HTGR Technology Course for the Nuclear Regulatory Commission

May 24 – 27, 2010

Module 5b

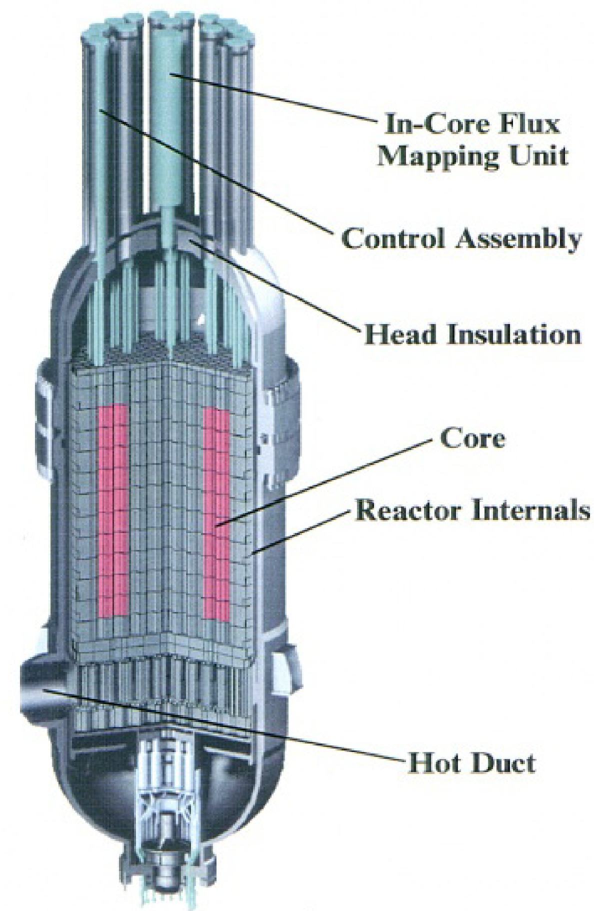
Prismatic HTGR Nuclear Design

Alan Baxter

General Atomics

Outline

- HTGR core nuclear design basics
- Temperature coefficients
- Decay heat
- Analytical tools
- Code verification and validation



HTGR Nuclear Design Shaped by System Requirements and Materials

- **Graphite is the moderator and structure, not metal and water**
  - High temperature solid moderator
  - hard thermal spectrum
  - fixed burnable poison
  - Large physical dimensions
  - low power density

- **Helium is the coolant not water**
  - Coolant is transparent to thermal neutrons
  - Coolant has no phase changes

- **Fuel is carbide-clad, small ceramic, particles not metal clad UO_2**
  - PyC/SiC carbide clad is primary fission product release barrier
  - Fuel operates at high temperatures with wide margin to failure
  - Double heterogeneity in physics modeling of the fuel

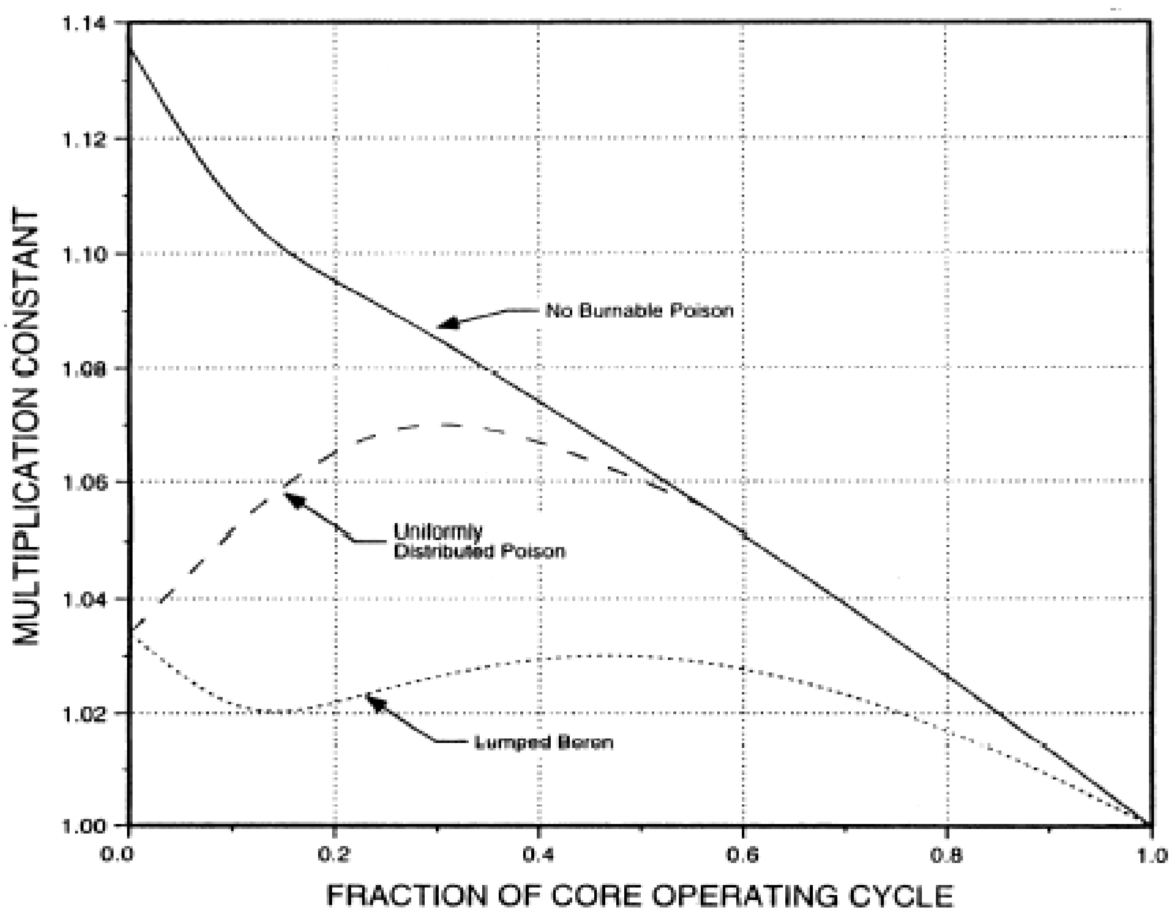
- **Modular HTGR has an annular, not cylindrical, core**
  - In-core control rods withdrawn during startup
  - Reflector rods used for control at power

