核反应堆物理计算的现状和展望

—讲述一个计算反应堆物理分析师自己的故事

吴泽云 博士

NIST Center for Neutron Research, 100 Bureau Drive, Gaithersburg, MD

Department of Mat. Sci. and Eng., University of Maryland, College Park, MD

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Outline

- Career Development so far
- Retrospect on R&D in Computational Reactor Physics
- Reactor Analysis Examples
  - Core Design for a Small Modular BWR
  - Transient Safety Analysis for a Sodium Fast Reactor
  - Feasibility Study for the NIST New Research Reactor
- Summary and Moving Forward
- Q & A

B.S. @ Tsinghua University

- Beijing, China
- Engineering Physics
- *Alma mater* Forever
- Mathematics & Physics
- Engineering & Technology

****Get Out with Nothing about Reactor Physics****

Ph.D. @ Texas A&M University

Get Out with Something about Reactor Physics

Post-doc at NC State (1st Real Job)

H. S. Abdel-Khalik

CASL

****Get Out with Fancy about Reactor Physics****

Post-doc at Purdue (2nd Real Job)

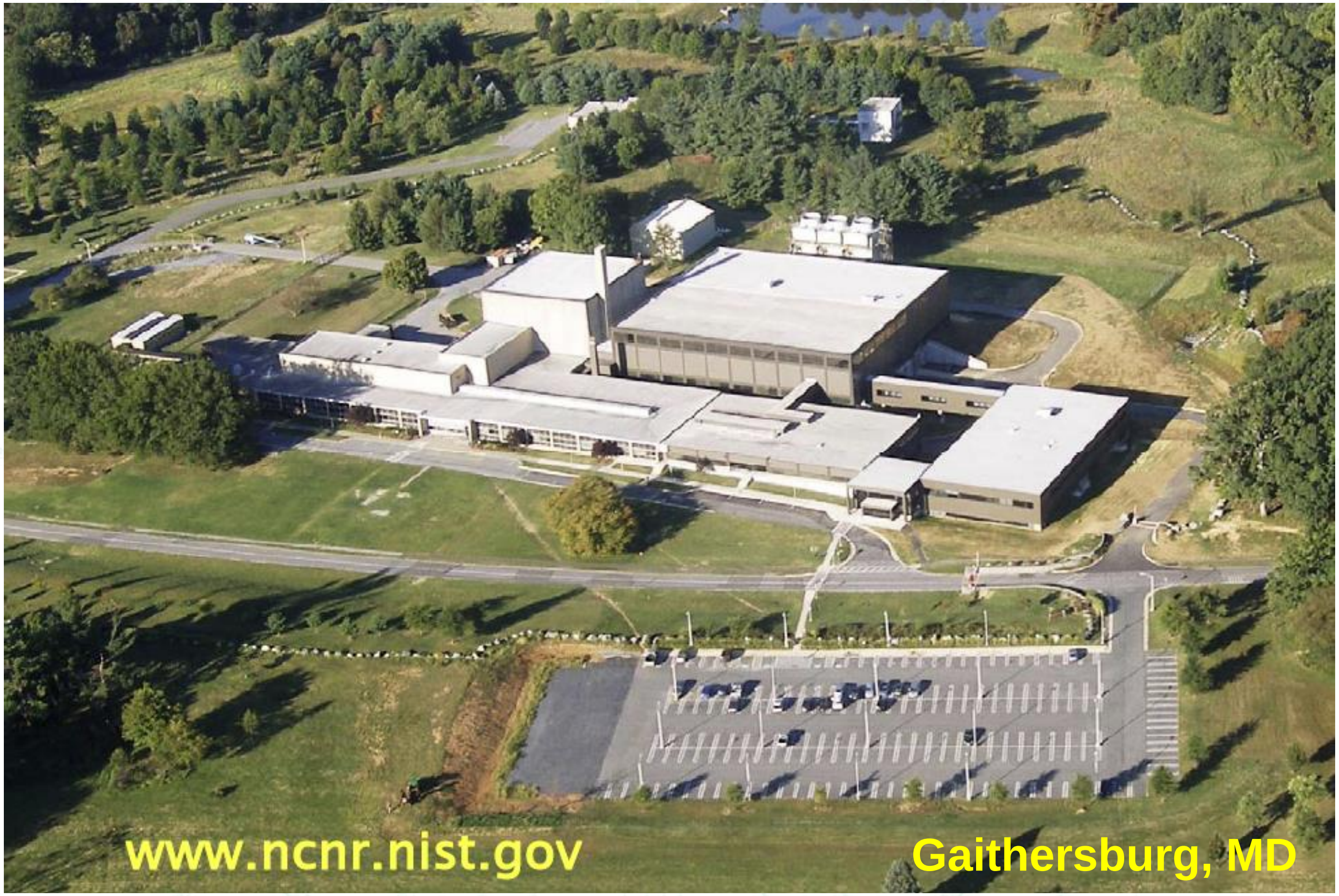
W. S. Yang

Argonne NATIONAL LABORATORY

- Fast Reactor Physics
- Statistics and Dynamics
- Resonance Self Shielding
- Reactivity Feedback
- BWR & FR
- CASMO/PARCS/Relap5
- MC^2-3/DIF3D/Rebus-3

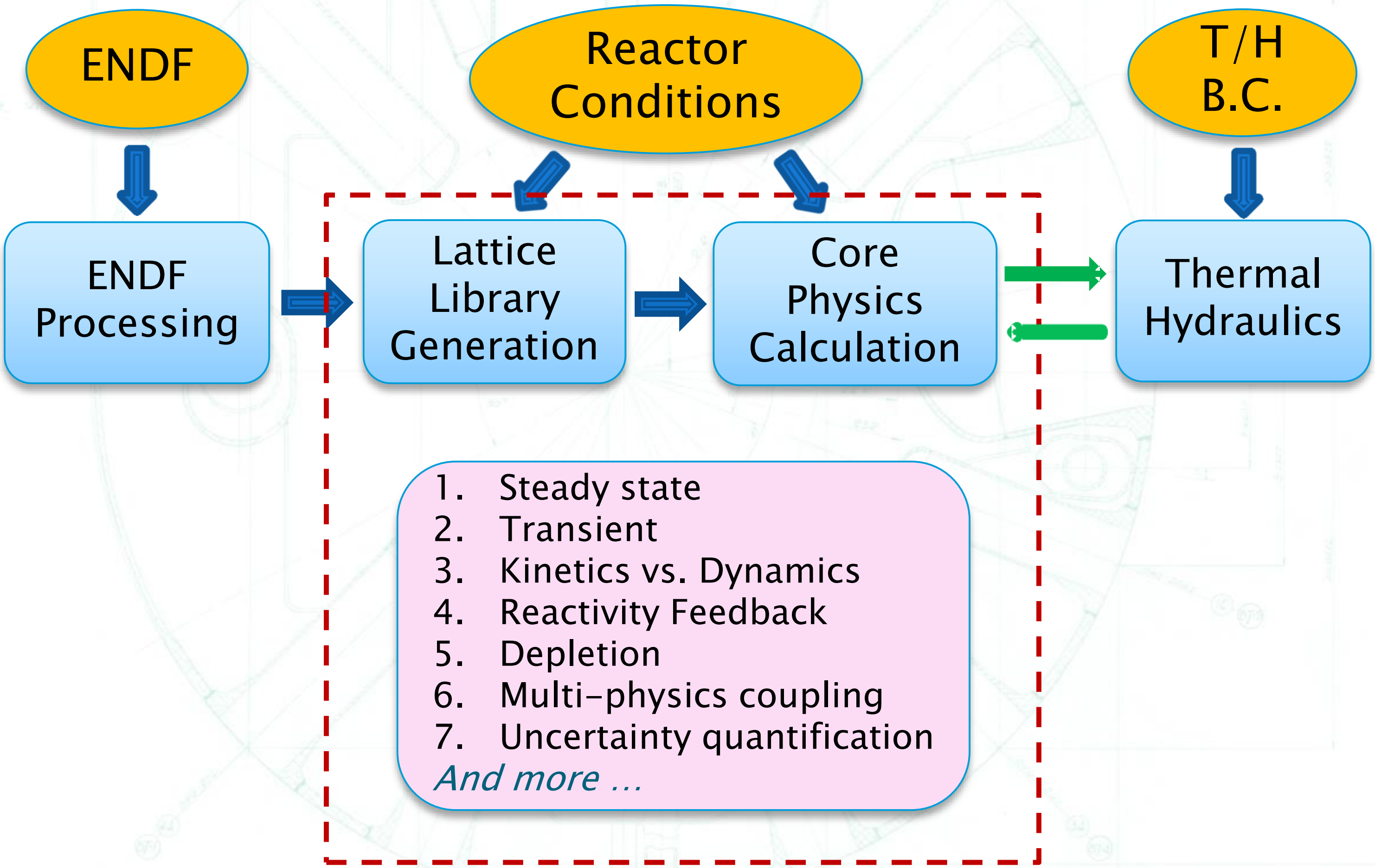
Get Out with Sophistication about Reactor Physics

Practice Reactor Physics at NIST (Current Job)

