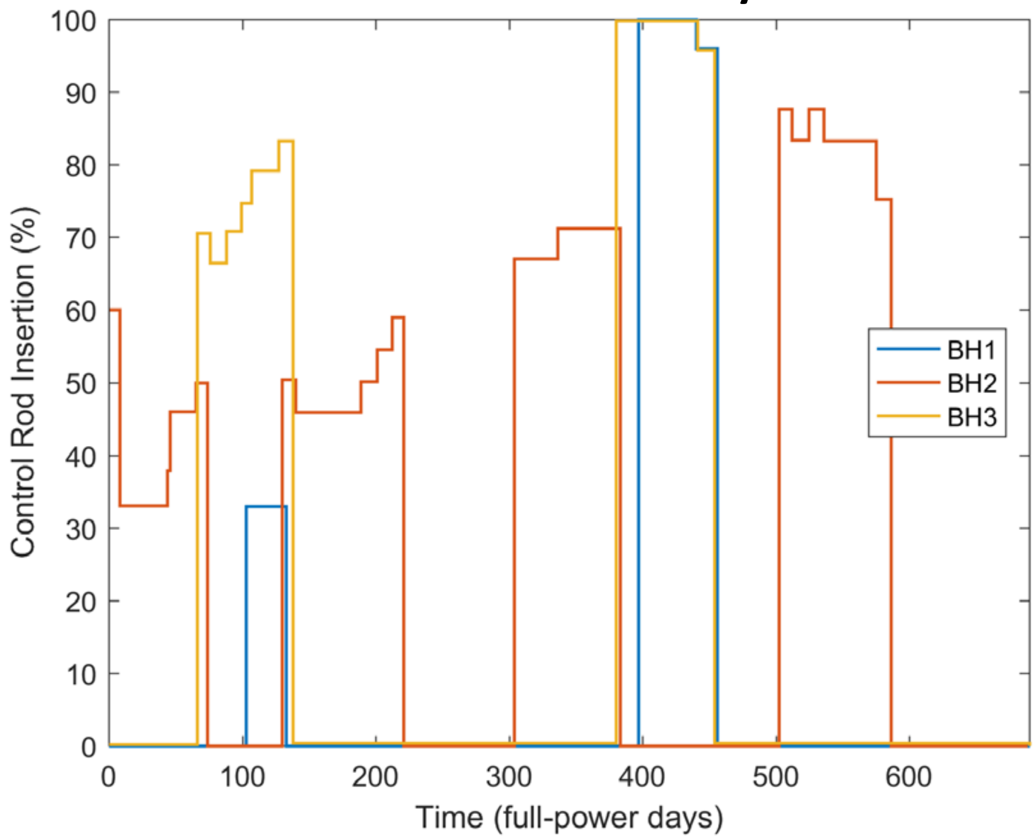
- Model is based on GE 10x10 design
  - 74 UO_2 rods
  - 18 $UO_2 + Gd_2O_3$ rods
  - 2 Large Water rods
- Axial enrichment heterogeneity
- 7 axial layers with heterogenous enrichment and part length rods.
- Natural uranium is used at lattice lower and upper axial layers.



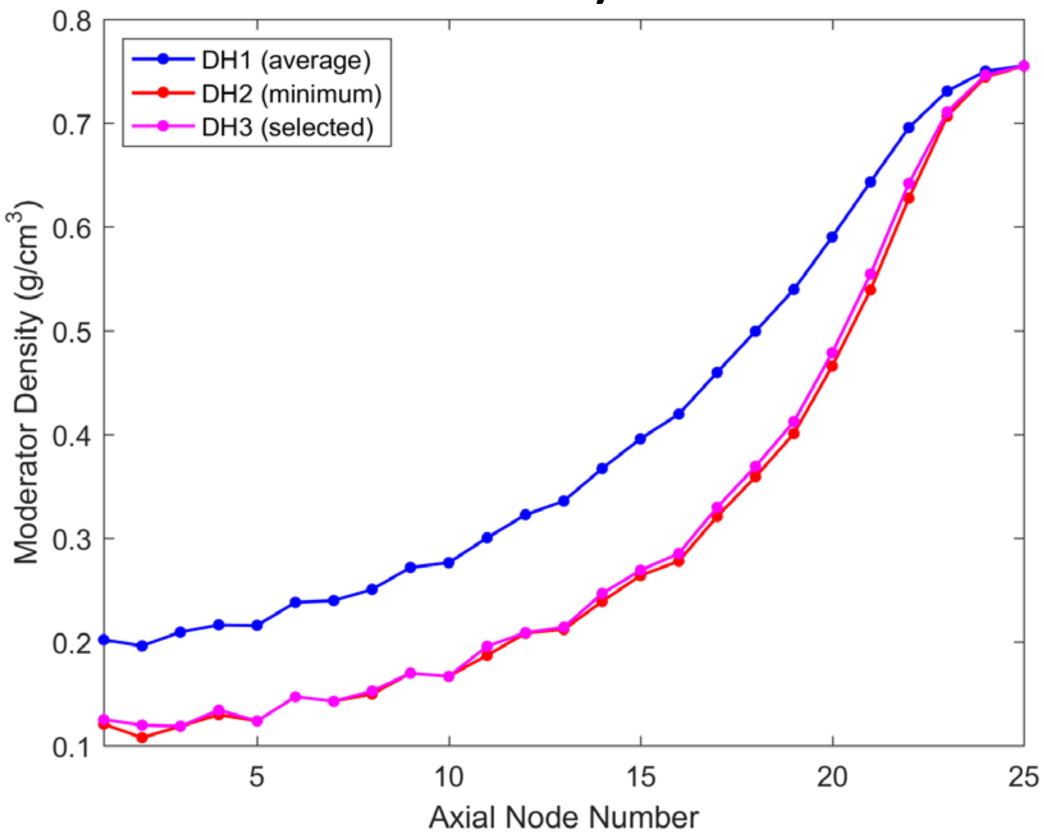
Fensin, M.L., 2004. Optimum Boiling Water Reactor Fuel Design Strategies to Enhance Reactor Shutdown by the Standby Liquid Control System, Master Thesis. University of Florida, USA.

BWR Data and Resources (Marshall, ORNL)