HTGR Nuclear Characteristics
- A Comparison -

Nuclear Properties

Core	Modular - HTGR	LWR
Power density, w/cc	5.8-6.6	58 - 105
Linear heat rate, kW/ft	1.6	19
Avg. therm-neutron energy, eV	0.22	0.17
Average Uranium Enrichment	15.5%	4.00%
Moderator (at 0.025 eV)	**Graphite**	**Water**
Diffusion Coefficient D, cm	0.86	0.16
Diffusion Length L, cm	54	2.75
Migration length M, cm	57	6
Collisions to thermalize	~18	~1
Σa (cm^{-1})	0.00029	0.022
Σs c(m^{-1})	0.41	3.45

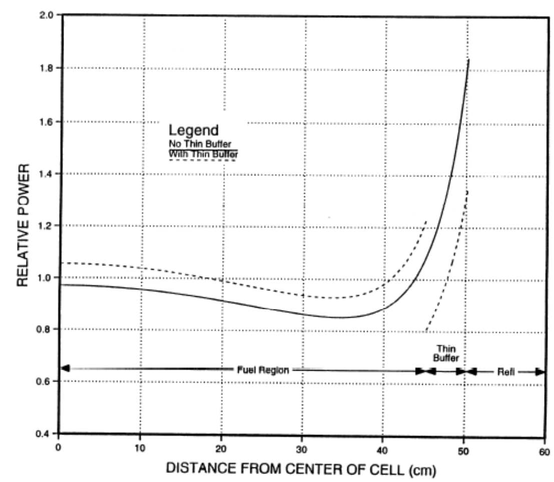
Use of Fixed Lumped Boron Poison (LBP) for HTGR Reactivity Control



Self shielding of the lumped boron (B4C) used to control poison burnout and core reactivity behavior over a fuel cycle to minimize control rod requirements

Modular HTGR Fuel and LBP is Zoned to Control Power Distribution

- **Fuel and burnable poison loadings are varied radially within core annular rings and axially within fuel columns (zoning)**
  - To maintain stable power shapes with control rod motion and fuel burnup
  - To keep peak fuel temperatures within acceptable limits
- **The uranium loading in the fuel rods adjacent to the core/reflector boundary is reduced to minimize the reflector thermal peaking effect**



- **Reflector control rods are used for reactivity control during normal operation, and the control sequence is varied for more uniform burnup, and control of power peaks**

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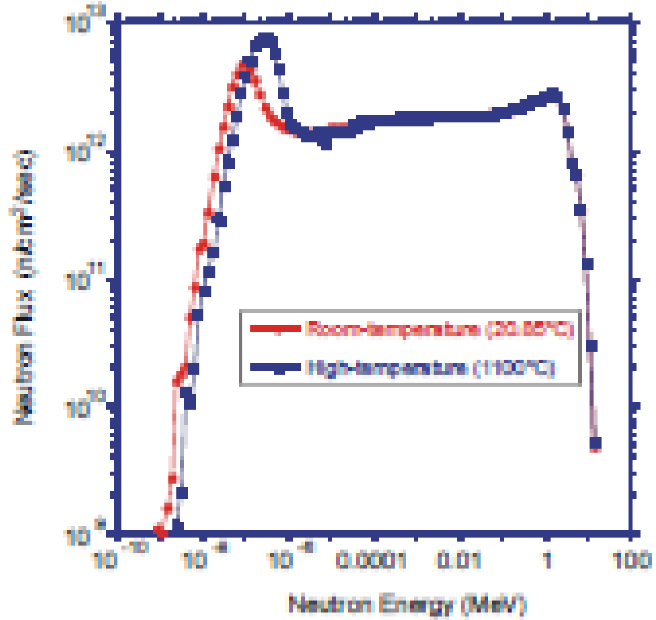
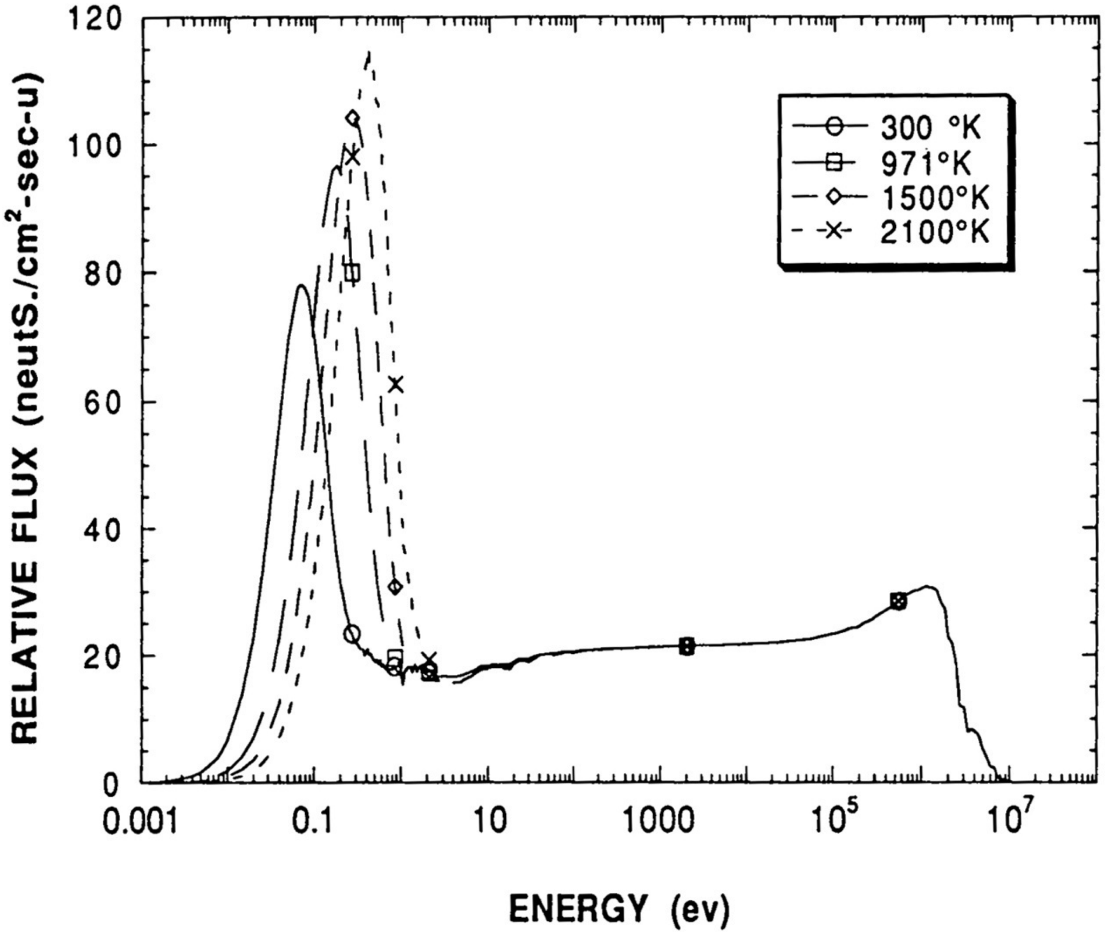


