NGNP Thermal Fluids Experiments: Design Basis and Approach

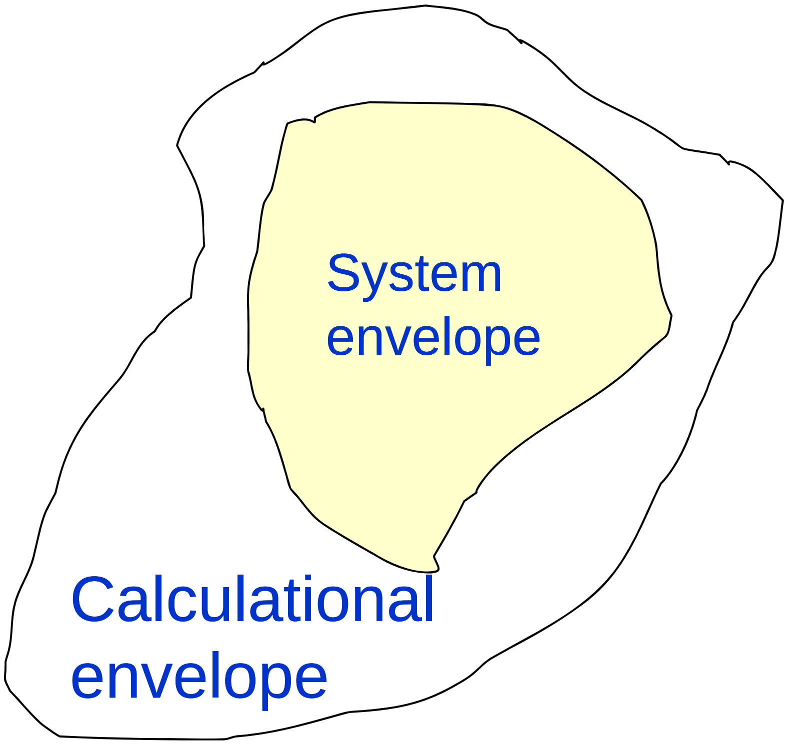
August 11, 2009,
Park City, UT

Contents of Presentation…

- *Experimental program: approach & philosophy*
- *Effort to find available data*
- *Outline of thermal-fluids experimental program*
- *Discussion on integral facility—designing, defining required instrumentation, planning experiments, performing experiments, and using the data to validate software*
- *Summary*

TH Analysis Needs for Advanced Systems…

- Are set by the operational and accident envelopes of the system
- And can only be satisfied if the calculational envelope of the TH software is demonstrated to either match or encompass the system operational and accident envelopes.



TH Software Calculational Envelope…

- *Is defined by the physics in the software*
- *And confirmation that the software physics models properly calculate the key phenomena (V&V).*
- *Successful V&V can only be achieved if an adequate, high-fidelity data matrix and/or exact analytical solution set are available to benchmark the calculational results.*
- *The Evaluation Model contains all the software that must be validated.*

Validation Scope Defined Using Following Approach…

Scenario Identification: Operational and accident scenarios that require analysis are identified

↓

PIRT: Important phenomena are identified for each scenario

↓

Validation: Analysis tools are evaluated to determine whether important phenomena can be calculated

Yes ↓ (to Analysis) — No ↓ (to Development)

Development: If important phenomena cannot be calculated by analysis tools, then further development is undertaken

Yes ↑ (back to Validation)

Analysis: The operational and accident scenarios that require study are analyzed

Each Reactor Scenario must be Evaluated in Context of…

Relevant potential accidents:

- Phenomenology and sequence timing
  - What happens when?
  - Influence of geometry, break size, break location (orientation)
  - Graphite structural material (nuclear or non-nuclear)
- Are there factors that may combine to cause unexpected result, e.g., "cliff-edge" behavior or unanticipated turn of events?

Impacts type of system

Design implications

- Mitigation systems?
- Accident management procedures?

Nature of system: redundancies, diversities, etc.

Credible break size:

- Design basis?
- Beyond design basis?
- Best Estimate or conservative approach (Code of Federal Regulations [CFR])
- Acceptance criteria?

For Thermal-Fluids Analysis…

- *Both systems analysis software and computational fluid dynamics (CFD) software will be used.*
- *Systems analysis software will be used to analyze the overall behavior of reactor or experiment during selected scenarios*
  a. *RELAP5-3D will generally be used*
  b. *MELCOR will be used for fission product tracking.*
- *CFD software will be used to study more detailed flow behavior and to study the location and magnitude of localized hot spots and possibly unacceptably large thermal gradients.*
  - *Commercial CFD will be used most extensively (probably STAR-CCM+)*
  - *Specialized advanced CFD will be used selectively*

Before final plant behavior analyses are performed…

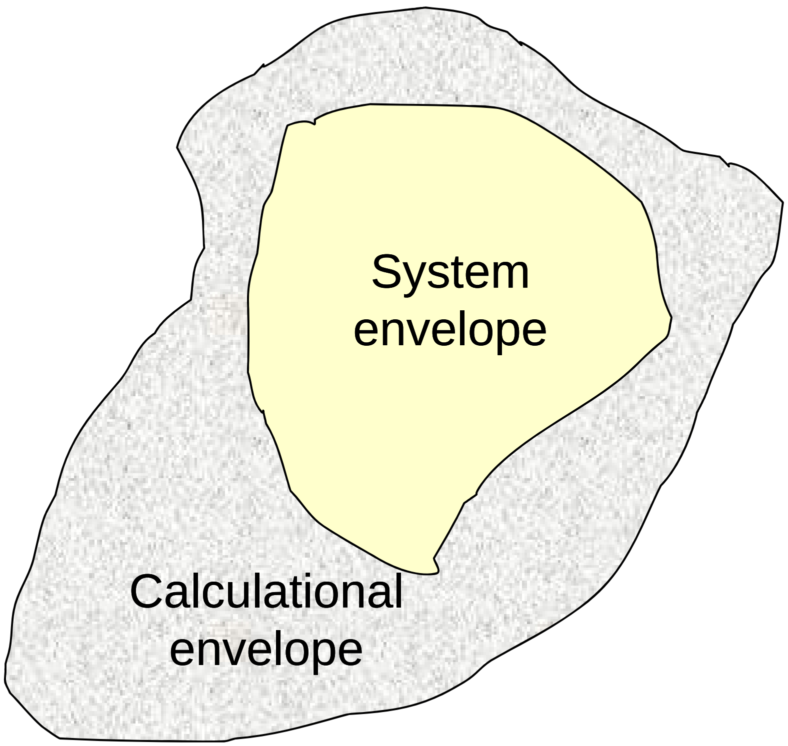
- *Both the systems analysis and CFD software must be adequately validated.*
- *The methodology used to ensure the thermal-analysis software are adequately validated is based on that used for the Westinghouse AP600 and outlined in Regulatory Guideline 1.203.*
- *The approach taken will focus on identifying the data needs, defining the experiments required to provide the data needs, and then performing the validation studies.*
- *The experiments designed and performed will be standard problems that provide adequate data to perform validation calculations for both system analysis and CFD software.*
- *It is intended to make each experiment a standard problem.*

Using Computational Fluid Dynamics (CFD) for Advanced Reactor Analysis

- Complex fluid phenomena and lack of data motivate the need for computational fluid dynamics in safety analysis and licensing
- CFD is still in a developmental stage – fidelity depends upon engineering judgment and knowledge of the application
- For the NRC to accept a CFD analysis, codes need to be validated and qualified, just as materials must be qualified for reactor applications
- NGNP is forming an ASME Committee on Standard for Verification and Validation of System Analysis and Computational Fluid Dynamics Software for Nuclear Applications (V&V30)
- V&V30 is the follow-on to the ASME V&V20 which formulated the *ASME Standard on Verification and Validation in Computational Fluids Dynamics and Heat Transfer*

