Advanced Reactors RD&D: Technical Challenges

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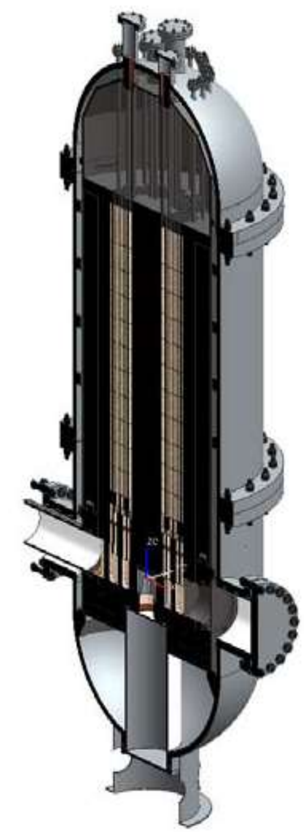
DOE-NE Program on Advanced Reactor Technologies

R&D focused on Advanced, Small and Modular Reactor Concepts

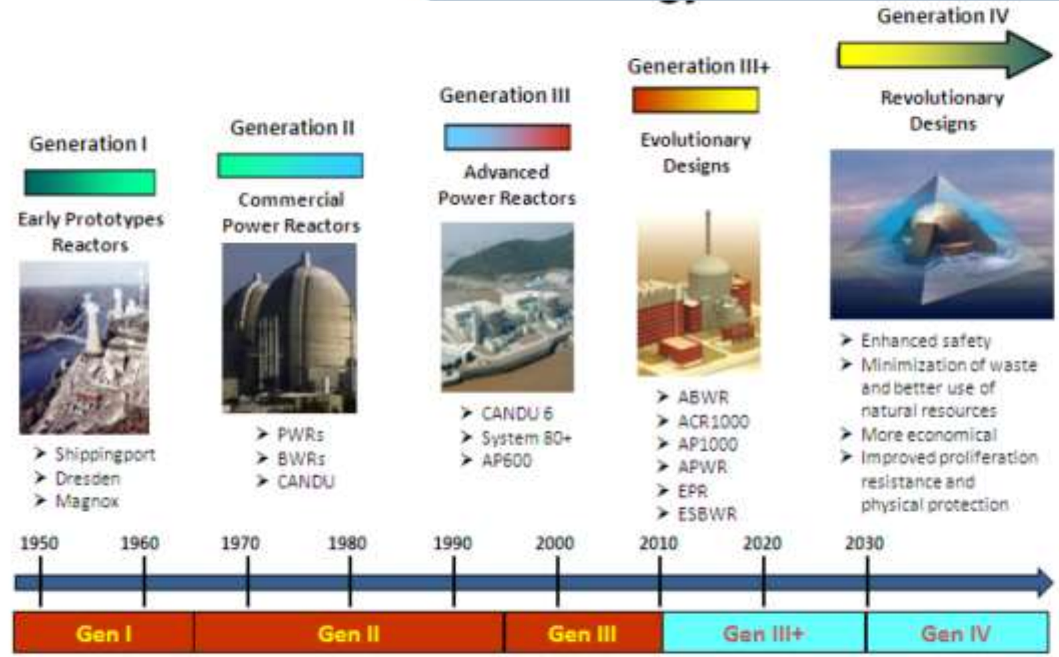
- Fast Reactor Technologies
  - For actinide management and electricity production
  - Current focus on sodium coolant
- High Temperature Reactor Technologies
  - For electricity and process heat production
  - Current focus on gas- and liquid salt-cooled systems
- Advanced Reactor Generic Technologies
  - Common design needs for advanced materials, energy conversion, advanced instrumentation and control strategies, decay heat removal systems and modeling methods
- Advanced Reactor Regulatory Framework
  - Development of licensing requirements for advanced reactors
- Advanced Reactor System Studies
  - Analyses of capital, operations and fuel costs for advanced reactor types



High Temperature Test Facility
Oregon State University

Generation IV Nuclear Systems

- Six Generation IV Systems considered internationally
- Often target missions beyond electricity
  - High temperature energy products
  - Fuel cycle benefits