Control Blade History



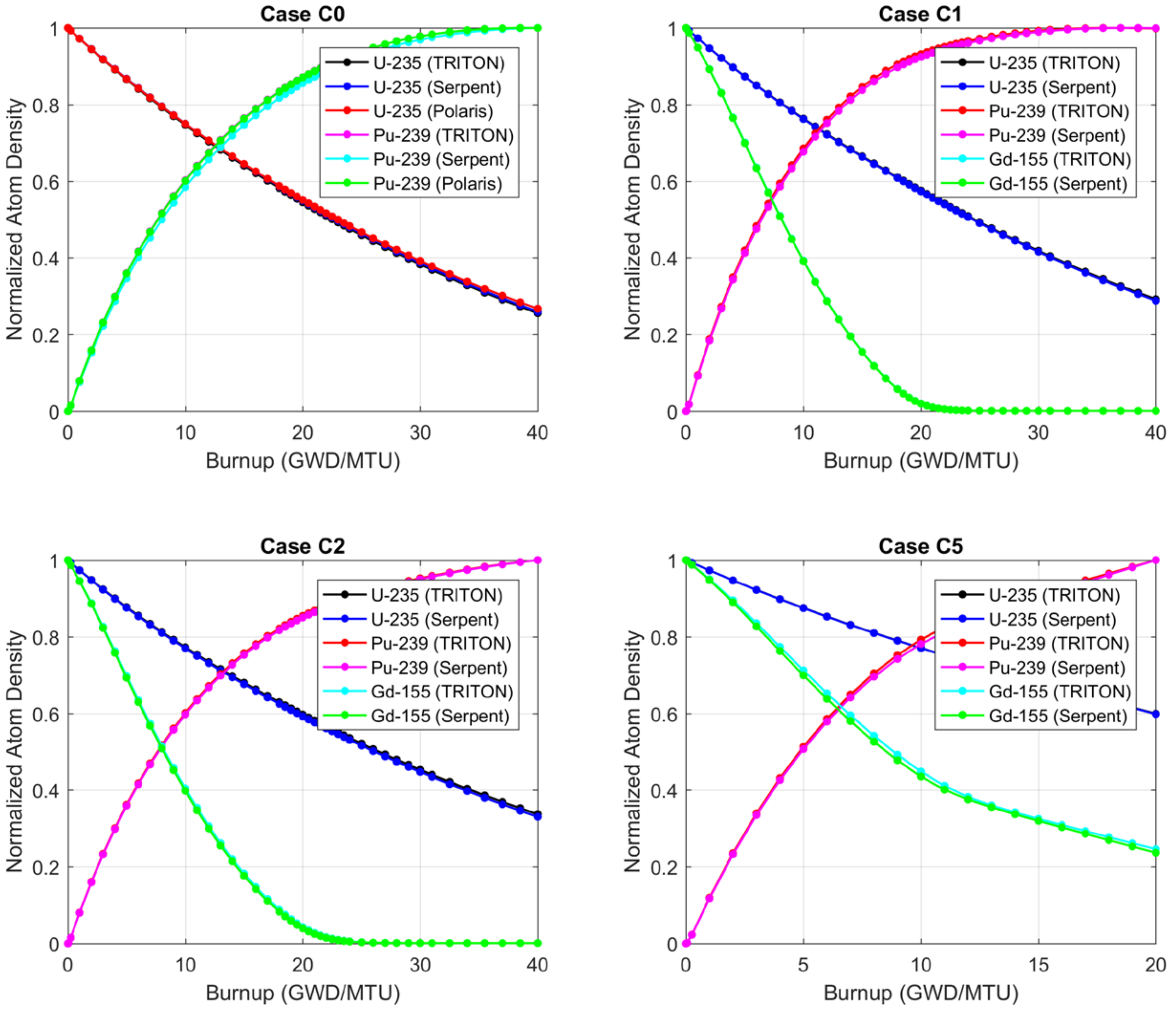
Axial coolant density distribution



W. Marshall, Axial moderator density distributions, control blade usage, and axial burnup distributions or extended BWR burnup credit, Tech. Rep., NUREG/CR-7224, ORNL/TM-2015/544, US Nuclear Regulatory Commission, ORNL 2015

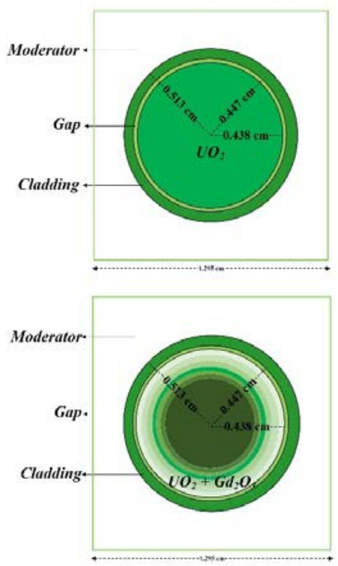
Model Benchmarking

- Relevant isotopic concentration is benchmarked using different codes
- Different 2D and 3D cases are considered.
- Good agreement is observed

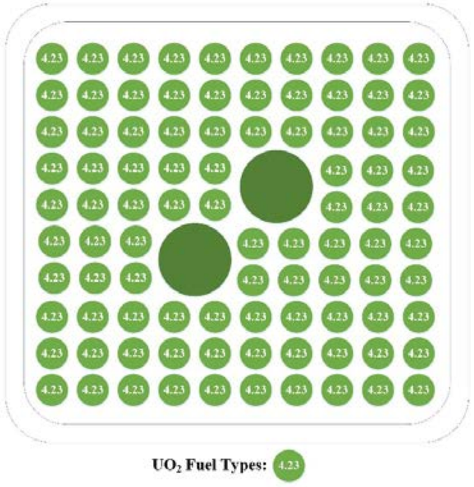


BWR 2D Cases (T-DEPL)

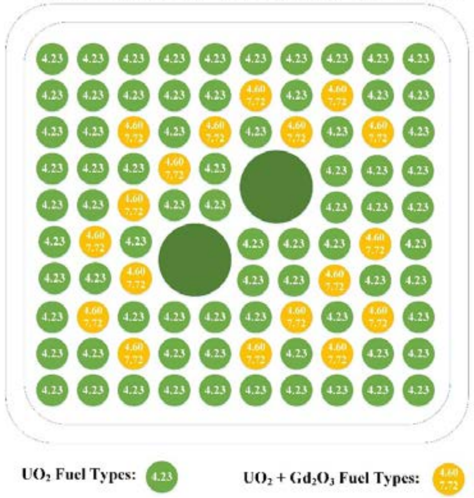
CS0: UO_2 and Gadolinium pin-cell



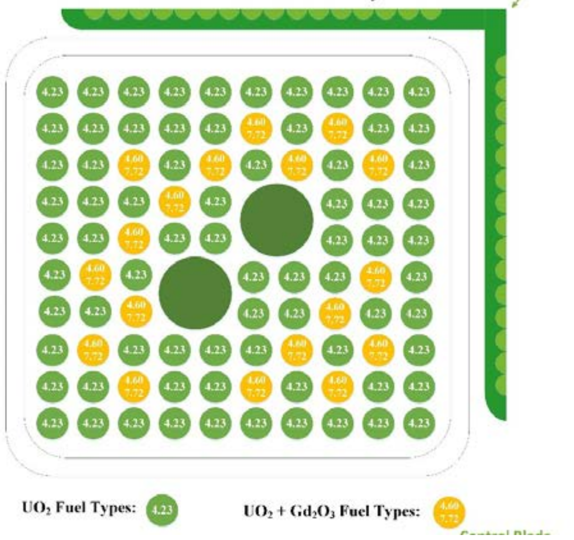
C0: UO_2-only lattice



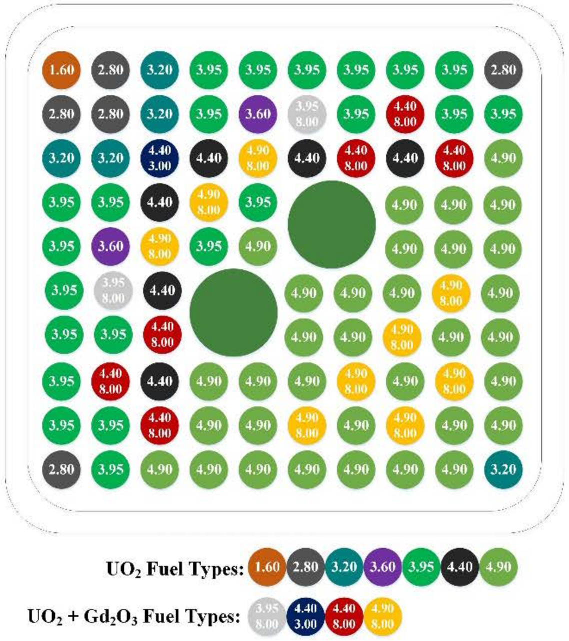
C1: UO_2 and Gad with uniform enrichment



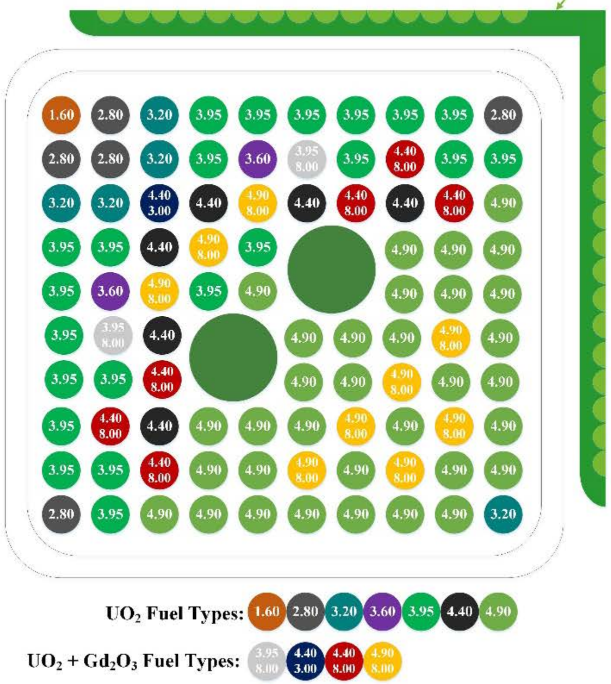
C2: C1 with Control blade fully inserted



