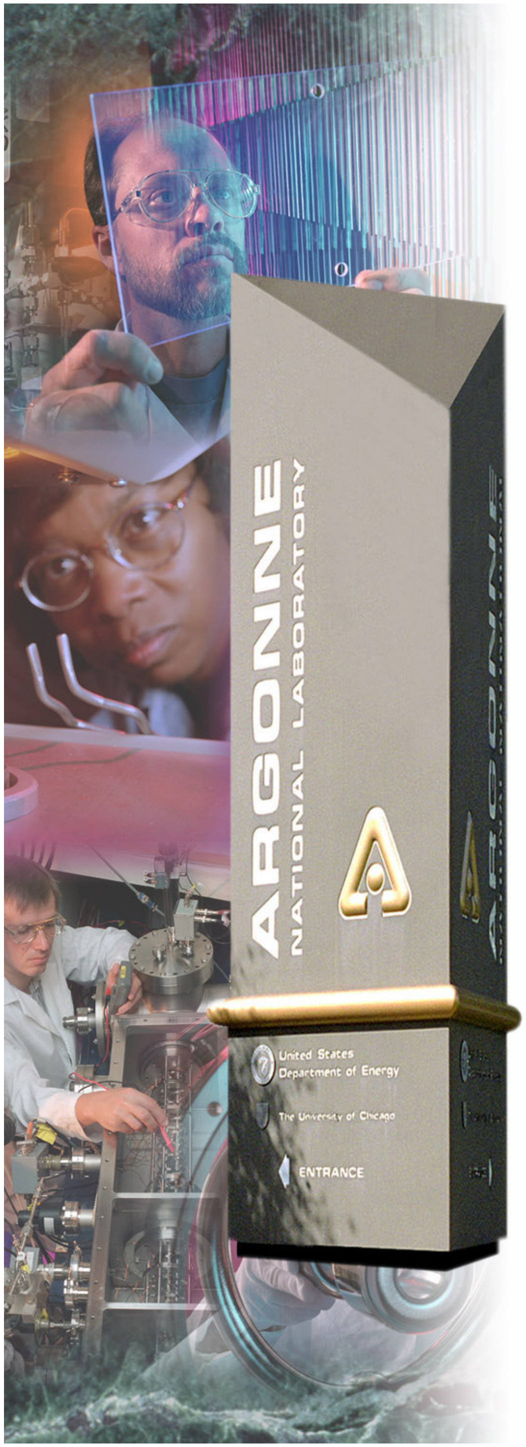

U.S.: Summary of 2003 Gen IV Reactor Physics Workshops

Workshop on Reactor Physics Advances for Design and Analysis of Gen IV Nuclear Energy Systems
Chicago, Illinois
April 30, 2004

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Nuclear Engineering Division
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Office of Science
U.S. Department of Energy

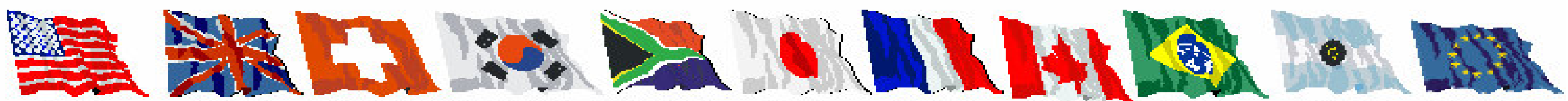
A U.S. Department of Energy
Office of Science Laboratory
Operated by The University of Chicago

Generation IV Systems

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

Prospective Participants in Steering Committees

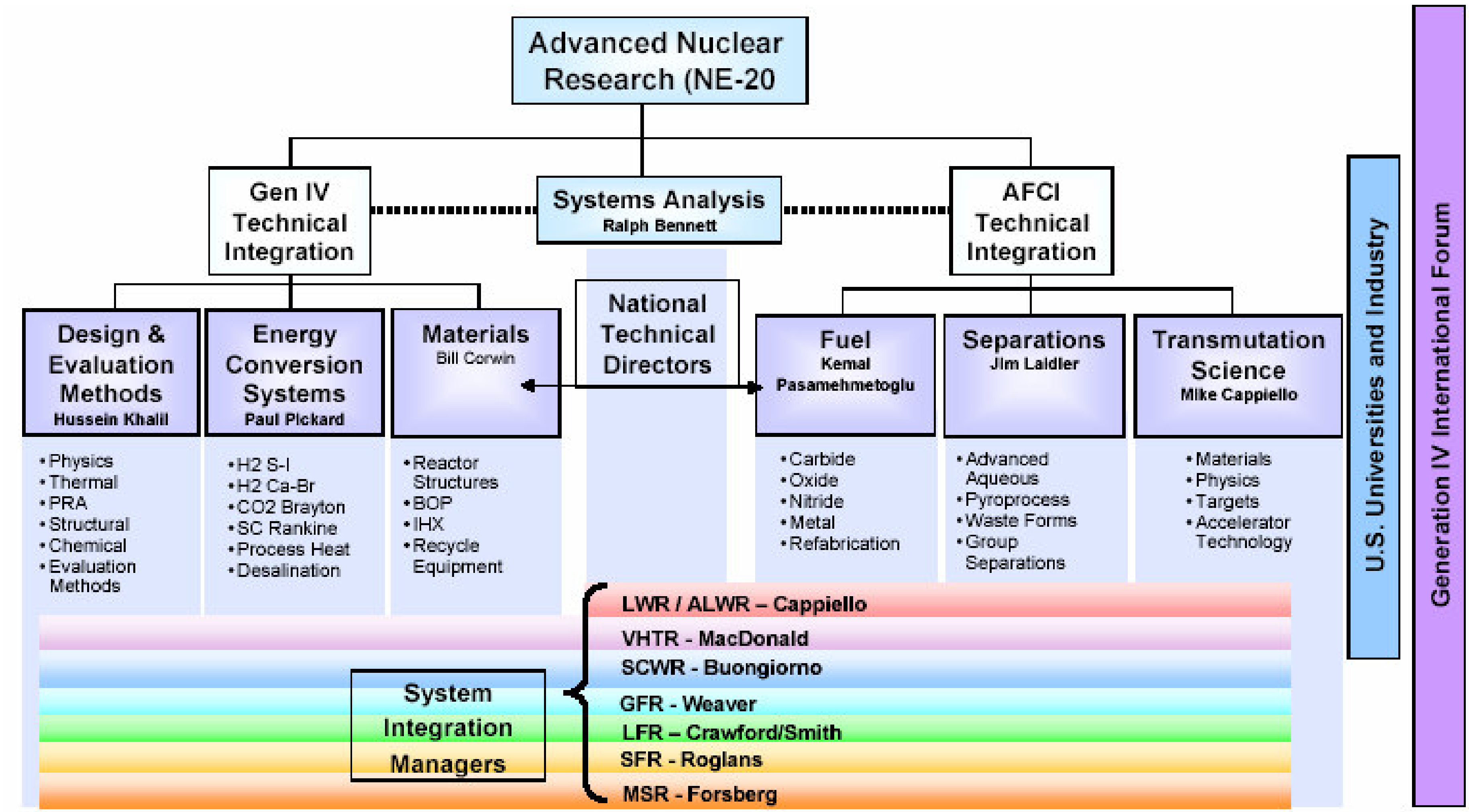
GIF Countries



	U.S.A.	United Kingdom	Switzerland	South Korea	South Africa	Japan	France	Canada	Brazil	Argentina	European Union
VHTR	*	*	*	*		*	*				*
GFR	*	*	*	*	*	*	*				
LFR	*		*	*		*	*				*
SFR	*	*		*		*	*				*
SCWR	*					*		*			*
MSR	*						*				*

*** Formation being negotiated or pending**

USDOE-NE Program Integration



U.S. Gen IV System Design and Evaluation Workshops to Assess Design Capabilities

- **Conducted workshops to assess**
  - Analysis needs for Gen IV systems
  - Capabilities of existing analysis tools (computer codes and databases)
  - Ongoing work to advance analysis capabilities
- **Workshops held in the following topical areas**
  - Reactor physics design analysis — Feb 18-19, 2003 (ANL)
  - T-H and safety analysis — Mar 18-19, 2003 (INEEL)
  - Nuclear data needs — Apr 24-25, 2003 (BNL)
- **Workshops attended by lab, university and industry representatives**
  - Presentations and discussion sessions
  - Conclusions and recommendations formulated
- **September 2003 report submitted to USDOE**
  - "Assess needs and opportunities for crosscutting design and safety analysis capability enhancement "

