High Temperature Gas-cooled Reactor: Core Design

Advanced Reactor Technologies

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NRC HTGR Training July 16-17, 2019

HTGR Core Design – Overview

- General Attributes of Modular Prismatic and Pebble Bed HTGRs
  - Common features and physics
  - Neutronics
  - Prismatic and Pebble Fuel
  - Thermal-Fluidics
  - Inherent Safety
- Plant Systems and Power Conversion
  - Reactivity Control
  - Instrumentation and Control
  - Helium Conditioning
  - Power Conversion
- Normal Operation and Power Maneuvers

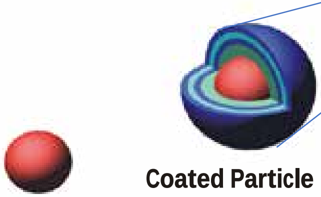
Fort St. Vrain Fuel Blocks
(General Atomics)

Gun drilling long holes in Ft. St. Vrain fuel elements
Today - drilled with numerically controlled machines

Vollman, R. Prismatic HTGR Core Design Description, Module 5A -HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

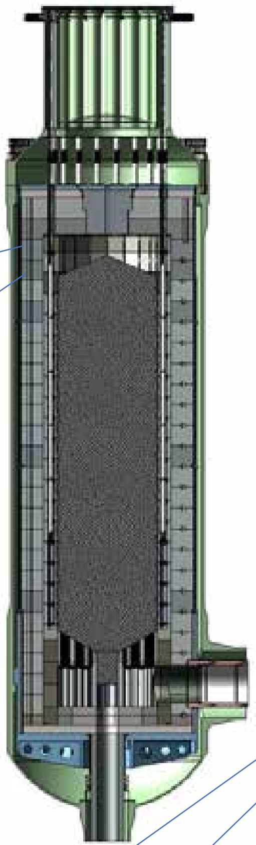
Modular HTGR(s) from the bottom up

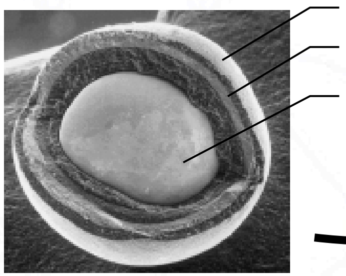
- UO_2 or UCO (ceramic) kernels
- Tristructural isotropic (actually 4 layers around the kernel)
- Pressed into a semi-graphitic matrix and shaped into either ‘compacts’ or pebbles
- Cylindrical or annular cores

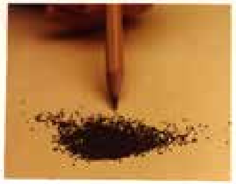


Coated Particle

Pebble



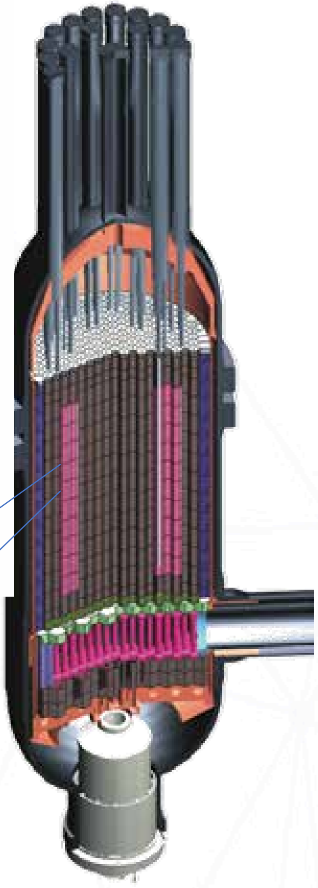




Particles

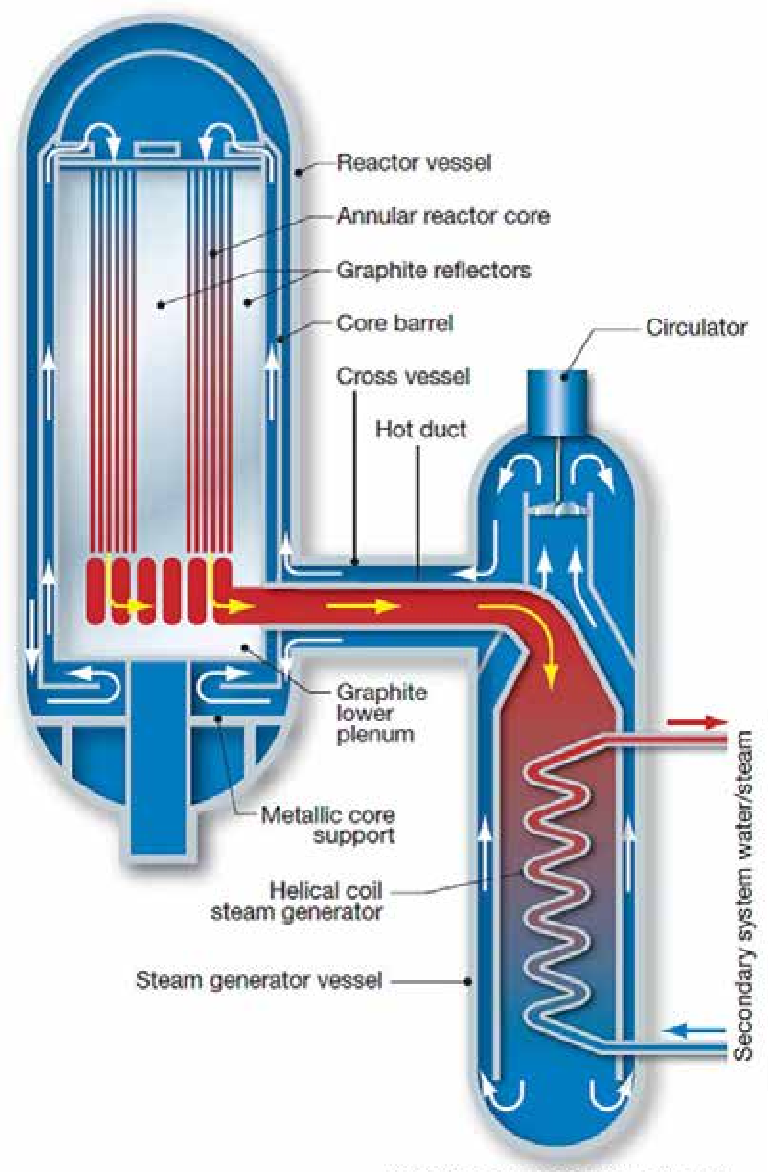
Compacts

Fuel Elements



Relevant Attributes of Modular HTGRs

- Graphite-moderated and reflected
- Cooled (usually) by helium (~7 MPa)- Molten salt is being explored (and nitrogen has been proposed)
- Large DT_c (>400°C) across the core (top to bottom) compared to 30°C for an LWR
- Fuel: TRISO fuel particles in a carbonaceous matrix
- Uninsulated reactor vessel
- Large aspect ratio: heat escapes radially via conduction and radiation if forced cooling is lost. This attribute also limits the power density (~400 MWt for PBRs; ~600MWt for prismatic reactors)
- Slow temperature response during accidents (high heat capacity and low power density)



(1 of 2 steam generators shown)

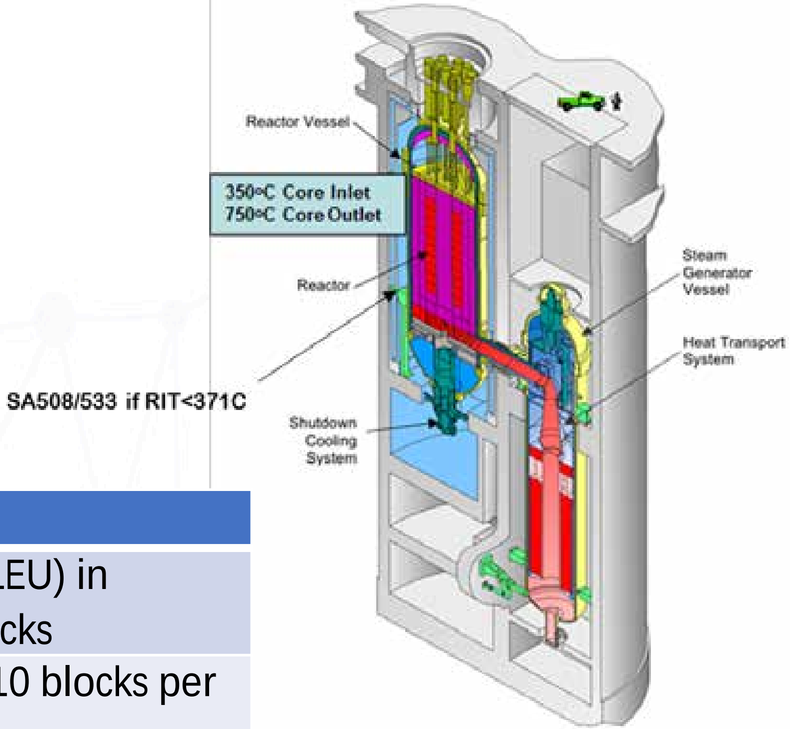
LWRs vs HTGRs in a Nutshell

Item	HTGR	LWR
Moderator	Graphite	Water
Coolant	Helium	Water
Average coolant exit temperature	700-950°C	310°C
Structural material	Graphite	Steel
Fuel clad	SiC and PyC	Zircaloy
Fuel	UO_2, UCO	UO_2
Fuel damage time at temperature	UCO - No failures for at least 150 hrs @ 1800°C*	1260°C
Power density, W/cm^3	4 to 6.5	58-105
Migration Length, cm	6	57

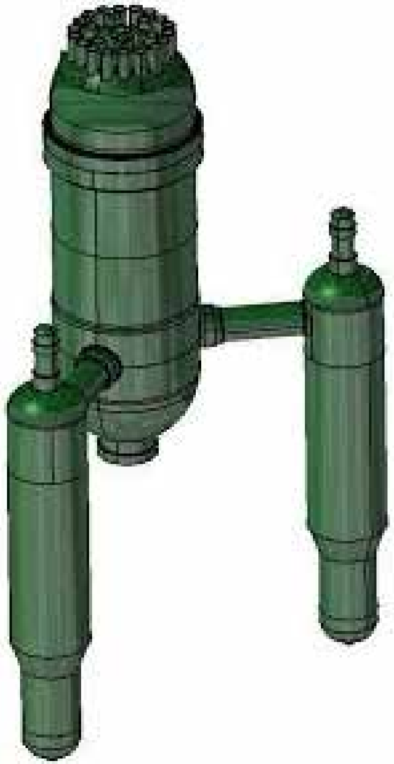
* Not a hard limit; based on statistical failure rates. Typical duration of peak fuel temperature is less than 100 hrs for a Loss of Forced Cooling event

Shenoy, A.. (General Atomics) History and Evolution of HTGRs, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Common Primary Loop Features – Framatome Steam Cycle-HTGR

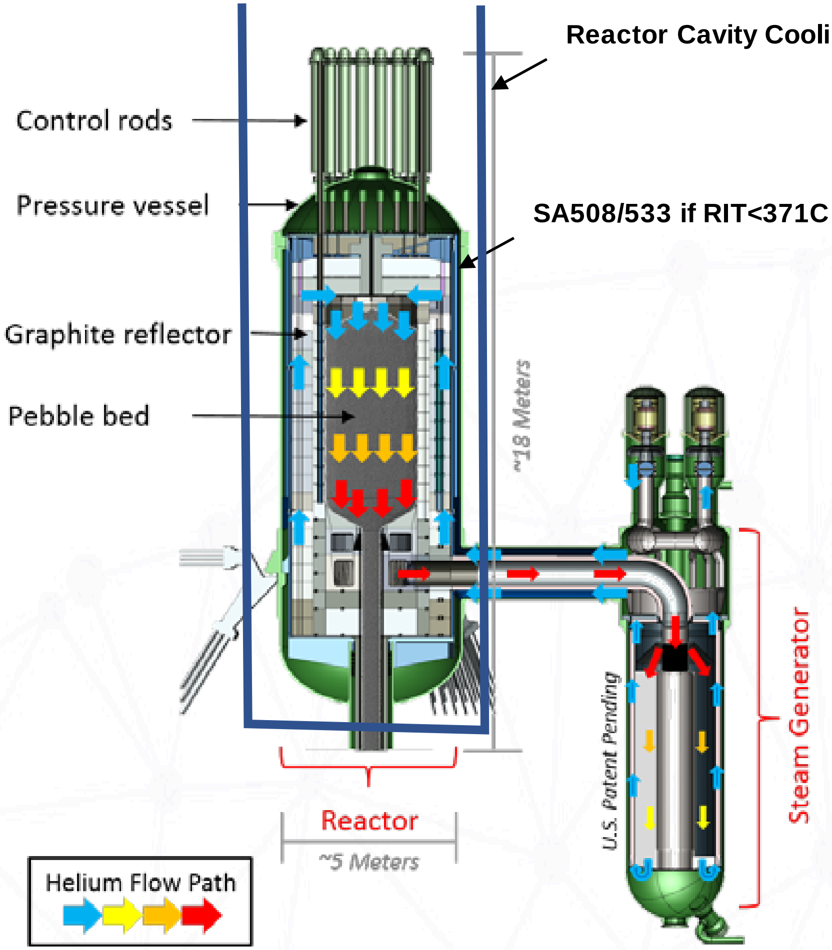


Parameter	
Fuel	TRISO (<20% LEU) in Compacts/Blocks
Core Geometry	102 columns,10 blocks per column
Reactor Power	625 MWt
Reactor Outlet Temperature	750°C
Reactor Inlet Temperature	325°C
Primary	He at 6 MPa
Secondary (x2)	Steam @ 16.7 MPa, 566°C



Framatome 625 MWt Prismatic SC-HTGR *(framatome.com) – Heat Transport System (HTS) supports process heat applications*

Common Primary Loop Features



Reactor Cavity Cooling System

SA508/533 if RIT<371C

Xe-100 200 MWt Pebble Bed HTGR
(x-energy.com)

Parameter	
Fuel	TRISO (~15% LEU) in Pebbles
Core Geometry	~300K Pebbles in a Cylindrical Bed
Reactor Power	200 MWt
Reactor Outlet Temperature	750°C
Reactor Inlet Temperature	260°C
Primary	He at 6 Mpa
Secondary (×2)	Steam at 16.5 MPa, 565°C

Reactor (Vessel) Cavity Cooling System

- Active or passive heat removal via absorption of thermal radiation (shine) emitted from a hot uninsulated reactor pressure vessel
- Ultimately rejects heat to the atmosphere
- Air-cooled, water-cooled, or hybrid configurations

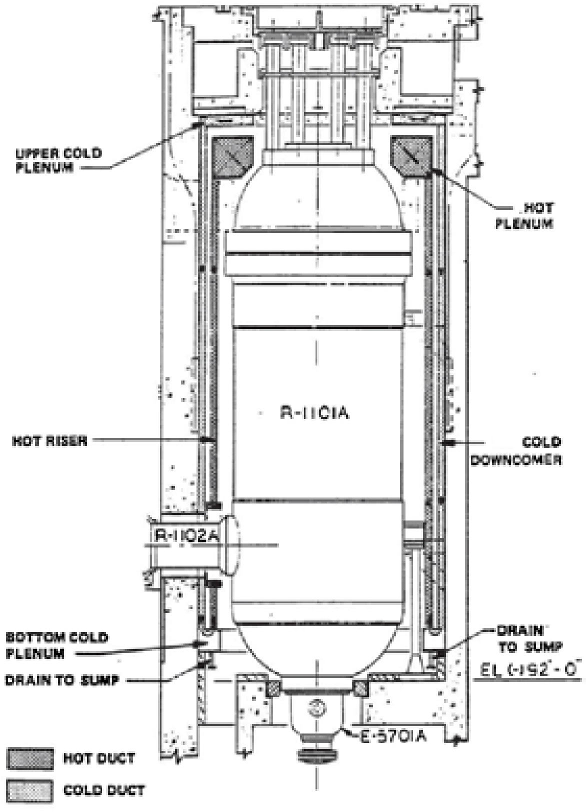
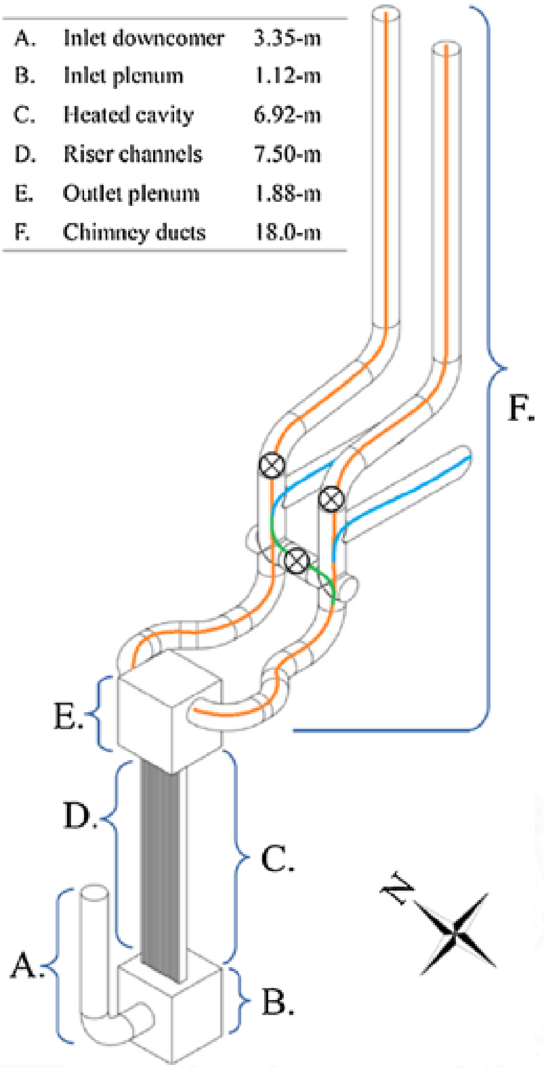


FIGURE 5.5-6
RCCS COOLING PANEL CONFIGURATION – ELEVATION
HIGH TEMPERATURE GAS-COOLED REACTOR PRELIMINARY SAFETY INFORMATION DOCUMENT HTGR-86-024



A.	Inlet downcomer	3.35-m
B.	Inlet plenum	1.12-m
C.	Heated cavity	6.92-m
D.	Riser channels	7.50-m
E.	Outlet plenum	1.88-m
F.	Chimney ducts	18.0-m

Lisowski, D.D. et al, Experimental Observations of Natural Circulation Flow in the NSTF, Nuclear Engineering and Design 306, (2016) 124-132.

Physics of HTGRs

Graphite dominates

Core composition
HTR-PM

	v/o	m/o
Carbon	60.6	96.0
Helium	39.0	0.2
UO_2	0.4	3.8

- **Neutronics**
  - Core looks very homogeneous and diffusive, longer mean free path
  - Slightly harder spectrum than LWRs (more negative temperature feedback)
  - Good Pu-burner but MA buildup is high
- **Thermal-fluidics and Accident Behavior**
  - Graphite acts as a thermal buffer – absorbs heat during reactivity insertions and conducts (or radiates) it away
  - Time constant is much longer than neutronics
- **Mechanical**
  - Holds the core together and ‘creeps’ to relieve stress
- **Fission Product Retention in fuel element (block)**
  - Holds much of what little FP escape from the TRISO fuel
- **Spent fuel**
  - Large volume, low heat production, geochemically stable

Massimo,L. “The Physics of High Temperature Reactors”, ebook ISBN 9781483280288

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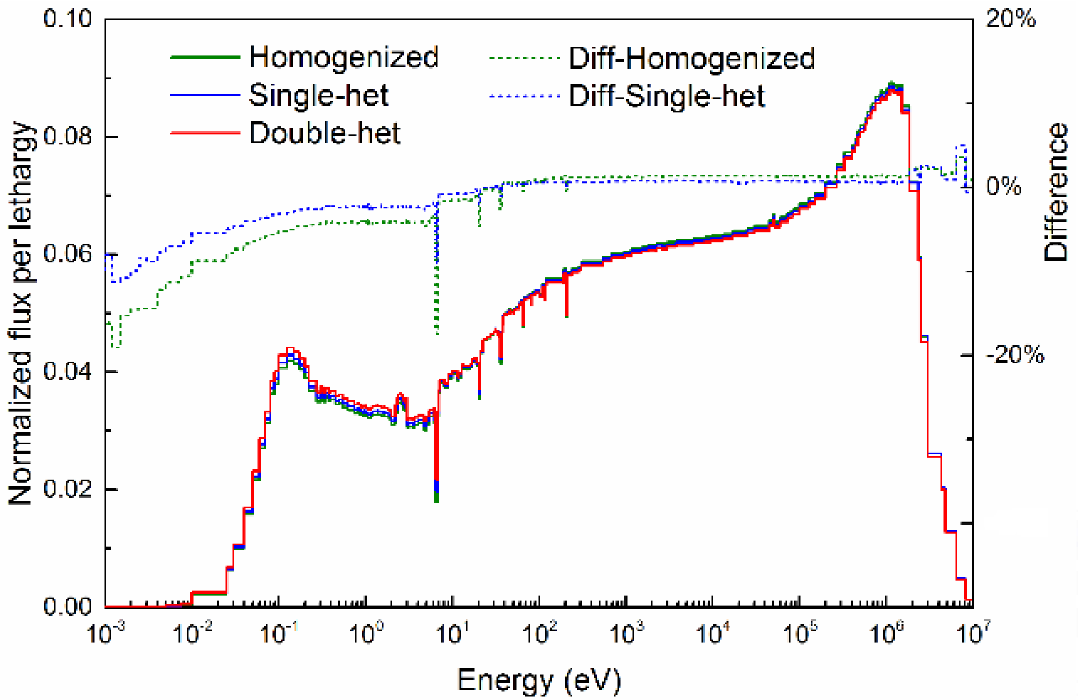
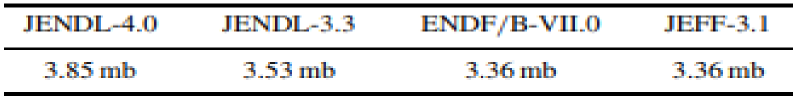


Fu & Hou, Jason & Ivanov, Kostadin. (2016). Effect Of Double Heterogeneity Treatment On Neutronics Modeling of HTGR Unit Cell.

