ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

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Next Generation Nuclear Plant (NGNP)
A
Very High Temperature Gas-cooled Reactor (VHTR)

Idaho National Engineering and Environmental Laboratory (INEEL)

Workshop for Universities
Hilton Hotel, Gaithersburg, MD
March 4-5, 2004

OUTLINE

- General characteristics of high temperature gas reactors
- Highlights of prismatic block NGNP point design results
- Highlights of the pebble bed NGNP point design results
- Additional work needed on the point designs
- Code development and verification needs
- Materials research

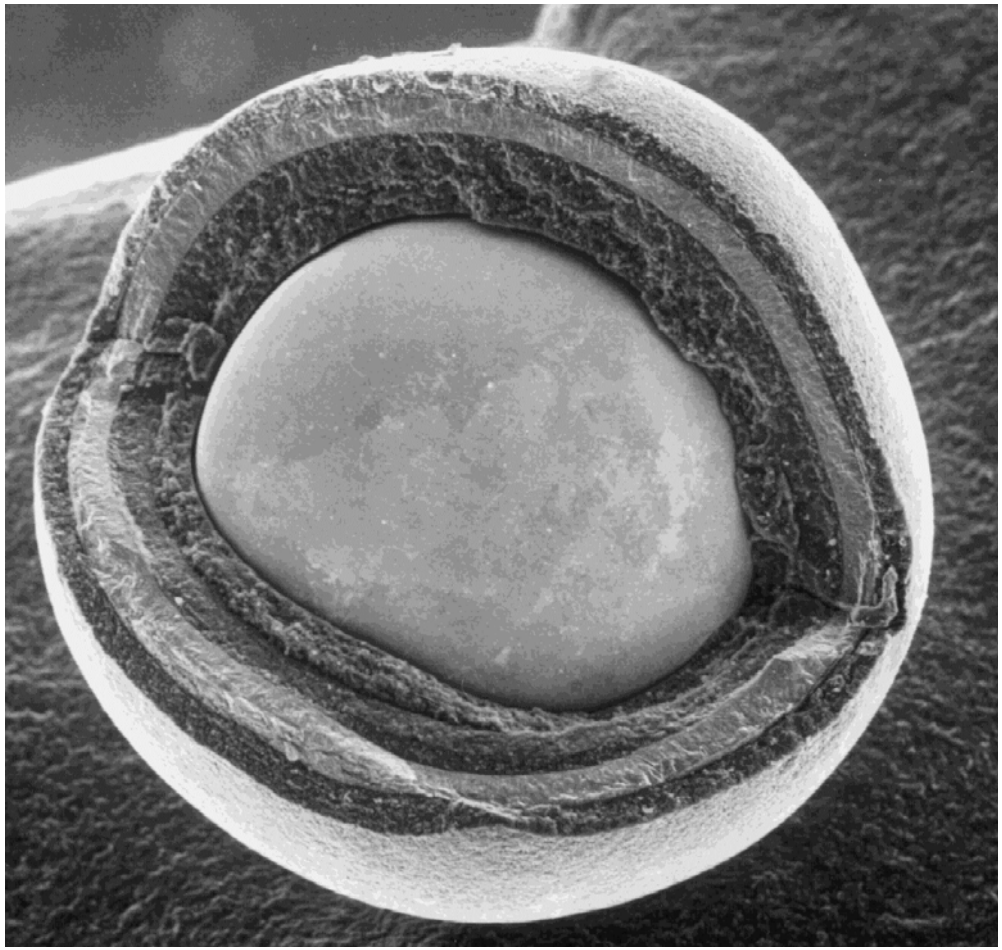
THIS RESEARCH AREA INCLUDES

The High Temperature Gas Reactor Is the Current Reference Design

- **Attractive inherent characteristics**
  - Helium coolant - inert, single phase
  - Refractory coated fuel - high temp capability, low fission product release
  - Graphite moderator - high temp stability, long response times
- **Passively safe design**
  - Relatively low power density
  - Annular core
  - Large negative temperature coefficient
  - Passive decay heat removal system
  - Silo installation

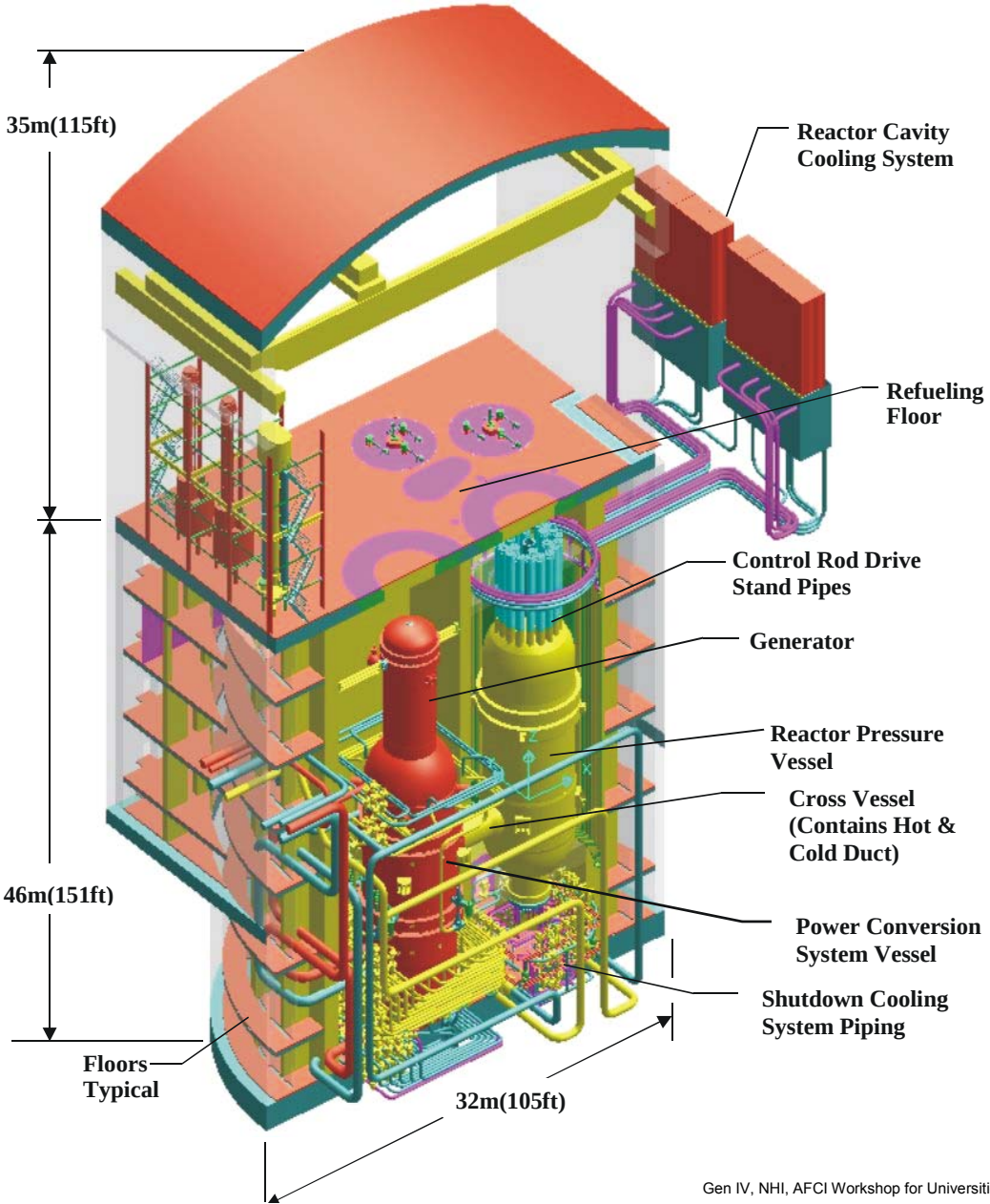
THIS RESEARCH AREA INCLUDES

The TRISO coated fuel particle provides the primary containment for the fission products



THIS RESEARCH AREA INCLUDES

The GT-MHR Plant Layout was the starting point for our analyses



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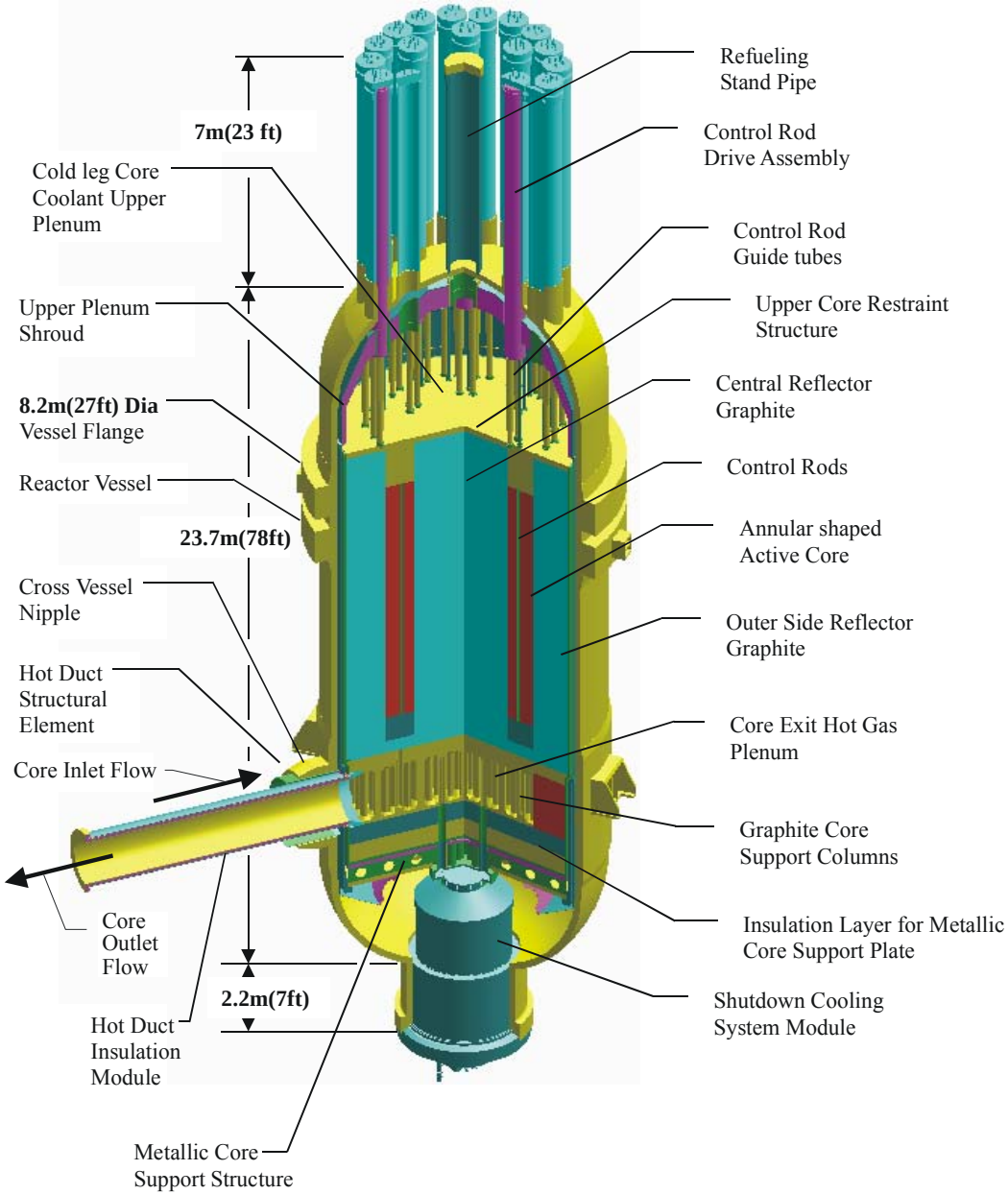
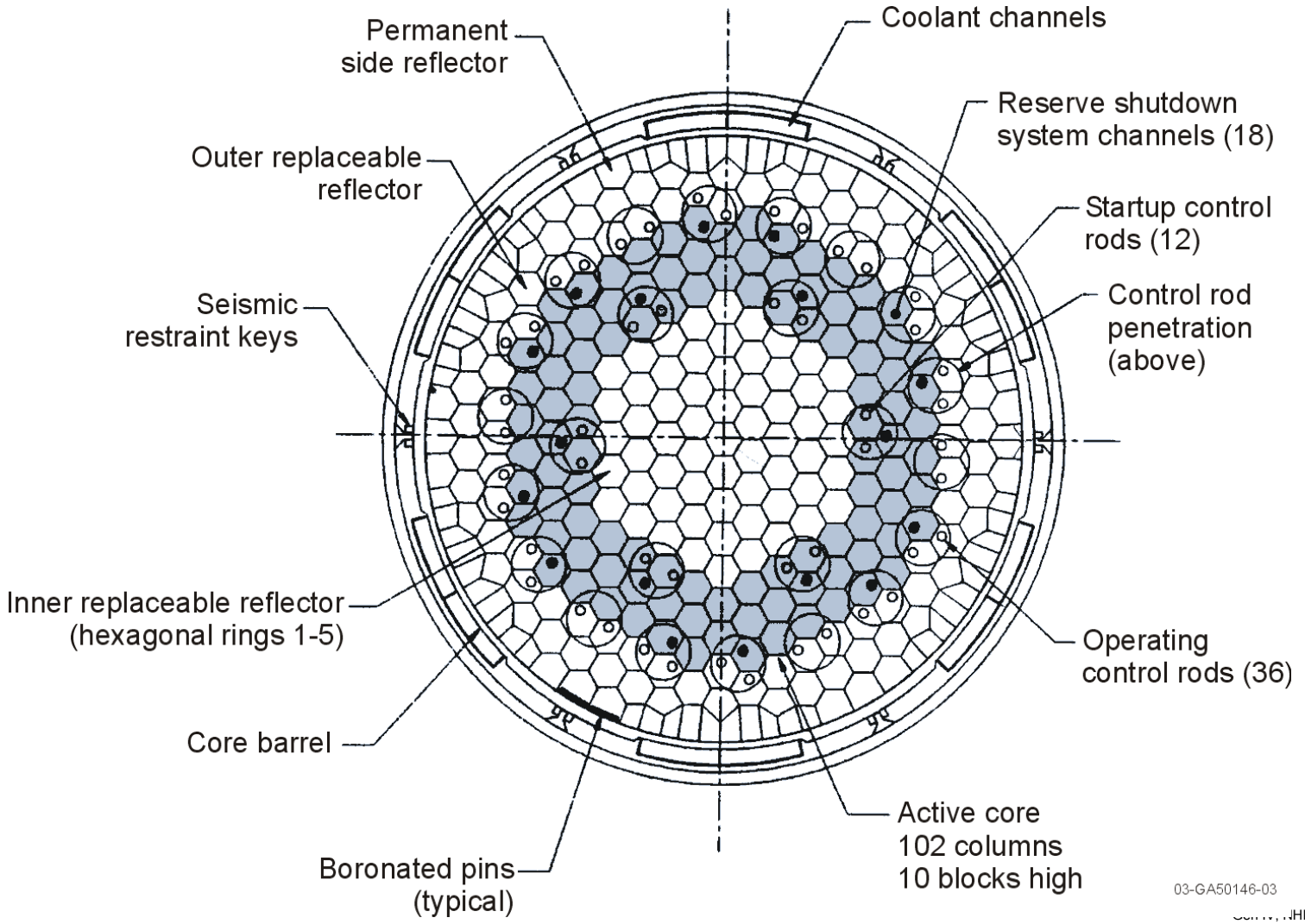