Standard Problems

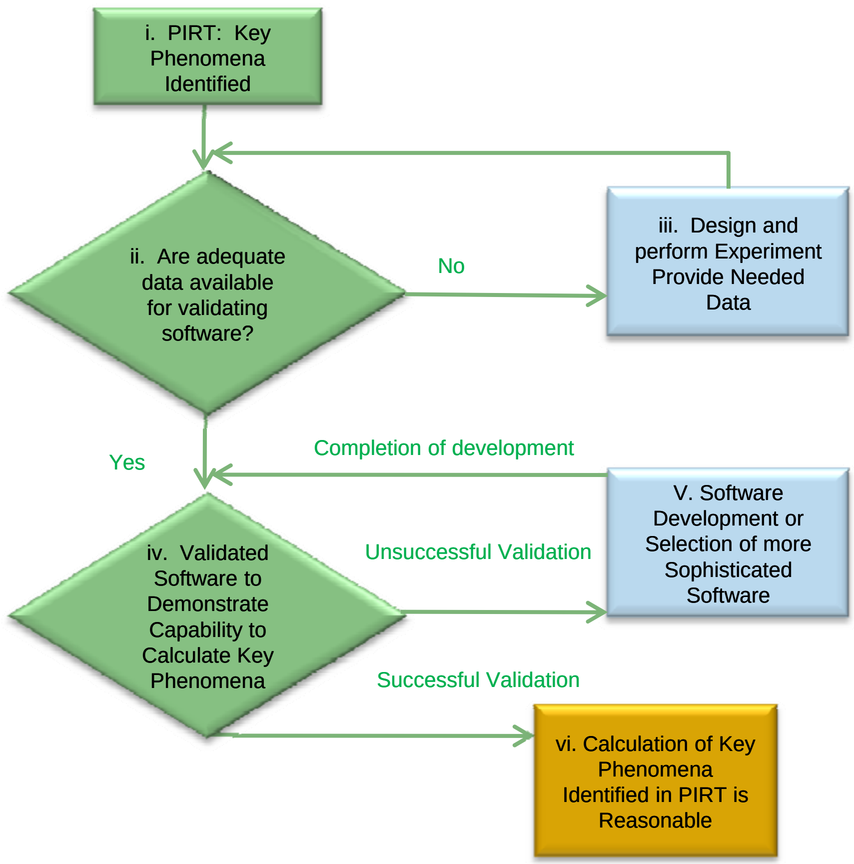
- The Committee will define a new ASME Standard for specifying the applicable software validation matrix, experiment scaling and design procedures, and methodology to ensure completed validations define a software calculational envelope that encompasses anticipated operating conditions
- The validation matrix will be composed of *standard problems* designed to measure the degree to which the software of interest achieves the desired capabilities



Calculational envelope = domain of qualification

Fulfilling Data Needs

- Some data are available in literature
- Some data may be available for R&D partners, e.g., Generation IV International Forum (GIF) for VHTR
- The balance of data needs must be obtained by designing and performing experiments

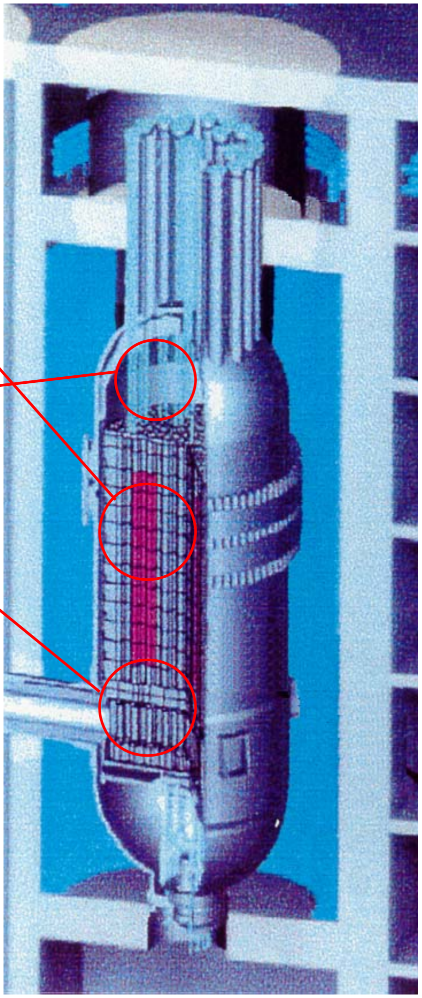


Experiment Design

- All experiments are scaled to prototype plant
- Whenever possible, try to obtain validation data for key phenomena at two different scales—to obtain cross-check. Hence, a separate-effects experiment combined with an integral-effects experiment, both at different scales, enhances confidence level of findings
- Sometimes experiments must be designed at the most fundamental level to quantify and prioritize the physics.
- This leads to lower experimental uncertainties

Key Phenomena: Reactor Physics, CFD and Systems Analysis

- Normal operation at full or partial loads
  - Core power distribution
  - Coolant flow and temperature distributions through reactor core channels (“hot channel”)
  - Mixing of hot jets in the reactor core lower plenum (“hot streaking”)
- LOFA (pressurized cool down)
  - Mixing of hot plumes in the reactor core upper plenum
  - Coolant flow and temperature distributions through reactor core channels (natural circulation)
  - Rejection of heat by natural convection and thermal radiation at the vessel outer surface
  - Hydrodynamic stability: RCCS water cooling system
- LOCA or (depressurized cooldown)
  - Prediction of reactor core depressurized cooldown - conduction and thermal radiation
  - Rejection of heat by natural convection and thermal radiation at the vessel outer surface
  - Water ingress: steam generator tube rupture



Database Survey

- Nuclear reactors
  - Research reactors
  - Commercial reactors

- Experimental facilities
  - Large scale gas loops
  - Small scale gas loops for Separate Effects Tests
  - Component test loops

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Korea Atomic Energy Research Institute

Test-Matrix Structure

- **Accidents (7+)**
  - **Air ingress**
  - **Water ingress**
  - **Loss of flow**
  - **Control rod withdrawal**
  - **ATWS**
  - **Load change**
  - **Hydrogen side upset**

- Phenomena (11+)
  - Convection HT
  - Conduction HT
  - Radiation HT
  - Flow distribution
  - Pressure drop
  - Bypass flow
  - Thermal mixing
  - Molecular diffusion
  - Natural circulation
  - Jet discharge
  - Bulk CO reaction
  - Graphite oxidation

- **Components (14+)**
  - **Top & Lower plenum**
  - **Core & reflector**
  - **Riser**
  - **Internal insulation**
  - **CEDM**
  - **Hot gas duct**
  - **IHX**
  - **Circulator**
  - **PCU (integrated)**
  - **Turbine**
  - **Compressor**
  - **Recuperator**
  - **Valve**
  - **RCCS**