Reactor Physics Design Issues

System	Issues	
VHTR	•Fuel double heterogeneity •Stochastic behavior of pebble movement (for PBR variant)	•Graphite scattering treatment •Neutron streaming through coolant channels •Core/reflector interfacial effect
GFR	•Data for actinides, coolant (e.g., Pb, Bi) and fuel matrix candidate materials •Neutron streaming •Full-core transport effects •Spectral transition at core periphery •Modeling of reactivity feedback coefficients including expansion feedback	
LFR		
SFR		
SCWR	•Similar to BWRs •Increased heterogeneity	•Strong coupling of neutronics and T-H •Neutron streaming
MSR	•Evolution of mobile-fuel composition •Delayed neutron precursor loss	•Modeling of nuclear, thermal, and physio-chemical processes

Priorities for Future Development – Reactor Physics Methods (1)

- **While existing neutronic analysis tools may be largely adequate for early pre-conceptual design development and viability phase evaluations, improvements are needed:**
  - Treatment of the double heterogeneity effects of concepts using coated particle fuel (e.g., the VHTR)
  - Treatment of the stochastic fuel behavior for the pebble-bed variant
  - Improved modeling for spectral transitions at core periphery
  - Streaming effects in low-density coolant reactors
  - Definition and prediction of neutron damage
- **Priority should be placed on identifying previous integral experiment measurements of greatest relevance to advanced systems and on documenting/preserving their specifications and measured results**
- **Additional experiments should be defined to address significant deficiencies that may exist in the available experimental database**

Priorities for Future Development – Reactor Physics Methods (2)

- **Consistent methodology and/or set of standards should be developed for verification and validation of analysis methods and data for application to advanced systems**
- **Deterministic and Monte Carlo methods will be needed for design of advanced systems**
  - High-fidelity methods for whole-core analysis should continue to be pursued, including robust Monte Carlo methods with burnup and feedback capability
- **Development of a standardized physics design code system for the VHTR should be initiated given the targeted development pace**
  - Relevant existing capabilities should be identified
  - International VHTR benchmark activity should be launched to support effort
- **Coupling of neutronic, thermal-hydraulic (T-H), fuel behavior, and structural models should be planned and accommodated in the early stages of code development or adaptation**
- **Define target accuracies for core performance parameters**

Priorities for Future Development – Nuclear Data

- **High burnup operation of the VHTR might require re-evaluation of some transuranics data (cross sections, decay data, and fission yields) that have not typically been important in thermal-reactor design analysis**
  - Some new differential measurements may also ultimately be needed for selected nuclides, depending on the detailed spectral conditions and specific fuel cycle strategy adopted for the VHTR
- **Fast spectrum systems (GFR, LFR, and SFR) to be deployed for actinide management within a closed fuel cycle would require additional evaluation of data for transuranics, particularly minor actinides, as well as integral measurements for validation of the basic data and their processing tools**
- **Non-conventional structural or fuel-matrix materials may also necessitate new evaluations or measurements of basic data**
- **Systematic approach based on sensitivity and uncertainty analysis required for further specifying data needs**
  - Covariance data that are largely missing from current evaluated nuclear data files (e.g., ENDF/B) needed for this approach
  - Covariance information should be developed expeditiously

Nuclear Data Issues – Minor Actinides (MIT Study: P. Hejzlar)

- Neutronic analyses for advanced systems (ADS and Gen IV) rely on nuclear data libraries
- Data discrepancies exist for hard spectrum systems: order of tens of percent

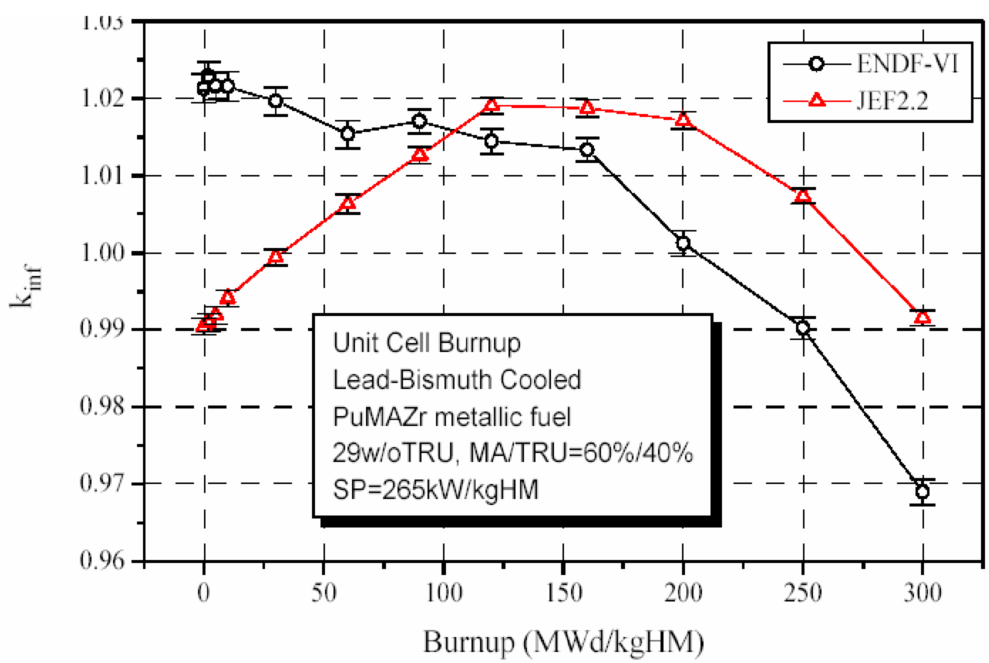


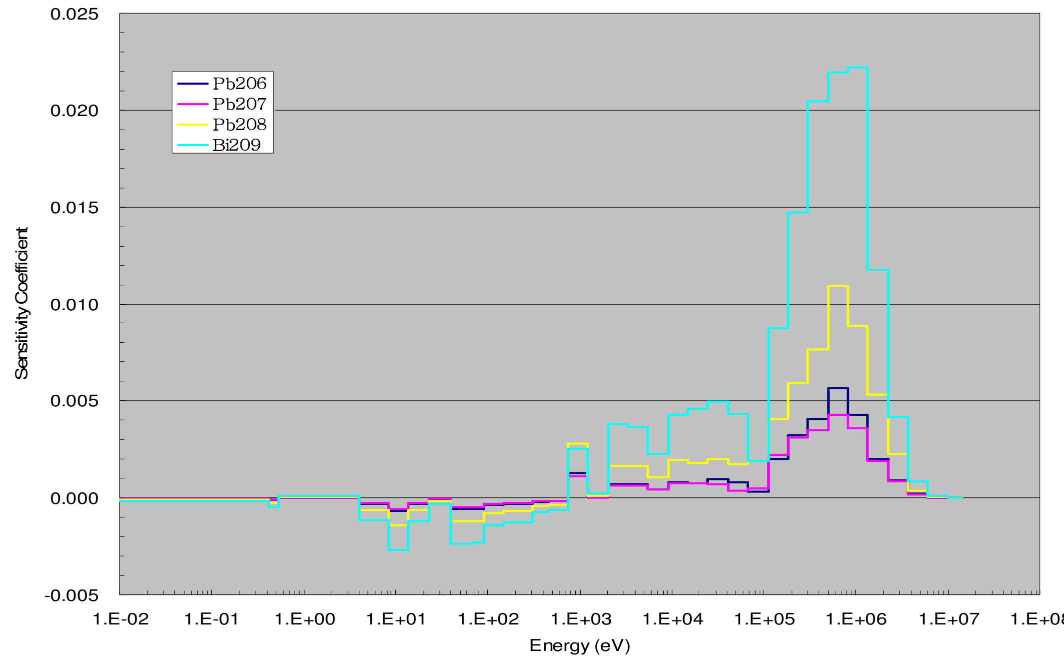
TABLE I. Spectrum-average One-Group Cross Sections*.

