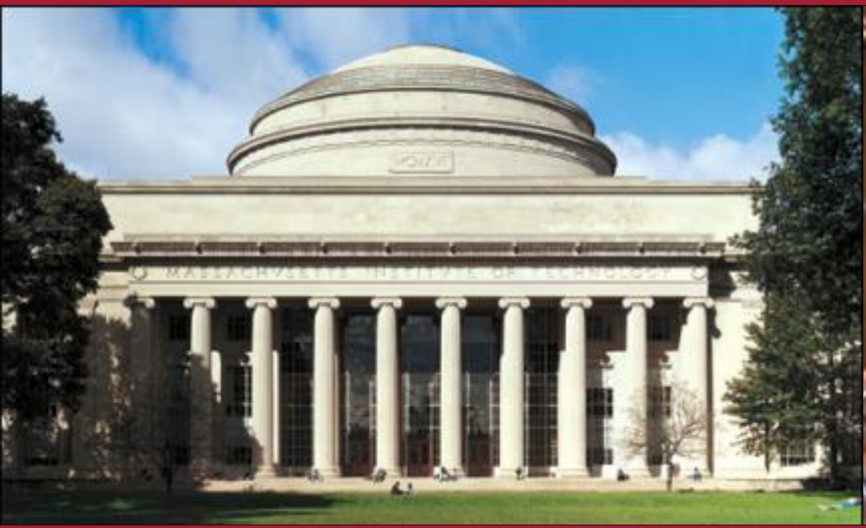
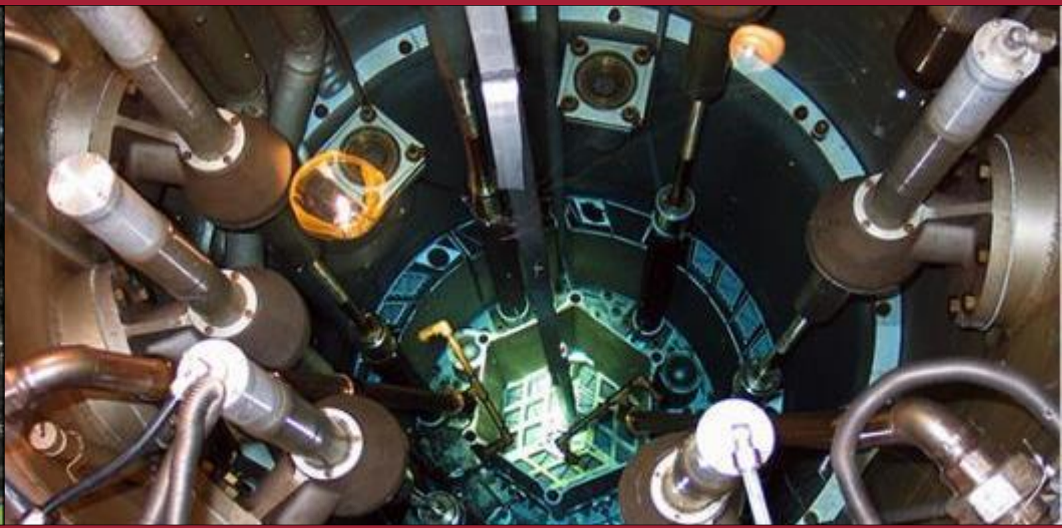
MIT NUCLEAR REACTOR LABORATORY

an MIT Interdepartmental Center





Technical Analysis and Administrative Issues of Criticality Study for Different MITR Facilities

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Group Leader – Reactor Physics, Research Scientist, Reactor Engineer, Criticality Officer

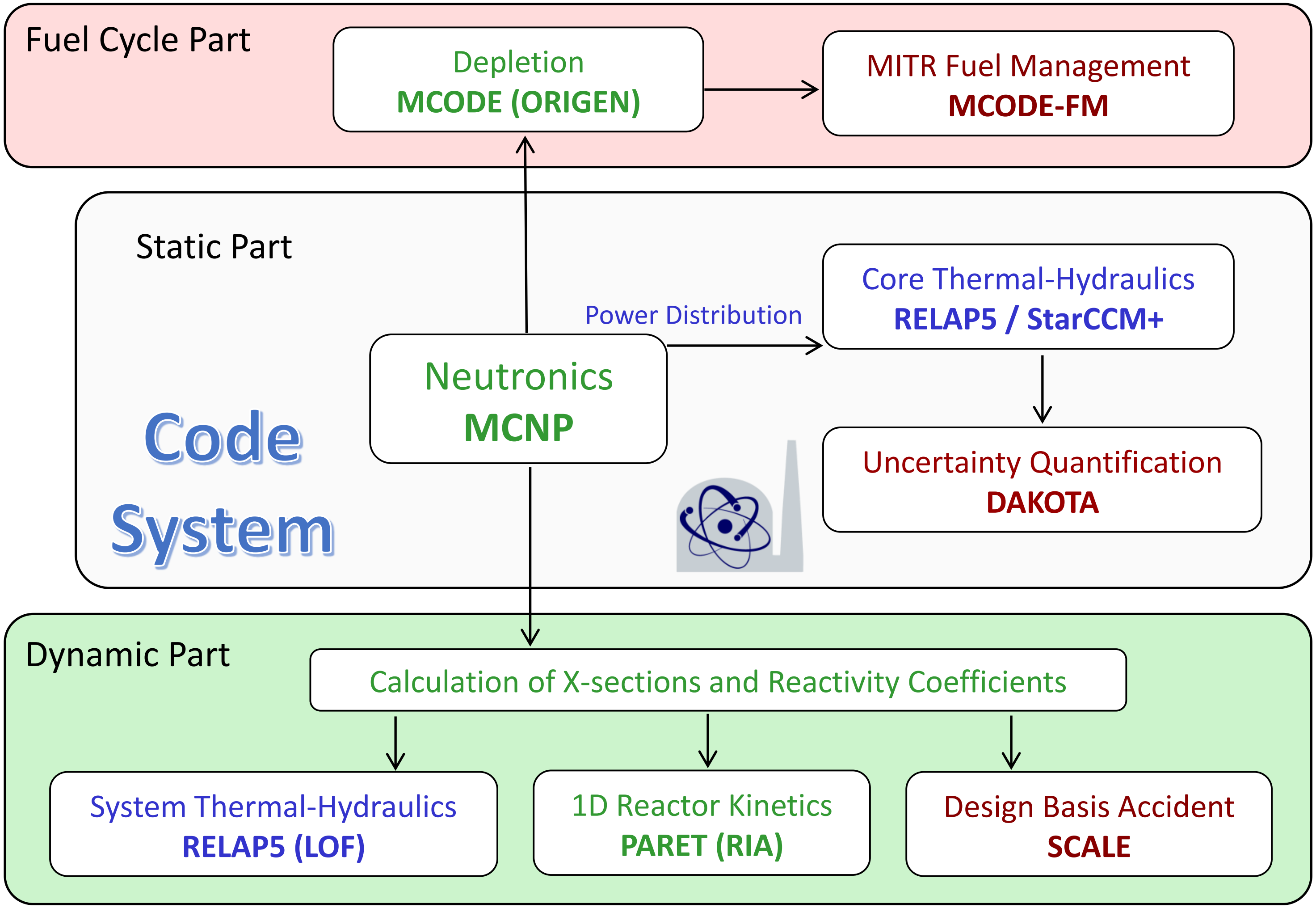
TRTR 2017 Meeting, San Diego, 09/18/2017

MIT Research Reactor (MITR)