GT-MHR Reactor System Cutaway Showing Internals, Core, Control Rod Guide Tubes, and Shutdown Cooling System

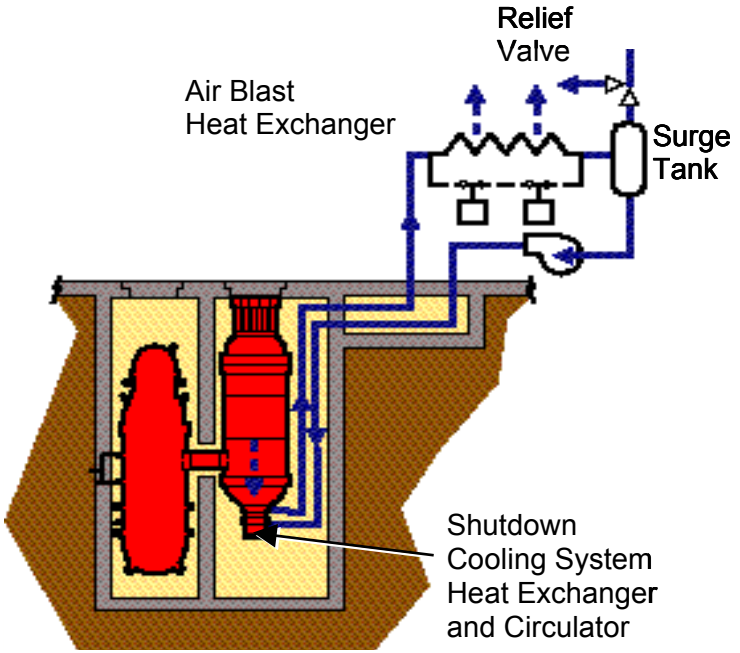
THIS RESEARCH AREA INCLUDES

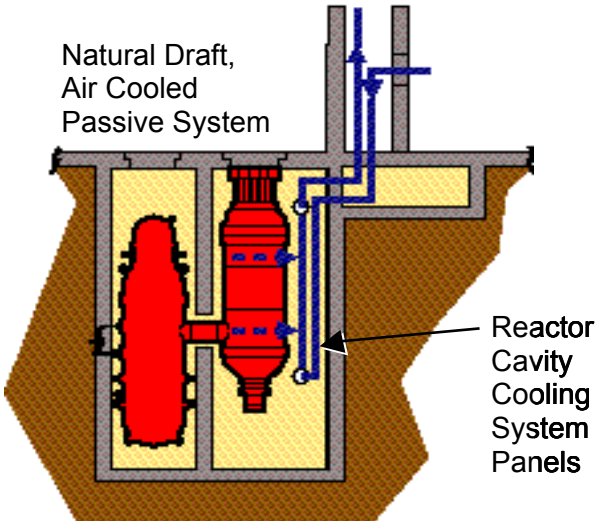
Annular Reactor Core Limits The Fuel Temperatures During Accidents

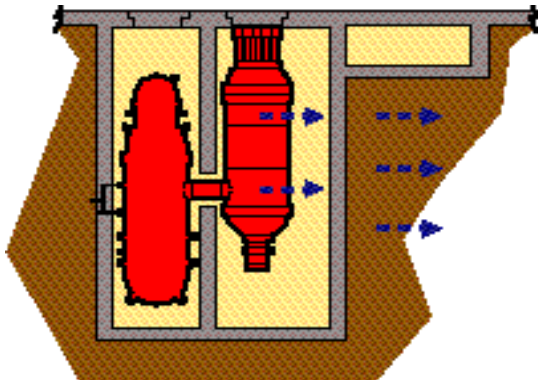


THIS RESEARCH AREA INCLUDES

Multiple Decay Heat Removal Paths







A) Active Shutdown Cooling System

B) Passive Reactor Cavity Cooling System

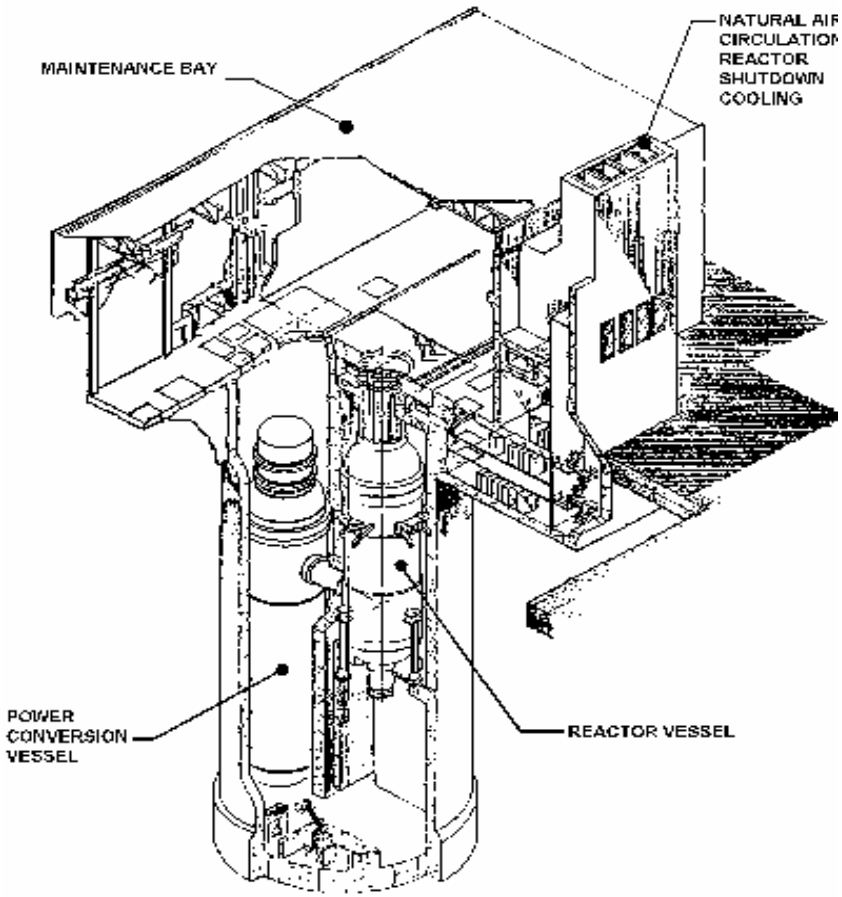
C) Passive Radiation and Conduction of Afterheat to Silo Containment (Beyond Design Basis Event)

. . . DEFENSE-IN-DEPTH BUTTRESSED BY INHERENT CHARACTERISTICS

THIS RESEARCH AREA INCLUDES

Vented Low-Pressure Containment

- Closed, vented, below-grade building housing reactor and power conversion system
- Volume provides holdup and retention of fission products.
- Vent prevents overpressure, protecting building and reactor cavity cooling system.
- Not challenged by transients in other barriers - boundaries are independent



THIS RESEARCH AREA INCLUDES

Advantages Over High-Pressure Containment

- **Fewer fault modes, fewer active components, requires less operator supervision**
- **Matched to GT-MHR accident behavior**
  - Vented early in transient when radionuclides released are low
  - Closed later in transient when fuel sees maximum temperatures
- **Compatible with highly-reliable passive air-cooled reactor cavity cooling system design**
- **Much lower cost than conventional containment without sacrificing safety**

THIS RESEARCH AREA INCLUDES

Comparison of Example VHTR Operating Conditions and Features with GT-MHR and Fort St. Vrain

Condition or Feature	Fort St. Vrain HTGR	GT-MHR	VHTR
Power Output (MWt)	841	600	600 - 900 (Depends on core heigh
Average power density (w/cm^3)	6.3	6.5	4 - 6.5
Coolant and Pressure (MPa / psia)	Helium @ 4.83 / 700	Helium @ 7.12 / 1032	Helium @ 7.12 / 1032
Moderator	Graphite	Graphite	Graphite
Core Geometry	Cylindrical	Annular	Annular
Safety Design Philosophy	Active Safety Sys	Passive	Passive
Plant Design Life (yrs)	30	60	60
Core outlet temp. (°C)	785	850	1000
Core inlet temp. (°C)	406	488	490 (Needs to be optimized
Fuel – Coated Particle	HEU-PyC/SiC Th/ 93% ^{235}U	LEU-PyC/SiC	a) LEU-PyC/SiC b) LEU-PyC/ZrC
Fuel Max Temp – Normal Operation (°C)	1260	1250	a) ~1250 b) ~ 1400

FY03 ACCOMPLISHMENTS

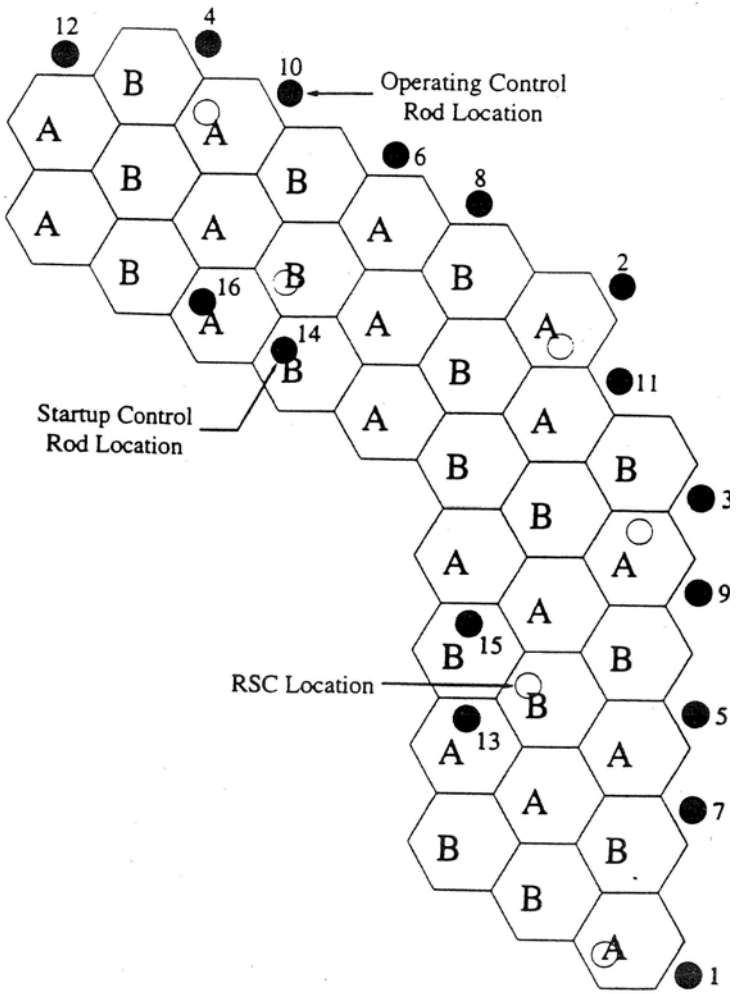
NGNP Core Point Design

- **Two design requirements :**
  - inherent safety under all accident or transient conditions
  - 1,000 °C outlet gas temperature
- **The first condition drives us immediately to an annular core configuration with substantial inner reflector graphite mass for absorption and temporal storage of thermal energy during the transients.**
- **The second condition requires either higher inlet temperatures or significant revisions in the core thermal-hydraulic design.**

FY03 ACCOMPLISHMENTS

Block NGNP Steady-State Design (1)

- **The inlet pressure and temperature were specified**
- **POKE was used to calculate**
  - the flow distribution in 1/3 of the core
  - the temperatures of the coolant, graphite, and fuel at each axial location for each column
  - the axial pressure distribution in each column and the overall pressure drop across the core
- **The power distribution is based on 3-D core-physics calculations for the 600-MWt GT-MHR, fueled with low-enriched uranium and operating at the middle of an equilibrium cycle.**



FY03 ACCOMPLISHMENTS

Block NGNP Steady-State Design (2)

- Effects of reducing bypass flow

	Bypass Flow Fraction		
	0.2	0.15	0.1
Max Fuel Temperature (°C)	1361	1334	1309
Max Outlet Temperature (°C)	1169	1145	1124
Core Pressure Drop (psid)	8.1	9.0	10.0