Actinide	JEF2.2 σ_f(b)	JEF2.2 σ_c(b)	ENDF-VI σ_f(b)	ENDF-VI 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 $\pm$0.006

Effects of Pb and Bi Cross Sections

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



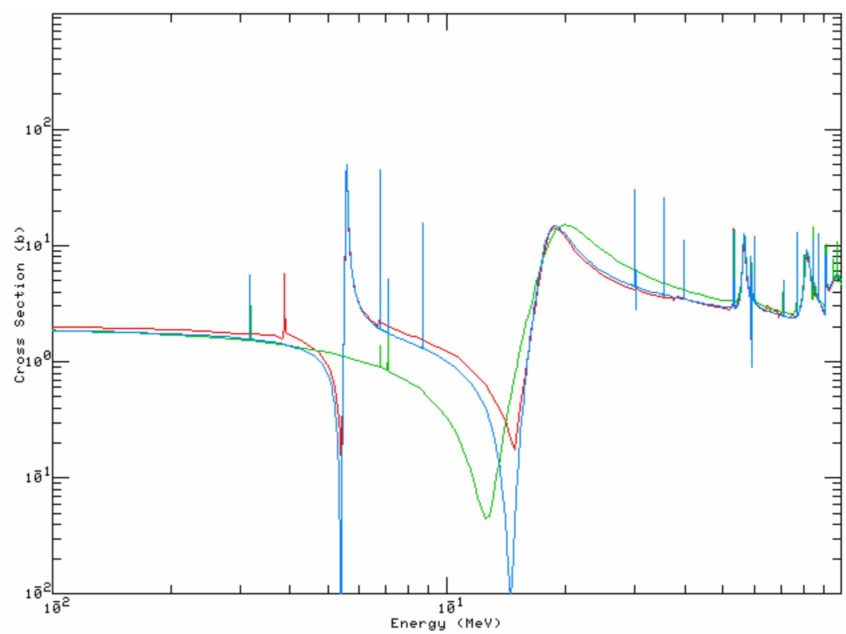
Sensitivity Functions of k_{eff} wrt Elastic Scattering Cross Sections

Differences in multiplication factor (k_{eff}) due mainly to differences in elastic scattering cross sections

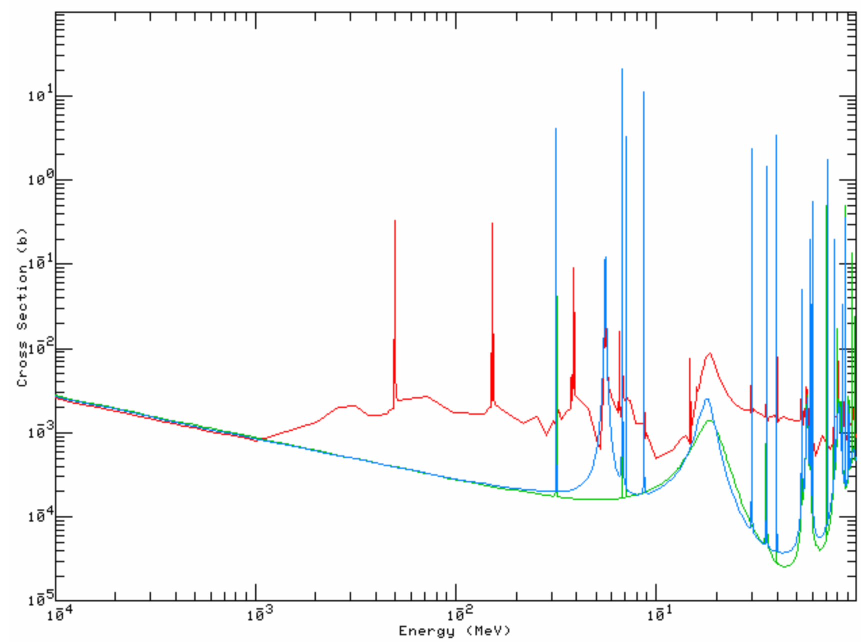
Total and Capture Cross Sections of Si

- **Silicon is proposed to be used in VHTR, GFR, SCWR, and LFR as fuel coating, fuel matrix, reflector, and structural materials**
- **Current evaluated libraries show large differences**

ENDF/B-VI.7 **Pre-ENDF/B-VII** **JEF-2**



Total



Capture

Parameters Affected by Nuclear Data Uncertainties

- Criticality (multiplication factor)
- Reactivity feedback coefficients (e.g., Doppler, Coolant Void)
- Kinetics parameters (e.g., Effective Delayed Neutron Fraction)
- Reactivity loss during irradiation (Excess reactivity)
- Peak power value
- Conversion ratio of sustainable cores
- Transmutation potential of burner cores
- Max Dpa, maximum helium- and hydrogen-production, maximum (helium-production)/Dpa
- Decay heat, radiotoxicity, and neutron and gamma radiation levels

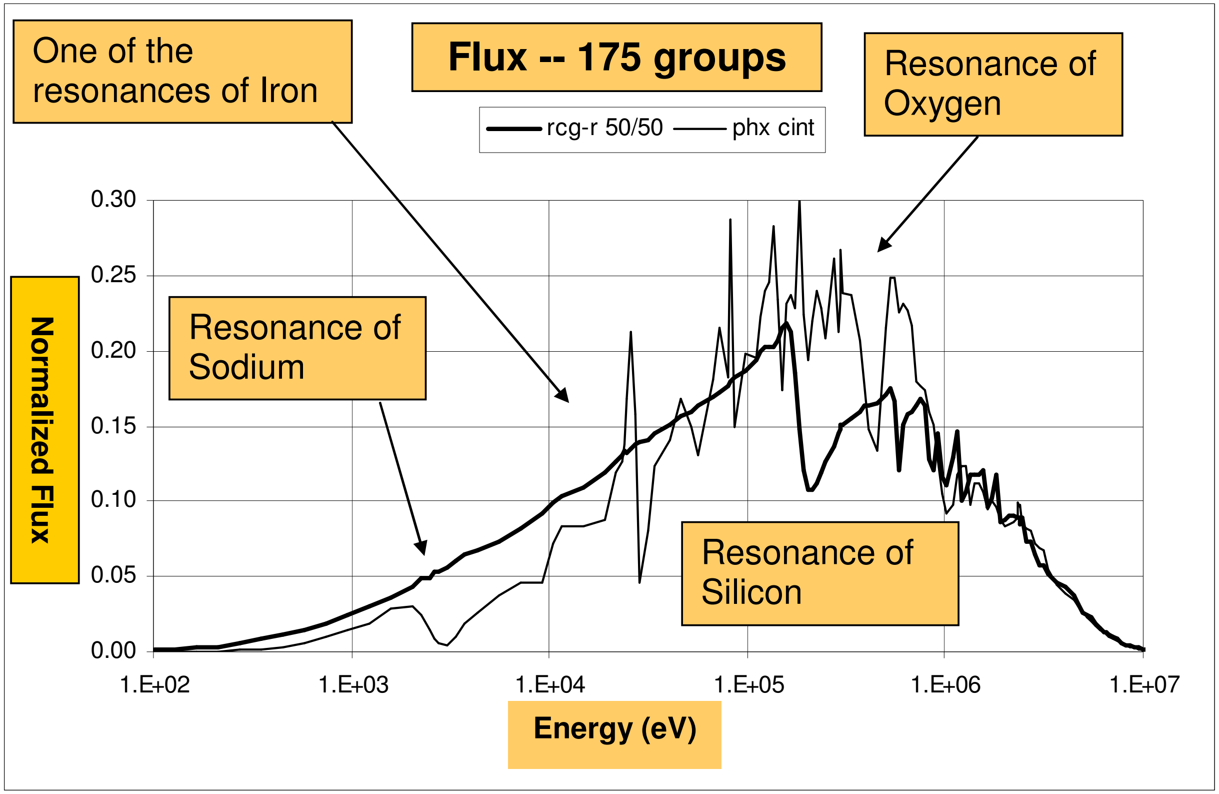
Some Recent or Planned International Reactor Physics Experiments

- ASTRA (RRC KI & South Africa), PROTEUS (PSI) and HTR-10 (China) experiments for VHTRs (PBMR)
- HTTR (Japan) experiments for VHTRs (block-type)
- Fast reactor experiments in U.S., France, Japan, Russia
- Planned ENIGMA project at CEA for GFR physics studies
- VENUS and MALIBU experiments (Belgium) for MOX fuel
- OSMOSE experiments (France)
- MUSE at CEA for evaluating accelerator driven systems (ADS)
- Planned TRADE and RACE projects for ADS at-power studies

