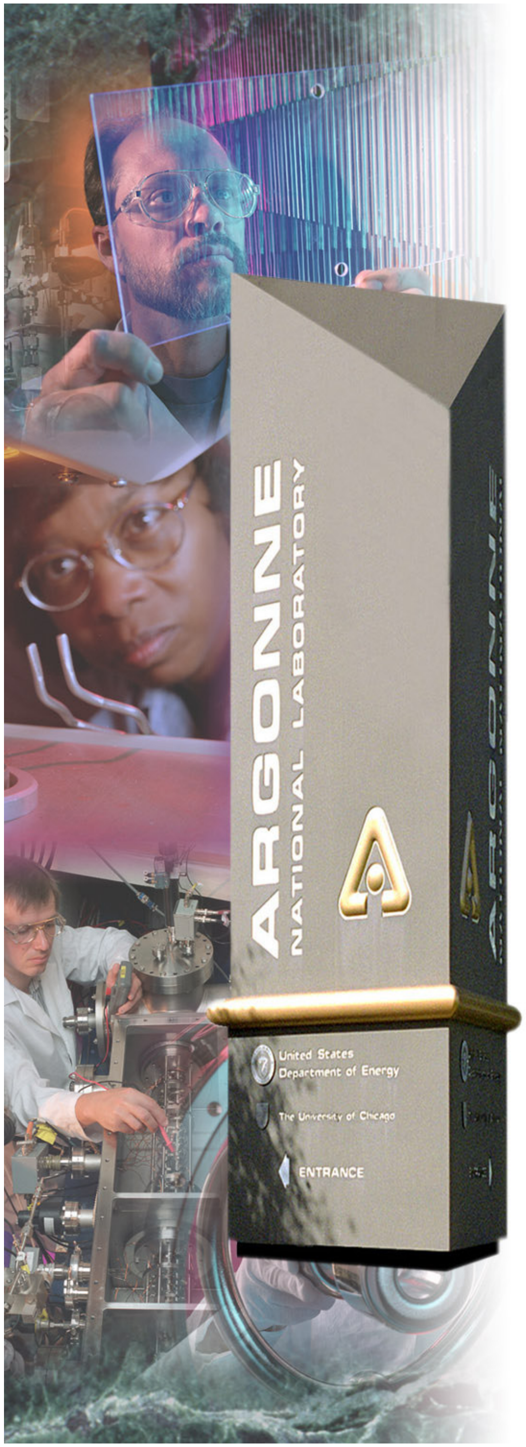

Advances In Reactor Physics Analysis Needed for Design of Generation IV Systems

Presented at

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by

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Office of Science
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Topics

- Generation IV background
- Features of Gen IV systems
- Physics analysis issues
- Status of current capabilities
- Priorities for future development

Generation IV

Nuclear energy systems deployable no later than 2030 in both developed and developing countries, for generation of electricity and other energy products

Generation I

Early Prototype Reactors

- Shippingport
- Dresden, Fermi I
- Magnox

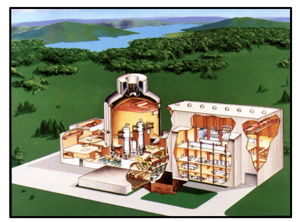
Generation II

Commercial Power Reactors

- LWR-PWR, BWR
- CANDU
- VVER/RBMK

Generation III

Advanced LWRs



- ABWR
- System 80+
- AP600
- EPR

Near-Term Deployment

Generation III+ Evolutionary Designs Offering Improved Economics

Generation IV

- Highly Economical
- Enhanced Safety
- Minimal Waste
- Proliferation Resistant

Gen I | Gen II | Gen III | Gen III+ | Gen IV

1950 | 1960 | 1970 | 1980 | 1990 | 2000 | 2010 | 2020 | 2030

Role of Physics Analysis

Economics	• Compact core and shield configuration • Minimum fissile and reactivity control requirements • Fuel management optimization (e.g. cycle length)
Sustainability	• Achievement of high average burnup (resource utilization for once through cycle) • Waste characteristics (toxicity, decay heat)
Safety & Reliability	• Verifying fission power & decay heating within heat removal capabilities • Excess reactivity minimization, favorable reactivity Δ's for temperature and material density changes
Proliferation Resistance & Physical Protection	• Accurate accounting for life-cycle flows of fissile materials into and out of reactor • Isotopic makeup (weapons attractiveness)

Generation IV International Forum

- **Ten nations are cooperating to advance Generation IV nuclear energy systems**
  - Argentina, Brazil, Canada, France, Japan, Korea, South Africa, Switzerland, UK, US
  - International organizations (Euratom, OECD-NEA, and IAEA) also participate
- **GIF Charter (July 2001)**
  - Establish and foster collaborations on future nuclear energy systems
  - Establish guidelines for collaborative R&D projects and reporting of their results
  - Regularly review progress and set future directions
- **Current GIF focus**
  - Agreements for multilateral collaborations
  - Joint R&D projects, based on Generation IV Roadmap

Generation IV Systems

System	*Neutron Spectrum*	*Fuel Cycle*	*Size*	*Applications*	*R&D*
Very High Temp. Gas Reactor (VHTR)	Thermal	Open	Med	Electricity, Hydrogen Production, Process Heat	Fuels, Materials, H_2 production
Supercritical Water Reactor (SCWR)	Thermal, Fast	Open, Closed	Large	Electricity	Materials, Safety
Gas-Cooled Fast Reactor (GFR)	Fast	Closed	Med to Large	Electricity, Hydrogen, Actinide Management	Fuels, Materials, Safety
Lead-alloy Cooled Fast Reactor (LFR)	Fast	Closed	Small	Electricity, Hydrogen Production	Fuels, Materials compatibility
Sodium Cooled Fast Reactor (SFR)	Fast	Closed	Med to Large	Electricity, Actinide Management	Advanced Recycle
Molten Salt Reactor (MSR)	Thermal	Closed	Large	Electricity, Hydrogen Actinide Management	Fuel, Fuel treatment, Materials, Safety and Reliability

Generation IV R&D Focus

- **Thrust of the GIF collaborations is to develop and demonstrate the Gen IV systems**
  - vs. generic R&D
  - Steering committees formed for VHTR(NGNP), SCWR, GFR, and SFR
  - US participates in all, with NGNP receiving highest priority and funding
- **Early emphasis is on resolving key viability questions**
- **Conceptual design development is an integral part of the R&D**
  - Provide focus for technology development (fuels, materials, energy conversion, recycle)
  - Insure compatibility/integration of different technologies
  - Provide basis for evaluating performance