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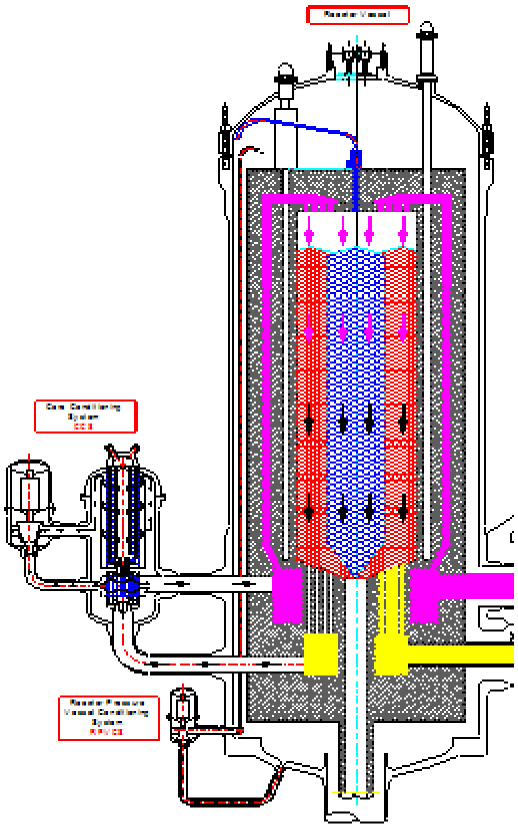
System Components…



System	Component
Reactor Vessel*	Inlet Plenum
	Riser
	Upper Plenum and Components (e.g., Control Rod Assembly Surface Inside Vessel)
	Reflectors (Includes Bypass External to Core)
	Core(Includes Core Bypass Component)
	Fuel (Fuel Integrity)
	Outlet Plenum and Components
Hot Duct	Annular Outlet (Hot) & Inlet (Cold) Pipe
Power Conversion System (Direct Cycle)	Turbine
	Recuperator
	Precooler
	Low & High Pressure Compressor
	Intercooler
	Intermediate Heat Exchanger (IHX)
Reactor Cavity Cooling System	Reactor Cavity
	Downcomers, Piping and Headers
	Air Cooler
	Chimney
	Air Duct
Shutdown Cooling System**	Coolers

Test-Matrix: Nuclear Reactors

			Accident							Phenomena												Components														
			Air Ingress (LPCC)	Water Ingress	Loss of Flow (HPCC)	Control Rod Withdrawal	ATWS (Core Heat-Up)	Load Change	Hydrogen Side Upset	Conv.Heat Transfer	Cond. Heat Transfer	Radiation Heat Transfer	Flow Distribution	Pressure Drop	Bypass Flow	Thermal Mixing	Molecular Diffusion	Natural Circulation	Jet Discharge	Bulk CO Reaction	Graphite Oxidation	Top & Lower Plenum	Core & Reflector	Riser	Internal Insulator	CEDM	Hot Gas Duct	IHX	Circulator	PCU (Integrated)	Turbine	Compressor	Recuperator	Valve	RCCS	
Facility	**Nation**	**Type***	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**	**13**	**14**	**Facility**
ML-1	USA	-												O									O				O	O			O	O				ML-1
Dragon	USA	PMR					O																O			O		O	O							Dragon
Peach B	USA	-						O															O					O								Peach B
UHTREX	USA	PMR																					O													UHTREX
FSV	USA	PMR	O	O	O		O						O		O								O			O		O	O							FSV
HHV	Germany	PBMR																													O					HHV
AVR	Germany	PBMR		O	O		O				O											O						O								AVR
THTR300	Germany	PBMR											O													O	O									THTR300
OGL-1	Japan	PMR																					O													OGL-1
HTTR	Japan	PMR	O		O						O	O									O	O	O		O	O		O	O						O	HTTR
HTR-10	China	PBMR			O	O	O							O		O						O	O		O	O			O							HTR-10

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Nuclear Reactors

Name	Nation	Design Specification				Fluid	Type	Objective	Status	
		Pressure	Temp.	Flow	Power					
		MPa	℃	kg/sec	MW					
ML-1	USA	**2**	**650**		**3.3**	**N2**	**RR**	**First gas turbine**	**1961-1965**	U4
Dragon		2	750		20	He	RR	Block core demo.	1964-1976	1
Peach B		2.3	728		115	He	RR	Fuel rod core demo.	1966-1976	1, U4
UHTREX		4	1300	1.29	3	He	RR	VHTR demo., Ceramic fue	1966-1970	1
FSV		4.9	785		842	He	CR	Comercial Reactor	1977-1989	1, U4
HHV	Germany	5	850	200	90	He	RR	High-T Turbine	Stop 1981	G7
AVR		2.3	950	4	49	He	RR	Pebble core demo.	1966-1988	G7,G10
THTR-300		3.9	750		750	He	CR	Comercial Reactor	1985-1989	1, U4
OGL-1	Japan	3.5	1000	0.1	0.1	He	RR	Fuel Irradiation	1977-1995	J4
HTTR		4	850		30	He	RR	Advanced PMR demo.	Operation	J2
HTR-10	China	3	700	4.32	10	He	RR	Advanced PBMR demo.	Operation	C1-C7

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Test-Matrix : SET or Component Tests Facilities

			Accident							Phenomena												Components														
			Air Ingress (LPCC)	Water Ingress	Loss of Flow (HPCC)	Control Rod Withdrawal	ATWS (Core Heat-Up)	Load Change	Hydrogen Side Upset	Conv.Heat Transfer	Cond. Heat Transfer	Radiation Heat Transfer	Flow Distribution	Pressure Drop	Bypass Flow	Thermal Mixing	Molecular Diffusion	Natural Circulation	Jet Discharge	Bulk CO Reaction	Graphite Oxidation	Top & Lower Plenum	Core & Reflector	Riser	Internal Insulator	CEDM	Hot Gas Duct	IHX	Circulator	PCU (Integrated)	Turbine	Compressor	Recuperator	Valve	RCCS	
Facility	Nation	Type*	1	2	3	4	5	6	7	1	2	3	4	5	6	7	8	9	10	11	12	1	2	3	4	5	6	7	8	9	10	11	12	13	14	Facility
CFTL	USA	PMR																								O										CFTL
PLUM		-																					O													PLUM
SANA	Germany	PBMR	O		O		O				O	O											O													SANA
NACOK		PBMR	O		O								O	O				O				O													O	NACOK
PNP		PBMR											O	O		O		O				O					O									PNP
AVA		PBMR														O						O					O									AVA
EVO		PBMR																													O					EVO
ADI		PBMR																																		ADI
THERA		PBMR	O																		O		O													THERA
MASEX	Russia	Both								O																										MASEX
ASTRA		Both																								O										ASTRA
HETAP	France	Both																					O													HETAP
HETHIMO		Both																					O		O											HETHIMO
INSULATOR	Japan	PMR									O														O		O									INSULATOR
HOT DUCT		PMR									O																O									HOT DUCT
INCORE		PMR								O			O			O		O				O	O	O												INCORE
PBMM	S. Africa	PBMR																												O						PBMM
SNU-1	Korea	Both								O		O																							O	SNU-1
SNU-2		PBMR									O												O													SNU-2
KAIST-1		PMR	O																				O													KAIST-1
KAIST-2		Both								O				O														O								KAIST-2

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SET or Component Test Facilities