Target Accuracies for Gen IV Reactor Physics Parameters (tentative)

	System Development Phase	
Parameter	**Viability**	**Performance**
Multiplication factor, k_{eff}	<0.5%	<0.1%
Local power density	~5%	~1%
Control element worth	2-5%	1%
Reactivity coefficients: Large effects Small effects	 10% 20%	 1-5% 10%
Kinetics parameters	5%	2%
Local nuclide densities: Major constituents Minor constituents	 5% 10-20%	 1% 2-5%

Comparison of FR and GFR Spectra (CEA Study)



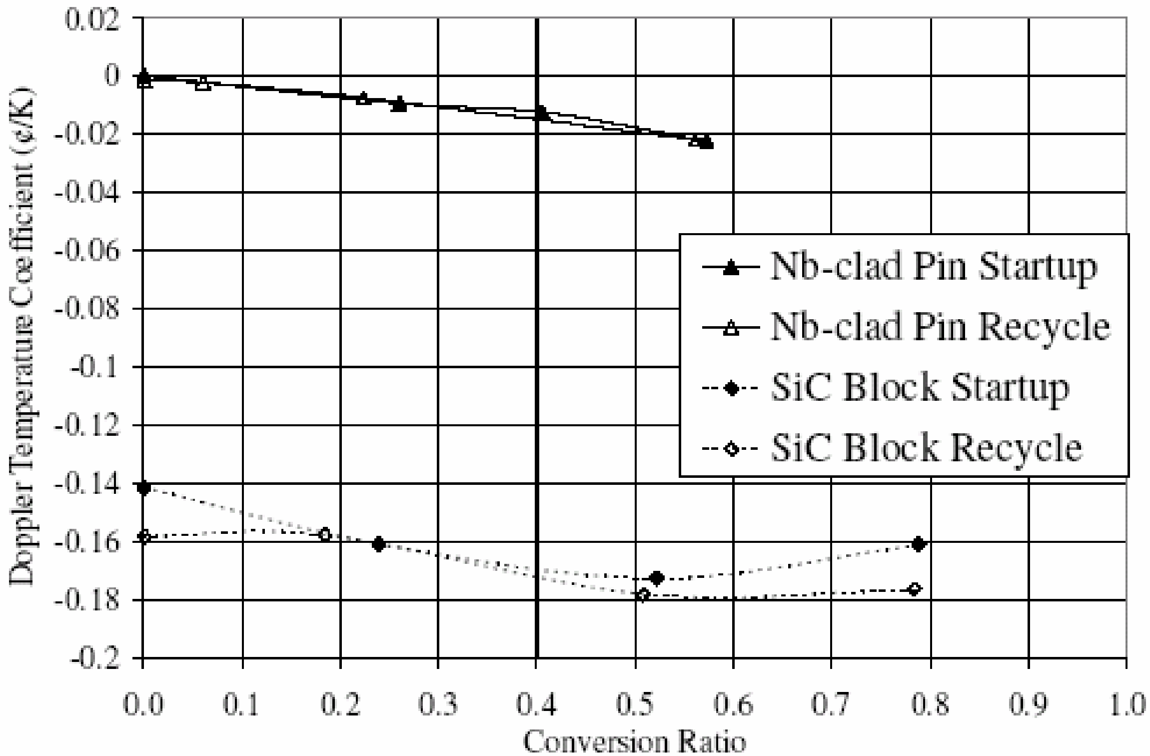
- **Differences observed in neutron spectrum, streaming, behavior of reflectors (reactivity, power distributions), reactivity of control rods, Doppler**
- **Advanced systems could use degraded (vs. typical) Pu isotopics**
- **Need of integral experiments and a few selected cross-section data to support GFR, particularly for matrix, structure, reflector materials**

Experimental Neutron Investigation on Gas-Reactors at MAsurca (ENIGMA) Program Support

- **Planning is underway at CEA for GFR physics experiments to be conducted at the MASURCA facility**
- **First phase of the experiments will be from 2005 to 2007**
  - Define MASURCA configurations (Phase I) that are similar in their neutronic characteristics to candidate GFR designs by June 2004
  - Validate methods, codes and data for neutronics calculations of GFRs
- **Use of different axial and radial reflectors surrounding has been studied (i.e., innovative materials such as C, Zr, ZrC, Zr_3Si_2, and SiC instead of the traditional stainless steel reflector)**

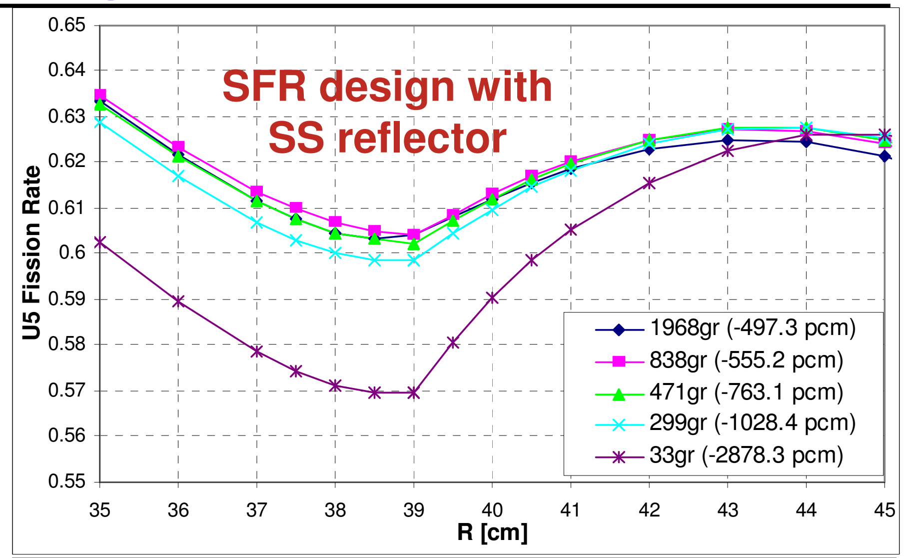
SiC as Matrix Materials for Block-type GFR Gives Large Doppler Coefficient Value

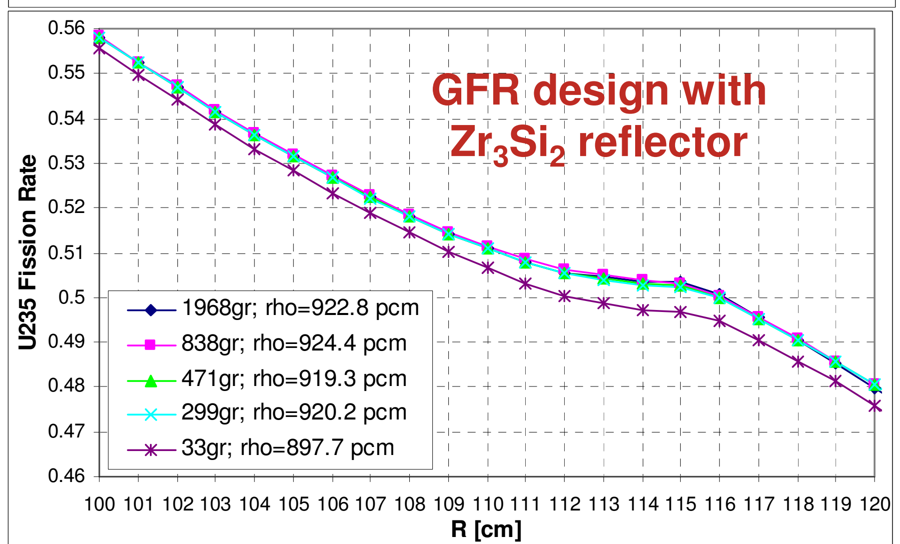
- **Doppler temperature coefficient fairly large at low conversion ratio with SiC as matrix material in block-type GFR design**
- **Trend associated with the significant shift in neutron spectrum as the conversion ratio is reduced**
  - For the CR=0 case, the in-core SiC inventory is nearly twice as large as in the reference design
- **Resulting shift in neutron spectrum increases resonance absorption in fertile TRU isotopes (e.g., ^{240}Pu)**
- **Issue needs further verification**