NSUF Nuclear Science User Facilities

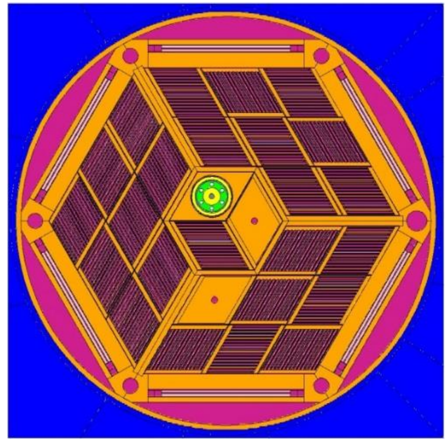
- Part of interdepartmental Nuclear Reactor Laboratory
- Built on the MIT campus in 1958, upgraded in 1976
- 6 MW_{th} - the 2nd largest university reactor in U.S.
- Light water-cooled, heavy water-reflected
- Operates 24/7, up to 10-week cycles

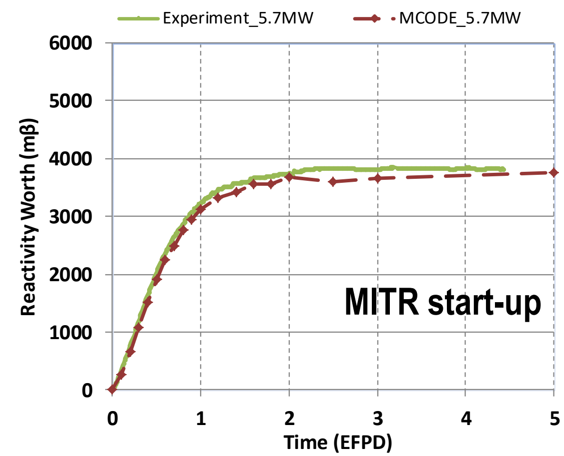
Code System

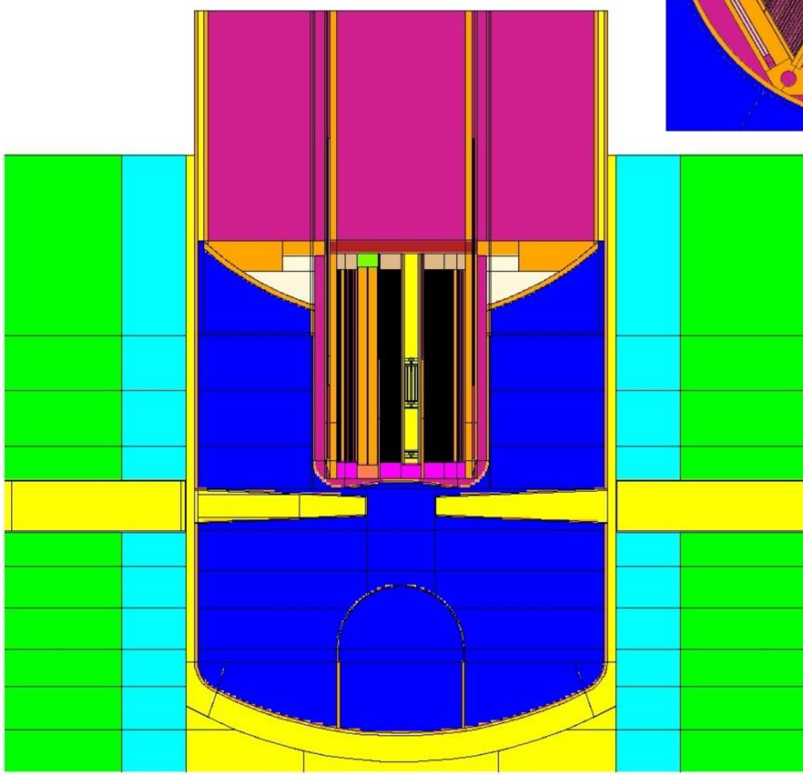

Fuel Cycle Part
Depletion
MCODE (ORIGEN)
MITR Fuel Management
MCODE-FM
Static Part
Power Distribution
Core Thermal-Hydraulics
RELAP5 / StarCCM+
Neutronics
MCNP
Code
System
Uncertainty Quantification
DAKOTA
Dynamic Part
Calculation of X-sections and Reactivity Coefficients
System Thermal-Hydraulics
RELAP5 (LOF)
1D Reactor Kinetics
PARET (RIA)
Design Basis Accident
SCALE

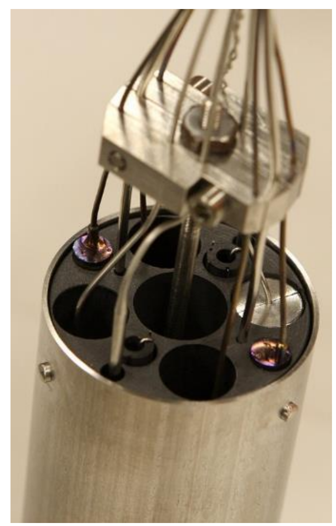
MITR Modeling & Fuel Management

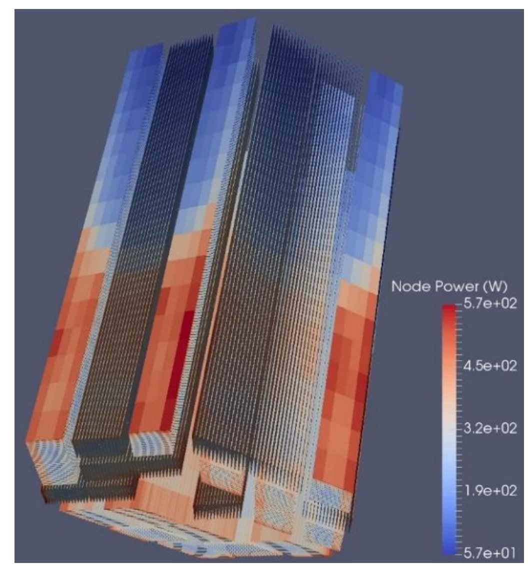
- Detailed MCNP modeling
- Extensive experimental validations
- Criticality (shim bank height) search
- Tracking rhomboid-shaped fuel elements being rotated and/or flipped











