Numerical Benchmark Results for 1000MWth Sodium-cooled Fast Reactor

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Content

- **Description of 1000 MWth SFR Core Concepts**
  - Background
  - Description of 1000MWth SFR Cores
- **ANL Fast Reactor Analysis Code Suite**
- **Comparison of Benchmark Results**
- **Conclusions and Potential Future Work**

- **Backup**
  - Informational Note of ANL Methods for Reactivity Feedback Calculations
  - Large SFR core power distribution
  - Proposed schedule

Background

- **Advanced Burner Reactor (ABR) was developed to demonstrate transmutation, cost reduction, fast reactor safety, and qualify fuels and materials**
  - 1000 MWt core concepts were developed to be utilized as reference or benchmark core concepts
  - Studies were performed to achieve low conversion ratio (~0.25) and break-even ratio (~1.0)
  - Heterogeneous recycle of TRU considered as an alternative approach
- **Advanced/Innovative core concept studies are being performed under US-DOE Nuclear Reactor Technology Programs**
  - Impact of power increase up to 2000 MWt or 3000 MWt
  - Impact of fuel forms with carbide and nitride fuels
  - Cost reduction studies with fission gas vented fuel, advanced shielding and structural materials, additives in primary coolant, etc.
  - Advanced core concepts such as small modular reactor, ultra-long life core, traveling wave reactor, etc

Reference 1000 MWth ABR Core Concepts

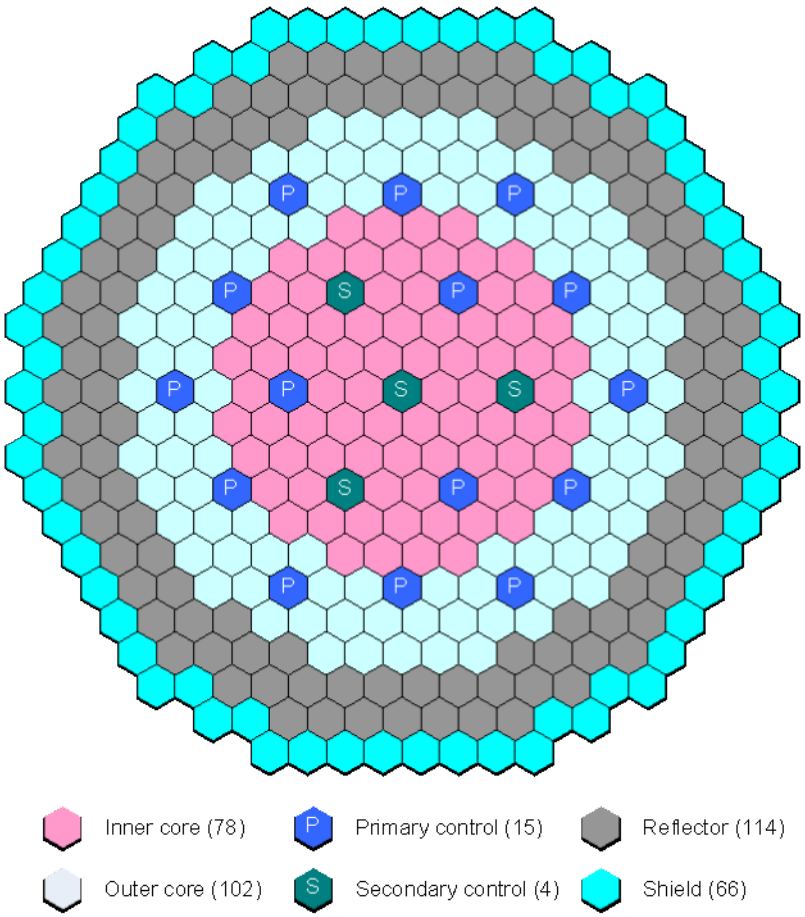
- **Design goals**
  - Compact core with cycle length of one-year
  - Maximize discharge burnup within proposed design constraints
  - Small excess reactivity and moderately low TRU conversion ratio (~0.7)
  - Coolant outlet temperature of 510°C with mean ΔT of 155°C
- **Design constraints and considerations**
  - Able to exchange between ternary metallic (U-TRU-Zr) and mixed oxide (UO_2+$TRUO_2$) fuels
  - Fuel smeared density of 75% for metallic fuel and 85% for oxide fuel
  - Maximum TRU mass fraction is less than US irradiation experiences (~30%)
  - HT-9 cladding with peak fast fluence limit of 4 x 10^{23} n/cm^2
  - Peak 2σ cladding inner wall temperature of metallic fuel less than 650°C
  - Peak 2σ fuel center line temperature less than fuel melting temperature
- **Reference ABR-1000 was developed targeting relatively quick licensing based on US-DOE fast reactor experiences**
  - Optimizations and innovations were not attempted

Development of Numerical Benchmark Cores

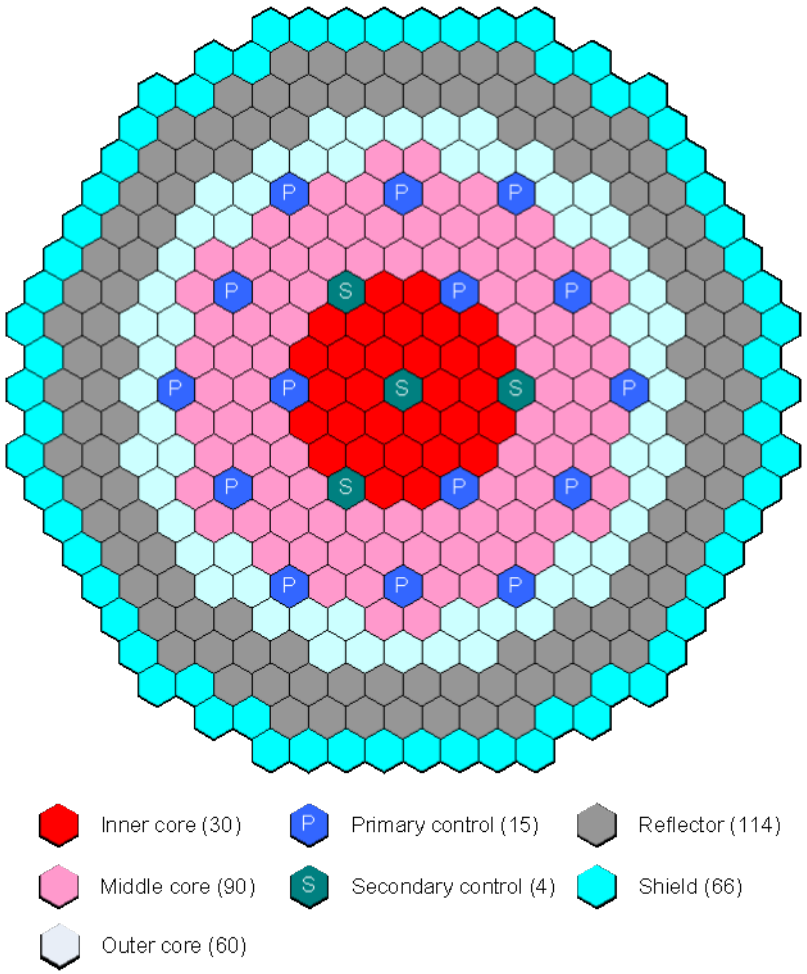
- **Medium size benchmark cores were developed using reference ABR-1000 core concepts**
  - Generally, data and fuel composition of benchmark core were obtained from reference ABR core concepts at nominal operating condition
  - Design parameters were adjusted by material thermal expansions and irradiation-induced fuel swelling (in particular, metallic fuel)
  - Fuel compositions were obtained at equilibrium conditions, but it was simplified
    - Active core is axially divided into five zones and uniform composition is assumed in each zone
  - Simplified fission product model was adopted with pseudo fission product
    - Natural Molybdenum is used as pseudo fission product
  - Nominal operating temperatures

	Metal Core	Oxide Core
Coolant temperature, °C	432.5	432.5
Structure temperature, °C	432.5	432.5
Ave. fuel temperature, °C	534.0	1027.0