System	Neutron spectrum	Coolant	Outlet coolant Temp. °C	Fuel cycle	Size (MWe)
VHTR (Very high temperature reactor)	thermal	helium	900-1 000	open	250-300
SFR (Sodium-cooled fast reactor)	fast	sodium	550	closed	30-150, 300-1 500, 1 000-2 000
SCWR (Supercritical water cooled reactor)	thermal/fast	water	510-625	open/closed	300-700 1 000-1 500
GFR (Gas-cooled fast reactor)	fast	helium	850	closed	1200
LFR (Lead-cooled fast reactor)	fast	lead	480-800	closed	20-180, 300-1 200, 600-1 000
MSR (Molten salt reactor)	Epithermal/fast	fluoride salts	700-800	closed	1 000

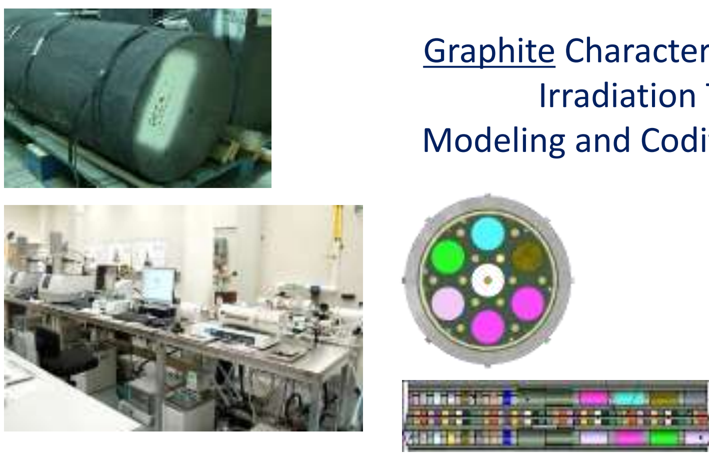
Comparison of Key Reactor Characteristics

	Gen III ALWR	*Gen IV VHTR*	*Gen IV SFR*
Applications	electricity generation	electricity generation, heat supply	electricity generation, actinide management
Power, MW_{th}	3000-4500	600-800 (block) 300-400 (pebble)	800-3500 (loop or pool plant)
Power Density, W/cm^3	50-100	≤ 6.5	200-400
Primary Coolant (T_{Outlet}, ºC)	H_2O (300-350)	He (850-1000)	Na (510-550)
Primary System Pressure (MPa)	15.5	7.1	0.1
Fuel Material	UO_2	UO_2, $UC_{0.5}O_{1.5}$	(U,TRU) oxide, metal alloy
Fuel Form	pellet	Triso coated particle	pellet or slug
Fuel Element / Assembly	square pitch pin bundle	hex block, pebble	triangular pitch pin bundle with duct
Moderator	light water	graphite	none
Number of coolant circuits	2	1 or 2	3
Core Structural Material	zirconium alloy	graphite	ferritic steel
Power Conversion Cycle	steam Rankine	direct or indirect He Brayton	superheated steam Rankine, or S-CO_2 Brayton

HTGR Technology Development and Qualification Needs

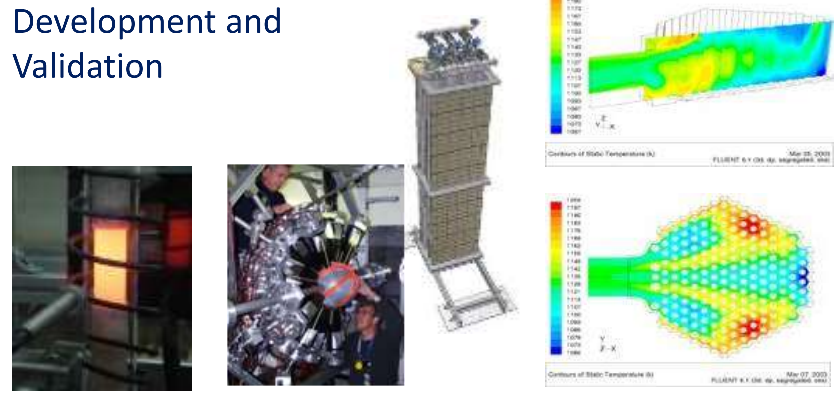
High Temperature Materials Characterization, Testing and Codification