Power Distribution in MITR Core

MITR Modeling & Fuel Management

- Detailed MCNP modeling
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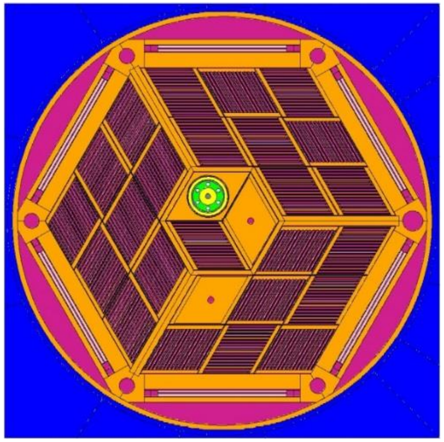
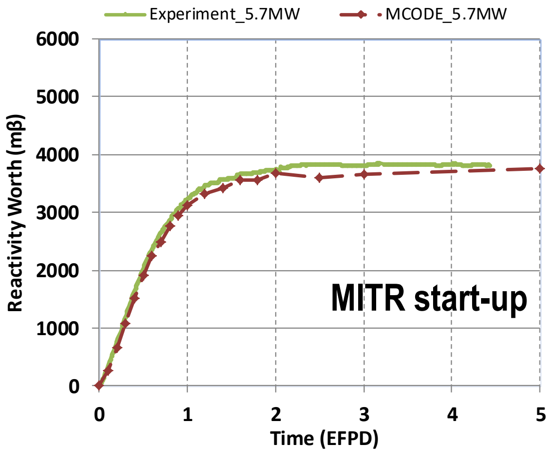
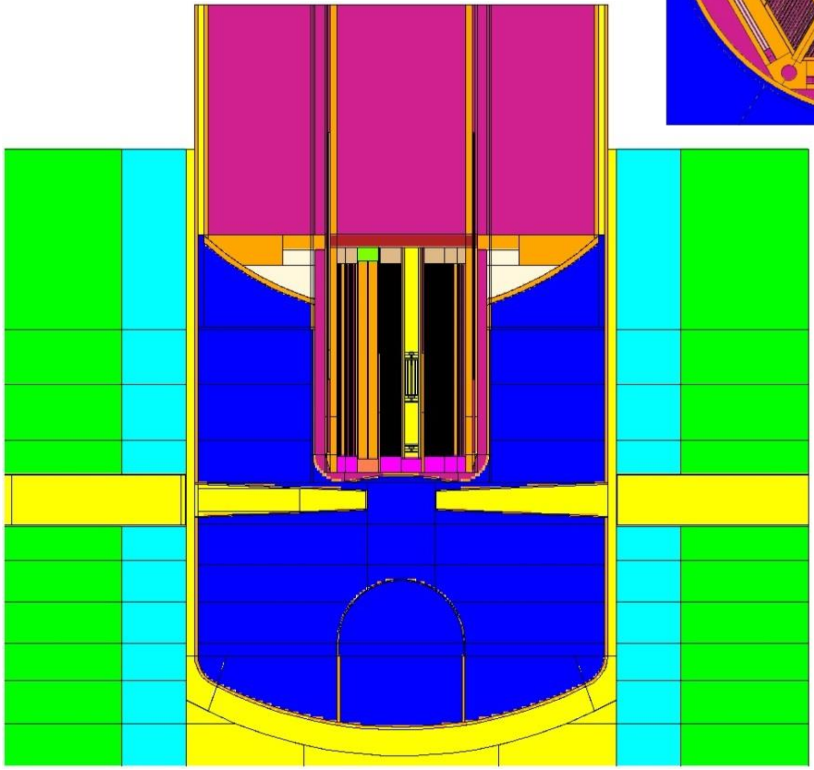
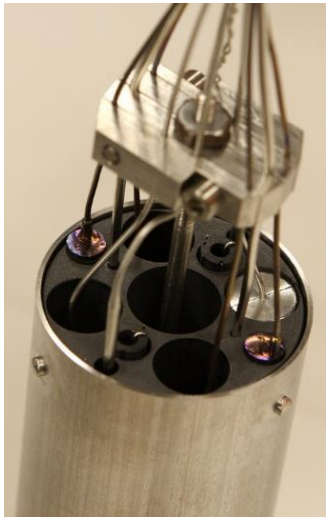
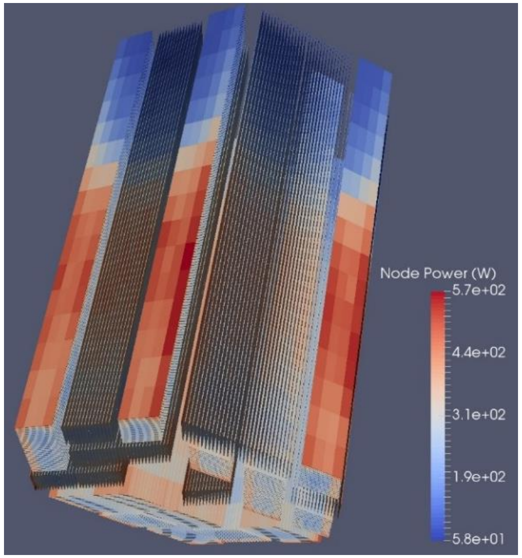










Power Distribution in MITR Core

Background

- During recent years, U.S. Nuclear Regulation Commission (NRC) enhances the criticality safety regulations, emphasis being placed on the validation requirements for the corresponding neutronics calculations.

- In the past two years, there are four criticality studies being required to the Criticality Officer for analyzing multiple MITR facilities with fissionable material involved:
  I. Wet Storage Systems (Spent Fuel Pool and Wet Storage Ring)
  II. Special Nuclear Material Vault
  III. Exponential Graphite Pile (Storage and Operation)

- Most existing criticality reports (if there is any) for the above mentioned facilities are out dated and lack of sufficient technical details

- There are needs to perform up-to-date calculations for the license renewal (and/or accommodate the new regulation requirements)

Objectives

1. **Technical:** There is a clear trend that NRC pushed to implement neutronics validations for the calculation results, where newer versions of ANSI/ANS Standards (Series 8) is particularly requested to be followed. <u>*How other Research Reactors accommodate this request?*</u>

2. **Administrative:** At least at MITR, there is no specific/clear funding source supporting criticality safety analysis and validation report. <u>*How other Research Reactors solve the financial issue?*</u>

Criticality Safety Analyses

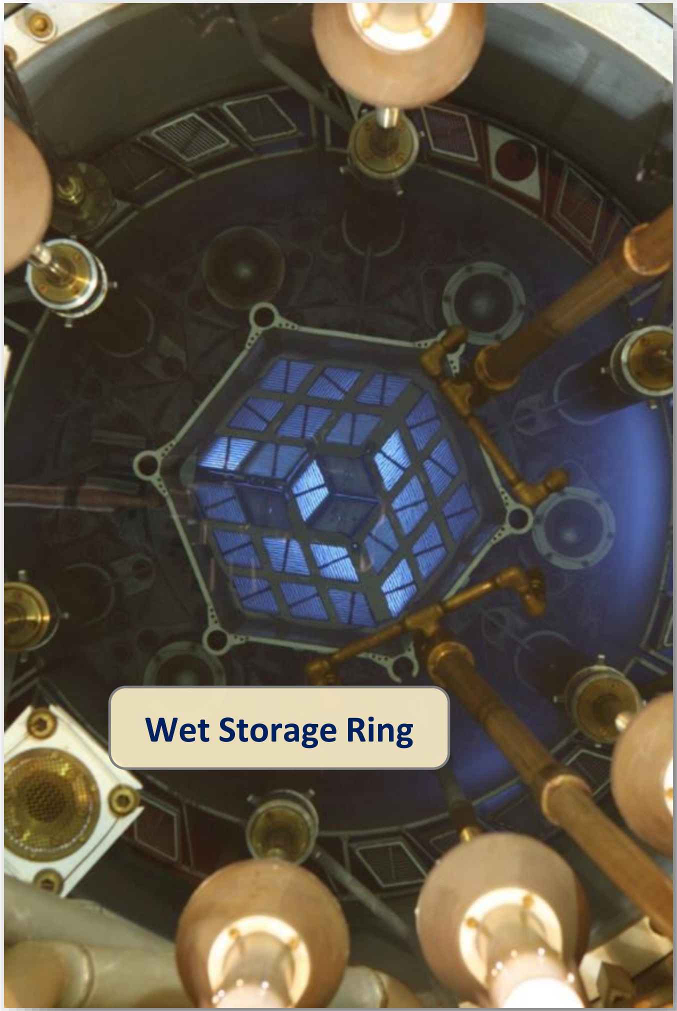
1. **Wet Storage Systems** (Spent Fuel Pool and Wet Storage Ring)

2. **Special Nuclear Material Vault**

3. **Exponential Graphite Pile** (Storage and Operation Configurations)

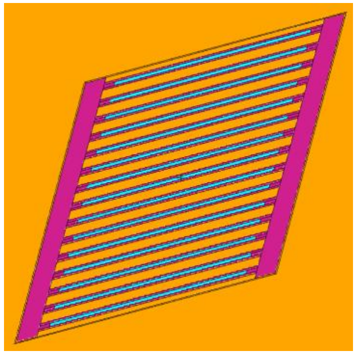
All cases shall satisfy the MITR technical specifications, i.e., k_{eff} shall be less than 0.90 (NRC limit is 0.95) with sufficient safety margins, by considering double contingency – typically over (or double) batching and light-water flooding.