Modular HTGR Temperature Coefficients

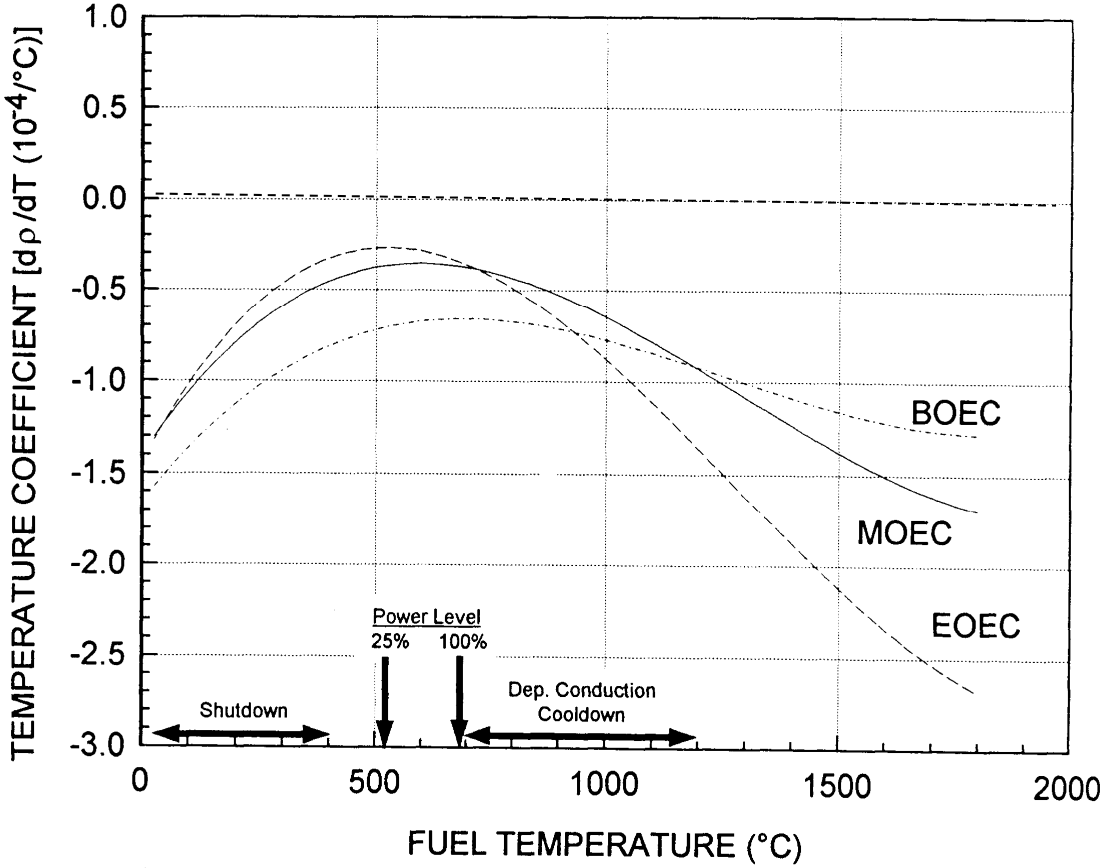
- **Except for control rod motion, the only significant reactivity effect in modular HTGRs is that caused by changes in core temperature**
  - Helium is essentially transparent to thermal neutrons
  - Core dimensional changes are negligible
- **Reactivity decreases as core temperature increases**
  - Ensures the passive safety of the system
  - Large prompt negative Doppler effect from the fuel
  - Core moderator effect is slightly slower and negative
  - Reflector effect is slower, small, and can be slightly positive

Modular HTGR Flux Spectrum as a Function of Operating Temperature



As core temperature Increases, the flux spectrum moves into the U-238 and Pu-240 resonance absorption cross section range.