Benchmark Core Configurations



Metal Core



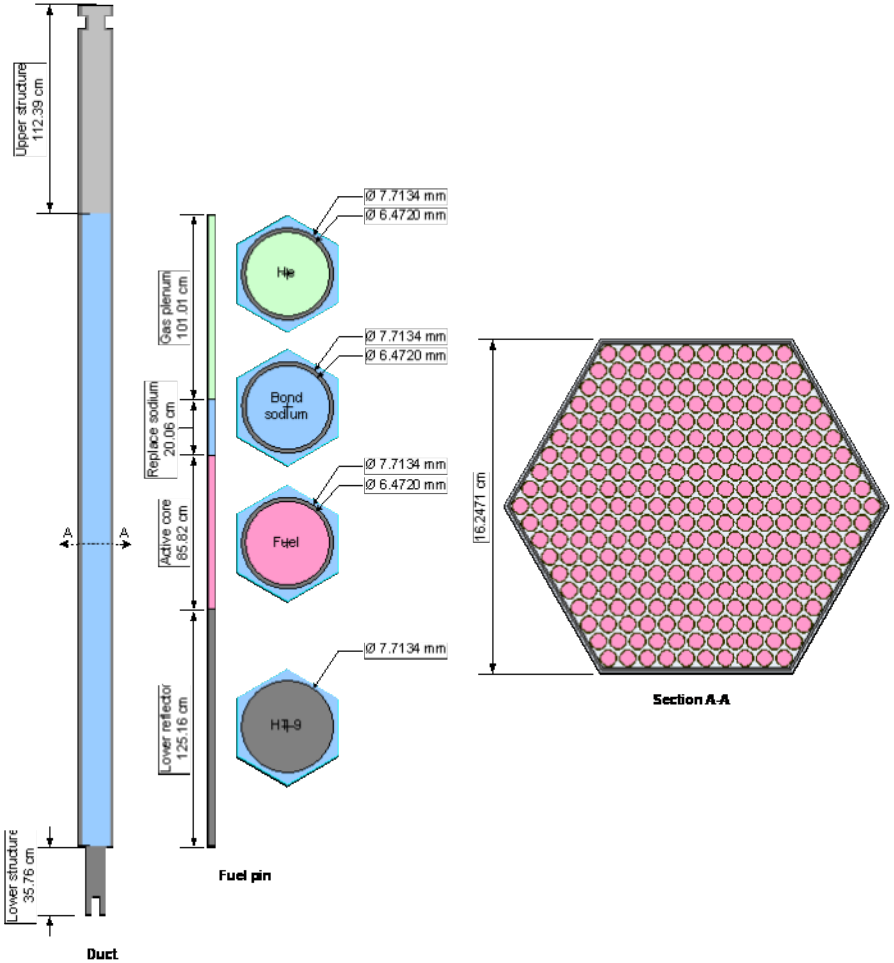
Oxide Core

Subassembly Parameters at Nominal Condition

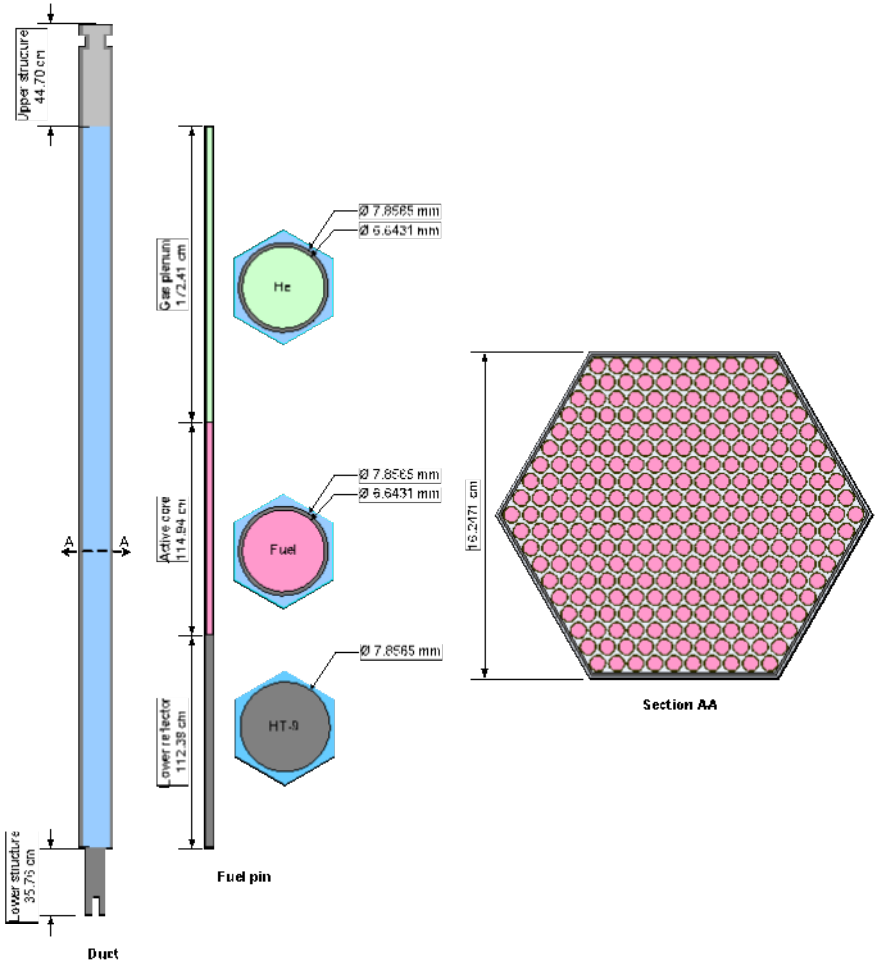
	Metal Core	Oxide Core
Overall length of subassembly, cm	482.20	482.20
- Lower structure	35.76	35.73
- Lower reflector	125.16	112.39
- Active core height	85.52	114.94
- Replaced bond sodium	20.06	-
- Upper gas plenum	101.01	171.41
- Upper structure	112.39	44.70
Subassembly pitch, cm	16.2471	16.2471
Subassembly duct outer flat-to-flat distance, cm	15.8123	15.8123
Subassembly duct wall thickness, cm	0.3966	0.3966
Number of fuel pins per subassembly	271	271
Cladding outer radius, cm [a)]	0.3857	0.3928
Cladding inner radius, cm	0.3236	0.3322
Volume fraction, %		
- Fuel	39.0	41.1
- Sodium	35.3	33.3
- HT-9	25.7	25.6