Name	Nation	Design Specification				Fluid	Objective	Status	Ref./ Source
		Pressure	Temp.	Flow	Power				
		MPa	℃	kg/sec	MW				
CFTL	USA	10.7	600	0.47m3/s	5	He	Component test	1967	U1
PLUM		227psia	1000F	0.24lb/s	0.2	He	Fuel test	1967	U2
SANA 1	Germany	-	1200	-	0.05	He	Fundamental test		G1, G2
NACOK			1,200	0.003	0.174	Air	Air ingress, Natural circulation	Operation	G3, G4
PNP		0.1-0.6	90	9	0.42	N2	1/5.6 model		G5
AVA		0.13	40	20	2	He	1/2.9 flexiglass model		G6
EVO		2.7	750		160	He	PCU Direct cycle plant	Stop 1981	G7
EVA-II		4	950	3.8	10	He	Steam reformer bundle		G11,G13
ADI		7	950	4.5	0.2	He	PNP Rx. Thermal insulator		G9,G9-1
THERA			1,000			O2,CO2	Graphite oxidation		
MASEX	Russia			300 m3/hr		He-air	Heat and mass transfer	Start 1991	R2
ASTRA		0.25	700		0.07	N2	Fuel handling, CEDM test	Operation	R2
HETAP	France	10	850			He	Metallic heater test	Operation	F1
HETHIMO		10	1,000	0	0.04	He	Insulator, Piping	Operation	F1
INSULATOR	Japan	5	400	0.5	100	He	Insulator, Incore Struc.	Start 1975	J3
HOT DUCT		4	1000	0.007		He	Thermo-Structure	Start 1978	J3
INCORE		4	400	0.25	0.1	He	Core Bottom & Graphite		J3
PFTL	S. Africa					ball	Pebble flow		SA1
PBMM						He	PCU test		SA2
SNU-1	Korea	atm	300	40 m/s	0.005	Air	RCCS, Radiation	Dismentled	K1, K2
SNU-2		atm	50	65 m/s		Air	CFD, contact conductance	Operation	K3
KAIST-1		atm	1500	0.15 m/s		CO,O2	Air ingress, Graphite Oxidation	Operation	K4
KAIST-2		atm	400	0.035	0.03	Air	Heat exchanger	Operation	K5

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Experiment Design Must Be Quite Rigorous…

- Designed to capture key phenomena.
- Scaled to provide a direct link between subscaled experimental facility and prototypical plant.
- Low, quantified uncertainties.
- Experiment design should consider decomposition of behavior in system component to lowest level that can be modeled by software to ensure each component is properly being calculated by software physics.

Thermal-Hydraulic Phenomena: Experiment Planning

- **Normal operation at full or partial loads**
  - Coolant flow and temperature distributions through reactor core channels ("hot channel")
  - Mixing of hot jets in the reactor core lower plenum ("hot streaking")
- **Loss of Flow Accident (LOFA or "pressurized cooldown")**
  - Mixing of hot plumes in the reactor core upper plenum
  - Coolant flow and temperature distributions through reactor core channels (natural circulation)
  - Rejection of heat by natural convection and thermal radiation at the vessel outer surface
- **Loss of Coolant Accident (LOCA or "depressurized cooldown")**
  - Prediction of reactor core depressurized cooldown - conduction and thermal radiation
  - Rejection of heat by natural convection and thermal radiation at the vessel outer surface