Core Temperature Coefficient Shows Effect of Increased Resonance Absorption Over a Cycle



BOEC = Beginning of Equilibrium Cycle

MOEC = Middle of Equilibrium Cycle

EOEC = End of Equilibrium Cycle

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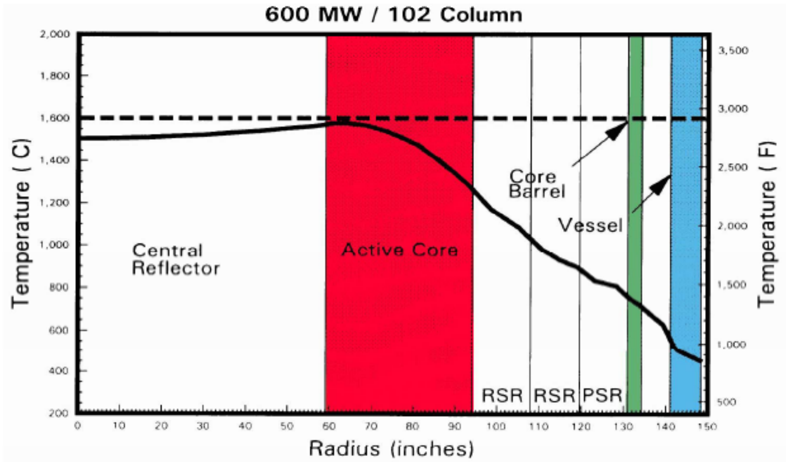
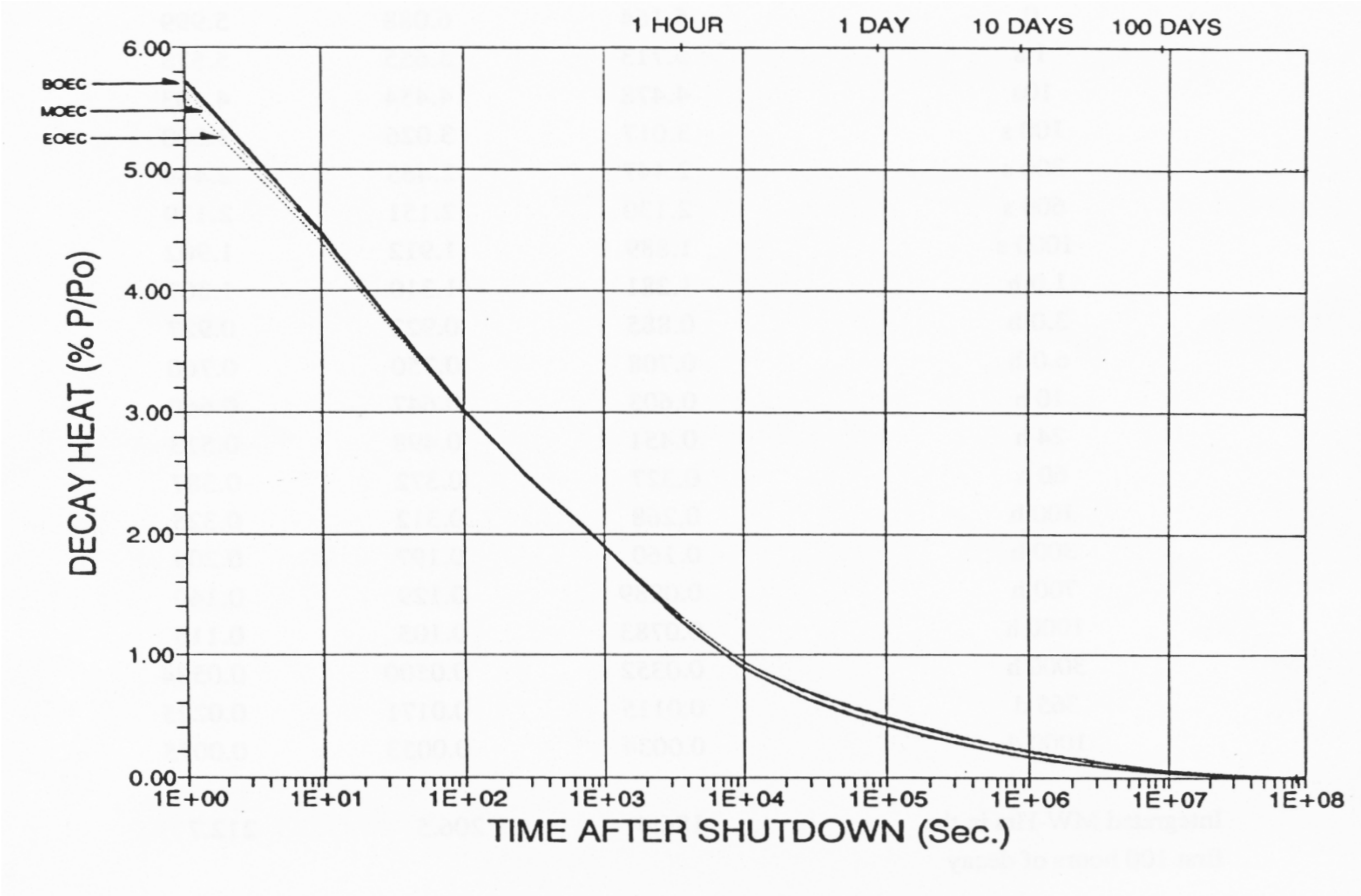


Modular HTGR Core Decay Heat

- **Core Decay Heat calculated using 1100 nuclide depletion chain model (GARGOYLE):**
  - Includes heavy metals, structure, impurities and fission products
  - GARGOYLE (0D burnup code)has been benchmarked using the ANSI LWR decay heat standard
    - Agreement to within 0.1% at all times
  - Essentially no variation in the decay heat curve during a cycle
  - Distribution of decay heat in core and reflector calculated using Monte Carlo (MCNP)
  - During heatup transients peak fuel and vessel temperatures reached between 80-120 hours after loss of forced circulation