C3: UO_2 and Gad with variable radial enrichment

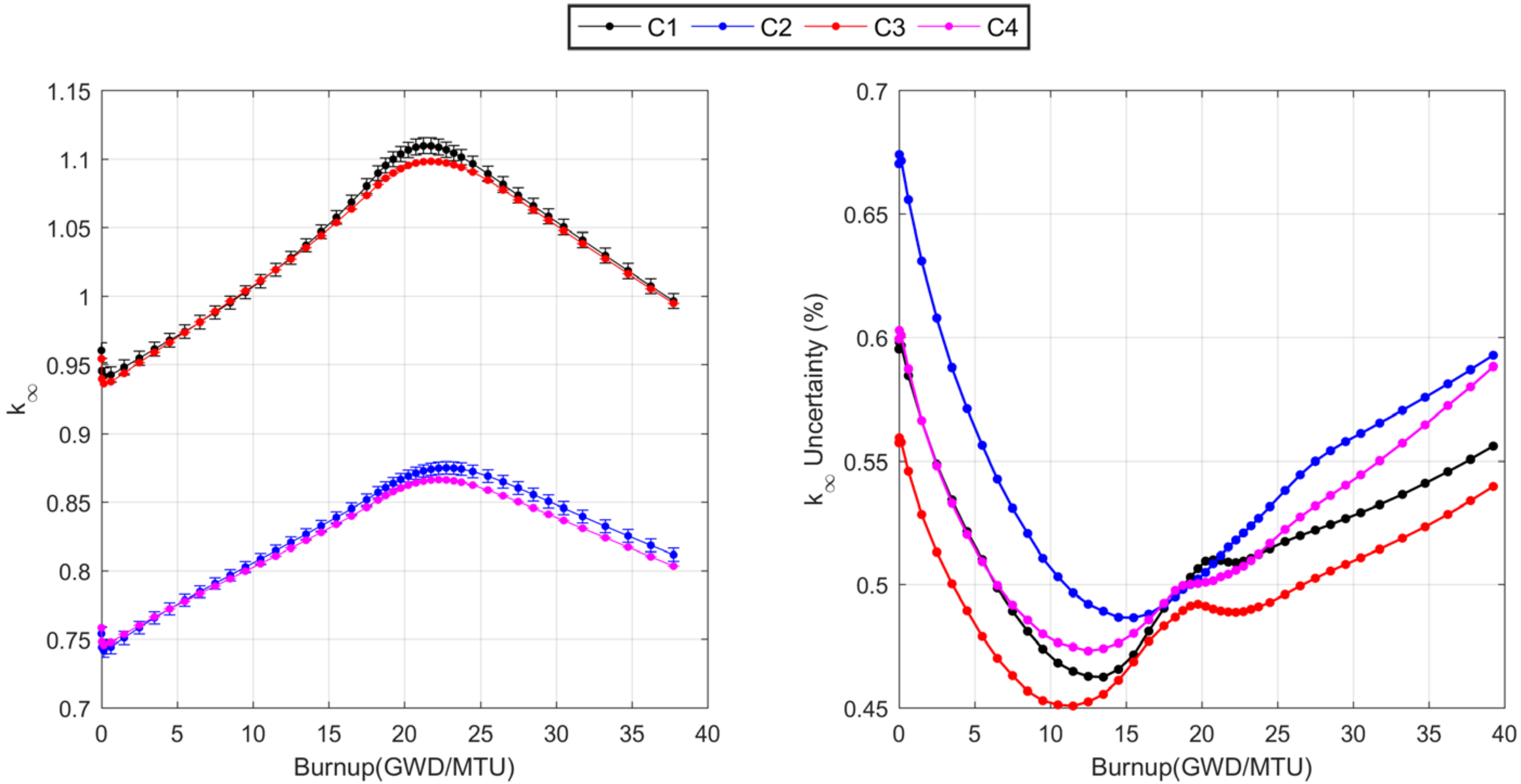


C4: C3 with control rod fully inserted



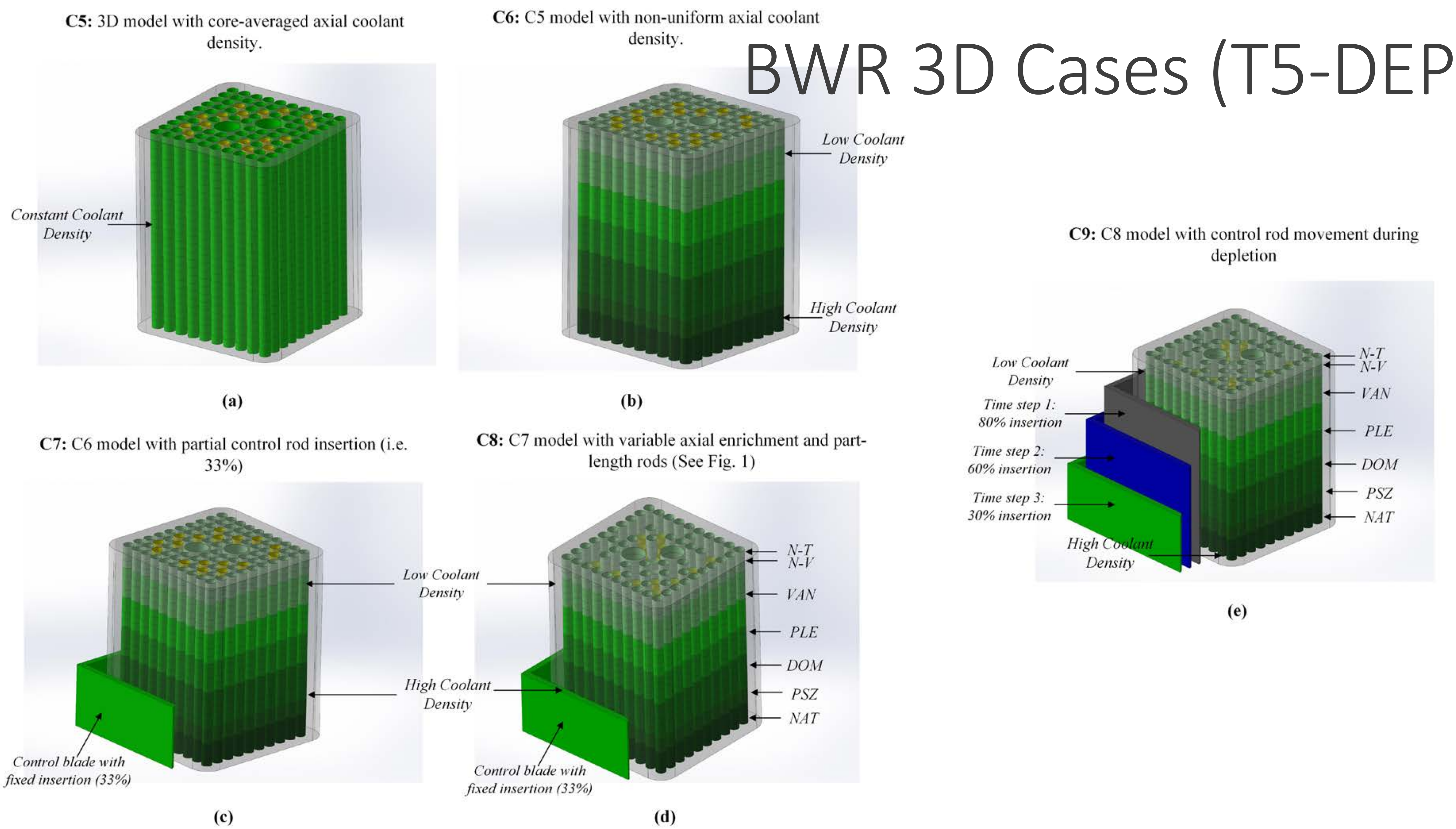
M. I. Radaideh, D. Price, T. Kozlowski, "Uncertainty quantification of BWR criticality safety simulations ," Proceedings of Physics of Reactors (PHYSOR-2018), Cancun, Mexico, April 22-26, pp. 2866-2877 (2018).