Core Neutronics

0.0253eV capture cross section of C-12



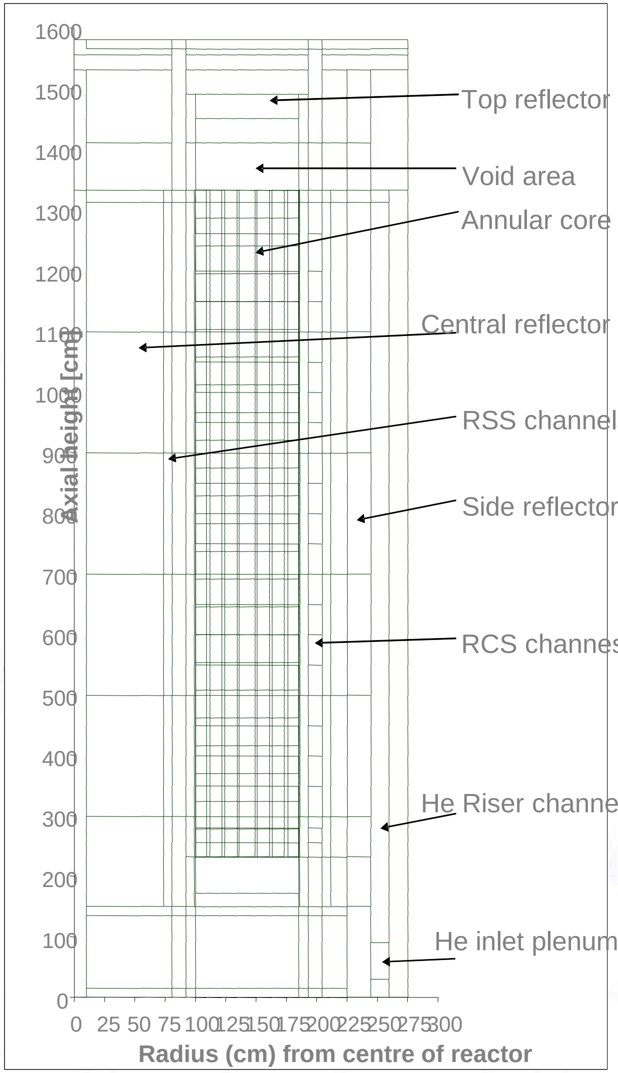
JENDL-4.0	JENDL-3.3	ENDF/B-VII.0	JEFF-3.1
3.85 mb	3.53 mb	3.36 mb	3.36 mb

Goto, et al, Impact of revised thermal neutron capture cross section of carbon stored in JENDL-4.0 on HTTR criticality cross section, J of NS&T, Jan. 2012.

- So much graphite…
  - Criticality benchmark evaluations (Bess, 2014) frequently overpredicted k_{eff} by several hundred pcm until new measurements dropped s_c by ~0.3 mb. (under-prediction resulted)
  - Relatively large uncertainties in neutronic calculations (e.g. XS input uncertainties lead to ~600 pcm keff uncertainty (1 std.dev))
  - Fortunately, safety parameters (e.g. rod worth, power peaking) are largely insensitive (e.g. <1.5% variation in local block power) to these XS uncertainties (Strydom, 2018)
- Large temperature and burnup variation along *z*
  - Need to discretize the core along *z.*
  - Must couple (at least loosely) to thermal-fluidics
- Large mean free path (mfp)
  - Neutronic coupling between blocks or pebble bed ‘zones’ – single assembly lattice calculations do not capture the leakage effects

John D. Bess, Leland M. Montierth, Oliver Köberl and Luka Snoj (2014) Benchmark Evaluation of HTR-PROTEUS Pebble Bed Experimental Program, Nuclear Science and Engineering, 178:3, 387-400,DOI: 10.13182/NSE14-13

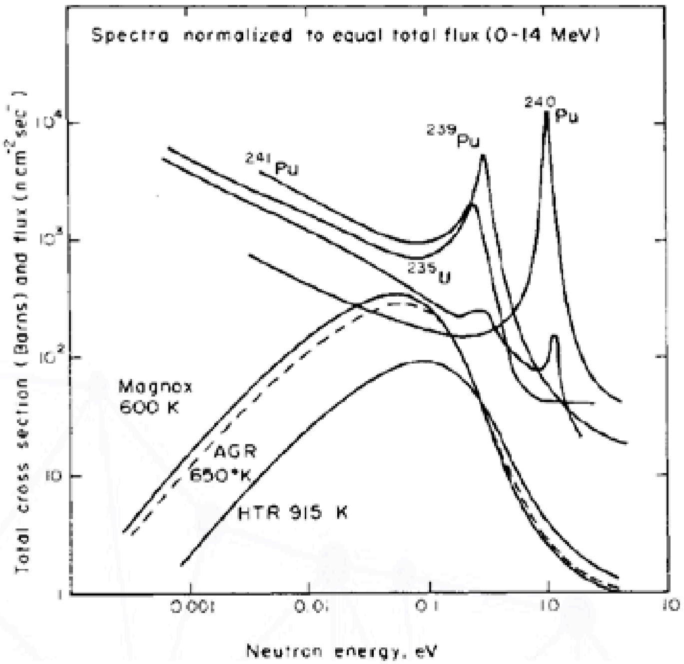
G. Strydom, P. Rouxelin (2018). IAEA CRP on HTGR UAM: Propagation of Phase I cross section uncertainties to Phase II neutronics steady state using SCALE/SAMPLER and PHISICS/RELAP5-3D. Proc. of HTR2018, Warsaw, Poland.



Typical R-Z discretization of the PBMR-400 Core

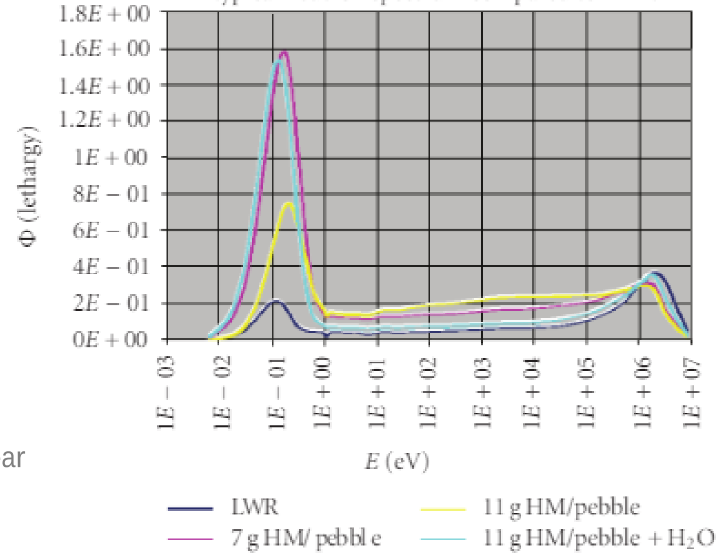
Graphite vs. H_20 as Moderators

	H2O	Graphite
Average Thermal Energy (eV)	0.17	0.22
Enrichment %	3-5	8-16
Moderating Ratio $(\xi\Sigma_s/\Sigma_a)$	62	216
# scatters to thermal	~18	~114
Mean free path (cm)	0.3	3.9
Migration Length (cm)	57	6



HTR-pebble bed

typical neutron spectrum compared to LWR



- Greater buildup of minor actinides
- Stronger negative fuel temperature feedback
  - HTGR: -7 pcm/K
  - PWR: -1 to -4 pcm/K

Bomboni, Eleonora and Cerullo, Nicola and Lomonaco, Guglielmo and Romanello, Vincenzo. (2008). A Critical Review of the Recent Improvements in Minimizing Nuclear Waste by Innovative Gas-Cooled Reactors. Science and Technology of Nuclear Installations. 10.1155/2008/265430.

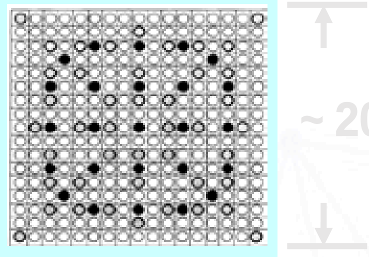
Relative Size (mfp) of Fuel and Core

mean
free path

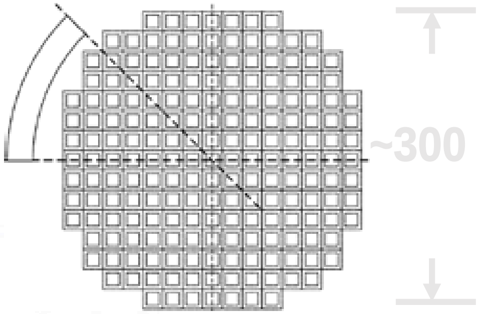
LWR

1 cm

Assembly



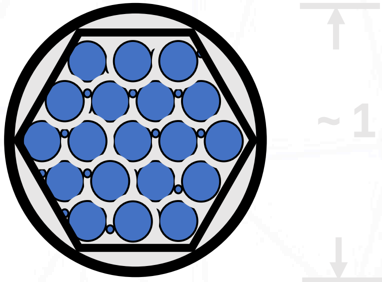
Core

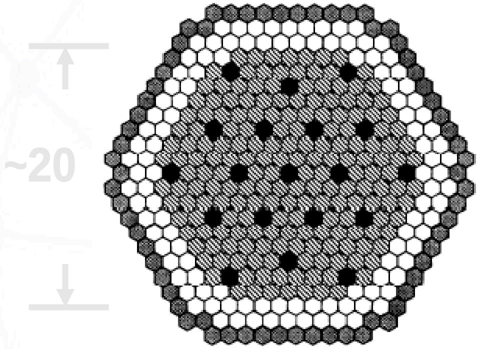


weak
coupling,
strong
local
resolution

SFR

5-8 cm

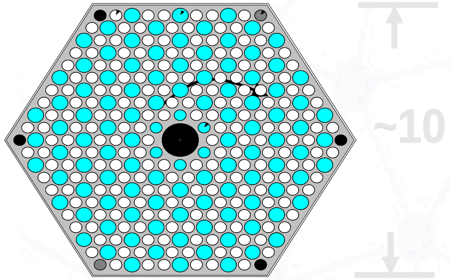




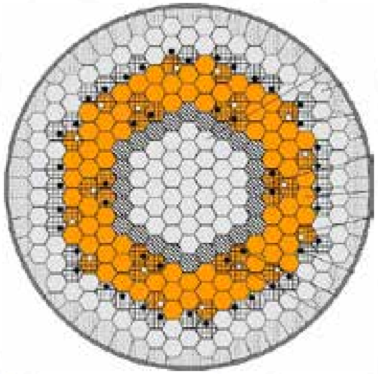
Strong
coupling,
weak local
resolution

HTGR

3-4 cm

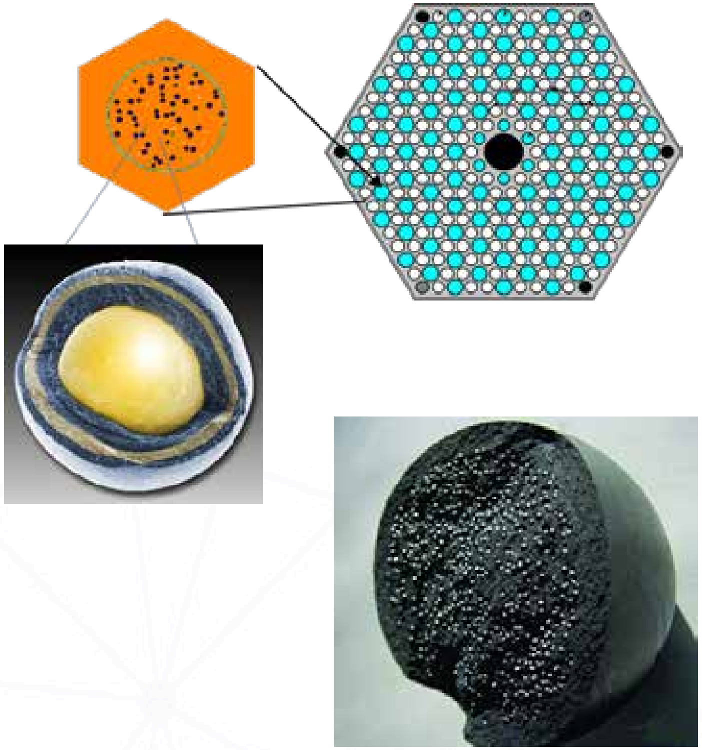


*Moderate
coupling,
Moderate local
resolution*



Cross-Generation Considerations

- 3 or 4 levels of heterogeneity
- More scattering in the resonance region
- Long migration area
- Reflectors (and control rods in them)
- Uncertainties in nuclear data
- Good agreement can be obtained by using:
  - More groups (8-26)
  - A supercell method for capturing leakage and generating cross sections for the control rod regions in the reflector
  - ‘SuperHomogenization’ or discontinuity factors for harmonizing transport and diffusion reactor rates
  - Discretize in the axial dimension



H. Gougar, A. Ougouag, W. Yoon, “Multiscale Analysis of Pebble Bed Reactors,” Proceedings of 5th International Topical Meeting on High Temperature Reactor Technology, (HTR 2010), Prague, 2010.
Laboure, V., Ortensi, J., an Hummel, A., :HTTR 3-D Cross-Section Generation with Serpent and MAMMOTH, INL/EXT-18-51317, September 2018.

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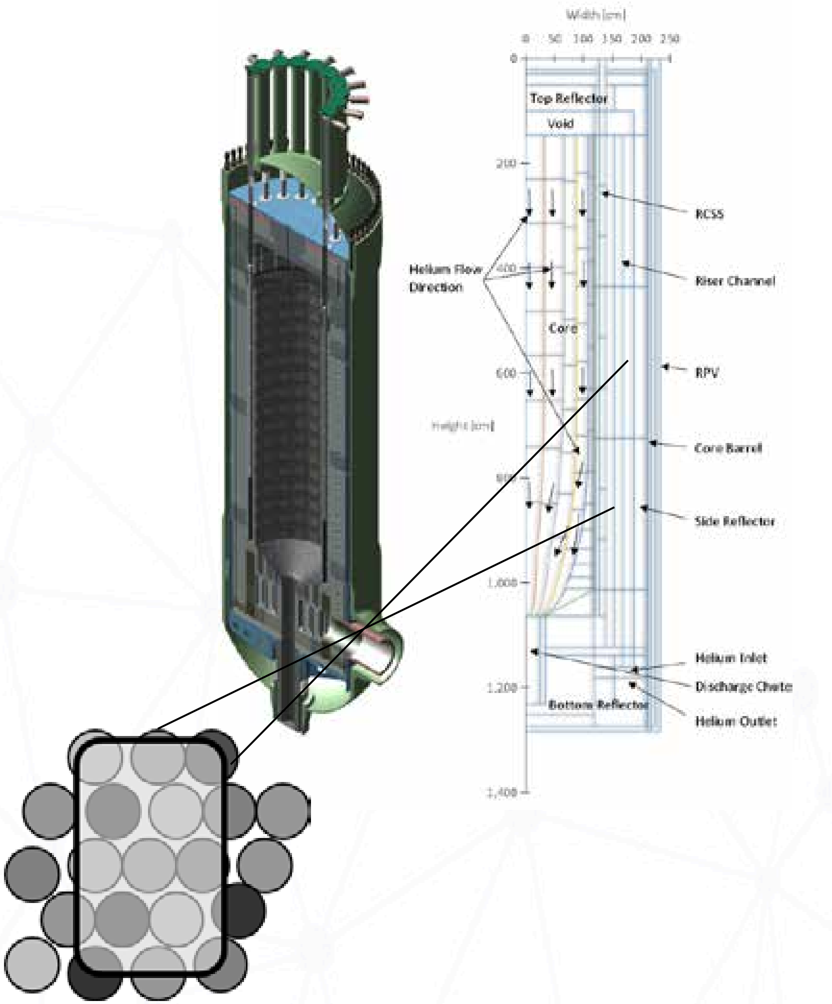
Fuel Elements in HTGRs

Pebble Bed Fuel Considerations

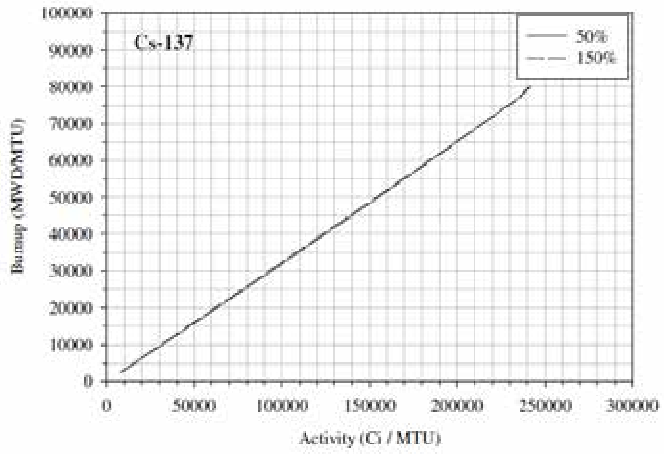
- Lack of ‘natural’ assemblies; cross sections are computed for somewhat arbitrarily chosen ‘spectral zones’ to account for variations in temperature and composition
- Fuel movement and reshuffling
    - Loaded from the top (unless it’s cooled with molten salt)
    - Pebbles roughly follow axial flowlines; radial motion toward a discharge chute. Burnup is solved along these.
    - Partially burnt pebbles sent back to the top (requires online burnup measurement)
    - If the power and fuel pebble design are kept constant, eventually the core reaches an equilibrium burnup profile
    - Online fueling allows for a very low excess reactivity
    - Analysis of the ‘Running-in’ Period (which can be a few years) poses a challenging design problem

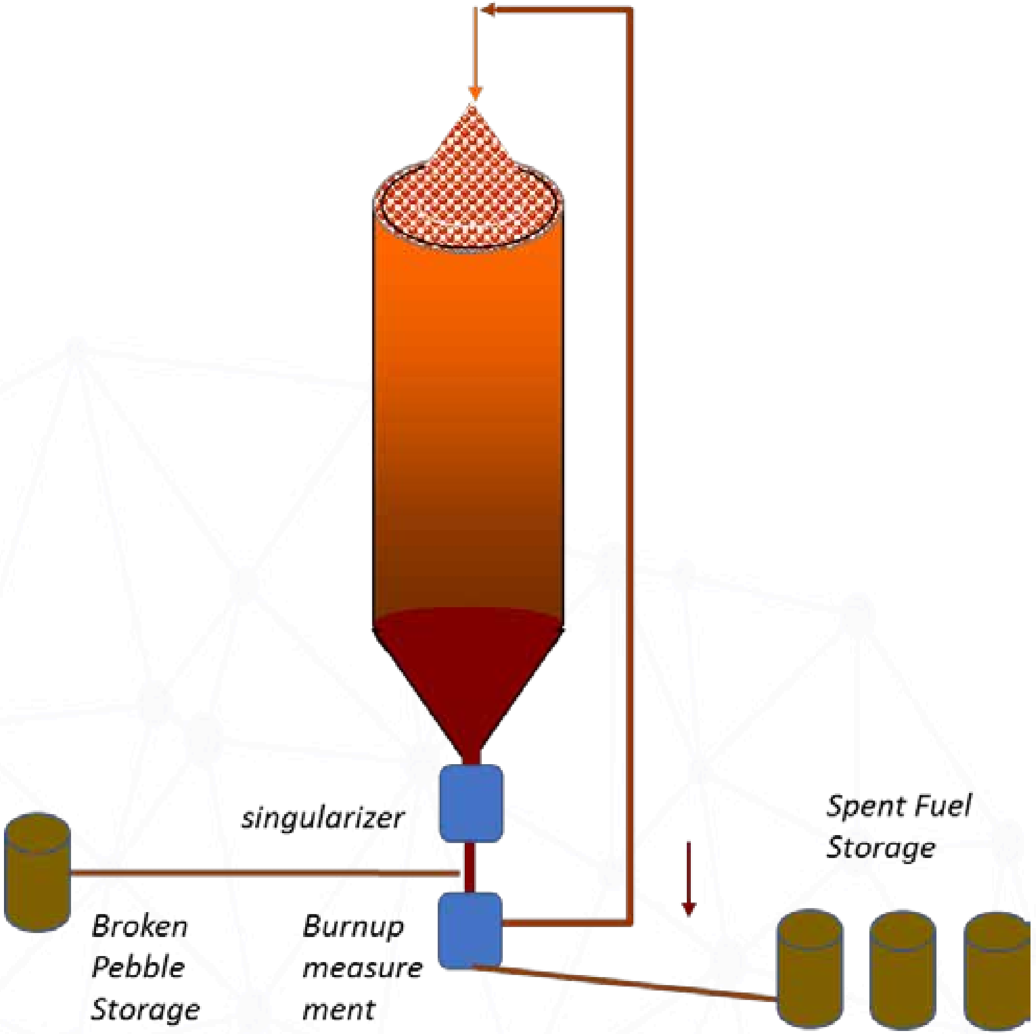


A.M. Ougouag, H.D. Gougar, R.S. Sen, “Identification of Spectral Zone Boundaries in Pebble Bed Reactors,” Proceedings of 9th International Topical Meeting on High Temperature Reactor Technology, (HTR 2018), Warsaw, 2018.