Fuel Fabrication, Irradiation, and Safety Testing



Graphite Characterization, Irradiation Testing, Modeling and Codification

Design and Safety Methods Development and Validation



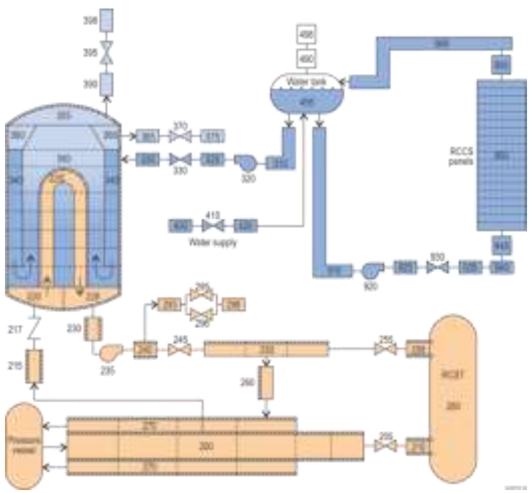
Safety Behavior of VHTR

- Inherent characteristics
  - Inert, single phase helium coolant
  - Multi-layer coated robust fuel particles prevent releases
  - High temperature stable graphite structure and moderator

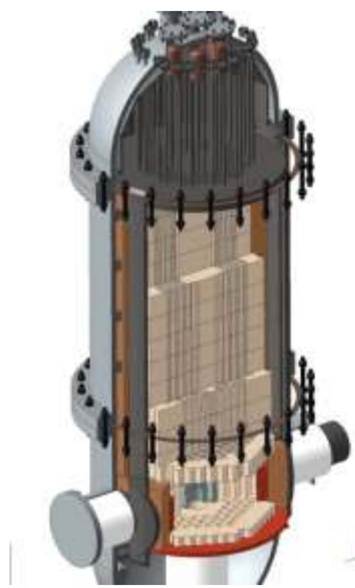
- Passively safe design
  - Slow heat-up of large graphite structures
    - In combination with low power density, implies long response times
  - Passive decay heat removal by radiation to cavity cooling
  - Annular core with negative temperature coefficients
  - No coolant voiding and/or change in moderation with temperature

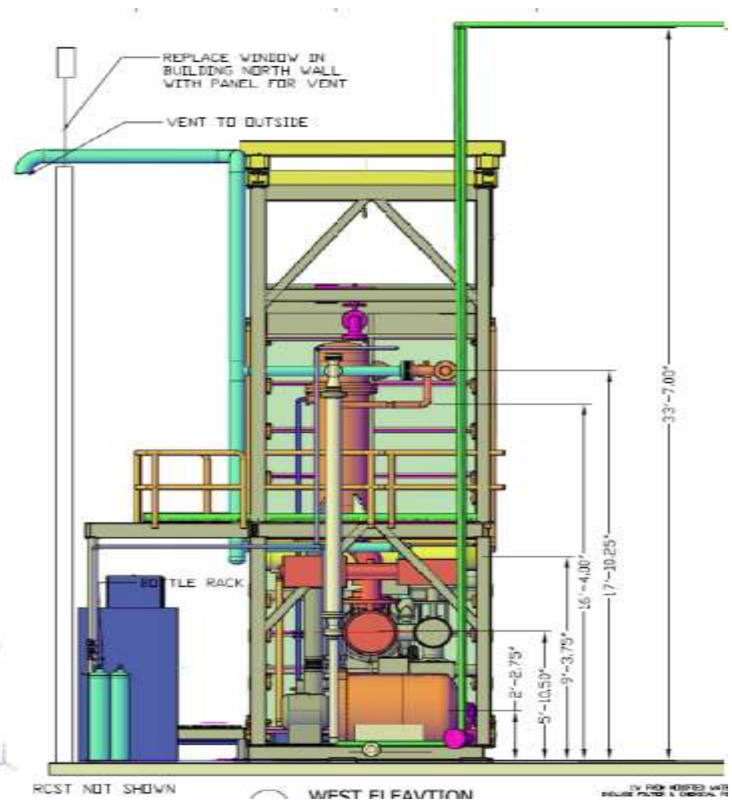
HTR Methods Validation – Thermal Fluidics

- **High Temperature Test Facility at Oregon State - Designed to validate models of depressurization and air ingress**