2D Results (k_∞ vs Burnup)

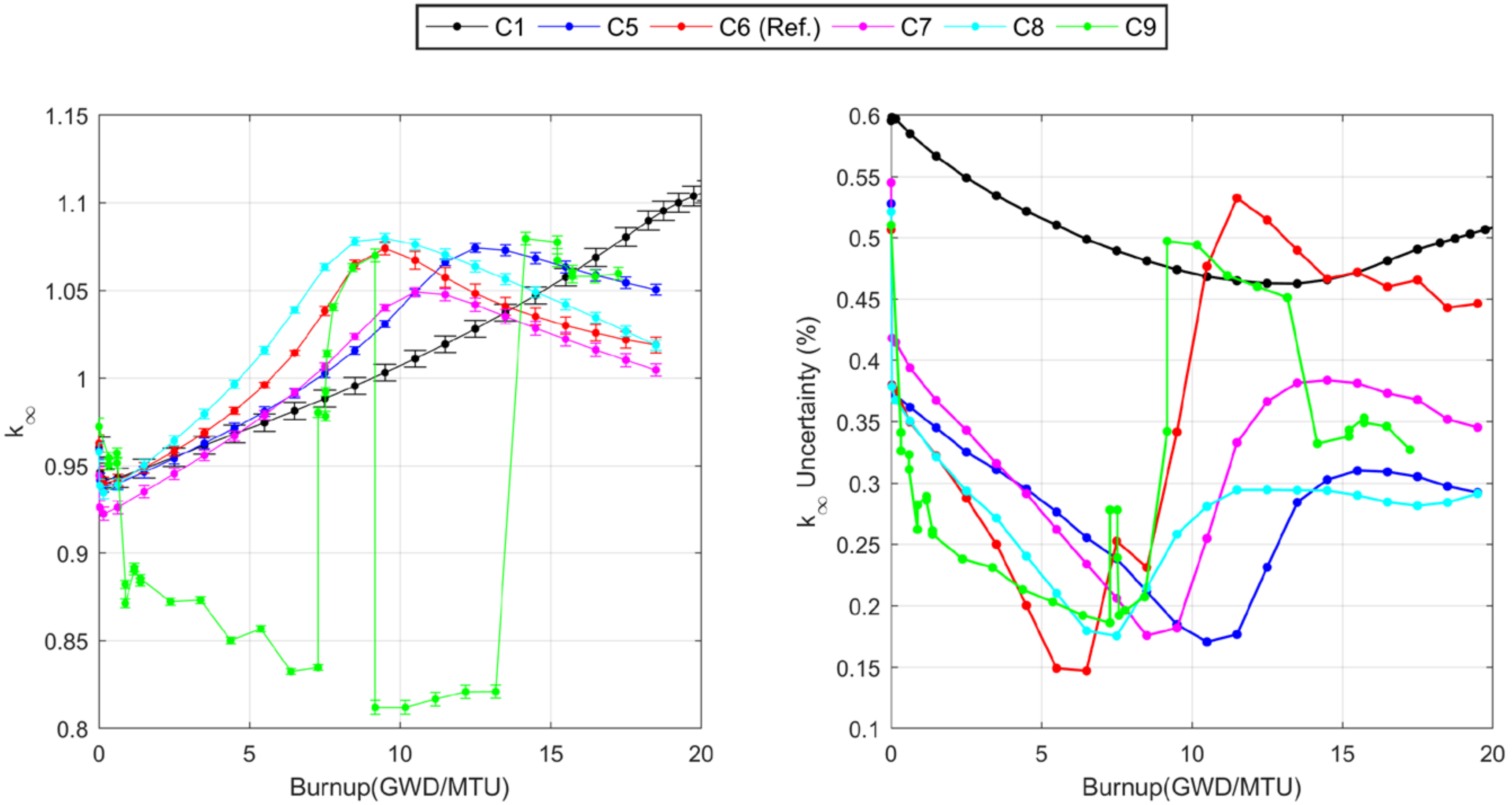


D. Price, M. I. Radaideh, T. Kozlowski (2018)

BWR 3D Cases (T5-DEPL)



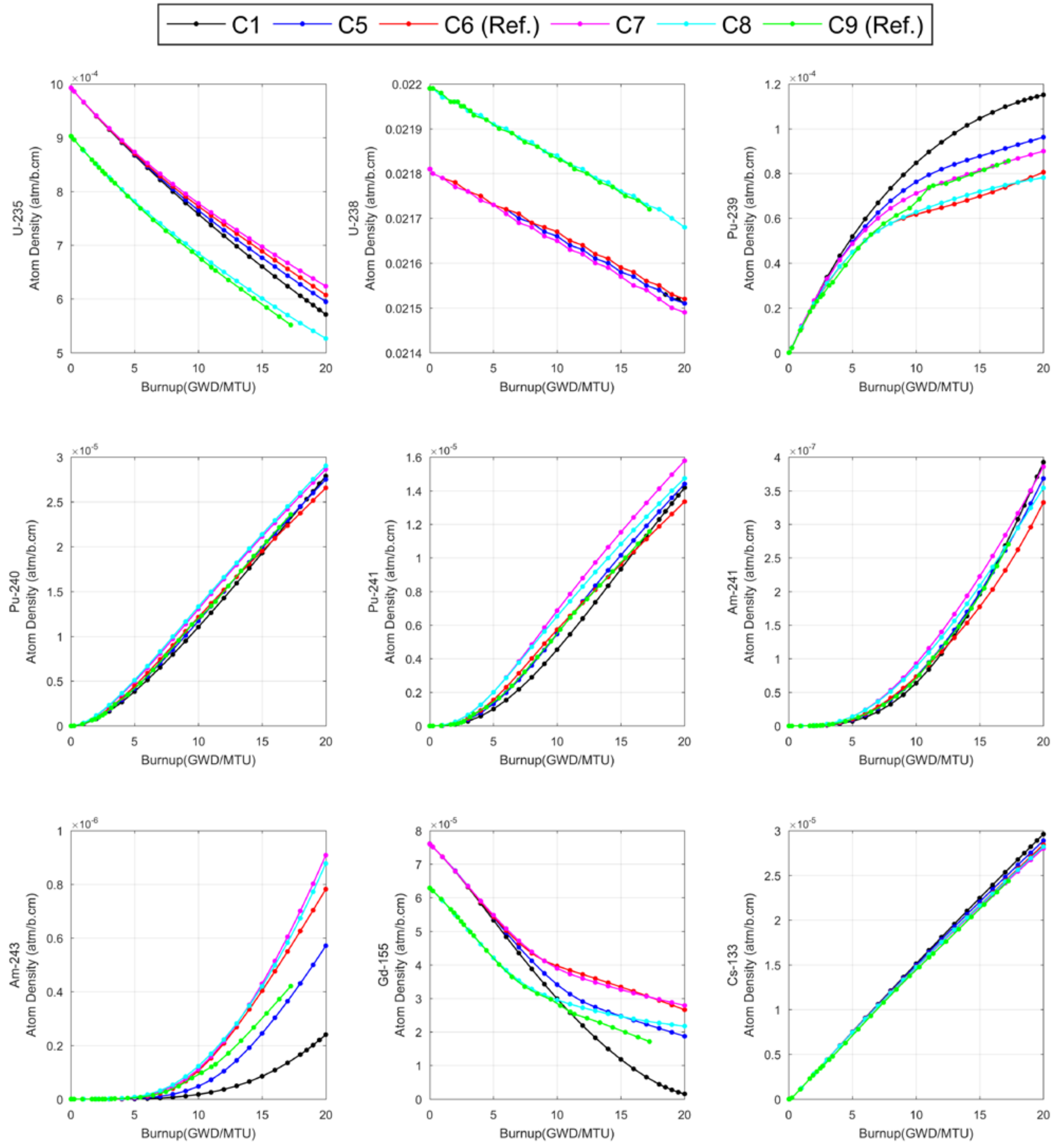
3D Results (k_∞ vs Burnup)



D. Price, M. I. Radaideh, T. Kozlowski (2018)

3D Results: Isotopic Inventory

- **The 2D case C1** is considered the reference case to which other cases are compared.
- **Initial U-235 and U-238 amounts differ for C8 and C9** compared to other cases due to the vanished locations and axial heterogeneity.
- **3D cases deplete Gd-155 slowly** due to the axial burnup profile at the boundaries.
- **Control rod movement (C9)** enhances Pu-239 breeding due to the spectrum hardening.
- **The 2D case reaches the reactivity peak faster** due to the uniform depletion.