PBR Fuel Handling

- Pneumatic transfer
- Burnup Measurement
- Spent Fuel Storage





singularizer

Spent Fuel Storage

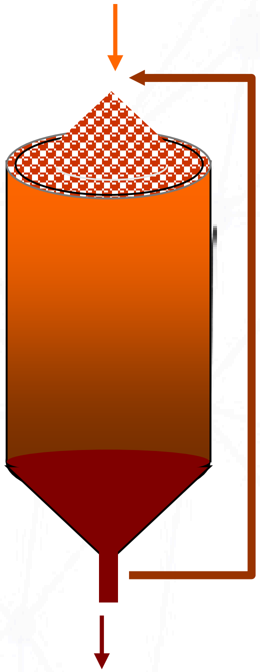
Broken Pebble Storage

Burnup measure ment

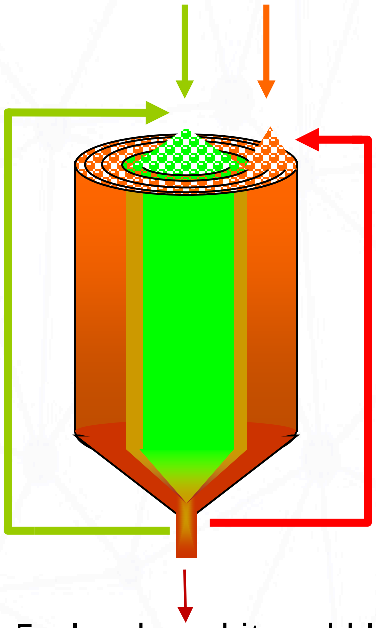
Su, Bingjing and Zhao, Zhongxiong and Chen, Jianwei and I. Hawari, Ayman. (2006). Assessment of on-line burnup monitoring of pebble bed reactor fuel by passive neutron counting. Progress in Nuclear Energy - PROG NUCL ENERGY. 48. 686-702. 10.1016/j.pnucene.2006.06.013.

PBR Fuel Zoning possibilities

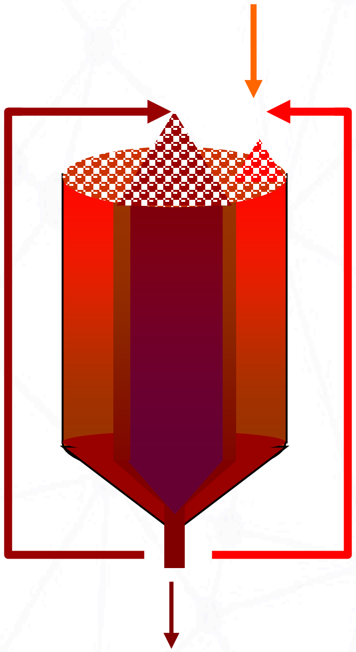
- Pebble flow is largely axial and incompressible
- Mixing between ‘streamlines’ is minimal, allowing (for most design and analysis purposes) the Bateman equation to be solved along the flow lines
- Flow is subjected to drag forces along reflector walls (variable residence time)
- Cylindrical or annular cores, multiple pebble types, and different loading patterns are possible (cylindrical vessels with a single pebble type are the most common)



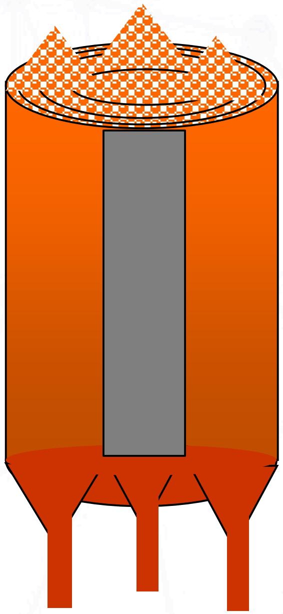
Single pebble type
Cylindrical core (Xe-100)



Fuel and graphite pebbles
Cylindrical core (early PBMR concept)



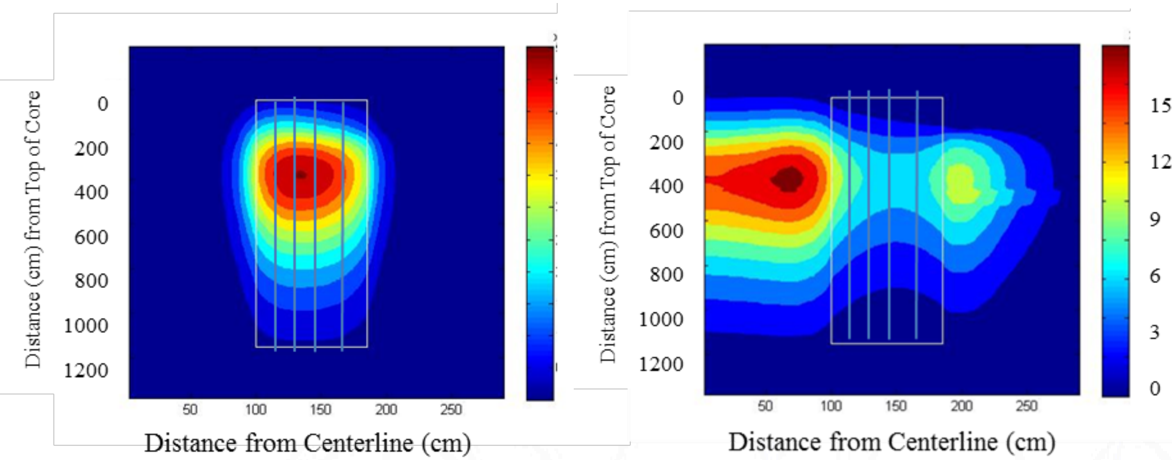
Single pebble type
with burnup zoning (THTR)

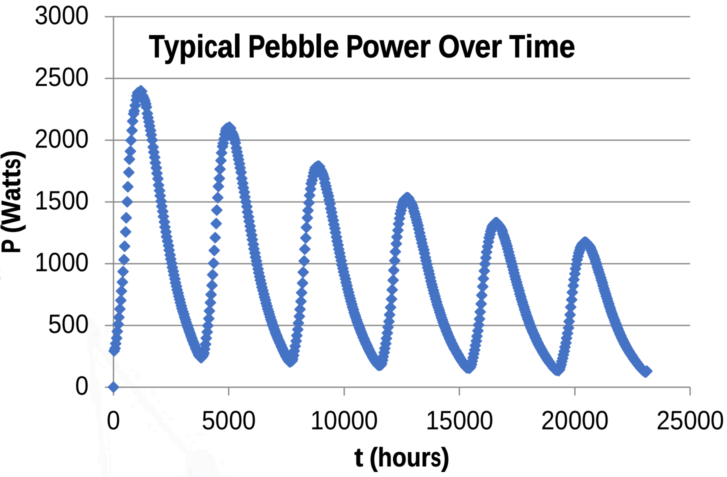


Single pebble type
Annular core (PBMR-400)

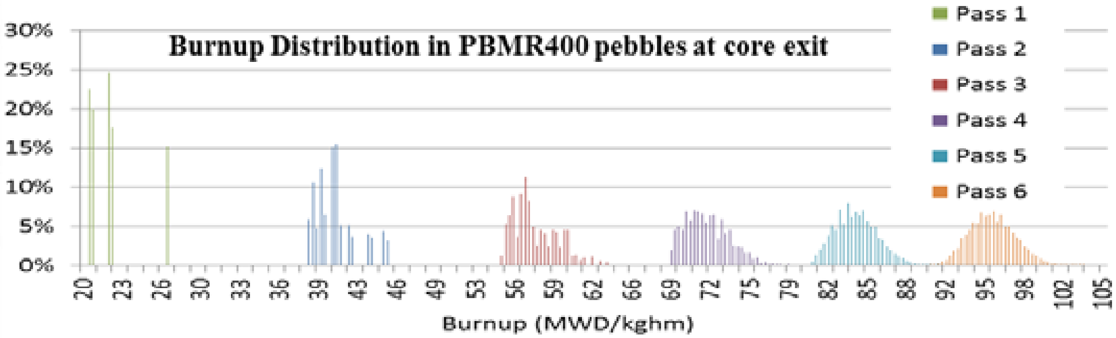
Resulting Profiles

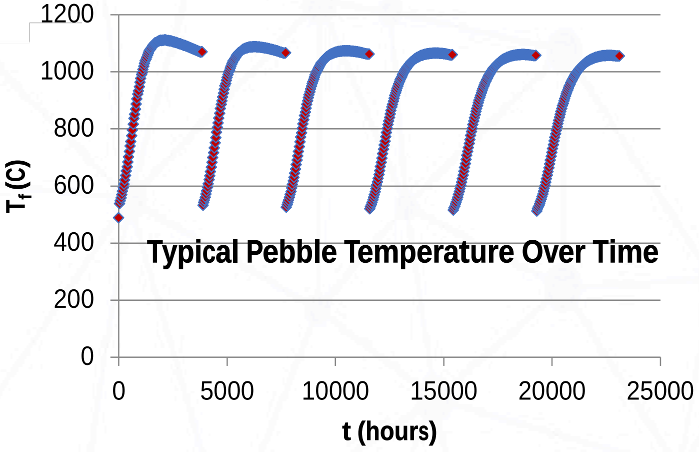
Fast and thermal flux profiles in the PBMR-400 equilibrium core (6-pass core with inner reflector)





Spectral variations leads to a burnup distribution in pebbles leaving the core

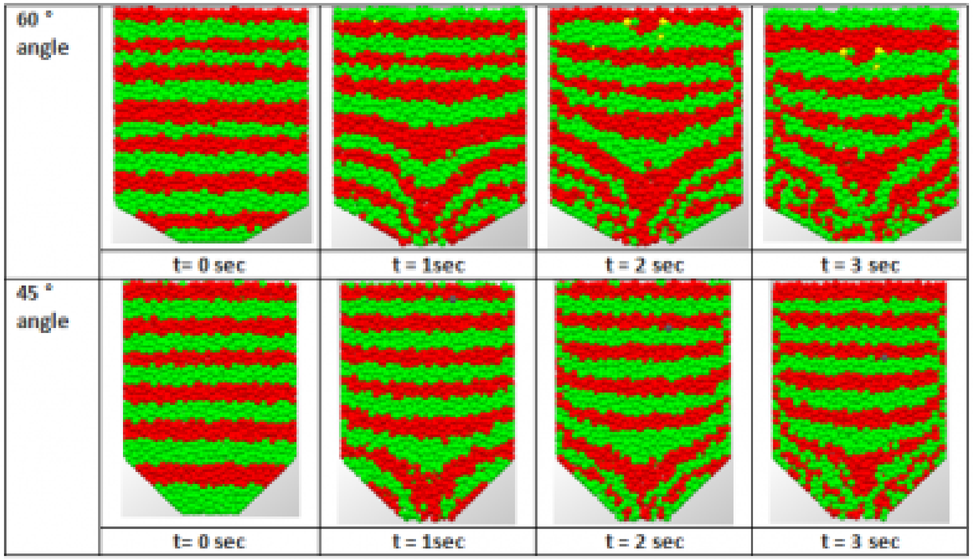




Gougar, H.D., Ougouag, A. M., Terry, W. K., and Ivanov, K. I., "Automated Design and Optimization of Pebble Bed Reactor Cores, Nuclear Science and Engineering, Oct. 2008.

PBR Fuel Flow Modeling

- Inter-pebble and pebble-wall friction and the geometry of the vessel lead to non-uniform radial flow patterns
- Flow lines were originally determined experimentally; now DEM codes are used (PEBBLES, LIGGGHTS- LAMMPS, PFC-3D)
- Earthquakes can be modeled



Cogliati, J., "PEBBLES: A Computer Code for Modeling Packing, Flow, and Recirculation of Pebbles in a Pebble Bed Reactor," Proceedings of 5th International Topical Meeting on High Temperature Reactor Technology, (HTR 2010), Prague, 2010.

C. H. Rycroft, G. S. Grest, J. W. Landry, and M. Z. Bazant, Analysis of Granular Flow in a Pebble-Bed Nuclear Reactor, Phys. Rev. E 74, 021306 (2006).
PFC3D – Itasca Consulting Group.

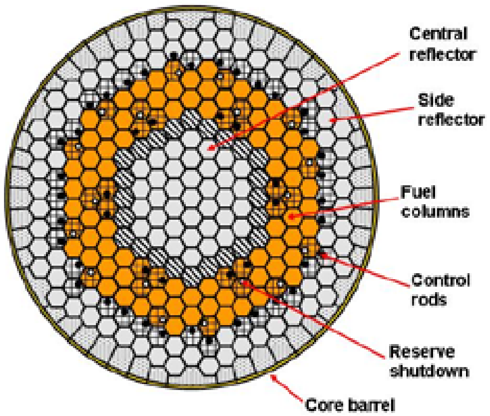
More on Pebble Motion

- Earthquakes cause pebble bed to settle
- A settling induced reactivity insertion and subsequent power transient require~~s~~ some computational horsepower to simulate
- Fortunately this does not appear to be much of a safety issue – temperature feedback shuts down the reactor with a relatively mild heatup
- Block shifting may interfere with control rod motion
- The real hazards from earthquakes are the stress put on pipes and other components

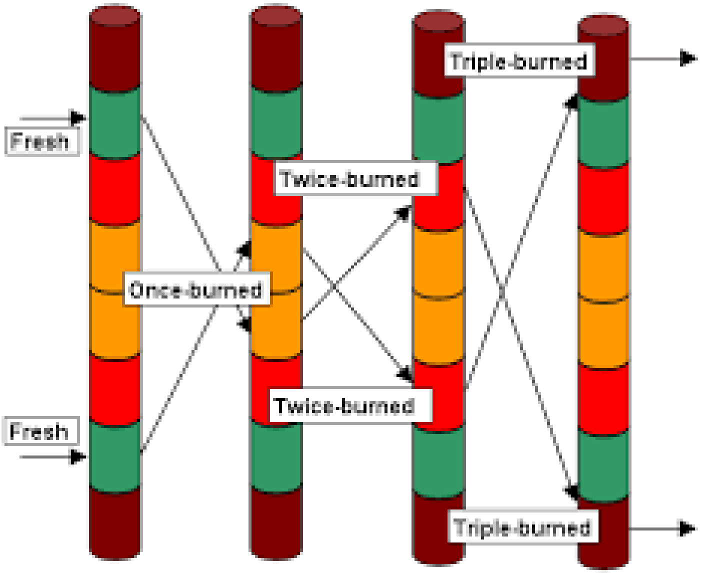
Solid Volume Fraction of randomly-packed spheres = ~0.59-0.64

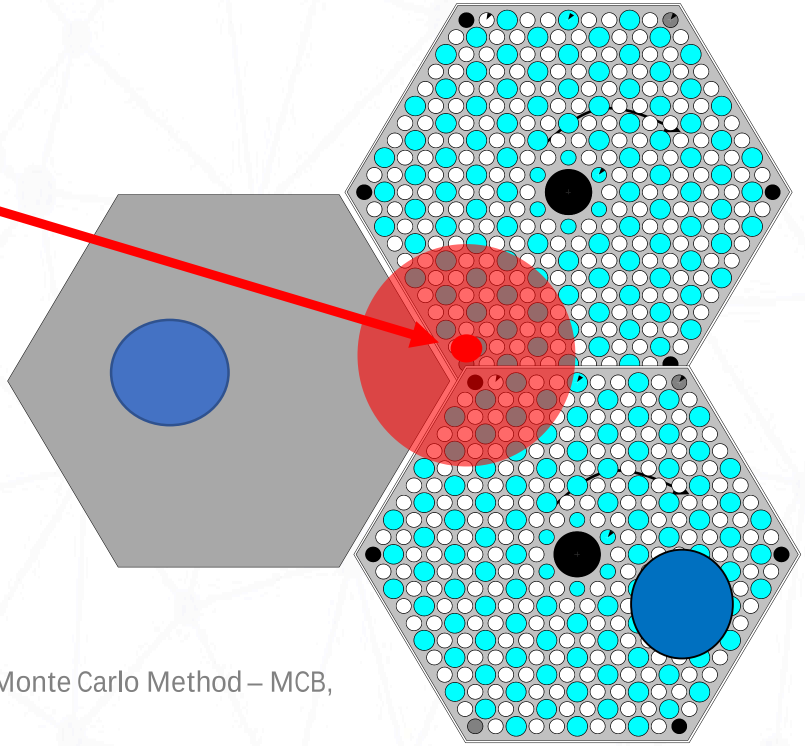
Ougouag and Cogliati. “Earthquakes and Pebble Bed Reactors: Time-dependent Densification”. Joint International Topical Meeting on Mathematics and Computation and Supercomputing in Nuclear Applications (M&C + SNA 2007) Monterey, California, April 15-19, 2007

Prismatic Fuel Considerations



- Compacts in blocks with engineered coolant channels – more heterogeneous than PBRs – batch-loaded
- Burnable poison pins are used to flatten the power and hold down reactivity over the cycle
- Shutdown rods are inserted into the fuel blocks – normally out (holes become streaming pathways)
- Fuel reshuffling can be 3D, but generally not (uneven swelling of blocks?). Axial shuffling generally preferred.