Range of System Characteristics

	VHTR	SCWR	GFR	LFR	SFR
Power, MW_{th}	600-800 (block) ~300 (pebble)	~2000-3600	600, 2400	25-400	800-3500
Power Density, W/cm^3	≤ 6.5	≤ 70	100 (50-200)	25-100	200-400
Primary Coolant (T_{Outlet}, ºC)	He (1000)	SC H_2O (450-500)	He (600-850) SC CO_2	Pb (500-800) Pb-Bi (500-550)	Na (510-550)
Fuel Material	UO_2, $UC_{0.5}O_{1.5}$	UO_2	(U,TRU) carbide, nitride, oxide	(U,TRU) nitride	(U,TRU) oxide, metal alloy
Fuel Form	Triso particle	solid pellet	CerCer dispersion, solid solution, coated particle	solid pellet	pellet or slug
Fuel Element/ Assembly	hex block, pebble	LWR or ACR type pin bundle	hex block, plate, pin, or particle	triangular pitch pin bundle	triangular pitch pin bundle w/duct
Moderator	graphite	water rods (PV) D_2O (PT)	None	None	None
Core Structural Material	graphite	F-M SS, Ni alloy	SiC matrix or cladding, TiN, ODS steel	F-M SS, SiC/SiC composite	ODS ferritic steel

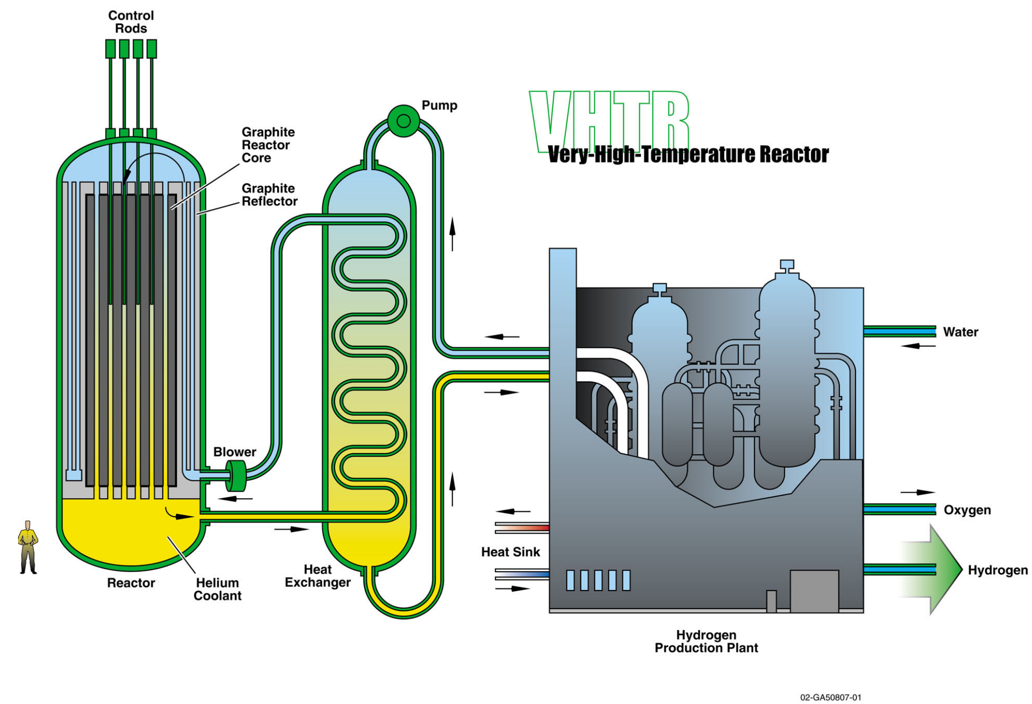
Very-High-Temperature Reactor (VHTR)

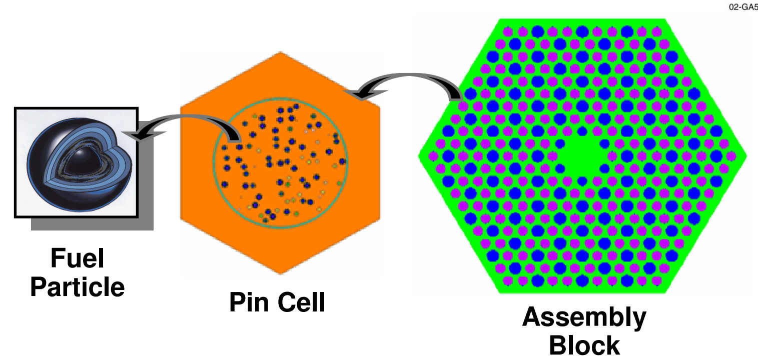
Nominal Characteristics

- ***He coolant, direct cycle***
- ***1000°C outlet temperature***
- ***Prismatic block or pebble bed core***
- ***Coated particle LEU ceramic fuel, dispersed in graphite moderator***
- ***Prismatic block or pebble bed core configuration***

Physics analysis issues

- ***Fuel double heterogeneity***
- ***Stochastic nature of pebble flow (for PBR variant)***
- ***Strong coupling of nuclear and thermal behavior***
- ***Graphite scattering data***
- ***Core/reflector interfacial effect***



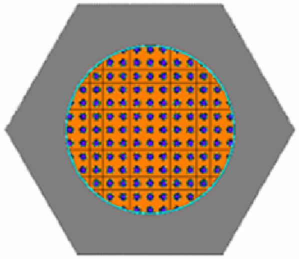


Evaluation of Physics Codes for VHTR Analysis

- **Older code systems (e.g., *VSOP*) available, but there is a need to**
  - Improve upon their modeling fidelity, efficiency and user friendliness
  - Verify and validate predictions to modern standards

- **Review of available benchmark tests is underway**
  - *Critical experiments:* VHTRC (Japan); ASTRA, GROG, RBMK criticals (RF); KATHER (Germany); PROTEUS (Switzerland)
  - *Reactor measurements:* HTR-10 (China); HTTR (Japan); DRAGON (UK); Peach Bottom, FSV, TREAT (US)

- **Comparisons of deterministic and Monte Carlo (MCNP) results also underway**
  - Regular (lattice) distribution of particles typically used to approximate actual stochastic distribution; cuts particles at compact boundary

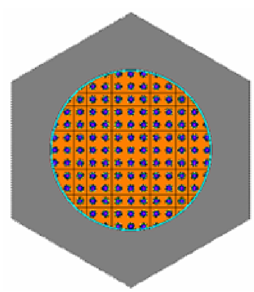


Particle Heterogeneity and Distribution Effects

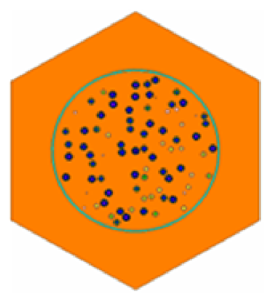
k_∞ Estimates:

			GT-MHR, UO_2	GT-MHR, $(TRU)O_{1.7}$	VHTR, $UC_{0.5}O_{1.5}$
MCNP4C	Random Distribution		**1.57335 ± 0.00040**	**1.25838 ± 0.00040**	**1.53280 ± 0.00082**
	Lattice Distribution	SC	1.57279 ± 0.00039	1.25427 ± 0.00071	1.52978 ± 0.00071
		BCC	1.57118 ± 0.00041	1.25317 ± 0.00070	1.53160 ± 0.00071
		FCC	1.57276 ± 0.00041	1.25192 ± 0.00077	1.52890 ± 0.00073
DRAGON			1.57565 (93)	1.26794 (599)	1.54393 (470)
WIMS8			1.57121 (-86)	1.25326 (-325)	1.52993 (-122)
Double heterogeneity effect			1.4 % $\Delta\rho$	13.1 % $\Delta\rho$	2.3 % $\Delta\rho$