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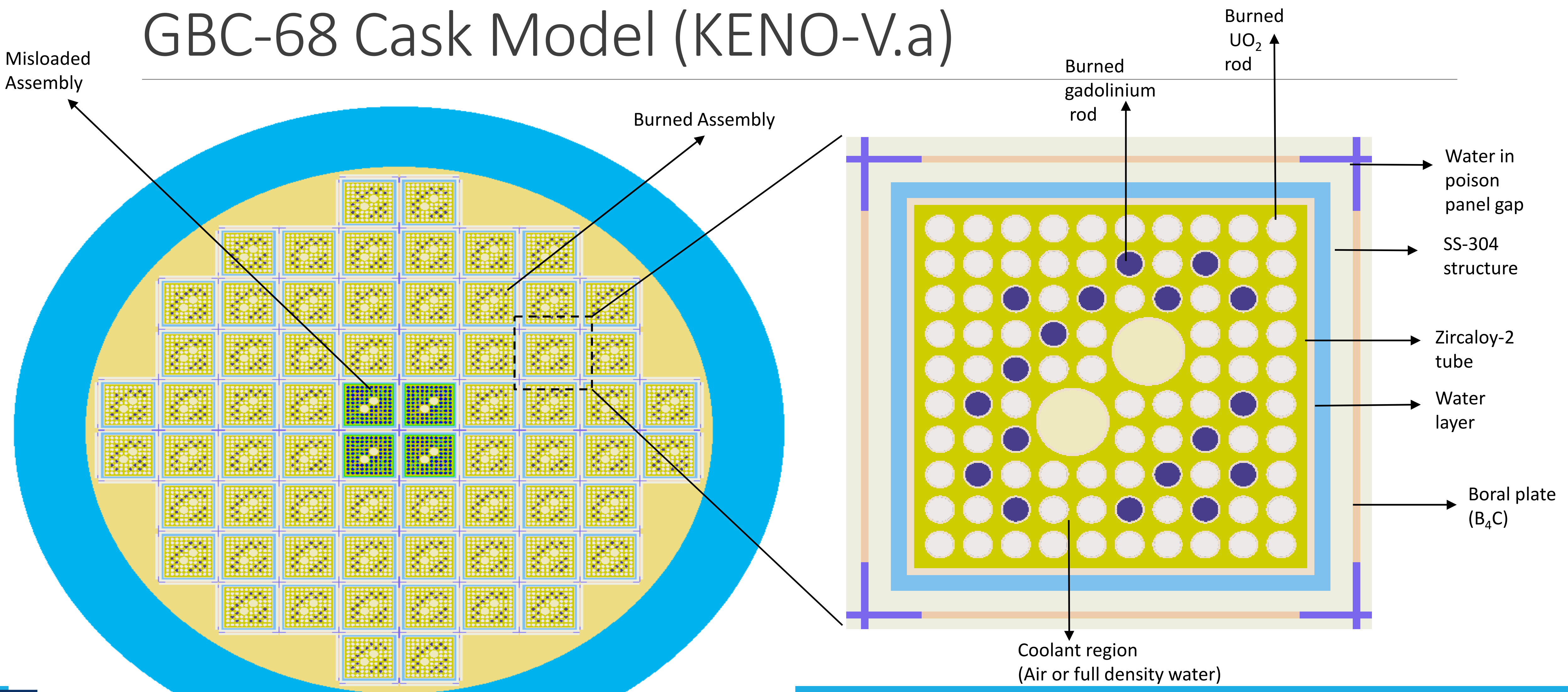
- Background
- Advanced BWR Depletion Cases (T-Depl and T5-Depl)
- **Criticality Calculations (KENO-V.a)**
- Sensitivity and Uncertainty Analyses (TSUNAMI-3D)
- BWR Isotopic Uncertainty Quantification (Sampler)
- Summary

Note: the cask was assumed flooded by full density water to simulate accident scenarios

GBC-68 Cask Model (KENO-V.a)



D.E. Mueller, J.M. Scaglione, J.C. Wagner, S.M. Bowman, Computational Benchmark for Estimated Reactivity Margin from Fission Products and Minor Actinides in BWR Burnup Credit, ORNL Report to the NRC, (NUREG/CR-7157, ORNL/TM-2012/96), pp. 1-8 (2013b).

Cask Criticality

- Results from fuel discharged at 40GWD/MTU
- Rodded cases yield more critical cask
- Heterogeneous enrichment has little effect on cask criticality
- Cask criticality results for C7-C9 are in progress

Case Number	Description	k_{eff}
C0	Pure UO2	0.65864
C1	Homo	0.67702
C2	Homo, CR	0.72709
C3	Hetero	0.67792
C4	Hetero, CR	0.72966
C5	Constant void	0.71671
C6	Void distribution*	0.65610

All statistical uncertainty is within 10 pcm

*based on DH1, the void distribution based on the average throughout the core

Contents

Nuclear Data Sensitivity (Cask k_{eff}) via TSUNAMI-3D

Case C0

Nuclide	Reaction1; Reaction 2	Δk/k
U-238	n, gamma; n, gamma	0.220
Pu-239	fission; fission	0.207
U-235	nubar; nubar	0.151
Pu-239	fission; n, gamma	0.131
Pu-239	n, gamma; n, gamma	0.095