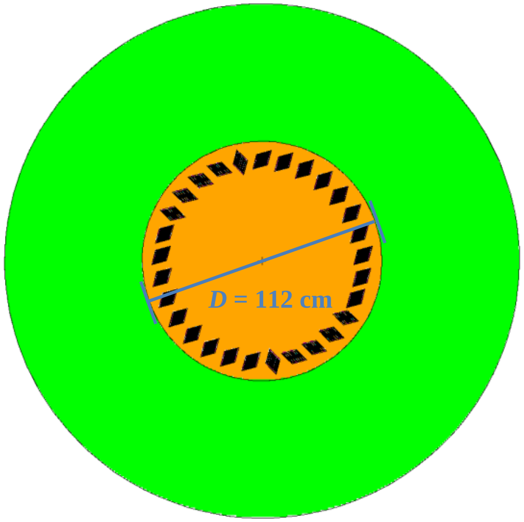
Wet Storage Systems

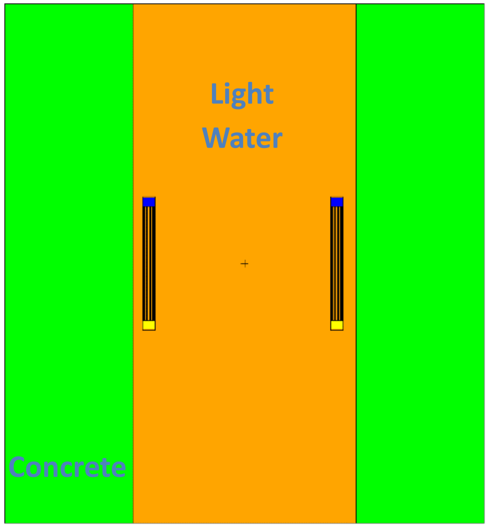


- NRC issued a Generic Letter, asking reactors to address degradation of neutron-absorbing materials in wet storage systems for reactor fuel
- We were trying to demonstrate our wet storage systems are able to maintain sub-criticality without any neutron-absorbing materials

Wet Storage Ring – Modeling







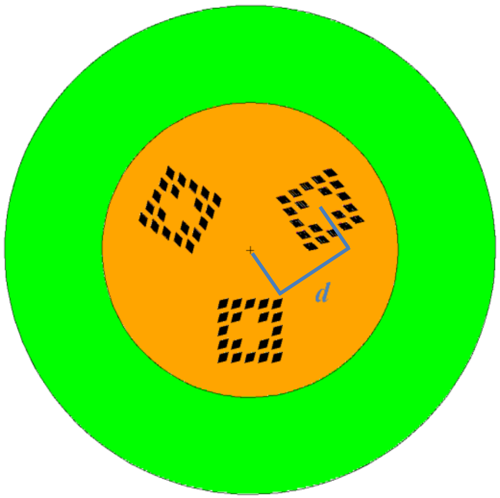
1) No neutron-absorbing materials (i.e., cadmium liners for the MITR case) are included in the MCNP model. This is a very conservative assumption, since it is highly unlikely that cadmium is degraded to zero level.

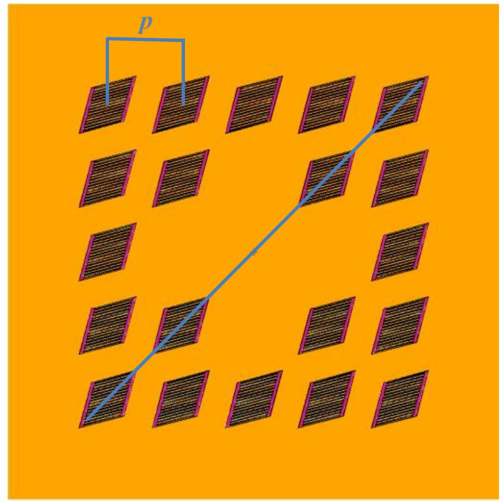
2) No structural components, such as depleted shim blades, metallic racks, storage containers, and etc., are taken into account. There is only full density (room temperature) light-water surrounding the fuel elements in the MCNP model. This is also a conservative assumption, since it will result in higher k_{eff}.

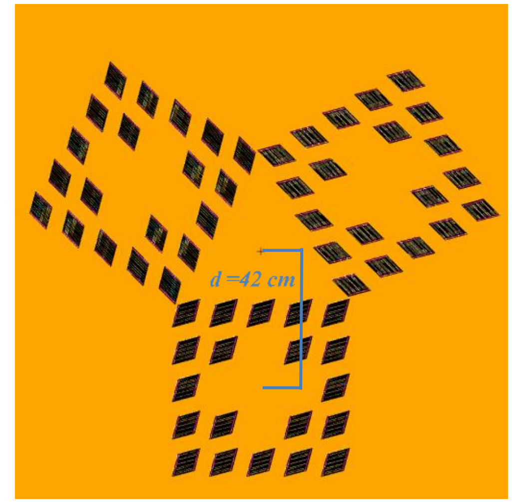
3) All fresh fuel elements are used in the calculations. Such an approach is again on the conservative side, since additional fissile materials are included.

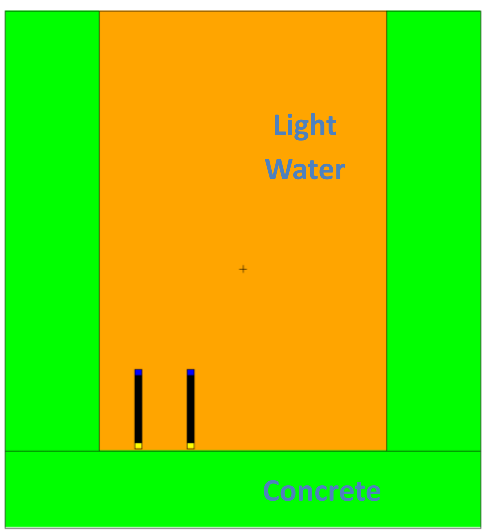
Results: 0.70496 $\pm$ 0.00060

Spent Fuel Pool – Modeling









Loading Configurations	Results
25 – Full Fuel Elements Loading	0.96533 ± 0.00057
24 – 1 Central Element Out	0.90794 ± 0.00057
23 – 1 Central + 1 Neighboring Elements Out	0.82360 ± 0.00057
21 – 1 Central + 3 Neighboring Elements Out	0.82267 ± 0.00062
21 – 0 Central + 4 Neighboring Elements Out	0.78881 ± 0.00057
20 – 1 Central + 4 Neighboring Elements Out	0.77633 ± 0.00057
13 – 12 Corner Elements (3 each) Out	0.87151 ± 0.00061
9 – Form a 3×3 Square	0.82541 ± 0.00068