RELAP5 Model of Ex-Vessel Components

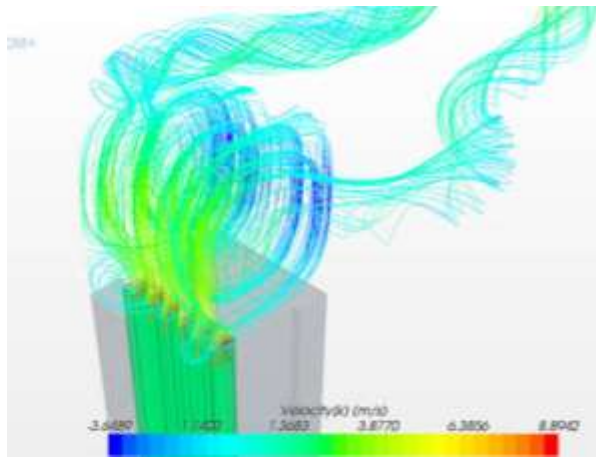




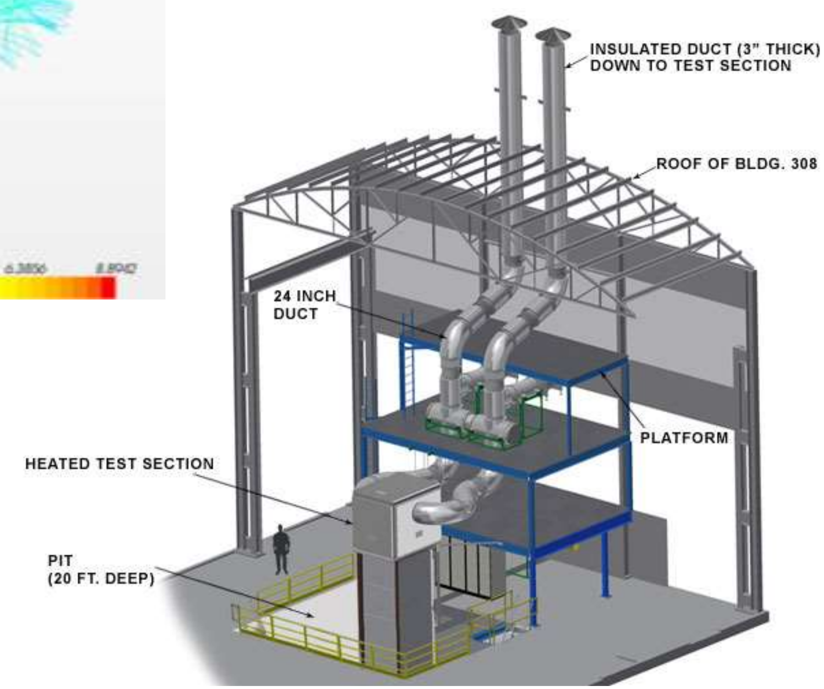
Cutaway View of HTTF

HTR Methods Validation – Reactor Cavity Cooling Systems

- **Natural Circulation Shutdown Test Facility (ANL) - Designed to validate models of passive vessel cooling systems**



Surface corrosion on riser ducts as seen from within outlet plenum



Fast Reactor R&D Priorities

- **Capital investment in reactors is the dominant cost of any nuclear fuel cycle; thus,**

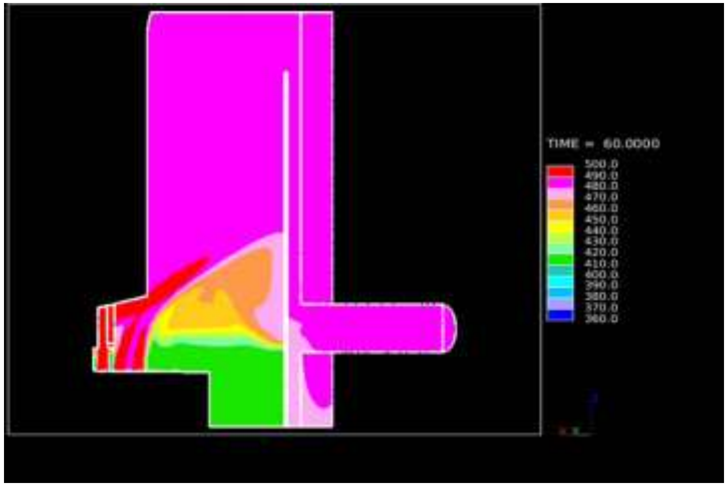
- **The primary <u>research focus is capital cost reduction</u> through application of innovative technology solutions**
  - Improved design approach – components and maintenance
  - Advanced structural materials to reduce commodities
  - Advanced energy conversion to improve size/efficiency
  - Advanced modeling and simulation to optimize performance
  - Fuel development to improve fuel cycle costs
  - Safety modeling to validate and assure margins