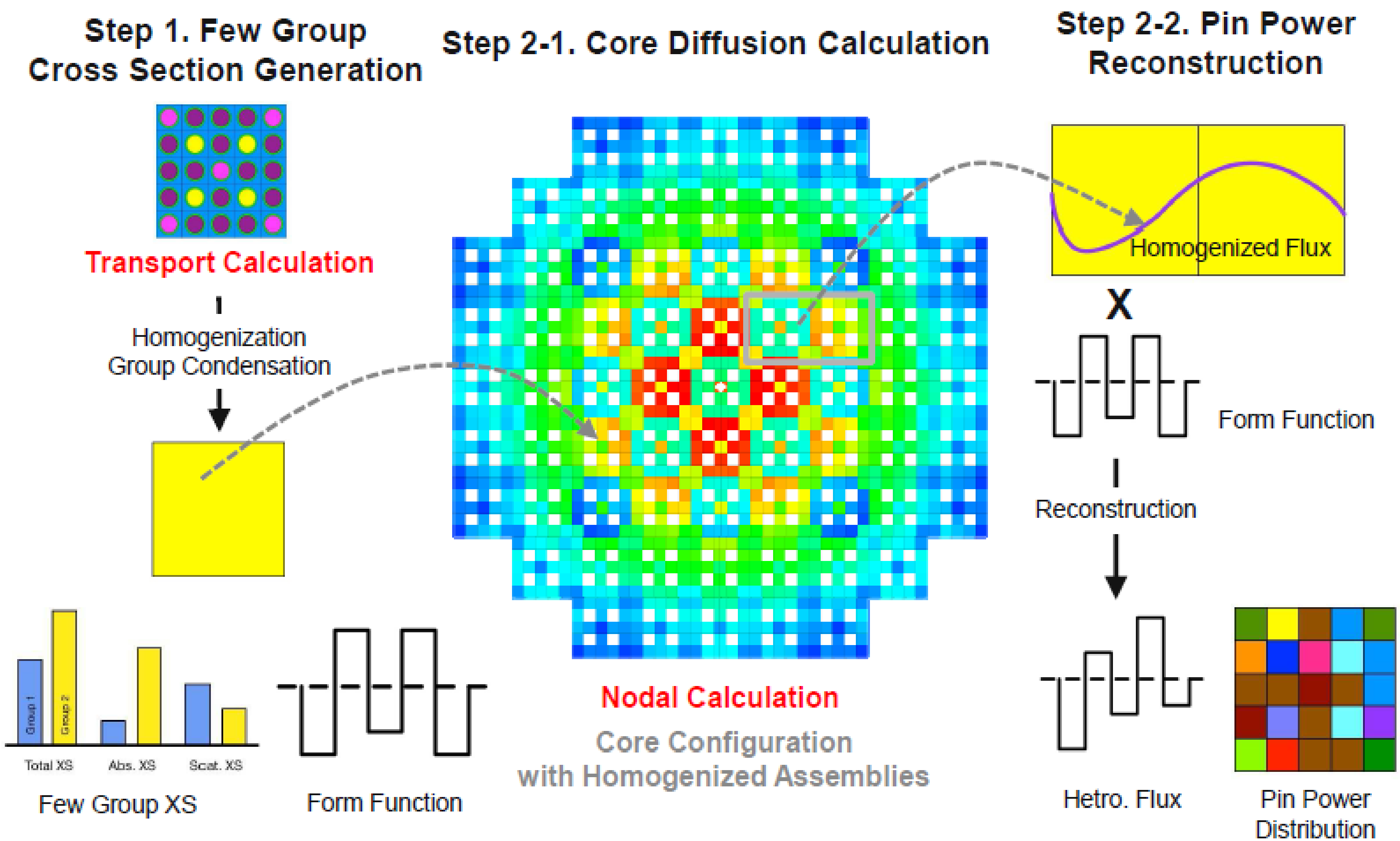


****Equipped with Realities about Reactor Physics****

Main Modules in Reactor Physics Calculation



Two-Step Reactor Physics Calculation (LWR)



(Courtesy of Dr. W. S. Yang's Reactor Physics Lectures)

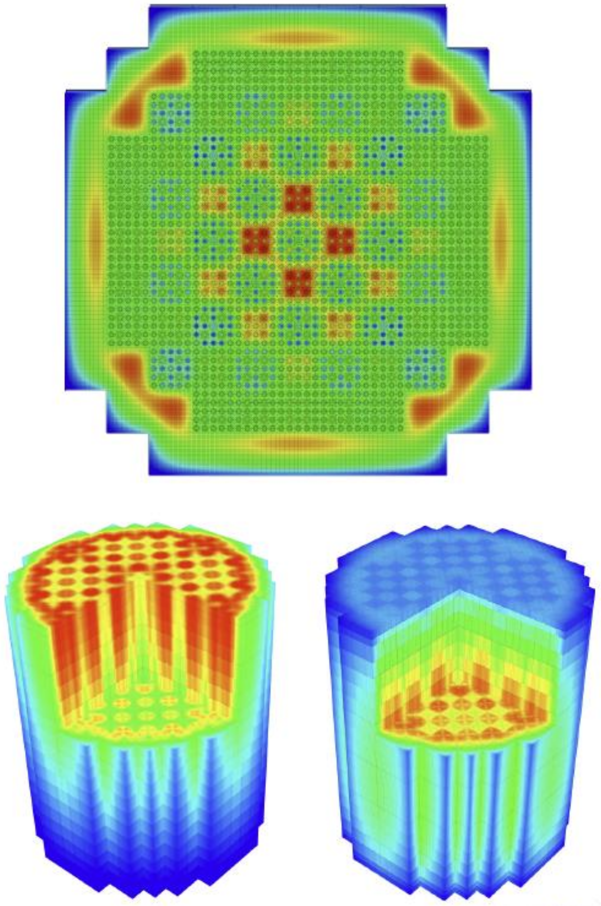
Reactor Physics Code Suites based on the Two-Step Approach

Lattice/Core Physics Code	Laboratory	Country
CASMO (HELIOS)/SIMULATE	Studsvik	U.S.
CASMO/PARCS	Purdue	U.S.
MC2/DIF3D+VARIANT	ANL	U.S.
WIMS/CITATION	ORNL	U.S.
ALPHA/PHOENIX-P (PARAGON)/ANC-9 (SP-NOVA)	Westinghouse	U.S.
LANCER (TGBLA)/PANCEA	GNF/GE	U.S.
DRAGON-4.0/DONJON5	EPM	Canada
WIMS-AECL/RFSP	AECL	Canada
APPOLO-2.5/CRONOS2	CEA	France
CASMO/MICROBURN-B/P	AREVA	France
WIMS/PANTHER	BNFL/BE	UK
MOSRA-SRAC/MOSRA-LIGHT	JAEA	Japan

Direct Whole Core Transport Calculation

Required Improvements on M&S:

- Detailed geometric and compositional representation for all assemblies in core
- Sufficient polar angle representation to see axial differences in fuel rods
- Anisotropic scattering expansion of at least P_2
- Fine energy resolution at least 30 groups
- Multi-dimension OTF resonance self-shielding
- Intra-pellet spatial resolution of Doppler effect
- Localize T-H code to treat various feedback
- Sub-pin level depletion (track large number of isotopes ~300 for each fuel pin
- Short and middle range transient calculation (efficient time integration for transient analyses)
- Big, cheap, robust computer and disk storage farms



(Courtesy of Dr. Kord Smith's Reactor Physics Presentation)

Reactor Codes based on Whole Core Transport

Reactor Code	Laboratory	Country
Attila	LANL	U.S.
DeCart	KAERI	Korea
nTRACER	SNU	Korea
MPACT	U-Mich	U.S.
APPOLO-3	CEA	France
DRAGON-5	EPM	Canada

Reactor Physics Codes based on Monte Carlo

Monte Carlo Code	Laboratory	Country
MCNP-6	LANL	U.S.
SCALE 6.0	ORNL	U.S.
MC21	KAPL/BAPL	U.S.
MVP/GMVP	JAEA	Japan
MCU-6	KI	Russia
TRIPOLI-4	CEA	France
MORET	IRSN	France
McCARD	SNU	Korea
MONK	ANSWERS	U.K.
SERPENT	VTT	Finland
OpenMC	MIT	U.S.
RMC	Tsinghua	China

Reactor Analysis Example #1:

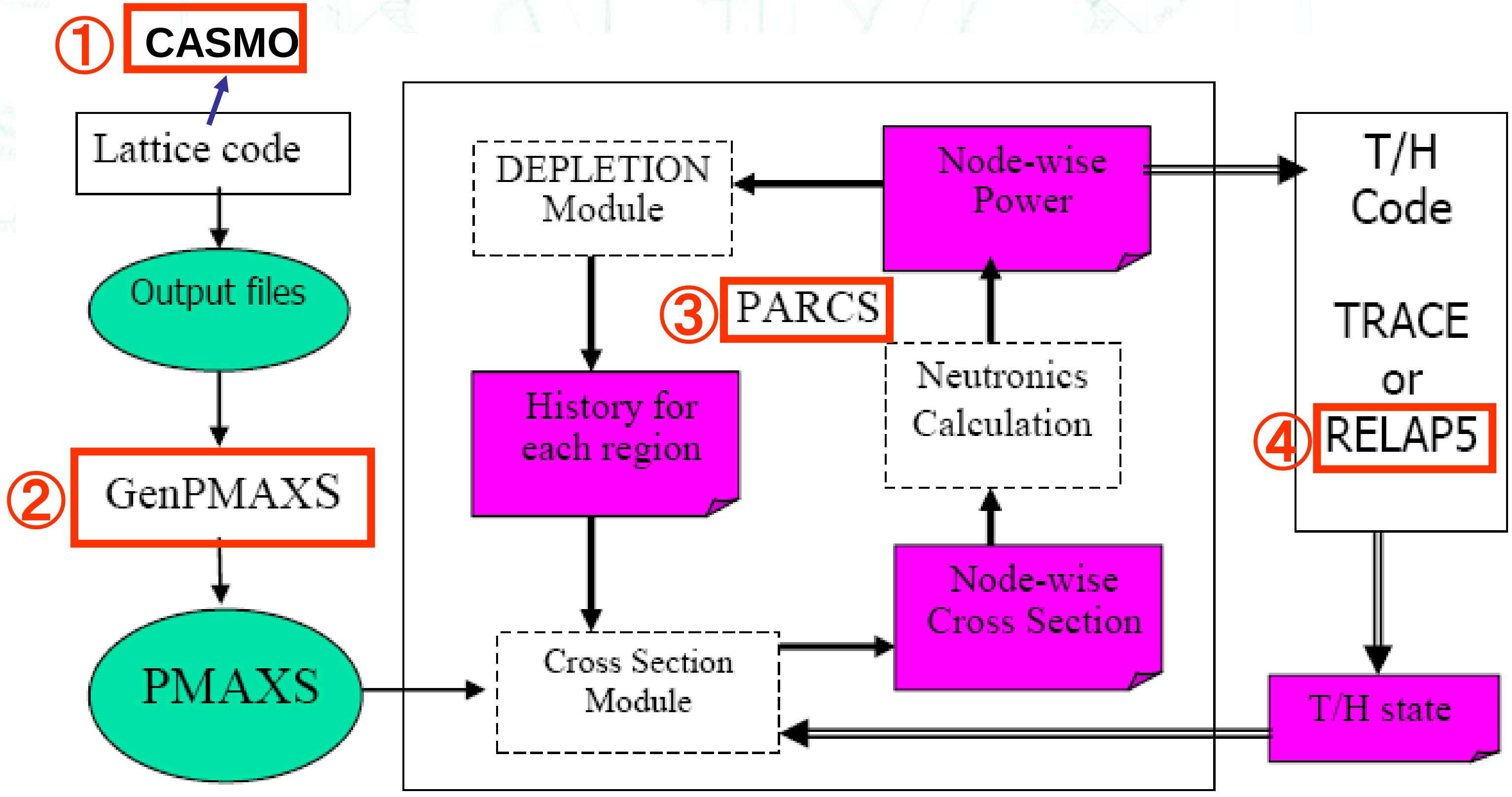
Core Design Studies for a 50-Mwe BWR-Based SMR with Long-Life Core (NMR-50)

Small Modular Reactors (SMR)

- The size of the reactor unit is “small”
- Reactors can be deployed modularly

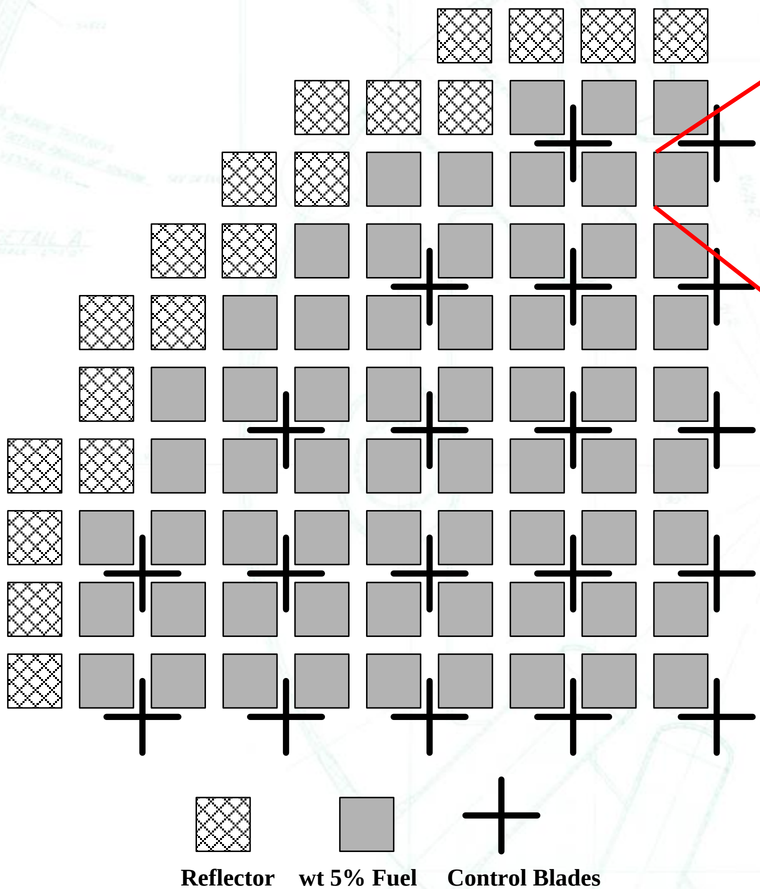
Name	Vendor	Power (MWe)	Cycle Length (years)	Fuel ^{235}U (wt.%)	Type
IRIS	WESC	335	2.5 – 4	4.95	PWR
mPower	B&W	180	4	5.00	PWR
NuScale	NuScale	45	2	< 4.95	PWR
HPM	LANL	25	10	19.75	LMFR
NMR-50	Purdue	50	~ 10	5.00	BWR

Neutronics Analysis Code Suite for NMR-50

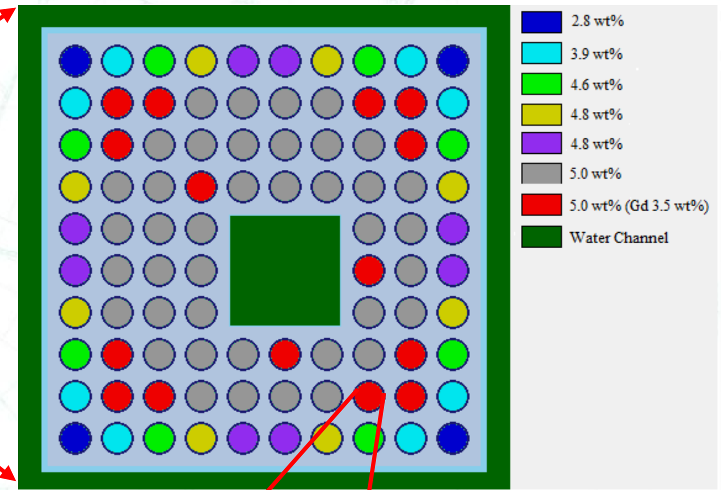


Ref. Y. Xu and T. Downar, “GenPMAXS-V6: Code for Generating the PARCS Cross Section Interface File PMAXS”, GenPMAXS manual, University of Michigan, March (2012)

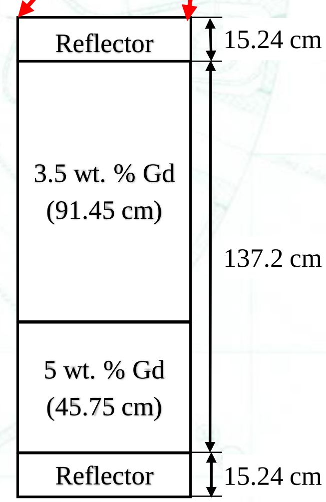
Single Assembly Core Design for NMR-50



Radial view of the quarter core



Schematics of the fuel assembly



Axial zoning of the Gd fuel rod.