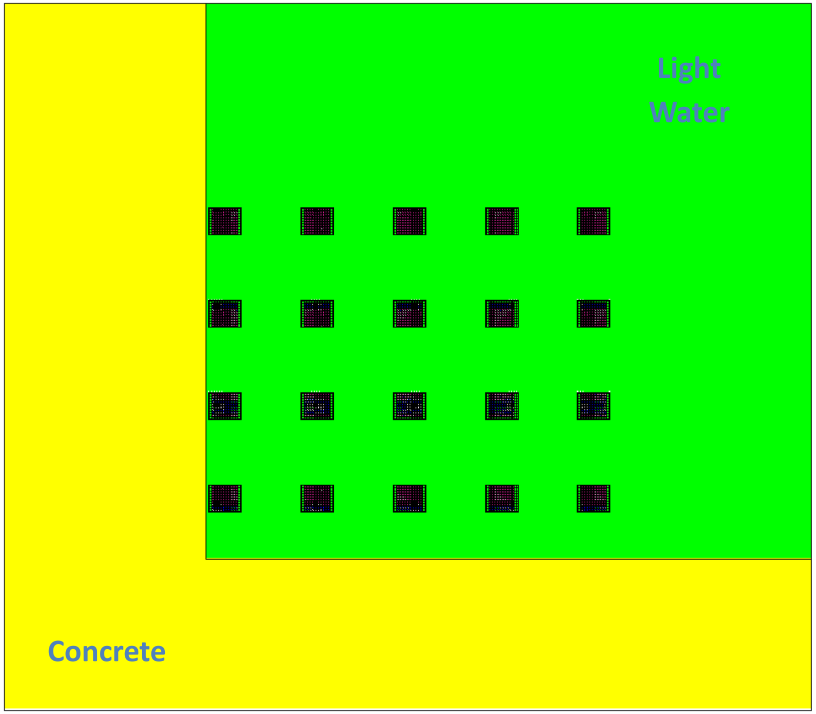
Pitch (p)	Results
11.0 cm	0.81219 ± 0.00070
11.5 cm (Ref)	0.77633 ± 0.00057
12.0 cm	0.74265 ± 0.00058

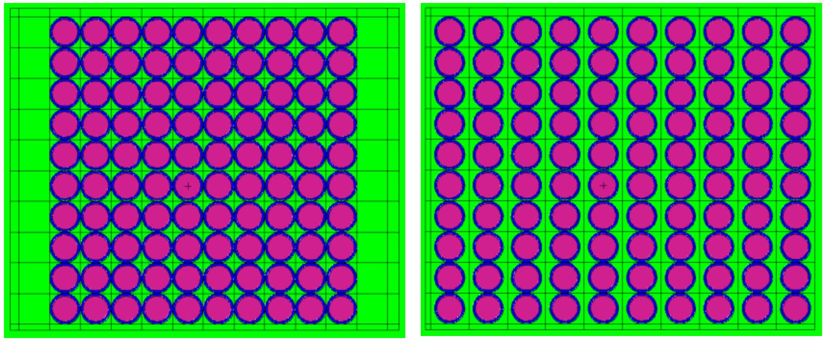
Distance (d)	Results
60.0 cm	0.77678 ± 0.00042
48.0 cm	0.77933 ± 0.00065
42.0 cm (Min)	0.81340 ± 0.00074

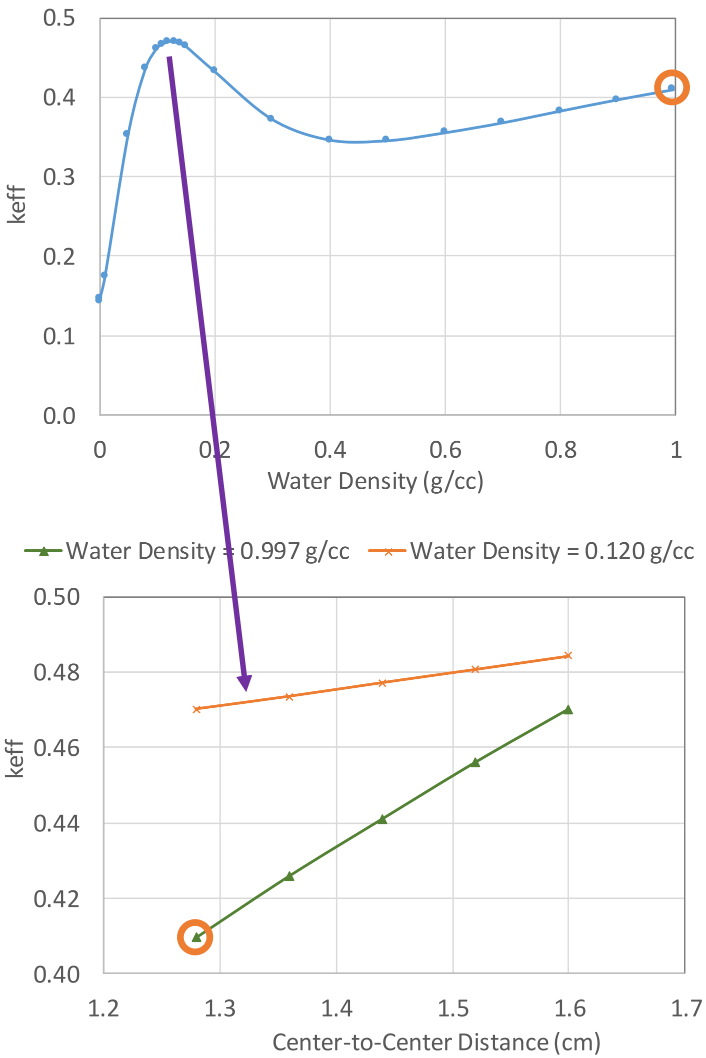
Special Nuclear Material Vault

- Special nuclear material inventory started to build-up since 1960s.
- No criticality safety analysis was required for the past several license renewals (every 10 years) until the most recent one in 2016.

