High Temperature Gas-cooled Reactor: Accident Analysis

Advanced Reactor Technologies

Idaho National Laboratory

Hans Gougar, PhD

Nuclear Engineer

Gerhard Strydom

National Technical Director – DOE Advanced Reactor Technologies Gas-Cooled Reactor Campaign

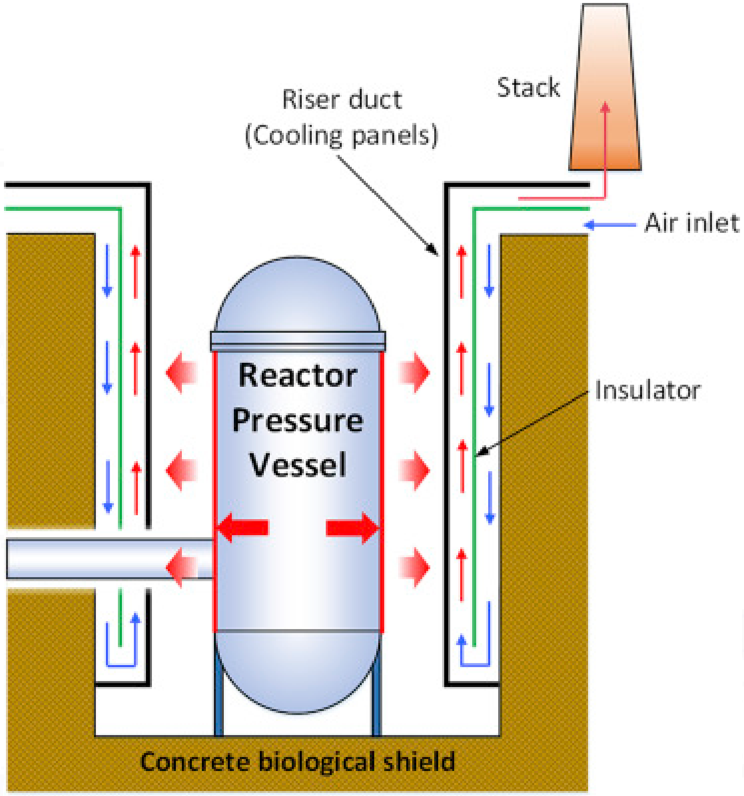
NRC HTGR Training July 16-17, 2019

ADVANCED REACTOR TECHNOLOGIES

HTGR Accident Analysis – Overview

- Types of Potential Accidents and Reactor Response
- Codes and Tools
- Experimental Validation
- Safety Analysis Approach
- Licensing Modernization Project
- Use of PRA in LMP, ASME/ANS, Non-LWR PRA Standard
- Methods for Incorporating Passive System Reliability into a PRA

Reactor Cavity Cooling

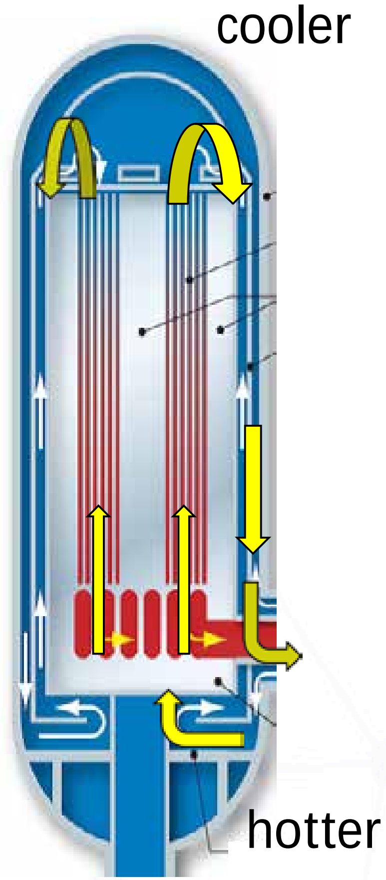


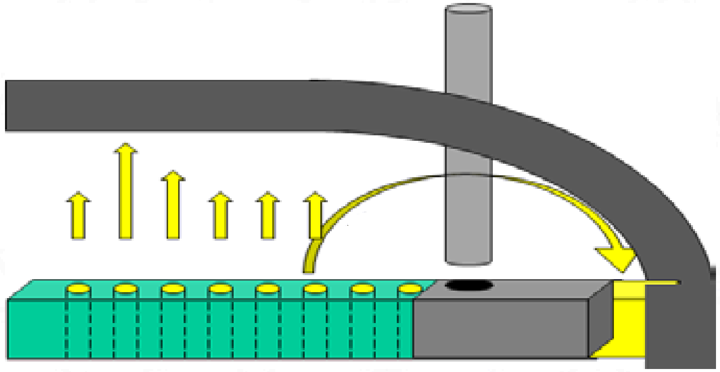
Dong-Ho Shin, Chan Soo Kim, Goon-Cherl Park, Hyoung Kyu Cho,Experimental analysis on mixed convection in reactor cavity cooling system of HTGR for hydrogen production, International Journal of Hydrogen Energy, Volume 42, Issue 34, 2017.