a) Cladding radius is slightly increased by smearing wire-wrap to cladding

Schematics of Driver Subassembly

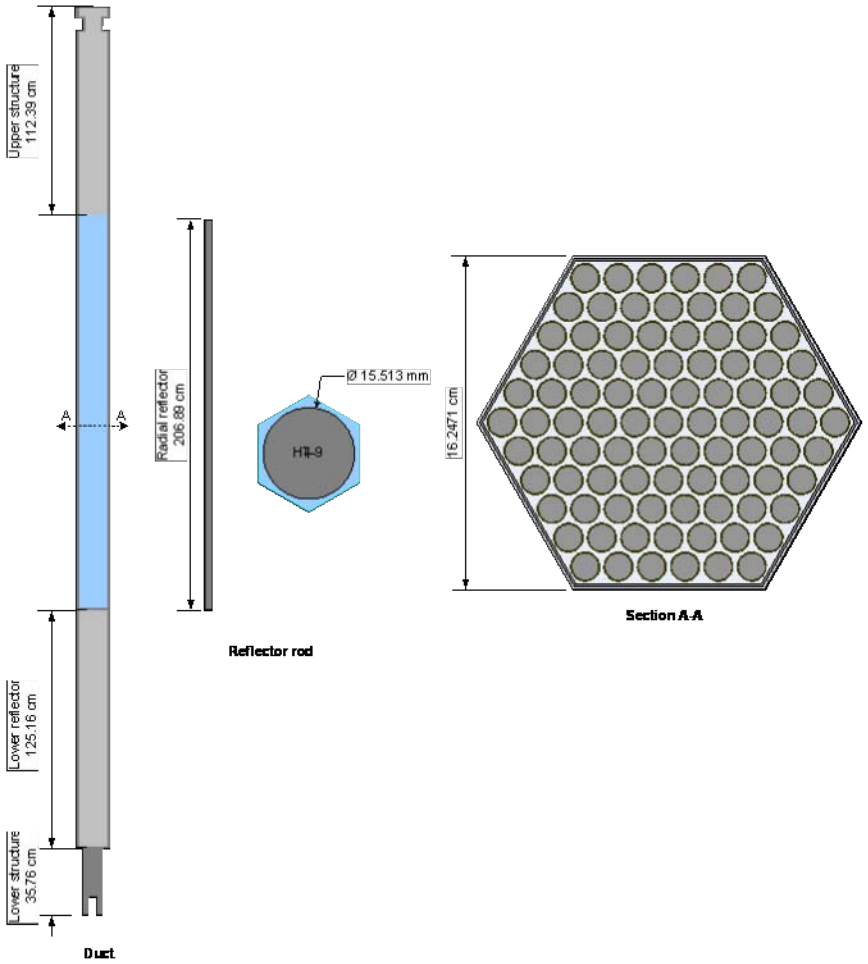


Metal Core

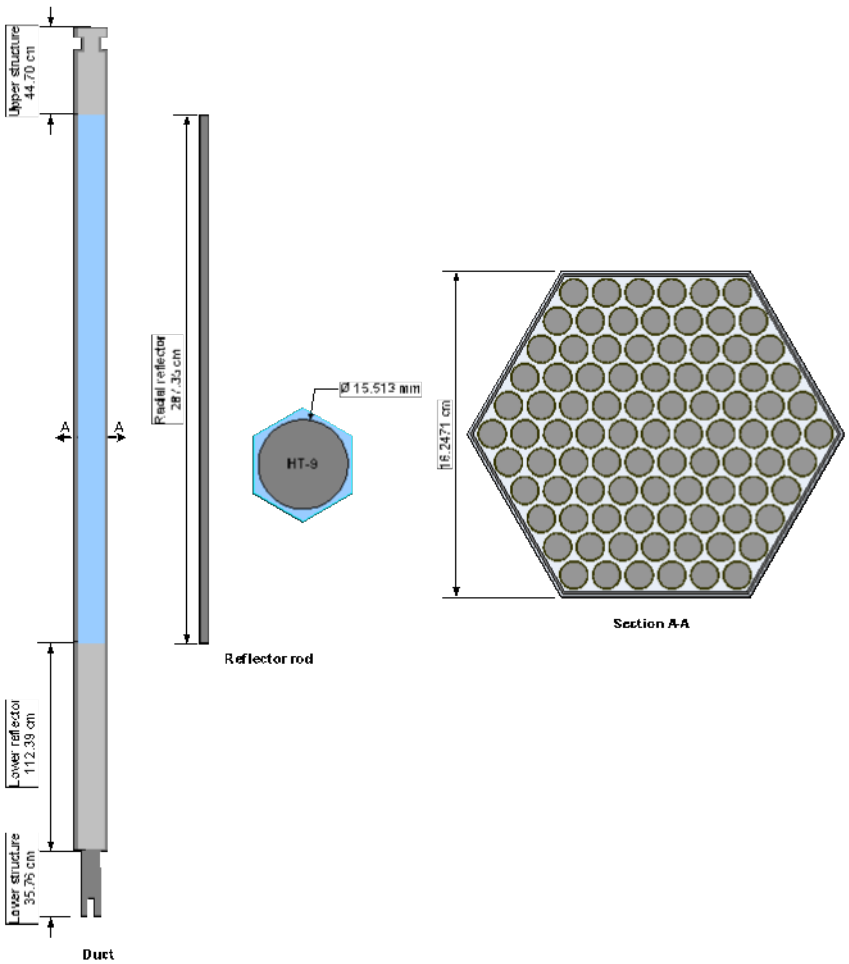


Oxide Core

Schematics of Radial Reflector

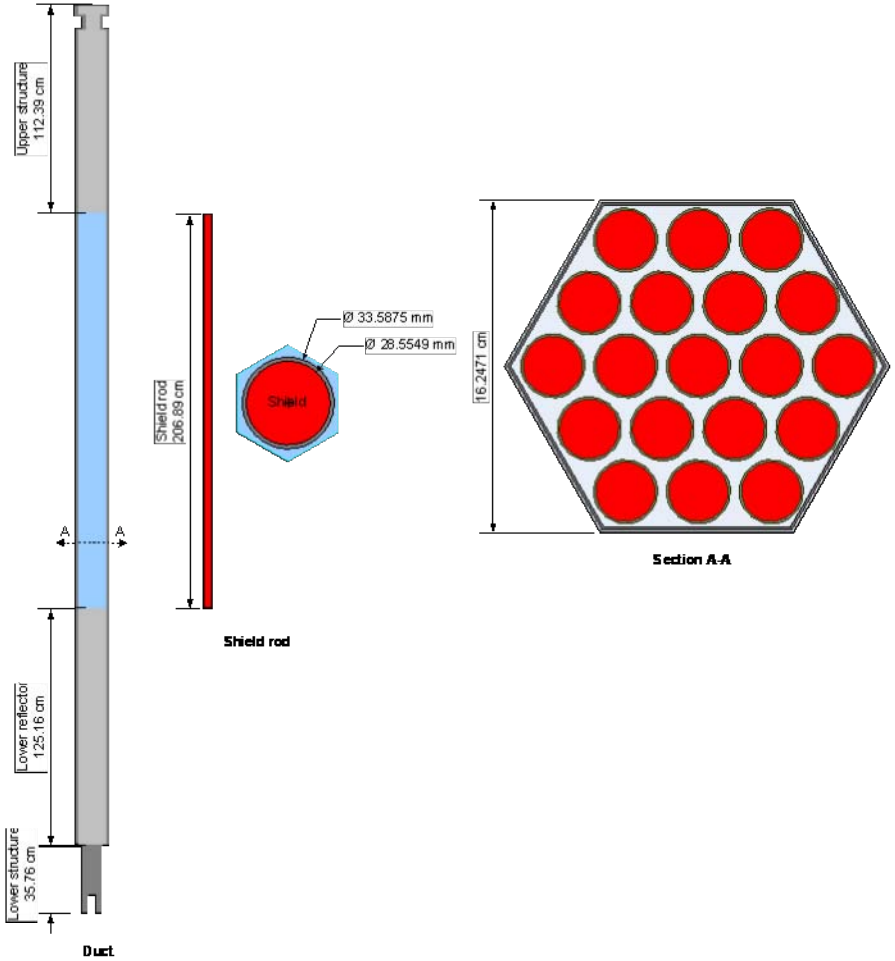


Metal Core

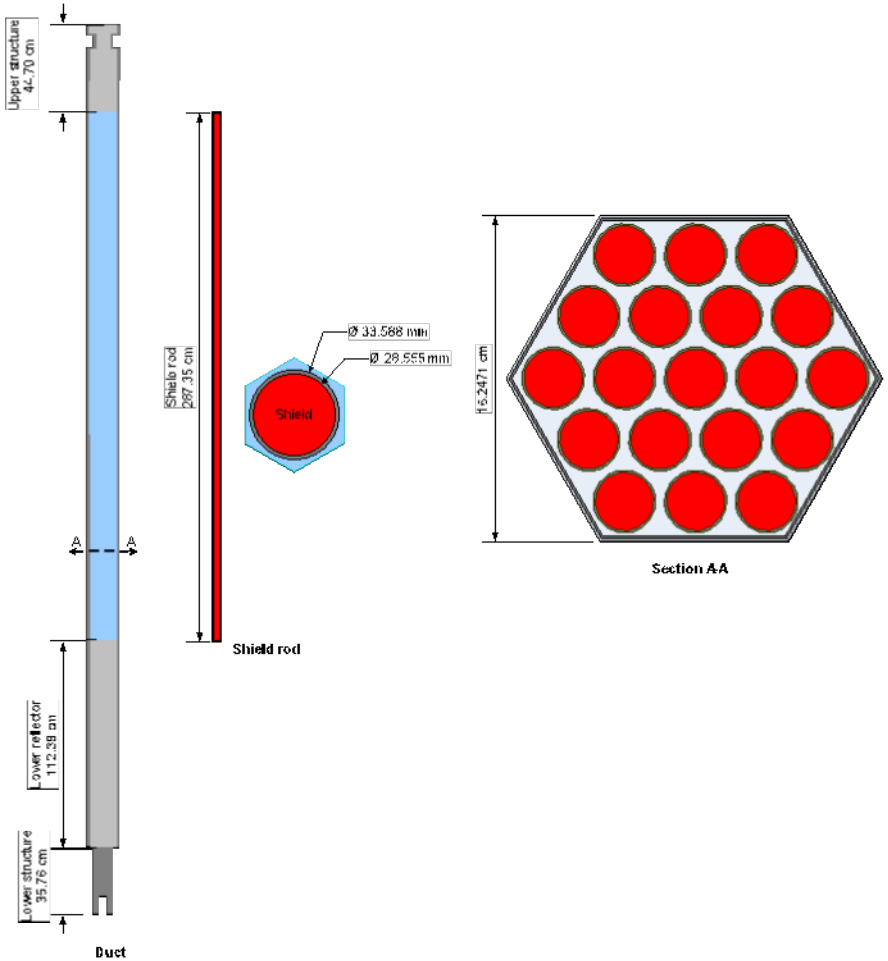


Oxide Core

Schematics of Radial Shield

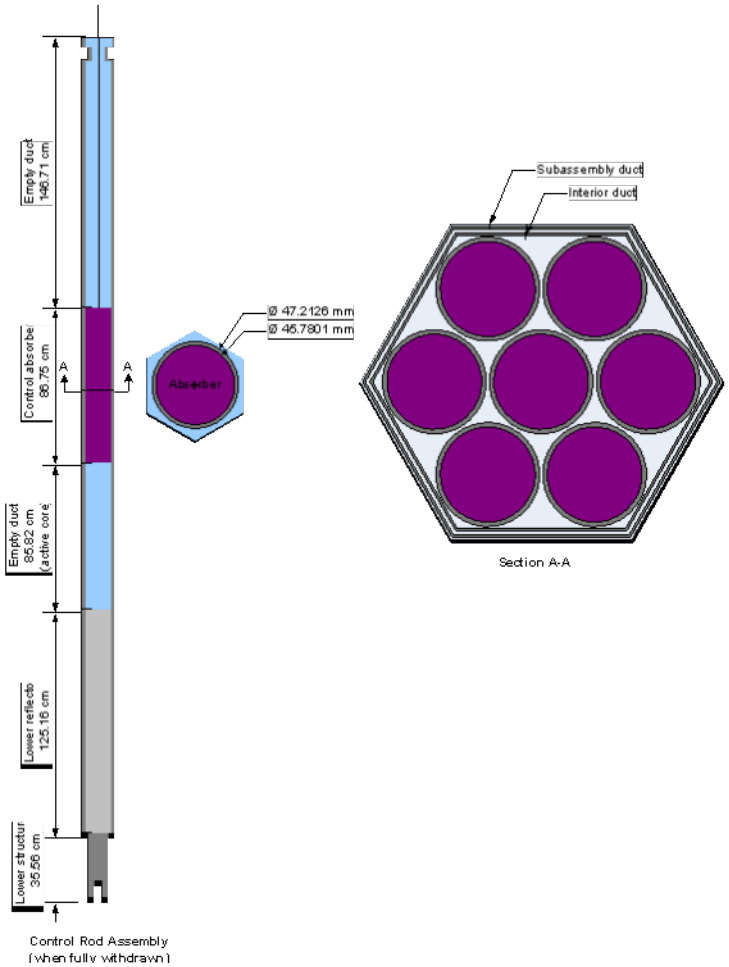


Metal Core

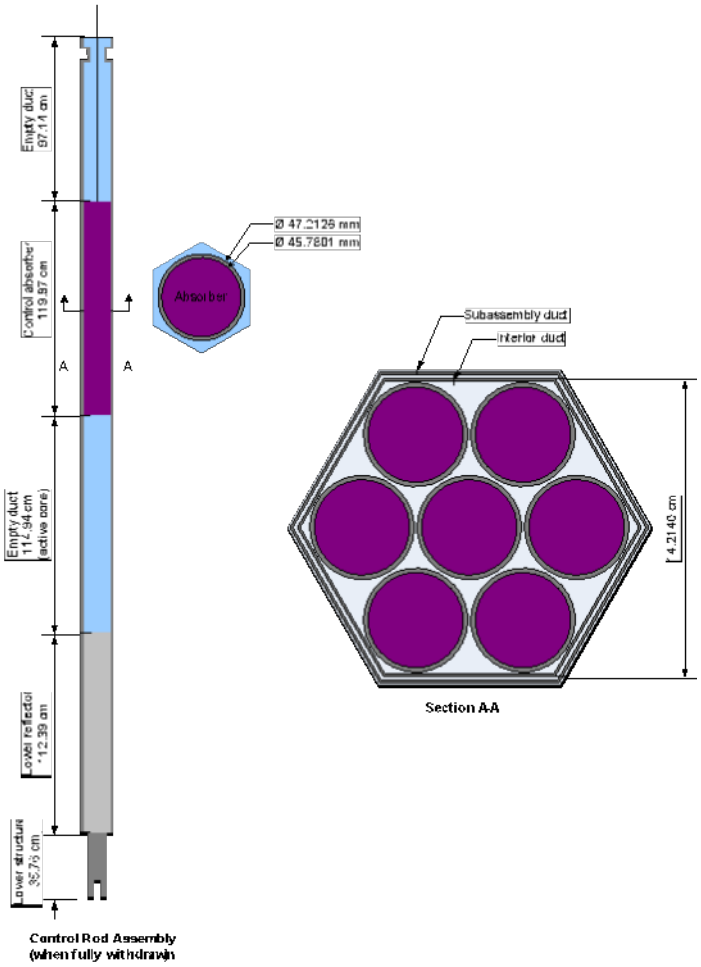


Oxide Core

Schematics of Control Subassembly



Metal Core



Oxide Core

Fuel Composition

- **Fuel composition at BOC, %**

	Metal Core		Oxide Core		
	Inner core	Outer core	Inner core	Middle core	Outer core
U	81.4	75.4	79.5	76.3	69.6
Pu	17.1	22.4	18.8	21.5	27.4
MA	1.5	2.3	1.7	2.1	3.0
TRU	18.6	24.6	20.5	23.7	30.4
Fissile	10.2	13.0	10.6	11.8	15.1

Isotopic number densities are provided in benchmark descriptions

- **EOC is defined by full power irradiation for 328.5 days, which is one-year cycle length with 90% capacity factor**
- **It was assumed that all control assemblies are fully withdrawn from active core during irradiation**
- **EOC fuel compositions are not provided in this benchmark (Participant will evaluate them)**

Expected Results

- **Following values are expected**
  - Core multiplication factors
  - Sodium Void worth ($\Delta\rho=\rho_{void}-\rho_{nominal}$)
    - Void state is defined by voiding sodium completely in active core
  - Doppler constant ($K_D = (\rho_{high}-\rho_{nominal})/\ln 2$)
  - Effective delayed neutron fraction
  - Average nuclide masses per each core at EOC
  - Radial power distribution
  - Control rod worth ($\Delta\rho=\rho_{ARI}-\rho_{ARO}$)
    - Both primary and secondary rods are inserted to bottom of active core

- **Reference**
  - T. K. Kim et al, "Core Design Studies for a 1000 MWth Advanced Burner Reactor," Annals of Nuclear Energy 36 (2009)

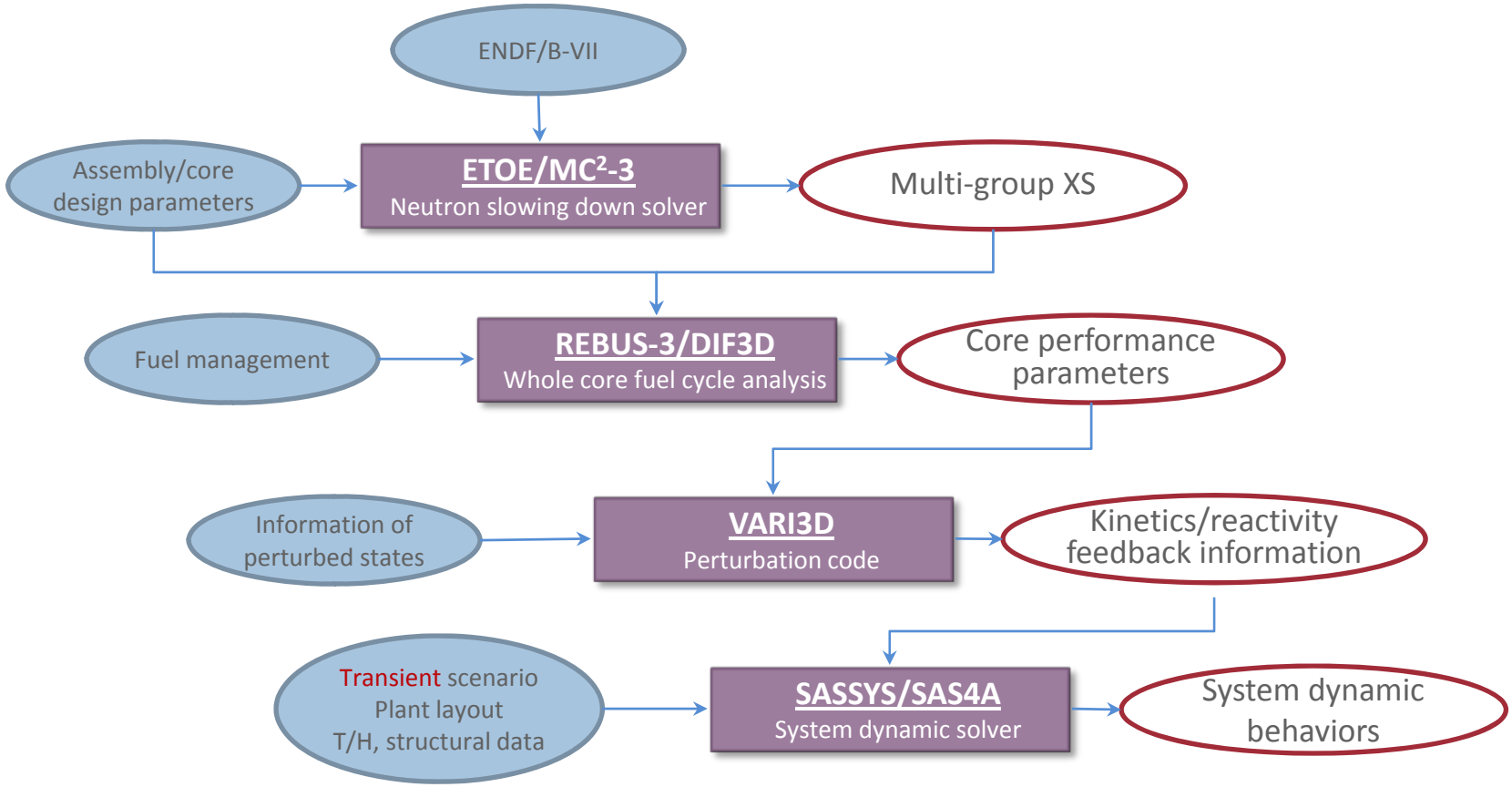
Participants of Medium Core Benchmarks

Participant	Country	Primary evaluator	Computation tools
ANL	U.S.A	T. K. Kim	REBUS-3/DIF3D, VARI3D, MCNP5
CEA	France	L. Buiron	ERANOS
KTH/UIUC	Sweden/USA	R. Garcia and T. Kozlowski	SERPENT
KFKI	Hungary	P. Vertes	NOTRAKET program system