Modular HTGR Core
Decay Heat After Shutdown During Equilibrium Cycle



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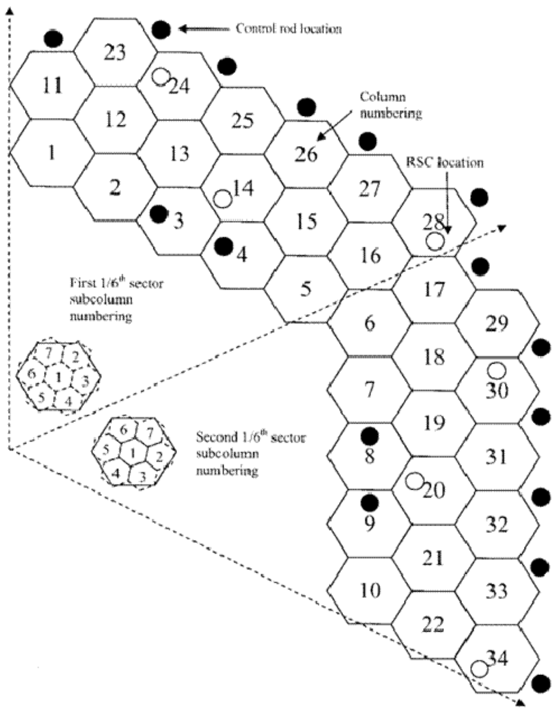
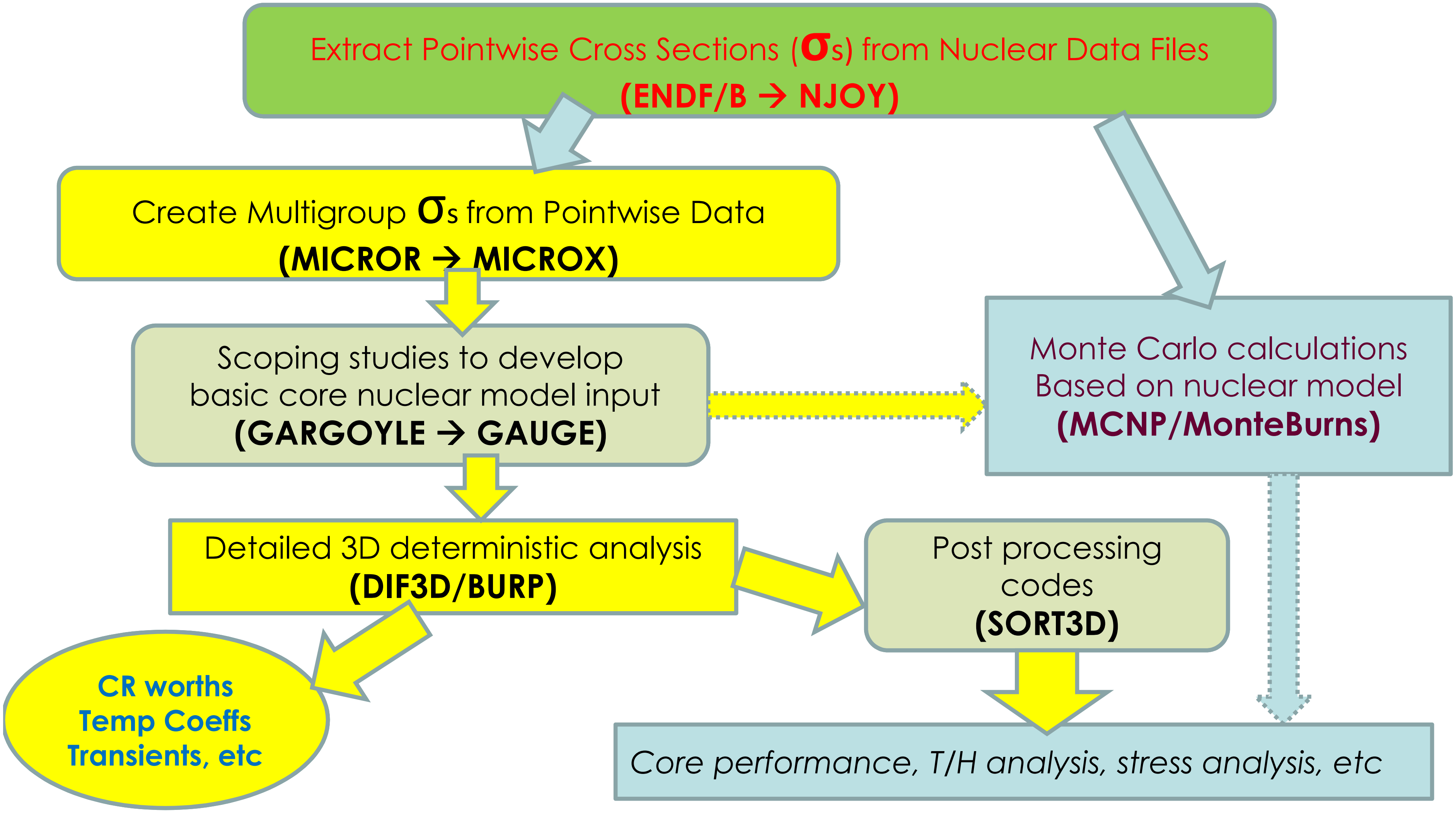


Analytic Tools Must Address Specific Prismatic HTGR Nuclear Design Issues

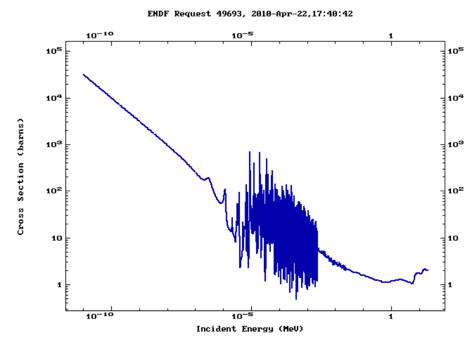
- **Accurately model the physics of HTGR cores**
  - Multiple heterogeneities (TRISO particle, fuel rods, graphite blocks)
  - Temperature dependent neutron scattering in graphite
  - Cross section resonance effects
- **Generate broad group cross sections that yield accurate results in diffusion and depletion calculations**
  - Depends on local composition
  - Strong absorbers and interface effects
  - Modular HTGRs have neutronically "thin" cores (7 to 8 mean free paths)
- **Adequately reproduce local reaction rates**
  - Modeling of lumped burnable poisons