Integral Facility/ RCCS

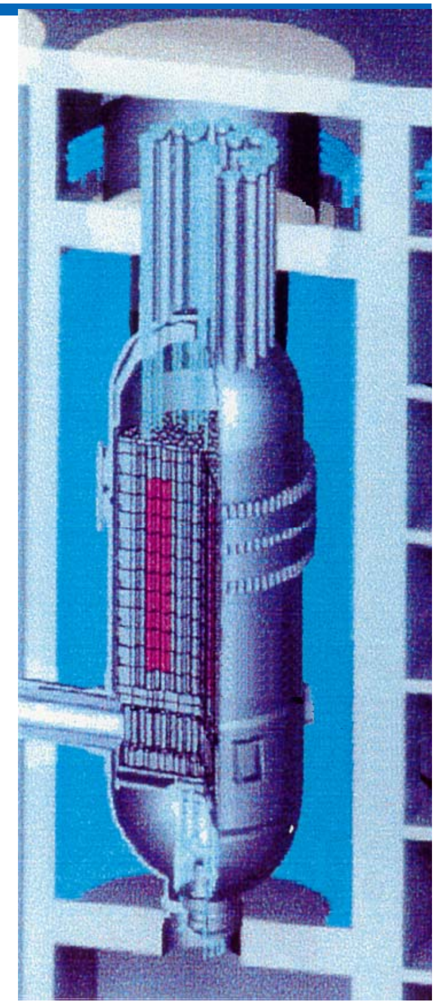
Lower Plenum Exp

Plenum-to Plenum Exp

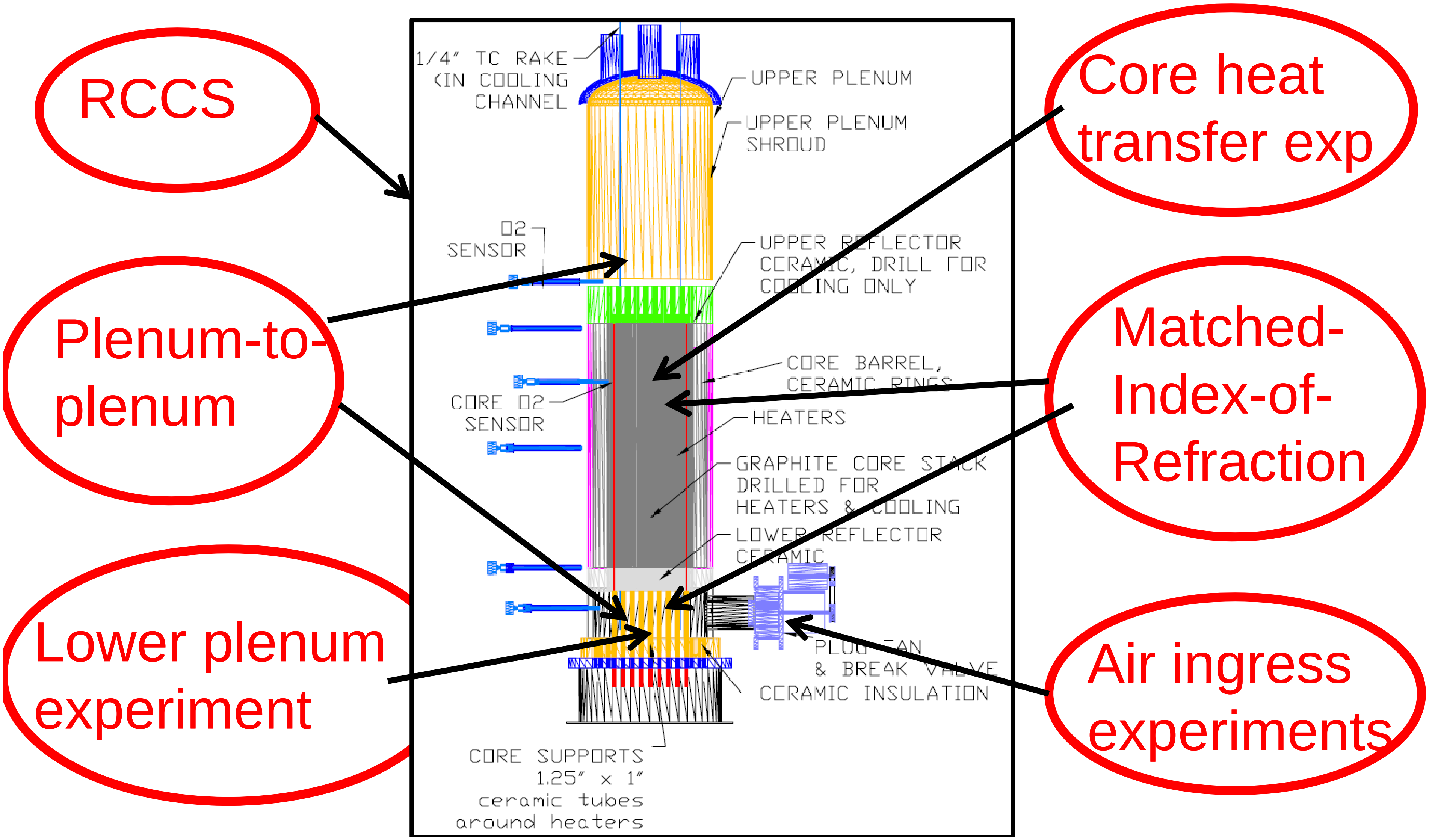
Core Exp

MIR Exp

Air Ingress Exp



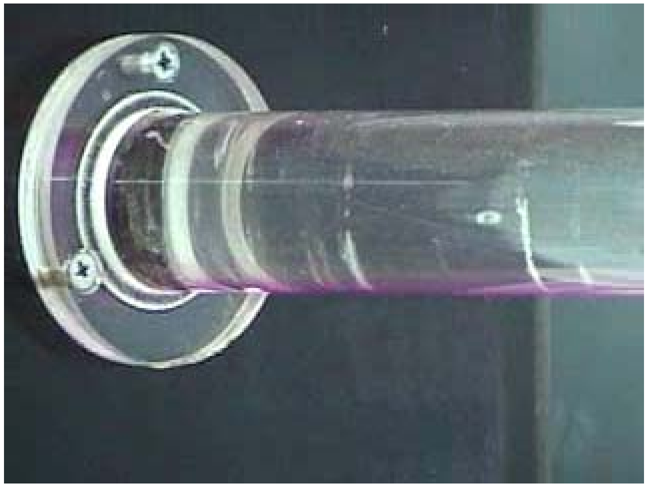
Relationship of Separate-Effects Experiments to Integral Experiment…



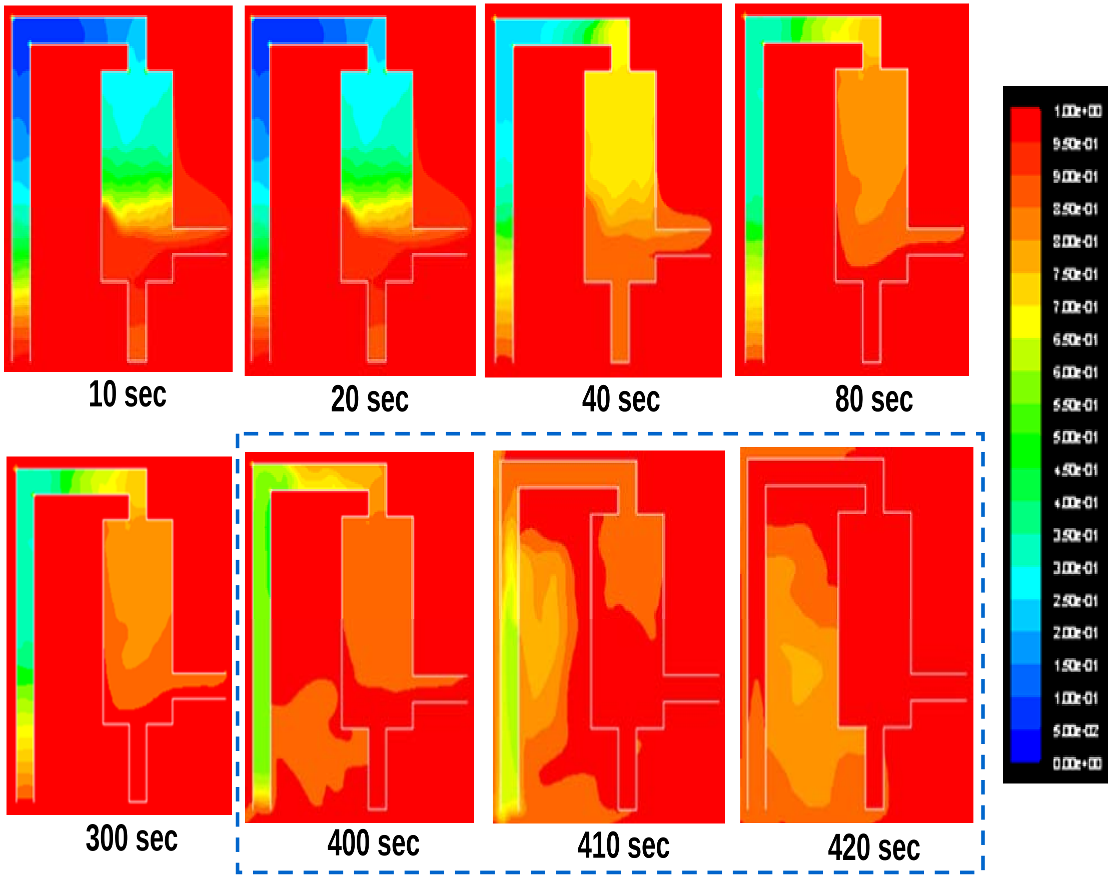
Air Ingress

- Problem: oxygen flowing naturally into the core after a pipe break initiates rapid graphite oxidation → core degradation and inventory release
- Figure of Merit: time to onset of natural circulation (how soon after the break does natural circulation of air begin?)
  - restricts response, drives design changes
  - molecular diffusion (days) or density-driven stratified flow (minutes)
- Design and Licensing
  - Plausibility - Design basis or beyond?
  - Large and small breaks; double-ended guillotine?
- Modeling: what is the right answer?
  - System codes (e.g. RELAP) don't capture the flow physics
  - CFD codes predict it but have not been validated
- Validation
  - ISU Stratified Flow Experiments, OSU HTTF

Density Gradient Driven Flow Experiments and Simulations



Calculated Mass Fraction for Density Driven Air-ingress Experiment



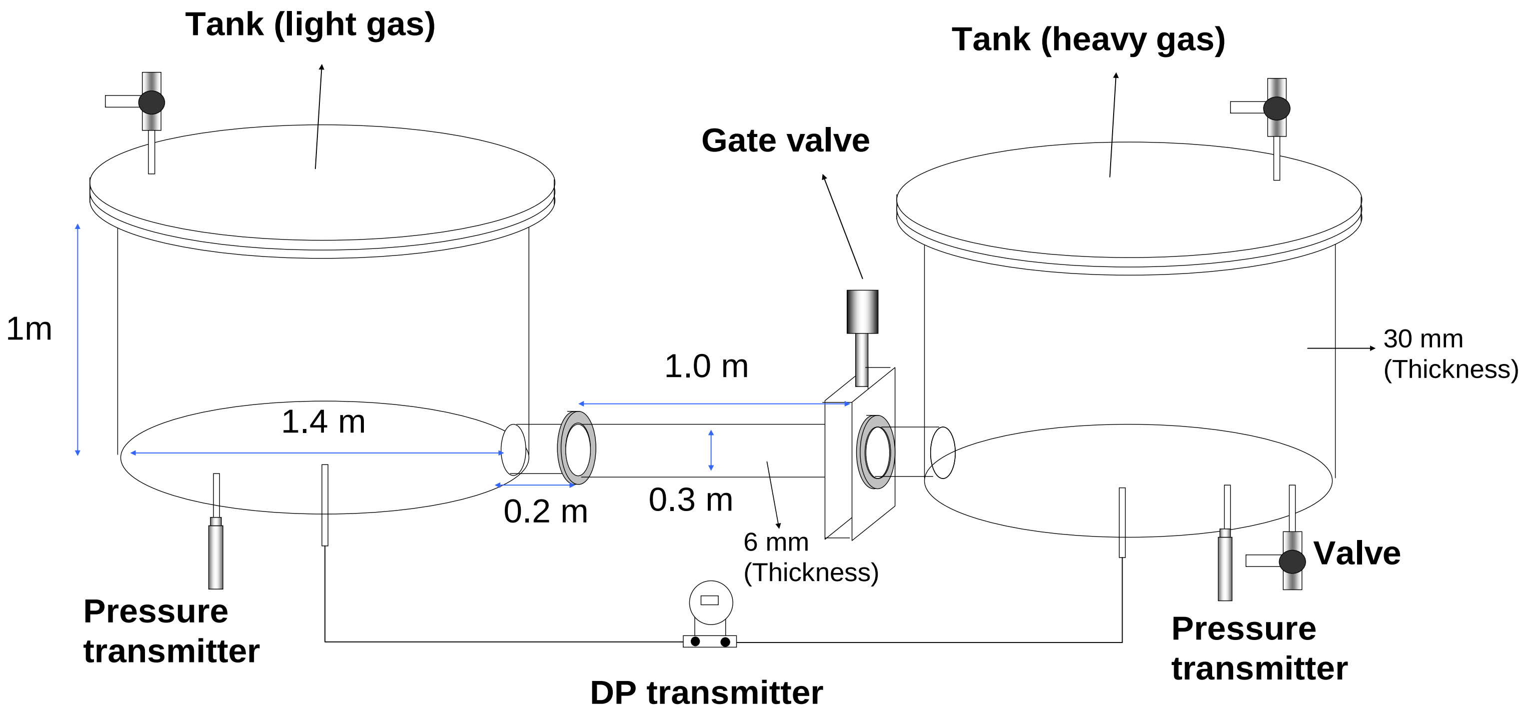
Onset of natural convection occurred at around 400 sec.