Neutronics Results for NMR-50 at BOC

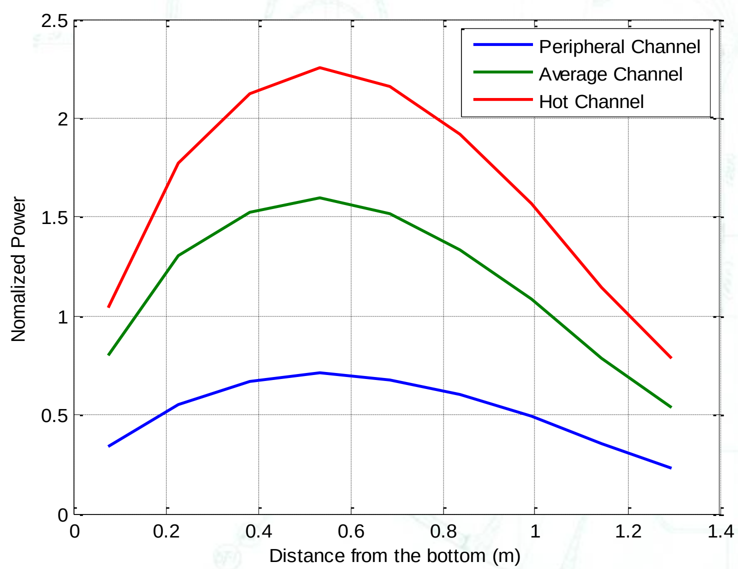
Initial CR Positions

			0	0
		0	0	0
	0	0	0	2192
0	0	0	2192	2392
0	0	2192	2392	2392

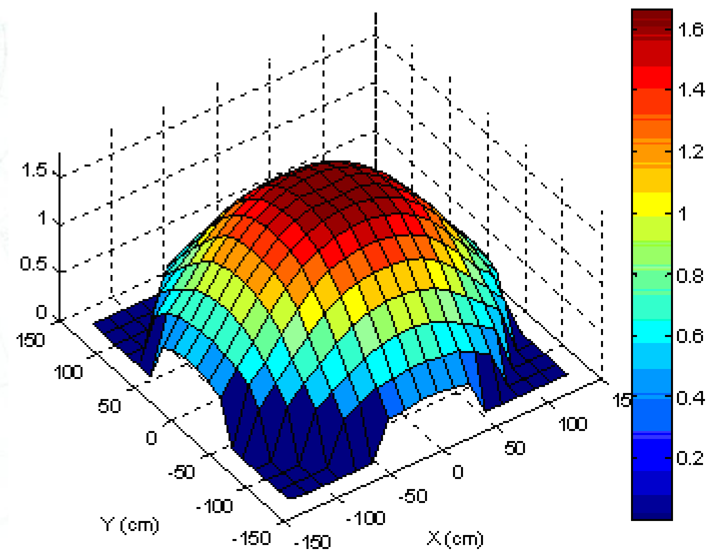
Final CR Positions

			0	0
		0	0	0
	0	0	0	229
0	0	0	229	249
0	0	229	249	270

Fig. Control rod insertion positions for criticality search at BOC. The notch value of a fully inserted control rod is 3192.



Axial power distribution for different flow channel



Radial power distribution

Steady State T/H Safety Performance at BOC

Property	SBWR-600[1]	NMR-50
Average LPD (kW/m)	16.60	5.16
Total peaking factor	2.73	2.98
MFLPD (kW/m)	45.30	15.36
MCPR (minimum)	1.32	2.25

[1]. Simplified Boiling Water Reactor Standard Safety Analysis Report (SSAR)," General Electric, 25A5113 Rev. A, August, 1992.

Core Performance of NMR-50 in 10 Years Fuel Cycle

Burn time (years)	Avg. Burnup (GWd/T)	k_{eff}	Control blade notch[a]	MFLPD (kW/m)	MCPR
0.00	0.00	0.99988	1455	15.36	2.25
1.00	3.06	1.00560	14394	17.78	2.55
2.00	6.12	1.00135	28101	17.61	2.36
3.00	9.18	1.00062	40818	18.66	2.17
4.00	12.24	1.00005	38856	13.13	2.29
5.00	15.31	1.00010	34602	12.48	2.47
6.00	18.37	1.00009	27262	12.92	2.07
7.00	21.43	1.00009	23346	11.97	2.34
8.00	24.49	1.00010	19139	12.39	2.57
9.00	27.55	1.00011	14490	14.06	2.84
9.99	30.61	1.00010	7963	15.80	2.79

[a]The notch value is the sum of notches for all inserted control blades.

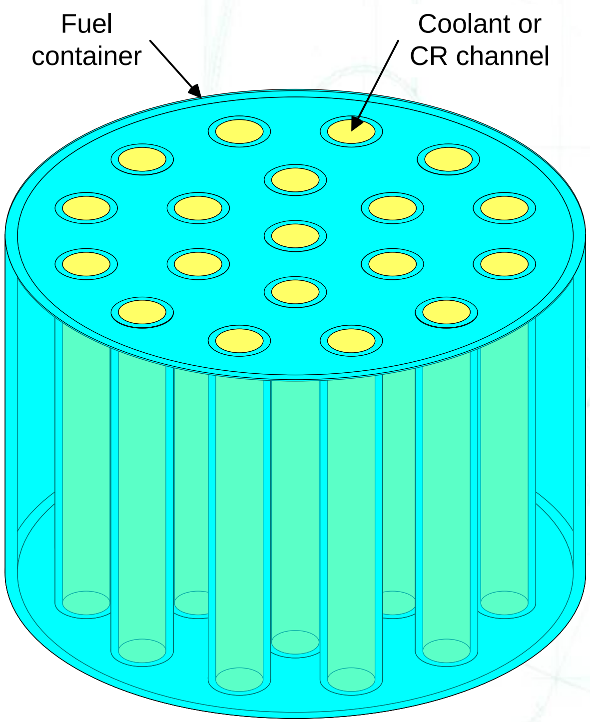
Reactor Analysis Example #2:

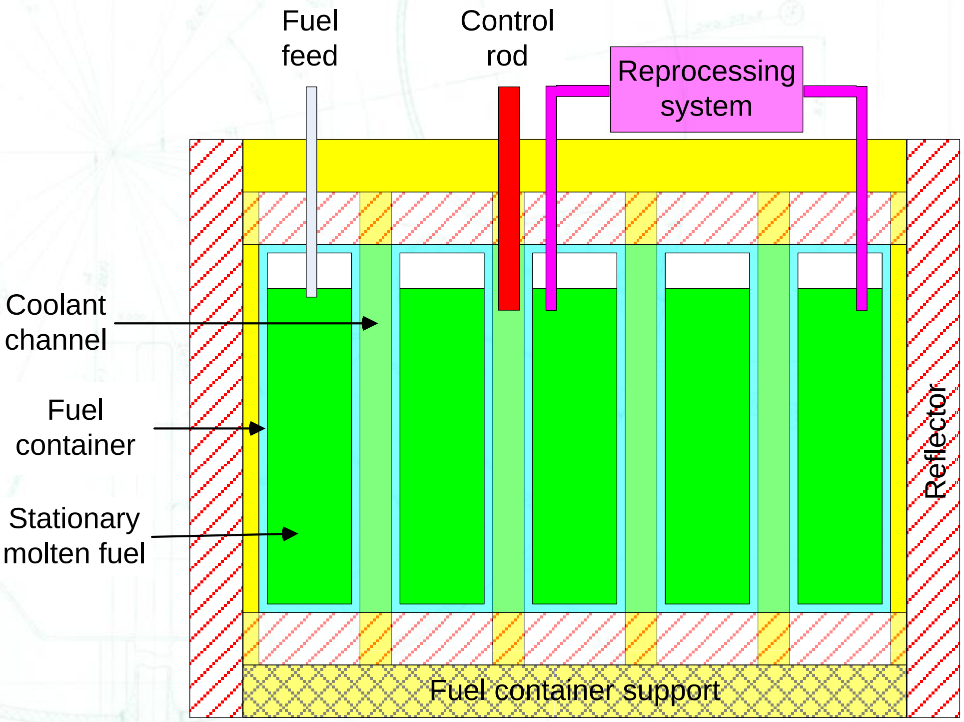
Transient Safety Analysis for a Stationary Liquid Fuel Fast Reactor (SLFFR)

Stationary Liquid Fuel Fast Reactor (SLFFR) Concept

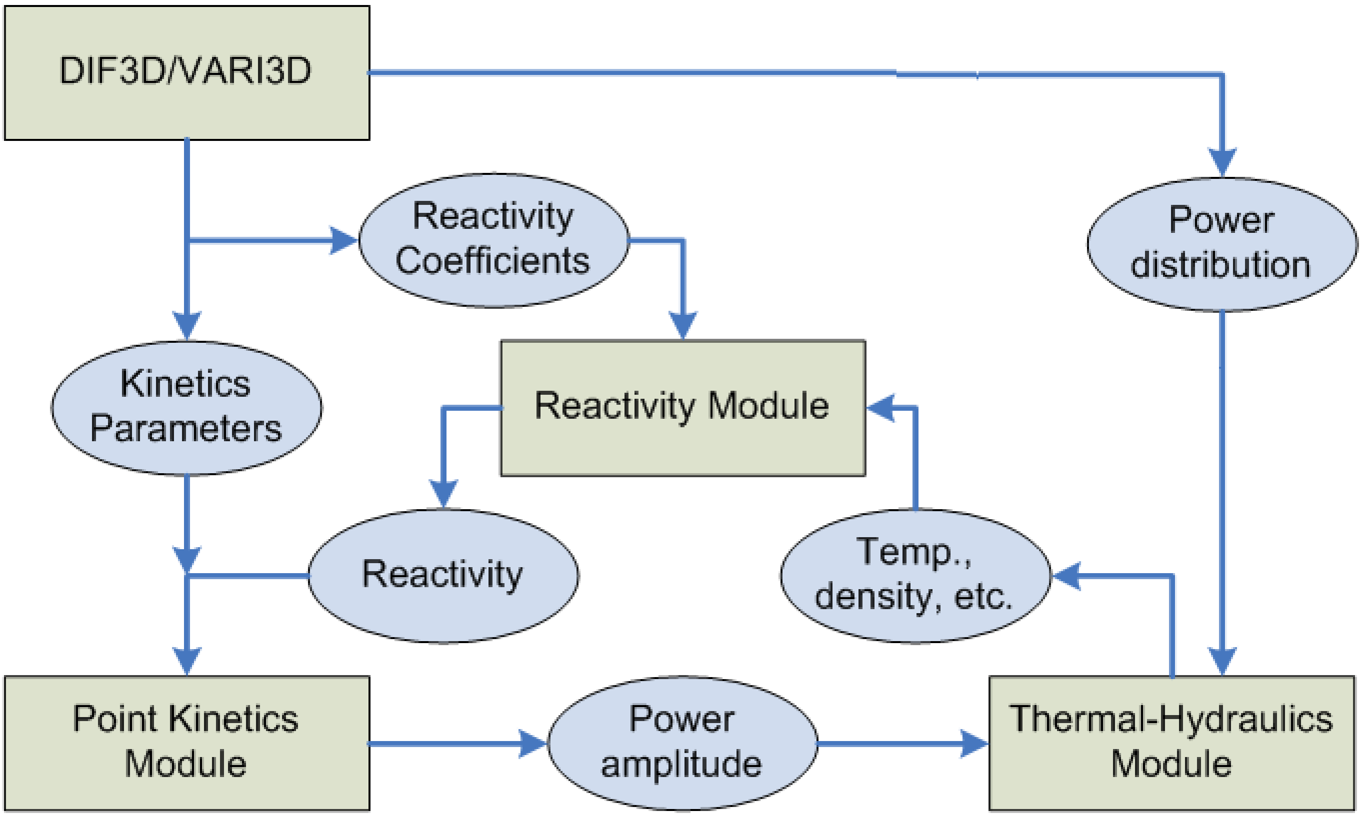
SLFFR is a new type of fast reactor system based on stationary (non-flowing) molten metallic fuel and a co-located reprocessing system