Material Impact on Core-Reflector Interface Physics

- **Proximity of core and reflector (no blanket) in advanced fast reactor designs (e.g., SFR, GFR, and LFR)**
- **With few broad groups (e.g., 33), proper treatment of interfacial effects is quite important for flux and reactivity calculations, particularly with stainless steel (SS) reflector**
  - Detailed macrocell calculation is required for interfacial treatment
- **Differences due to cross-section resonance structures**
  - Resonances appearing above 10 KeV for the isotopes Si and Zr are more isolated than for the isotopes ^{56}Fe, ^{52}Cr and ^{58}Ni present in SS
- **Zr and Si trends need further evaluation by rigorously processing resonance cross-sections with more energy groups and comparison to Monte Carlo calculations and measurements**





Reactor Physics Issues for SFR Design

- **Inaccuracies in diffusion theory solution due to incorrect transport cross sections have been observed**
- **In "high leakage" configurations, this requires more detailed coupling of material regions (core/reflector/blanket) when cross sections are generated**
  - Using transport or more energy groups is an alternative, but not necessarily a sure fix
- **Core performance, material reactivity coefficients and reactivity expansion coefficients can all be affected significantly**

VARIANT Approx.	Low Leakage Configuration		High Leakage Configuration	
	Flooded	Voided	Flooded	Voided
$P_1\ S_0$	0.9983	1.0093	0.9972	0.9901
$P_3\ S_0$	1.0139	1.0282	1.0169	1.0176
$P_5\ S_0$	1.0166	1.0319	1.0203	1.0231
$P_1\ S_1$	1.0188	1.0334	1.0300	1.0328
$P_3\ S_1$	1.0267	1.0429	1.0401	1.0473
$P_5\ S_1$	1.0276	1.0441	1.0413	1.0493
VIM	1.0278 +/- 0.0001	1.0400 +/- 0.0001	1.0363 +/- 0.0002	1.0307 +/- 0.0001

***P_xS_y: x is the order of the angular dependence of flux; y is the order of anisotropy of scattering X-S**

Monte Carlo Methods for Core Analysis

Pro

- **Monte Carlo capabilities have been used for preliminary design of Generation IV concepts**
- **Permits explicit analysis of core fuel arrangements**
- **Attractive because of the ability to represent accurately nuclear data details and to treat heterogeneity effects and complex geometries**
- **Method is particularly useful for analysis of specified designs**
  - If only a limited exploration of design and operational parameters is required, then such a calculation approach may be sufficient

Con

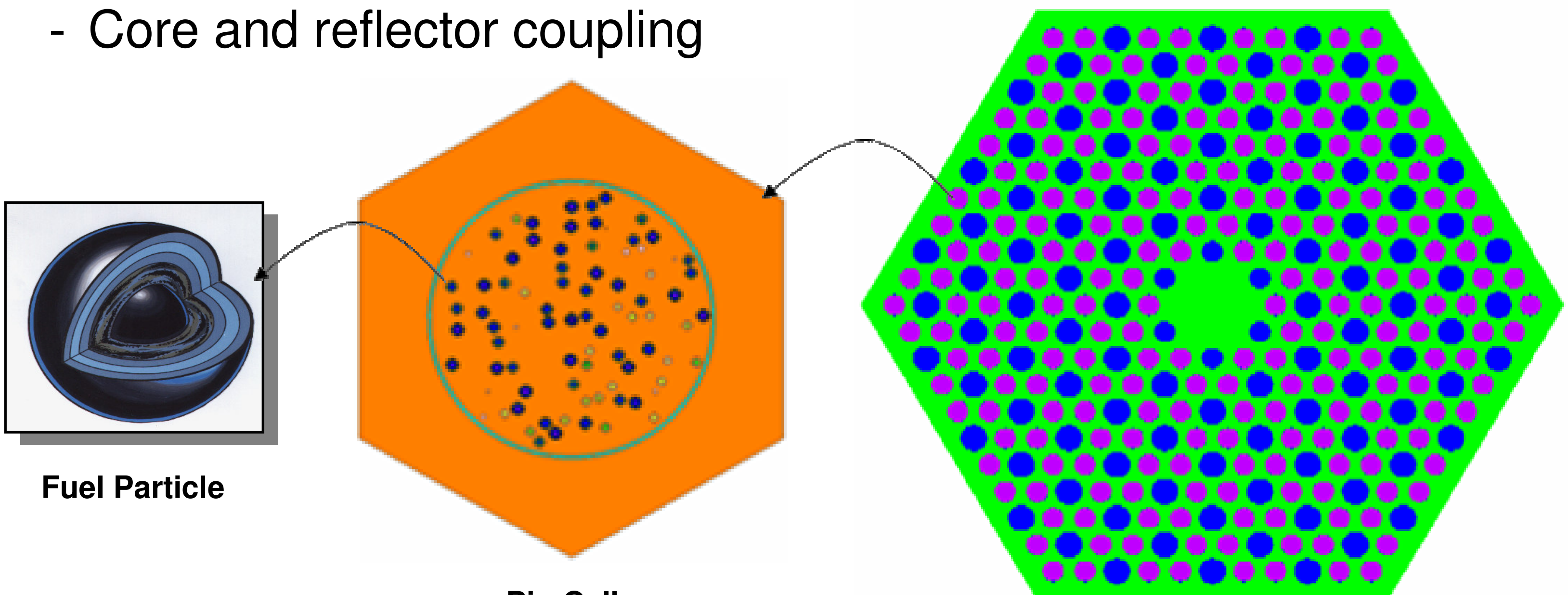
- **Computationally intensive calculation**
  - For acceptably low statistical uncertainty
  - Modern computer architecture has alleviated problem
- **Not sufficiently efficient for use in parametric and trade studies required for developing an optimized design**
- **Propagation of uncertainties in depletion calculations**
- **Thermal feedback capability**
- **Perturbation capability for sensitivity studies**
- **Efficient time-dependent capability**

Developmental Capabilities and Activities

- **Efforts are being made to improve the current nodal approaches that employ assembly homogenization**
  - Quasi-diffusion theory approach
  - High-order boundary perturbation method
  - Heterogeneous deterministic transport theory methods
- **Application of spatially adaptive quadrature sets and wavelets are being investigated to improve angular discretization in S_n methods**
- **Whole-core heterogeneous transport calculation capabilities are being developed based on MOC to eliminate homogenization step**
- **Whole-core diffusion theory code is being developed to represent continuous refueling and resulting pebble motion in PBMR reactors**

VHTR Physics

- **Very high temperature reactor (VHTR) is a leading candidate for the next generation nuclear plant (NGNP)**
- **Physics analysis of VHTR requires modeling capabilities**
  - Double heterogeneity of coated particle fuel
  - Neutron streaming in coolant channels
  - Core and reflector coupling



Fuel Particle

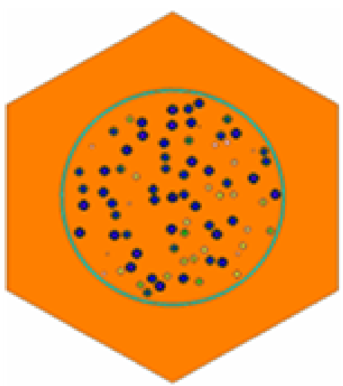
Pin Cell

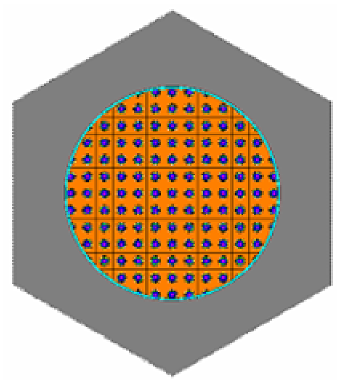
Assembly Block

VHTR Issues – Fuel Block Calculations

Computational model		NGNP, UCO	GT-MHR, UO_2	GT-MHR, TRU
Homogenized compact		1.48040 ± 0.00075	1.53966 ± 0.00046	1.07997 ± 0.00088
Stochastic distribution	Mean	**1.53280**[a)]	**1.57335**	**1.25838**
	STD	0.00082	0.00040	0.00040
Regular lattice distribution	SC	1.52978 ± 0.00071	1.57279 ± 0.00039	1.25427 ± 0.00071
	BCC	1.53160 ± 0.00071	1.57118 ± 0.00041	1.25317 ± 0.00070
	FCC	1.52890 ± 0.00073	1.57276 ± 0.00041	1.25192 ± 0.00077
DRAGON		1.54393 (470)	1.57565 (93)	1.26794 (599)
WIMS8		1.52993 (-122)	1.57121 (-86)	1.25326 (-325)
Double heterogeneity effect		2.3 % $\Delta\rho$	1.4 % $\Delta\rho$	13.1 % $\Delta\rho$