***) MCNP5 and SERPENT are Monte Carlo codes**

****) KFKI used KENO/DANTSYS.**

ANL Fast Reactor Analysis Code Suite



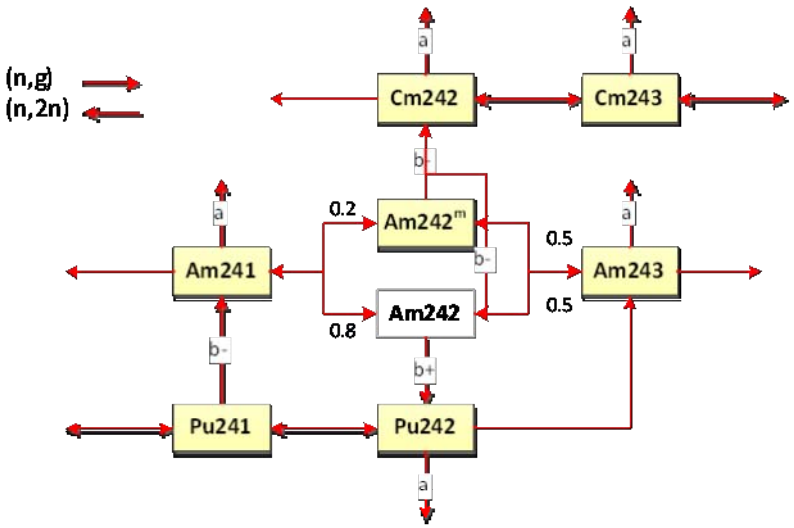
VIM	RCT	SE2-ANL	DIF3D-K/VARIANT-K	LIFE-METAL	Etc
Monte Carlo code	Intra-assembly reconstruction	Steady-state T/H solver	Space-dependent kinetics	Thermal/mechanical fuel	

ANL Neutronics Codes

- **ETOE-2/MC^2-3:** Generates multi-group cross sections based on detailed spectrum calculations (2082 group) for individual compositions
- **DIF3D:** Multi-group steady-state diffusion and transport theory solvers
  - Finite difference diffusion theory method, nodal diffusion theory method, variational nodal transport theory method
- **REBUS-3:** Fast reactor fuel cycle analysis code
  - Equilibrium or non-equilibrium cycle analysis for various fuel management schemes
  - Various flux solvers available: all DIF3D options, TWODANT, and MCNP
  - External cycle model: Flexible reprocessing, re-fabrication, and modeling for time delays between various processes and radioactive decays
- **VARI3D:** Perturbation theory code
  - First-order and exact perturbation theory
  - Computes reactivity feedback coefficients
  - Computes sensitivity of bi-linearly weighted reaction rate ratios to cross section and number density variations
  - DIF3D flux and adjoint solutions (finite difference diffusion theory option)

Assumptions in ANL Calculations

- **Depletion Chains**
  - Short half-life isotopes are simplified
- **Fission product model**
  - In simplified model, it is assumed that two Nat. Mo isotopes are generated per fission as pseudo-fission product
- **Both fission and capture energy were represented**
- **Solutions**
  - 21-group diffusion theory equations were solved by DIF3D/nodal option (one-hexagon with ~20 cm axial segment)
  - Exact and/or first order perturbation theory was used to calculate kinetics parameters and reactivity feedback coefficients



Results of Medium Size Metallic Core

	BOC				EOC			
	ANL	CEA	KTH	KFKI	ANL	CEA	KTH	KFKI
keff	1.02592	1.03715	a) 1.0406	0.9817	1.00855	1.00997	1.0230	0.9656
βeff	0.0033	0.0036	0.0035	0.0033	0.0033	0.0035	0.0035	0.0030
Sodium void worth, $	7.96	5.61	5.50	-0.16	8.26	7.64	4.29	-0.004
Doppler constant, $	-0.88	-1.03	-0.05	-0.09	-0.90	-1.02	0.02	0.002
Control rod worth, $	-63.2	-54.5	-51.7	-53.2	-66.0	-56.7	-57.0	-100.6

a) STD = ± 0.0008

- **MCNP results at BOC**

Temperature	Library	Metallic
300K	ENDF/B-V	1.03406 ± 0.0006
	ENDF/B-VII	1.02937 ± 0.0006
900K	ENDF/B-VII	1.02233 ± 0.0006
HOT *)	ENDF/B-VII	1.02273 ± 0.0006

***) HOT condition: Coolant = 600K, Structure=900K, Fuel=1200K**

Results of Medium Size Oxide Core

	BOC				EOC			
	ANL	CEA	KTH	KFKI	ANL	CEA	KTH	KFKI
keff	1.01869	1.03530	a) 0.9917	0.9656	1.01019	1.00803	0.9862	0.3153
βeff	0.0032	0.0035	0.0034	0.0031	0.0032	0.0034	0.0034	0.0004
Sodium void worth, $	7.19	5.57			7.34	6.03		-0.15
Doppler constant, $	-1.91	-2.29		-0.006	-1.91	-2.24		-0.026
Control rod worth, $	-69.0	-62.2		-61.8	-71.4	-64.3		-193.9

a) STD = ± 0.0008

- **MCNP results at BOC**

Temperature	Library	Oxide
300K	ENDF/B-V	1.02675 ± 0.0006
	ENDF/B-VII	1.03499 ± 0.0006
900K	ENDF/B-VII	1.02158 ± 0.0006
HOT *)	ENDF/B-VII	1.02192 ± 0.0006

***) HOT condition: Coolant = 600K, Structure=900K, Fuel=1200K**

EOC Core Inventory (kg)

	Metallic				Oxide			
	ANL	CEA	Diff.(%)	KFKI	ANL	CEA	Diff.(%)	KFKI
U	10,019.4	10,008.5	0.1	8927.9	10,771.4	10,780.1	0.1	11041.9
Pu	2,549.5	2,561.8	0.5	2247.2	3,343.6	3,334.7	0.3	3395.5
Am	144.6	145.8	0.9	130.3	202.5	201.5	0.5	210.7
Cm	61.3	63.0	2.8	51.8	96.7	98.4	1.7	94.8

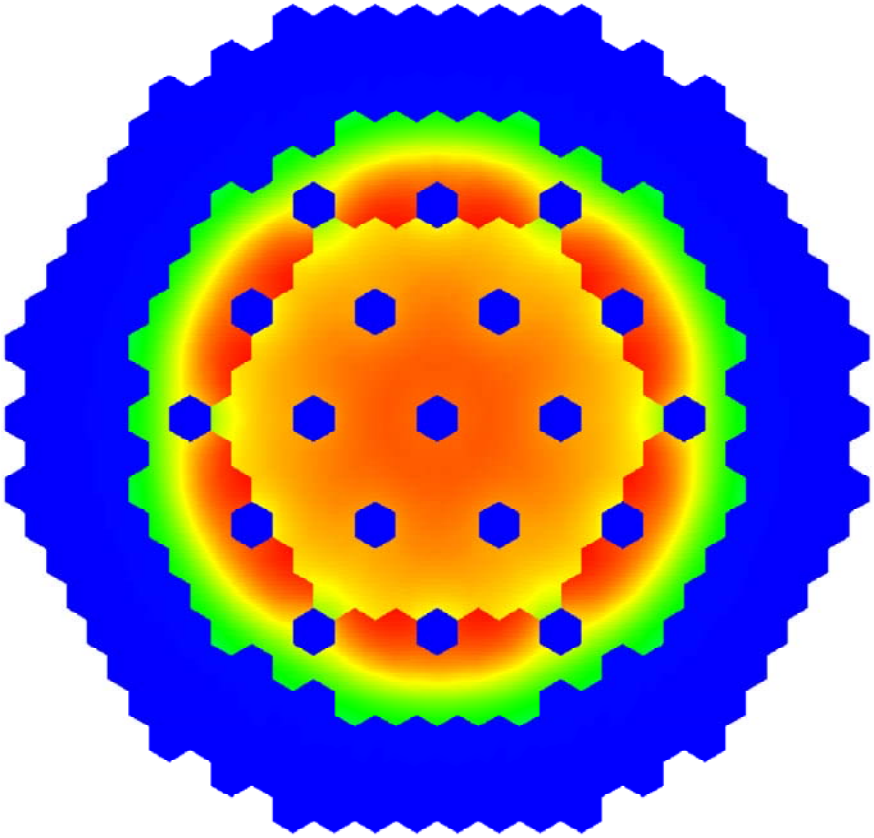
- **Relatively large differences in Cm mass between ANL and CEA are due to different decay chains**

EOC Number Densities

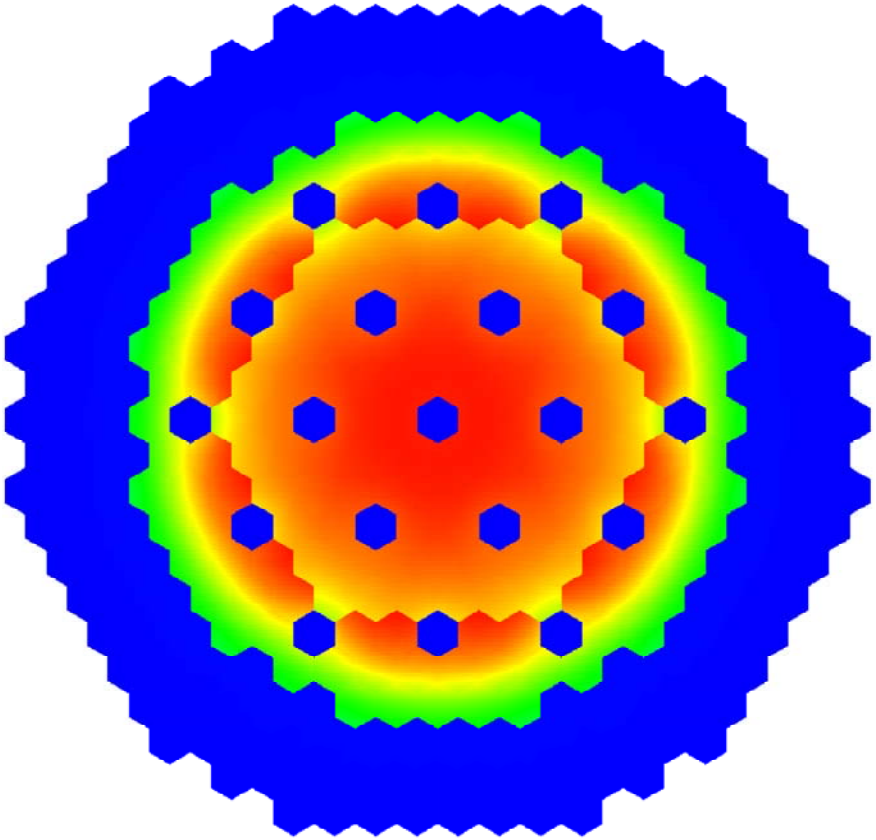
	Metallic Core			Oxide Core		
	ANL	CEA	Diff.(%)	ANL	CEA	Diff.(%)
U234	8.54E-06	8.61E-06	0.8%	1.74E-05	1.75E-05	0.6%
U235	9.79E-05	9.81E-05	0.2%	1.13E-04	1.13E-04	-0.2%
U236	1.25E-05	1.25E-05	-0.1%	1.97E-05	1.98E-05	0.8%
U238	7.18E-02	7.19E-02	0.1%	8.68E-02	8.69E-02	0.1%
Np237	2.57E-04	2.61E-04	1.3%	2.95E-04	2.98E-04	1.0%
Pu238	5.21E-04	5.24E-04	0.7%	7.83E-04	7.87E-04	0.6%
Pu239	9.47E-03	9.40E-03	-0.8%	1.27E-02	1.26E-02	-0.7%
Pu240	5.79E-03	5.82E-03	0.4%	9.09E-03	9.15E-03	0.6%
Pu241	8.89E-04	8.67E-04	-2.6%	1.52E-03	1.48E-03	-2.7%
Pu242	1.32E-03	1.32E-03	0.0%	2.04E-03	2.04E-03	-0.2%
Am241	5.23E-04	5.24E-04	0.2%	8.26E-04	8.24E-04	-0.3%
Am242g	0.00E+00	1.66E-07	100.0%	0.00E+00	2.57E-07	100.0%
Am242m	4.14E-05	3.99E-05	-3.6%	6.32E-05	6.27E-05	-0.9%
Am243	4.40E-04	4.34E-04	-1.3%	6.58E-04	6.53E-04	-0.8%
Cm242	2.96E-05	3.07E-05	3.7%	4.44E-05	4.76E-05	6.7%
Cm243	2.17E-06	2.39E-06	9.3%	4.23E-06	4.58E-06	7.7%
Cm244	2.84E-04	2.93E-04	3.1%	4.75E-04	4.87E-04	2.4%
Cm245	7.14E-05	7.06E-05	-1.1%	1.35E-04	1.33E-04	-1.8%
Cm246	3.81E-05	3.85E-05	0.8%	8.25E-05	8.27E-05	0.3%