Cetnar, J. et al, Assessment of Pu and MA utilisation in deep burn Prismatic HTR by Monte Carlo Method – MCB, Project PUMA, AGH-University of Science and Technology, Krakow, Poland, 2013

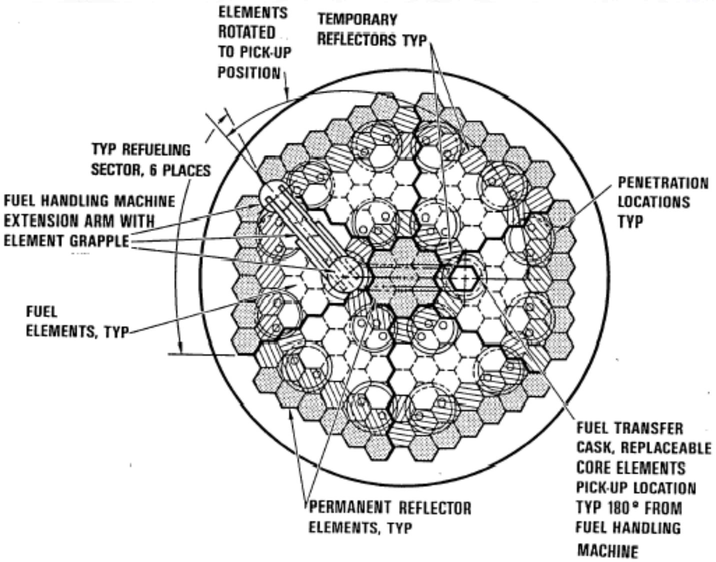
Tolerances in General Atomic’s Neutronic Codes (C-E)/E)

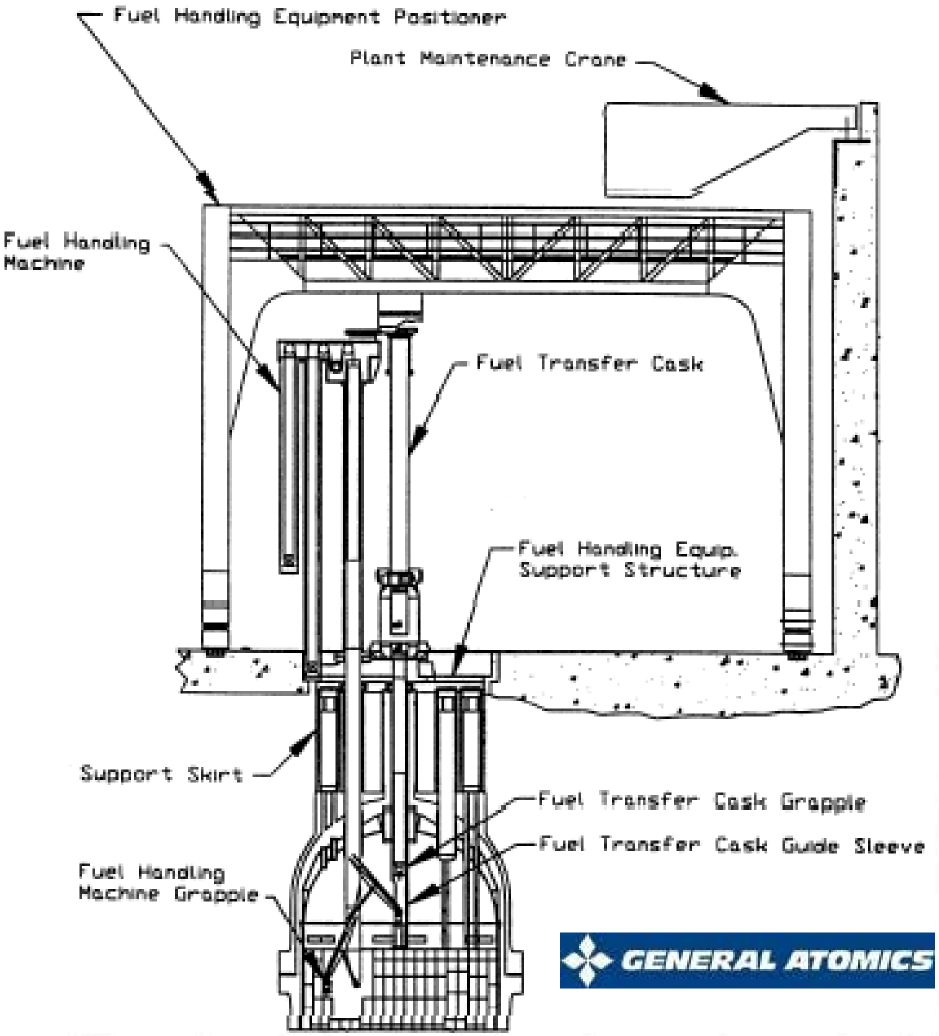
Facility	**Temp. Defect**	**C. R. Worth**	**Power Distr.**	**K-$_{\text{eff}}$**	**Water Ingress**	**Decay Heat**
HEU-CORES						
Peach Bottom Critical	±14%	-11%	±10%	±0.7%	DA	-
Peach Bottom	-11% to +4%	-6% to +10%	±10%	±0.7%	-	DA
HTGR Critical	+6%	+4% to 13%	-	-0.1% to +1.0%	-	-
Fort St. Vrain	-9% to +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%	-	
HITREX-2	-	-	±10%	±0.5%	-	

Baxter, A.. (General Atomics) Module 5b - Prismatic Nuclear Design, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Prismatic Fuel Handling – MHTGR

Fort St. Vrain Fuel Handling Machine (FHM)



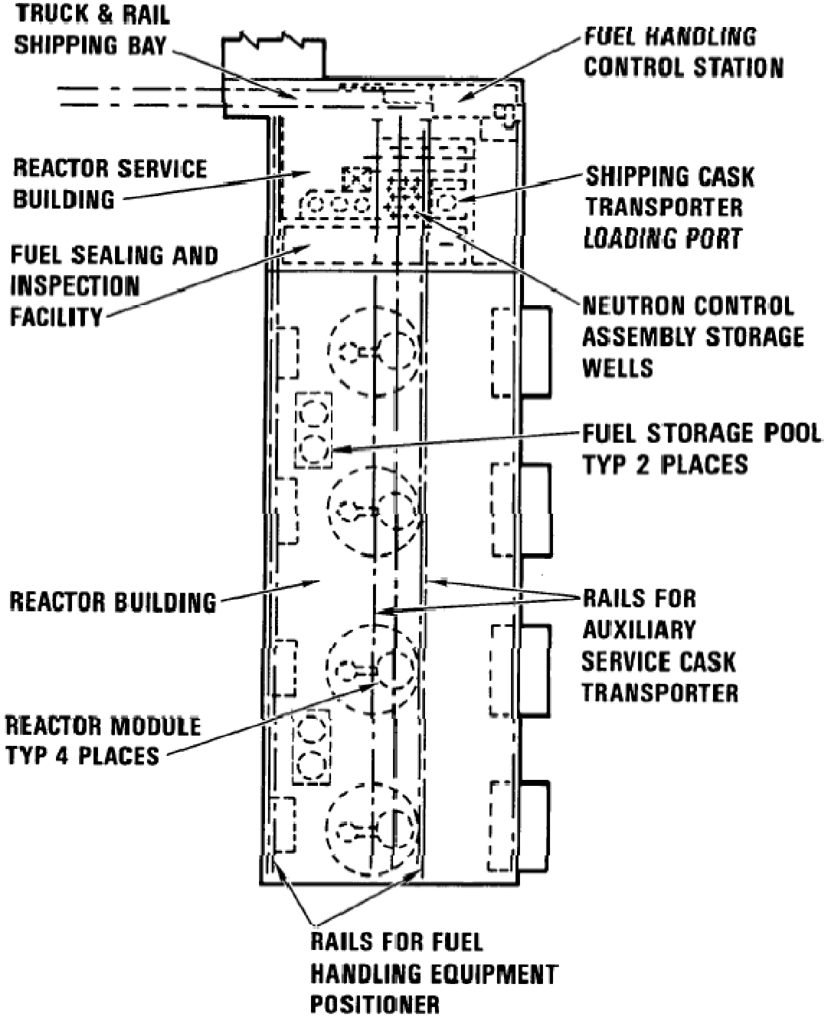


GENERAL ATOMICS

Vollman, R. (General Atomics) Prismatic HTGR Core Design Description, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Prismatic Fuel Handling – MHTGR (cont.)

Fuel Loading Deck of the
Fort. St. Vrain Core



Layout of a 4-module MHTGR

HTGR Core Design – Overview

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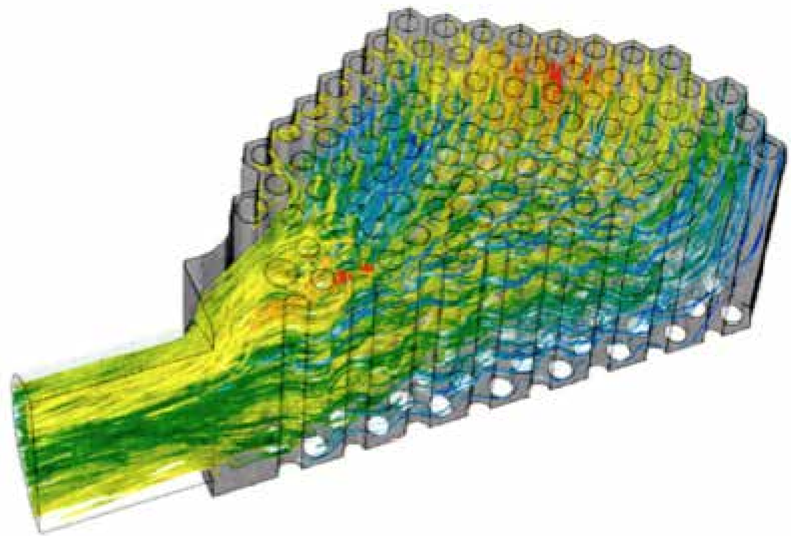
  - Power Conversion
- Normal Operation and Power Maneuvers

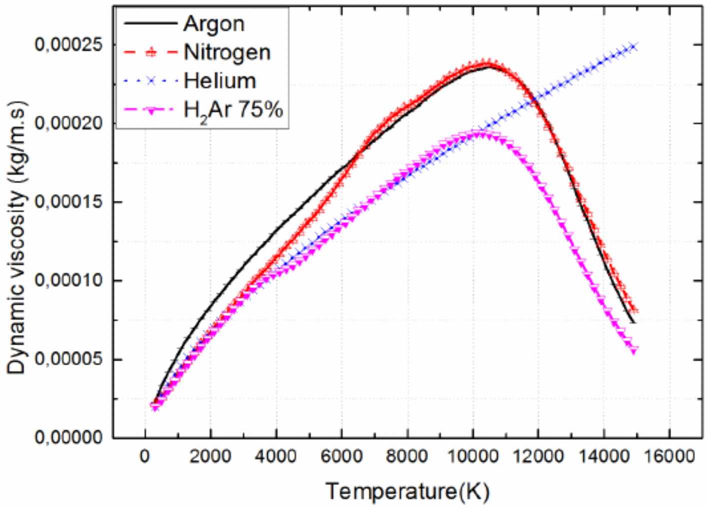
Coolant flow in Lower Plenum



Petti, D. et al (2019). Current Status of VHTR Technology Development.

Thermal-Fluidics

- Downward flow
  - Inlet coolant directed upward along the inside of the RPV to keep it and the Control Rod structures cool
  - Flow reverses during LOFC
  - Complex mixing structure at core outlet to prevent thermal ‘hot-striping’ and stress on downstream components
- So much carbonaceous material…
  - Thermal transients are relatively slow
  - Heat transfer via conduction/radiation after a loss of force flow
- Helium
  - Neutronically transparent and chemically inert
  - Viscosity increases with temperature (potential stagnation in hot channels)



$$\mu(T) = 3.953\text{E}{-7}T^{0.687}\,\text{Ns}/m^2$$

Abderrahmane, Aissa, Mohamed, Abdelouahab, Noureddine, Abdelkader, El Ganaoui, Mohammed , Pateyron, Bernard. (2013). Ranz and Marshall correlations limits on heat flow between a sphere and its surrounding gas at high temperature. Thermal Science. 10.2298/TSCI120912090A.

Melese and Katz, “Thermal and Flow Design of Helium-Cooled Reactors”, American Nuclear Society, ISBN 0-89448-027-8, 1984.

Temperature Feedback

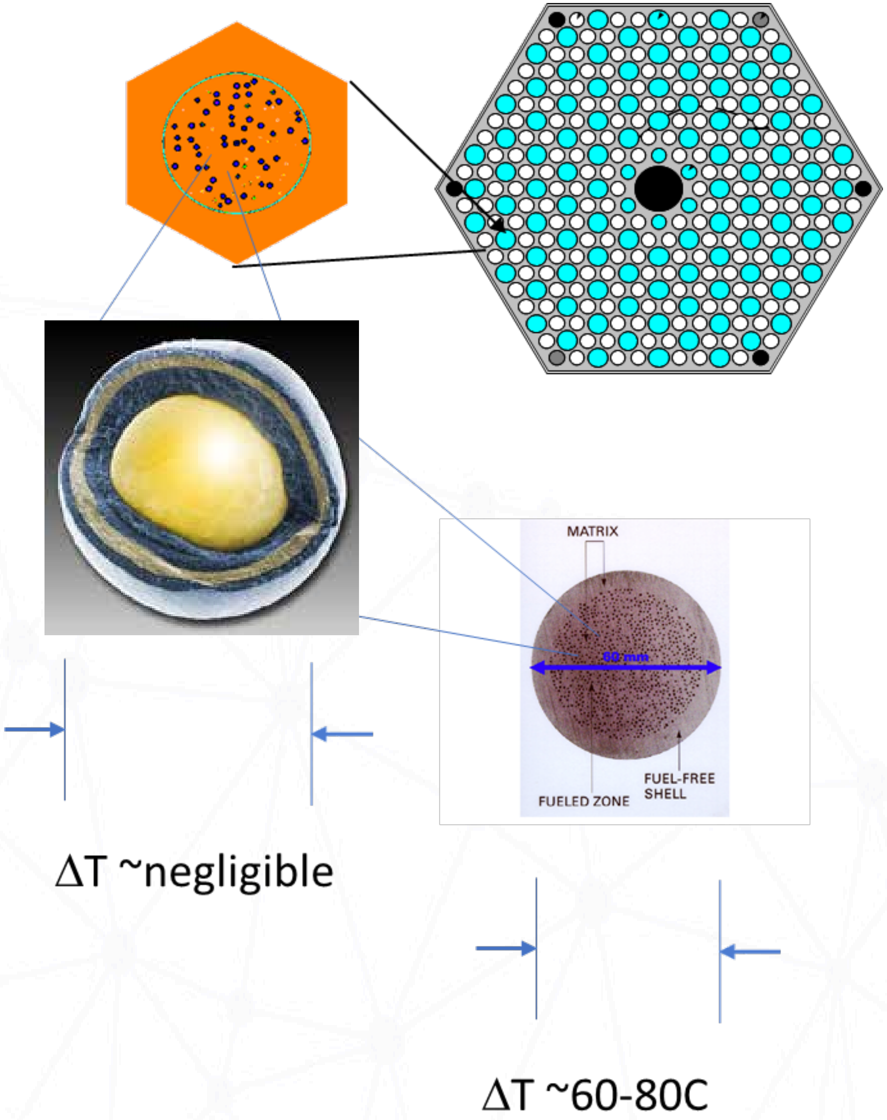
- The core will shut itself down in the event of a loss of coolant
- Enables load following with He mass flow control

Temperature Coefficients	Unit	Under Operating Conditions
Fuel (Doppler coefficient of mainly ^{238}U)	Dp/°C	- 4.4 x10^{-5}
Moderator	Dp/°C	- 1.0 x10^{-5}
Reflector regions (all together)	Dp/°C	+ 1.8 x10^{-5}
Total	**Dp/°C**	**- 3.6 x10^{-5}**

Venter, P. (PBMR) Module 6b – Pebble Bed HTGR Nuclear Design, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

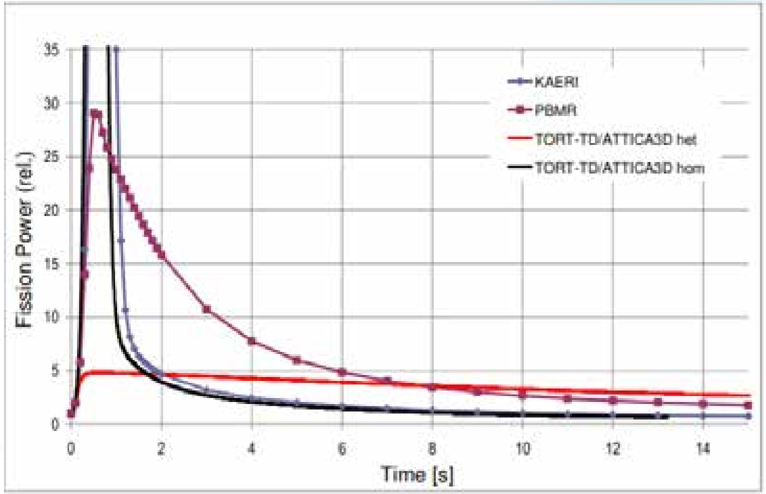
Heat Deposition

- Kernels are small, but still larger than the recoil distance of fission products Þ most of the fission heat is deposited in the kernel, but…
- This heat dissipates easily into the surrounding matrix, so for all but the most extreme (BDB) reactivity spikes, the particles are largely in thermal equilibrium with the surrounding matrix, even during transients
- This allows one to define the ‘fuel temperature’ as the compact or fueled region of the pebble

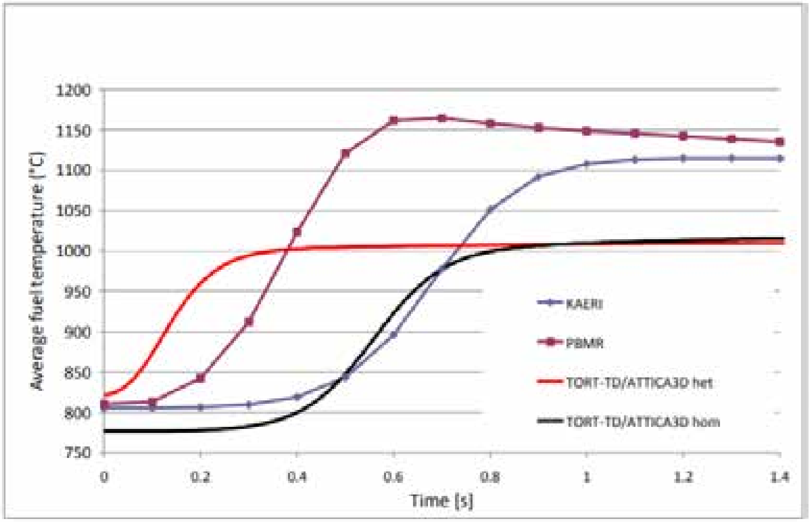


Explicit Particle Heat Deposition Models

- Some codes have been developed with a ‘subgrid’ model of heat deposition only in the kernel and transient heat conduction out of the particles and into the matrix
- Results show very different fuel temperature and power trajectories between ‘smeared’ and explicit models for large (and in some cases unphysical) transients
- The smeared fuel models are generally much more conservative – kernel-limited heat deposition leads to faster Doppler turnaround



Power and temperature excursion during Total Rod Ejection (0.1 cm) – this scenario is precluded by design



Lapins, Janis and Seubert, A and Buck, Michael and Bader, Jo and Laurien, E. (2011). Tort-td/AtticA3D: A Coupled Neutron Transport and Thermal Hydraulics Code System for 3-D Transient Analysis of Gas Cooled High Temperature Reactors. 10.13140/2.1.3526.3369.

Ortensi,J., Boer, B, and Ougouag, A,. Thermo-mechanical Analysis of Coated Particle Fuel Experience a Fast Control Rod Ejection, Proceedings of the 5th International Topical Meeting on High temperature Reactor Technology (HTR2010), Prague, October 2010.

Hu, Jianwei and Uddin, R., 3D Thermal Modeling of TRISO Fuel Coupled with Neutronic Simulation, LA-UR-10-00442, Los Alamos national Lab, 1 January 2010.