- Effects of controlling flow distribution

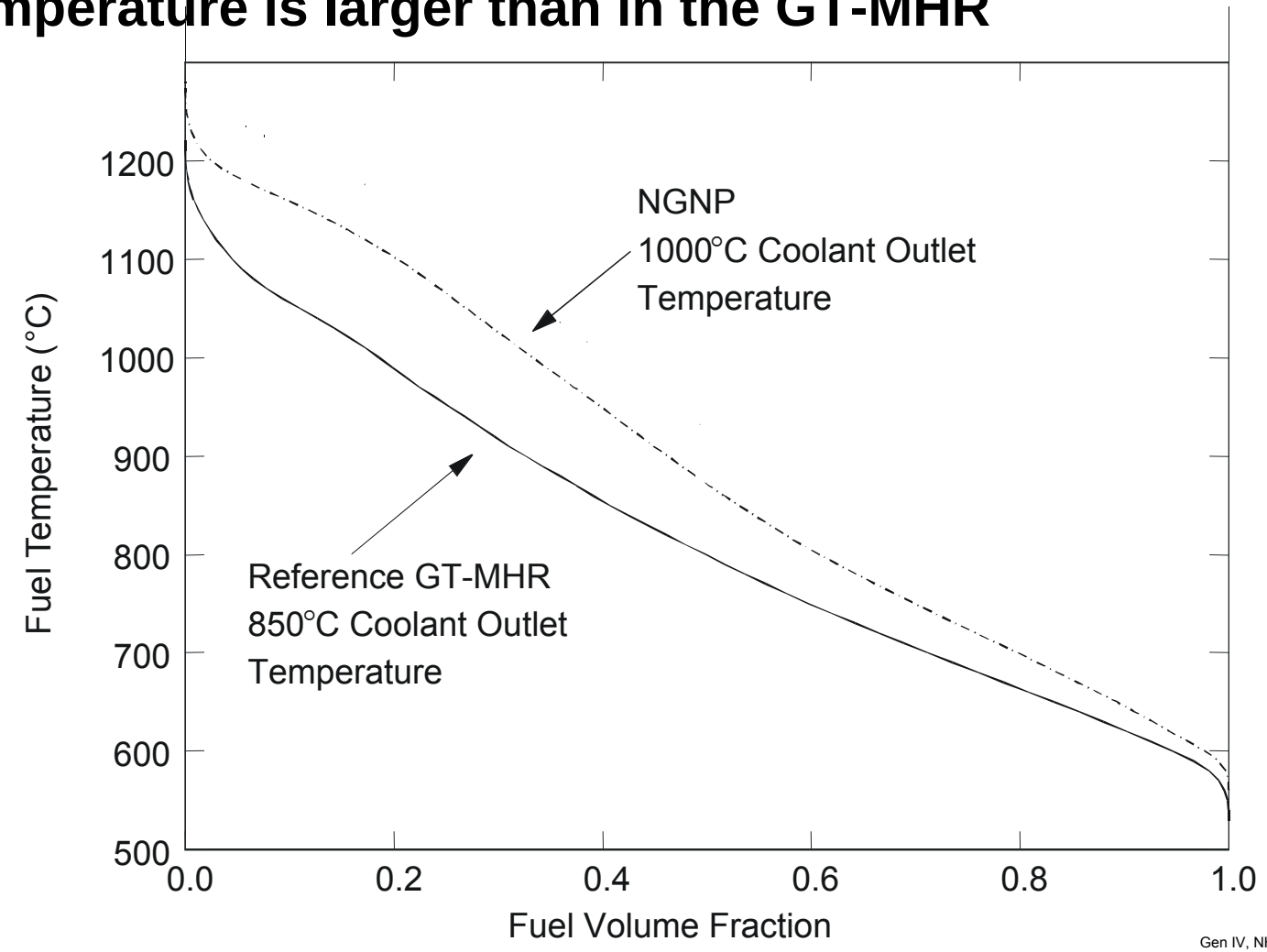
	None	Optimized by POKE	Optimized by POKE
Inlet Temperature (°C)	641	641	491
Flow Rate (kg/s)	320	320	226
Average Outlet Temperature (°C)	1000	1000	1000
Max Fuel Temperature (°C)	1309	1204	1239
Max Outlet Temperature (°C)	1124	1030	1042
Core Pressure Drop (psid)	10.0	14.5	6.9

- **Note that these results do not include local uncertainties in power and flow (hot channel factors)**

FY03 ACCOMPLISHMENTS

Block NGNP Steady-State Design (3)

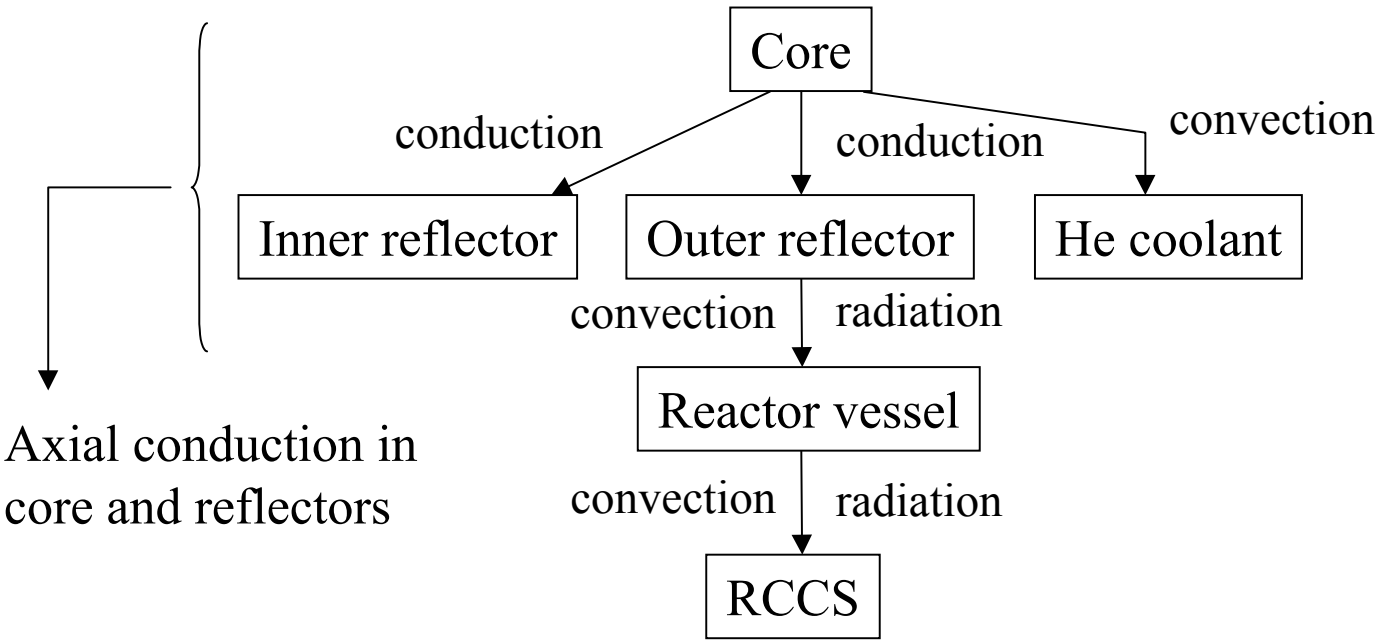
The volume fraction of fuel in the NGNP above a given temperature is larger than in the GT-MHR



FY03 ACCOMPLISHMENTS

Block NGNP Low Pressure Conduction Cool-Down Accident Analyses (1)

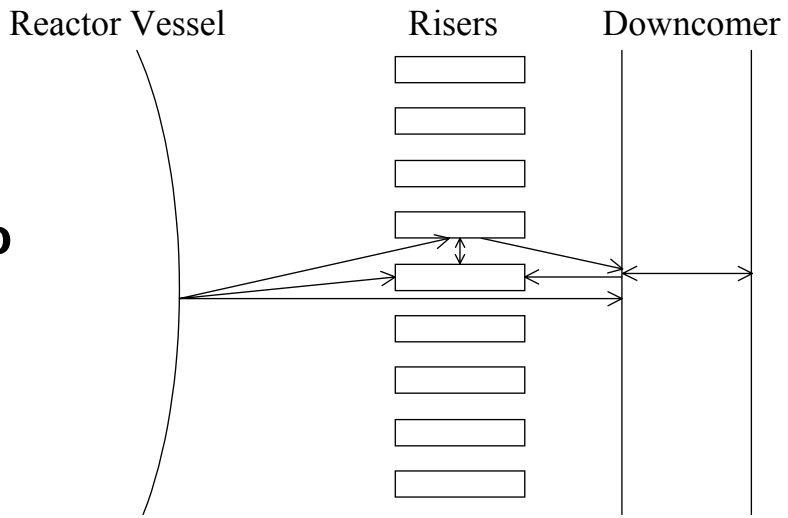
- **The RELAP5-3D/ATHENA computer code was used at the INEEL to analyze this event**
- **Helium is the working fluid in the primary system**
- **The containment and reactor cavity cooling system (RCCS) are modeled as containing dry air**



FY03 ACCOMPLISHMENTS

Block NGNP Low Pressure Conduction Cool-Down Accident Analyses (2)

- The RCCS is modeled as an air-cooled system
- Air at 43 °C enters the inlet plenum above the downcomer, then flows through the downcomer (which is attached to the containment wall) to the bottom of the reactor compartment, where it is distributed to the riser channels
- The hot air leaving the risers is collected in a plenum, then discharged back to the atmosphere

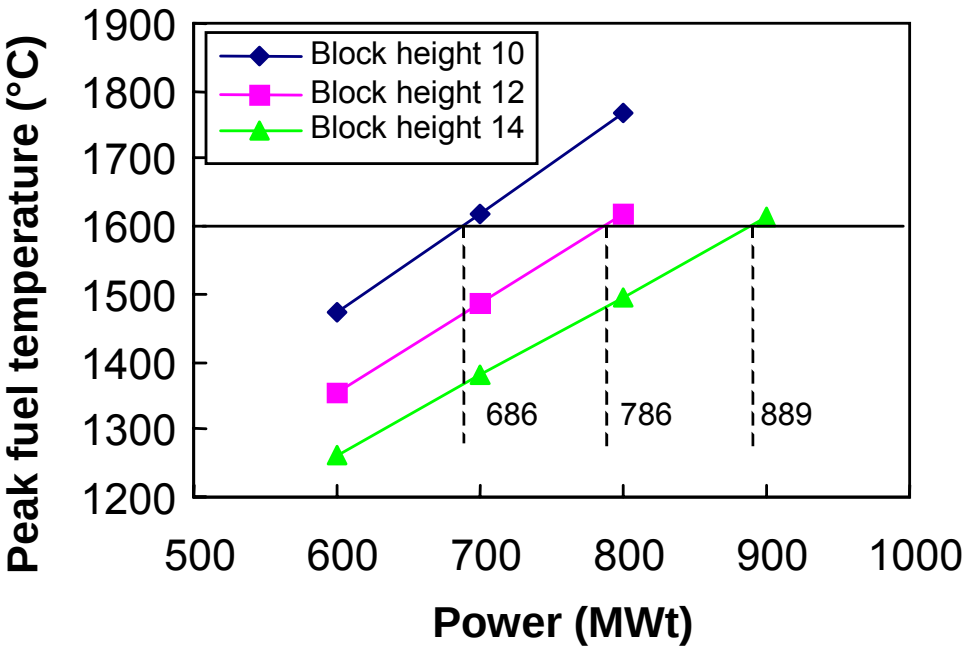


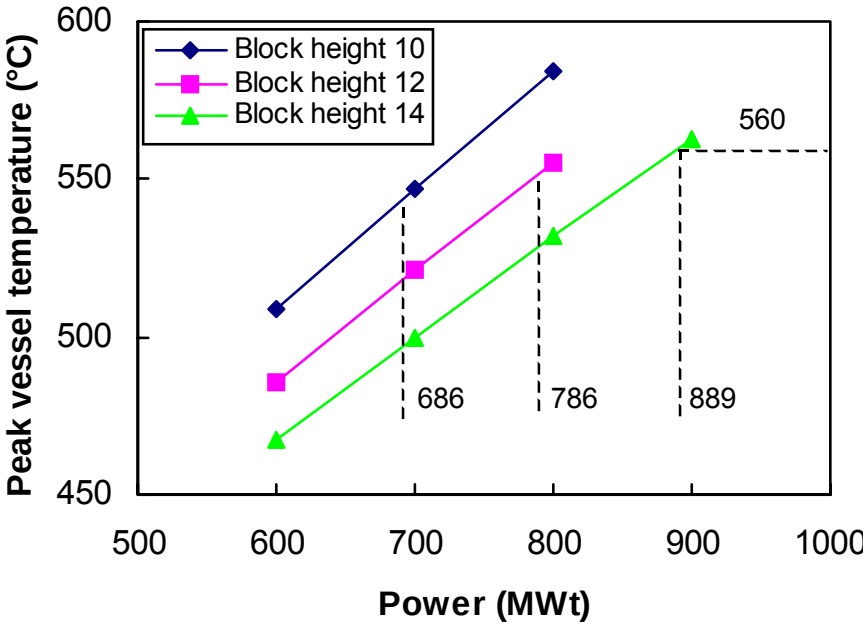
- Emissivity values of 0.8 were used for the core barrel, reactor vessel, and RCCS structures
- An emissivity of 0.1 was used for the RCCS insulated downcomer wall facing the reactor vessel

FY03 ACCOMPLISHMENTS

Block NGNP Low Pressure Conduction Cool-Down Accident Analyses (3)

The maximum fuel temperatures during the LPCC transient with 10% core bypass determine the allowable power as a function of core height

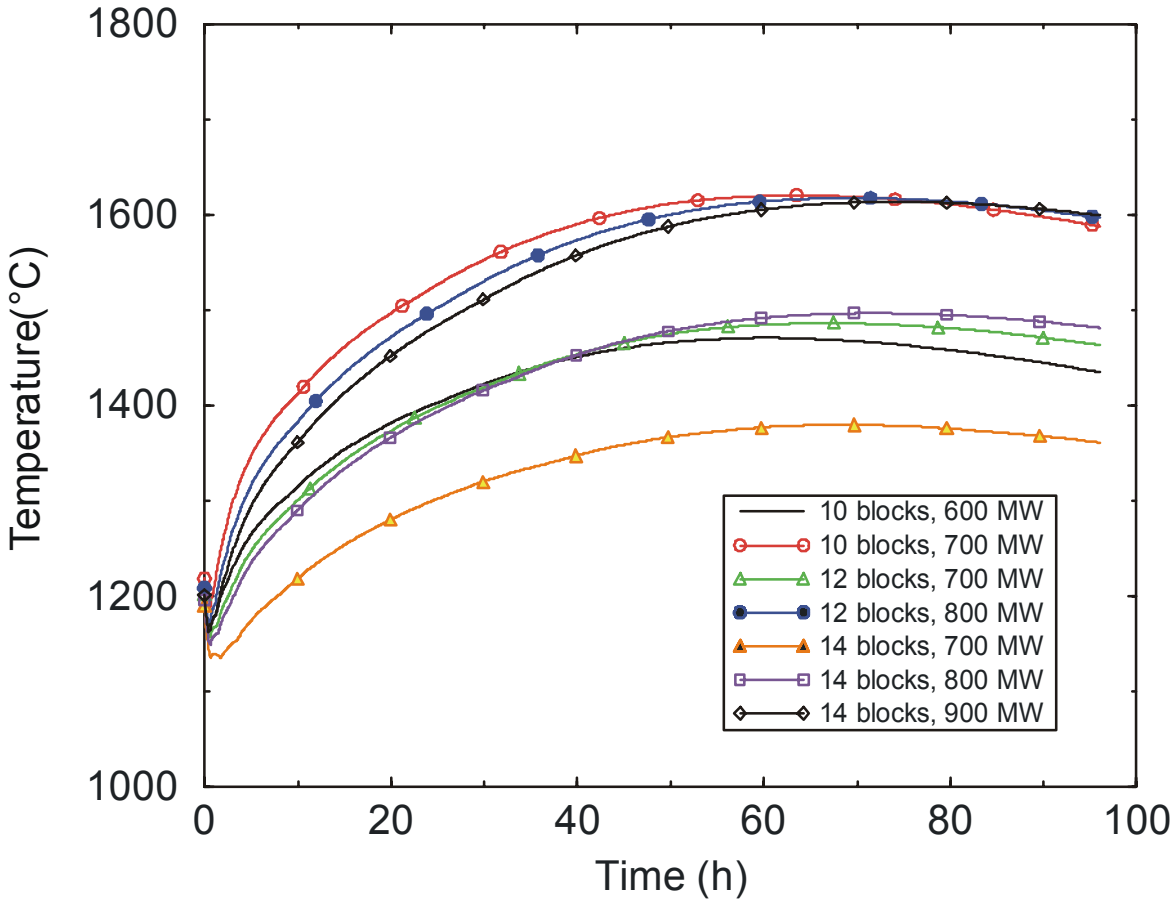




FY03 ACCOMPLISHMENTS

Block NGNP Low Pressure Conduction Cool-Down Accident Analyses (4)