- **Am242 chain was simplified in ANL calculations**

Power Distributions -Metallic Core

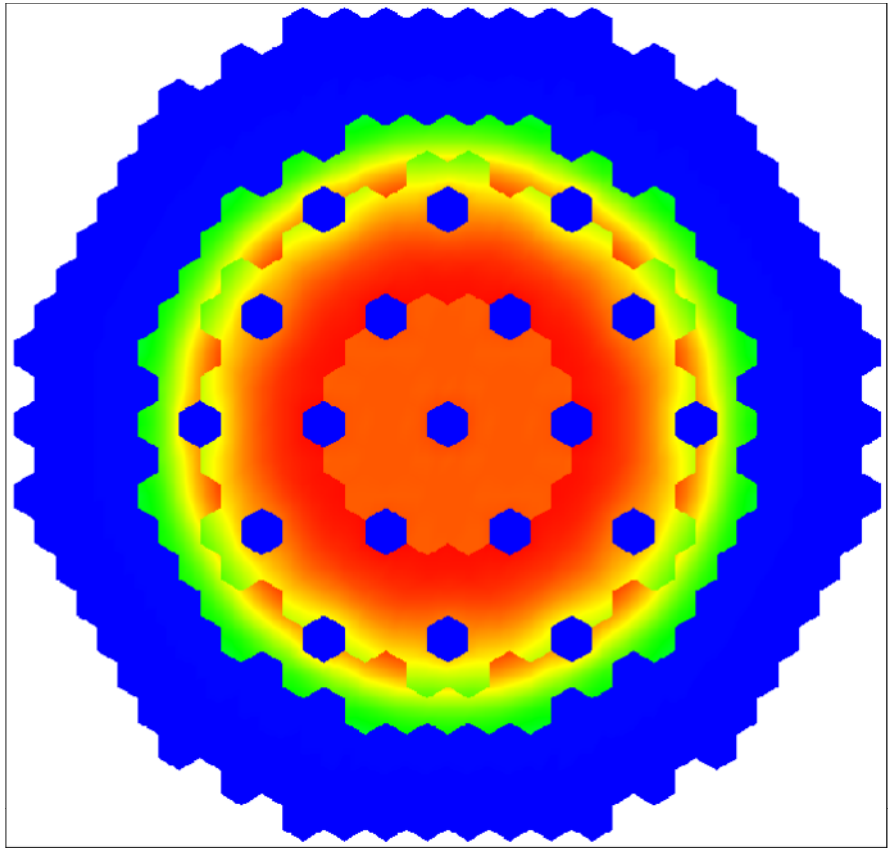


BOC

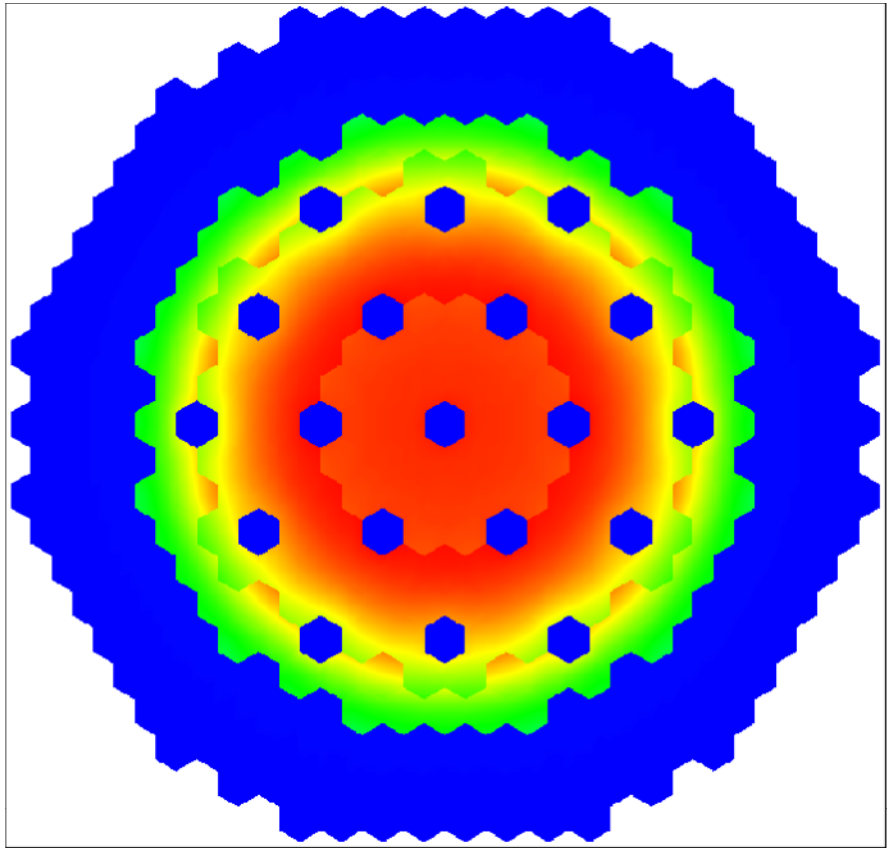


EOC

Power Distributions -Oxide Core



BOC



EOC

Power of Metallic Core (MW, 1/6 Core)

		ANL		CEA		Difference (%)	
Ring number	position number	BOC	EOC	BOC	EOC	BOC	EOC
1	1	0.1	0.1	0.0	0.0		
2	1	6.3	6.7	6.3	6.6	-0.7%	-1.8%
2	2	6.3	6.7	6.3	6.6	-0.7%	-1.8%
3	3	6.2	6.5	6.1	6.4	-0.7%	-1.6%
3	4	6.2	6.6	6.2	6.5	-0.7%	-1.6%
4	1	0.2	0.2	0.0	0.0		
4	2	6.1	6.4	6.0	6.3	-0.7%	-1.5%
4	3	6.1	6.4	6.0	6.3	-0.7%	-1.5%
5	1	5.7	5.9	5.7	5.8	0.1%	-0.6%
5	2	5.8	6.0	5.8	6.0	-0.7%	-1.3%
5	3	5.9	6.1	5.8	6.0	-1.0%	-1.5%
5	4	5.8	6.0	5.8	6.0	-0.7%	-1.3%
6	1	5.2	5.3	6.6	6.4	21.2%	18.2%
6	2	5.5	5.6	5.5	5.6	-0.3%	-0.6%
6	3	5.6	5.7	5.5	5.6	-1.4%	-1.6%
6	4	5.6	5.7	5.5	5.6	-1.4%	-1.6%
6	5	5.5	5.6	5.5	5.6	-0.3%	-0.6%
7	1	0.2	0.2	0.0	0.0		
7	2	6.2	6.0	6.1	6.0	-0.5%	-0.2%
7	3	6.6	6.3	6.5	6.3	-1.3%	-0.9%
7	4	0.2	0.2	0.0	0.0		
7	5	6.6	6.3	6.5	6.3	-1.3%	-0.9%
7	6	6.2	6.0	6.1	6.0	-0.5%	-0.2%
8	1	4.1	4.0	4.0	3.9	-2.5%	-1.9%
8	2	4.9	4.7	4.8	4.6	-2.7%	-2.0%
8	3	5.5	5.3	5.4	5.2	-2.6%	-1.7%
8	4	5.8	5.6	5.6	5.5	-2.9%	-2.0%
8	5	5.8	5.6	5.6	5.5	-2.9%	-2.0%
8	6	5.5	5.3	5.4	5.2	-2.6%	-1.7%
8	7	4.9	4.7	4.8	4.6	-2.7%	-2.0%
9	1	0.1	0.1	0.0	0.0		
9	2	0.1	0.1	0.0	0.0		
9	3	3.9	3.8	3.7	3.7	-4.1%	-3.0%
9	4	4.3	4.2	4.1	4.0	-4.9%	-3.6%
9	5	4.5	4.3	4.3	4.2	-5.0%	-3.7%
9	6	4.3	4.2	4.1	4.0	-4.9%	-3.6%
9	7	3.9	3.8	3.7	3.7	-4.1%	-3.0%
9	8	0.1	0.1	0.0	0.0		
9	9	0.1	0.1	0.0	0.0		

Power of Oxide Core (MW, 1/6 Core)