Core Thermal-Fluidics: Prismatic

- To first order, heat transfer during power operation can be captured with 1-D pipe flow models and 2-D heat conduction

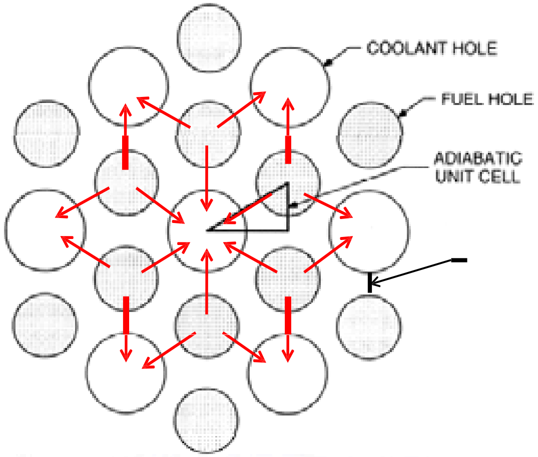
$$Nu_{FD} = 0.023\ \mathrm{Re}_b^{0.8}\ \mathrm{Pr}_b^{1/3}(\mu_b/\mu_w)^{0.14}$$

Seider-Tate Correlation for single phase flow in a circular channel

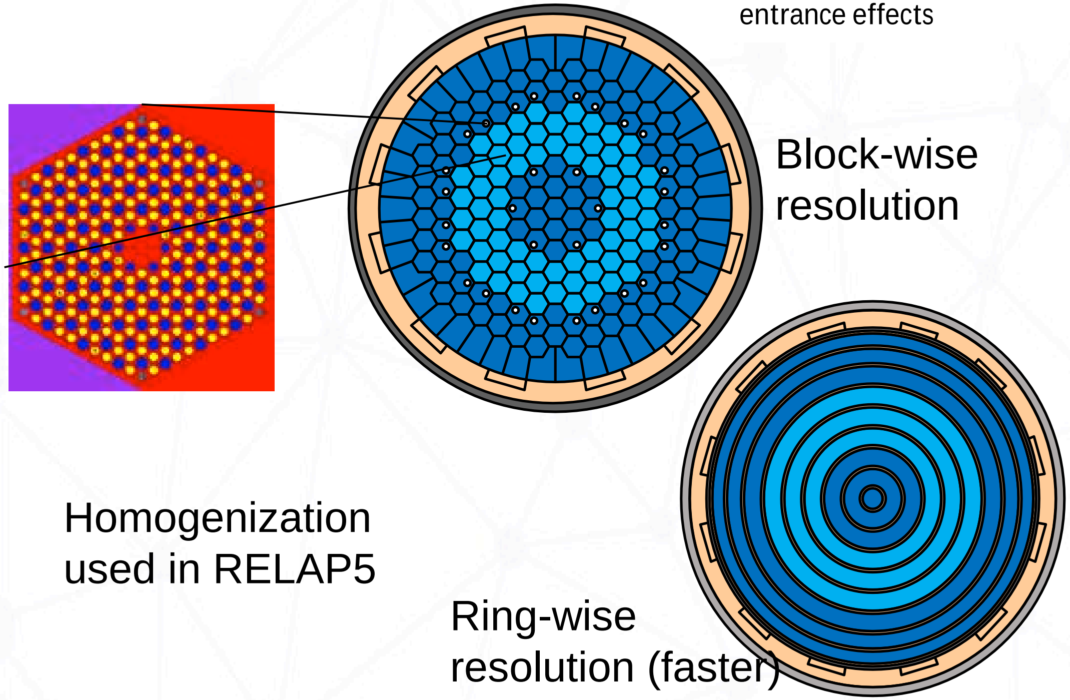
$$Nu_{FD} = 0.021\mathrm{Re}_b^{0.8}\mathrm{Pr}_b^{0.4}(T_w/T_b)^{-0.50}$$

$$Nu = Nu_{FD}[1 + (z/D)^{-0.70}]$$

McEligot Correlation for fully developed flow and entrance effects



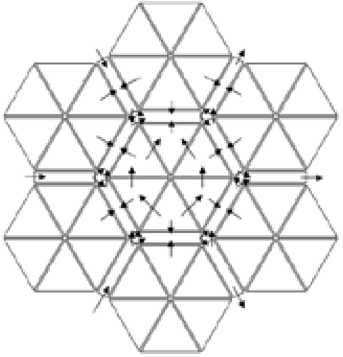
Seker, V.. (2007). Multiphysics methods development for high temperature gas reactor analysis. ETD Collection for Purdue University.



Block-wise resolution

Homogenization used in RELAP5

Ring-wise resolution (faster)

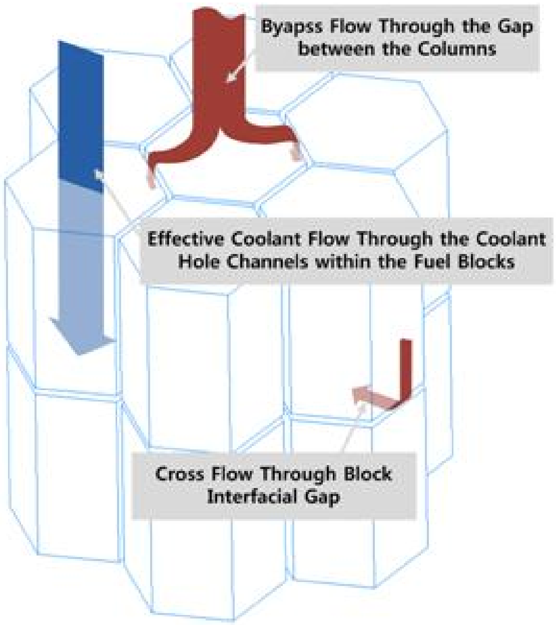


Homogenization/ Network Model used in AGREE/GASNET

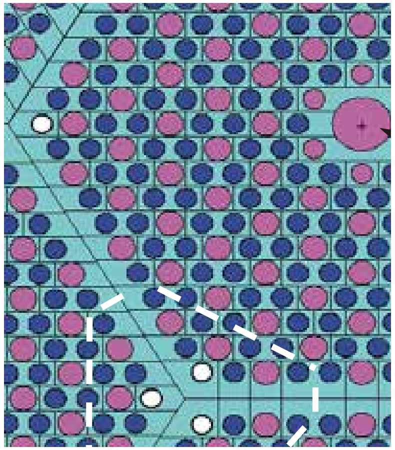
Triangle-wise resolution

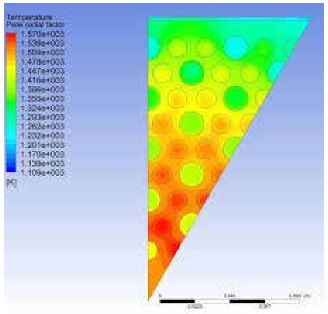
Core Thermal-Fluidics: Prismatic (cont.)

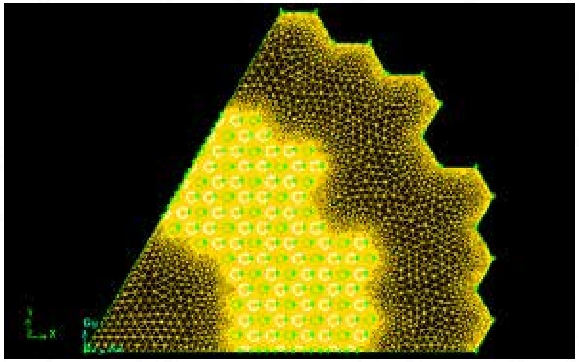
- Dimensional changes in graphite lead to alternate coolant pathways (bypass flow) – significantly altering the temperature profile in the core and reflector. Bypass flows can be modeled as extra channels in network codes.
- Little momentum upon loss of pumping power, coolant quickly slows (relaminarization) and is then driven by buoyancy. If there are significant bypass gaps, radiation across the gaps becomes a dominant heat transfer mechanism
- Transient analysis are still performed with the simple, homogenized block (or subblock) models. Coarse mesh CFD methods may be an adequate compromise (PRONGHORN?)



Richard W. Johnson, Hiroyuki Sato, and Richard R. Schultz. CFD Analysis of Core Bypass Phenomena. United States: N. p., 2009. Web. doi:10.2172/974775.







Core Thermal-Fluidics: Pebble Bed (cont.)

Convective heat transfer in a packed bed

$$h = \frac{Nu k_{He}}{D_p}$$

$$Nu = f_\varepsilon Nu_s$$

$$Nu_s = 2 + \sqrt{Nu_l^2 + Nu_t^2}$$

$\varepsilon \sim 0.40$

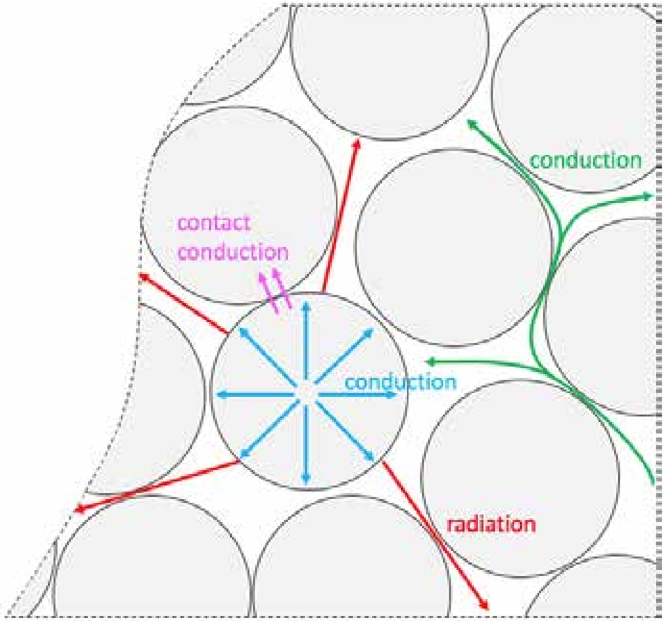
$$f_\varepsilon = 1+1.5(1-\varepsilon)$$

$$Nu_l = 0.664 \left(\frac{\mathrm{Re}}{\varepsilon}\right)^{1/2} \mathrm{Pr}^{1/3}$$

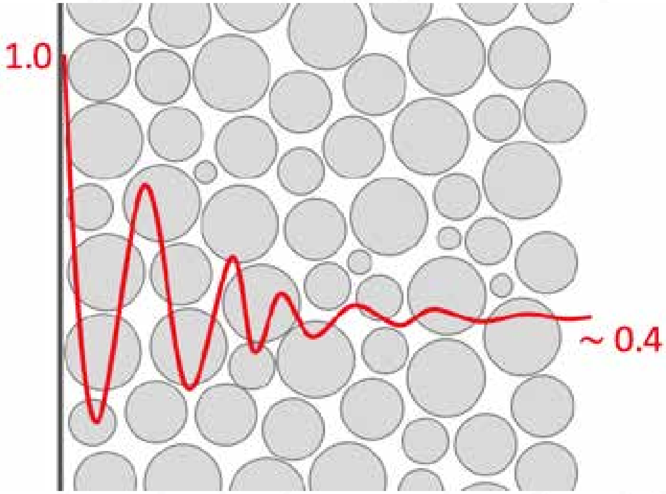
Laminar component

$$Nu_t = \frac{0.037\left(\frac{\mathrm{Re}}{\varepsilon}\right)^{0.8} \mathrm{Pr}}{1 + 2.443\left(\frac{\mathrm{Re}}{\varepsilon}\right)^{-0.1}\left(\mathrm{Pr}^{2/3} - 1\right)}$$

Turbulent component



Other correlations have been developed to capture variable porosity, wall effects, radiation and conduction under low flow conditions



CFD models of local geometries have been executed and avoid many of these empirical assumptions

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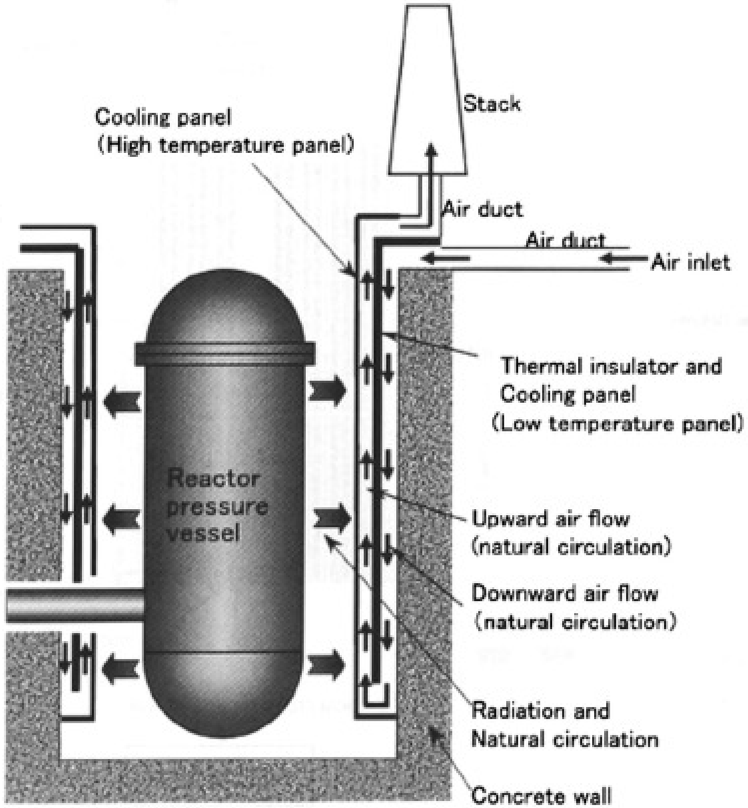
Pressure Vessel in the Reactor Cavity

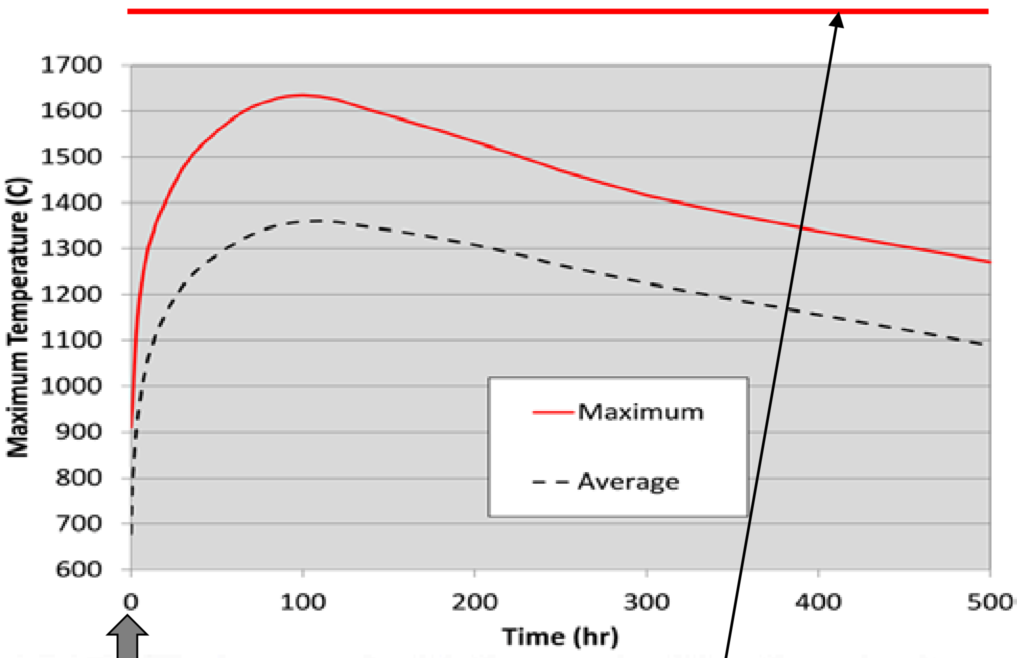


Kuniyoshi Takamatsu, Tatsuya Matsumoto, Koji Morita, New reactor cavity cooling system (RCCS) with passive safety features: A comparative methodology between a real RCCS and a scaled-down heat-removal test facility, *Annals of Nuclear Energy*, Volume 96, 2016.

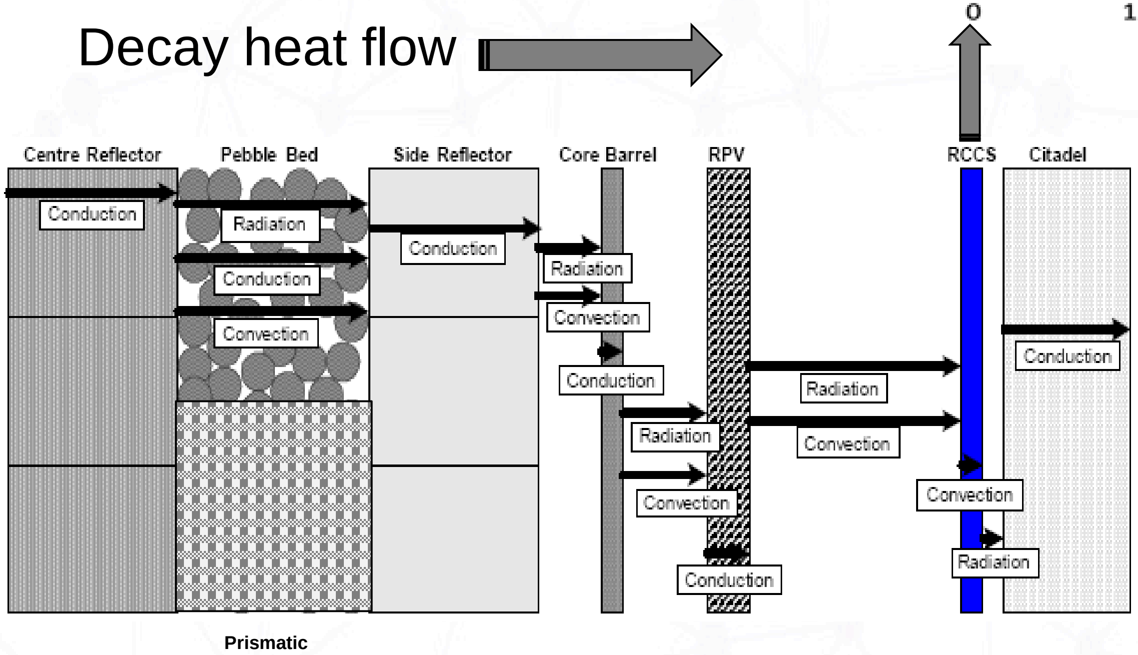
Inherent Safety

'Grace period' (no operator intervention) measured in tens or hundreds of hours

Core temperatures during a DLOFC
AREVA Technical Document 12-9251926-001,
Summary Report-SC-HTGRE Demonstration
Reactor



Decay heat flow

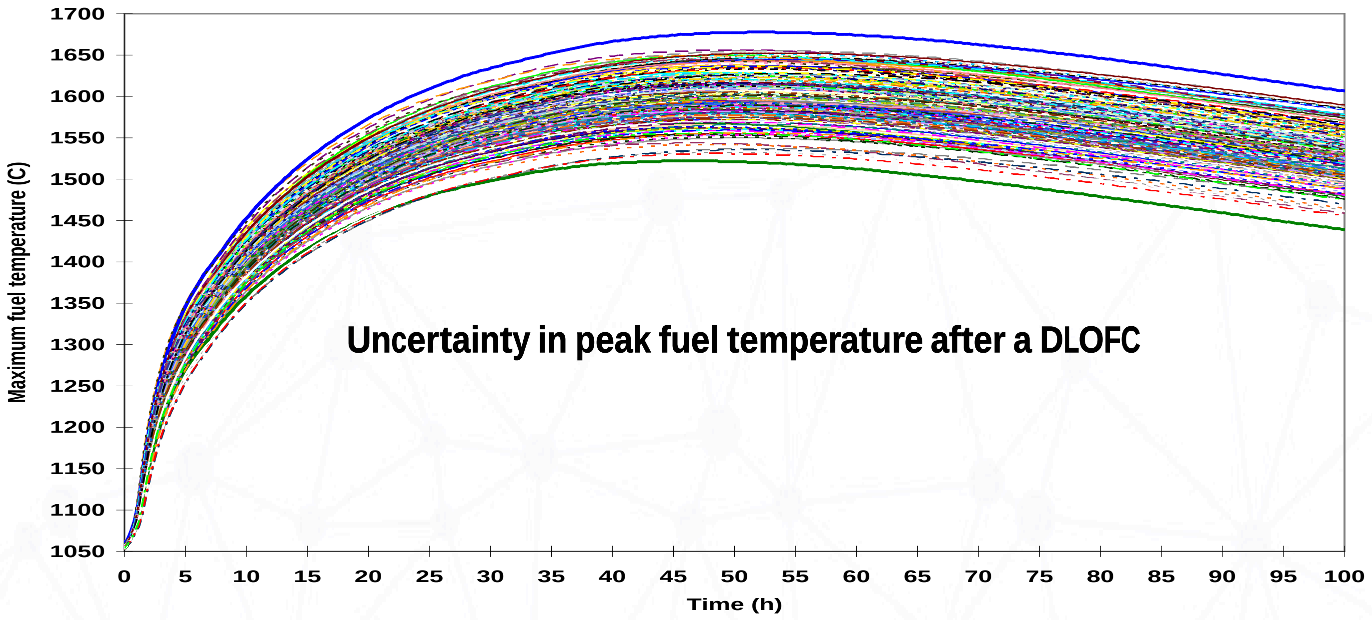


1800C – No appreciable UCO particle failures observed in AGR heating test at this temperature although accelerated diffusion of certain FP (Sr, Cs, Eu) is observed.