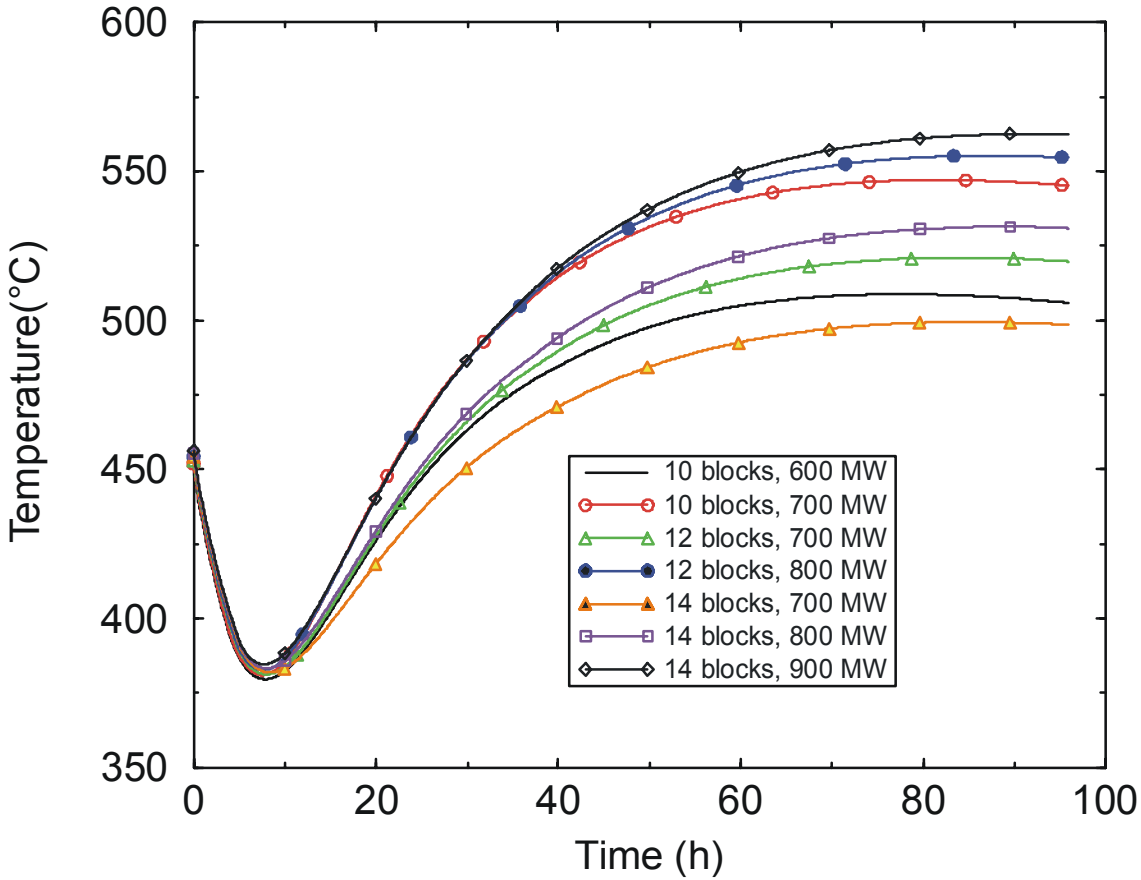
The peak fuel temperatures during the LPCC transient with 10% core bypass occur after about 60 hours



FY03 ACCOMPLISHMENTS

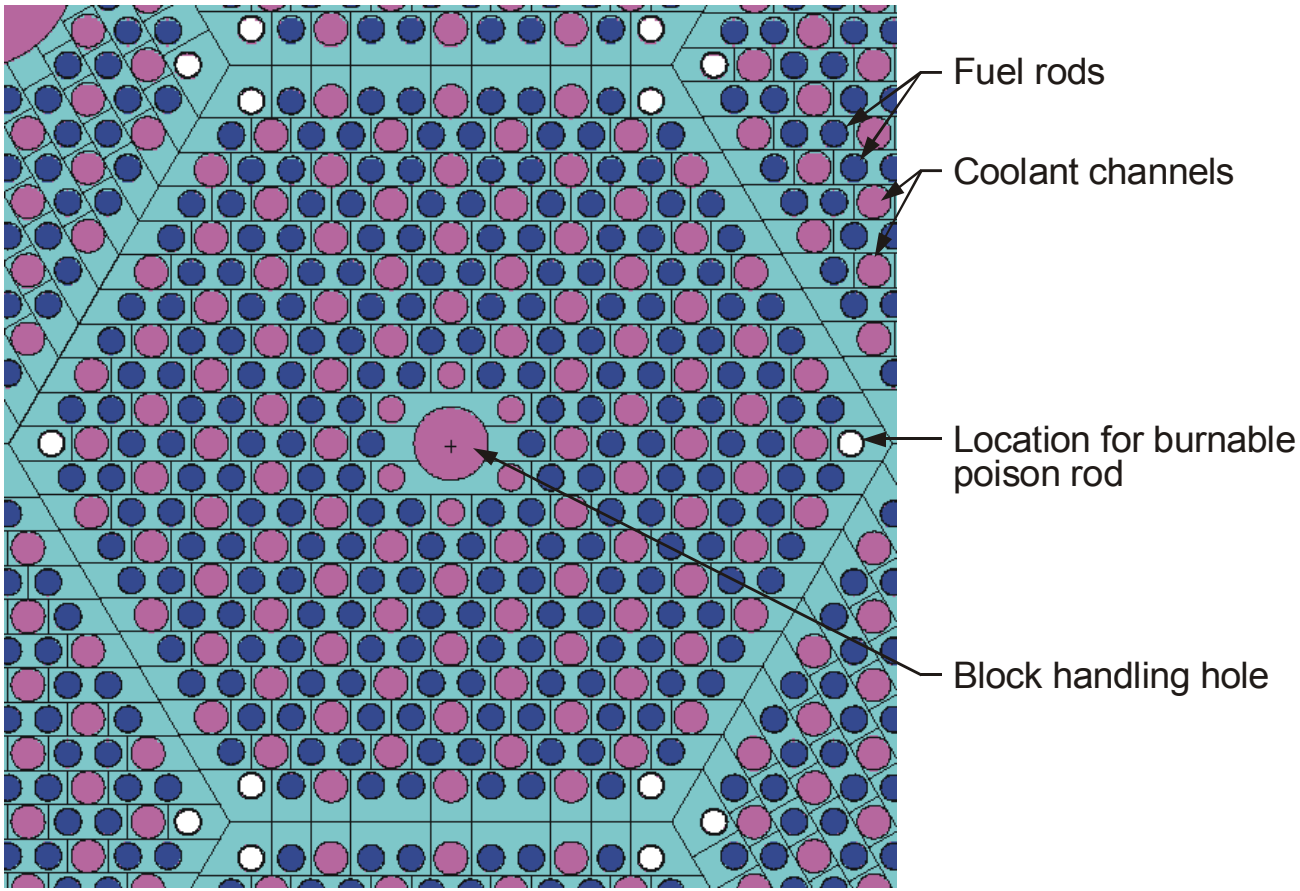
Block NGNP Low Pressure Conduction Cool-Down Accident Analyses (5)

The peak reactor vessel temperatures during the LPCC transient with 10% core bypass occur after about 80 hours



Block NGNP Core Neutronics Design (1)

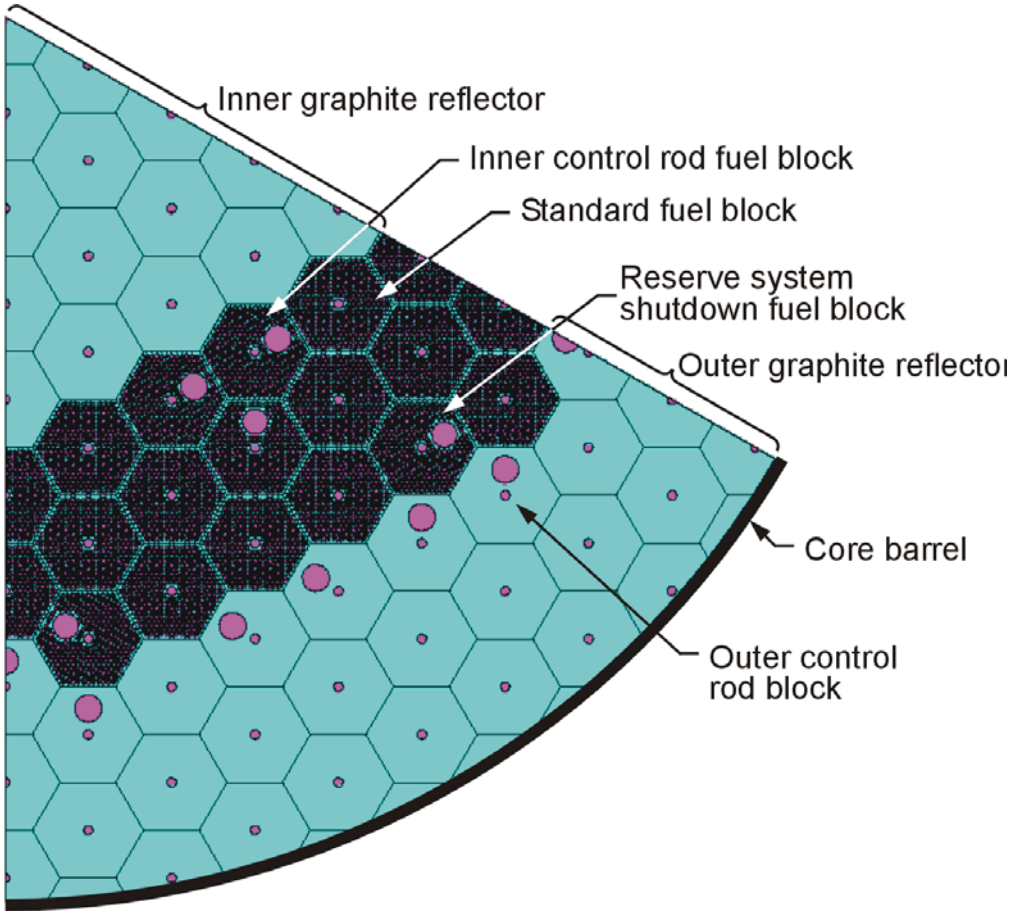
All of the fuel rods, coolant channels, and other core features were explicitly defined in the MCNP-ORIGEN block models



FY03 ACCOMPLISHMENTS

Block NGNP Core Neutronics Design (2)

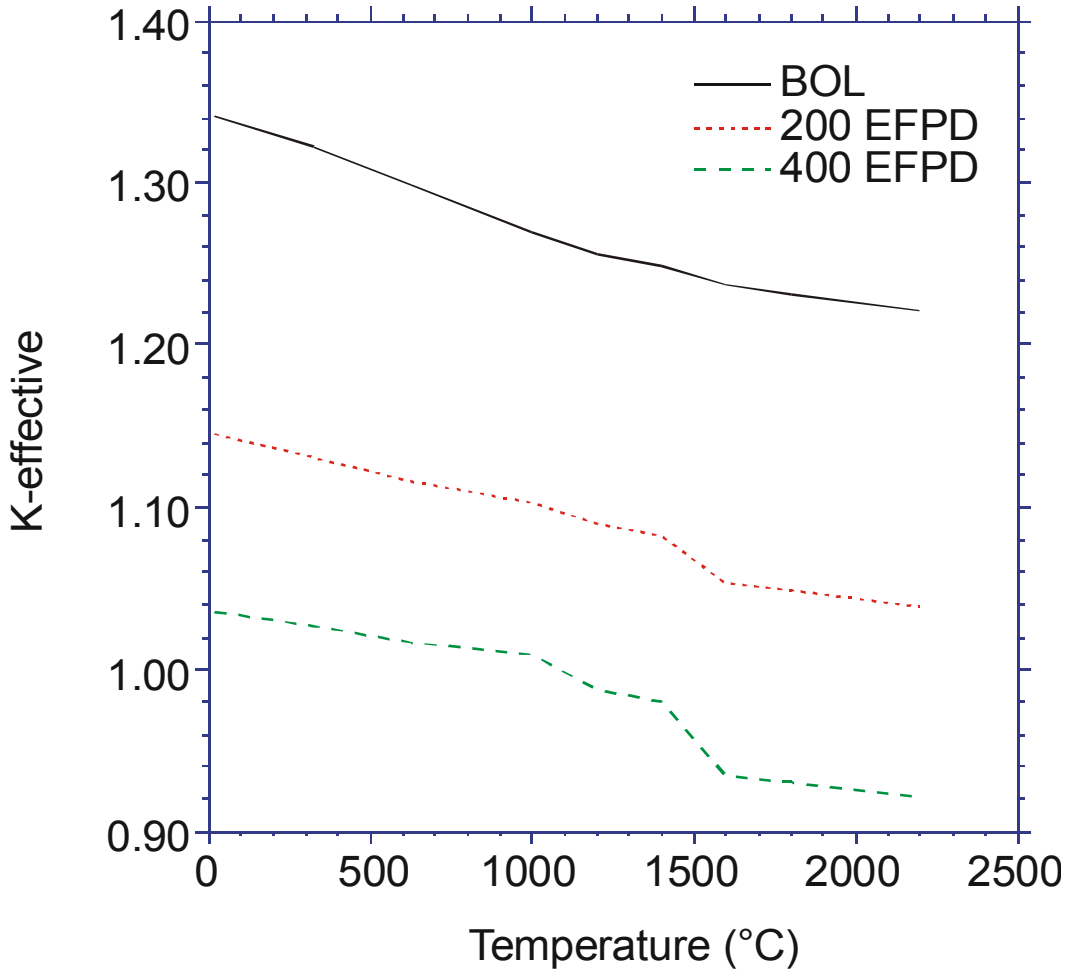
- The core models were 1/6 core radial wedge models with reflective boundary conditions applied to the azimuthal planes
- Both 1/6-core single block and full core height (including top and bottom reflector blocks) models were run
- The reactivity was about the same for the 1/6-core single block and full core height model