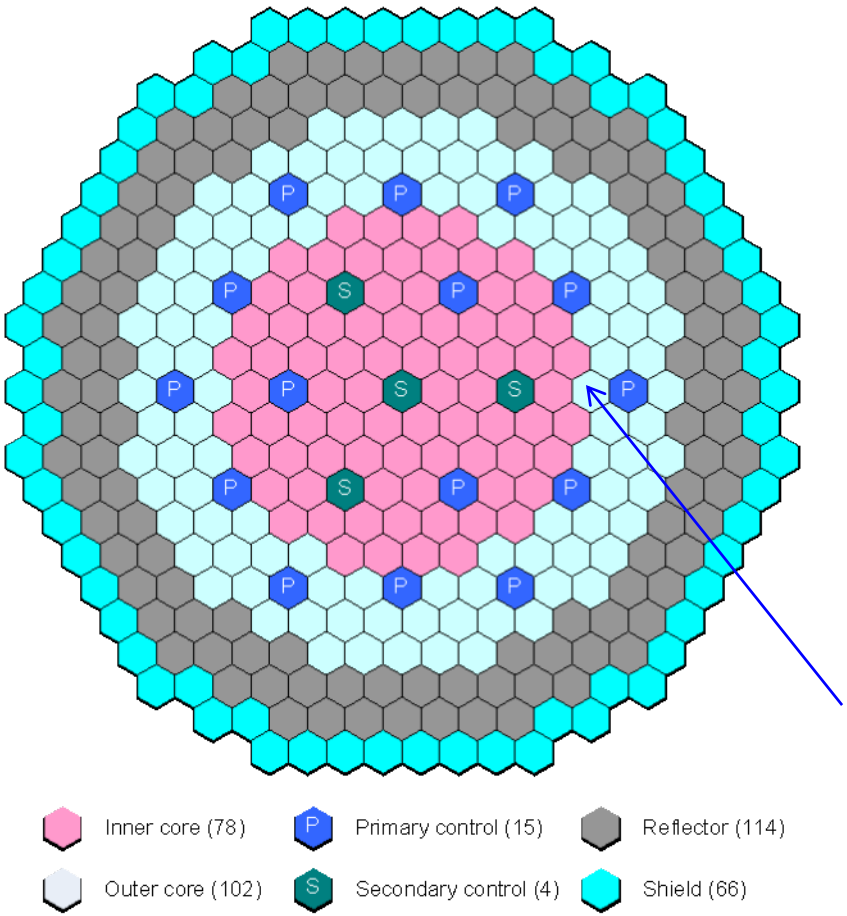
		ANL		CEA		Difference (%)	
Ring number	position number	BOC	EOC	BOC	EOC	BOC	EOC
1	1	0.1	0.1	0.0	0.0		
2	1	6.1	6.6	6.3	6.6	2.7%	0.4%
2	2	6.1	6.6	6.3	6.6	2.7%	0.4%
3	3	6.1	6.5	6.2	6.5	2.0%	0.1%
3	4	6.1	6.5	6.2	6.5	2.4%	0.4%
4	1	0.1	0.2	0.0	0.0		
4	2	6.1	6.4	6.2	6.4	1.5%	-0.1%
4	3	6.1	6.4	6.2	6.4	1.5%	-0.1%
5	1	6.5	6.5	6.5	6.5	0.6%	-0.1%
5	2	6.6	6.7	6.7	6.7	1.0%	0.0%
5	3	6.6	6.8	6.7	6.8	1.2%	0.2%
5	4	6.6	6.7	6.7	6.7	1.0%	0.0%
6	1	5.8	5.7	5.7	5.7	-0.7%	-0.6%
6	2	6.1	6.1	6.1	6.1	0.0%	-0.2%
6	3	6.2	6.3	6.2	6.3	0.1%	-0.3%
6	4	6.2	6.3	6.2	6.3	0.1%	-0.3%
6	5	6.1	6.1	6.1	6.1	0.0%	-0.2%
7	1	0.1	0.1	0.0	0.0		
7	2	5.3	5.2	5.2	5.2	-1.4%	-0.8%
7	3	5.6	5.6	5.6	5.5	-1.0%	-0.6%
7	4	0.1	0.1	0.0	0.0		
7	5	5.6	5.6	5.6	5.5	-1.0%	-0.6%
7	6	5.3	5.2	5.2	5.2	-1.4%	-0.8%
8	1	4.2	4.1	4.1	4.0	-2.3%	-1.0%
8	2	5.0	4.8	4.9	4.7	-2.5%	-1.3%
8	3	5.6	5.3	5.5	5.3	-1.8%	-0.6%
8	4	4.9	4.8	4.8	4.7	-2.0%	-1.1%
8	5	4.9	4.8	4.8	4.7	-2.0%	-1.1%
8	6	5.6	5.3	5.5	5.3	-1.8%	-0.6%
8	7	5.0	4.8	4.9	4.7	-2.5%	-1.3%
9	1	0.1	0.1	0.0	0.0		
9	2	0.1	0.1	0.0	0.0		
9	3	3.9	3.7	3.8	3.7	-2.9%	-1.3%
9	4	4.3	4.1	4.2	4.0	-3.2%	-1.6%
9	5	4.4	4.2	4.3	4.1	-3.1%	-1.5%
9	6	4.3	4.1	4.2	4.0	-3.2%	-1.6%
9	7	3.9	3.7	3.8	3.7	-2.9%	-1.3%
9	8	0.1	0.1	0.0	0.0		
9	9	0.1	0.1	0.0	0.0		

Comparison of Power Distribution

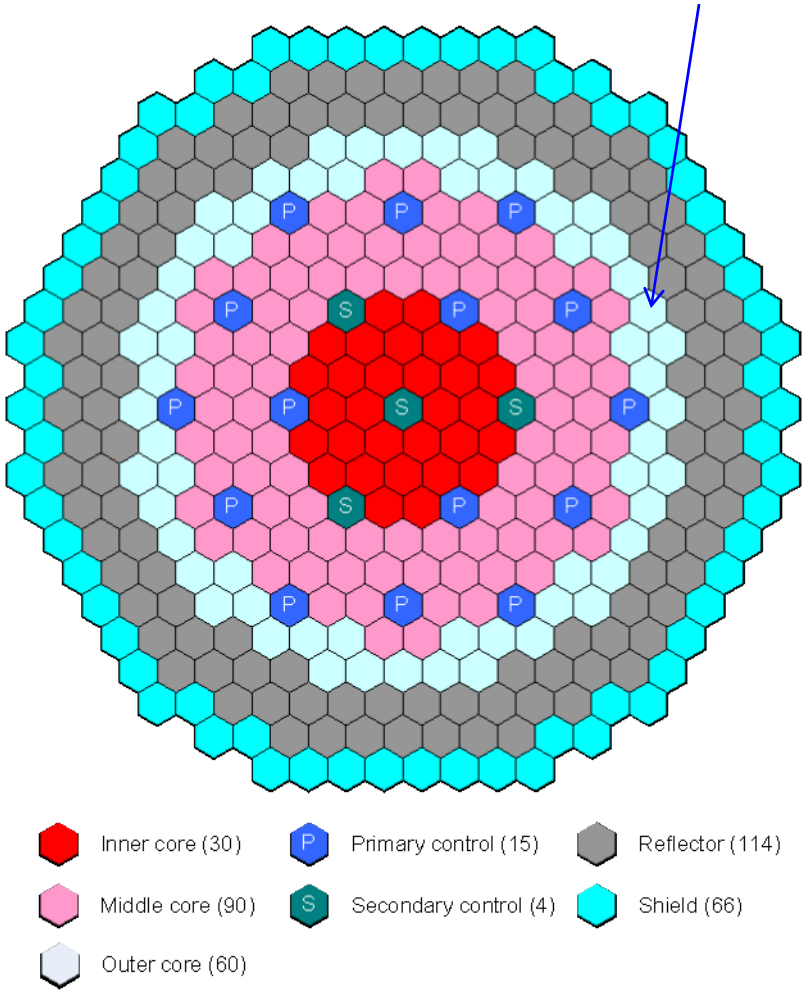
	Medium Metallic		Medium Oxide	
	BOC	EOC	BOC	EOC
RMSE (MW)	0.28	0.23	0.10	0.04
Peak location	(6,1)	(6,1)	(9,4)	(9,4)
Ave. error,%	2.5%	2.2%	1.7%	0.7%

- **Definitions**
  - RMSE (Root Mean Square Error) = $\sqrt{\frac{1}{n}\sum_i \left(p_A^i - p_C^i\right)^2}$

Location of Peak Power Differences



Metal Core



Oxide Core

Conclusions and Potential Future Work

- **Four organizations (ANL, CEA, KTH/UIUC, and AEKI/KFKI) have participated in Medium size SFR benchmark**
  - Wide variation in core multiplication factor was observed
    - BOC keff: metallic core of 0.982 – 1.041, oxide core of 0.966 – 1.0353
    - Reactivity swing: metallic core of 1.7% - 2.7% Δk, oxide core of 0.6 – 2.7% Δk
  - Effective delayed neutron fraction is generally similar within the range 0.030 – 0.0035
- **Comparison of ANL and CEA results**
  - ANL calculated higher sodium void worth and control rod worth, but lower Doppler constant
  - Core inventories are similar, except for higher minor actinides (in particular, Cm)
  - RMSE of power distribution is about ~0.2 MW for metallic core, and ~0.1 MW for oxide core, which are about 3% and 2% differences on average
- **Detailed analysis is required to identify differences, maybe with more data from additional participants**
  - Potential reasons can be due to different interpretations of benchmark, methods/assumptions, power evaluation scheme, etc.

SFR Benchmark Schedule (by F. Varaine)

- **T0: May 2011**
  - Initiate expert group activities
  - Provide core data for neutronic calculation
- **T0 + 6 months : Dec. 2011**
  - Presentation of neutronic evaluations
  - Discussion and agreement on feedback coefficients
- **T0 + 12 months : Jun. 2012 (***ANS/ICAPP meeting at Chicago***)**
  - Presentation of feedback coefficients
  - Discussion and agreement on transient to be calculated
- **T0 + 18 months : Dec. 2012**
  - Presentation and discussion of transition results
  - Discussion of final report
- **T0 + 24 months : Jun. 2013**
  - Final report and recommendation for further activities

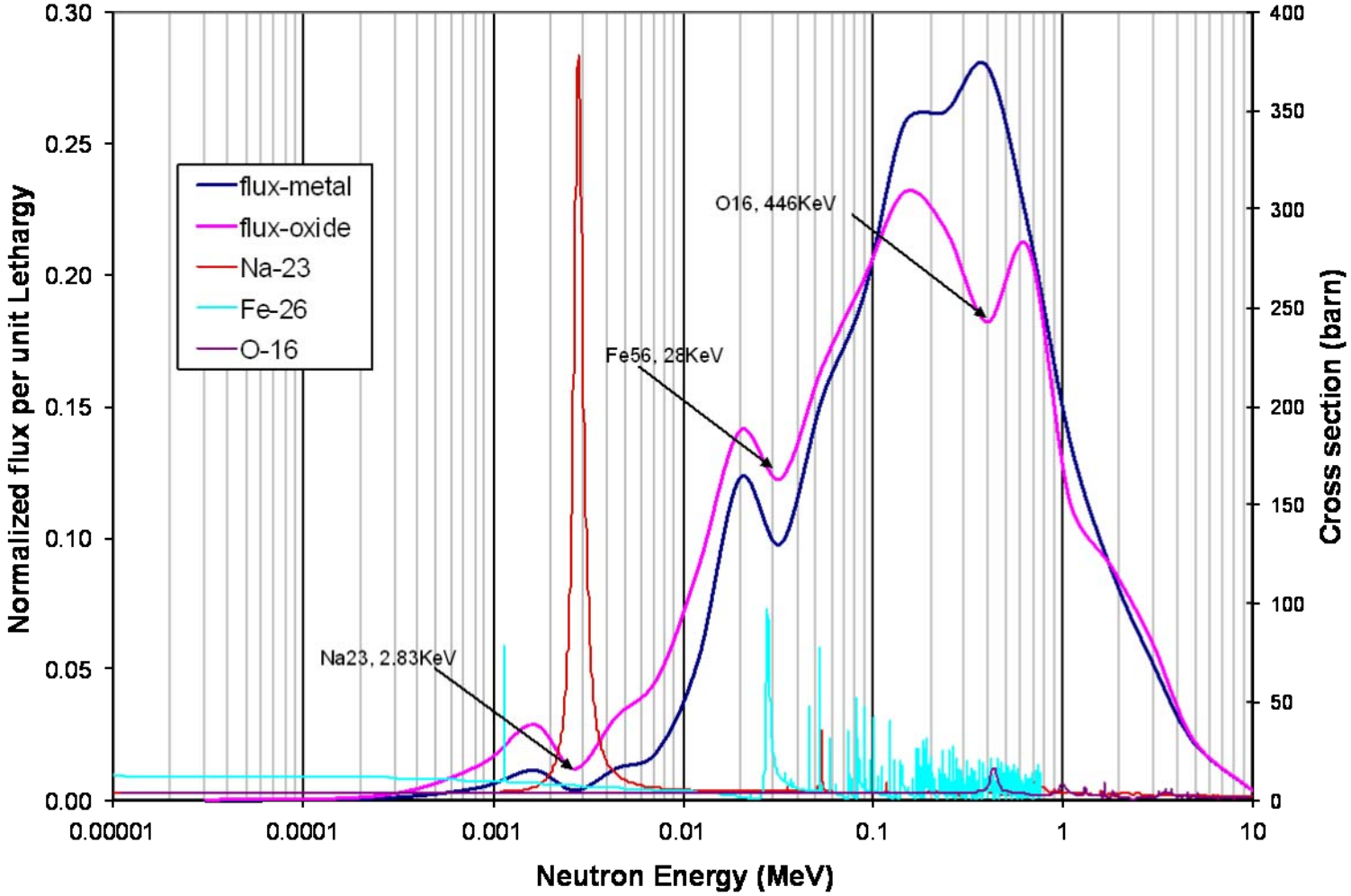
Backup

- **Informational Note for Reactivity Feedback Calculation in ANL**

Kinetics Parameters and Reactivity Feedback Coefficients

- **Compute kinetics and reactivity feedback coefficients using perturbation theory code (VARI3D) and steady-state diffusion equation solver (DIF3D)**
- **Compute following parameters using VARI3D because neutron importance and spatial distributions are required for safety analysis**
  - Effective delayed neutron fraction and prompt neutron lifetime
  - Coolant sodium density coefficient
  - Coolant sodium void coefficient
  - Fuel and structural material worth
  - Doppler coefficient
- **Compute following parameters using DIF3D because integral parameters are important**
  - Uniform radial expansion
  - Uniform axial expansion
  - Control rod driveline expansion