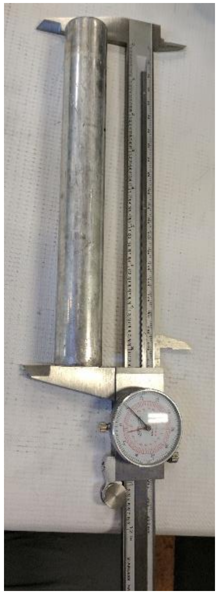
SNM Vault – Modeling

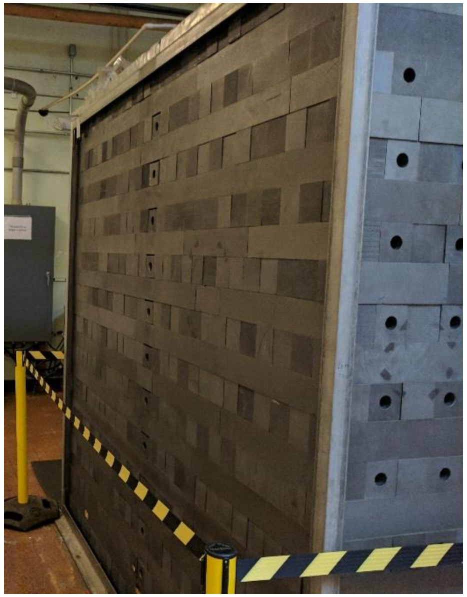


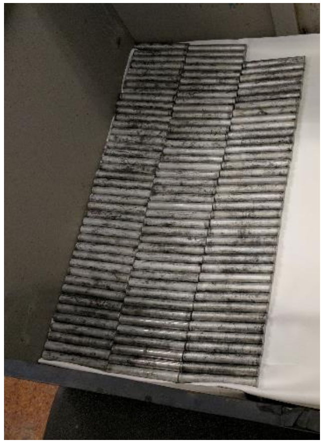




Exponential Graphite Pile

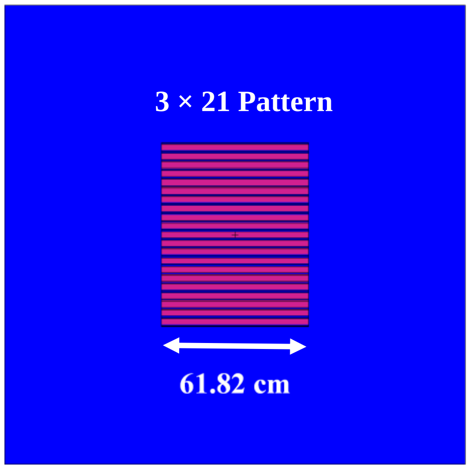


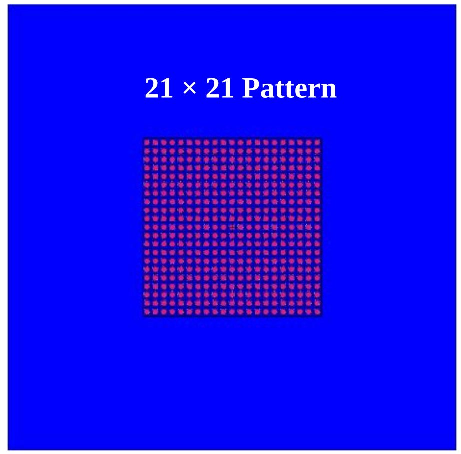


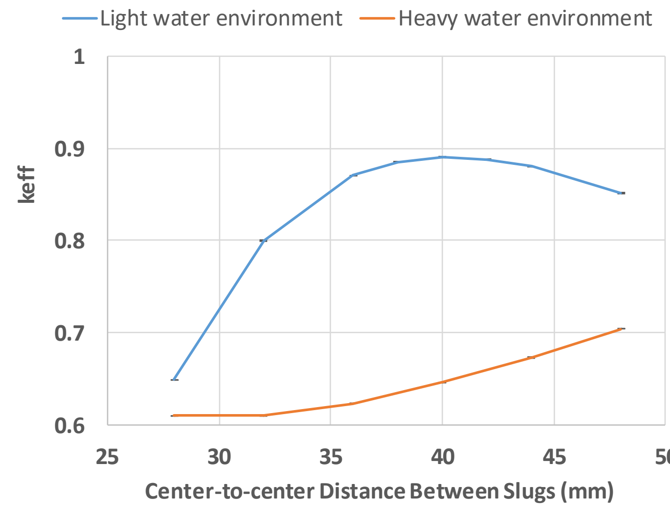


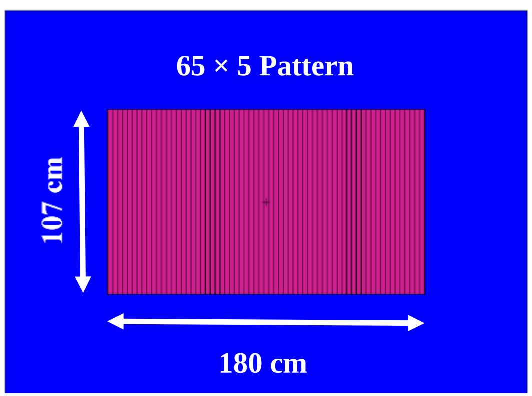
Fuel Slug Storage – Modeling

"Neutronically Optimal" Storage Configuration

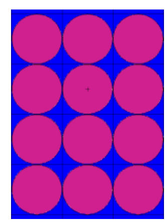


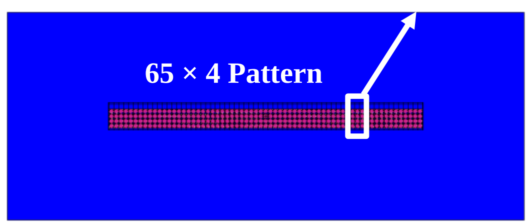






"Realistic" Storage Configuration





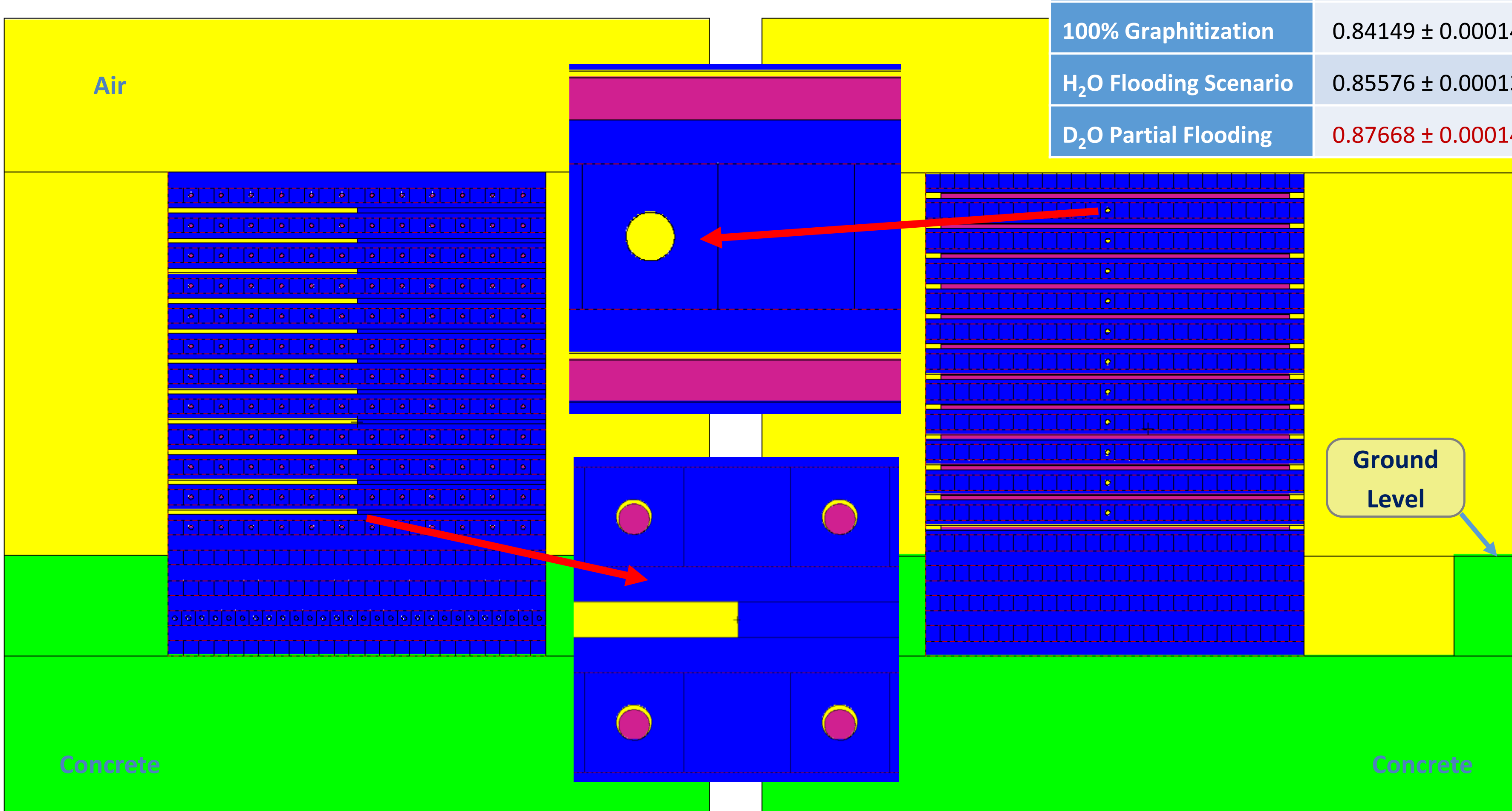
Results: 0.55541 ± 0.00030 (light-water)
0.68889 ± 0.00030 (heavy-water)

Graphite Pile – Modeling

	$k_{eff} \pm 1\sigma$
Reference Case	0.84821 ± 0.00014
100% Graphitization	0.84149 ± 0.00014
H_2O Flooding Scenario	0.85576 ± 0.00013
D_2O Partial Flooding	0.87668 ± 0.00014