FY03 ACCOMPLISHMENTS

Block NGNP Core Neutronics Design (3)

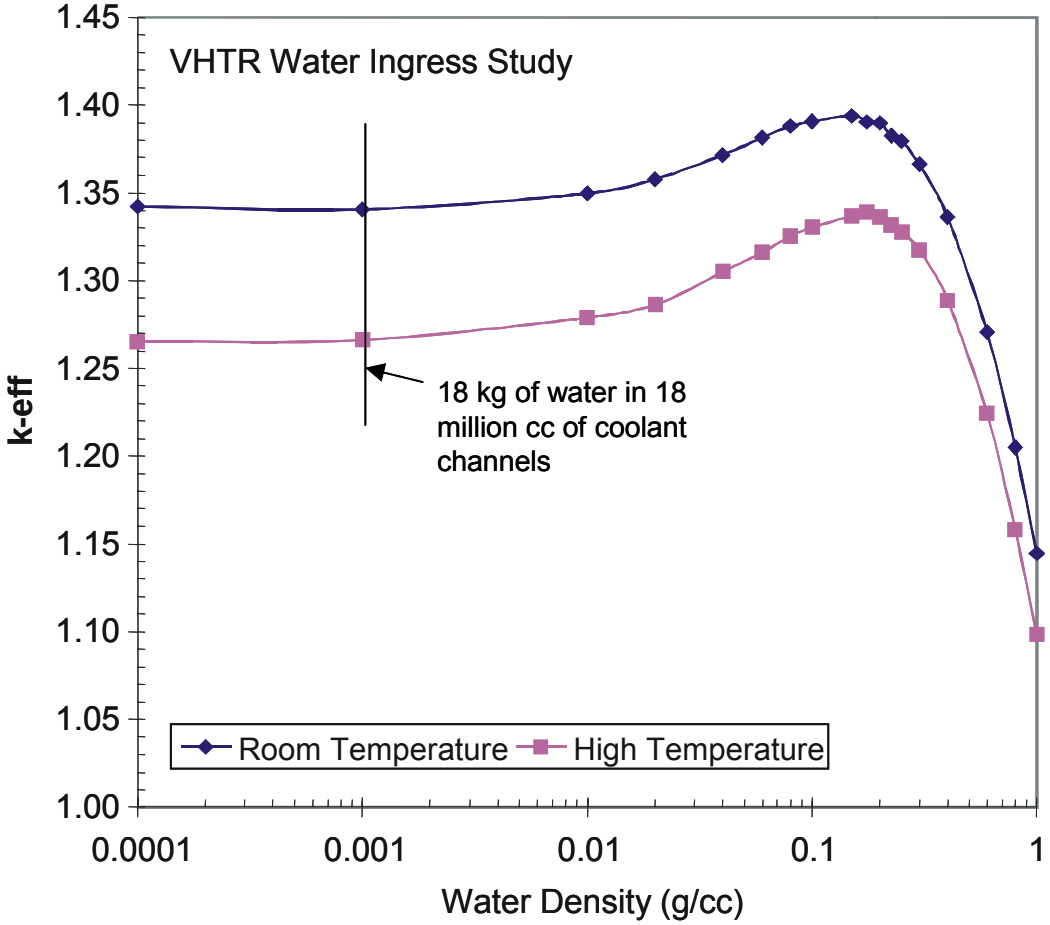
The isothermal core k-effective as a function of core temperature is negative



FY03 ACCOMPLISHMENTS

Block NGNP Core Neutronics Design (4)

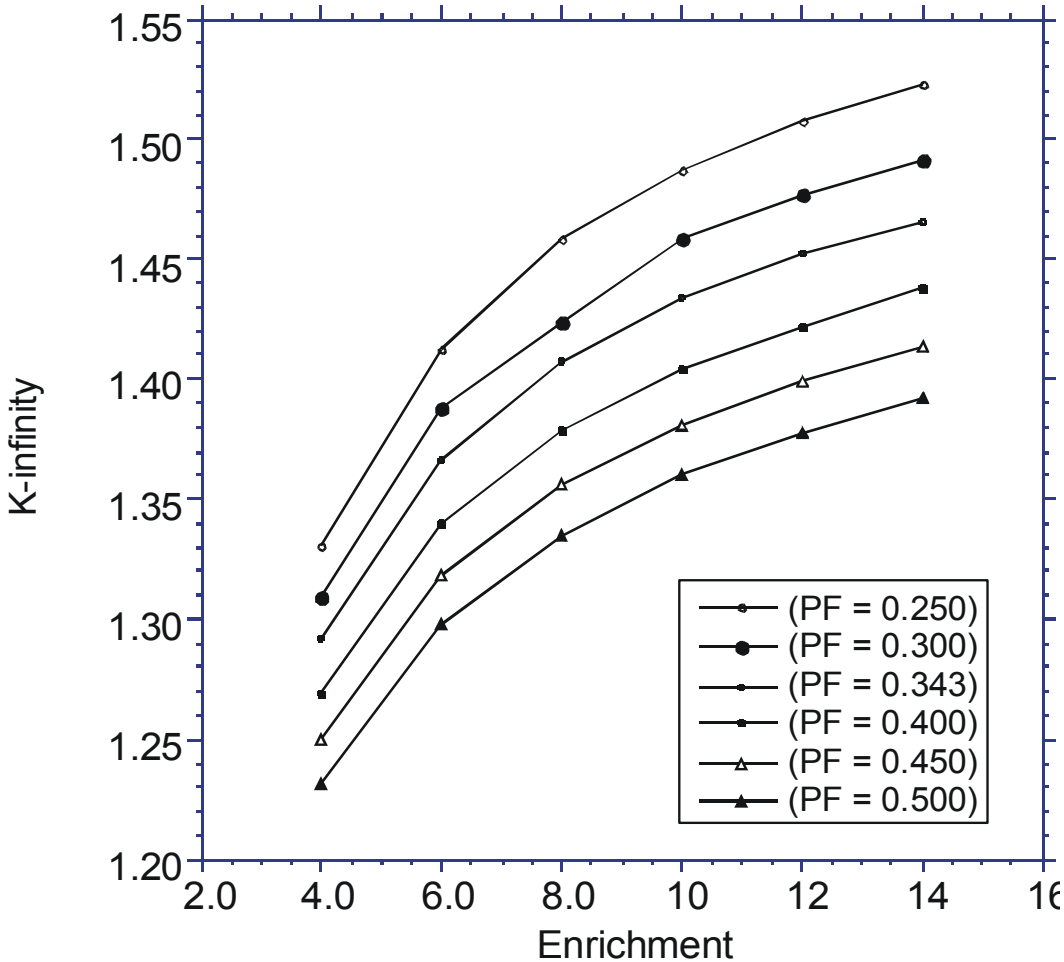
Significant water leakage into the prismatic NGNP results in a significant reactivity insertion. Complete flooding results in a reduction in reactivity.



FY03 ACCOMPLISHMENTS

Block NGNP Core Neutronics Design (5)

Standard block lattice k-infinity versus fuel enrichment and particle packing fraction (PF)



- The k-infinity values decrease as the packing fraction increases
- The larger packing fractions allow heavier U-235 loading with suppressed reactivity at BOL (due primarily to thermal neutron self-shielding)
- This effect can be exploited for the goal of increasing the NGNP power cycle length

FY03 ACCOMPLISHMENTS

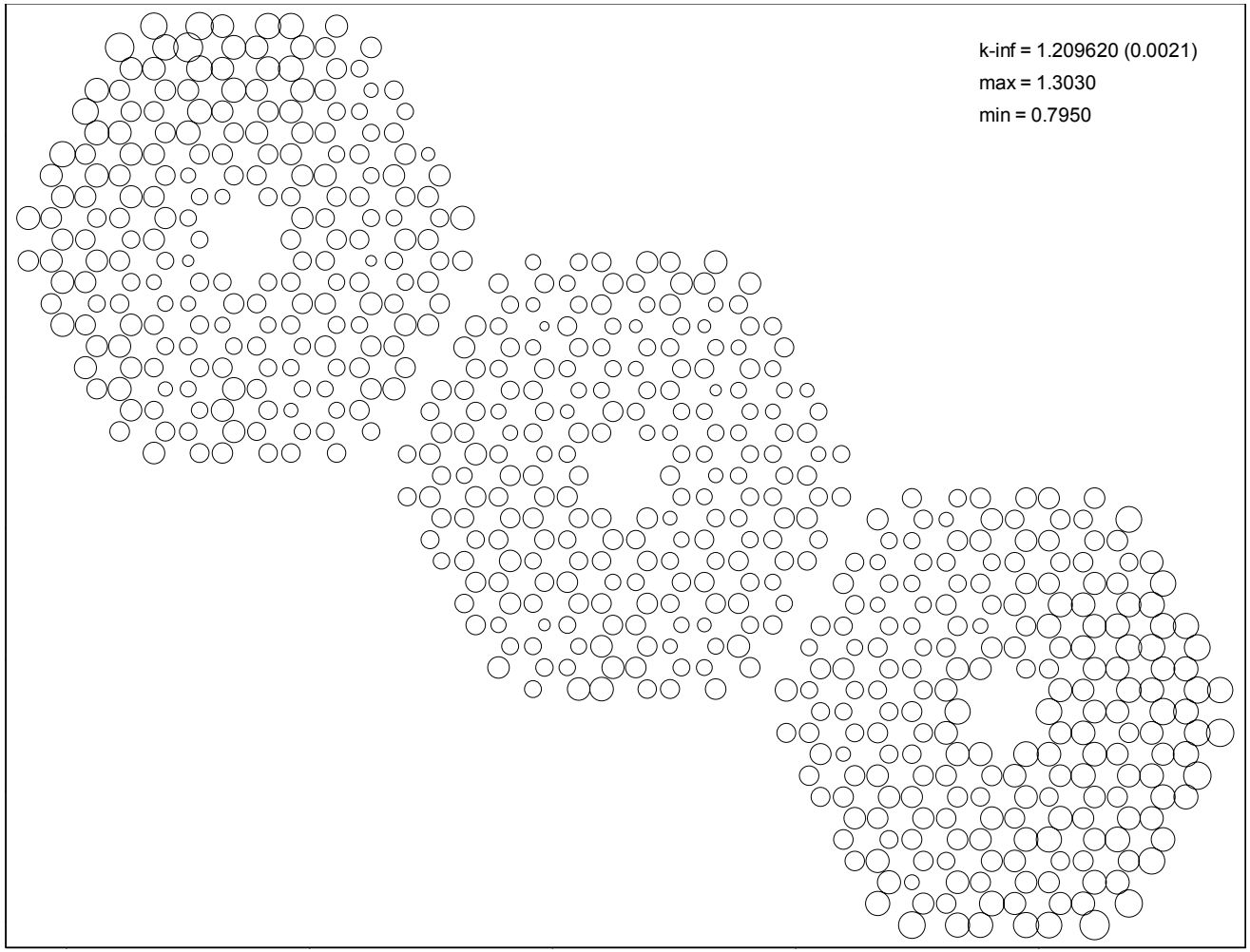
Block NGNP Core Neutronics Design (6)

- **The radial power peaking at the core-inner reflector interface can be significant (about 1.6 in a uniform core)**
- **Peaking parametric studies currently under investigation involve the following options:**
  1. Loading discrete burnable poison rod locations with burnable poisons
  2. Substitution of burnable poison rods for fuel rods near the interface
  3. Variation of the particle packing fraction in the fuel rods near the interface
  4. Variation of the uranium enrichment in the fuel rods near the interface
  5. Burnable poison loading in the inner graphite reflector blocks
  6. Use of different burnable poisons (B-10, Gd, Er, etc)
  7. Mixing Gd particles with fuel particles in the compacts near the inner reflector

FY03 ACCOMPLISHMENTS

Block NGNP Core Neutronics Design (7)

- **Three B_4C rods in each fuel block at the inner reflector interface only. The B_4C is 1.0% of the graphite shim and matrix mixture.**



WORK IN PROGRESS FOR FY04

Block NGNP Open Design Questions (1)

- What is the optimum fuel particle diameter, fuel compact diameter, and packing fraction?
- Is there a more optimal block design?
- How much more can we reduce the radial power peaking?
- What is the optimum fuel management strategy (a four ring configuration with in-out and top down block movements may be best)?
- What is the power peaking in the individual fuel microspheres (using double heterogeneity models)?
- How high can we make the core and still retain mechanical and neutronic stability?
- Do we want to minimize the power in the lower region of the core?
- What are the optimal control locations?

Block NGNP Open Design Questions (2)

- What is the lowest possible coolant inlet temperature for a plant with an efficiency near 50%?
- How well can we orifice the flow channels and how hot is the hot channel?
- Will we have adequate mixing in the lower plenum?
- How well can we control the core bypass flow and how much does it change during irradiation?
- Do longer, higher power cores make economic sense?