Pressurized Loss of Forced Cooling (PLOFC)

aka Pressurized Conduction Cooldown

- Blower trip leads to loss of forced flow through core. Doppler shuts down fission within first few seconds
- Forced downflow quickly yields to gravity-driven upflow through channels (or bed) - the transition flow is complex
- Core increases in temperature over many hours, then cools
- The hotter lower vessel structures drive ‘plenum-to-plenum’ currents and complex recirculation patterns
- RCCS pulls off heat from RPV
- If unmitigated (e.g., shutdown cooler), hot plumes impinging on upper plenum structures may damage CR guide tubes and the RPV head

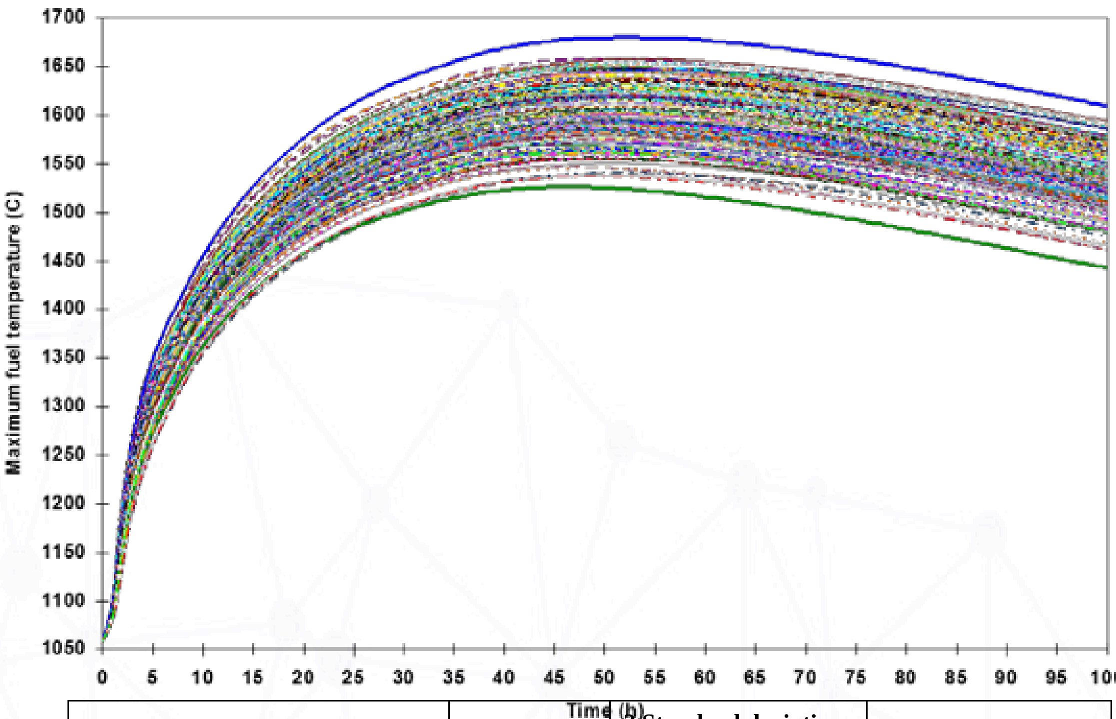




Valentin, F. I., N. Artoun, M. KawaJI and D. M. McEligot, 2018. Forced and mixed convection heat transfer at high pressure and high temperature in a graphite flow channel. J. Heat Transfer, 140, pp. 122502-1 to -10

DLOFC Uncertainties

- IAEA CRP-5 PEBBED model of the PBMR-400
- DLOFC transient sampled 200 times with SUSA uncertainty quantification code
- Input parameters sampled statistically.
- Obtains "band" of 200 peak fuel temperatures as function of time.
- 95th/95th tolerance limits of ~60°C observed (<4%).
- Only a small fraction of the fuel volume (<5%) reaches these temperatures for less than 150 hrs!



Parameter	Mean value	2 Standard deviations (2σ) value	PDF Type
Reactor power	400 MW	±8 MW (2%)	Normal & Uniform
Reactor inlet gas temperature (RIT)	500°C	±10°C (2%)	Normal & Uniform
Decay heat multiplication factor	1.0	±0.057 (5.7%)	Normal & Uniform
Fuel specific heat multiplication factor	1.0	±0.06 (6%)	Normal & Uniform
Reflector specific heat multiplication factor	1.0	±0.10 (10%)	Normal & Uniform
Fuel conductivity multiplication factor	1.0	±0.14 (14%)	Normal & Uniform
Pebble bed effective conductivity multiplication factor	1.0	±0.08 (8%)	Normal & Uniform
Reflector conductivity multiplication factor	1.0	±0.10 (10%)	Normal & Uniform

G. Strydom, 2010. PEBBED Uncertainty and Sensitivity Analysis of the CRP-5 PBMR DLOFC Transient Benchmark with the SUSA Code. INL/EXT-10-20531.