Random Distribution in Compact

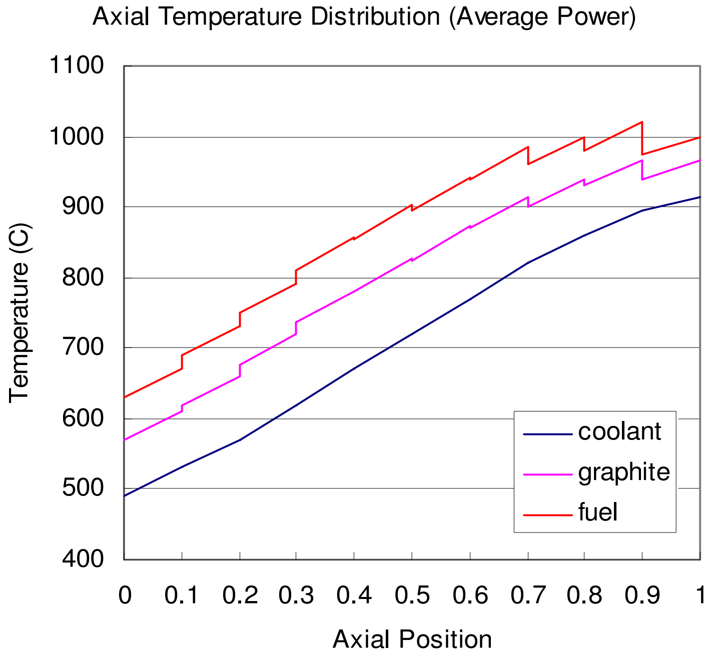




Lattice Distribution in Compact

Thermal Feedback Effect in VHTR Modeling

- **Development of thermal feedback scheme**
  - Importance of thermal feedback in prismatic NGNP design was examined
    - *Axial temperature distribution vs. average temperature*



	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

Recommendations Concerning IRPhE Database (Dec. 2003 Meeting in France)

- **Preservation of past specifications, experiments and measured results pertinent to Gen IV systems has been identified as a need**
  - Consistent with the goals of IRPhE Project
- **IRPhE and Gen IV Design and Evaluation Methods activities could be leveraged**
- **Proposal**
  - Convene subgroups to evaluate the IRPhE database for data of interest to the International Gen IV systems
  - Participants could be from U.S., Japan, France, Britain, South Korea and Canada
  - Select meaningful (clean) experiments (useful C/E)
  - Determine if data are adequate for the validation and verification of each pertinent system
  - Prepare report on findings

Backup

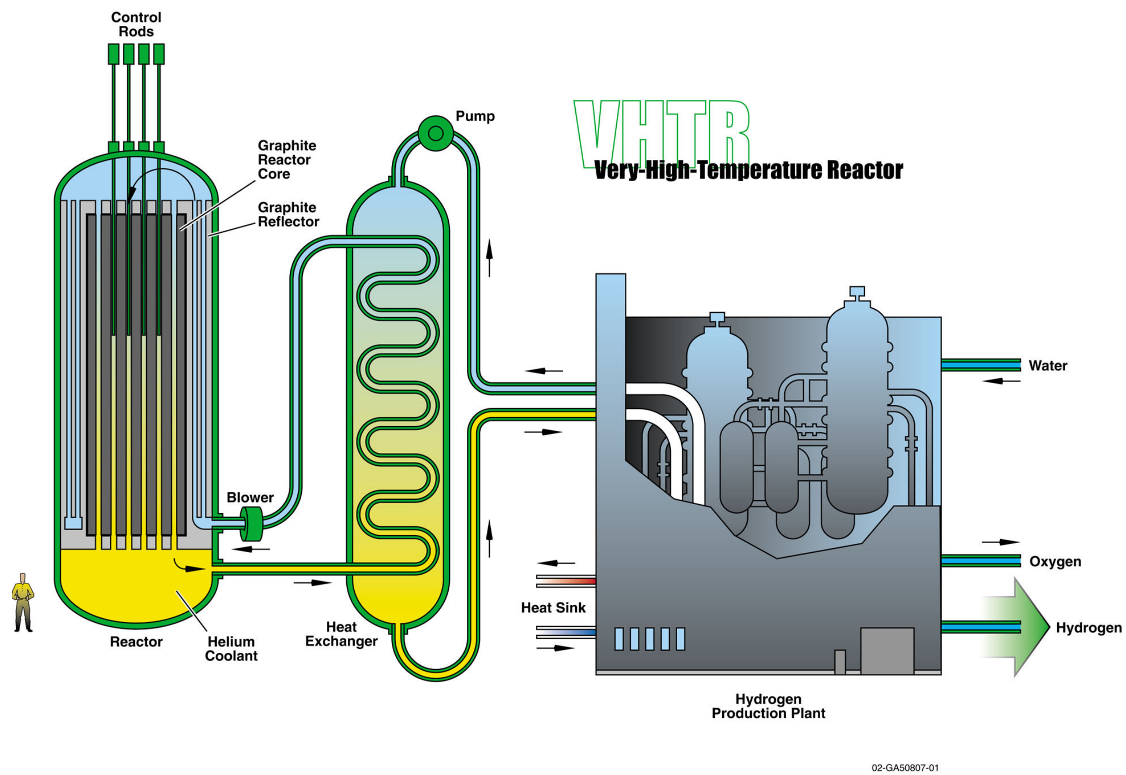
- Backup Information
- Backup information

Very-High-Temperature Reactor (VHTR)

Characteristics

- He coolant, direct cycle
- 1000℃ outlet temperature
- 600 MW_{th}, nominally based on GT-MHR
- Coated particle fuel
- Solid graphite block core

- High thermal efficiency
- Hydrogen production
- Passive safety



Reactor physics issues

- Fuel double heterogeneity
- Stochastic behavior of pebble movement (for PBR variant)
- Graphite scattering treatment
- Neutron streaming through coolant channels
- Core/reflector interfacial effect

Gas-Cooled Fast Reactor (GFR)

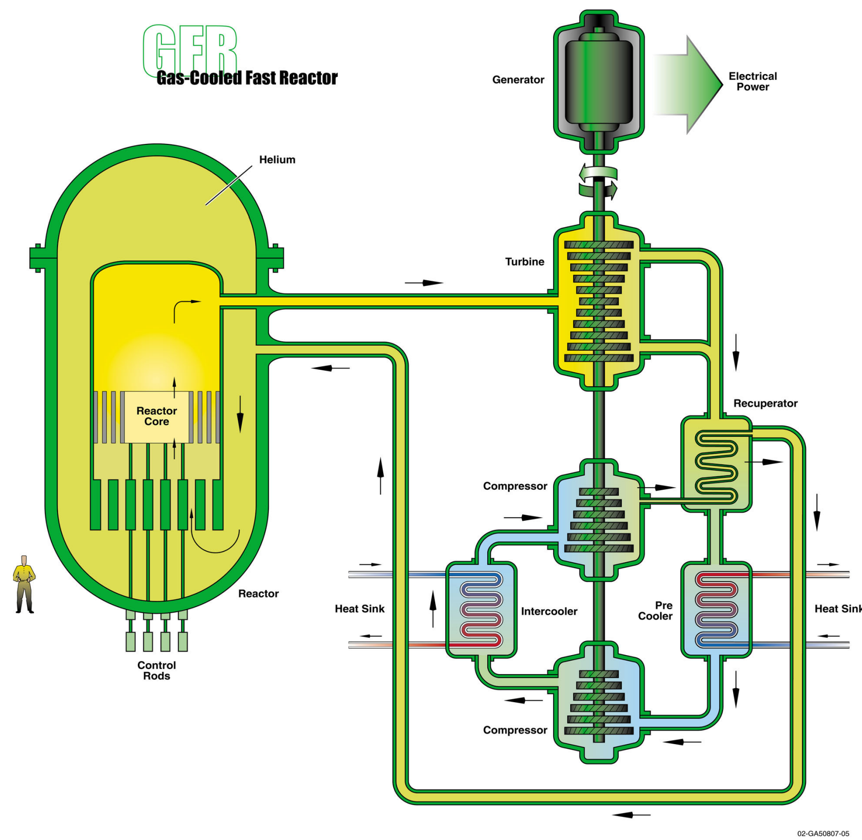
Characteristics

- He (or SC CO_2) coolant, direct cycle gas-turbine
- 850°C outlet temperature
- 600 MW_{th}/288 MW_e
- U-TRU ceramic fuel in coated particle, dispersion, or homogeneous form
- Block, pebble, plate or pin core geometry