FY03 ACCOMPLISHMENTS

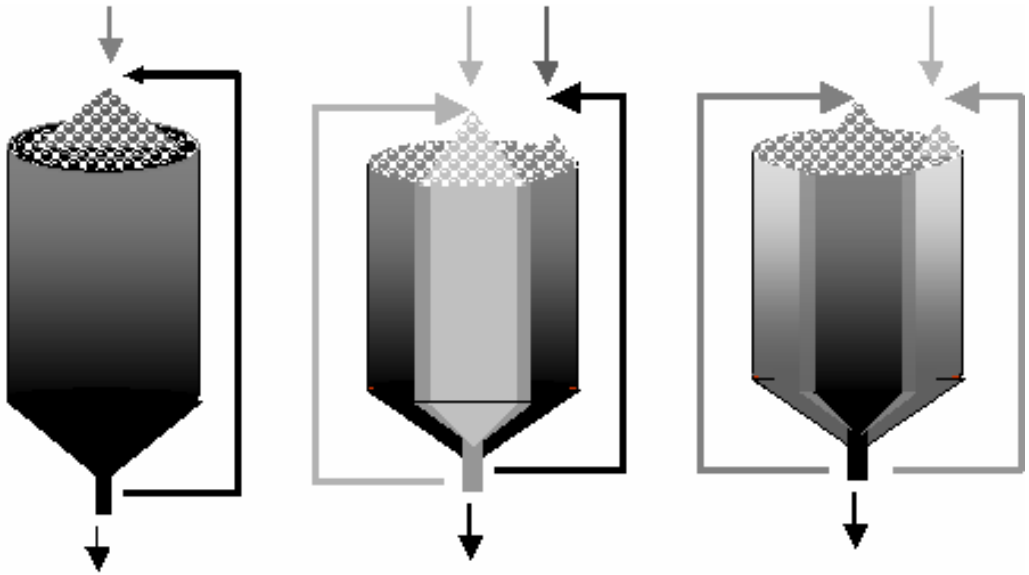
Pebble Bed NGNP

- **The same two design requirements :**
  - inherent safety under all accident or transient conditions
  - 1,000 °C outlet gas temperature
- **Again, mandate an annular core configuration with substantial inner reflector graphite mass for absorption and temporal storage of thermal energy during the transients**
- **And require significant revisions in the core thermal-hydraulic design**

FY03 ACCOMPLISHMENTS

Pebble Bed NGNP - PEBBED Fuel Cycle Analysis

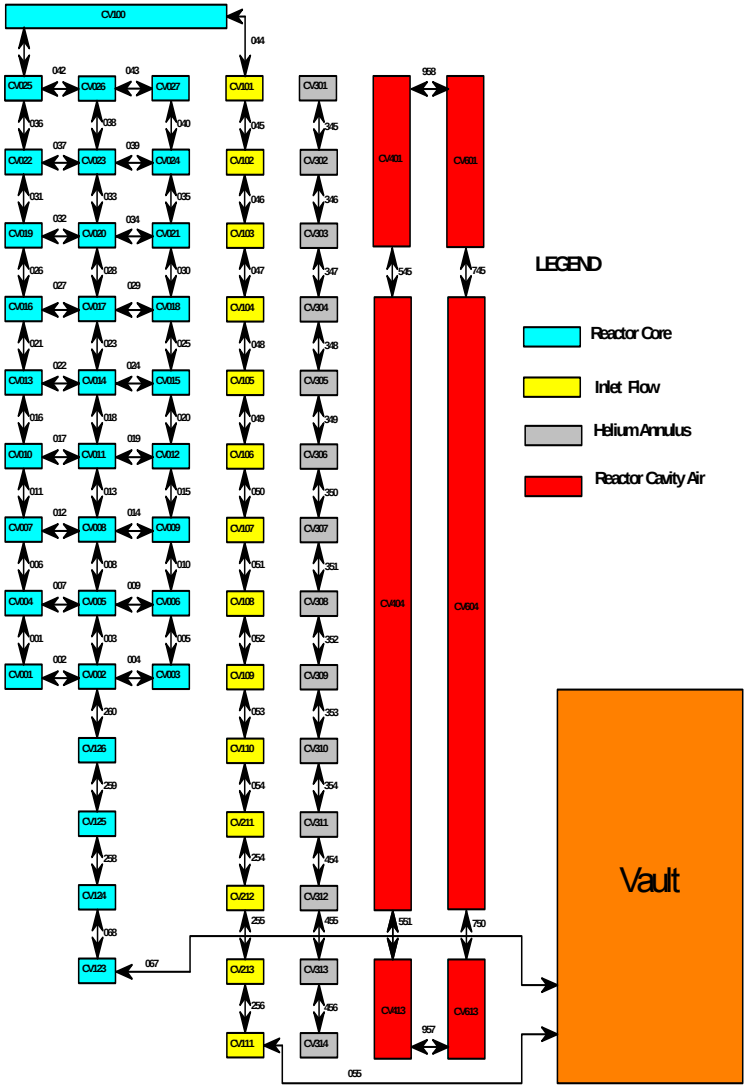
- Simultaneous, self-consistent solution to diffusion and nuclide balance equations in a reactor with a flowing core
- Exact solution of nuclide density over mesh
- Entry plane burnup computed for arbitrary, user-defined recirculation patterns
- Modules for estimating nominal and accident fuel temperatures
- Automated optimization technique



FY03 ACCOMPLISHMENTS

Pebble Bed NGNP – MELCOR Analyses

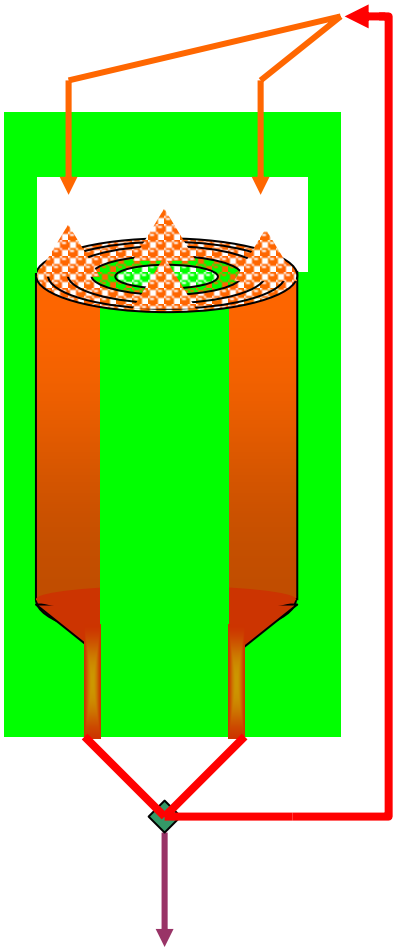
- Integrated systems-level code developed at Sandia
- Used for LWR severe accident analysis by the NRC but flexible enough to analyze other reactor types
- Modified by the INEEL to treat graphite oxidation and alternate fluids



FY03 ACCOMPLISHMENTS

Search for the NGNP-300/600

- **Base design – PBMR***
- **Replaced pebble inner reflector with solid graphite**
- **Optimized the pebble design**
- **Investigated size of inner reflector**
- **Sampled a range of inner and outer fuel annulus radii**
- **Tweaked height and discharge burnup**

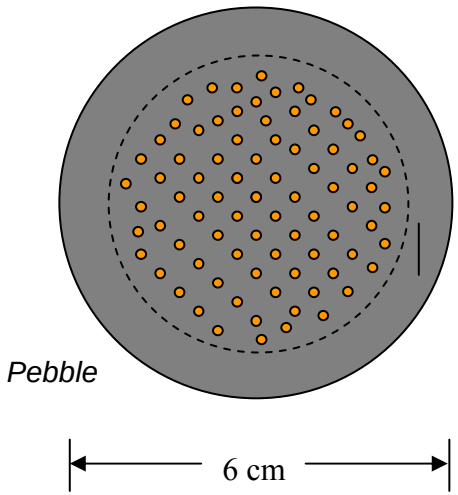


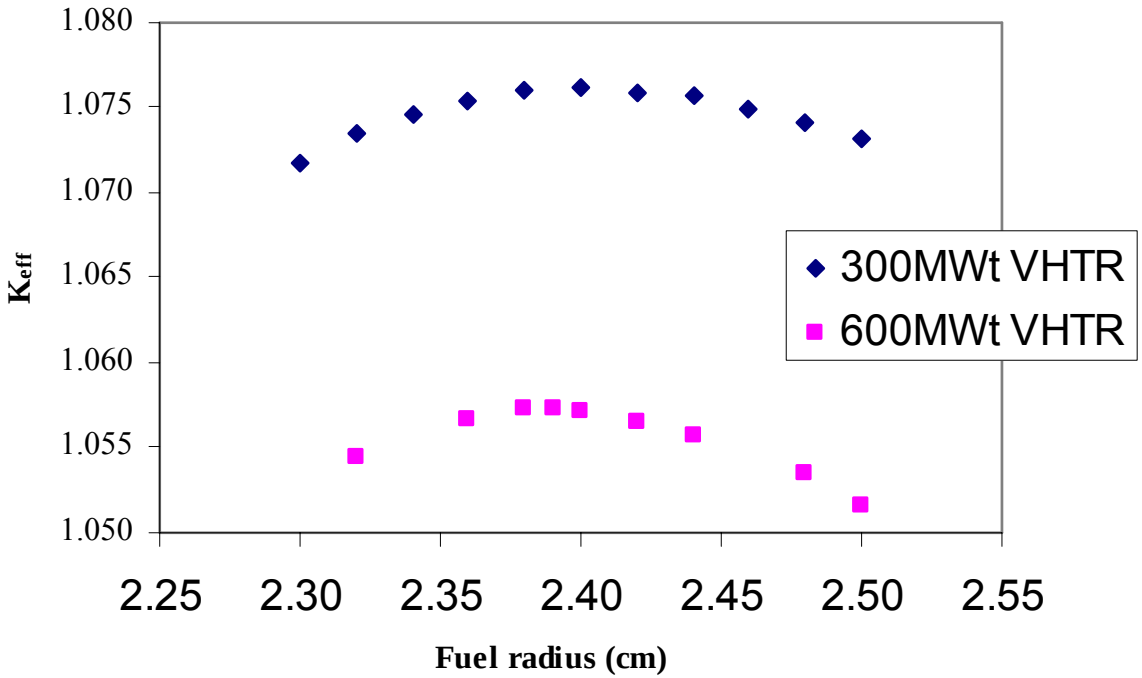
***PBMR (268):**	***O.D. (m)***	***GT-MHR (600):***	***O.D. (m)***
Inner reflector	*~1.75*	*Inner reflector*	*2.96*
Fuel annulus	*3.5*	*Fuel annulus*	*4.83*
RPV	*6*	*RPV*	*7.66*
Height	*8.4*	*Height*	*7.93*

FY03 ACCOMPLISHMENTS

Pebble Bed NGNP - Pebble Optimization

Experimented with pebble dimensions to improve neutronic characteristics





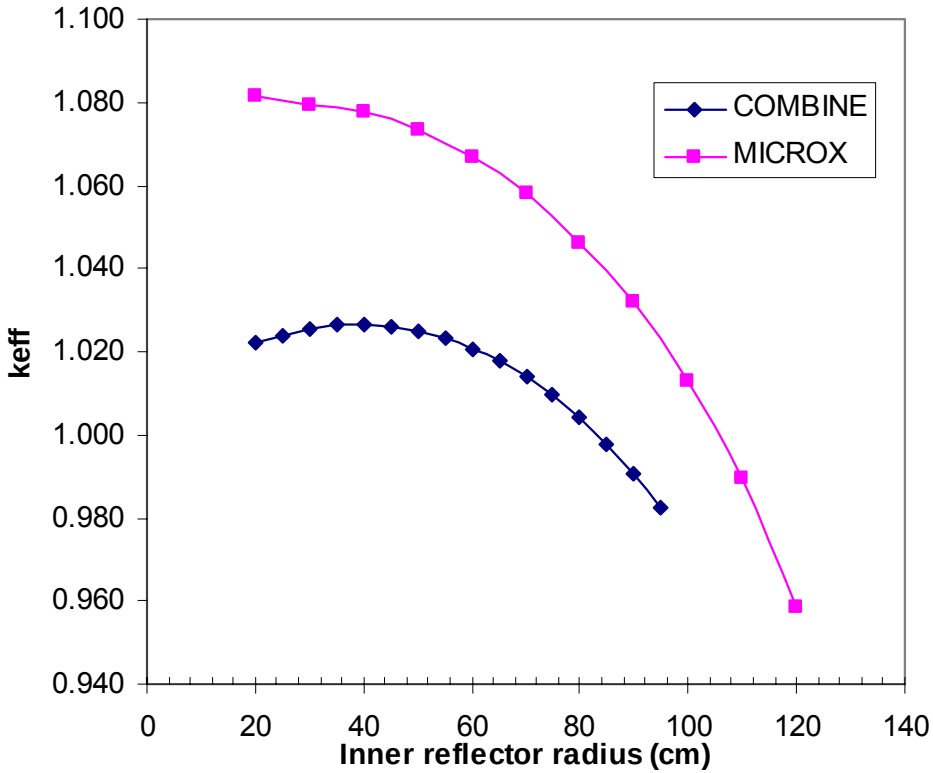
MICROX/PEBBED K_{eff} vs Fuel Zone Radius - 300MWt Core

FY03 ACCOMPLISHMENTS

Pebble Bed NGNP Inner Reflector Study – 300 MWt

Also experimented with the reflector dimensions to improve neutronic characteristics

- This is one of the results that show the need for better treatment of cross sections
- However, both codes show that the performance of the reactor is sensitive to the inner reflector design



FY03 ACCOMPLISHMENTS

Pebble Bed NGNP - Attributes of Best Candidates (1)

	PBMR	VHTR-300	VHTR-600
Power (MW)	268	300	600
Inlet Temp (^oC)	503	600	600
Outlet Temp (^oC)	908	1000	1000
Coolant Flow Rate (kg/s)	125.7	288.4	288.4
Pumping Power (MW)	2.88	5.9	26
Active Core Volume (m^3)	81.8	79.8	119
Core Radius (cm)	175	175	119.00
Inner Reflector Radius (cm)	87ish	40	150
Outer Reflector Thickness (cm)	75	75	75
Active Core Height (m)	840	875	950
Pressure Vessel Diameter (m)	6	6	7.5
Mean Pebble Temperature (^oC)	863	863	863
Peak Pebble Temperature (^oC)	1040	1027	1028
Peak temperature drop across Pebble (^oC)	54	67	76
Peak Pebble Power (W)	1379	1791	1791
Mean Core Power Density(W/cc)	3.28	3.76	5.03
Peak Core Power Density (W/cc)	6.78	7.9	9.31
PEAK DCC Temperature (^oC)	1419	1608	1584
Steam Ingress (0.001g/cc) Reactivity ($)	0.30	0.41	0.09

FY03 ACCOMPLISHMENTS

Pebble Bed NGNP - Attributes of Best Candidates (2)

	PBMR Nom	VHTR-300	VHTR-600
k_{eff}	1.073	1.073	1.073
Discharge Burnup (MWD/kgHM)	80.0	94.3	82.6
enrichment	8%	8%	8%
HM loading (g)	9.00	7.96	7.86
# particles per pebble	15000	13271	13106
Pebble Injection Rate (pebbles/day)	372	400	923
# passes per pebble	10	12	9
Residence Time (days)	879	1082	701
HM Mass Daily Throughput (g/day)	3344	3183	7260
HM Mass Daily Throughput per MWD	12.5	10.6	12.1
particles/MWD (net)	21024	18047	21084