Air Ingress

- The amount of air that re-enters the primary system is a function of relative gas inventories and break location/orientation
- Oxidation of graphitic structures may ensue – mostly in the lower plenum; degrading structural integrity and perhaps causing further FP release if unmitigated. CO likely is generated
- Nuclear grade graphite <u>does not burn</u> (Windes, 2017) - but it does oxidize. Much of the oxygen is consumed by the lower graphite structures
- Moorman (2011) disagrees. Graphite oxidation remains misunderstood – much official OECD and IAEA documentation still erroneously refers to "graphite fires" at Windscale and Chernobyl accidents
- Graphite oxidation is temperature dependent.
  - Is it better to allow building circulation to cool the core structures or bottle it up to prevent O_2 exposure?

Oxidation/Degradation of graphite samples

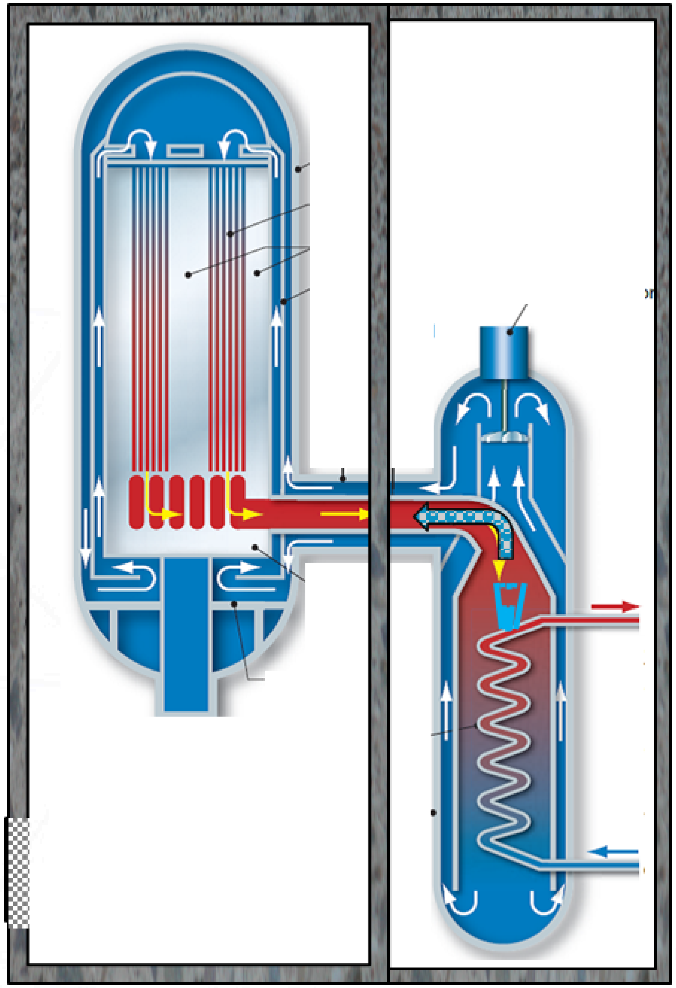
Type D Fire Extinguisher (graphite powder) used on electrical fires

Issue: How much oxygen can actually get back in? Sensitive to building air inventory and engineered vent pathways.

Windes, W. et al, "Discussion of Nuclear-Grade Graphite Oxidation in Modular High Temperature Gas-Cooled Reactors, 2017.
Moorman, R., "Phenomenology of Graphite Burning in Air Ingress Accidents of HTRs", 2011.
Srinivasan, M., and Carlson, D. (US NRC), "Enhanced Graphite Oxidation Under Potential Accident Scenarios", Proceedings of the 12th International Nuclear Graphite Specialists Meeting, Jeju, Korea, September 20-23, 2011.

Steam Generator (SG) Tube Rupture

- SG rupture sends water/steam into the RPV. Rupture may cause surrounding tubes to fail
- Reactivity insertion event (extra moderator)
- Moisture penetrates and oxidizes graphite surfaces. It picks up residual fission products normally trapped there. CO and volatile hydrocarbons formed
- Primary pressure relief valve opens, releasing circulating and leached FP into the building
- Relief valve closes but may reopen if more water enters and flashes. After 2-3 valve cycles, it is assumed to fail open
- Event is classified as a DLOFC with additional FP release



Issue: Amount (and phase) of water entering the core depends upon location of break. Fun multiphysics problem.