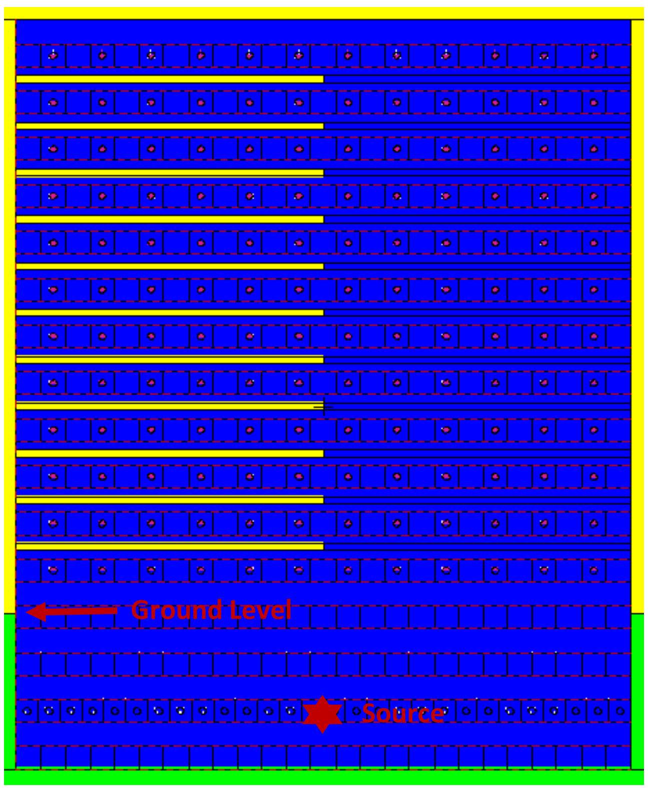
“Front Face” (Vertical Cross-section)

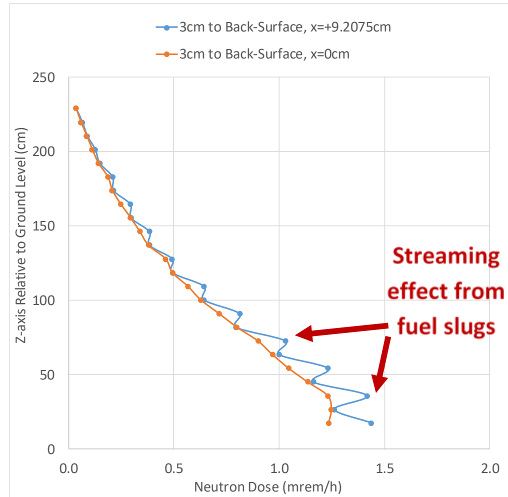
“Side Face”

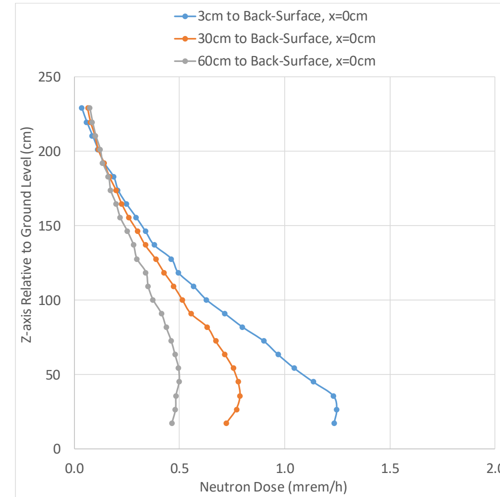


Neutron Doses – Pedestal Source

- A 10-curie Pu-Be source loaded at pedestal channel
- At 30 cm from pile surfaces, total radiation level < 1.0 mrem/h







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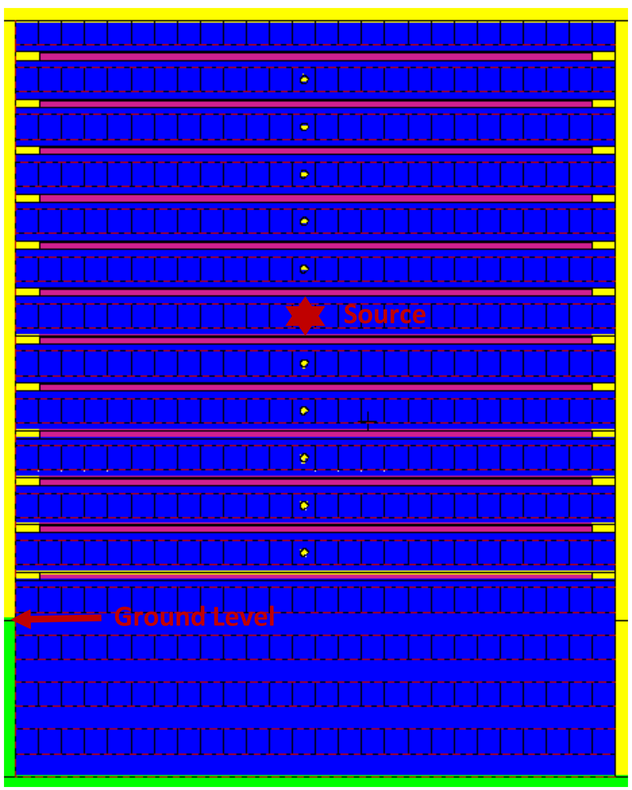
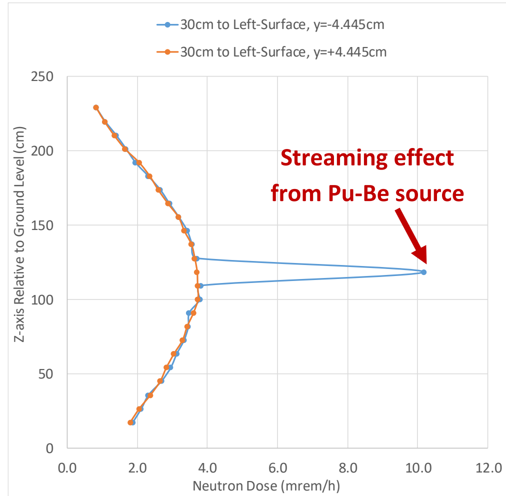
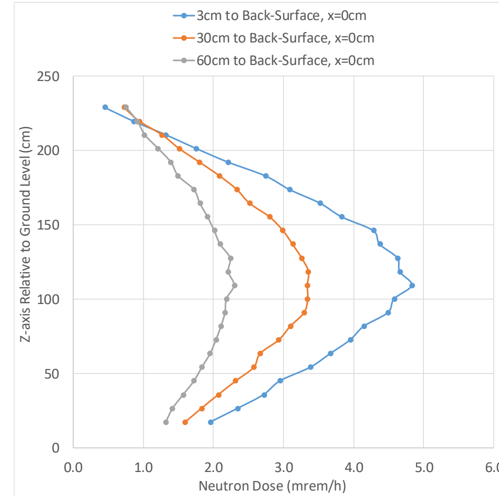
11																									
10		0.022		0.045		0.066		0.082		0.094		0.101		0.101		0.095		0.083		0.066		0.046		0.022	
9																									
8		0.047		0.099		0.145		0.182		0.208		0.223		0.223		0.210		0.184		0.147		0.101		0.048	
7																									
6		0.075		0.158		0.231		0.291		0.334		0.357		0.358		0.337		0.296		0.236		0.162		0.077	
5																									
4		0.107		0.225		0.330		0.416		0.479		0.512		0.514		0.484		0.424		0.338		0.232		0.111	
3																									
2		0.145		0.305		0.447		0.565		0.651		0.697		0.701		0.660		0.578		0.460		0.315		0.150	
1																									
0		0.189		0.399		0.588		0.745		0.861		0.925		0.930		0.876		0.766		0.608		0.415		0.197	
-1																									
-2		0.241		0.511		0.757		0.965		1.121		1.208		1.217		1.145		0.998		0.790		0.537		0.255	
-3																									
-4		0.302		0.643		0.960		1.234		1.445		1.566		1.582		1.485		1.288		1.013		0.684		0.323	
-5																									
-6		0.371		0.793		1.197		1.561		1.851		2.024		2.051		1.917		1.647		1.281		0.857		0.402	
-7																									
-8		0.441		0.954		1.465		1.954		2.367		2.628		2.671		2.477		2.093		1.597		1.051		0.488	
-9																									
-10		0.514		1.125		1.775		2.455		3.086		3.519		3.600		3.287		2.693		1.985		1.271		0.582	
-11																									
-12		0.629		1.395		2.295		3.365		4.491		5.348		5.524		4.907		3.818		2.658		1.629		0.737	