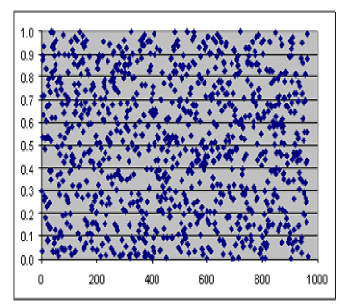
Lattice Distribution



Random Distribution (Planar)



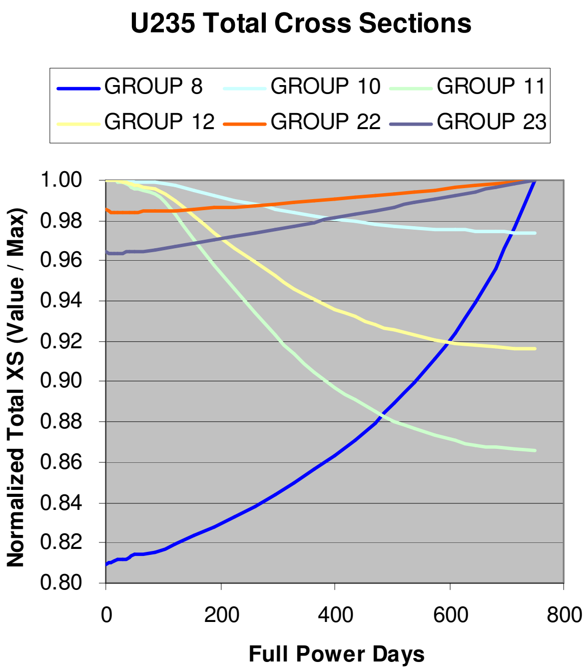
Random Distribution (Axial)

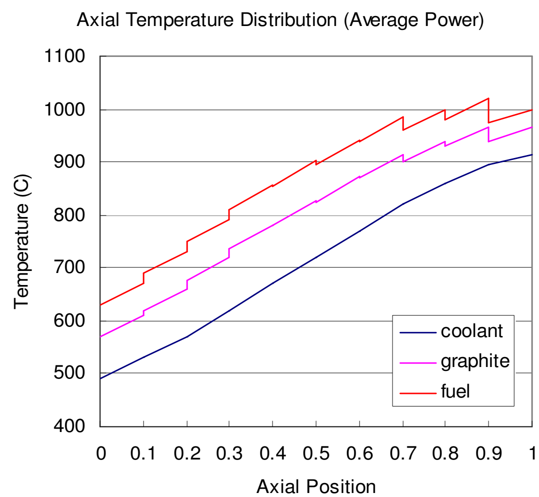


Representation of Effective Cross Sections

Need consistent power and temperature distributions

Effective cross sections depend on burnup and other nodal variables

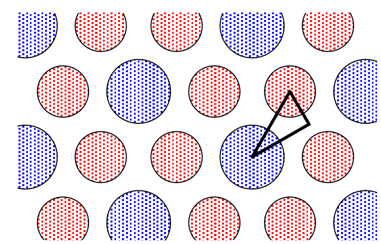


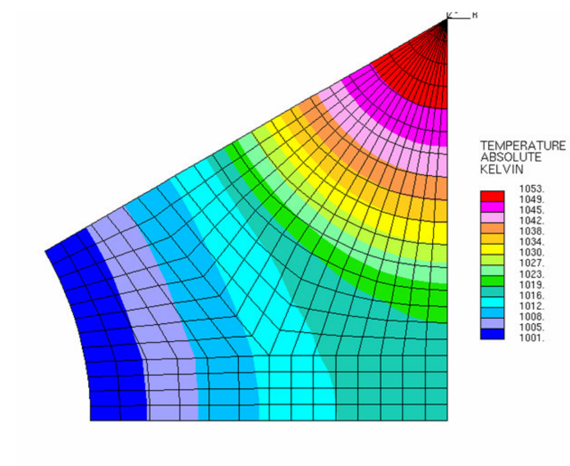


	4 Group		69 Group	
Configuration	A	B	A	B
Eigenvalue	1.3942	1.3943	1.3913	1.3922
Axial Zone	**Power (MWt)**			
1	111.3	89.5	111.8	93.0
2	152.0	135.6	147.3	134.1
3	149.5	152.2	142.8	146.7
4	116.6	134.4	118.4	133.6
5	70.5	88.2	79.7	92.5

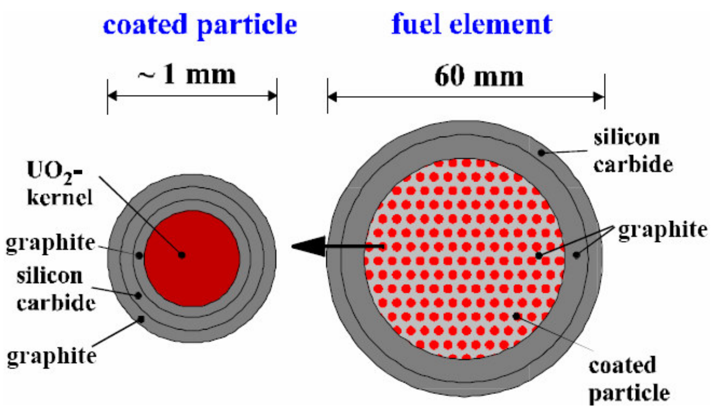
A uses axial temperature distributions
B uses average temperatures

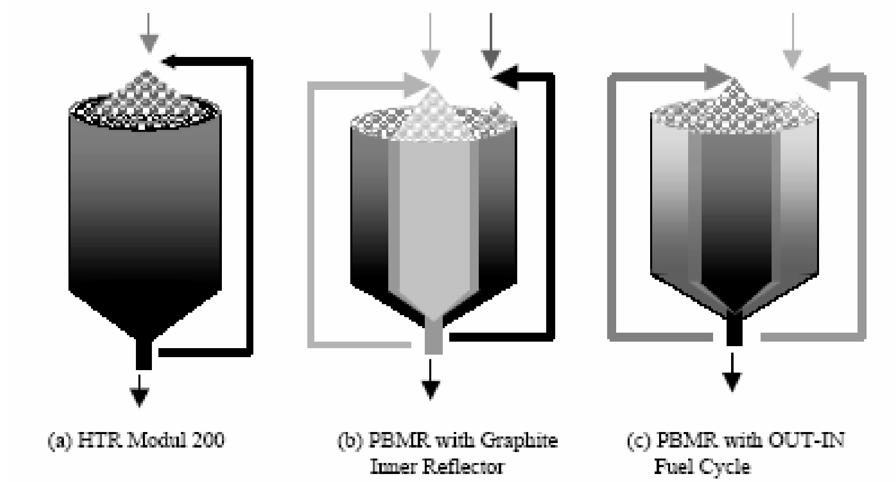
1-D model of unit-cell heat conduction is adequate for static analyses