- Waste minimization
- Efficient electricity generation

Reactor physics issues

- Core configuration dependent
- Neutron streaming
- Data for actinides and fuel matrix candidate materials

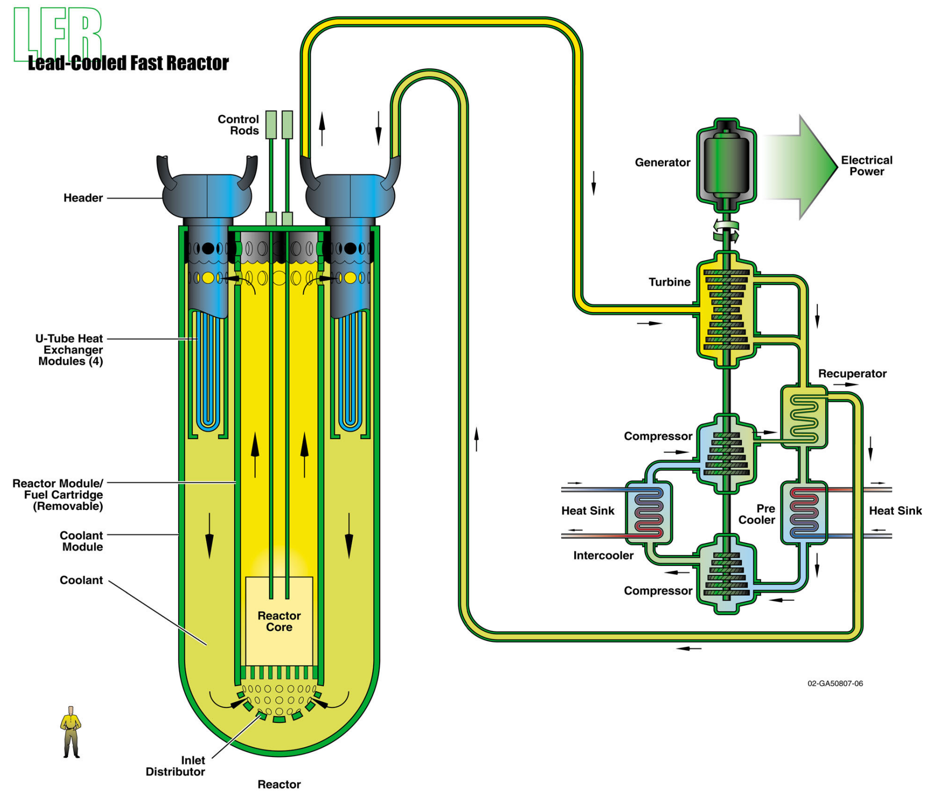


Lead-Cooled Fast Reactor (LFR)

Characteristics

- Pb or Pb/Bi coolant
- 550℃ to 800℃ outlet temperature
- U-TRU nitride or Zr-alloy fuel pins on triangular pitch
- 120–400 MWe
- 15–30 year core life
- Core refueled as a cartridge

- Distributed energy generation
- Transportable core
- Passive safety and operational autonomy



Reactor physics issues

- Data for actinides, Pb, Bi
- Spectrum transition at core edge
- Reactivity feedback coefficients

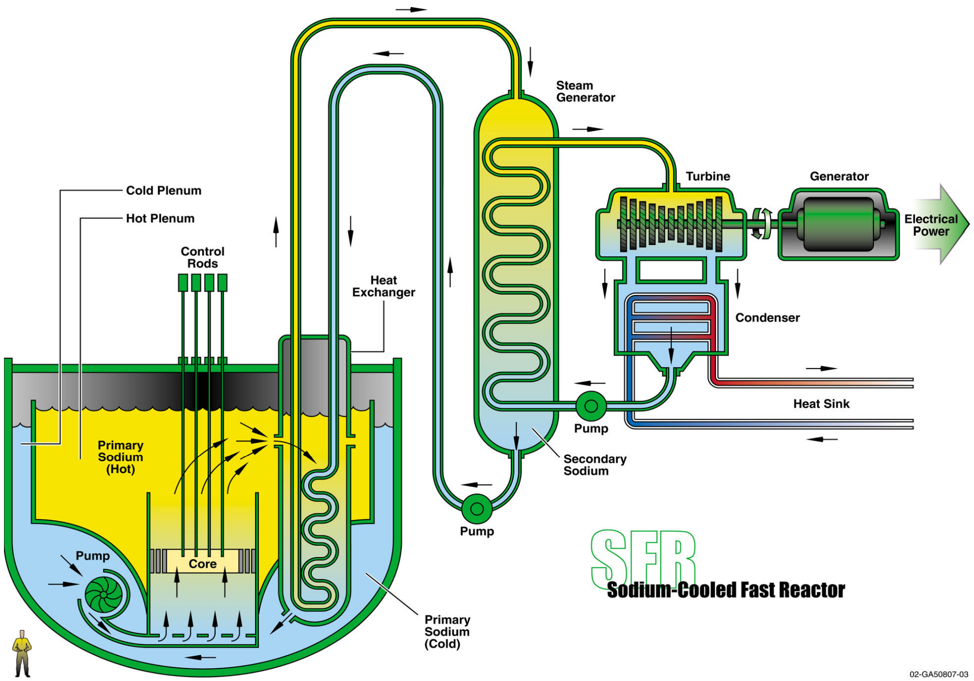
Sodium-Cooled Fast Reactor (SFR)

Characteristics

- Sodium coolant, 550°C T_{out}
- 150 to 1500 MWe
- U-TRU oxide or metal-alloy fuel
- Hexagonal assemblies of fuel pins on triangular pitch
- Homogenous or heterogeneous core
- Consumption of LWR discharge actinides
- Efficient fissile material generation

Reactor physics issues

- Actinide data
- Full-core transport effects
- Spectral transition at core periphery and beyond
- Accurate modeling of expansion feedback

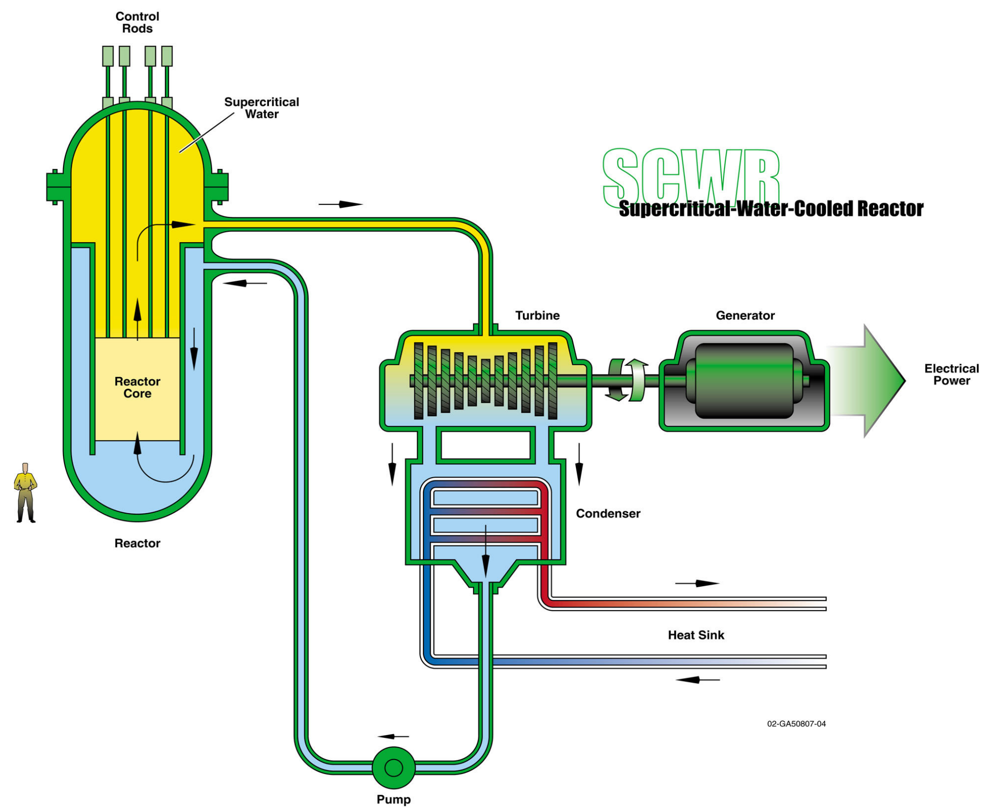


Supercritical-Water-Cooled Reactor (SCWR)