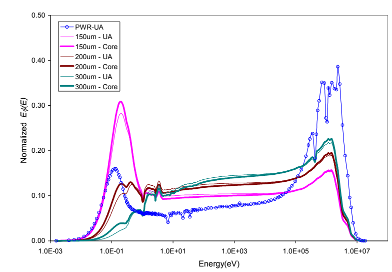
Analytic Tools for HTGR Prismatic Core Nuclear Design
Design Sequence and Code Examples



Prismatic HTGR Analytic Tools
- A Summary -

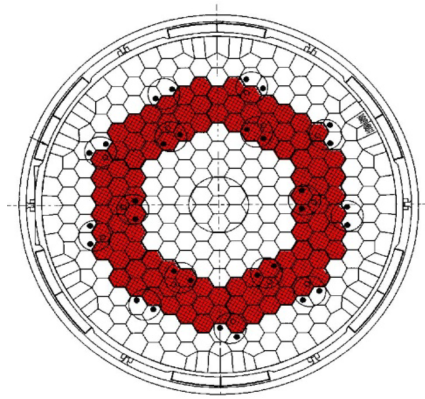
- **NJOY**
  - Extracts and process ENDF/B nuclear data
- **MICROR**
  - Develops multigroup fast and thermal cross sections from NJOY input
- **MICROX**
  - An integral transport theory flux spectrum code, which solves the neutron slowing down and thermalization equations on a detailed energy grid for a two-region lattice cell

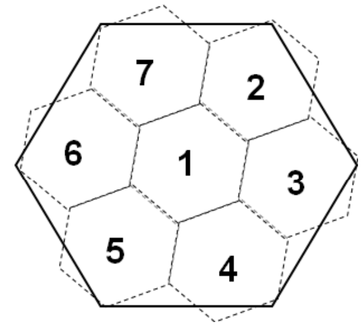




Prismatic HTGR Analytic Tools
- A Summary -

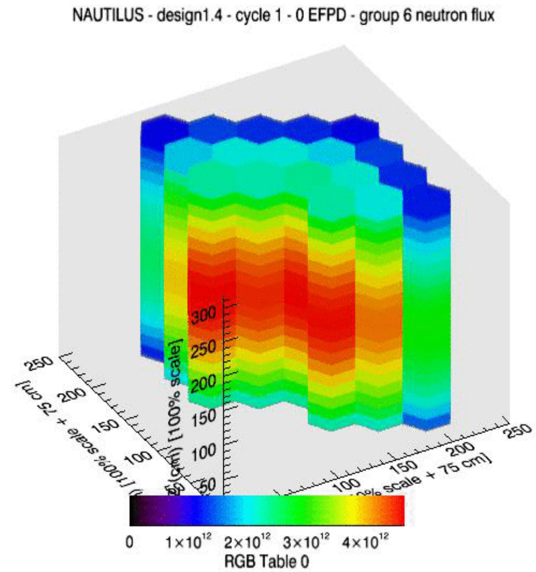
- **GARGOYLE**
  - 0D diffusion depletion code for determining core segment fuel loadings
- **GAUGE**
  - Two-dimensional few group neutron diffusion, triangular spatial mesh, depletion code
  - Can be used to calculate burnup histories for large reactors with hexagonal core configurations
- **MCNP**
  - Radiation transport code for nuclear analysis using Monte Carlo methods
- **MonteBurns**
  - Provides burnup capability for Monte Carlo calculations





Prismatic HTGR Analytic Tools
- A Summary -

- **DIF3D**
  - Solves the multigroup diffusion theory eigenvalue, adjoint, fixed source, and criticality (concentration search) problems in 1, 2, and 3 space dimensions
  - Can handle orthogonal (rectangular or cylindrical), triangular and hexagonal geometries
  - Core models at GA apply a nodal subhex geometry
- **BURP**
  - BURP (Burnup Replacement Package), designed to work in conjunction with DIF3D
  - Provides core nuclide depletion capability when used in tandem with existing static DIF3D neutron diffusion models



3-D animation sample (1/3rd core symmetry)

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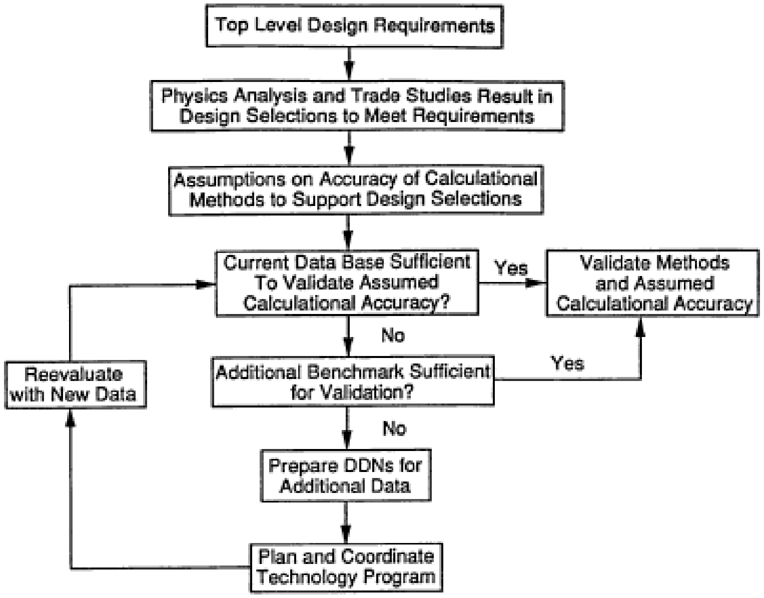