PEBBED Code for PBR Analysis (INEEL)



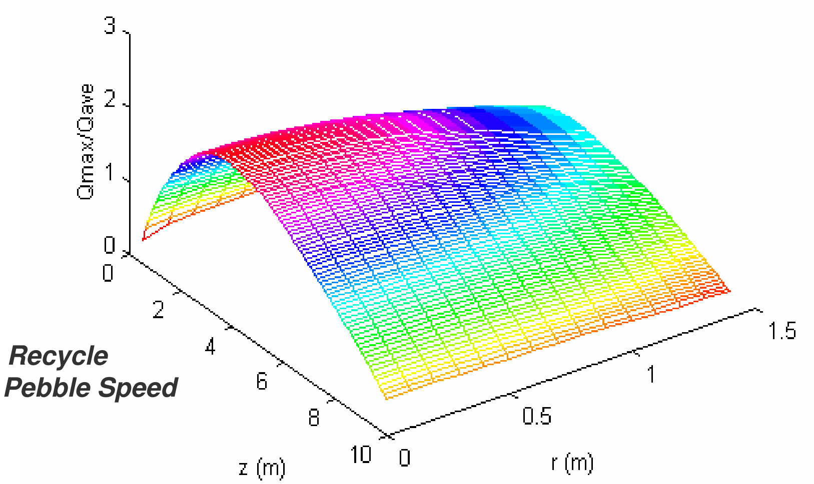


Simultaneous, self-consistent solution to diffusion and nuclide balance equations

- Analytic solution of nuclide density over mesh
- R-Z neutronics solver upgradeable

Pebble recirculation and flow patterns defined by user

Random Recycle
Uniform Pebble Speed



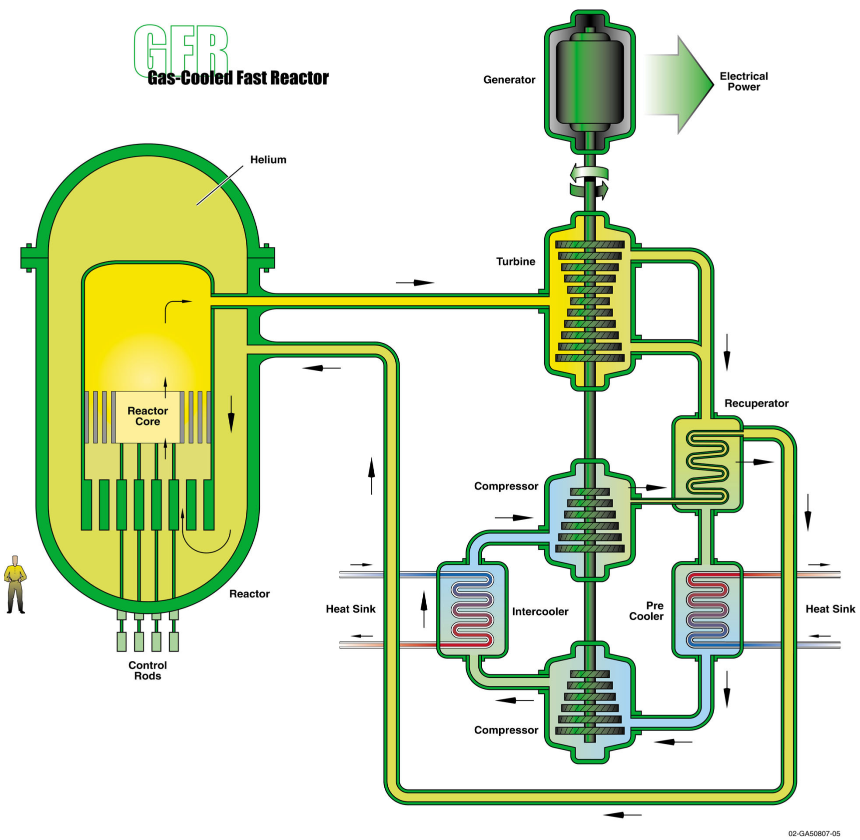
Gas-Cooled Fast Reactor (GFR)

Nominal Characteristics

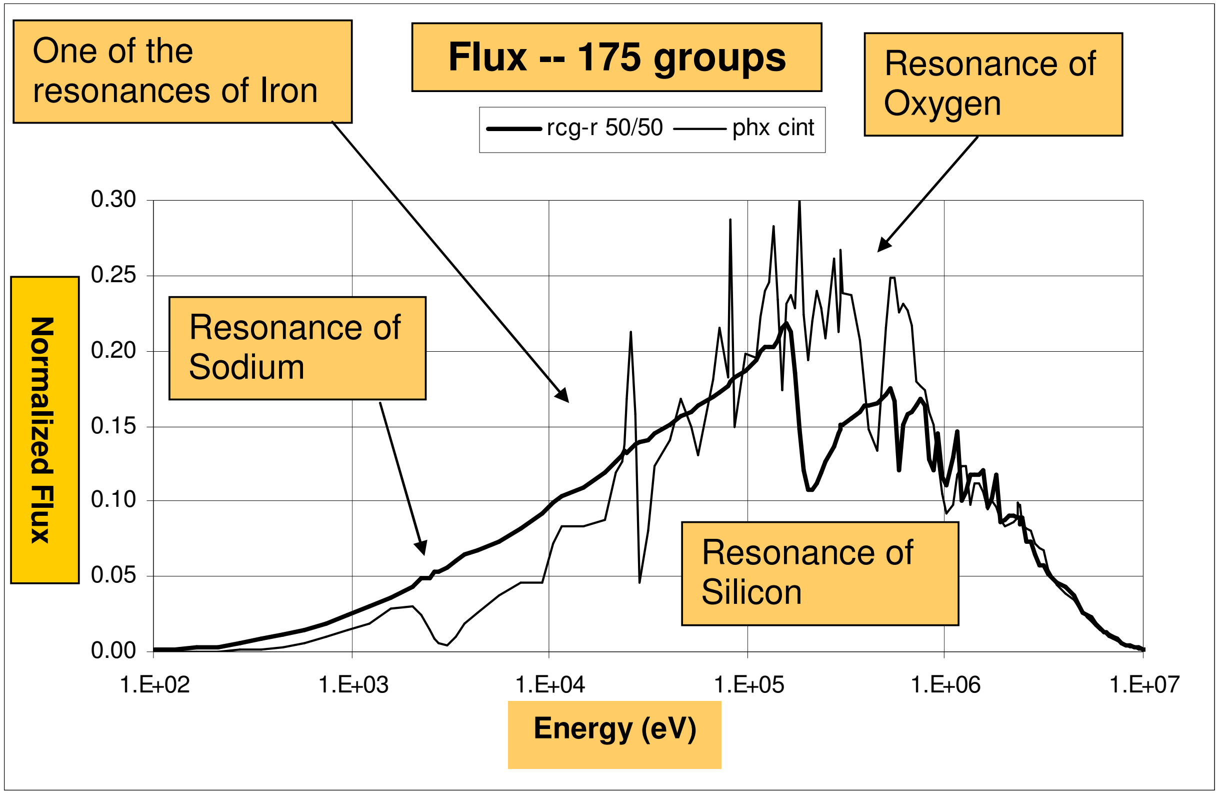
- ***He coolant, 850°C outlet temperature***
- ***600 MWth/288 MWe***
- ***Full TRU recycle***
- ***U-TRU ceramic fuel in CerCer dispersion, solid solution, or coated particle form***
- ***Block, pin, or plate core geometry***
- ***No fertile blanket***