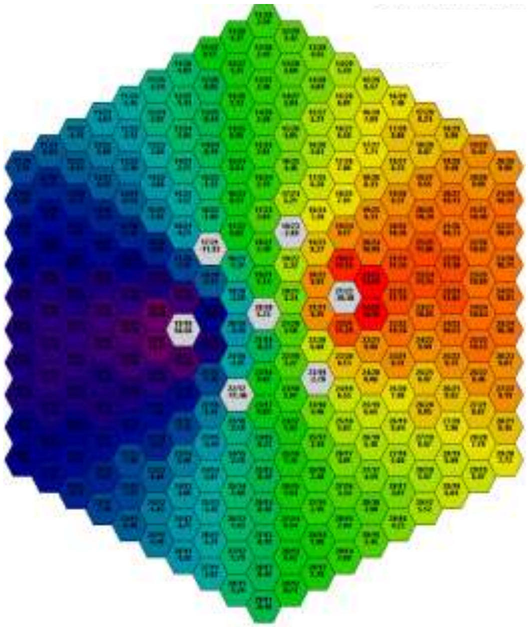
Safety Implications of SFR Design Approach

- Superior heat transfer properties of liquid metals allow:
  - Operation at high power density and high fuel volume fraction
  - Low pressure operation with significant margin to boiling
  - Enhanced natural circulation for heat removal
- Inherent safety design
  - Multiple paths for passive decay heat removal envisioned
  - Tailored reactivity feedbacks to prevent core damage
- High leakage fraction implies that the fast reactor reactivity is sensitive to minor geometric changes
  - As temperature increases and materials expand, a net negative reactivity feedback is inherently introduced
- Favorable inherent feedback in sodium-cooled fast reactors (SFR) have been demonstrated
  - EBR-2 and FFTF tests for double fault accidents

CRP Safety Benchmarks

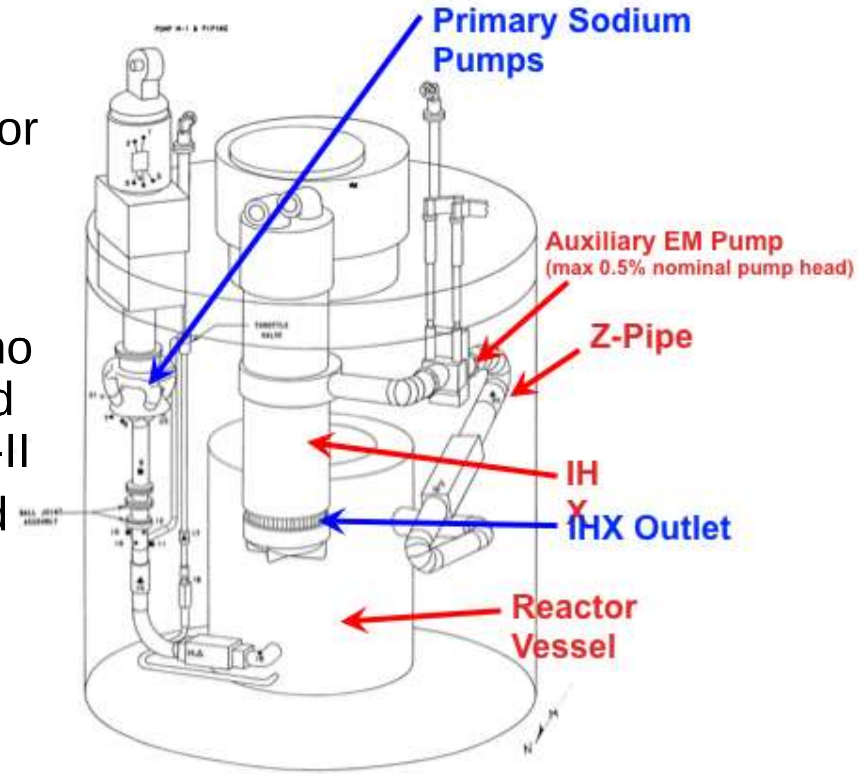
- IAEA coordinated research projects on SFR safety experiments
  - Benchmark analysis of 1995 MONJU natural circulation cooling experiments
    - Coolant flow and temperature distributions in the upper plenum following pump trip and scram
    - Confirmation of passive decay heat removal and evaluation of thermal stratification in the upper plenum
    - Assessment of sharp temporal and spatial temp. gradients in reactor vessel and upper core structures
  - Benchmark analysis of 2009 Phenix end-of-life tests
    - Control rod withdrawal tests for a sequence of asymmetric control rod movements to determine flux tilt
    - Evaluation of sodium natural convection for a protected loss of heat sink test