Neutron Doses – Central Source

- a 10-curie Pu-Be source loaded at graphite pile center
- At 30 cm from pile surfaces, total radiation level < 4.0 mrem/h







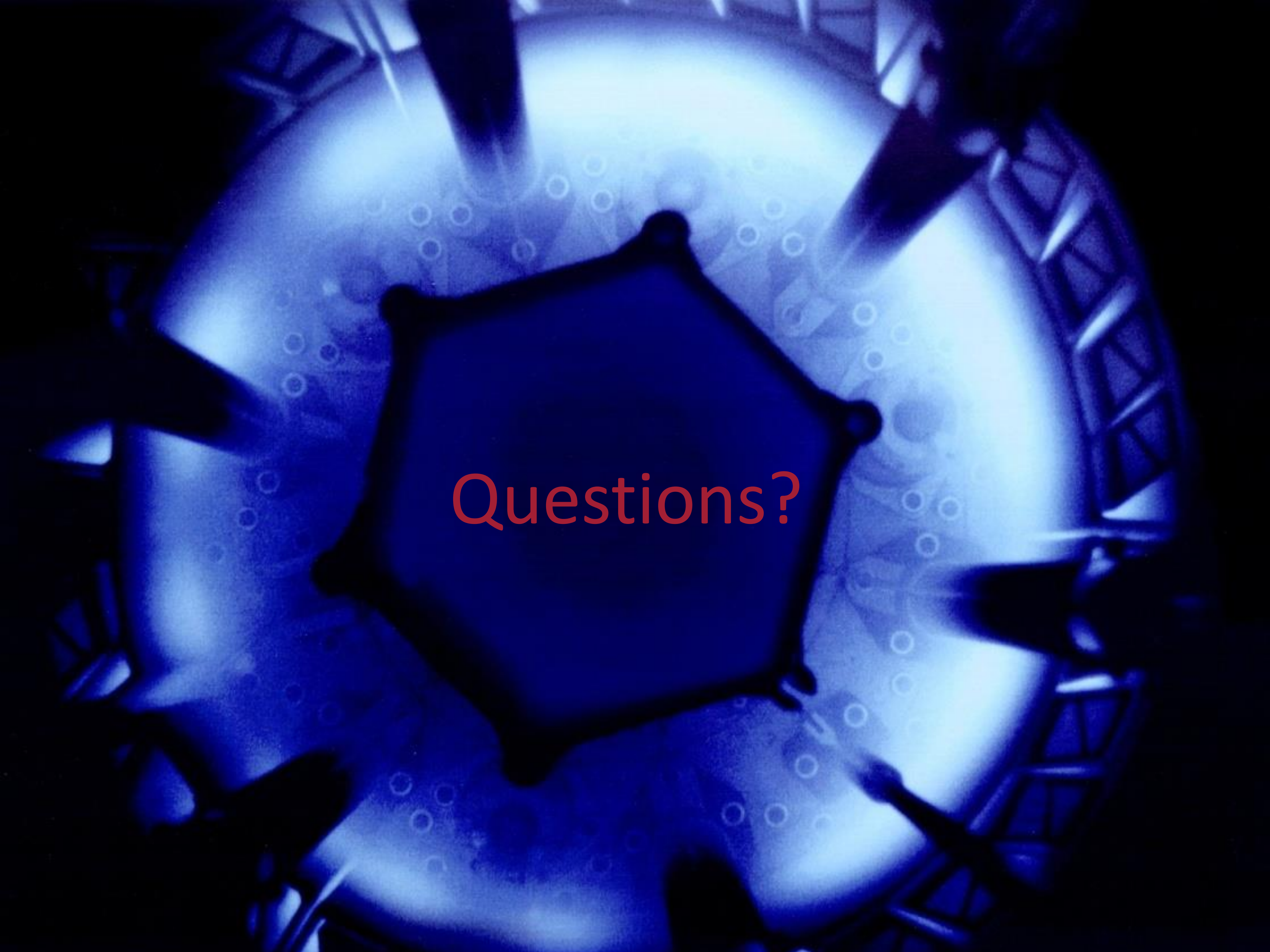
	-12	-11	-10	-9	-8	-7	-6	-5	-4	-3	-2	-1	0	1	2	3	4	5	6	7	8	9	10	11	12
11																									
10		0.064		0.137		0.208		0.271		0.321		0.348		0.347		0.318		0.267		0.204		0.134		0.062	
9																									
8		0.137		0.296		0.452		0.596		0.710		0.775		0.773		0.704		0.587		0.444		0.291		0.134	
7																									
6		0.208		0.454		0.700		0.939		1.141		1.258		1.254		1.130		0.925		0.688		0.444		0.204	
5																									
4		0.275		0.603		0.946		1.300		1.621		1.822		1.815		1.601		1.276		0.926		0.589		0.268	
3																									
2		0.328		0.728		1.162		1.638		2.107		2.424		2.413		2.078		1.603		1.133		0.709		0.320	
1																									
0		0.361		0.805		1.299		1.866		2.456		2.935		2.909		2.416		1.821		1.264		0.783		0.351	
-1																									
-2		0.368		0.817		1.317		1.888		2.481		2.959		2.932		2.441		1.843		1.282		0.795		0.357	
-3																									
-4		0.347		0.766		1.217		1.707		2.183		2.504		2.493		2.153		1.671		1.187		0.746		0.338	
-5																									
-6		0.307		0.670		1.042		1.419		1.756		1.964		1.956		1.736		1.395		1.020		0.654		0.300	
-7																									
-8		0.256		0.552		0.843		1.117		1.344		1.475		1.471		1.332		1.102		0.829		0.542		0.251	
-9																									
-10		0.205		0.436		0.655		0.850		1.003		1.087		1.085		0.996		0.841		0.646		0.429		0.202	
-11																									
-12		0.168		0.349		0.518		0.662		0.770		0.829		0.827		0.766		0.656		0.511		0.344		0.165	

Summary and Discussion

Summary: Several criticality safety analyses for MITR facilities have been presented. All cases satisfy MITR technical specifications, i.e., k_{eff} less than 0.90 (NRC limit is 0.95) with sufficient safety margins, by considering double contingency.

1. **Technical:** There is a clear trend that NRC pushed to implement neutronics validations for the calculation results, where newer versions of ANSI/ANS Standards (Series 8) is particularly requested to be followed. *How other Research Reactors accommodate this request?*

2. **Administrative:** At least at MITR, there is no specific/clear funding source supporting criticality safety analysis and validation report. *How other Research Reactors solve the financial issue?*


Questions?