Physics analysis issues

- ***Data for Pu and MA***
- ***Use of unconventional fuel matrix, cladding, and reflector materials***
- ***Neutron streaming***
- ***Spectrum transition near interface between core and reflector***



Comparison of GFR and SFR Spectra (CEA Study)

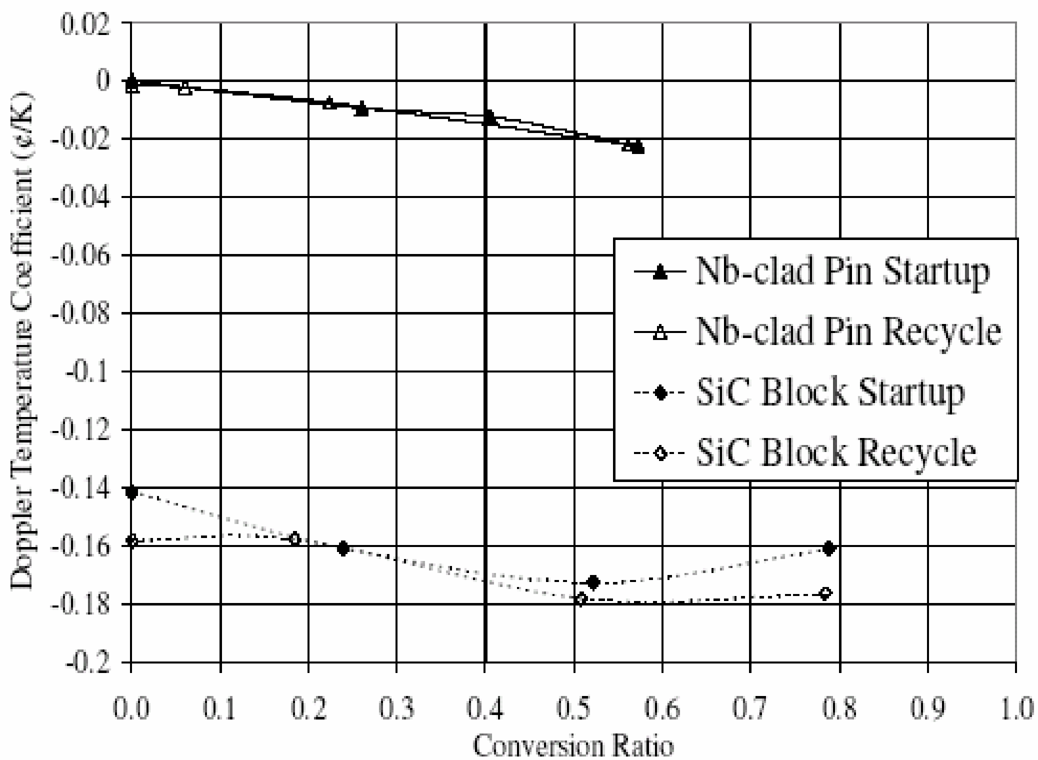


Differences in fuel composition, streaming, reflector properties affect neutron spectrum

Transmutation systems could use degraded Pu

Need integral experiments and re-evaluations of data for matrix, structure, reflector materials

SiC Matrix for Block-type GFR Enhances Doppler Coefficient



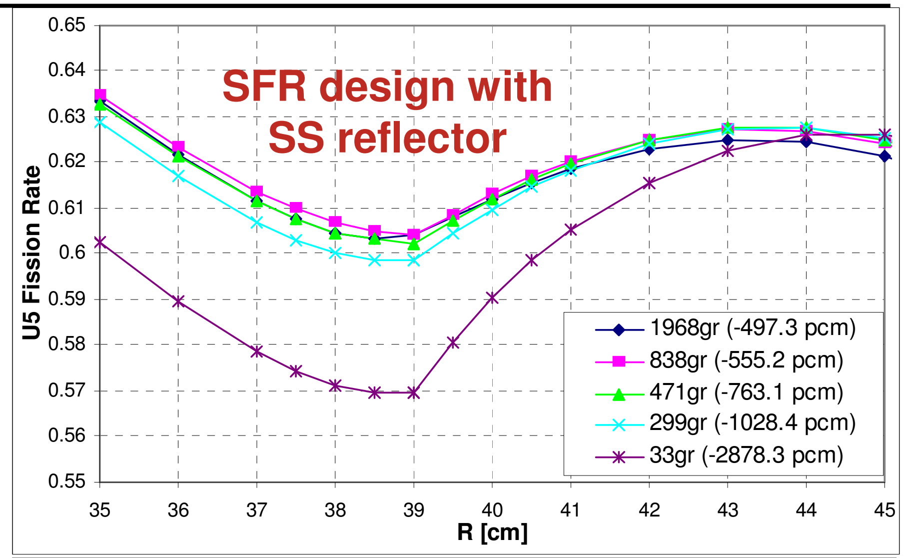
Spectrum is significantly softer with SiC dispersion fuel

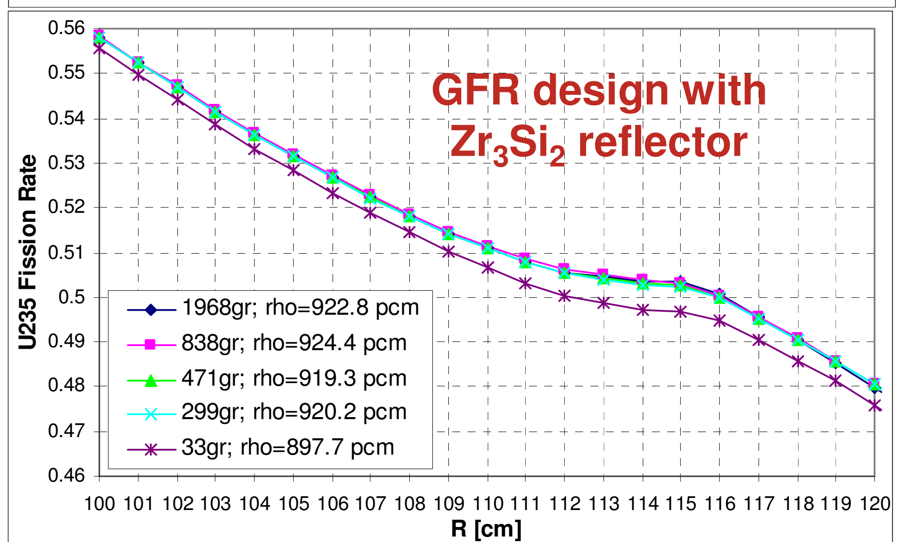
Doppler coefficient strongly negative even for CR = 0