CFD simulations at INL, PBMR, KAIST– qualitative agreement on phenomena

Planned Air Ingress Experiments…

Isothermal Experiment in the Horizontal Circular Pipe

- Focused on the separate effect of stratified flow phenomena
- A simple scaling method used for pipe sizing and test conditions

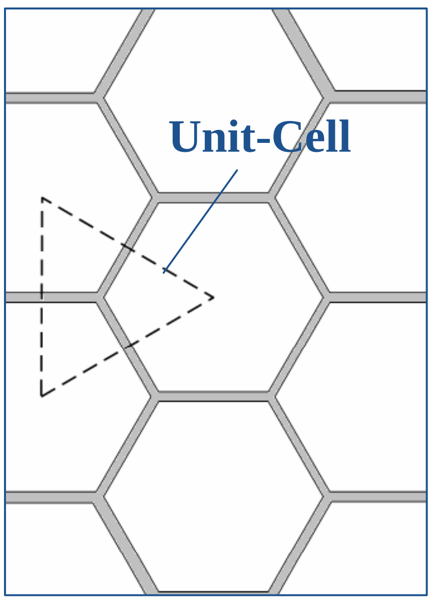


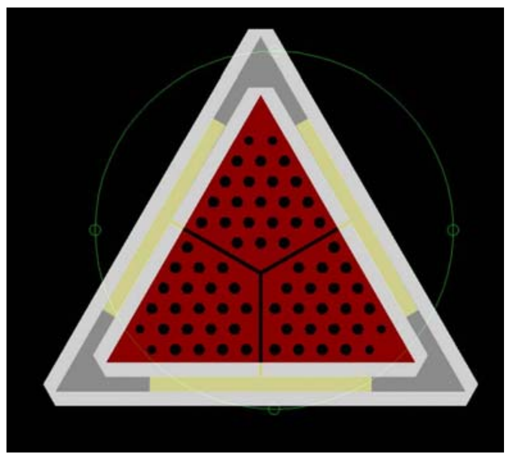
Bypass Flow

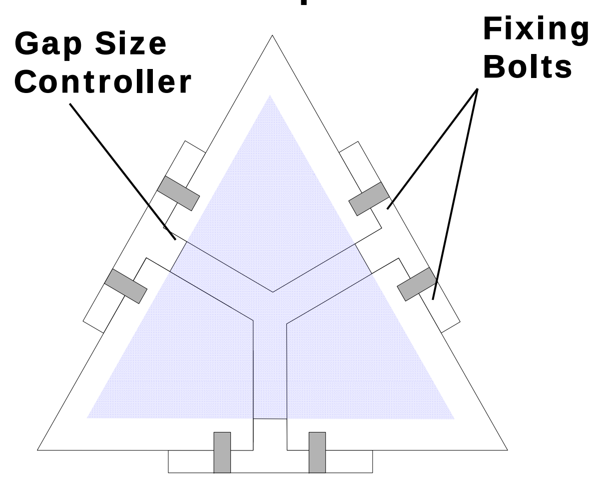
- Problem: Unknown fraction (~10-25% ?) of the primary coolant flows through reflector rather than through the core. Fuel temperatures are higher and more uncertain than what is computed.
- Figure of Merit: Peak and average fuel temperatures
  - Fuel and graphite are being tested and qualified based upon anticipated temperatures – these may not envelope actual conditions
- Design and Licensing
  - Operating power and temperature may need to be reduced to provide enough margin to cover the uncertainty
  - Current fuel/graphite testing conditions may be inadequate
- Modeling: what is the right answer?
  - Bypass flow is a function of graphite block structure which changes with temperature and irradiation
  - CFD codes are needed to model complex flows
  - Prismatic fuel operates at higher temperatures – less margin
  - Statistical uncertainty in pebble bed local burnup and packing fractions contribute to greater uncertainty
  - Graphite dimensional changes are a complex function of temperature and fluence – a multiphysics problem (structural, neutronic, CFD)
- Validation
  - KAERI, matched-index-of-refraction

Bypass Flow

- MIR experiments planned at INL
- Small scale experiments can be performed at universities
- KAERI is conducting single and multi-block experiments







Courtesy: Korea Atomic Energy Research Institute

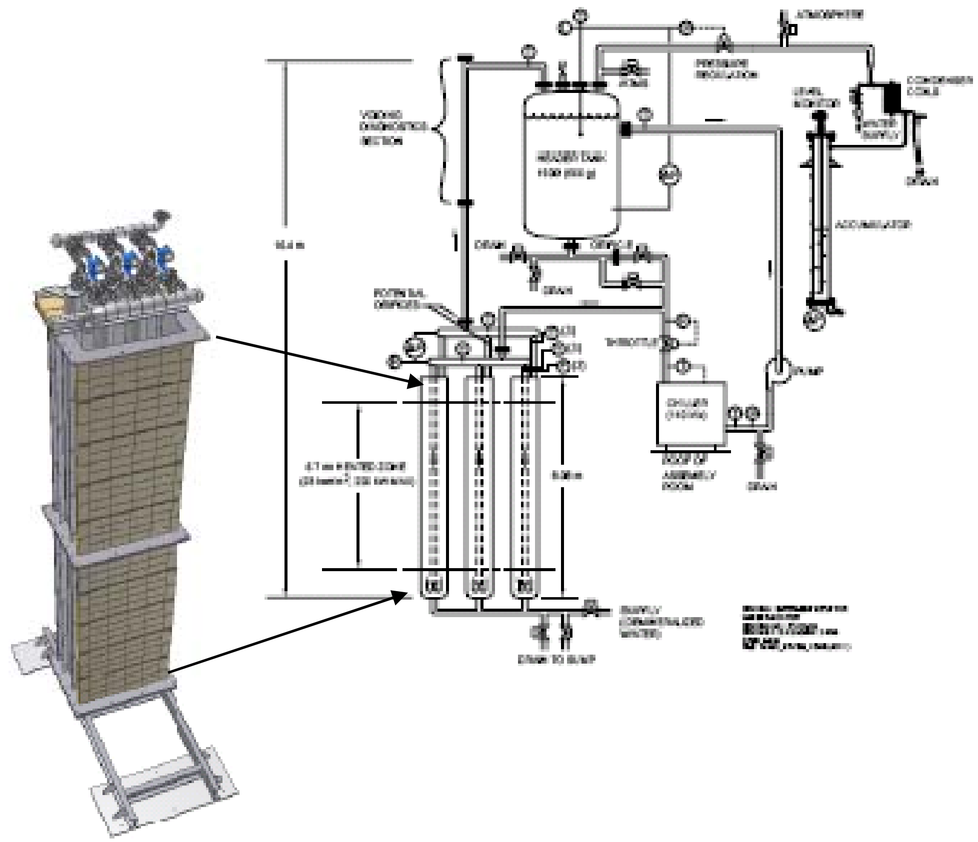
- **Unit-Cell**
  - **By connecting the center of three hexagonal blocks**
  - **Three cases of the block combination**
    - **. 1 fuel block + 2 reflector blocks (F1)**
    - **. 2 fuel blocks + 1 reflector block (F2)**
    - **. 3 fuel blocks (F3)**