Fast Reactor Safety and Licensing R&D: EBR-II Benchmarks

- **EBR-II Shutdown Heat Removal Tests (SHRT)**
  - Importance of passive decay heat removal capability of advanced reactor designs has been emphasized in the aftermath of Fukushima accident
  - Potential of SFRs to survive even more severe accident initiators with no core damage has been demonstrated during the testing program with EBR-II
  - Two EBR-II SHRT tests are analyzed as the benchmark problems:
    - SHRT-17 (1984): A protected LOF from 100% power and flow
    - SHRT-45R (1986): An unprotected LOF from 100% power and flow



Fast Reactor Safety and Licensing R&D: EBR-II Benchmarks Participants

Four year project

– 2013: Blind analyses
– 2014: Preliminary assessments and model revisions
– 2015: Uncertainty evaluations and parametric analyses
– 2016: Documentation of the contributions in an IAEA report

EBR-II CRP: Participating Organizations	
China: *CIAE, Xian Jiaotong University, North China Electric Power University*	Korea, Republic of: *KAERI, KINS*
France: *IRSN*	The Netherlands: *NRG*
Germany: *KIT*	*Russia: IBRAE*
Italy: *ENEA, POLITO, UNIPI (GRNSPG)*	Switzerland: *PSI*
India: *IGCAR*	USA: *ANL, TerraPower*
Japan: JAEA, *Fukui University, Kyushu University*	

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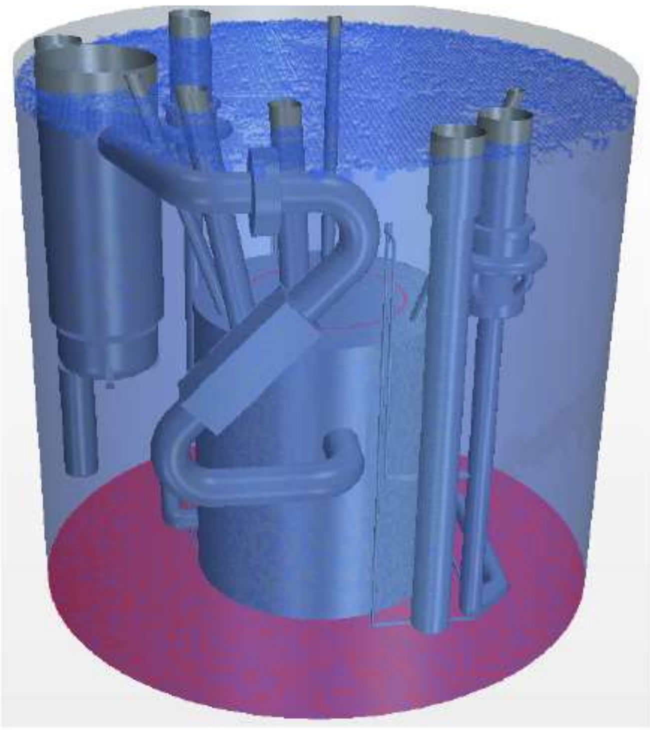
IAEA EBR-II Benchmark Project: Lessons Learned

- The two primary pumps and associated piping must be modeled individually
- A perfect mixing model is inadequate for the upper plenum
- Modeling thermal stratification in the cold pool is more important than first thought and may be necessary to improve temperature predictions at the pump inlets and entrances to the inlet plena
- Heat transfer between the Z-pipe and the cold pool important
- Leakage paths between the cold pool and the upper plenum appear to be important and must be investigated by parametric evaluations
- Thermal stratification in the Z-pipe and the upper plenum needs to be investigated in conjunction with the bypass flow through leakage paths
- Heat transfer between the instrumented subassemblies and the surrounding subassemblies requires detailed modeling, particularly for XX10
- Gamma heating needs to be modeled near the instrumented subassembly flowmeters and throughout steel pin subassemblies



New <u>advanced reactor </u>initiatives

- Industry-led (cost share) R&D Projects
- Licensing Framework for non-LWRs
- Advanced test or demonstration reactor study

Questions?