- Eutectic TRU alloy fuel is contained in a thick container
- Liquid metal coolant such as sodium, lead, LBE flows through flow channels penetrating the fuel container





Computational Modules for SLFFR Safety Analysis



Mathematical Models Associated with the Modules.

1. Single Channel Thermal-Hydraulics Model:

$$\frac{\partial}{\partial t}\rho(t,z)+\frac{\partial}{\partial z}[\rho(t,z)v(t,z)]=0$$

$$\rho(t,z)c_p\left[\frac{\partial}{\partial t}T_c(t,z)+v(t,z)\frac{\partial}{\partial z}T_c(t,z)\right]=\frac{P_h}{A}q''(t,z)$$

$$\frac{\partial}{\partial t}[\rho(t,z)v(t,z)]+\frac{\partial}{\partial z}[\rho(t,z)v^2(t,z)]=-\frac{\partial}{\partial z}P(t,z)-\rho(t,z)g-\frac{f\rho(t,z)|v(t,z)|v(t,z)}{2D_h}$$

2. Heat Conduction Model:

$$\rho_f c_{p,f}\frac{\partial}{\partial t}T_f(r,z,t)=q'''(r,z,t)+\frac{1}{r}\frac{\partial}{\partial r}\left[rk_f\frac{\partial T_f(r,z,t)}{\partial r}\right]$$

$$\rho_w c_{p,w}\frac{\partial}{\partial t}T_w(r,z,t)=\frac{1}{r}\frac{\partial}{\partial r}\left[rk_w\frac{\partial T_w(r,z,t)}{\partial r}\right]$$

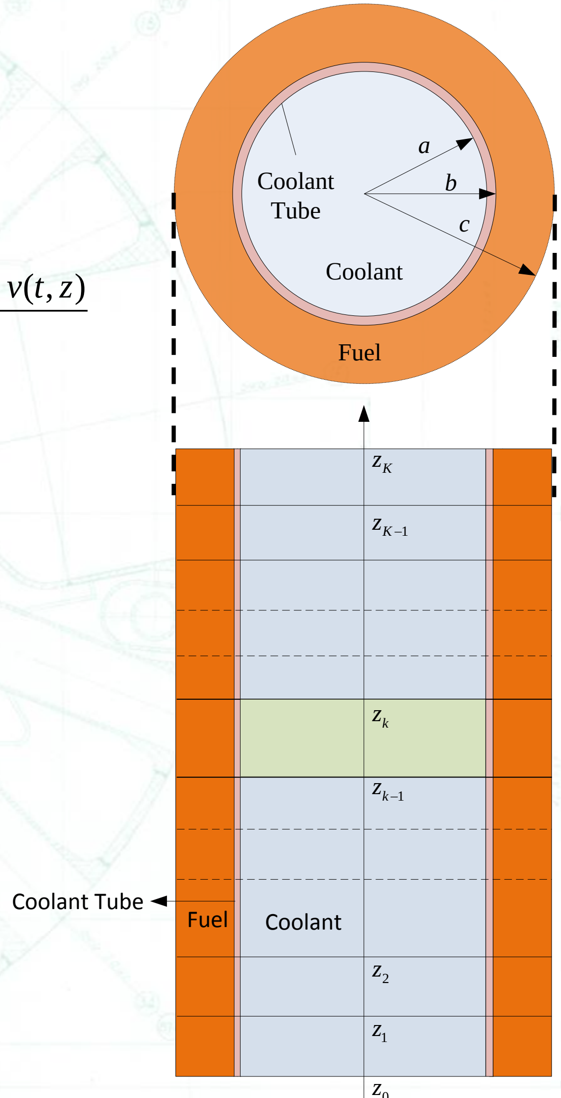
3. Point Kinetics Model:

$$\frac{dp(t)}{dt}=\frac{\rho(t)-\beta}{\Lambda}p(t)+\frac{1}{\Lambda}\sum_{i=1}^{6}\lambda_i\zeta_i(t)$$

$$\frac{d\zeta_i(t)}{dt}=\beta_i p(t)-\lambda_i\zeta_i(t),\qquad (i=1,\cdots,6)$$

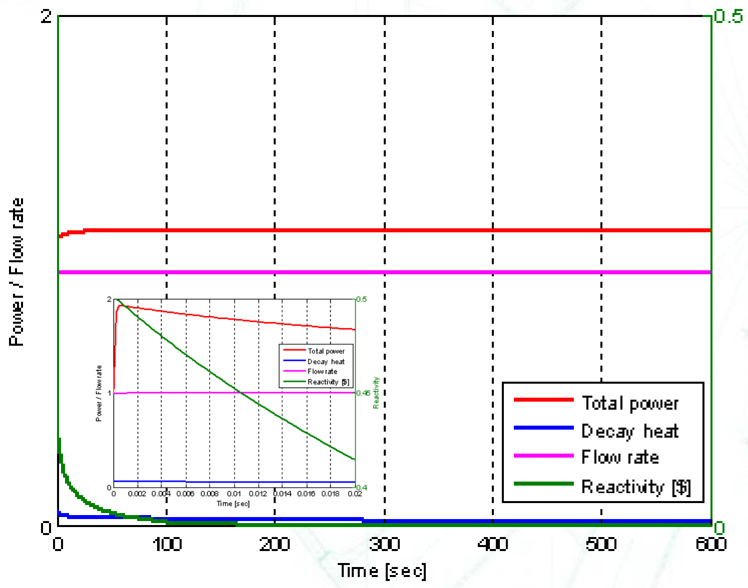
4. Reactivity Feedback Model:

$$\rho(t)=\rho_{ext}(t)+\delta\rho_D(t)+\delta\rho_{ax}(t)+\delta\rho_{re}(t)+\delta\rho_{Na}(t)$$

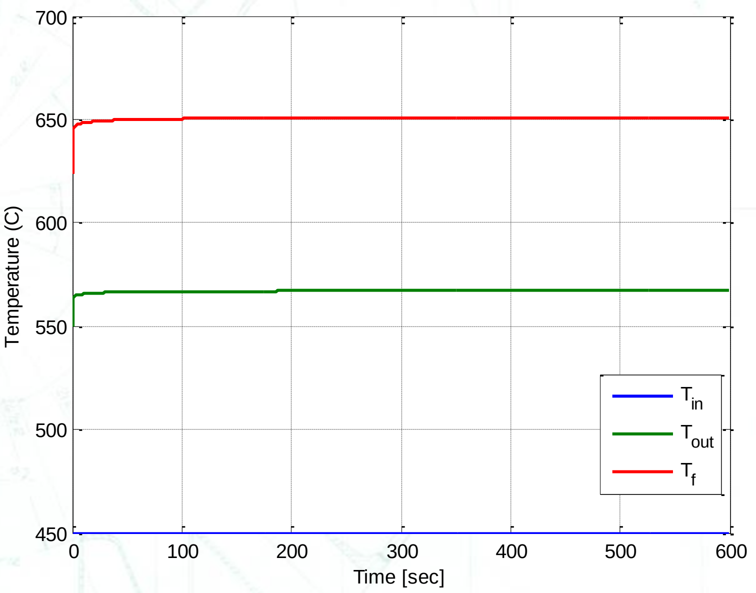


Unprotected Transient Overpower (UTOP) Accident

For a **UTOP** accident, it was assumed that a control rod runs out and introduces a positive step reactivity of 0.5$ while the flow and inlet temperature remain fixed, and that the reactor fails to scram.