RCCS

- Problem: Designs rely on the Reactor Cavity Cooling System to reject decay heat in the event that primary and secondary cooling systems are unavailable. RCCS systems have multiple cooling channels driven by natural circulation in mixed (laminar and turbulent) flow regimes.
- Figure of Merit: total heat flux into the RCCS
  - Depends upon surface properties, natural circulation heat transfer, geometry of flow channels
  - Not much data available for these flow regimes.
- Design and Licensing
  - Air-cooled vs. Water-cooled (depends upon containment)
- Modeling: what is the right answer?
  - System codes may not predict counter-flow phenomena (multiple channels with common plenum)
  - Asymmetric heat source (3D phenomenon)
  - Coupled system/CFD modeling (the system code provides the boundary conditions for the CFD analysis
- Validation
  - NSTF (ANL) - originally built to support Fast Reactor (PRISM) program
  - Other: Korea National University water-cooled RCCS experiment or HTTR

RCCS Experiments Planned

- Empty cavity, single tube, and multiple tube
- Constant wall temperature and constant heat flux
- Steady state and transient
- Water- and Air-cooled tests



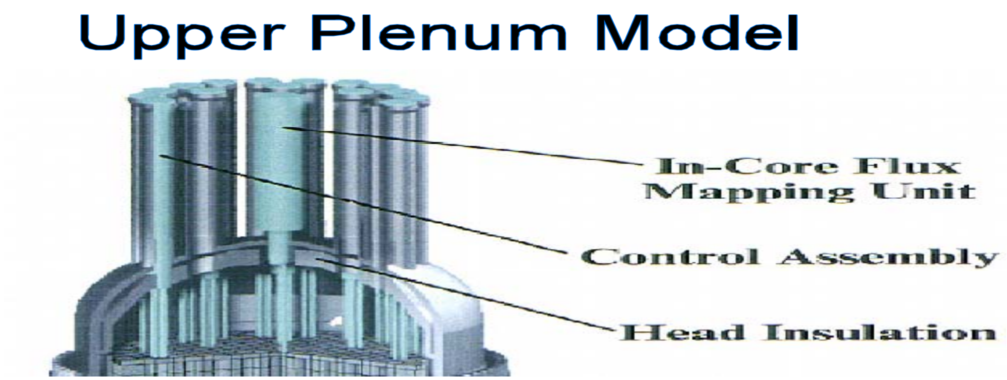
Natural Convection Shutdown Heat Removal Test Facility (NSTF)

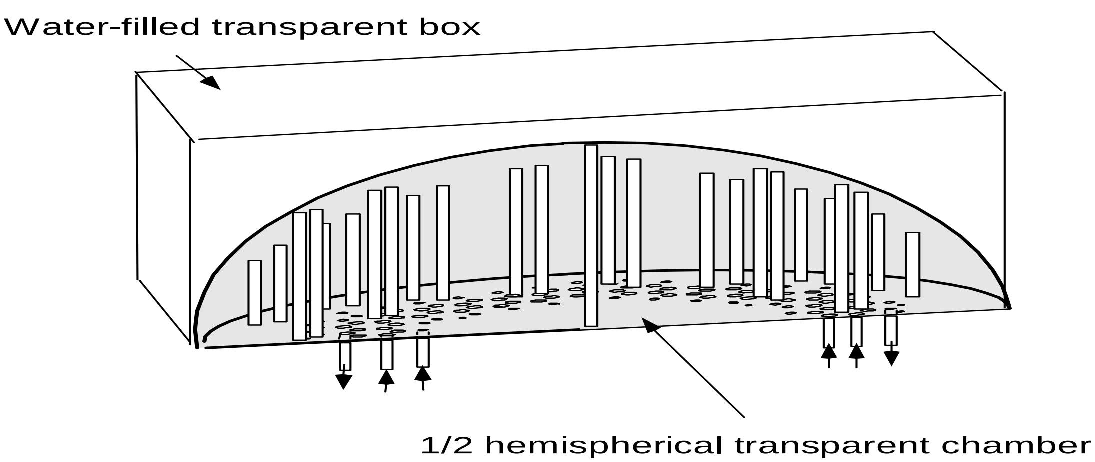
Plenum-to-Plenum…

- Problem: During loss-of-power the reactor system reverts from forced flow to natural circulation flow. The hottest regions of the core will cause hot buoyant plumes of gas (helium during LOFA and helium/air during LOCA) to move upward and impinge on ceiling of reactor where control rod drives are located
- Figure of Merit: Material temperatures in upper plenum
  - Depends upon power distribution—so is time dependent
  - Natural circulation in prismatic reactor may be considerably different from pebble-bed reactor
- Design and Licensing
  - Must demonstrate capability to predict material temperatures in upper plenum
- Modeling: what is the right answer?
  - System codes may not predict plume behavior correctly—particularly in pebble-bed reactor
  - CFD mesh for this type of analysis may be very large. Validations have not been done for this scenario previously
- Validation
  - Plenum-to-plenum experiments
  - HTTF