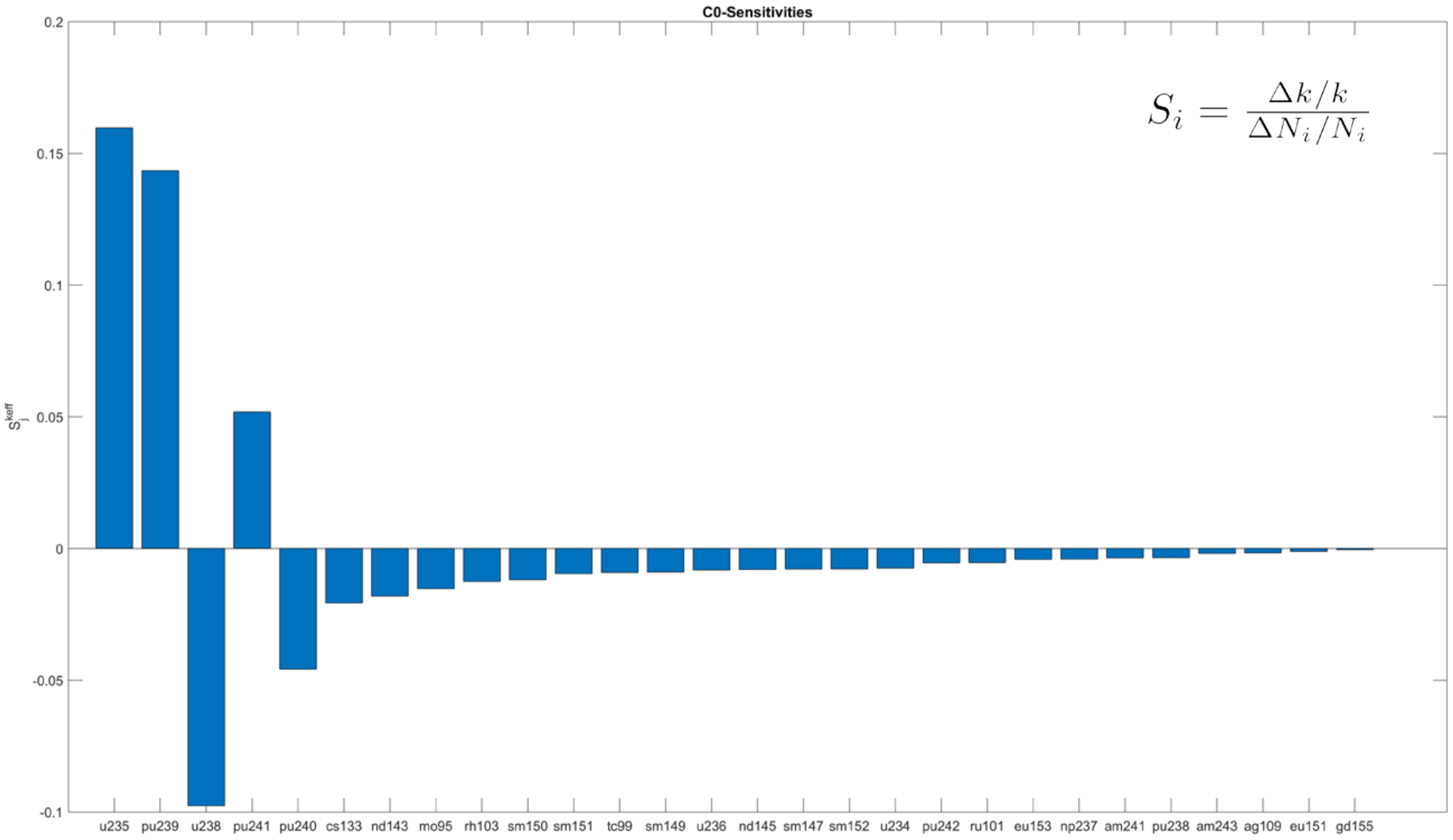
Case C1

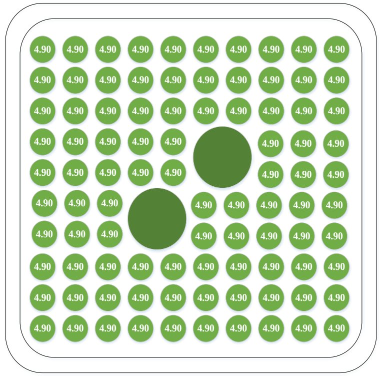
Nuclide	Reaction1; Reaction 2	Δk/k
U-238	n, gamma; n, gamma	0.209
Pu-239	fission; fission	0.193
U-235	nubar; nubar	0.159
Pu-239	fission; n, gamma	0.128
Pu-239	n, gamma; n,gamma	0.096

Case C2

Nuclide	Reaction1; Reaction 2	Δk/k
Pu-239	fission; fission	0.201
U-238	n, gamma; n, gamma	0.190
U-235	nubar; nubar	0.142
Pu-239	fission; n, gamma	0.142
Pu-239	n, gamma;n, gamma	0.114

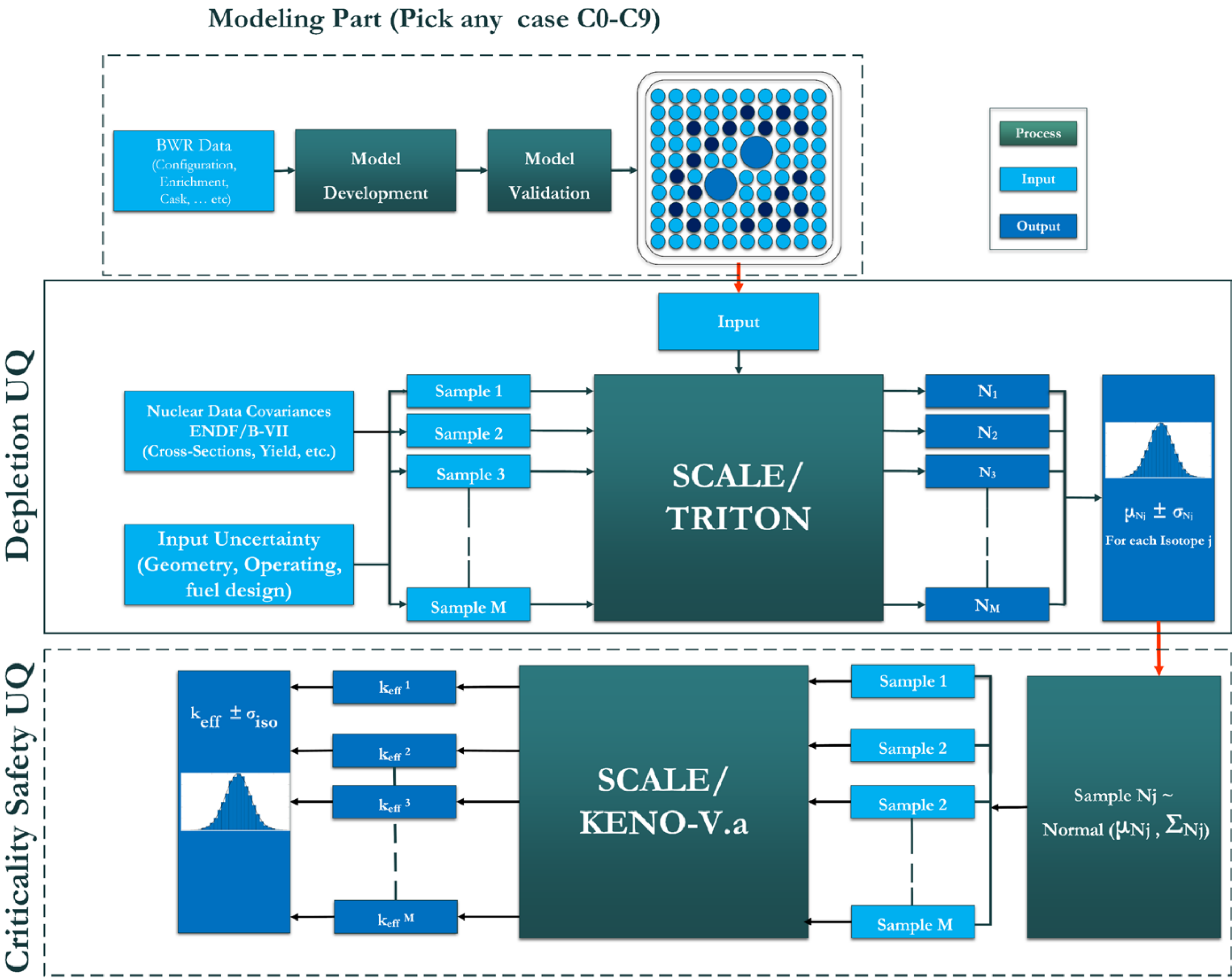
Cask k_{eff} Isotopic Sensitivities based on C0 depletion (EOL)