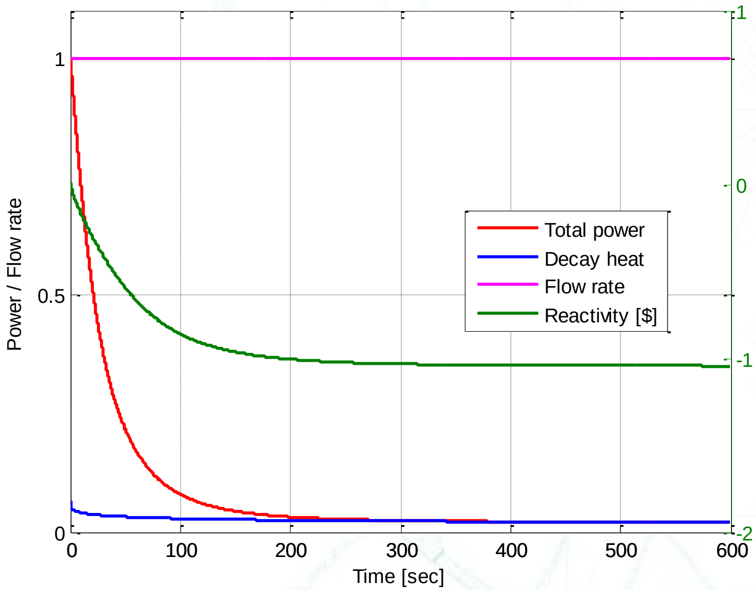
Power and reactivity transients in UTOP.



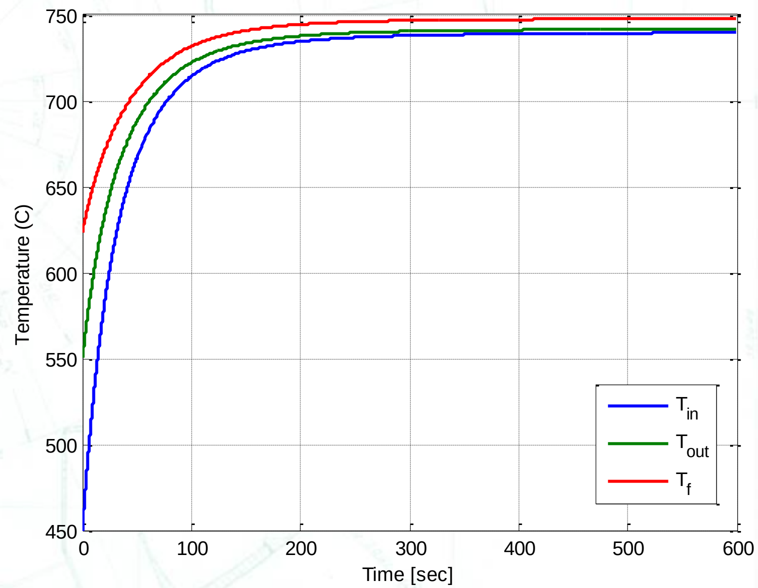
Fuel and coolant temperature transient in UTOP.

Unprotected Loss of Heat Sink (ULOHS) Accident

For a **ULOHS** accident, the steam turbine is tripped and isolated, resulting in the loss of heat ejection capability at the steam generator, and the reactor fails to scram. However, the decay heat removal system is assumed to be operational with 0.5-1% full power capability. The result of this accident is an increasing inlet temperature of the sodium coolant. With all systems continuing to operate, the coolant outlet temperature from the core also starts to rise.



Power and reactivity transients in ULOHS.

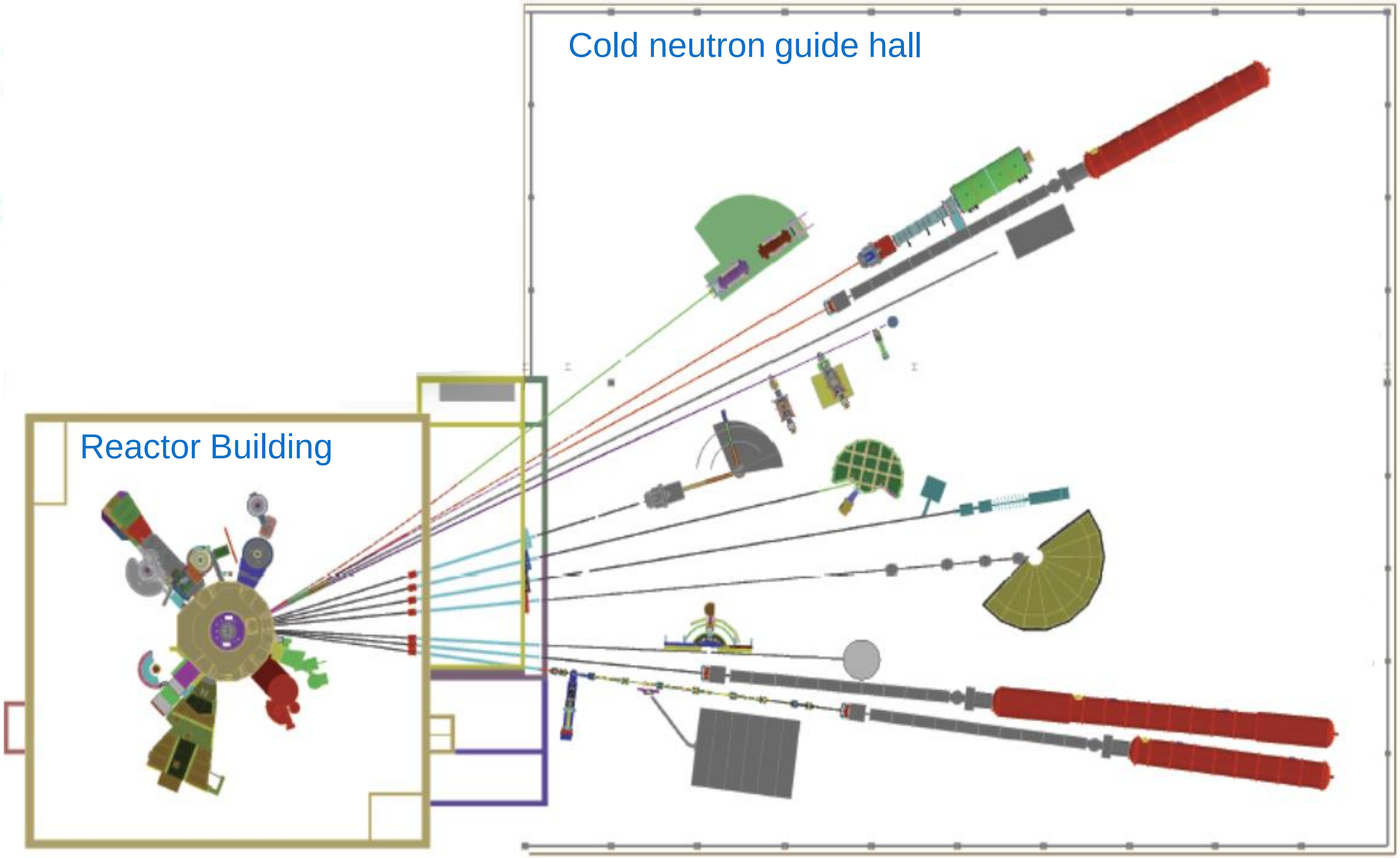


Fuel and coolant temperature transient in ULOHS.

Reactor Analysis Example #3:

Feasibility Study on a LEU Fueled Research Reactor for Advance Neutron Source at NIST

Scientific Utilization of the NBSR



NCNR has 28 instruments for various scientific experiments, 21 of them use cold neutrons (as of Dec. 2015), and hosts over 2,000 guest researchers annually, 70-80% of them are using cold neutrons.