Modeling of Core-Reflector Interface

- **For limited number of broad groups, representation of spectrum transition is particularly important with SS reflector**
  - Detailed macrocell calculation required for interface region

- **Results for GFR with Zr_3Si_2 reflector require further evaluation**
  - Zr and Si resonances above 10 KeV are more isolated than for ^{56}Fe, ^{52}Cr and ^{58}Ni
  - Process resonance XS with more energy groups
  - Compare to Monte Carlo calculations and measurements





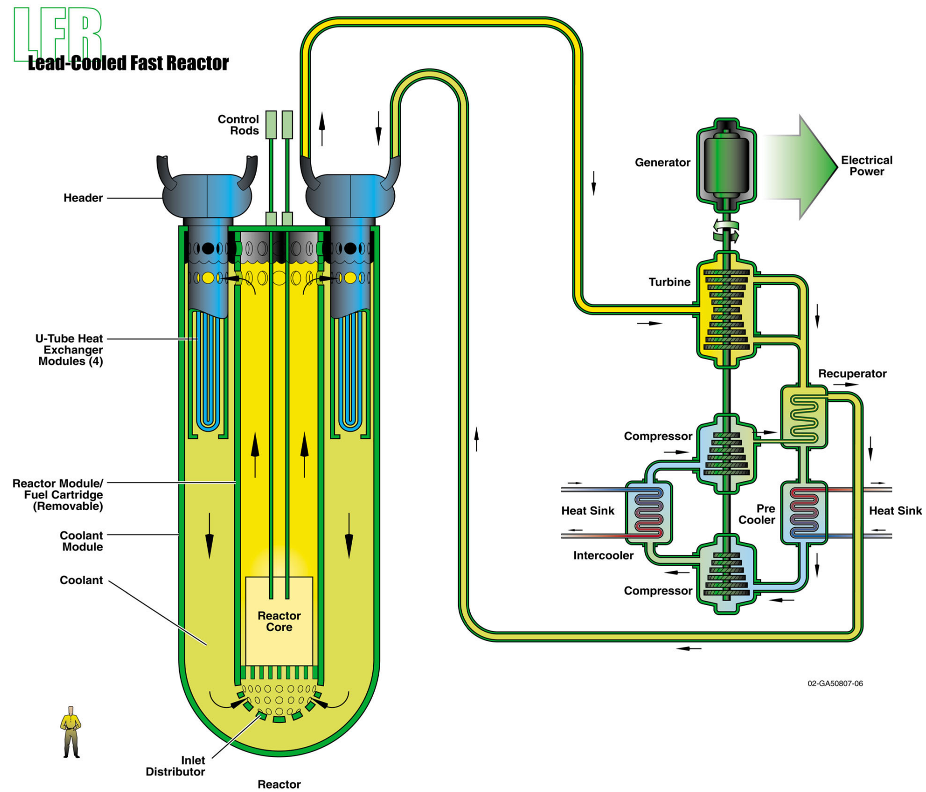
Lead-Cooled Fast Reactor (LFR)

Characteristics

- *Pb or Pb/Bi coolant*
- *500°C to 800°C outlet temperature*
- *Small, transportable reactor*
- *15–30 year core cartridge supplied by regional fuel cycle facility*
- *U-TRU nitride or Zr-alloy fuel pins on triangular pitch*

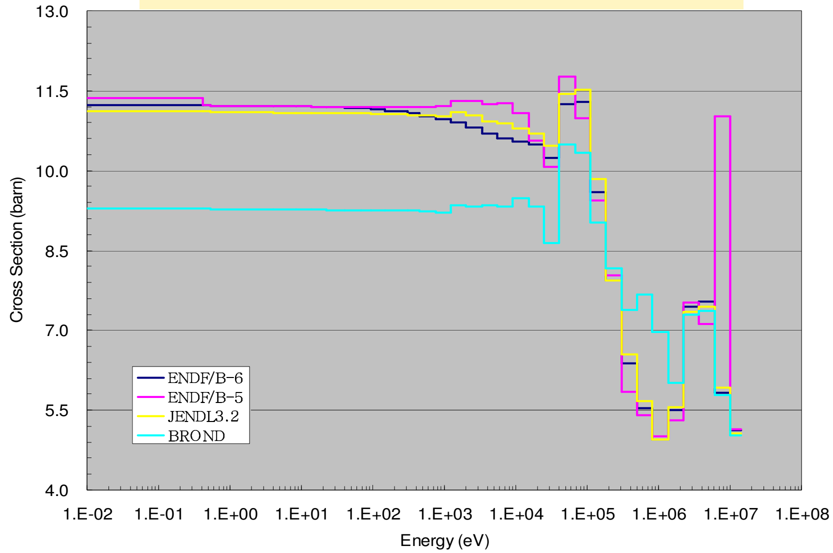
Physics analysis issues

- *Data for actinides, Pb, Bi*
- *Spectrum transition at core edge*
- *Verification of reactivity feedbacks for autonomous load follow*