FY03 ACCOMPLISHMENTS

Pebble Bed NGNP Point Design – Summary of Key Results

1. We learned how to design passively safe PBRs that produce more than 600 MWt
2. We found a way to improve both fuel utilization and safety by modifying the pebble design (adjust the fuel zone radius in the pebble to optimize fuel-to-moderator ratio)
3. We learned how to perform design optimization calculations automatically. We can now identify design parameters that optimize selected performance measures by performing hundreds of design calculations that evaluate a sequence of design parameter sets.
4. We learned how to calculate cross sections more accurately for PBRs, and we identified research needs for the further refinement of the cross section calculations

PLANS FOR FY05-07

Pebble Bed NGNP Open Design Issues (1)

- We need to assess designs with lower inlet temperatures
- The core pressure drops are high, especially for the 600 MWt version of the pebble bed NGNP; a cross-flow design would significantly reduce the pressure drop across the core
- Proper evaluation of reaction cross sections in the doubly heterogeneous configuration of the PBR requires better treatment of the Dancoff factor to account for shadowing effects

PLANS FOR FY05-07

Pebble Bed NGNP Open Design Issues (2)

- Find maximum power level for a passively safe pebble-bed NGNP
- Confirm passive safety of the highest power design
- Perform additional hot-spot analyses, i.e. further assess the stochastic nature of the pebble distribution, the possible collection of relatively reactive pebbles in regions of high neutron flux, and uncertainties in heat transfer across the core
- Determine time-dependency of neutron flux and burnup distributions during the run-in period
- Design the control rods for cold reactivity shutdown and reactor scram

PLANS FOR FY05-07

Validation of Reactor Physics and Core Design Analyses Tools

- All of the neutronics codes, even those with a long history of success in light-water reactor applications, need to be validated for use in gas-cooled reactor design and analysis.
- For the pebble-bed reactor, data are available from the critical facilities ASTRA (in Russia), CESAR II (in France), HTR-PROTEUS (in Switzerland), KATHER (in Germany), and GROG (in Russia).
- In addition, data are available from three actual power-producing pebble-bed reactors: AVR and THTR (in Germany) and HTR-10 (in China).
- For prismatic reactors, data are available from the critical facilities for the RMBK reactors in Russia, and from the critical facilities VHTRC (in Japan), GROG (in Russia), and TREAT (in the U.S.).
- Data are also available from the power reactors DRAGON (in the U.K.), Peach Bottom and Fort Saint Vrain (in the U.S.), and the currently operating HTTR (in Japan).

PLANS FOR FY05-07

Validation of Reactor Physics and Core Design Analyses Tools

- **The scope and suitability of the data from all these facilities are currently being evaluated**
- **Many of the facilities will provide data for checking the codes' ability to predict the neutron multiplication factor, k-effective**
- **The power reactors may provide data by which to assess the ability of PEBBED and the MOCUP protocol to predict asymptotic core states for pebble-bed reactors and depletion histories for prismatic core loadings, respectively**
- **The end product of the benchmarking program will be complete documentation of the verification and validation of the codes and USNRC approval for their use in a license application for the NGNP (or NRC/ACRS agreement that the codes are adequate for simulating the plant transient response)**
- **These computer codes will also be used for independent confirmation of the design submitted by the industrial consortium building the NGNP**

PLANS FOR FY05-07

Development and Validation of Reactor Thermal-Hydraulic and Mechanical Design Analyses Tools

- **The requirements imposed on the NGNP codes will be dictated by the design of the reactor and the thermal-hydraulic and mechanical phenomena it will exhibit both during normal operation and postulated accidents.**
- **Several key characteristics of the NGNP design mark a point of departure from previous reactors:**
  - Very high temperature operation
  - Coupling of the reactor to a hydrogen production system as well as an electrical generation system
  - Passively safe under all accident conditions

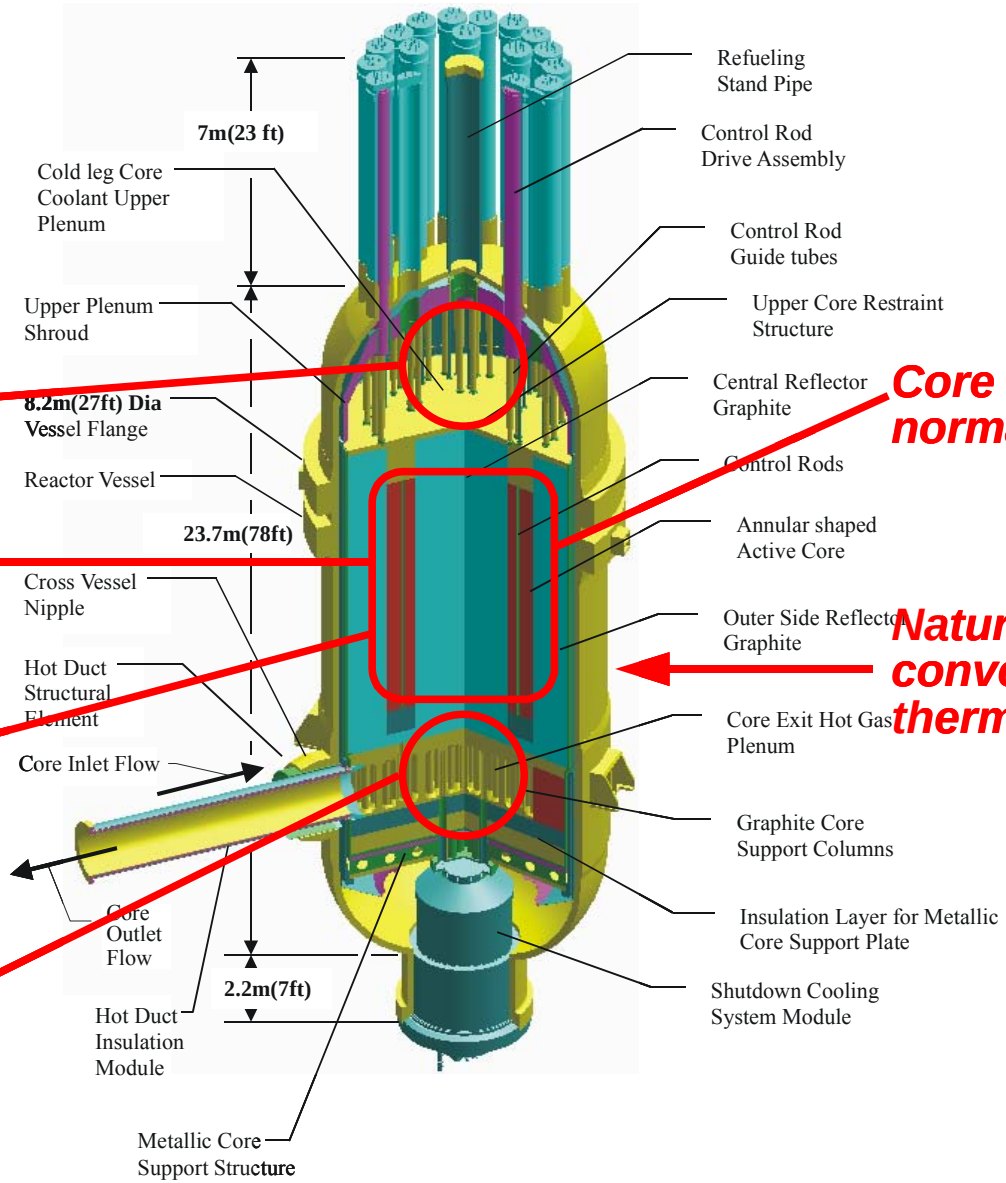
PLANS FOR FY05-07

Development and Validation of Reactor Thermal-Hydraulic and Mechanical Design Analyses Tools

- **The Phenomena Identification and Ranking Table (PIRT) process will be used to identify key system phenomena that the codes must be able to model accurately.**
- **Experimental data will be sought for these phenomena to be used to assess code capabilities.**
- **Shortcomings uncovered through assessment will lead to further code development and testing.**
- **To the extent that required data is not available, experiments will be planned and conducted.**
- **This iterative process will lead to a fully validated set of codes.**

PLANS FOR FY05-07

Reactor System Cutaway



Upper plenum - hot plume mixing - "LOF"

Core - depressurized cooldown

Flow between hotter/ cooler channels - "LOF"

Lower plenum - hot jet mixing

Core flow - normal operation

Natural convection and thermal radiation

PLANS FOR FY05-07

Development and Validation of Reactor Thermal-Hydraulic and Mechanical Design Analyses Tools

- **CFD analysis of the reactor during normal operation is needed to provide assurance that material temperature limits are not exceeded as a result of non-uniformities in coolant flow and power.**
- **System dynamics and interactions between the reactor and the balance of plant necessitate a system-wide analysis capability.**
- **The passive safety system of the NGNP relies on three heat transfer mechanisms (convection, conduction, and thermal radiation) in a coupled fashion.**
- **All this requires a system/CFD coupled code analysis capability.**

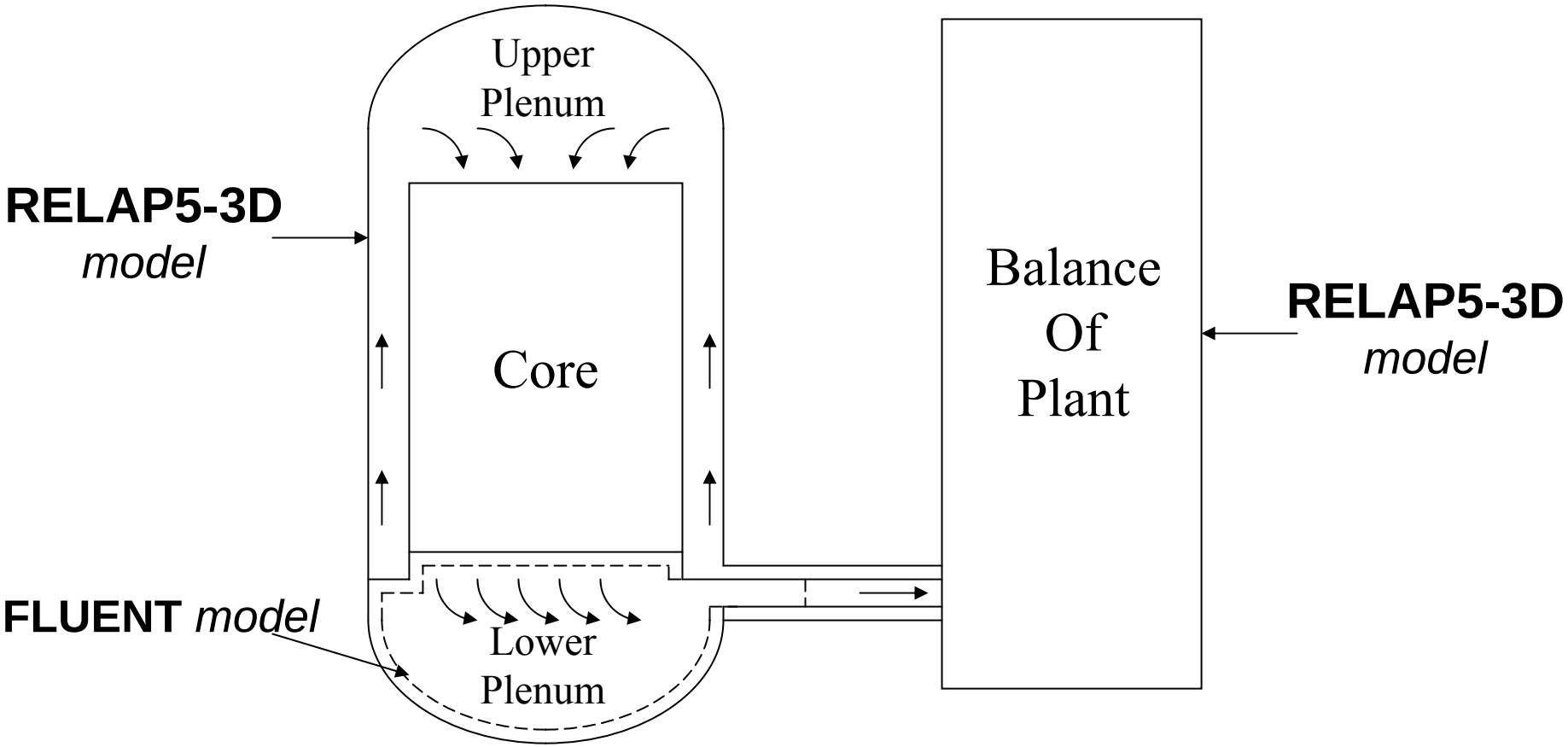
PLANS FOR FY05-07

Development and Validation of Reactor Thermal-Hydraulic and Mechanical Design Analyses Tools

- **The foundation for a thermal-hydraulic systems analysis capability directed specifically toward the NGNP has been under development for three years at the INEEL.**
- **This has resulted in the coupled RELAP5-3D/FLUENT code.**
- **RELAP5-3D provides a system-wide analysis capability and FLUENT provides the CFD capability.**
- **While the basic physical models in RELAP5-3D have been extensively validated for light water reactors, their applicability to the NGNP design must be demonstrated.**
- **A program encompassing validation, experiments, and further code development will accomplish this.**