Typical Neutron Spectrum and Cross Sections of Major Non-Actinide Nuclides



VARI3D – Perturbation Code

- **VARI3D solves adjoint equation of multi-group finite difference diffusion equation**
  - Compute effective delayed neutron fraction (β_{eff}) and other kinetic parameters

$$[\mathbf{L}-\lambda\mathbf{F}]\phi=0 \quad \text{vs.} \quad [\mathbf{L}-\lambda\mathbf{F}]^{\mathrm{T}}\phi^{*}=0$$

- **VARI3D solves first-order and exact perturbation equations**

$$\delta\lambda \cong \frac{\phi^{*\mathrm{T}}[\delta\mathbf{L}-\lambda\delta\mathbf{F}]\phi}{\phi^{*\mathrm{T}}\mathbf{F}\phi} \quad \text{vs.} \quad \delta\lambda = \frac{\phi_p^{*\mathrm{T}}[\delta\mathbf{L}-\lambda\delta\mathbf{F}]\phi}{\phi_p^{*\mathrm{T}}\mathbf{F}_p\phi}$$

  - Compute integral and local reactivity feedback coefficients
  - Compute sensitivity parameter of bilinear weighted reaction ratios

$$P=\frac{\phi^{*\mathrm{T}}\mathbf{H}\phi}{\phi^{*\mathrm{T}}\mathbf{G}\phi}$$

a) VARI3D Reference

Kinetics Parameters

- **Effective Delayed Neutron Fraction (β_{eff})**
  - β_{eff} accounts for importance of delayed neutron's energy and spatial distribution
  - β_{eff} is less than β in fast reactor due to lower energy of delayed neutron
  - β_{eff} depends on fissile (Pu vs. U), fuel form (metal vs. oxide), core configuration
    - β: Pu239=0.0022, U238=0.0158, U235=0.0068, Pu241=0.0052, MA<0.002
  - β_{eff} of conventional U-Pu fueled core is 0.003 – 0.004
    - Metallic core value is slightly higher than for oxide core (due to lower TRU mass fraction)
    - Homogeneous core value is higher than for heterogeneous core (due to higher U238 fission)
    - β_{eff} decreases as MA content increases
  - β_{eff} of U fueled core is 0.006 – 0.007, but it decreases as burnup increases
- **Prompt Neutron Lifetime (Λ)**
  - Prompt neutron lifetime is 0.3 – 0.4 µsec, depending on neutron spectrum
    - Metallic core has shorter lifetime compared to oxide code due to harder spectrum

Comparison of Core Performance

	SPRISM [a] (metal-BE)	ABR1000 [b]		SFR3500 [c] (metal)
		Metal	Oxide	
Core Configuration	heterogeneous	homogeneous	homogeneous	heterogeneous
Cycle Length, month	23	12	12	12
Number of batches	4	4	3	3
Average TRU mass fraction, %	~11	~22	~28	~14
Fissile conversion ratio	1.05	0.84	0.82	1.13
Average discharge burnup (GWd/t)	106	93	132	80
Peak fast fluence, $x10^{23}$ n/cm^2	3.9	4.1	4.1	3.1
Burnup reactivity swing, %Δk	0.1	3.2	2.6	-0.6
Average linear power, kW/m	18.9	23.3	17.6	N/A
Effective delayed neutron fraction		0.0033	0.0030	0.0035
Prompt neutron lifetime, μs	N/A	0.33	N/A	0.32

a) A. E. Dubberley, et al, *Superprism Oxide and Metal Fuel Core Designs*, ICONE 8, April 2-6, 2000, Baltimore, MD USA.
b) T. K. Kim, et al, ore design studies for a 1000 MWth Advanced Burner Reactor, Annals of Nuclear Energy 36 (2009) 331–336.
c) E. K. Fujita, The Neutronics and Fuel Cycle Performance of Interchangeable 3500 MWth Metal and Oxide Fueled LMRs, ANL-FRA-163, Argonne National Lab. March, 1989

Geometric Expansion Coefficients

- **Radial expansion – uniform expansion of grid plate**
  - Reduce fuel/structure densities uniformly
  - Radial expansion allows more leakage of neutrons axially
- **Axial fuel expansion – uniform expansion of fuel (depending fuel depletion behavior)**
  - Increase fuel length and reduce fuel density uniformly, which results in insertion of control rod
  - Axial expansion allows more leakage of neutrons radially
- **These feedbacks are important for fast reactor transient behavior**
  - Tied to different material temperatures (grid plate, fuel)
  - Thus, time constants are different

Neutron Balances for Radial and Axial Expansions

	Base case	Radial Expansion		Axial expansion	
	balance	balance	Δρ (%)	balance	Δρ (%)
Fission source	100.00	100.00		100.00	
(n,2n) source	0.18	0.18		0.18	
Absorption	68.64	68.68	-0.04	68.81	-0.17
Leakage	31.54	32.16	-0.62	31.69	-0.15
- Radial	17.49	17.71	-0.22	17.65	-0.16
- Axial	14.05	14.45	-0.40	14.04	0.01
Sum			-0.66		-0.32

- Axial leakage is dominant in radial expansion, but radial leakage is dominant in axial expansion
- Axial expansion gives absorption effect from control rod insertion
- Radial expansion coefficient is more negative in conventional SFR with shorter core height

Coolant Reactivity Feedback Coefficients

- **Spectral effect – positive reactivity effect**
  - Spectrum hardens as sodium density decreases, which increases fission yield neutrons from Pu-239 and threshold fission from major fertile
- **Leakage effect – negative reactivity effect**
- **Sodium capture effect – positive but small**
- **Coolant density coefficient**
  - computed by *first-order perturbation theory* to evaluate local coolant density change (for instance, due to temperature variation)
- **Coolant void worth**
  - evaluated using *exact perturbation theory* to account for spectrum change in coolant voided condition
  - In general, spectral effect of voided worth is ~10% more positive than that of coolant density worth

Coolant Void Worth by Components ($)

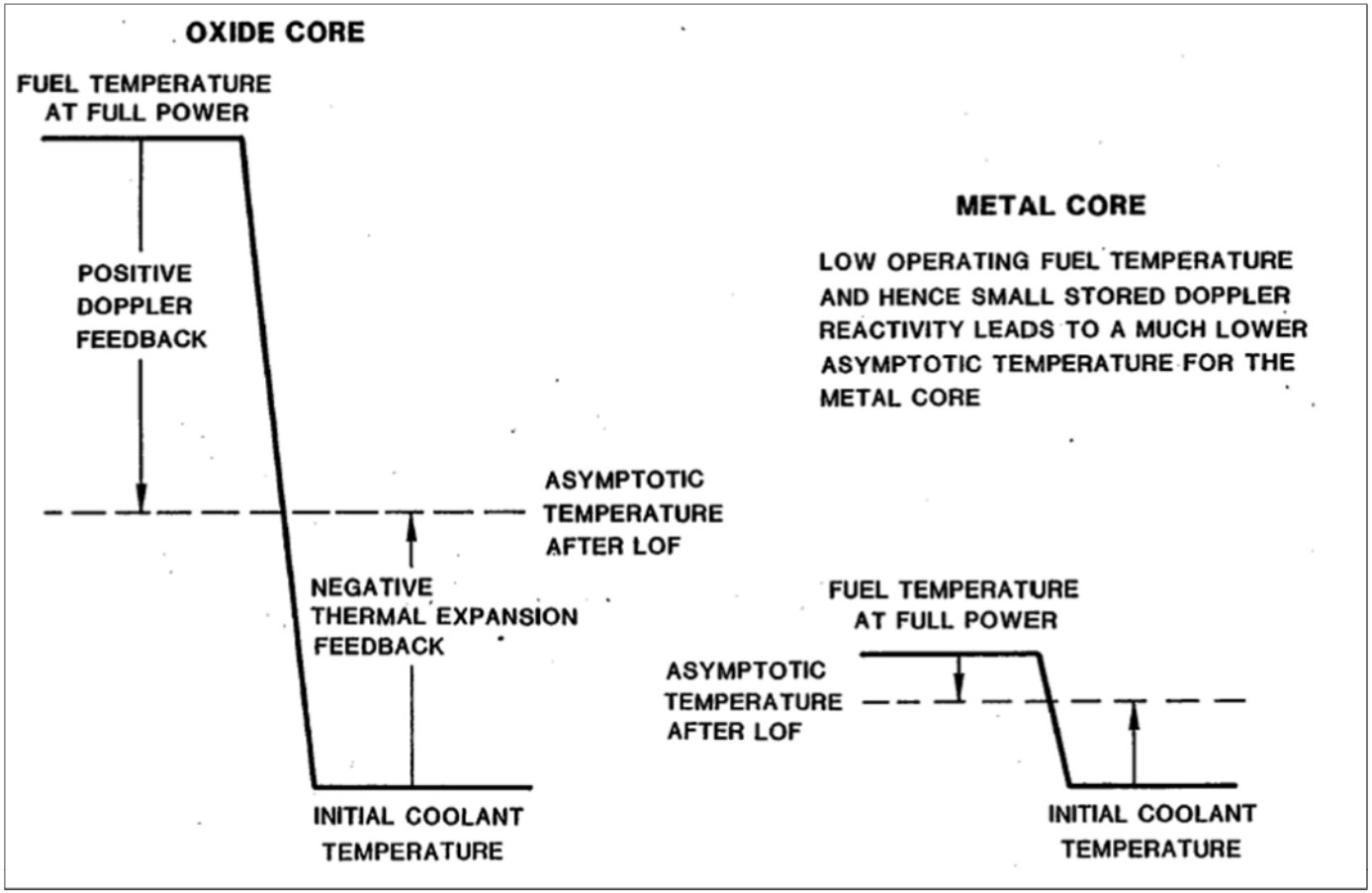
SFR size	Capture	Spectral	Leakage	Total
Medium size	0.5	9.1	-5.2	4.4
Small size	0.4	6.4	-5.8	1.0

- **Flowing coolant sodium voided in active and above-core regions**
- **Coolant void worth tends to increase with core size**
- **However, difficult to conceive transient situations that reach sodium boiling**
  - Low pressure system
  - More than 300°C margin to boiling
  - Overall reactivity feedback could be negative
  - Extensive report on void worth reduction – Khalil and Hill, NSE, 109 (1995)

Doppler Coefficient

- **Doppler coefficient arises primarily from U-238 resonance broadening, which is located at low energy tail (keV) of fast neutron spectrum**
- **Doppler coefficient depends on fuel temperature, U-238 content, and neutron spectrum softening**
  - Doppler effect enhanced by high U-238 content
  - Doppler enhance by spectral softening (i.e., voided Doppler is smaller than nominal Doppler)
  - For typical FR, Doppler effect has approximate 1/T dependence
- **There is also a structural Doppler reactivity effect (~1/3 fuel Doppler)**
  - However, tied to temperature of steel, not fuel (different timing)
- **Doppler feedback (which is always negative) is not helpful in all transients**
  - For example, when trying to cool the fuel to shutdown condition (e.g., ULOF), it is a positive feedback

Schematic View of Asymptotic Temperature After LOFWS in Oxide and Metal Cores



a) Y. I. Chang, Technical Rationale for Metal Fuel in Fast Reactor, Nuclear Engineering and Technology, Vol.39 No.3 June 2007

Quasi-Static Reactivity Balance with Integral Reactivity Parameters

- **Method for evaluating quasi-static fast reactor reactivity balance during Anticipated Transients Without Scram (ATWS) was developed**
  - D. Wade, E. Fujita, *Trends versus Reactor Size of Passive Reactivity Shutdown and Control Performance,* Nuclear Science and Engineering, 103 (1989).
  - Quasi-static fast reactor reactivity balance

$$\delta\rho = \mathbf{A}\left[P(t)-1\right] + \mathbf{B}\left[\frac{P(t)}{F(t)} - 1\right] + \mathbf{C}\delta T_{in}(t) + \delta\rho_{ext}$$

$P(t)$ = normalized reactor power

$F(t)$ = normalized coolant flow rate

$\delta T_{in}(t)$ = change in coolant inlet temperature

$\delta\rho_{ext}$ = externally applied changed in reactivity

$\mathbf{A}, \mathbf{B}, \mathbf{C}$ = integral reactivity parameters

  - Integral reactivity parameters (A, B, and C) are associated with reactivity feedback coefficients, core configurations, etc.