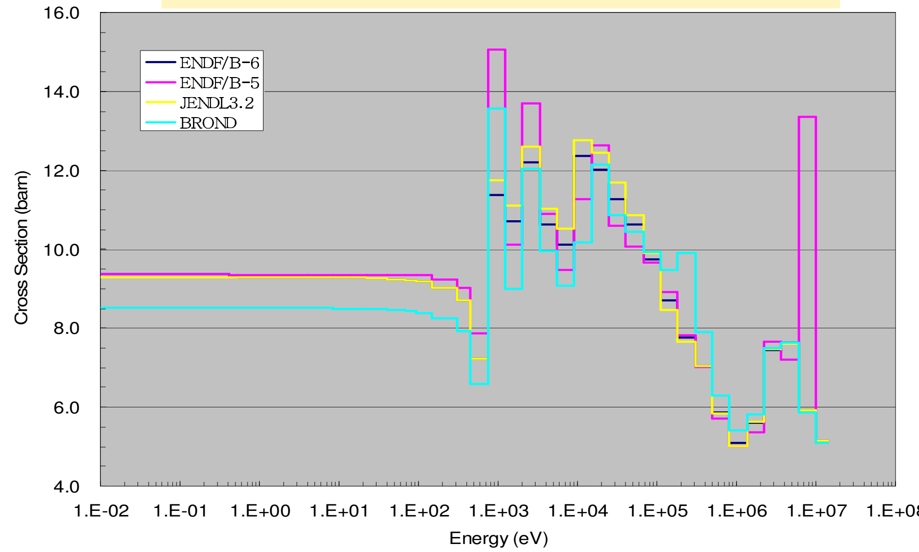


Cross Sections for Pb and Bi

Total Cross Section of Pb

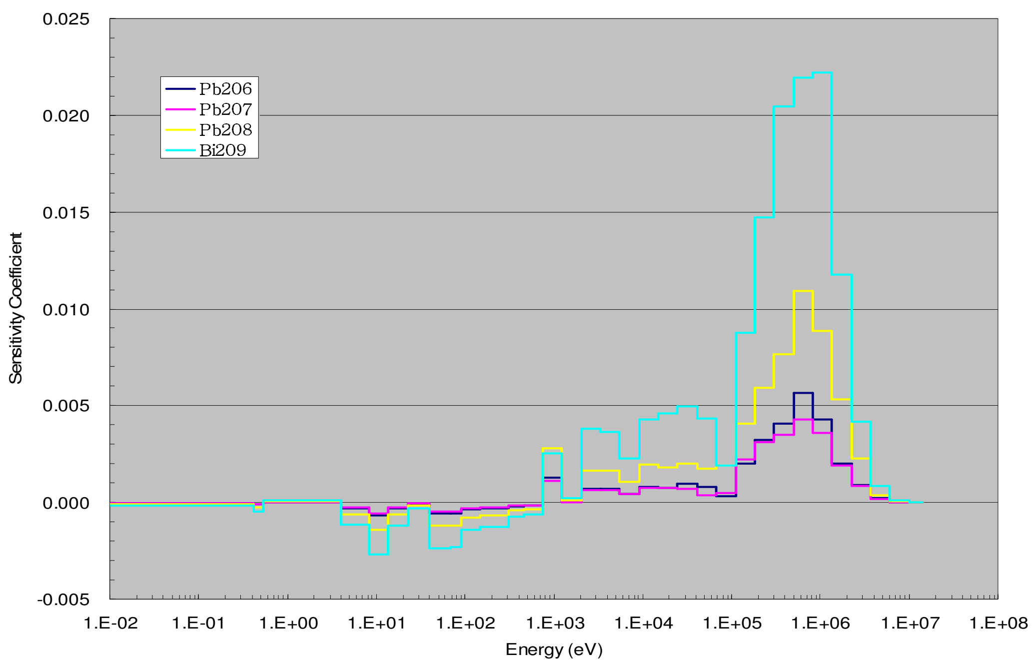


Total Cross Section of Bi209



- **Composition-dependent 33-group data generated using MC^2-2**
  - Actinides and structural material isotopes from ENDF/B-VI
  - Lead and bismuth
    - *ENDF/B-VI: Pb-206, Pb-207, Pb-208*
    - *ENDF/B-V: natural lead*
    - *JENDL-3.2: Pb-204, Pb-206, Pb-207, Pb-208*
    - *BROND-2.2: Pb-204, Pb-206, Pb-207, Pb-208*

Effects of Pb and Bi Cross Sections



Sensitivity of k_{eff} to Elastic Scattering Cross Section

Case		k_{eff}
Base (ENDF/B-VI)		0.98806
ENDF/B-V	Bi	0.96014
	Pb	1.00215
	Bi and Pb	0.97389
JENDL-3.2	Bi	0.98833
	Pb	0.97720
	Bi and Pb	0.97747
BROND-2.2	Bi	0.98802
	Pb	0.98761
	Bi and Pb	0.98629

Differences in k_{eff} result mainly from discrepancies in elastic scattering cross sections

Actinide Data Assessment

Data Type	PROFIL-1 C/E			Expt'l Uncert. (%)	Data Uncert. (%)
	JEF2.2	ENDF/B-V	ENDF/B-VI		
σ_{capt} **U-235**	0.95	0.99	0.95	1.7	7.6
σ_{capt} **U-238**	0.98	1.02	0.98	2.3	3.1
σ_{capt} **Pu-238**	0.98	**1.30**	**1.69**	4.0	48.0
σ_{capt} **Pu-239**	0.99	0.96	0.94	3.0	10.6
σ_{capt} **Pu-240**	1.14	1.07	0.99	2.2	23.7
σ_{capt} **Pu-241**	**1.24**	1.03	0.88	4.1	27.4
σ_{capt} **Pu-242**	**1.19**	1.11	1.06	3.5	24.9
σ_{capt} **Am-241**	1.02	0.87	0.83	1.7	20.6
σ_{capt} **Am-243**	0.99	**0.59**	0.82	5.0	20.6

Large σ_{capt} discrepancies for ^{241}Pu and ^{242}Pu from JEF2.2 library and for ^{238}Pu and ^{243}Am from ENDF/B libraries

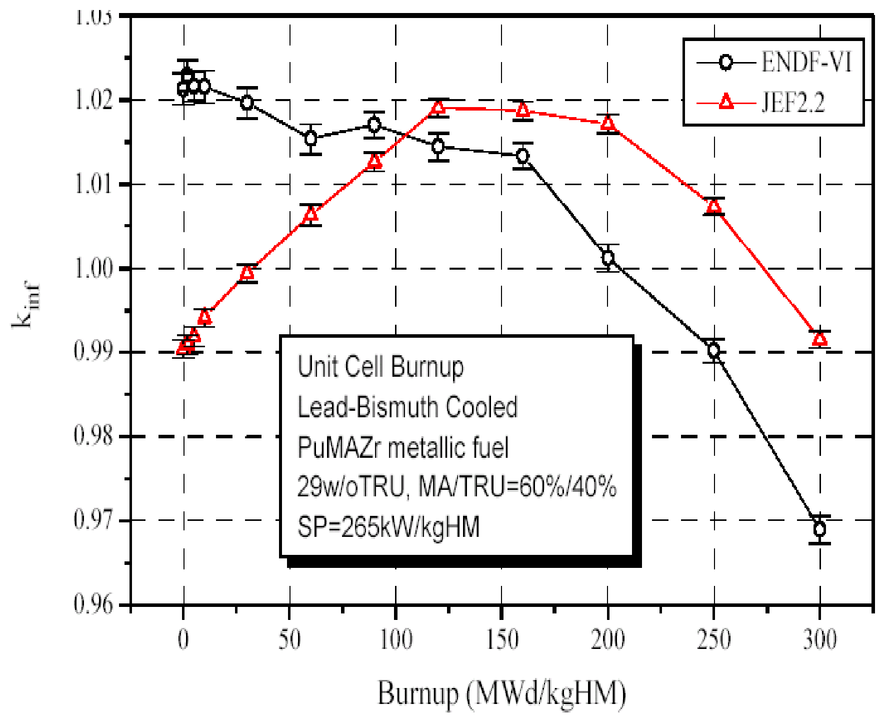
Estimated uncertainties due to data errors » corresponding measurement uncertainties

Effect of Discrepancies in Data for TRU Nuclides (MIT Analysis)

TABLE I. Spectrum-average One-Group Cross Sections*.