Code Verification and Validation (V&V)

- **Verification**
  - Ensures that a computer code correctly performs the mathematical operations specified in the numerical model used
  - Demonstrates substantially identical results when compared to known solutions
- **Validation**
  - Ensures that the computational method calculates the physical parameters of interest to within acceptable accuracy
  - Calculational results compared to experimental data, benchmark calculations, or results from other validated codes

Code Validation

- **GA HTGR prismatic core nuclear methods were validated for Peach Bottom and Fort St. Vrain HTGRs, and in the 1990s for the Modular HTGR design**
  - Dragon reactor startup experiments
  - Temperature-dependent graphite cross section measurements
  - Peach Bottom critical experiments and operation
  - Fort St. Vrain critical experiments, startup, and operation
  - High Temperature Lattice Test Reactor (HTLTR) measurements
  - HITREX-2 measurements
  - IAEA GCR Benchmark calculation results
  - Compact Nuclear Power Source (CNPS) measurements
  - AVR measurements
- **Re-validation, including new experiments and benchmarks, to meet latest NQA-1 requirements is required**

Results of Prior Validation of GA Modular HTGR Nuclear Design Codes

Facility	Temp. Defect	C. R. Worth	Power Distr.	K-eff	Water Ingress	Decay Heat
HEU-CORES						
Peach Bottom Critical	±14%	-11%	±10%	±0.7%	DA	-
Peach Bottom	-11% to +4%	-6% to +10%	±10%	±0.7%	-	DA
HTGR Critical	+6%	+4% to 13%	-	-0.1% to +1.0%	-	-
Fort St.Vrain	-9% to +12%	±10%	±15%	±0.5%	-	DA
HTLTR	±8%	-	-	-	-	-
KAHTER	-	DA	DA	-0.3% to +6%	±13%	-
DRAGON	DA	-11%	DA	-	-	DA
HEU/LEU CORES						
AVR	-25%	-5% to +15%	-	±11%	-	DA
LEU CORES						
HITREX-2	-	-	±10%	±0.5%	-	-
CNPS	-	-	-	±0.2%	-2% to +1%	-

DA = Data is available, but calculations have not yet been performed by GA

*** (Calculation - Experiment)/Experiment**

Fort St. Vrain Nuclear Calculations Were in Excellent Agreement with Startup Measurements

- **FSV Comparison — Calculation/Measurement**
  - Initial criticality — 1.001
  - Core shutdown margins
    - All rods in — 1.03
    - Max rod out — 0.92
    - Worth of 4 rod bank — 0.99
  - Hot-to-cold swing — 1.02
  - Core axial power distributions — ± 10%
  - Temperature defect — 1.03
- **Data confirms accuracy of methods for standard fuel cycles, and will aid in validation of Modular HTGR codes**

Accuracies Established for Early Modular HTGR 350MWt and 450MWt Nuclear Designs

Calculated Physics Parameters	PSID Allowed Calculational Uncertainty[a]
Temperature Defect	± 20%
Controlrod/bank reactivity worth	± 20%
Local power distributions	± 15%
Core reactivity (K-eff)	± 1.5%
Reactivity worth of water ingress	± 25%
Decay heat production	± 10%

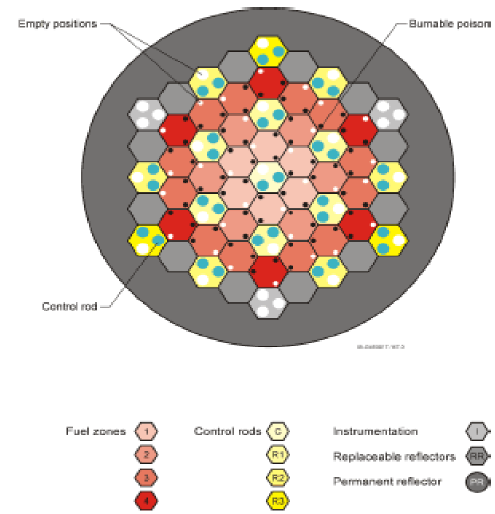
[a] Allowed 2σ standard deviation in C/E ratio

- **Allowed uncertainties based on:**
  - Validation (C to E) results
  - Sensitivity analysis to assure that safety criteria limits for the design could be met

Revalidation will Include Previous Data Plus New Benchmarks and Experiments

HTTR core benchmark

- Using both Monte Carlo and Deterministic modeling
- 30 fuel columns
- 12 replaceable reflector columns
- 16 control rod columns
- 3 instrumentation columns



Results for the IAEA Benchmark for the HTTR

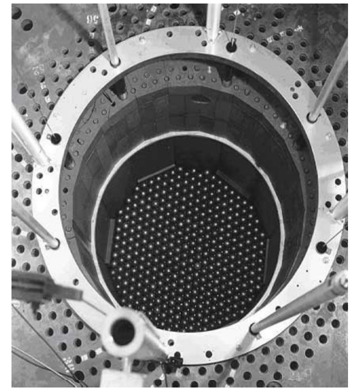
Country		Analytical results				Experimental results
		Japan	Russia	USA	France	
9MW operation	VCS heat removal	0.2 MW	0.133 MW	0.180 MW	0.178 MW	0. 22 MW
	RPV temperature (EL. 19-27 m)	~ 170 °C	165 °C	159 °C		~ 170 °C
30MW operation	VCS heat removal	0.77 MW	0.494 MW	0.67 MW	0.555 MW	0.81 MW
	RPV temperature (EL. 19-27 m)	370-380 °C	330-360 °C	330 °C		340-360 °C