Inherent (Passive) Safety Criteria

- **Sufficient conditions for acceptable asymptotic core outlet temperature**

$$\frac{\mathbf{A}}{\mathbf{B}} \le 1, \quad 1 \le \frac{\mathbf{C}\Delta T_c}{\mathbf{B}} \le 2, \quad \frac{\Delta\rho_{TOP}}{|\mathbf{B}|} \le 1$$

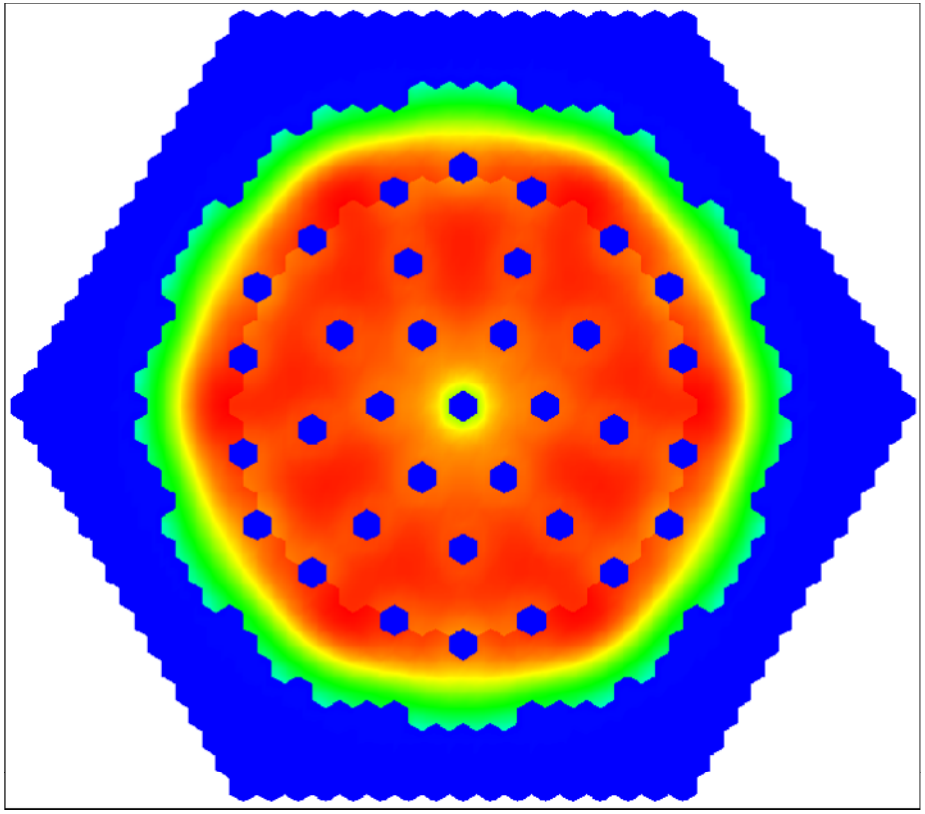
  - Evaluating these criteria gives indication for favorable inherent safety features of design concept although detailed safety analyses are required to confirm performance and margins

		ABR1000 [a)]	
		Metal Core	Oxide Core
A		-11	-73
B		-56	-41
C		-0.5	-0.4
Sufficient conditions	A/B ≤1	0.2	1.8
	1≤C ΔT/B ≤2	1.3	1.4
	Δρ/|B| ≤2	0.5	0.8

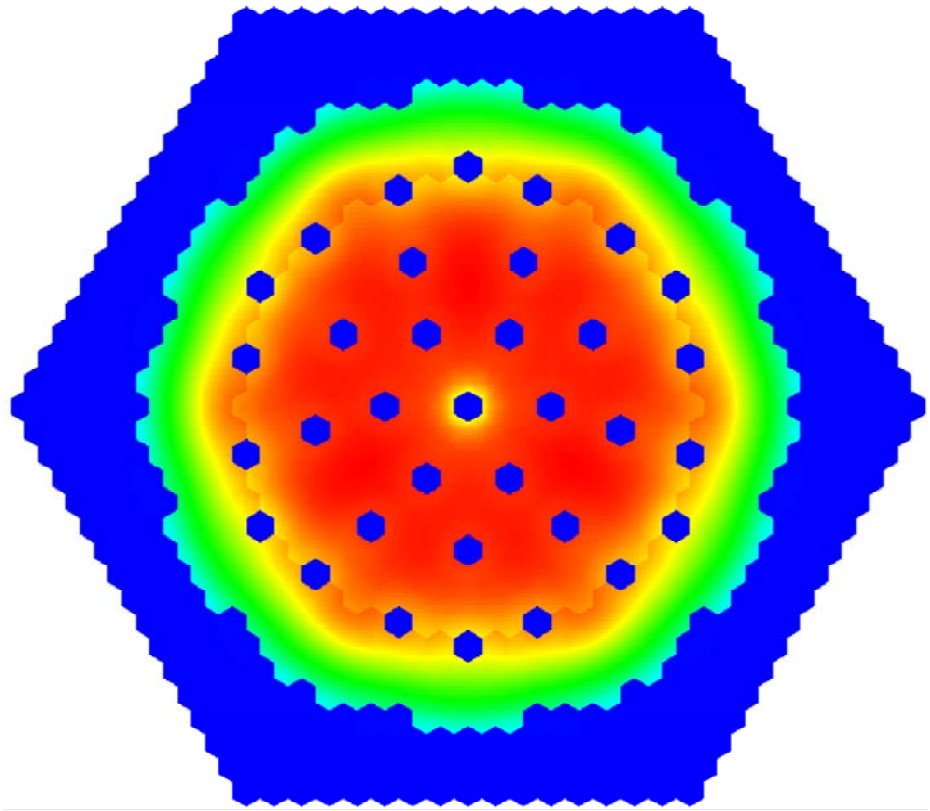
  - Above oxide core does not meet all conditions for unprotected accidents scenarios because of high fuel temperature and large Doppler feedback (additional design features are required to increase inherent safety margins).

a) T. K. Kim, et al, ore design studies for a 1000 MWth Advanced Burner Reactor, Annals of Nuclear Energy 36 (2009) 331–336.

Power Distributions – Large Size Oxide Core



BOC



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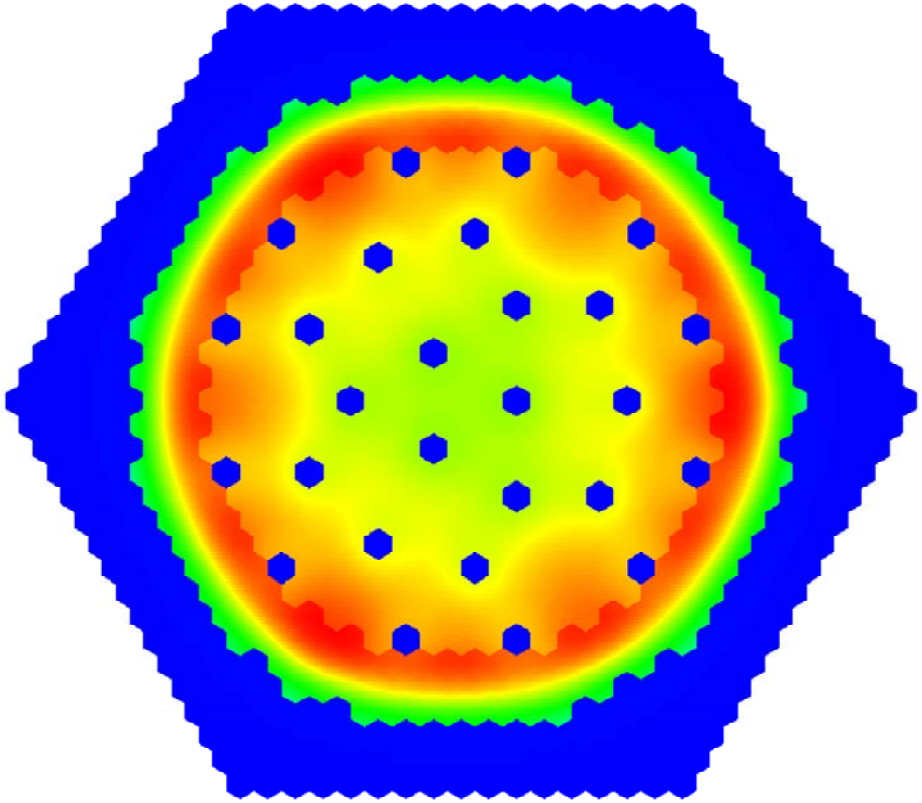
Power Distributions – Large Size Carbide Core



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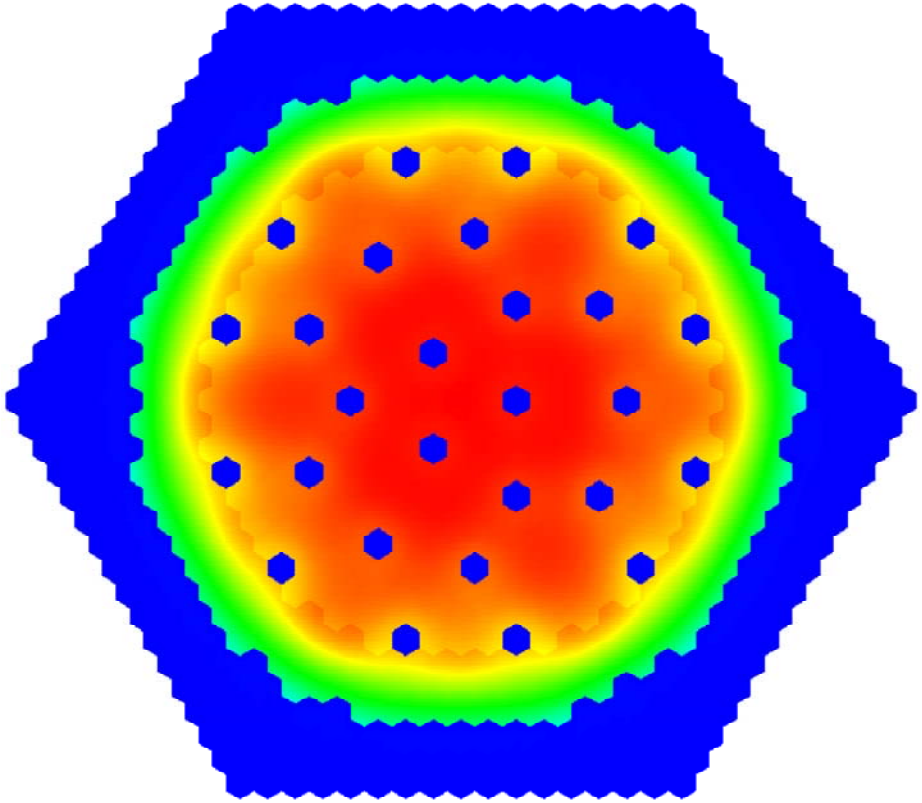

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