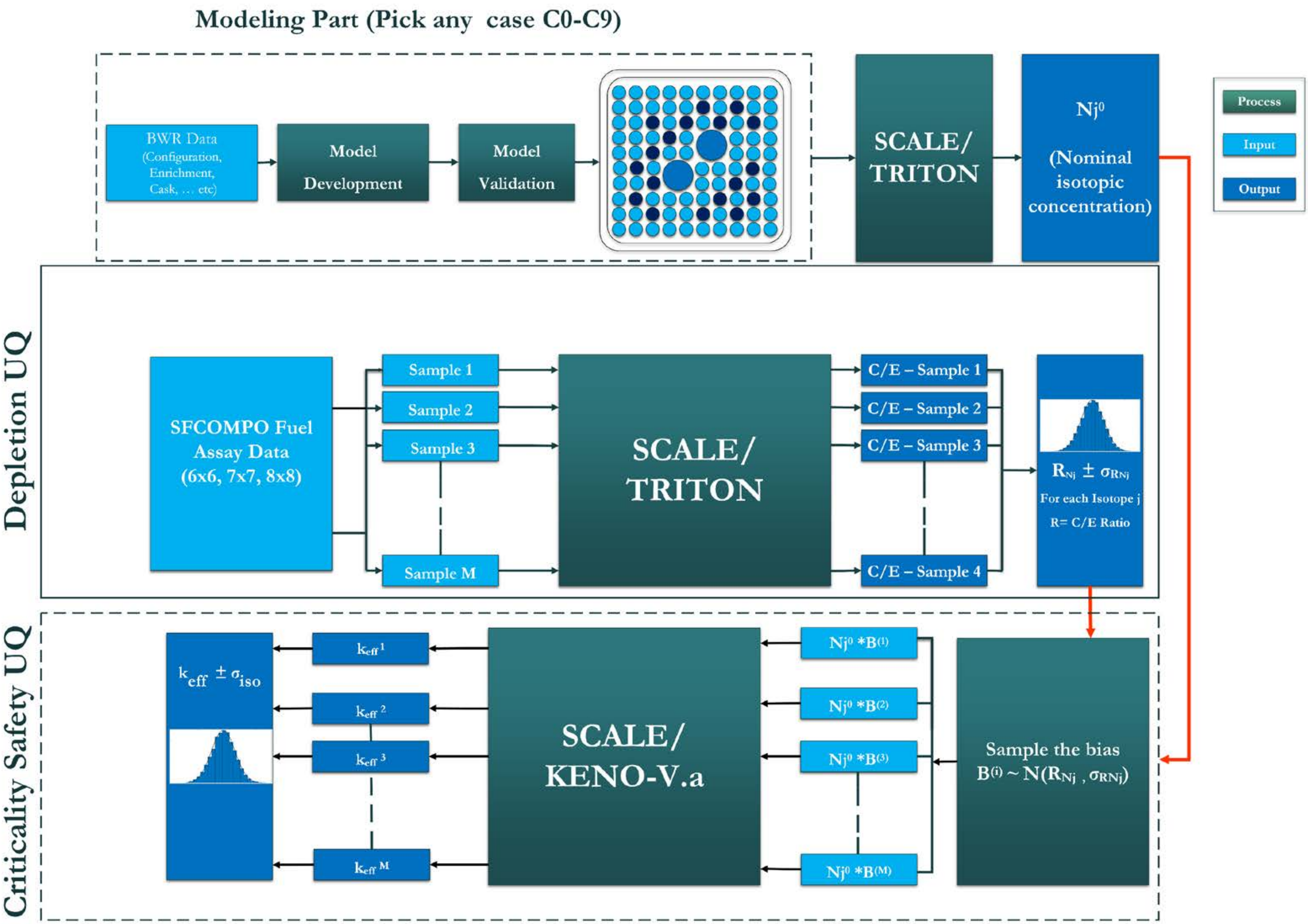


Contents

Computational vs Data-driven
(Computational method)



Computational vs Data-driven (Data method)



I. C. Gauld, "Strategies for application of isotopic uncertainties in burnup credit," ORNL/TM-2001/257, Oak Ridge National Laboratory, Oak Ridge, Tennessee (2003).

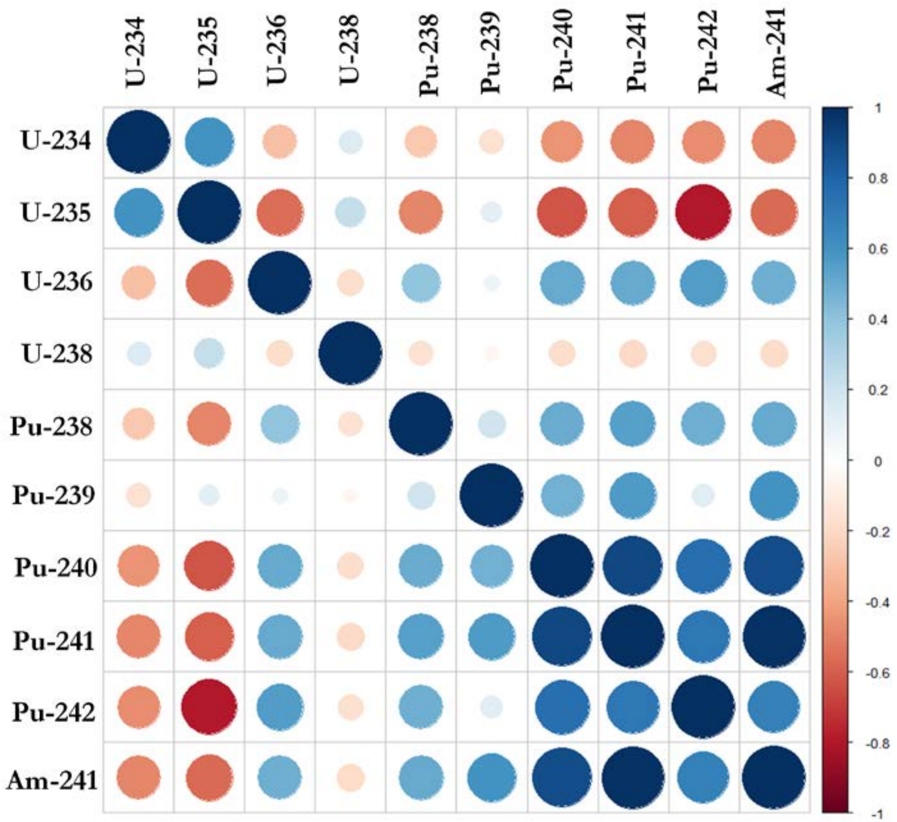
[3] S. M. Bowman, ”SCALE 6: comprehensive nuclear safety analysis code system,” Nuclear Technology, 174(2), pp. 126-148 (2011)
[8] G. Ilas, H. Liljenfeldt, ”Decay heat uncertainty for BWR used fuel due to modeling and nuclear data uncertainties,” Nuclear Engineering and Design, 319, 176-184 (2017).
[9] A. Hoefer, T. Ivanova, B. Rearden, D. Mennerdahl, O. Buss, ”Proposal for Benchmark Phase IV Role of Integral Experiment Covariance Data for Criticality Safety Validation,” Working Party on Nuclear Criticality Safety, EG UACSA, OECD/NEA (2014).

Computational-driven Approach

Input parameters' uncertainty for criticality safety applications

Parameter	1-σ (%)	Source
Geometry		
Pellet radius	0.14%	[9, 10]
Clad Inner Diameter	0.43%	[9, 10]
Clad Outer Diameter	0.46%	[9, 10]
Material		
U-235 wt%	0.60%	[8]
Gd_2O_3 wt%	1.67%	[8]
UO_2 Density	0.13%	[9, 10]
Operating		
Specific Power	1.67%	[8, 11, 12]
Coolant Density	3.33%	[8, 11, 12]
Fuel Temperature	3.33%	[8, 11, 12]
Nuclear Data		
Neutron Cross-sections	56group-COV	[3]
Fission Yield	56group-COV	[3]
Decay Data	56group-COV	[3]

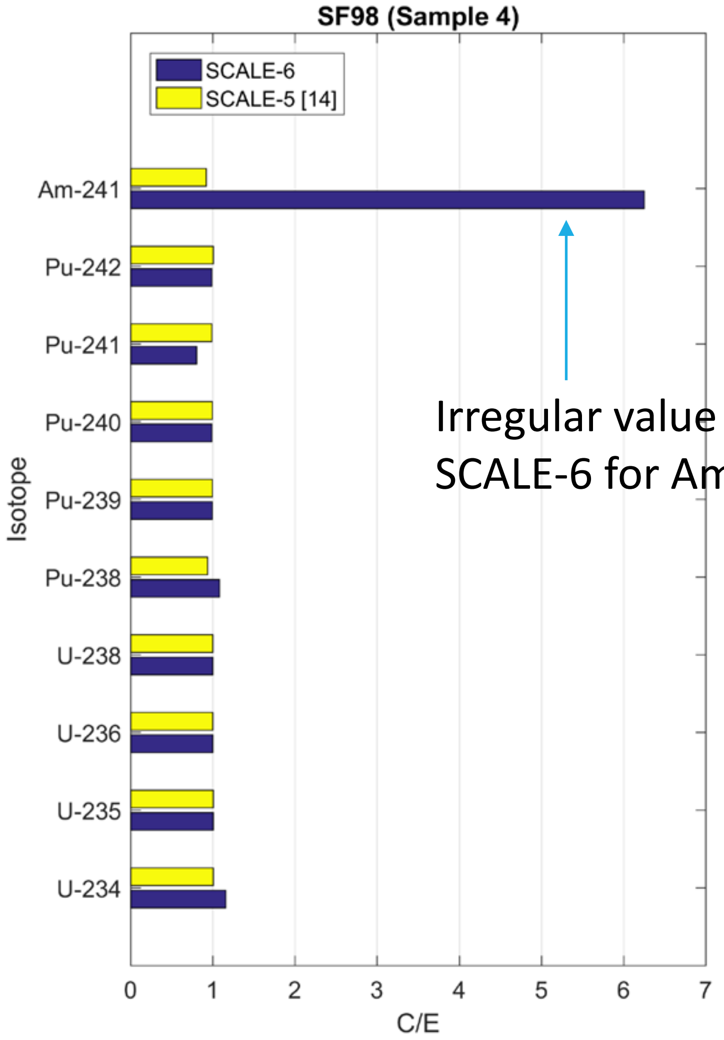
Correlation matrix between actinides based on case C1



[10] NEA Nuclear Science Committe, ”International Handbook of Evaluated Criticality Safety Benchmarks Experiments,” OECD Nuclear Energy Agency (1999).
[11] G. M. Grandi, J. A. Borkowski, ”Benchmark of SIMULATE-3K against the Frigg loop stability experiments,” In: CD Proceedings of Advances in Nuclear Fuel Management III, Hilton Head Island, USA (2003).
[12] M. Kruners, G. Grandi, M. Carlssonc, ”PWR transient xenon modeling and analysis using Studsvik CMS,” In: Proceeding of LWR Fuel Performance/TopFuel/WRFPM, Orlando, USA (2010).