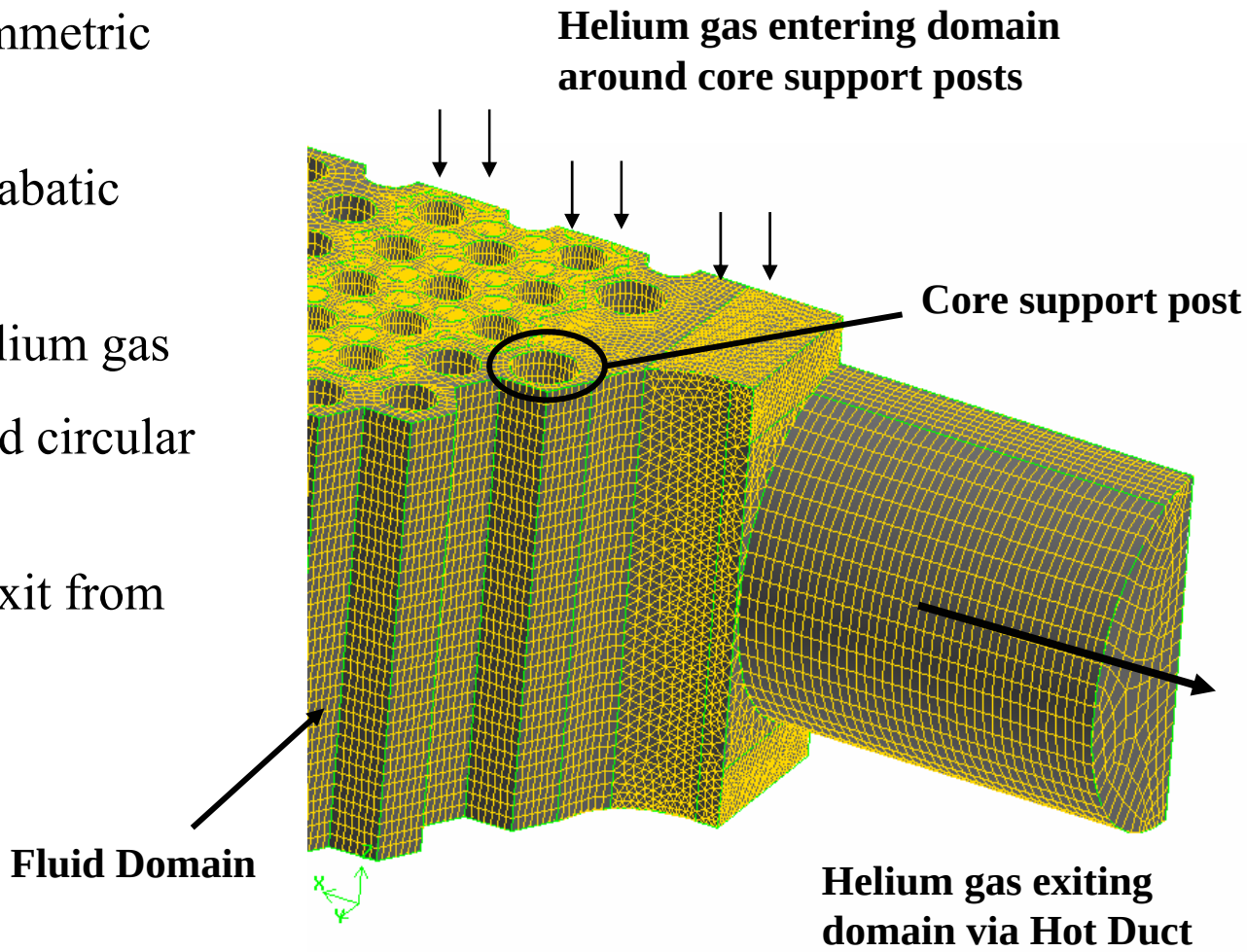
PLANS FOR FY05-07

RELAP5-3D Coupled to FLUENT To Enable Detailed Analysis of Lower Plenum Flow Patterns



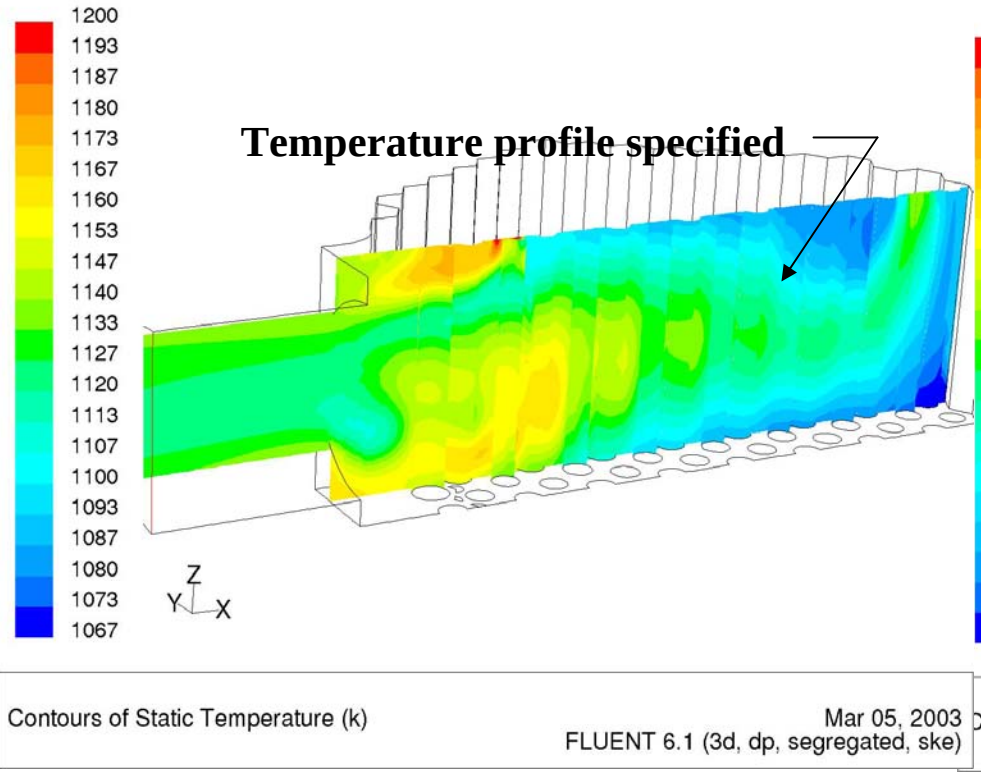
GT-MHR Lower Plenum: Model

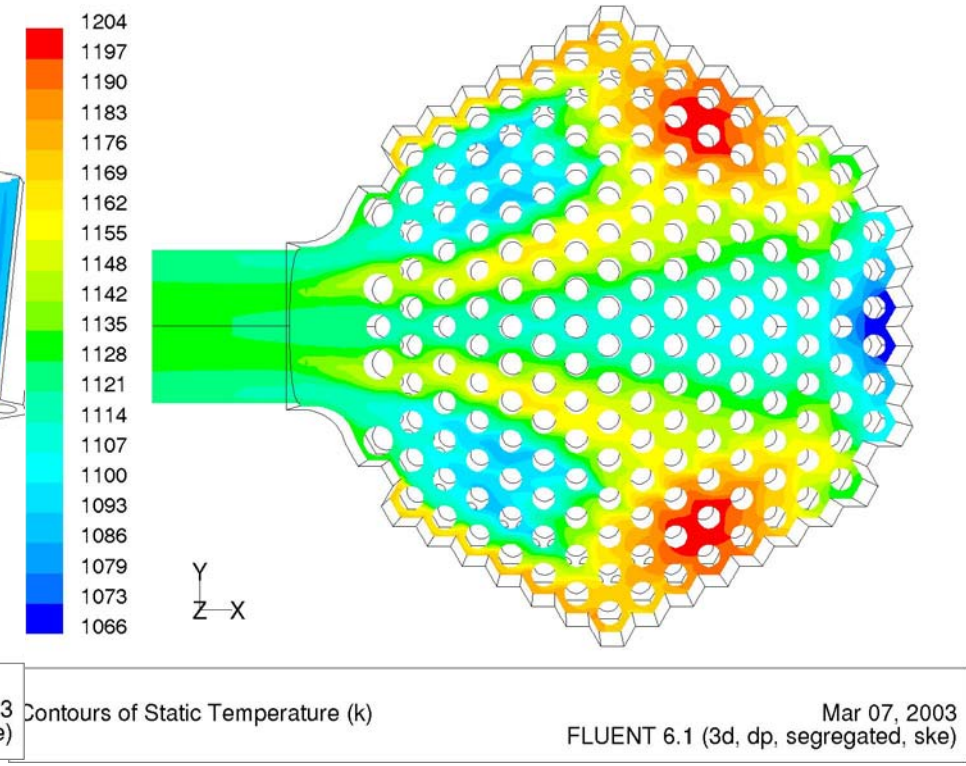
- ~ 750,000 cells, symmetric mesh
- Walls treated as adiabatic surfaces
- Working fluid is helium gas
- Inlets located around circular core support posts
- Hot duct provides exit from domain



PLANS FOR FY05-07

GT-MHR Lower Plenum: Mixing





PLANS FOR FY05-07

Development and Validation of Reactor Thermal-Hydraulic and Mechanical Design Analyses Tools

- **Data that are available for validation of both RELAP5-3D and FLUENT include the**
  - HTTR RCCS Mockup experiments (reactor cavity cooling),
  - SANA-1 experiments (heat removal from a pebble bed),
  - Fort St. Vrain data,
  - Bugey-1 Arret Program.
- **Code-to-code exercises from the IAEA HTGR CRP-3 Program may also be used (IAEA-TECDOC-1163).**
- **Data are also expected to be available from the HTTR reactor (Japan) and the HTR10 reactor (China).**

PLANS FOR FY05-07

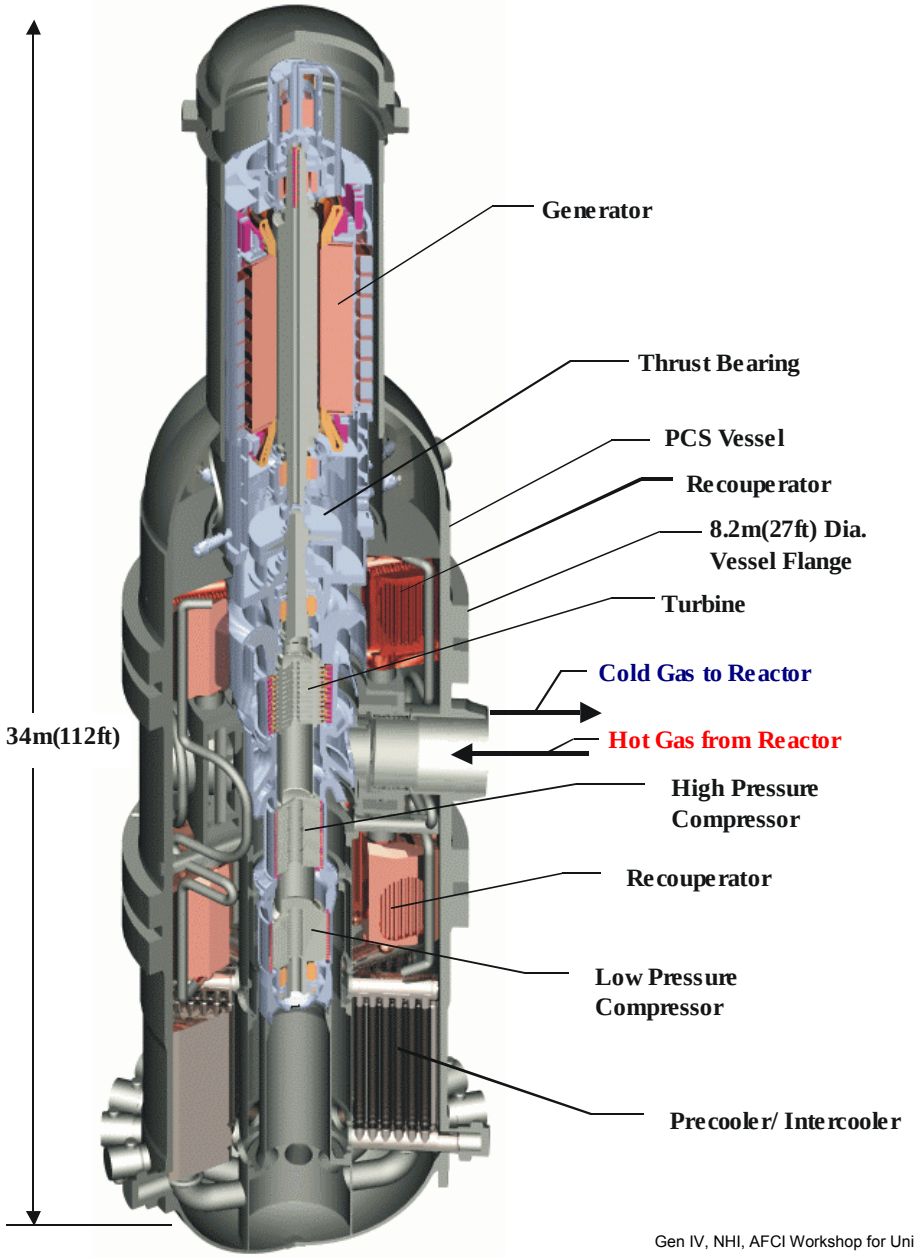
Power-Conversion Unit Assessments.

- There are a wide variety of possible electric power conversion unit (PCU) designs.
- The design can be either a direct or indirect Brayton cycle.
- If the cycle is an indirect cycle, the coolant on the secondary side might be helium or it might be, for example, a nitrogen/helium mixture.
- Also, an indirect cycle might have all gas turbines or some combination of gas and steam turbines.
- A direct cycle PCU might have all the turbines and compressors on a single shaft, or have multiple turbines and compressors on multiple shafts.
- And the shafts might be vertical or horizontal.
- The INEEL needs a capability to properly asses the various electric PCU designs that might be proposed and weigh their relative merits.
- The purpose of this work would be to develop that capability.
- Codes such as HYSYS or ASPEN may be used to study the thermodynamic efficiency of competing designs, but other codes will be needed to address the issues of cost, reliability, and safety.

PLANS FOR FY05-07

Thermal-Hydraulic Issues in the Power Conversion System

- ***Mixing of the gases during bypass events***
- ***Flow distributions among the recuperators and recuperator efficiency***
- ***Hot streaks at the turbine inlet***



PLANS FOR FY05-07

Risk & Regulatory Research

- Performa analysis to select optimal existing regulatory approach (10 CFR 50 or 10 CFR 52).
- Study application of advanced reactor regulatory frameworks to NGNP design.
- Develop methods for applying PRA techniques (identification of initiating events) to the preliminary design process that result in an optimal risk profile.
- Develop basis for probabilistic (reliability) criterion to replace single-failure criterion.
- Develop reactor safety metrics that apply to high temperature gas reactor designs (in lieu of core damage frequency [CDF] or large early releases [LERF]).
- Develop performance requirements and criteria for NGNP filtered confinement performance.
- Integrate source term studies and confinement criteria to develop a basis for the size of the emergency planning zone.

BUDGETS (FY03-04)

Functiona Area	Tasks	Performers	FY-03	FY-04
System Design and Evaluation	Pebble Bed Neutronics Analysis	INEEL	$121 K	
	Prismatic Neutronics Analysis	INEEL	$84 K	
	Pebble Bed / Prismatic Thermal-Hydraulics Analysis	INEEL	$97 K	
		General Atomics	$151 K	
		Penn State	3.2 K	
Fuel	Fuel Performance Double Heterogeneity Effects Study	INEEL		$103
	Fuel Unit Cell/Block/Core Parametics	INEEL		$118
	Fuel Performance Power Mitigation Study	INEEL		$108
	Fuel Management Study	ANL		$200
Total			**$456**	**$529**