Courtesy of F. Reitsma, IAEA

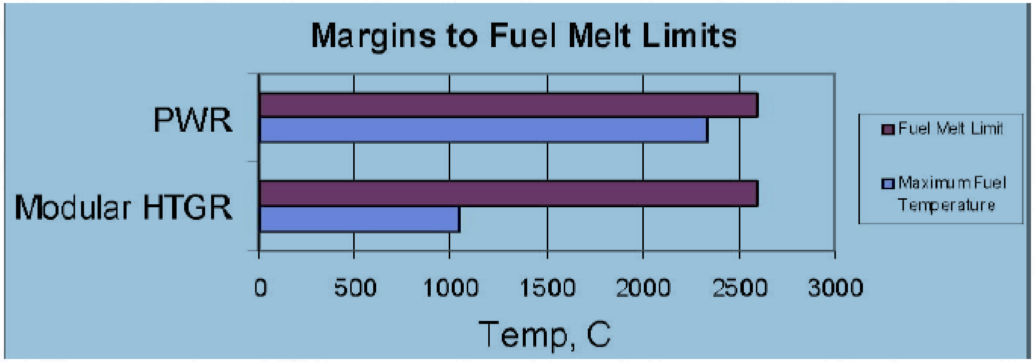
Large Margins to Particle Failure Temperature

1800°C beyond 150 h

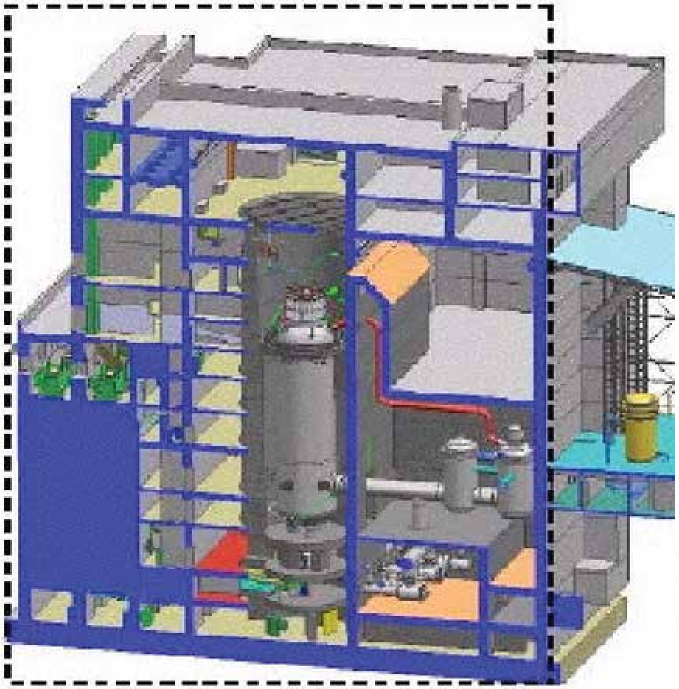


Uncertainty Analysis of Peak Fuel Temperature during a DLOFC in the PBMR400, G. Strydom, INL. Different trajectories obtained by varying input parameters over expected ranges (graphite conductivity, etc.)

Margins are large;
grace periods are long.



Radiological Release Sequence and the Vented Reactor Building Concept



Cutaway diagram of the PBMR-400 Demonstration Plant (PBMR (Pty) Co. Ltd)

Buildup: During operation small amounts FP diffuse out of the fuel/graphite (limited by He Purification System)

- Some (e.g. Ag) adsorb onto cooler surfaces in the primary loop
- Others (Eu, Cs, Sr) remain as ‘circulating inventory’

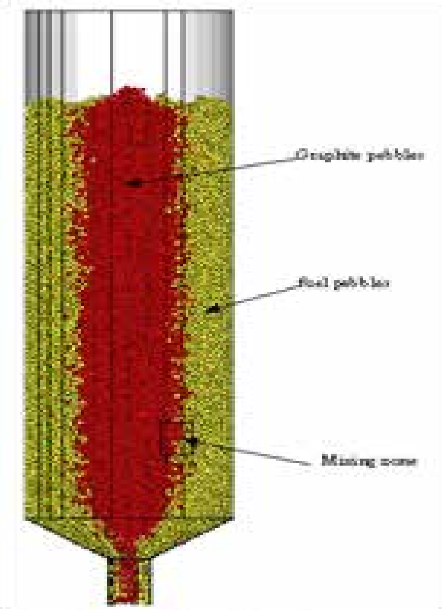
Puff: After a significant break, circulating inventory is released and vented from the building

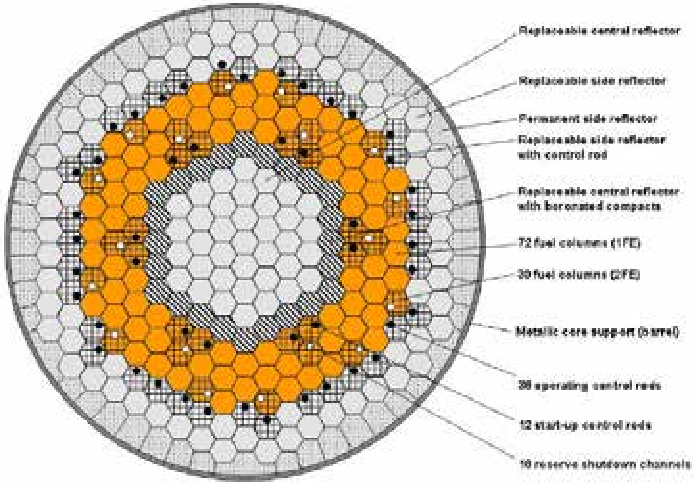
Cook (heatup/cooldown): After depressurization, the vents are closed. FP-driven heatup of the core drives additional releases from the fuel, some of which will eventually make its way out of the building.

D. A. Petti, R. R. Hobbins, P. Lowry and H. Gougar (2013) Representative Source Terms and the Influence of Reactor Attributes on Functional Containment in Modular High-Temperature Gas-Cooled Reactors, Nuclear Technology, 184:2, 181-197, DOI: 10.13182/ NT184-181

Core Analysis Summary

- Big graphite cores pose an interesting challenge for core modelers, especially for transient analysis
- Fortunately,
  - Safety parameters (fuel failure temperatures and fission product release rates) are not overly sensitive to neutronics parameters
  - Grace periods are long (many hours or days rather than minutes)
  - No coolant phase change
- High fidelity tools (Monte Carlo transport and CFD) are useful mainly for quantifying uncertainties; they are not essential for routine core design yet,but we're moving in that direction
- Still, some features of modular HTGRs pose challenges to traditional LWR methods (moving fuel, burnable poisons, spectral leakage). Modern tools are better suited to tackling these features in a rigorous way





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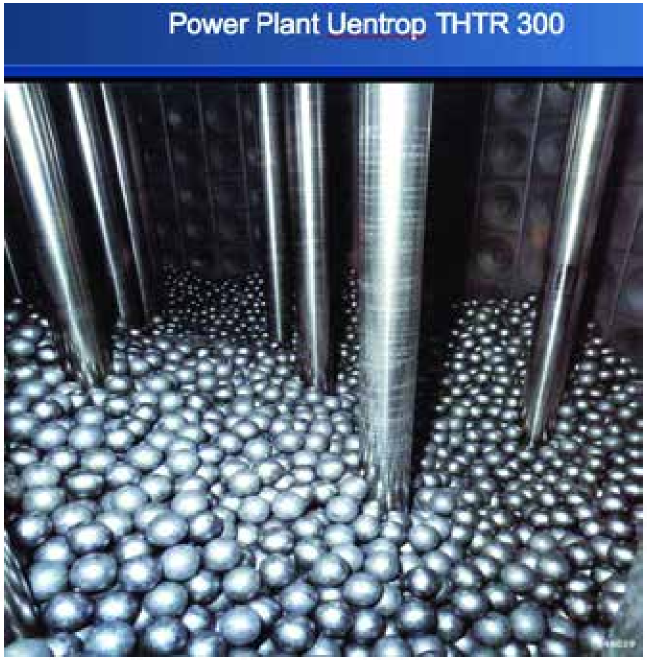
THTR featured a Shutdown CR system in which the rods were forced into the pebble bed. It was designed to be used only intermittently but unintended scrams were frequent. Broken pebbles were a result.

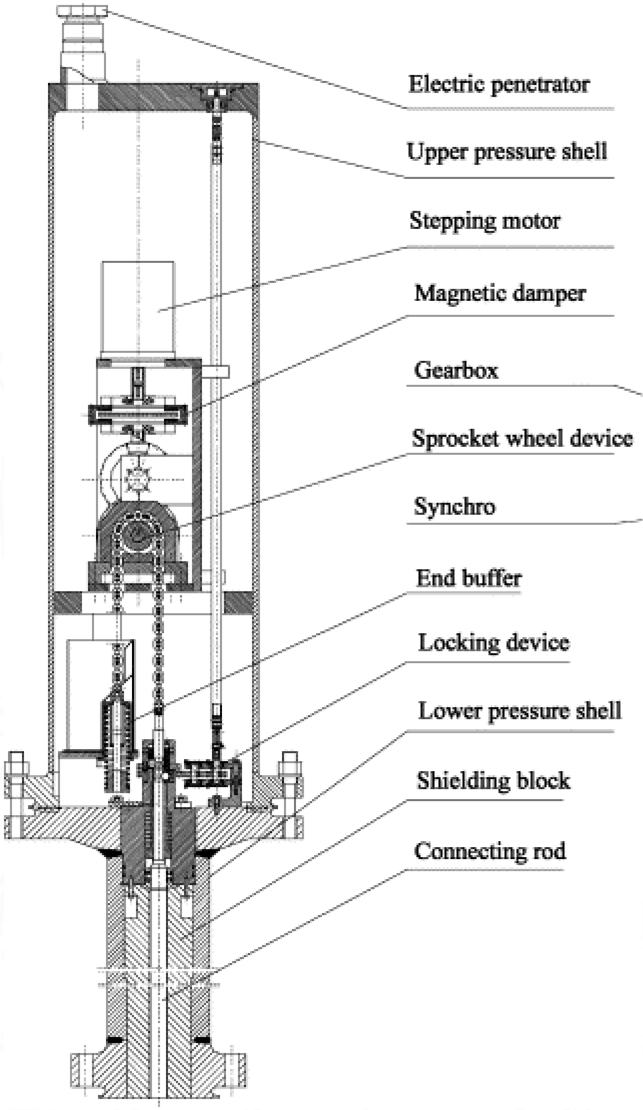


Daoud, H., Serries, F., & Schollmeyer, H. (1989). Operating experience with the THTR core control rods. Germany: INFORUM Verl. (available through IAEA INIS)

Reactivity Control Requirements

mHTGR-DC 26, NRC Reg guide 1.232

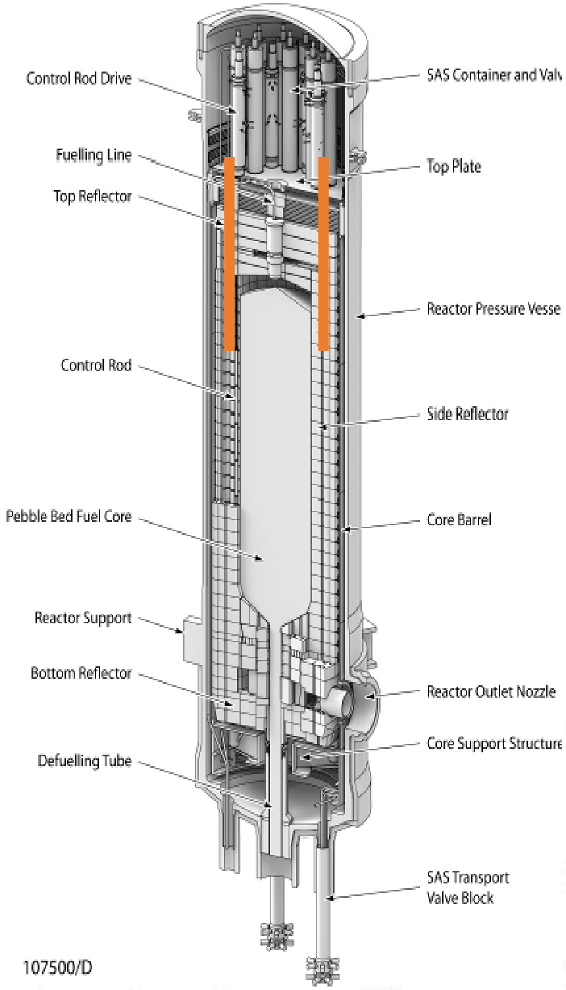
- A means of **inserting negative reactivity at a sufficient rate** and amount to assure… radionuclide release limits and He pressure design limits are not exceeded and safe shutdown is achieved…
- A means which is **independent and diverse** from the other(s), shall be capable of controlling the rate of reactivity..
- A means of inserting negative reactivity at a sufficient rate and amount to assure, … that the **capability to cool the core is maintained and a means of shutting down the reactor** and maintaining, at a minimum, a safe shutdown condition…
- A means for h**olding the reactor shutdown under conditions which allow for interventions** such as fuel loading, inspection and repair shall be provided.



Wu Yuanqiang, Diao Xingzhong, Zhou Huizhong, Huang Zhiyong, Design and tests for the HTR-10 control rod system, Nuclear Engineering and Design, Volume 218, Issues 1-3, 2002.

Reactivity Control

- Typical: Two independent rod banks
- Articulated rods suspended from drives by chains to be lowered into the radial reflector
- Bypass flow cools the rods
- May be partially inserted during power operation to provide Xe restart/load follow capability
- Some load following can be achieved with He flow control
- Prismatic – Shutdown rods can inserted into fuel blocks
- PBR – Small absorber spheres have been proposed for past designs (not in X-energy XE-100)



Both AVR and HTR-10 can be shut down without rods – circulators are stopped to affect a core heatup and Doppler shutdown.

Shutdown Cooling System (SCS)

SCS Protection System

Following detection of:

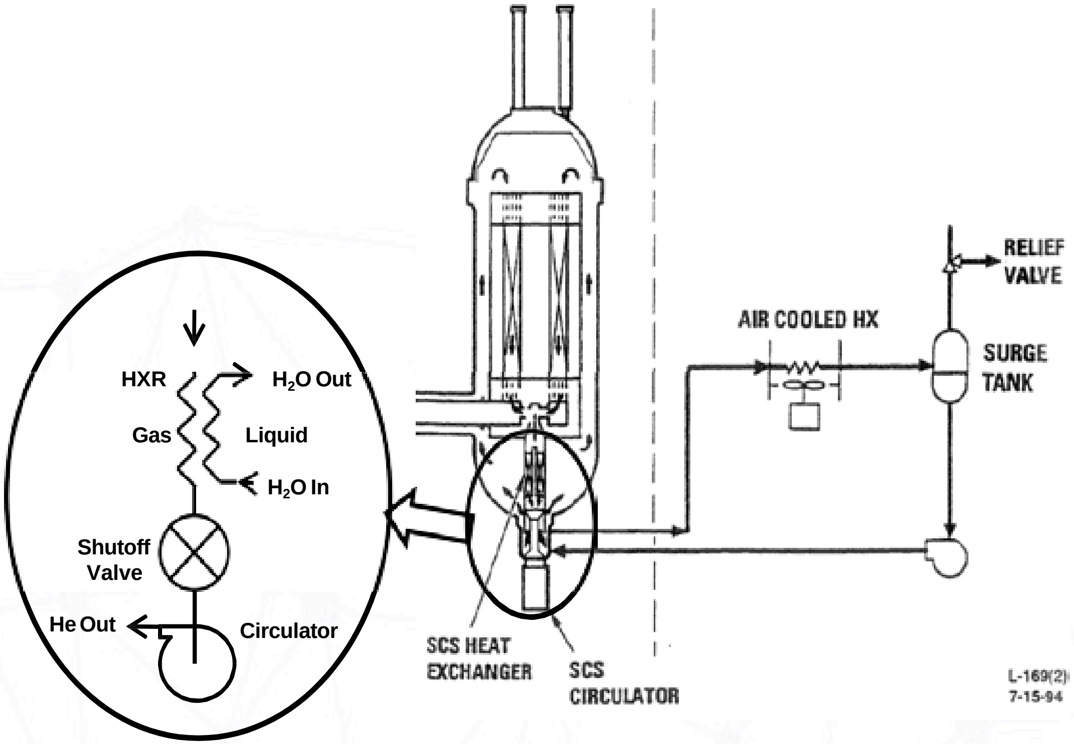
- Heat Exchanger Leaks
- Circulator Overspeed
- Low Cooling Water Flow
- Loss Of Net Positive Suction Head
- High Heat Exchanger temperatures

Actions:

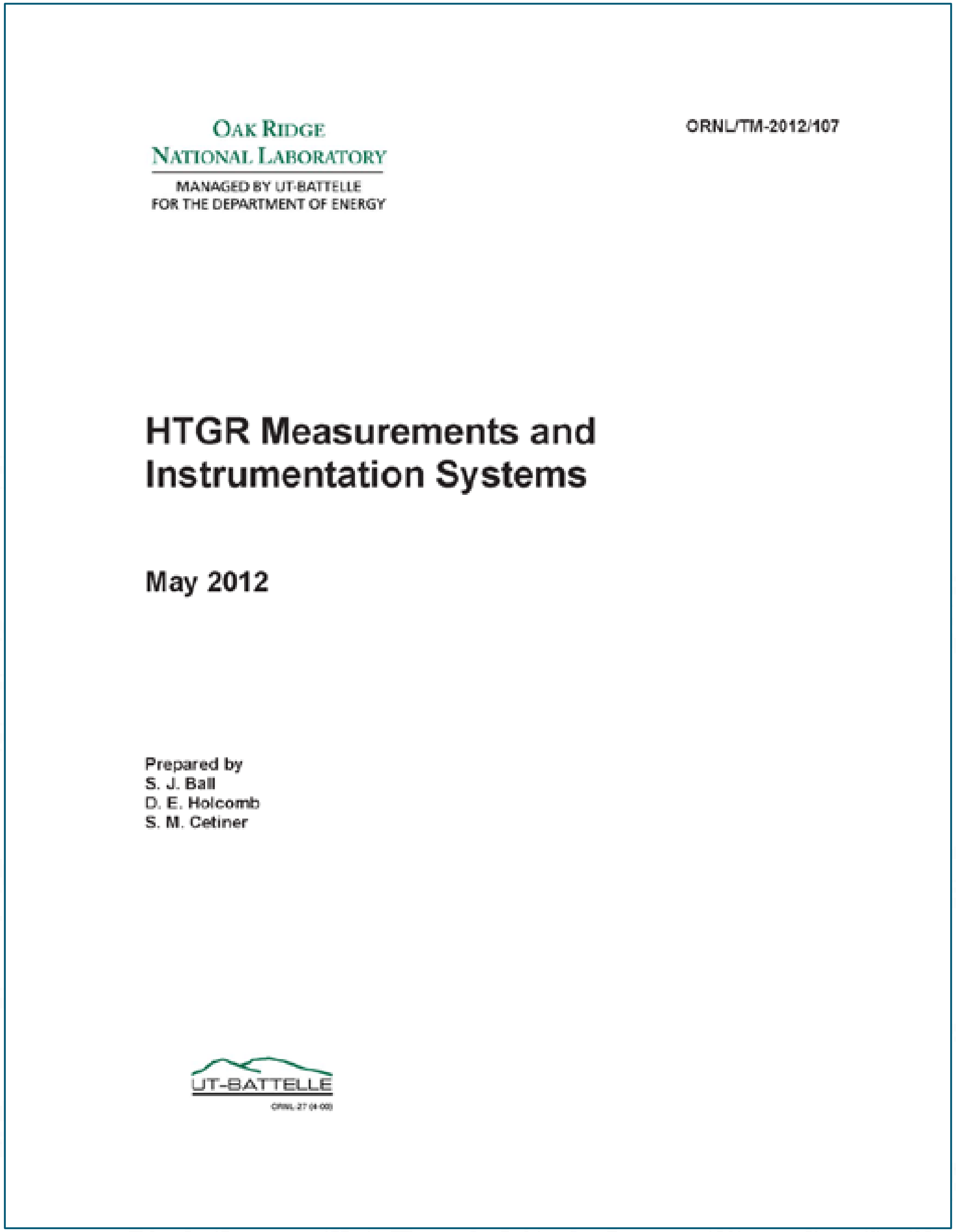
- Shutoff Valve actuated
- Circulator shutdown

Components List

- He Circulator
- He Shutoff Valve
- Gas to Liquid Heat Exchanger
- Control System
- Shutdown Water Cooling System
- Service Equipment