[14] U. Mertyurek, U., M. W. Francis, I. C. Gauld, "SCALE 5 analysis of BWR spent nuclear fuel isotopic compositions for safety studies," ORNL/TM-2010/286, Oak Ridge National Laboratory, Oak Ridge, Tennessee (2010).

Data-driven Approach



Irregular value by SCALE-6 for Am-241

$\bar{R}_n$: mean of the bias (C/E)
σ_{R_n}: standard deviation of the bias
N_s^n: number of experimental samples used

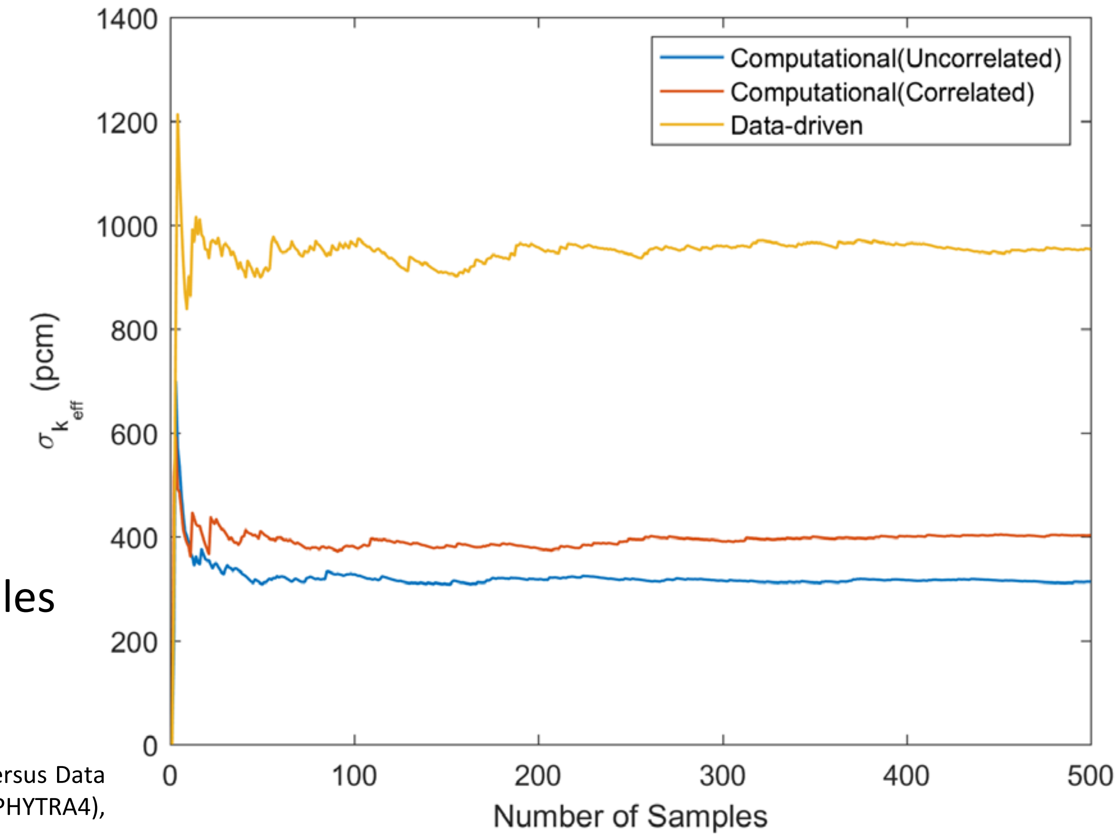
Nuclide	Geometry	$\bar{R}_n$	σ_{R_n}	N_s^n
U-234	8x8, 7x7	1.077	0.079	20
U-235	8x8, 7x7, 6x6	0.992	0.055	32
U-236	8x8, 7x7, 6x6	0.979	0.020	32
U-237	8x8, 7x7, 6x6	0.999	0.005	32
Pu-238	8x8, 7x7, 6x6	0.968	0.101	32
Pu-239	8x8, 7x7, 6x6	1.006	0.052	32
Pu-240	8x8, 7x7, 6x6	0.997	0.037	32
Pu-241	8x8, 7x7, 6x6	0.949	0.102	32
Pu-242	8x8, 7x7, 6x6	1.008	0.062	32
Am-241	8x8, 7x7	1.087	0.121	20

e, M. I. Radaideh, T. Kozlowski (2018)

Final cask k_{eff} Uncertainty due to isotopic Uncertainty

Method	Nominal k_{eff}	$\overline{k_{eff}}$	1-σ (%)	$\sigma_{k_{eff}}$ (pcm)
Computational (Uncorrelated)	0.79747	0.79762	0.39	314
Computational (Correlated)	0.79747	0.79812	0.50	403
Data-driven	0.79747	0.79601	1.20	953

The calculations were done using KENO-V.a + Sampler for 500 samples



M. I. Radaideh, D. Price, T. Kozlowski, " On Uncertainty Quantification of Isotopic Uncertainties Using Computational Versus Data Driven Approaches ," The 4th International Conference on Physics and Technology of Reactors and Applications (PHYTRA4), Marrakesh, Morocco, September 17-19 (2018).

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Summary

- Advanced BWR modelling is necessary to capture the complexities associated with the BWR design, which in turn yield an accurate criticality safety calculations.
- 2D modelling shows minimal effects of radial enrichment averaging.
- 3D models show different behaviour for U-235 and Gd-155 depletion, and Pu-239 breeding.
- In general, the uncertainty due to cross-section covariances decreases during the cycle
- The data driven method for performing UQ on the cask yields a higher uncertainty **(additional enchantment of the database will be done in future).**
- Correlation between isotopes tends to increase the k_{eff} uncertainty due to isotopic uncertainty.
- **Our future work will focus on combining the computational and data-driven approaches into one hybrid approach through quantifying the code bias.**
- Even when flooded, the BWR cask remains subcritical in all cases analysed.

Thank you!

- **<u>Our publications for more info</u>**
- M. I. Radaideh, D. Price, T. Kozlowski, ”Criticality and uncertainty assessment of assembly misloading in BWR transportation cask,” Annals of Nuclear Energy, 113, pp. 1-14 (2018).
- M. I. Radaideh, D. Price, T. Kozlowski, ”Uncertainty quantification of BWR criticality safety simulations ,” Proceedings of Physics of Reactors (PHYSOR-2018), Cancun, Mexico, April 22-26, pp. 2866-2877 (2018).
- M. I. Radaideh, D. Price, T. Kozlowski, ” On Uncertainty Quantification of Isotopic Uncertainties Using Computational Versus Data Driven Approaches ,” The 4th International Conference on Physics and Technology of Reactors and Applications (PHYTRA4), Marrakesh, Morocco, September 17-19 (2018).
- **<u>To come soon</u>**
- M. I. Radaideh, D. Price, D. O’Grady, T. Kozlowski, ”Advanced BWR Criticality Safety, Part I: Model Development, Model Benchmarking, , and Depletion with Uncertainty analysis,” to be submitted to Nuclear Engineering and Design (2019).
- D. Price, M. I. Radaideh, D. O’Grady, T. Kozlowski, ”Advanced BWR Criticality Safety, Part II: Cask Criticality, Burnup Credit, Sensitivity, and Uncertainty Analyses,” to be submitted to Nuclear Engineering and Design (2019).