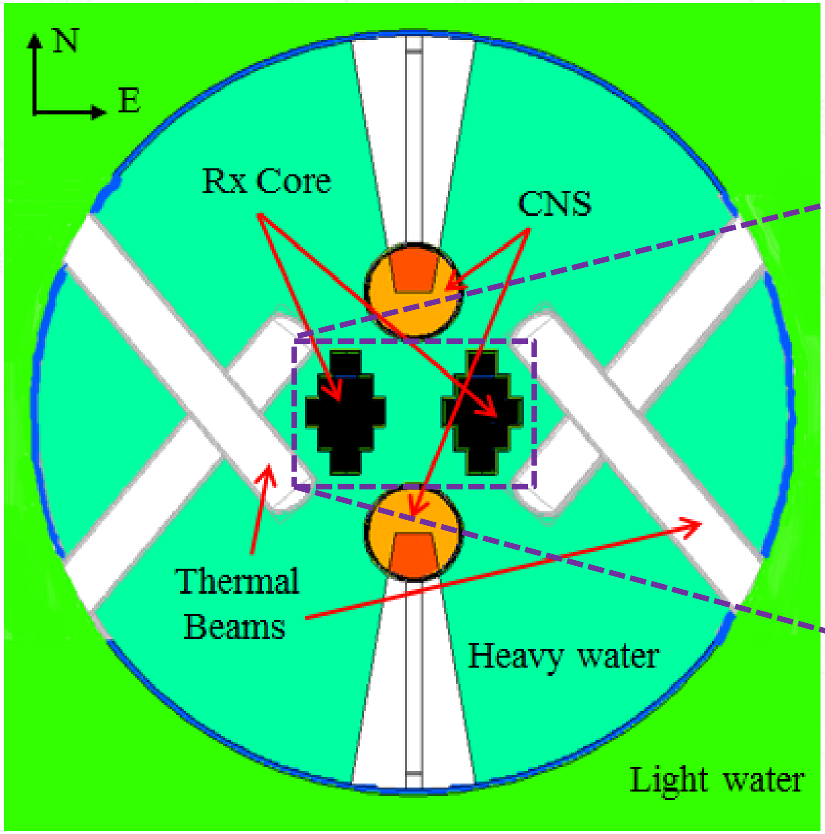
Main Design Parameters of New Reactor

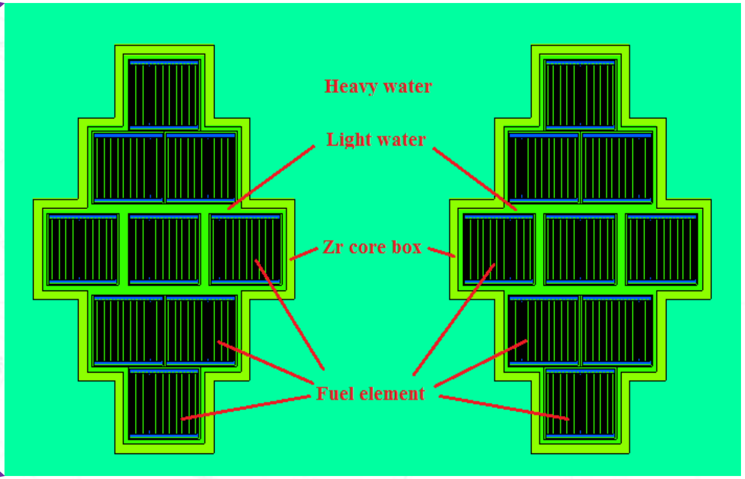
	New Reactor	NBSR
Reactor power (MW)	20 - 30	20
Fuel cycle length (days)	30	38.5
Fuel material	U_3Si_2/Al	U_3O_8/Al
Fuel enrichment (%)	19.75 (LEU)	93 (HEU)

Other Important Considerations:

- Compact core concept is employed in the design
- Principle objective is to provide cold neutron source (CNS)
- At least TWO CNSs are targeted in the new design
- Significantly utilize existing facilities and resources
- Combine latest proven research reactor design features

Schematics of the Split-Core Design

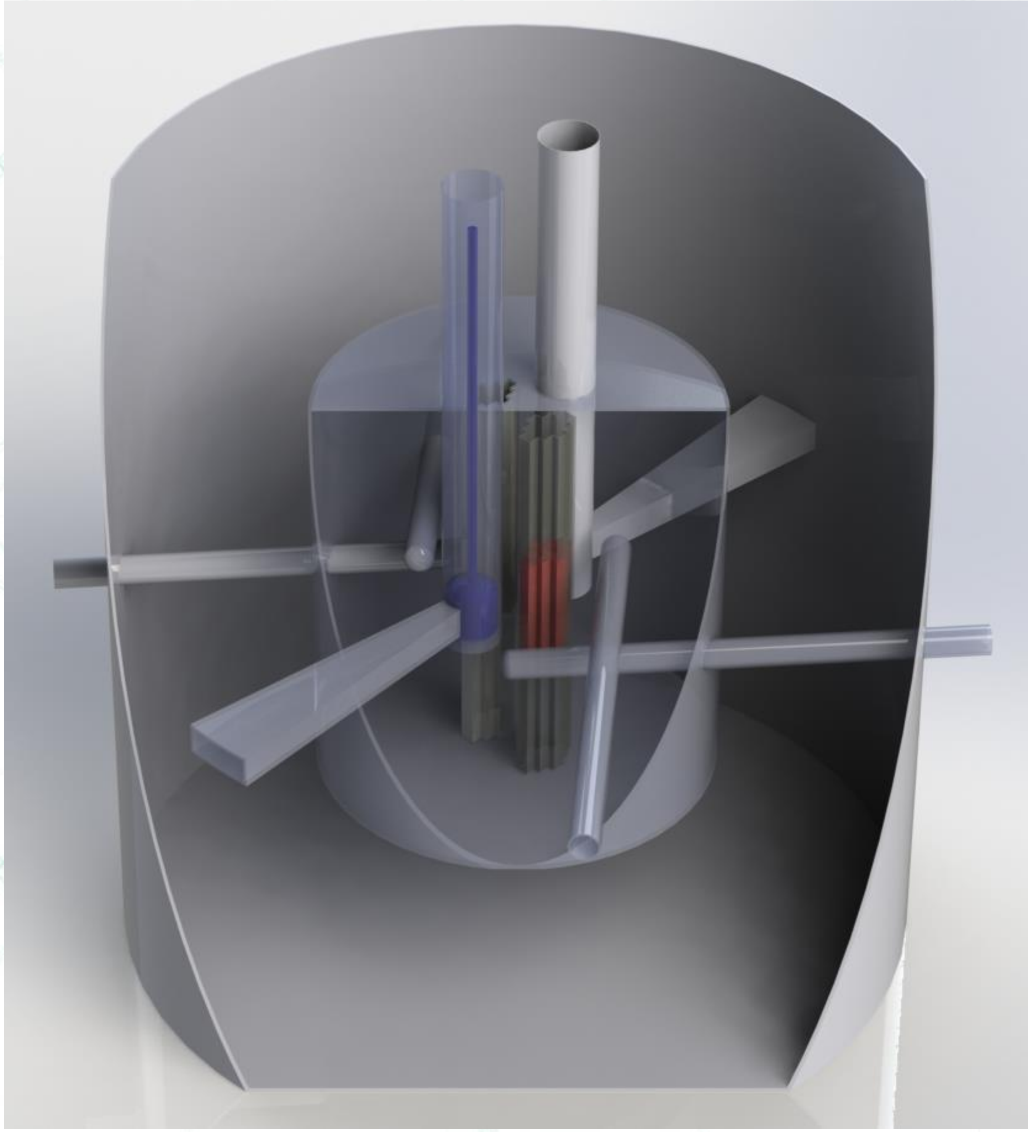




The core consists of total 18 fuel elements which are evenly distributed into two horizontal split regions.

The mid-plane of the split core reactor. Two cold neutron source (CNS) are placed in the north and south side of the core, and four thermal beam tubes are located in the east and west side of the core at different elevations.

