HTGR Technology Course for the Nuclear Regulatory Commission
May 24 – 27, 2010

Module 5b
Prismatic HTGR Nuclear Design

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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

- **Fuel is carbide-clad, small ceramic, particles not metal clad UO₂**
  - 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

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

- **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  - In-core control rods withdrawn during startup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 -

<table>
<thead>
<tr>
<th>Core</th>
<th>Modular - HTGR</th>
<th>LWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Power density, w/cc</td>
<td>5.8-6.6</td>
<td>58 - 105</td>
</tr>
<tr>
<td>Linear heat rate, kW/ft</td>
<td>1.6</td>
<td>19</td>
</tr>
<tr>
<td>Avg. therm-neutron energy, eV</td>
<td>0.22</td>
<td>0.17</td>
</tr>
<tr>
<td>Average Uranium Enrichment</td>
<td>15.5%</td>
<td>4.00%</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Moderator (at 0.025 eV)</th>
<th>Graphite</th>
<th>Water</th>
</tr>
</thead>
<tbody>
<tr>
<td>Diffusion Coefficient D, cm</td>
<td>0.86</td>
<td>0.16</td>
</tr>
<tr>
<td>Diffusion Length L, cm</td>
<td>54</td>
<td>2.75</td>
</tr>
<tr>
<td>Migration length M, cm</td>
<td>57</td>
<td>6</td>
</tr>
<tr>
<td>Collisions to thermalize</td>
<td>~18</td>
<td>~1</td>
</tr>
<tr>
<td>$\Sigma a$ (cm$^{-1}$)</td>
<td>0.00029</td>
<td>0.022</td>
</tr>
<tr>
<td>$\Sigma s$ (cm$^{-1}$)</td>
<td>0.41</td>
<td>3.45</td>
</tr>
</tbody>
</table>
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
Outline

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

TEMPERATURE COEFFICIENT [dρ/dT (10^{-4}^°C)]

FUEL TEMPERATURE (°C)
Outline

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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
Outline

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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

Extract Pointwise Cross Sections ($\sigma_s$) from Nuclear Data Files
(ENDF/B $\rightarrow$ NJOY)

Create Multigroup $\sigma_s$ from Pointwise Data
(MICROR $\rightarrow$ MICROX)

Scoping studies to develop basic core nuclear model input
(GARGOYLE $\rightarrow$ GAUGE)

Monte Carlo calculations Based on nuclear model
(MCNP/MonteBurns)

Detailed 3D deterministic analysis
(DIF3D/BURP)

Post processing codes
(SORT3D)

CR worths, Temp Coeffs, Transients, etc

Core performance, T/H analysis, stress analysis, etc
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**

- **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)](image-url)
• HTGR core nuclear design basics
• Temperature coefficients
• Decay heat
• Analytical tools
• Code verification and validation
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

<table>
<thead>
<tr>
<th></th>
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</tr>
</thead>
<tbody>
<tr>
<td><strong>HEU-CORES</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Peach Bottom Critical</td>
<td>±14%</td>
<td>-11%</td>
<td>±10%</td>
<td>±0.7%</td>
<td>DA</td>
<td>-</td>
</tr>
<tr>
<td>Peach Bottom</td>
<td>-11% to +4%</td>
<td>-6% to +10%</td>
<td>±10%</td>
<td>±0.7%</td>
<td>-</td>
<td>DA</td>
</tr>
<tr>
<td>HTGR Critical</td>
<td>+6%</td>
<td>+4% to 13%</td>
<td>-</td>
<td>-0.1% to +1.0%</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Fort St. Vrain</td>
<td>-9% to +12%</td>
<td>±10%</td>
<td>±15%</td>
<td>±0.5%</td>
<td>-</td>
<td>DA</td>
</tr>
<tr>
<td>HTLTR</td>
<td>±8%</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>KAHTER</td>
<td>-</td>
<td>DA</td>
<td>DA</td>
<td>-0.3% to +6%</td>
<td>±13%</td>
<td>-</td>
</tr>
<tr>
<td>DRAGON</td>
<td>DA</td>
<td>-11%</td>
<td>DA</td>
<td>-</td>
<td>-</td>
<td>DA</td>
</tr>
<tr>
<td><strong>HEU/LEU CORES</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>AVR</td>
<td>-25%</td>
<td>-5% to +15%</td>
<td>-</td>
<td>±11%</td>
<td>-</td>
<td>DA</td>
</tr>
<tr>
<td><strong>LEU CORES</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>HITREX-2</td>
<td>-</td>
<td>-</td>
<td>±10%</td>
<td>±0.5%</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>CNPS</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>±0.2%</td>
<td>-2% to +1%</td>
<td>-</td>
</tr>
</tbody>
</table>

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**
  - 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

<table>
<thead>
<tr>
<th>Calculated Physics Parameters</th>
<th>PSID Allowed Calculational Uncertainty&lt;sup&gt;(a)&lt;/sup&gt;</th>
</tr>
</thead>
<tbody>
<tr>
<td>Temperature Defect</td>
<td>± 20%</td>
</tr>
<tr>
<td>Controlrod/bank reactivity worth</td>
<td>± 20%</td>
</tr>
<tr>
<td>Local power distributions</td>
<td>± 15%</td>
</tr>
<tr>
<td>Core reactivity (K-eff)</td>
<td>± 1.5%</td>
</tr>
<tr>
<td>Reactivity worth of water ingress</td>
<td>± 25%</td>
</tr>
<tr>
<td>Decay heat production</td>
<td>± 10%</td>
</tr>
</tbody>
</table>

<sup>(a)</sup> 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

<table>
<thead>
<tr>
<th>Country</th>
<th>Analytical results</th>
<th>Experimental results</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Japan</td>
<td>Russia</td>
</tr>
<tr>
<td>9MW operation</td>
<td>VCS heat removal</td>
<td>0.2 MW</td>
</tr>
<tr>
<td></td>
<td>RPV temperature (EL. 19-27 m)</td>
<td>~ 170 °C</td>
</tr>
<tr>
<td>30MW operation</td>
<td>VCS heat removal</td>
<td>0.77 MW</td>
</tr>
<tr>
<td></td>
<td>RPV temperature (EL. 19-27 m)</td>
<td>370-380 °C</td>
</tr>
</tbody>
</table>
Additional HTGR Experimental Data Available for HTR & HTR-10

**HTR - PROTEUS**
Zero-power critical facility
- Graphite reflector
- Core: $R_c \sim 60$ cm, $H \sim 150$ 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