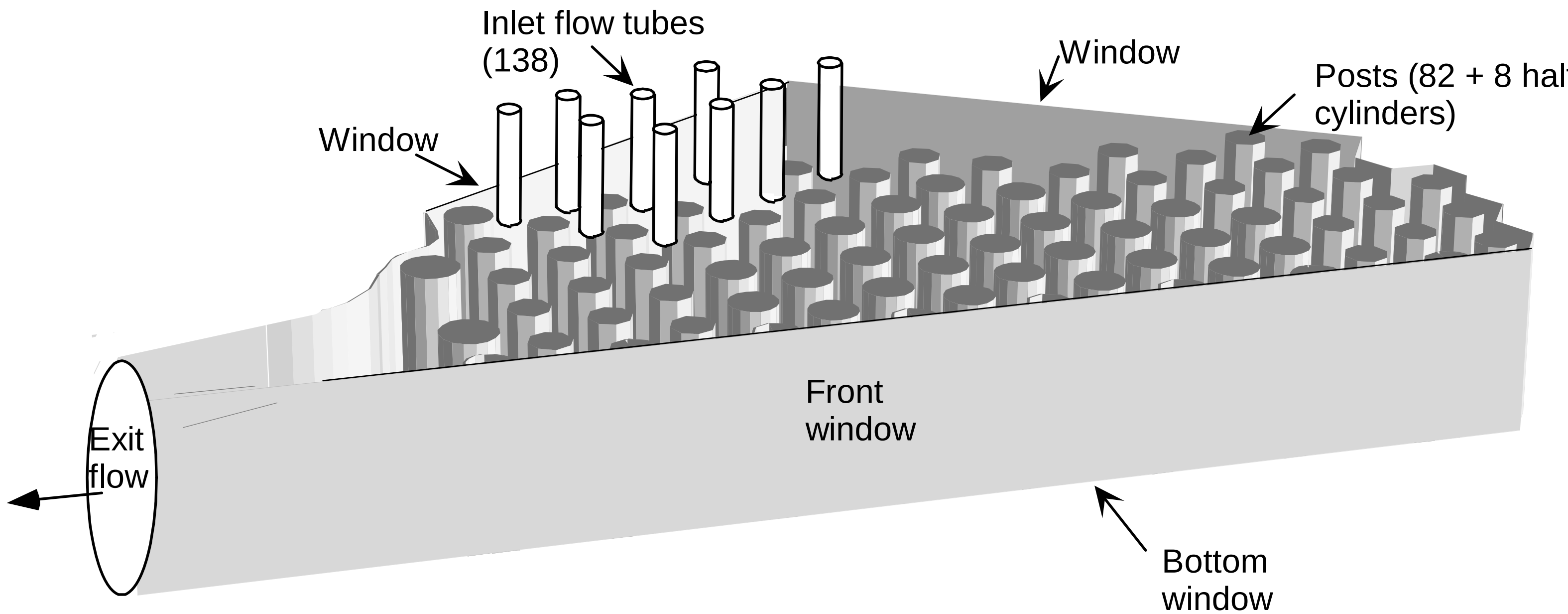
Plenum-to-plenum experiment…

- Designed to study natural circulation flow distribution through core as function of decay power.
- Study effect of hot plume impingement on ceiling of reactor vessel
- Experiment designed to have upper and lower plenum components



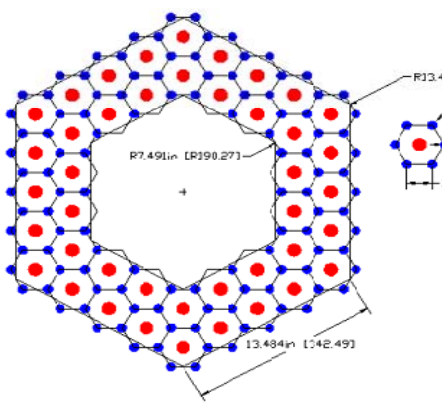


Plenum-to-plenum experiment: Lower plenum component…

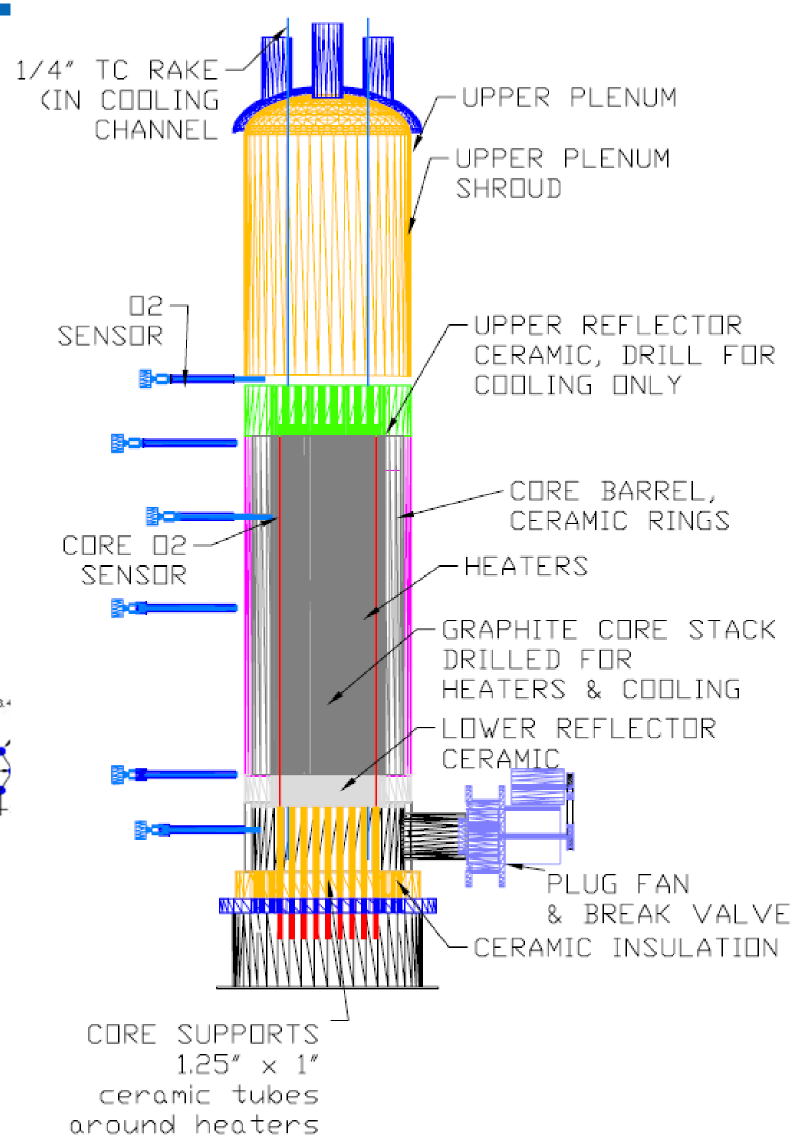


High Temperature Test Facility (HTTF): Integral Experiments

- 3 requirements imposed.
  1. Fluid property similitude.
  2. Geometric similarity.
  3. Friction and form loss similarity.
- Length scale selected as 1:4.
  - Natural circulation processes. Time scale→1:2. Therefore all transport processes scaled to 1:2.
  - Air ingress by molecu diffusion. Time scale→1:16.
- Diameter scaling ratio 1:7.54.



Heater rod configuration: plan view



HTTF Project—NGNP Activities…

- Review of HTTF scaling basis and approach.
- RELAP5-3D models and calculations: Study of facility behavior and its relevance to behavior in reference reactor: performed using RELAP5-3D models.
- CFD models and calculations: Study of facility behavior and its relation to behavior in reference reactor for the upper and lower plena and core.

HTTF Project—NGNP Activities (2)…

- Experiment design:
  1. Initial and boundary conditions: decay heat, thermal conductances and capacitances, initial flow and temperature distributions, etc
  2. Sizing of breaks, relief valves, etc
  3. Location of breaks and break orientation
  4. Study of potential atypicalities: environmental heat losses, experiment progression, etc
- Location and scope of experiment instrumentation
- Pre- and Post-Experiment analyses and software validation

Summary…

- To meet the thermal-hydraulic analysis needs for advanced reactors requires the implementation of a rigorous process that is designed to:

  "Ensure the calculational envelope of the TH software is demonstrated to either match or encompass the system operational and accident envelopes."

- *The calculational envelope of the TH software:*
  - *Is defined by the physics in the software*
  - *And confirmation that the software physics models properly calculate the key phenomena (V&V).*
  - *Successful V&V can only be achieved if an adequate, high-fidelity data matrix and/or exact analytical solution set are available to benchmark the calculational results.*
  - *The software should be validated using a rigorous approach that is approved by the licensing authorities.*
  - *Each of the NGNP experiments will be formulated into a standard problem.*