Characteristics

- Water coolant at supercritical conditions (~25 MPa)
- 510℃ outlet temperature
- 1700 MWe
- UO_2 fuel, clad with SS or Ni-based alloy
- Square (or hex) assemblies with moderator rods

- High efficiency, compact plant
- Thermal or fast neutron spectrum



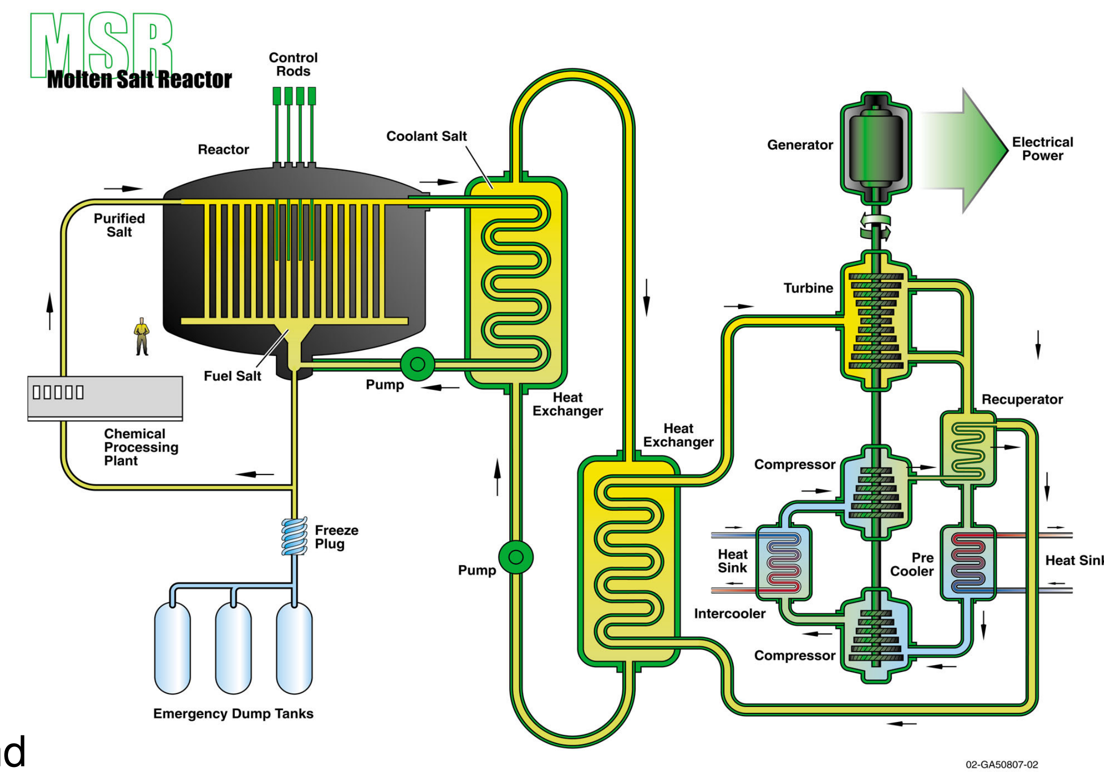
Reactor physics issues

- Similar to BWR's
- Increased heterogeneity
- Strong coupling of neutronics and T-H
- Neutron streaming

Molten Salt Reactor (MSR)

Characteristics

- Molten fluoride salt fuel
- 700–800℃ outlet temperature
- 1000 MWe
- Low pressure (<0.5 MPa)
- Circulating actinide-bearing fuel
- Graphite core structure to channel flow
- Actinide consumption
- Avoids fuel development and fabrication



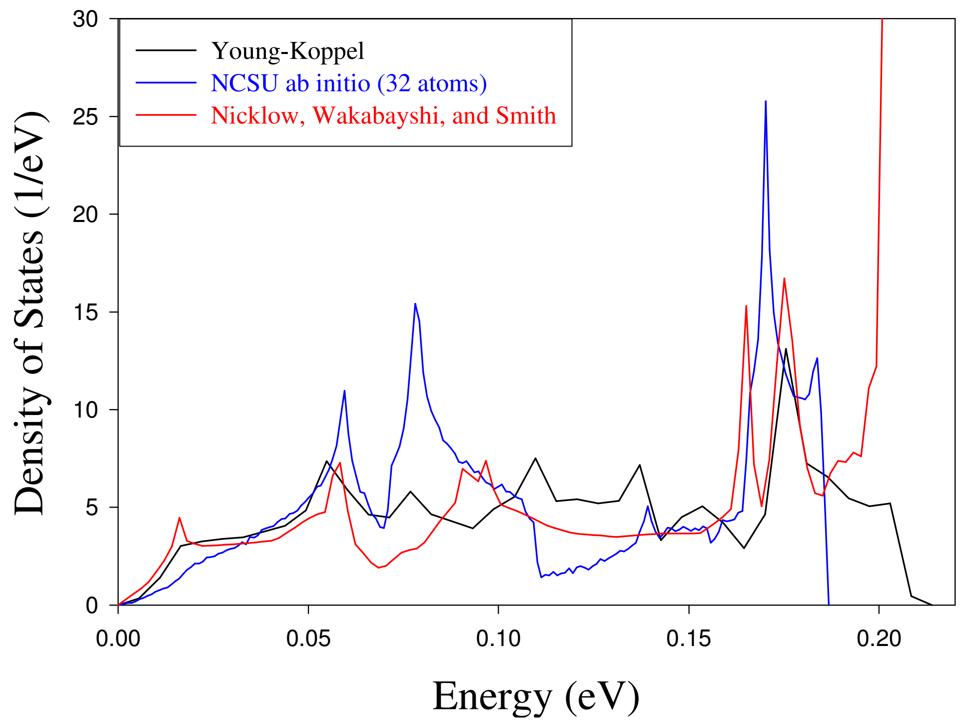
Reactor physics issues

- Evolution of mobile-fuel composition
- Modeling of nuclear, thermal, and physio-chemical processes
- Delayed neutron precursor loss

NCSU, ORNL, and Institute Balseiro (Argentina) Evaluation of Thermal-Neutron Scattering Data

- **Review the currently used thermal neutron scattering laws for various neutron moderators/reflectors as a function of temperature**
- **Update models and models parameters by introducing new developments in thermalization theory and condensed matter physics**
- **Benchmark the results against a well documented and representative set of experimental data sensitive to the neutron spectra**
- **In the case of graphite, perform a benchmark experiment by observing neutron slowing down as a function of temperatures equal to or greater than room temperature**
- **Understand the implications of the obtained results on the ability to accurately determine the operating and safety characteristics of a given reactor design**

Graphite Phonon Distributions



Identify Cross Sections with Uncertainties that have Greatest Impact on Gen IV System Performance?

- **Sensitivity studies, via GPT (Generalized Perturbation Theory), on major integral parameters using representative models of systems**
- **Uncertainty assessment**
  - Requires covariance data in multigroup format
- **Define target accuracy requirements and estimate required accuracy on nuclear cross section data ("inverse" problem)**
- **Use integral experiments for assessment of current knowledge on cross sections**
- **Data re-evaluation and/or measurement**

Development of Reactor Physics Computational Path for Analysis of NGNP

- **Develop a code suite for lattice and whole-core calculations**
- **Investigate and incorporate modeling improvements (e.g., whole-core capability with no homogenization)**
- **Identify exiting databases for support of code V&V and additional nuclear data needs**
- **Participate in international benchmark activities**
- **Develop plan for USNRC licensable capability with rigorous V&V**