Single Shutdown Cooling System Loop per Reactor Module



OAK RIDGE
NATIONAL LABORATORY
MANAGED BY UT-BATTELLE
FOR THE DEPARTMENT OF ENERGY

ORNL/TM-2012/107

HTGR Measurements and Instrumentation Systems

May 2012

Prepared by
S. J. Ball
D. E. Holcomb
S. M. Cetiner

UT-BATTELLE
ORNL-27 (4-00)

Helium Conditioning

- Removes chemical and radionuclide impurities from helium coolant
- Pressurizes, depressurizes, and controls the primary helium coolant inventory in conjunction with Helium Transfer and Storage System (HT&SS)
- Provides purified helium for purges and buffers
- Maintains primary coolant system at a slightly subatmospheric during refueling/maintenance
- Purifies helium pumped to storage
- Removes H_2O from primary circuit following water ingress event

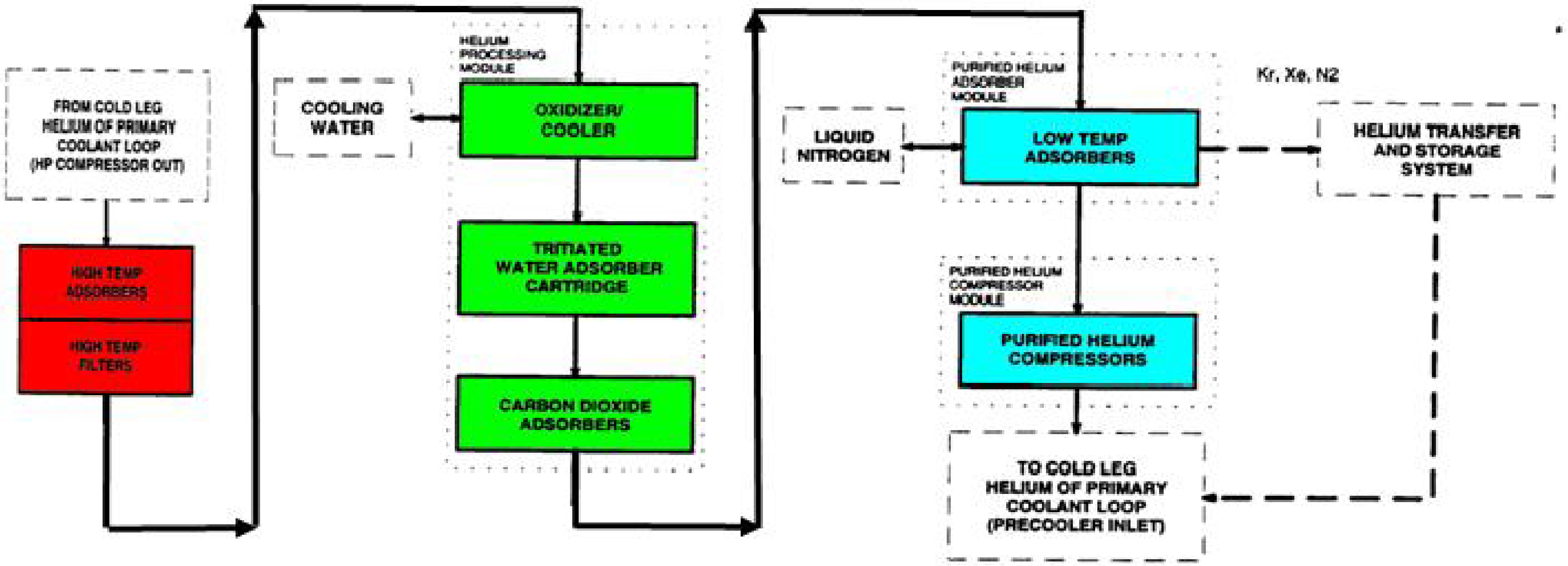
Hanson, D. (General Atomics) – Module 10c - Helium Inventory and Purification, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Helium Purification System Requirements (General Atomics)

- Each reactor module shall have an independent helium purification system
- Shall remove H_2O, CO, CO_2, H_2, N_2, O_2, H_2S, CH_4, and higher molecular weight hydrocarbons
- Shall allow depressurization of the Reactor Module (and/or adjacent module) within 24 hours after shutdown
- Shall include one regeneration train for two HPS
- Shall be sized to process a slipstream of the primary coolant, typically on the order of 1% of the primary loop volume flow rate

Hanson, D. (General Atomics) – Module 10c - Helium Inventory and Purification, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

HPS Train (General Atomics)



Contaminants Removed

Red
Non-condensable Radionuclides and Particulates

Green
H_2O, including tritiated H_TO and CO_2

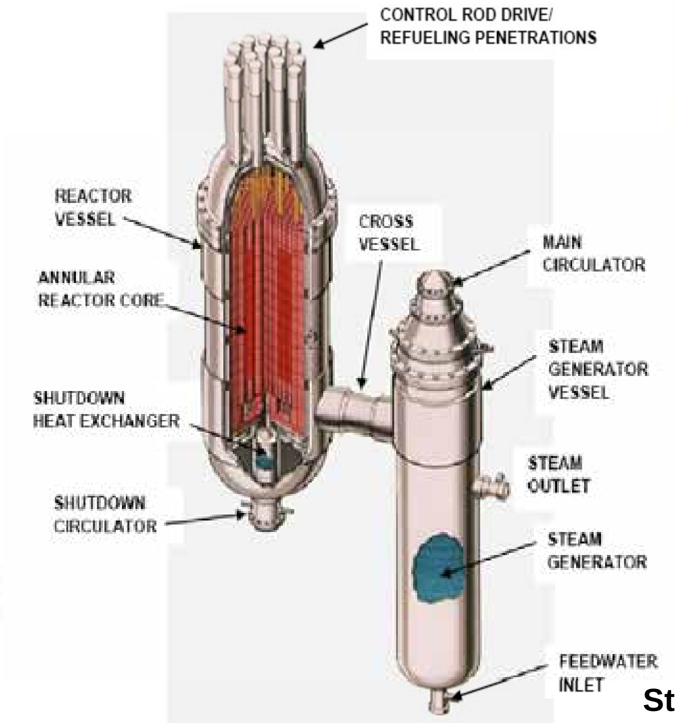
Blue
Kr, Xe, N_2

Lessons Learned from Fort St. Vrain (General Atomics)

- HPS and Helium Transport and Storage System (HT&SS) performed well in seven steam-cycle HTGRs
- Specific lessons from FSV (and AVR)
  - HPS overwhelmed by large H_2O ingresses; long times required for dry out of primary coolant circuit
  - Single transfer compressor required taking plant offline for compressor maintenance
- Components performed well except for Ti Getter Beds in FSV
  - FSV used Ti getter beds instead CuO oxidizers/driers for the removal of hydrogen and tritium
  - No operational consequences because H2 and H-3 sorbed onto core structures
- Design recommendations for future HTGRs:
  - Provide suitable drains for removal of standing water
  - Provide backup He transfer compressor
  - Use CuO oxidizer beds/driers for H2 and H-3 removal

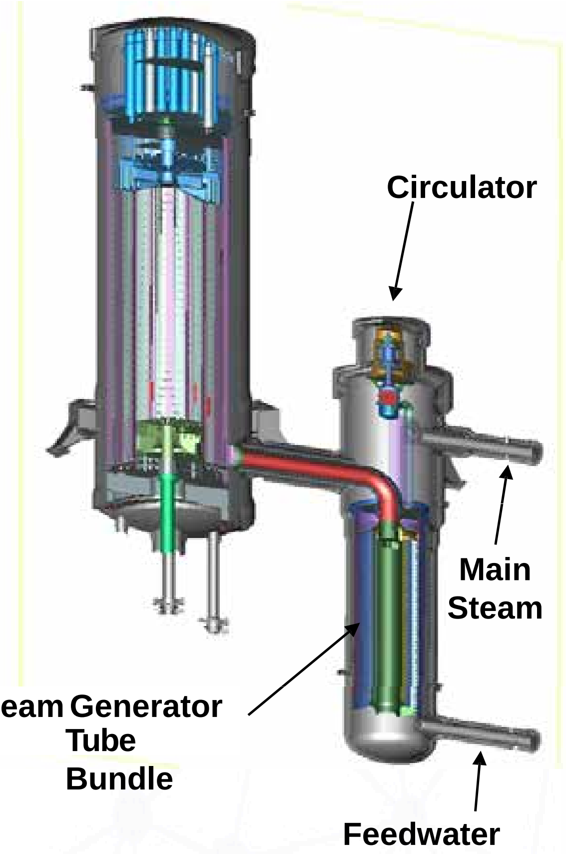
Hanson, D. (General Atomics) – Module 10c - Helium Inventory and Purification, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Power Conversion



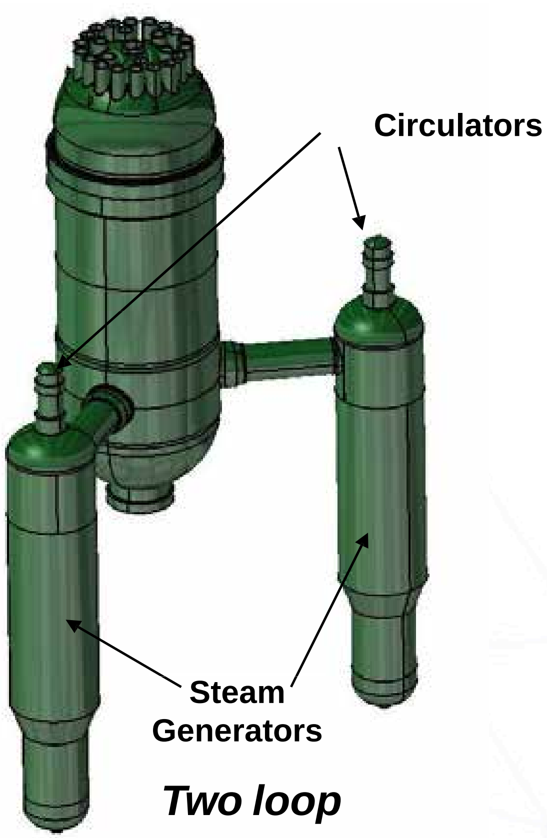
Single Loop

(MHTGR)



Single Loop

(PBMR-CG)

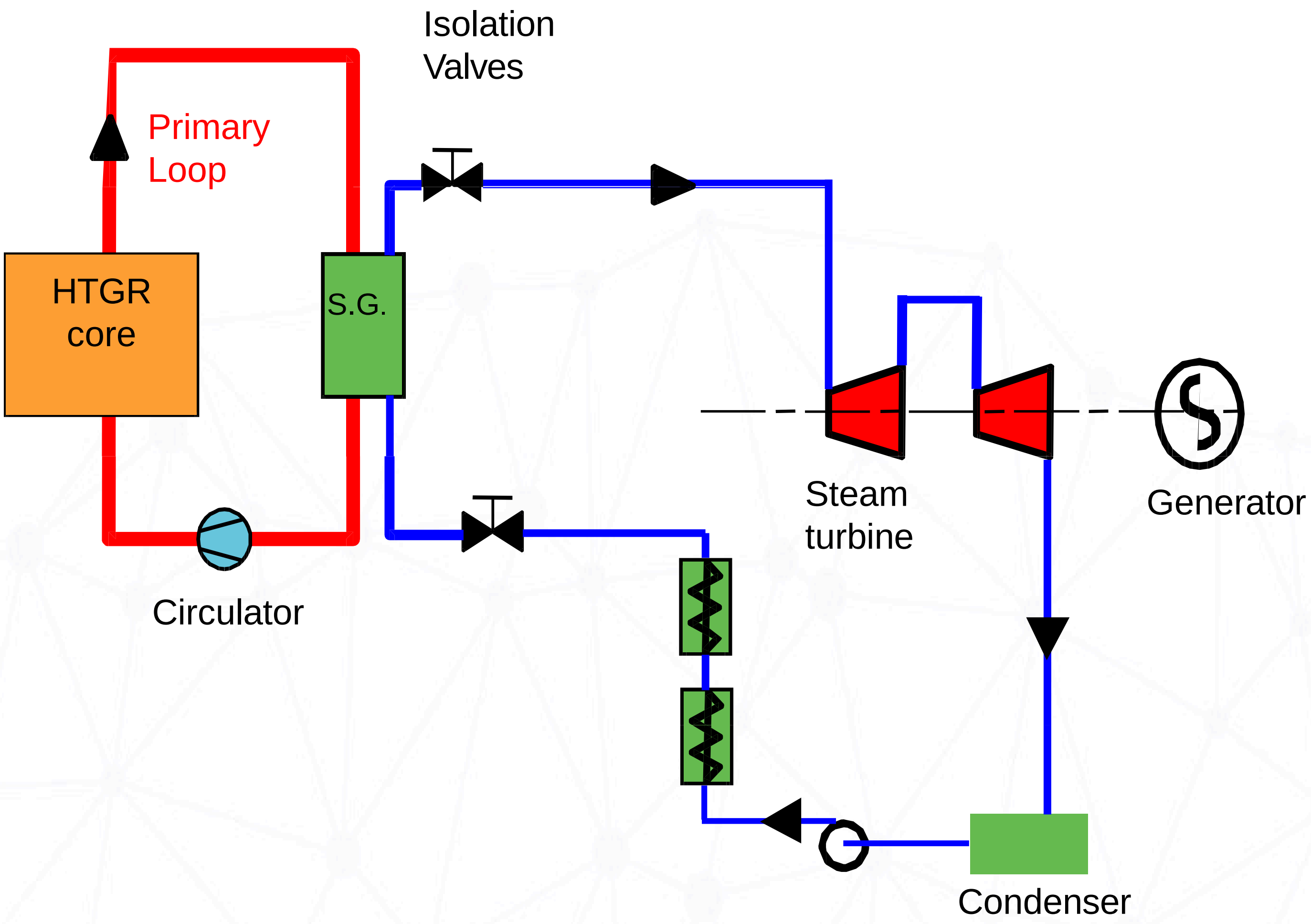


Two loop

(Framatome)

Lommers, L. (Framatome) – Module 10b - Steam Cycle Power Conversion System, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Conventional Rankine Cycle



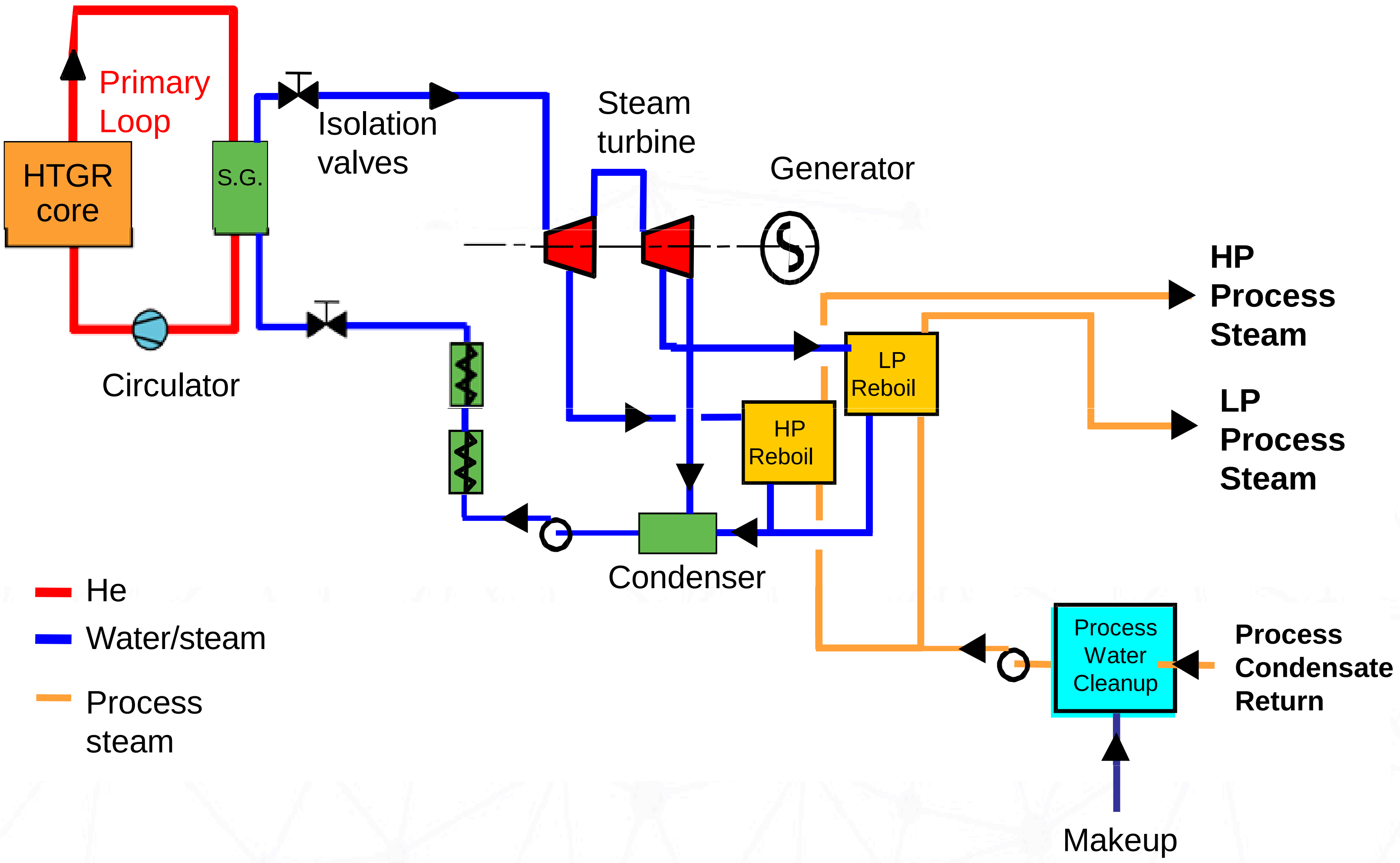
🟥 He

🟦 Water/steam

Steam and Process Heat Considerations

- Process steam pressure/temperature
- Process steam quantity
- Operating flexibility
  - Response to varying user steam demands
  - Flexibility for varying steam vs. electricity production
- Operational interaction between steam supply units and process users
- Process steam contamination concerns
- Feedwater quality control
- Process steam reliability concerns
  - Availability
  - Service interruption

Steam and Process Heat



Other Considerations

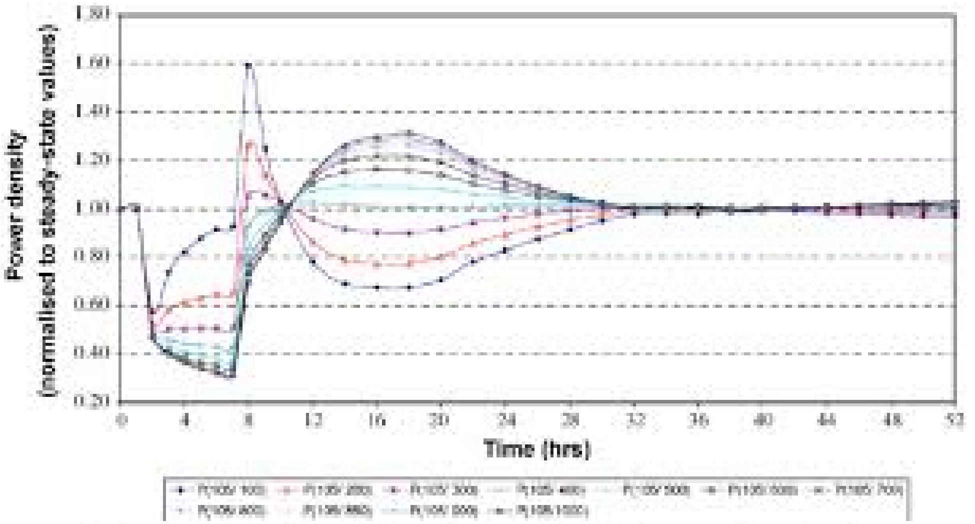
- Steam cycle and process heat components would use established fossil-driven technology
  - Coupling to an HTGR remains an issue
- Helium Circulators
  - Good experience from UK reactors
  - Magnetic bearings, submerged motors
  - Size is within vendor range
- Steam generators
  - Experience in HTGRs is more benign that PWRs (no shell-side CRUD)
  - HTGRs more robust
  - Problems with HTR-PM design delayed schedule
- Other Rankine cycle components
  - Well-within vendor experience base
- Reboiler (for Process Heat)
  - Used in fossil-drive process heat
  - New to HTGRs – will be customized

Anderson, N. and Sabharwall, P. (2014). RELAP5-3D transient modelling for NGNP integrated plant. Int. J. of Nuclear Energy Science and Technology. 8. 213 - 237. 10.1504/IJNEST.2014.063015.