Cut-away View of the Split-Core Design



Computational Toolkit Used in the Study

Neutronics

MCNP6.1

ENDF/B-VII.1

Power, Kinetics parameters, etc. →

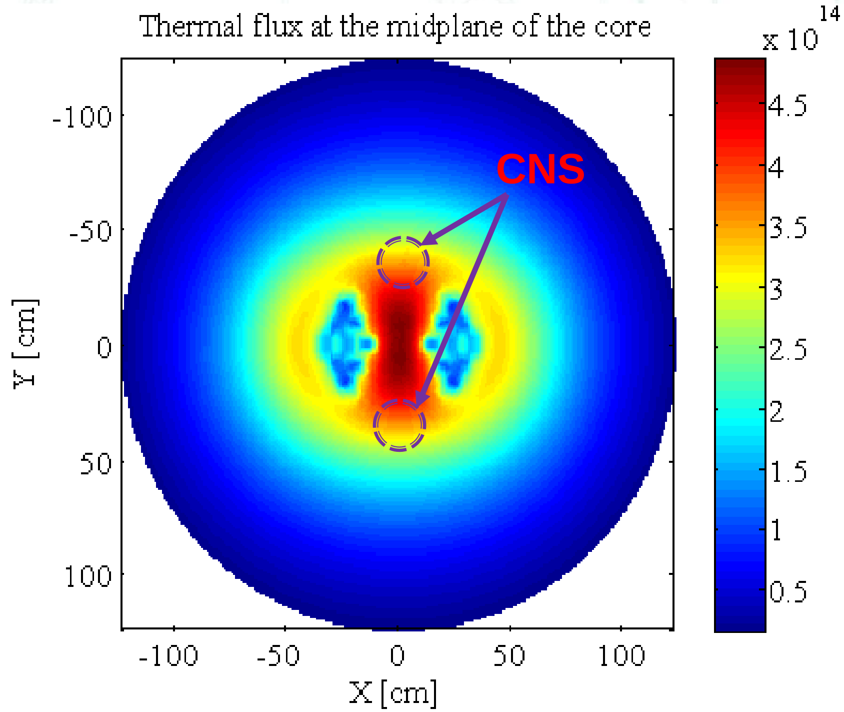
← Fuel, clad, coolant Temperature

Thermal Hydraulics

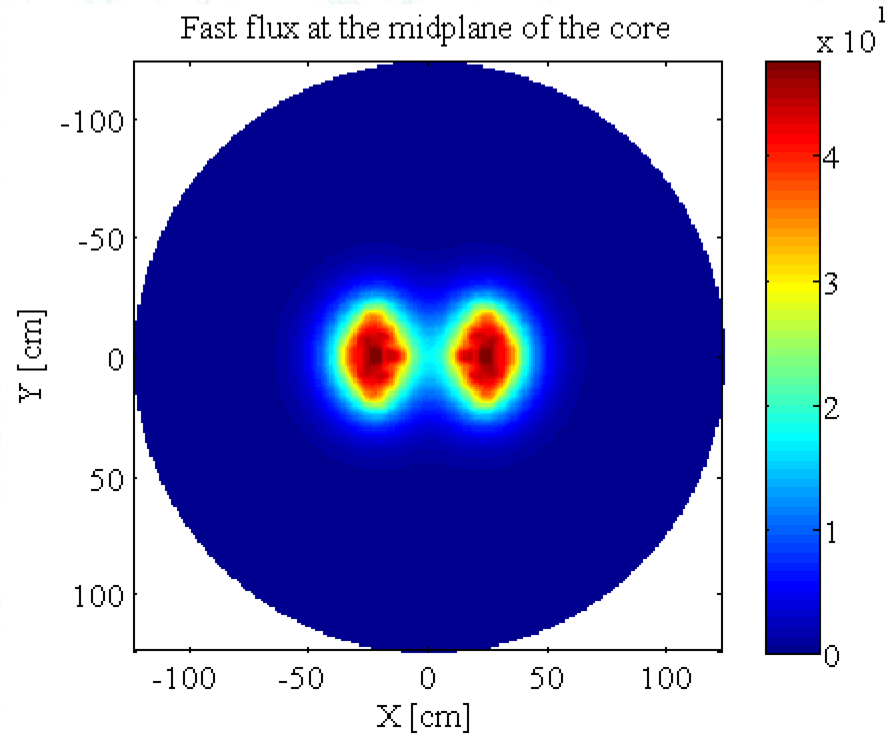
PARET/ANL

Channel Code

Top View of the Unperturbed Flux at EOC



Thermal Flux



Fast Flux

Maximum thermal flux at the core center ≈ 5×10^{14} n/cm^2-s.

Neutronics Performance Characteristics of the New Reactor

Reactor	NBSR	HFIR	BR-2	OPAL	CARR	FRM-II	NBSR-2
Country	U.S.	U.S.	Belgium	Australia	China	Germany	U.S.
Power (MW_{th})	20	85	60	20	60	20	20
Fuel	HEU	HEU	HEU	LEU	LEU	HEU	LEU
Max Φ_{th} ($\times 10^{14}$ n/cm^2-s)	3.5	10	12	3	8	8	5
Quality factor ($\times 10^{13}$ MTF/MW_{th})	1.8	1.2	2.0	1.5	1.3	4.0	2.5

The **Quality factor** is defined as the ratio of maximum thermal flux (MTF) to the total thermal power of the reactor

Cold Neutron Performance

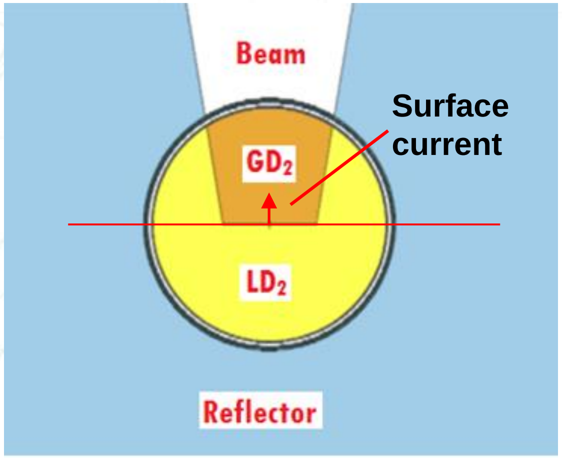
SU Results

Surface Current at the exit hole (n/cm2-s)			
E (ev)	North CNS	South CNS	NBSR CNS
5.00E-09	5.53E+11	5.68E+11	8.18E+10
Cell flux (n/cm2-s)			
E (ev)	North CNS	South CNS	NBSR CNS
5.00E-09	7.51E+13	7.57E+13	1.80E+13

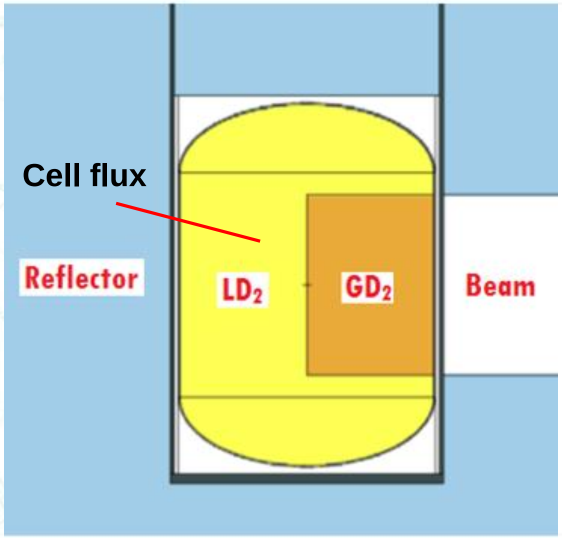
EOC Results

Surface current at the exit hole (n/cm2-s)		
E (ev)	North CNS	South CNS
5.00E-09	5.44E+11	5.46E+11
Cell flux (n/cm2-s)		
E (ev)	North CNS	South CNS
5.00E-09	7.35E+13	7.37E+13

The cold neutron flux produced by the new reactor outperforms the NBSR Unit-2 CNS by a factor of ~7.



Top View



Side View

Summary of What I have Talked

$$\frac{1}{v(E)}\frac{\partial}{\partial t}\left[\psi(\underline{r},E,\underline{\Omega},t)\right]=-\underline{\Omega}\cdot\underline{\nabla}\psi(\underline{r},E,\underline{\Omega},t)-\Sigma_t(\underline{r},E,t)\psi(\underline{r},E,\underline{\Omega},t)$$
$$+\int_0^\infty dE'\int_{4\pi} d\Omega'\Sigma_s(\underline{r},E'\to E,\underline{\Omega}'\to\underline{\Omega},t)\psi(\underline{r},E',\underline{\Omega}',t)$$
$$+\frac{\chi_p(E)}{4\pi}\int_0^\infty dE'\nu_p\Sigma_f(\underline{r},E',t)\phi(\underline{r},E',t)+\frac{1}{4\pi}\sum_{i=1}^{6}\chi_{di}(E)\lambda_i C_i(\underline{r},t)$$
$$+S_{ext}(\underline{r},E,\underline{\Omega},t)$$

Who am I?

What can I do?

Nuclear Reactor Design and Analysis

- Boiling Water Reactor (LWR)
- Sodium Fast Reactor (Gen-IV)
- Research Reactor (Non-Power)

What have I done?

So what's next?

Guru's Perspectives

A Winding Road Through Reactor Physics

- Real-time operator training simulators
  - Parallel computing methods
  - Dynamic fuel performance modeling
  - Robust 5 and 6 equation thermal-hydraulic models
  - Balance of plant systems and controller modeling
  - Blow-down and re-flood thermal/hydraulic effects
  - Non-linear, coupled, multi-physics solution methods
- BWR on-line core monitoring
  - Plant on-line data acquisition
  - On-line adaptive core modeling
  - JIT plant operations support
- HPC neutron transport methods
  - 2D/3D steady-state implementations
  - Transient time-integration methods for MOC
- HPC Monte Carlo for LWR analysis
  - On-the-fly cross section models
  - Massive tera-byte tallies and cross section generation
  - Convergence monitoring and solution acceleration

Quotes from Kord Smith's Eugene Wigner Lecture @ New Orleans ANS National Meeting in June 2016.

Moving Forward

Reactor Design and Analysis

- Accuracy & Efficiency,
- Sensitivities and Uncertainties,
- Multiphysics Modeling and Simulation,
- Probability Safety/Risk Analysis,
- Verification & Validation,

Method Development on Reactor Physics

- Hybrid Deterministic and Monte-Carlo Method,
- Large Scale Parallel Computational Transport Method,
- Nuclear Data Integrated Whole Core Transport Method,
- Multiscale Multiphysics Modeling and Simulation Method,
- Physics-based Uncertainty Quantifications Method.

Education Prerequisites

Undergraduate Level

- Introduction to Nuclear Engineering (Lamarsh),
- Nuclear Reactor Theory – Undergraduate Level (D&H or Ott),
- Nuclear Reactor Heat Transfer and Thermal Hydraulics (T&K),
- Radiation Measurements and Detections (Knoll),
- Mathematical Methods for Engineering (General and a lot),
- Senior Design in Nuclear Engineering (TBD).

Graduate Level

- Nuclear Reactor Theory – Graduate Level (D&H or Ott),
- Numerical Methods in Reactor Analysis (Papers and Manuals),
- Computational Methods of Neutron Transport (L&M),
- Dynamics of Nuclear Reactors (Hetrick),
- Neutron Transport Theory (Bell&Glasstone and D&M).

Special Topics

- Sensitivity and Uncertainty Analysis (Cacuci),
- Fast Reactor Physics (Waltar),
- Multi-physics Modeling and Simulation (Papers).

Thank you for your time!