Additional HTGR Experimental Data Available for HTR & HTR-10

HTR - PROTEUS

Zero-power critical facility

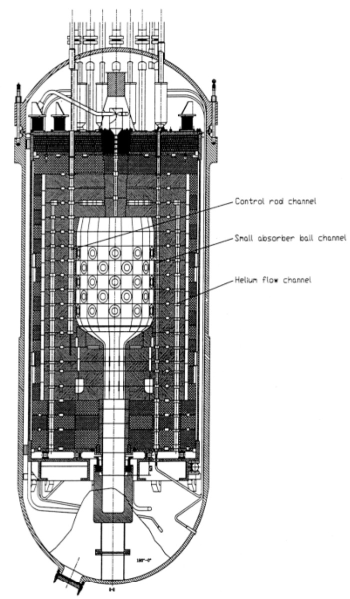
- Graphite reflector
- Core: R_c ~60 cm, H ~ 150 cm
- Fuel/mod sphere: R_s = 3 cm
- TRISO fuel with 5.966 g U/FS



HTR -10 (Beijing)

10 MW Pebble Bed Reactor

- Graphite reflector
- Core: R_c = 90 cm, H $\leq$ 197 cm
- TRISO fuel with 5 g U/Sphere
- 17% U235



Additional HTGR Experimental Data Available From ASTRA Critical Facility

Purpose

Experiments with reactor core cooling and heating

- ✓ *temperature coefficients up to 600°C*
- ✓ *critical parameters*
- ✓ *control rods worth*
- ✓ *control rods calibration characteristics*
- ✓ *spatial distribution of reaction rates*

Main technical characteristics

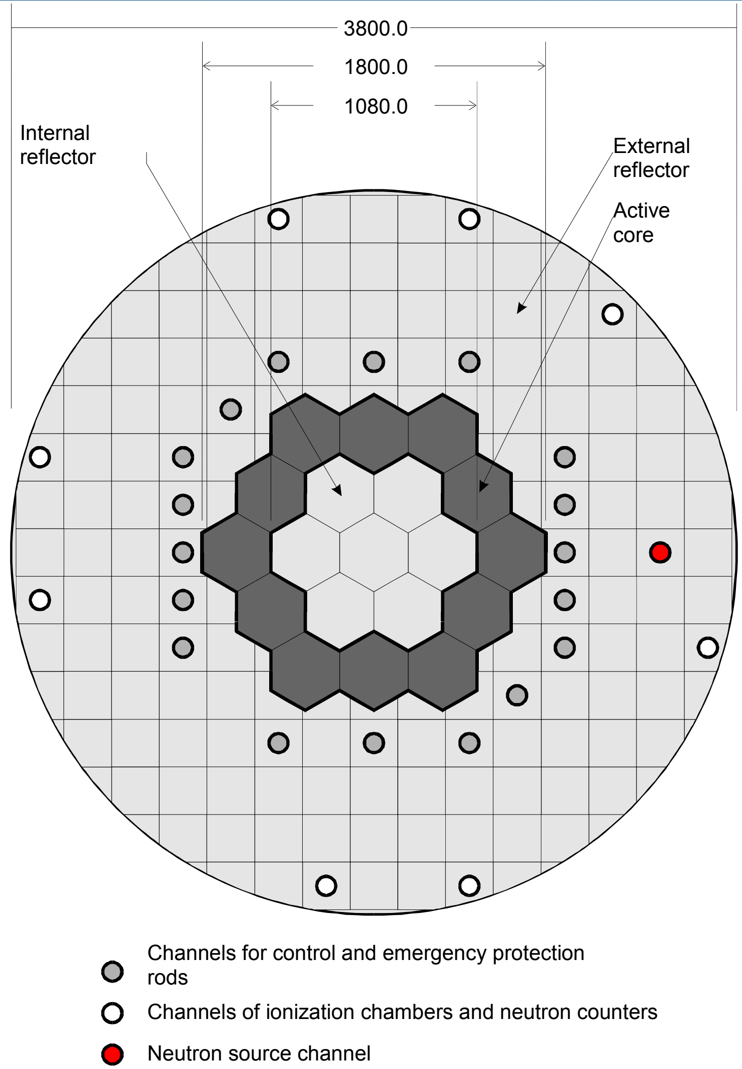
✓ *Geometry: cylinder H/D, mm*	*4600/3800*
✓ *Fuel*	
- pebble bed of spherical elements	
with diameter, mm	*60*
- with porosity from	*0.26 up to 0.39*
- Quantity of fuel elements	*≤ 50000*
- LEU with U-235 load, g/sphere	*0.51*
- enrichment, %	*up to 21*

Status

In operation

ASTRA Measurements and Planned Criticals

- **ASTRA Cold Criticals**
  - Pebble fuel
  - Reflector control rods
  - Measured
    - Core reactivity
    - Reflector control rod worth
    - Individual rod worths
    - Fission rate distributions
- **Phase 2 experiments planned for 2011**
  - Core temperature to 600°C
  - Plan to use block fuel
  - Measured
    - Core reactivity
    - Reflector control rod worth
    - Individual rod worths
    - Temperature coefficient
    - Fission rate distributions



SUMMARY

- **HTGR nuclear characteristics**
  - Physically large, but neutronically small and homogeneous
  - Relatively hard thermal neutron spectrum
  - Reactivity swing over a cycle minimized by the use of fixed, lumped, burnable poisons
- **Reactivity always decreases as core temperature increases, and is the only significant reactivity effect in the core**
  - Negative feedback effect ensures the passive safety of the system
- **Nuclear design codes have been developed and integrated for use on high temperature, gas-cooled reactors**
  - Codes have been validated with data from operating reactors and critical assemblies

Suggested Reading

- "NGNP Point Design – Results of the Initial Neutronics and Thermal- Hydraulic Assessments During FY-03," INEEL/EXT-03-00870, Revision 1, September 2003
- 450MW(t) MHTGR Core Nuclear Design, DOE-HTGR-90237, September 1993
- Reactor Physics Development Plan, DOE-HTGR-90348, Rev 0, December 1992