HTGR Core Design – Overview

- General Attributes of Modular Prismatic and Pebble Bed HTGRs
    - Common Features and Physics
    - Neutronics
    - Prismatic and Pebble Fuel
    - Thermal-Fluidics
    - Inherent Safety
- Plant Systems and Power Conversion
    - Reactivity Control
    - Instrumentation and Control
    - Helium Conditioning
    - Power Conversion
- Normal Operation and Power Maneuvers

100-50-100% Load Follow Trajectory



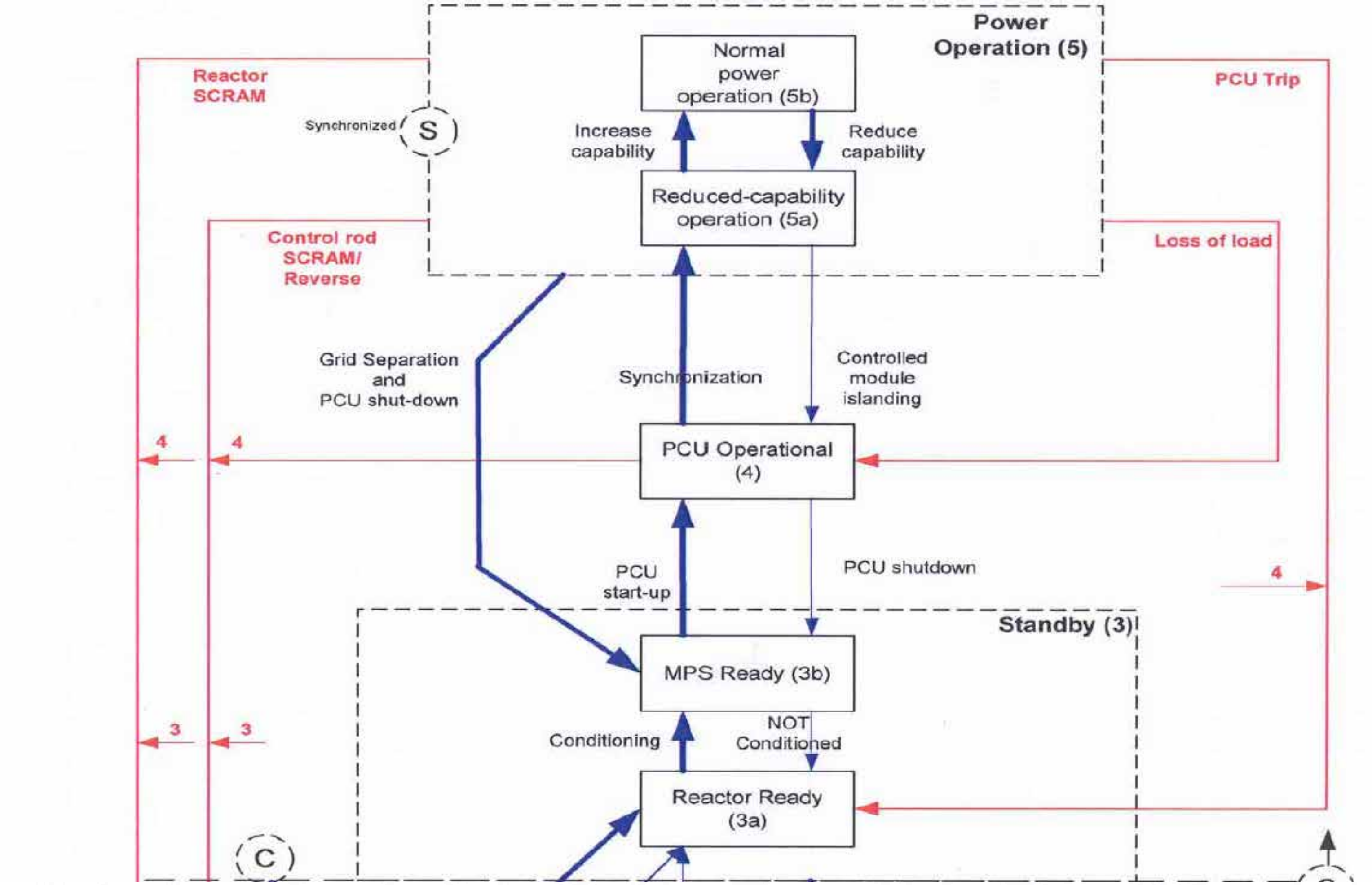
Strydom, G. (2019). Xenon-induced axial power oscillations in the 400 MW pebble bed modular reactor. Thesis (M.Sc. (Reactor Science))--North-West University, Potchefstroom Campus, 2008.

Power Operation/Load Follow

- Various maintenance, shutdown, standby and operational states are usually defined (PBMR example shown)
- Transitions between various modes/states can be complex (next slide)
- Convective heat transfer dominates during steady-state
- Flexible load-follow capability via helium mass flow rate control allows 100-40-100 e.g. power maneuvering to follow demand (PBMR limited to 1%/min)
- Load follow range mostly limited by excess fuel (+) and control rod (-) reactivity available to counter xenon swings

- Power Operation (Mode 5)
  - 100% MCR Load
  - 40% MCR load
- PCU Operational (Controlled Island Operation (Mode 4)
- Standby (Mode 3)
  - Main Power System ready
  - Reactor ready
- Shutdown (Mode 2)
  - Partial (control rods inserted only)
  - Intermediate (control rods and shutdown rods inserted)
  - Full (all rods and small absorber spheres inserted)
- Fueled Maintenance (Mode 1)
  - Helium Pressure Boundary closed
  - Open Power Conversion Unit
- Defueled Maintenance (Mode 0)

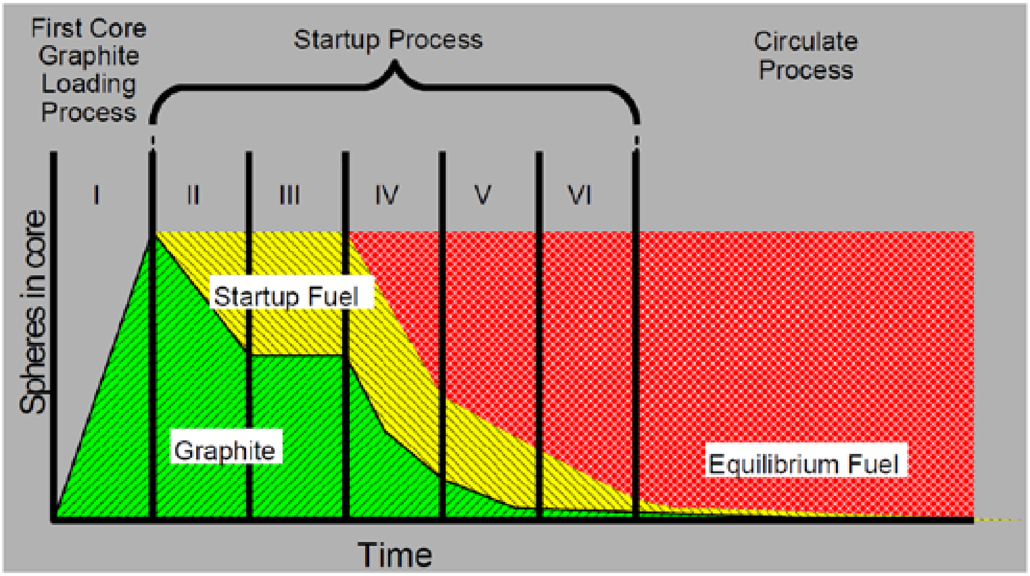
Operating Modes and States (PBMR example)



H.W. Chi, 2006. Presentation on PBMR Safety and Design Familiarization.

Transition from Startup to Equilibrium Core

- Core is initially filled with graphite spheres, and first critically is reached with mixture of graphite and fuel spheres.
- Core “running-in” phase is an optimization problem with multiple constraints:
  - peak fuel temperature <1130°C,
  - maximum power <4.5 kW/sphere,
  - minimize fuel costs - limit fuel types to two enrichments,
  - minimize time-to-full-power (revenue $ vs. time).
- Example – “revenue $ vs. time” (above) leads to discharging low-enriched start-up fuel out of the core as quickly as possible, but fuel (and fuel $) is wasted.



H. Chi, 2006: Presentation on PBMR Safety and Design Familiarization

Summary

- HTGRs occupy a special niche in the nuclear power world: really high temperatures for process heat, but still passively safe.
- (A few) HTGRs have been around awhile – a modular version is about to start up in China
- The low power density, coated particle fuel, and graphite effectively eliminate the possibility of a meltdown. Process heat user can set up operations next door.
- The physics are dominated by the graphite
- Neutronics can be challenging, but approximate methods work reasonably well if margins are quantified and care is taken with cross section generation. High fidelity neutronics are showing promise for reducing uncertainties.
- Thermal-fluidics can also be approximated with low order models, but higher fidelity models are desired. Full-core CFD is still out of reach for all but a few reference calculations
- Helium conditioning was demonstrated on Fort St. Vrain
- Steam cycle power conversions systems can exploit extensive technology developed for the fossil fuel industry; some specific components will need to be designed

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INL

Idaho National Laboratory

Backup

Diffusion Works Well

(as long as cross sections are properly generated)

$$\frac{1}{v}\frac{\partial\phi(\hat{r},t)}{\partial t} = D\nabla^2\phi(\hat{r},t) + \left(\frac{1}{k}\upsilon\Sigma_f - \Sigma_a\right)\phi(\hat{r},t)$$

With diffusion coefficient derived from transport theory

$$D = \frac{1}{3\Sigma_s(1-\bar{\mu})\left(1-\frac{4}{5}\frac{\Sigma_a}{\Sigma_t}+\cdots\right)}$$

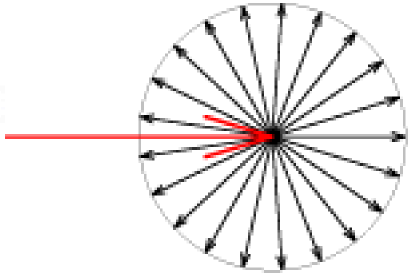
- Assumptions made in deriving the diffusion equation
  - Scattering dominates absorption
  - Flux does not change much over a distance of one mean free path
  - Scattering is isotropic (COM) or, at most, linearly anisotropic (LAB)
- Typical values for a pebble bed reactor

$$1)\ \Sigma_a \ll \Sigma_s$$

$$2)\ \lambda\frac{d\phi(x)}{dx} \ll \phi(x)$$

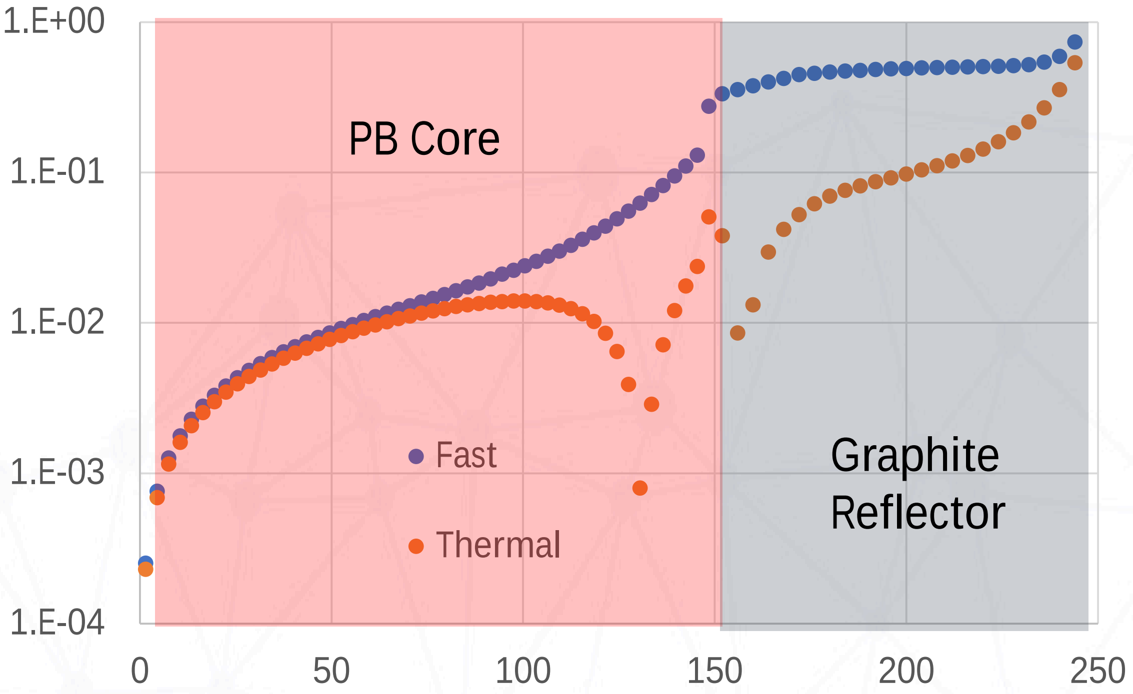
$$3)\ if\,\Sigma_a \ll \Sigma_t, D = \frac{1}{3\Sigma_s(1-\bar{\mu})} = \frac{\lambda_{tr}}{3}$$

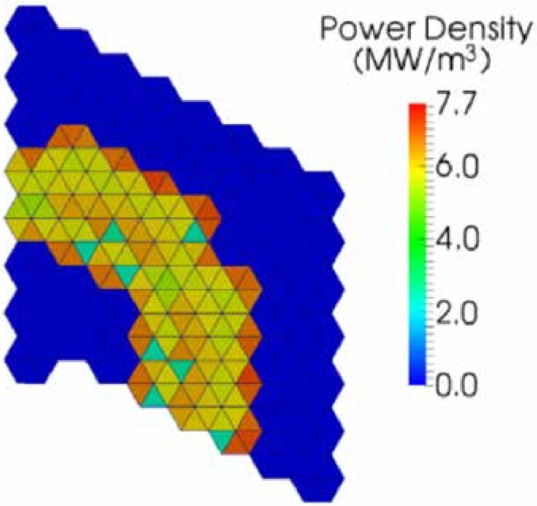
	S_s (cm^{-1})	S_a (cm^{-1})	S_t (cm^{-1})	/ (cm)
Fast	0.22	7.1E-4	0.23	4.4
Thermal	0.26	2.3E-3	0.26	3.8



Diffusion Near the Core-reflector Interface

$$\frac{\lambda}{\phi(x)} \frac{d\phi(x)}{dx} \ll 1?$$





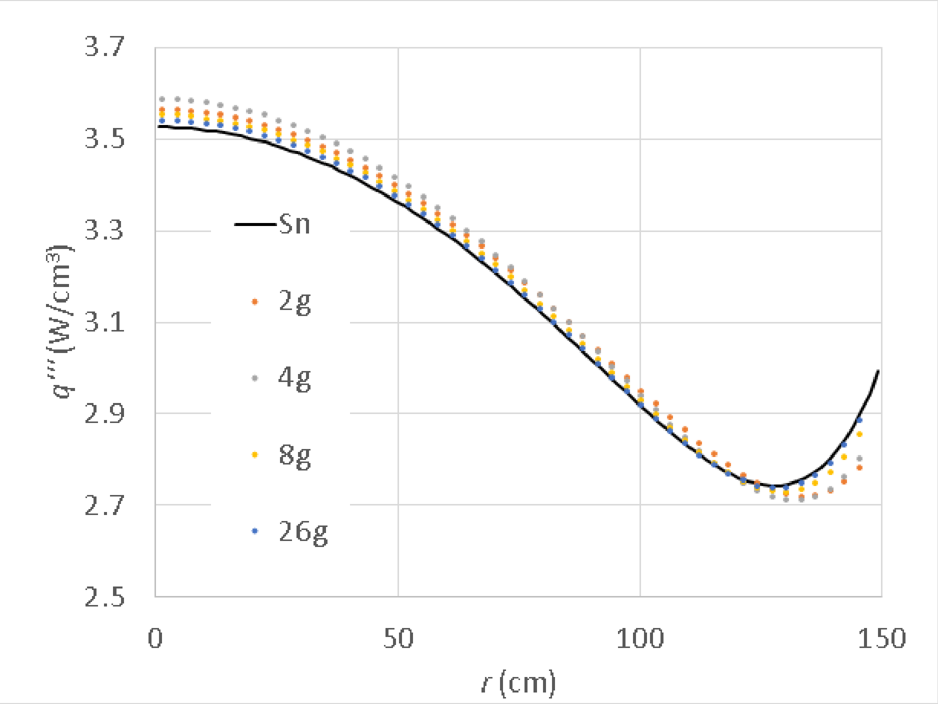
- This assumption is generally valid in the core. One runs into trouble near the core-reflector boundary and the outer boundary (where diffusion theory is known to fail)

Tak, N, et al, CAPP/GAMMA+ code system for coupled neutronics and thermo-fluid simulation of a prismatic VHTR core, Annals of Nuclear Energy 92 (2016)

Energy Group Structure and Accuracy

- Comparison of k_{eff} and power density generated with diffusion theory and S_n transport in a 1-D (radial core). Coarse groups structures may capture the global balance but finer group structures are needed to recover local reaction rates

Group Structure	Dk (pcm) – no leakage correction	Dk (pcm) – correction via supercell
2G	104	640
4G	-266	320
8G	178	160
26G	-75	-40



H. Gougar, A. Ougouag, J. Ortensi, C. Rabiti, “Suitability of Energy Group Structures Commonly Used in Pebble Bed Reactor Core Diffusion Analysis as Indicated by Agreement with Transport Theory for Selected Spectral Indices,” Proceedings of the 9th International Topical Meeting on HTR Technology (HTR 2018), Warsaw, October 2018.

Core Thermal-Fluidics: Pebble Bed

- The porous medium/nearly incompressible model yields pretty good results and is used in most steady state and transient analyses

$$\nabla \rho_G \cdot \mathbf{v} = q$$

Energy

$$\nabla p - \rho_G \cdot \mathbf{g} + \mathbf{R} = \mathbf{0}$$

Momentum

$$\mathbf{R} = \frac{\psi}{d} \cdot \frac{1-\varepsilon}{\varepsilon^3} \cdot \frac{|G|}{2\rho_G} \mathbf{G}$$

Mass

$$\nabla\lambda_G \nabla T_G - \nabla(\rho_G \cdot \mathbf{v} \cdot c_p \cdot T_G) + \alpha \cdot \frac{F}{V} \cdot (T - T_G) = 0$$

THERMIX-KONVEK

$$\varepsilon \frac{\partial \rho_f}{\partial t} + \nabla \cdot \left(\varepsilon \rho_f \vec{V}\right) = 0$$

$$\varepsilon \frac{\partial(\rho_f \vec{V})}{\partial t} + \nabla \cdot \left(\varepsilon \rho_f \vec{V}\vec{V}\right) + \varepsilon \nabla P - \varepsilon \rho_f \vec{g} + W \rho_f \vec{V} = \vec{0}$$

$$\varepsilon \frac{\partial(\rho_f E_f)}{\partial t} + \nabla \cdot \left(\varepsilon \rho_f H_f \vec{V}\right) - \varepsilon \rho_f \vec{g} \cdot \vec{V} - \nabla \cdot \left(\kappa_f \nabla T_f\right) + \alpha\left(T_f - T_s\right) - \dot{q}_f = 0$$

$$(1-\varepsilon)\rho_s C_{p,s} \frac{\partial T_s}{\partial t} - \nabla \cdot (\kappa_s \nabla T_s) + \alpha\left(T_s - T_f\right) - \dot{q}_s = 0$$

PRONGHORN

$$\Delta p = \psi \cdot \frac{1-\varepsilon}{\varepsilon^3} \cdot \frac{H}{d_p} \cdot \frac{1}{2\rho} \cdot \left(\frac{\dot{m}}{A}\right)^2$$

Pressure drop correlation derived from AVR and THTR operational data

$$\frac{\Delta p}{L} = \frac{150\mu(1-\varepsilon)\mu_o}{\varepsilon^3 d_p^2} + \frac{1.75(1-\varepsilon)\rho\mu_0^2}{\varepsilon^3 d_p^2}$$

General pressure drop correlation (Ergun) $\varepsilon \sim 0.40$

Novak, April and Zou, Ling and Peterson, John and C Martineau, R and Slaybaugh, R. (2018). Pronghorn: A Porous Media Thermal-Hydraulics Core Simulator and its Validation with the SANA Experiments.