	JEF2.2		ENDF-VI	
Actinide	σ_f(b)	σ_c(b)	σ_f(b)	σ_c(b)
Np237	0.307	1.190	0.304	1.220
Pu238	1.080	0.412	1.070	0.579
Pu239	1.670	0.357	1.650	0.342
Pu240	0.360	0.414	0.356	0.392
Pu241	2.190	0.466	2.190	0.311
Pu242	0.250	0.357	0.245	0.343
Am241	0.228	1.590	0.232	1.330
Am242m	2.750	0.430	3.330	0.270
Am243	0.174	1.330	0.181	1.140
Cm242	0.581	0.359	0.123	0.208
Cm243	2.880	0.149	2.230	0.173
Cm244	0.408	0.446	0.400	0.687
Cm245	2.310	0.247	2.010	0.261

*Maximum statistical error in σ of ±0.006



Analysis of TRAPU Experiments using ENDF/B-VI

Irradiated oxide pins with different initial Pu vectors – high proportions of ^{240}Pu, ^{241}Pu, ^{242}Pu

Large C/E discrepancies for discharged amounts of ^{237}Np, ^{243}Cm, ^{244}Cm, attributed to errors in $\sigma_{n,2n}$ for ^{238}U and σ_c of higher Pu isotopes, ^{241}Am, ^{243}Am, and ^{242}Cm

Isotope	TRAPU-1	TRAPU-2	TRAPU-3
U-234	0.96± 3.9 %	0.99± 3.8 %	1.03± 4.6 %
U-235	0.99± 0.4%	1.01± 0.4%	1.01± 0.4%
U-236	1.01± 0.8 %	1.03± 1.0 %	1.02± 0.9 %
Np-237	0.75± 6.8 %	0.75± 3.3 %	0.73± 3.2 %
Pu-238	0.96± 1.5 %	0.97± 1.0 %	0.99± 1.6 %
Pu-239	1.03± 0.6 %	1.02± 0.5 %	1.02± 0.4 %
Pu-240	1.02± 0.6 %	1.00± 0.6 %	1.00± 0.6 %
Pu-241	1.07± 0.6 %	1.03± 0.6 %	1.05± 0.6 %
Pu-242	1.08± 0.8 %	1.03± 0.6 %	1.02± 0.6 %
Am-241	0.99± 3.2 %	0.99± 3.9 %	1.00± 2.6 %
Am242M	0.91± 3.8 %	0.94± 3.1 %	0.93± 3.1 %
Am-243	1.05± 2.6 %	1.02± 3.9 %	1.06± 2.5 %
Cm-242	1.02± 3.9 %	1.00± 3.1 %	1.00± 2.7 %
Cm-243	-	0.51± 3.1 %	0.52± 3.2 %
Cm-244	0.66± 2.1 %	0.73± 2.3 %	0.75± 1.8 %

Suggested Priorities for Future Work: Nuclear Data

- **Systematically assess needs for further evaluation and measurement**
  - Pu, MA, Pb, Bi, unconventional GFR fuel matrix and reflector materials
  - Consider contributions of different materials/reactions to the uncertainty in key performance parameters
  - Requires covariance data in format suitable for application studies

- **Compare high fidelity calculations (deterministic and Monte Carlo) to integral measurements sensitive to materials/reactions in question**
  - Provides validation data in integral sense
  - Ensemble of measurements indicates adjustments to data and their correlated uncertainties

Priority should be placed on identifying past integral experiment measurements of greatest relevance to Gen IV systems and on preserving their specifications and measured results

Additional experiments to address identified discrepancies

Suggested Priorities for Future Work: Modeling Capabilities

- **Qualify and improve capabilities for VHTR analysis and design optimization**
  - Treatment of the double heterogeneity and random distribution of particles
  - Accounting for the stochastic nature of pebble flow (for the PBR variant)
  - Mutually consistent flux and thermal conditions

- **For fast reactors, assess and implement modeling procedures that accurately represent**
  - Spectral transitions at core periphery
  - Neutron streaming in low-coolant density configurations
  - Reactivity effects of thermal or radiation induced displacement of core structures

- **Implement and qualify standardized methods for computing dpa and for correlating damage (macroscopic manifestation) to dpa**

Suggested Priorities for Future Work: Modeling Capabilities (cont'd)

- **Advance Monte Carlo simulation capabilities**
  - Improve reliability of variance estimates for eigenvalue calculations
  - Estimate and propagate nuclide density uncertainties in depletion calculations
  - Speed up simulation, e.g., through improved variance reduction techniques and effective use of increasing computer capabilities

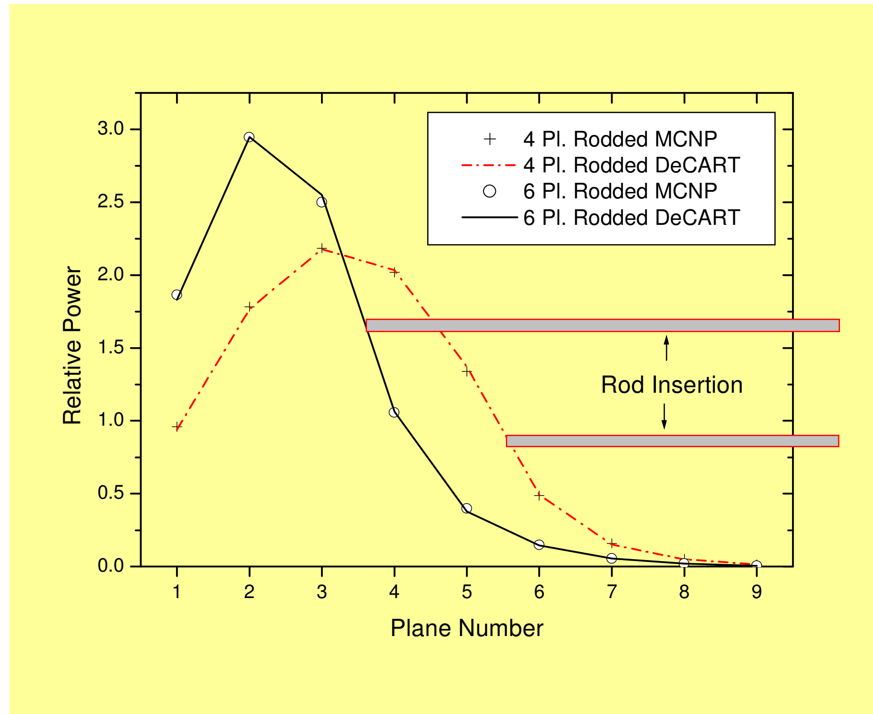
- **Improve efficiency (foremost human, but also machine effort)**
  - Greater automation, modularization, standardization of interfaces
  - Example: interpolation of XS data to specified temperature in MC simulation

DeCART – 3D Heterogeneous Transport Method

- **3D core transport calculation for detailed heterogeneous geometry**
  - Employs conventional lattice-code cross section library
  - Avoids approximations involved in assembly homogenization and energy group collapsing
  - Eliminate laborious cross section preparation for whole-core calculation

- **Transport solution method**
  - 2D planar MOC solution
  - 1D pin-wise diffusion/transport solution
  - 2D-1D coupling through transverse solution
  - Efficient multi-level acceleration schemes

Results for OECD/NEA C5G7 MOX Benchmark



Case	ε_k pcm	Max Error,%	RMS Error,%
2D	5	1.84	0.46
3D	4	1.89	0.50