Rod Bank Withdrawal and Seismic Events

- Both are part of the reactivity insertion event class
- These events are challenging for modelers because the reactor may stay critical if not scrammed. Coupled neutronic/thermal-fluid simulations are computationally demanding for anything but simple point kinetics/homogenized core models
- Control rods in HTGRs are generally ‘banked’ (grouped). A spurious control signal may cause uncontrolled withdrawal, the rate of which determines rate of energy deposition and ultimate temperature increase (Rod ‘ejection’ is prevented by core design)
- If rapid, the heat surge will shut down the reactor (Doppler) before particle failure conditions are attained
- Explicit modeling of kernel energy deposition indicates that the lower-order (smeared) fuel models over-predict power and fuel temperature
- Likewise, seismically-induced pebble bed settling is computed to result in a positive reactivity insertion on the order of a rod withdrawal event.
- Earthquake effects on other plant structures would need to be evaluated

HTGR Accident Analysis – Overview

- Types of Potential Accidents and Reactor Response
- Codes and Tools
- Experimental Validation
- Safety Analysis Approach
- Licensing Modernization Project
- Use of PRA in LMP, ASME/ANS, Non-LWR PRA Standard
- Methods for Incorporating Passive System Reliability into a PRA

Codes and Methods used for Past and Current HTGR Analysis – Prismatic

USED FOR LICENSING BENCHMARKED NEAMS

Purpose	Previously Used Codes	Codes for Today and tomorrow	Remark
Cross Section Generation	**MICROX MICROR**	SCALE/MCNP SERPENT	Slowing-down in graphite, heterogeneity, leakage, control rods
Criticality/Rod Worth Steady State Verification	**DIF3D**	Monte Carlo	
Steady State Design and Fuel Management	**DIF3D/BURP**	Monte Carlo with Burnup	
Time-dependent Reactor Dynamics	**?**	PARCS-AGREE, NEM-THERMIX, PHISICS-RELAP	Load-follow, steam ingress. PLOFC/DLFOC may work with point kinetics
Local Thermal-Fluidics	**TAC-2D, TREVER, DEMISE**	ANSYS, CFD	High fidelity conjugate heat transfer using finite element analysis
Core-wide Thermal Fluidics System Analysis	**DETRAC,TAP,** SINDA-FLUINT, RELAP5, GRSAC,	RELAP5-3D, AGREE, GASNET, RELAP7, SAM	1-D Channel Flow with input power trajectory. Flow mixing (network), Bypass flow
Thermomechanical Analysis	ANSYS	ANSYS, ABAQUS, COMSOL, GRIZZLY	2-D and 3-D solid mechanics with time history.
Seismic	ANSYS	ANSYS, MASTODON	
Fuel Performance Ex-Core FP transport	**GA/KFA**	PARFUME,COPA, TIMCOAT, BISON MELCOR, etc.	Fuel performance data and models may indicate that one need not take credit for retention in the building

Codes and Methods used for Past and Current HTGR Analysis – Pebble

USED FOR LICENSING BENCHMARKED NEAMS

Purpose	Previously Used Codes	Codes for Today and tomorrow	Remark
Cross Section Generation	**GAM-ZUT-THERMOS**	SCALE/MCNP SERPENT	Slowing-down in graphite, heterogeneity, leakage, control rods
Criticality/Rod Worth Steady State Verification	MCNP/ MonteBurns	Monte Carlo	
Steady State Design and Fuel Management	**VSOP** PEBBED	PARCS-AGREE MAMMOTH-PRONGHORN	Must account for flowing and mixing of fuel, including during the running –in period. Only VSOP does all of this currently
Time-dependent reactor dynamics	**TINTE**	PARCS-AGREE, NEM, RATTLESNAKE-PRONGHORN	Load-follow, steam ingress. PLOFC/DLFOC may work with point kinetics
Local Thermal-Fluidics	**ANSYS**	CFD (Fluent, **Star-CCM**, NEK5000)	High fidelity conjugate heat transfer using finite element analysis
Core-wide Thermal Fluidics System Analysis	**THERMIX-KONVEK**	RELAP5-3D, AGREE, GASNET, PRONGHORN, RELAP7,SAM, FLOWNEX, SURVEY	Porous medium conjugate heat transfer with subgrid pebble conduction for the core. Bypass flow in the reflector
Thermomechanical Analysis	**ANSYS**	ANSYS, ABAQUS, COMSOL, GRIZZLY	2-D and 3-D solid mechanics with time history.
Seismic	**ANSYS**	ANSYS, MASTODON	2-D and 3-D time-dependent structural mechanics with time history
Fuel Performance Ex-Core FP Transport	PANAMA, FIPREX-GETTER	PARFUME,COPA,TIMCOAT, STACY, BISON MELCOR, etc.	Semi-analytical models of FP transport in fuel.

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

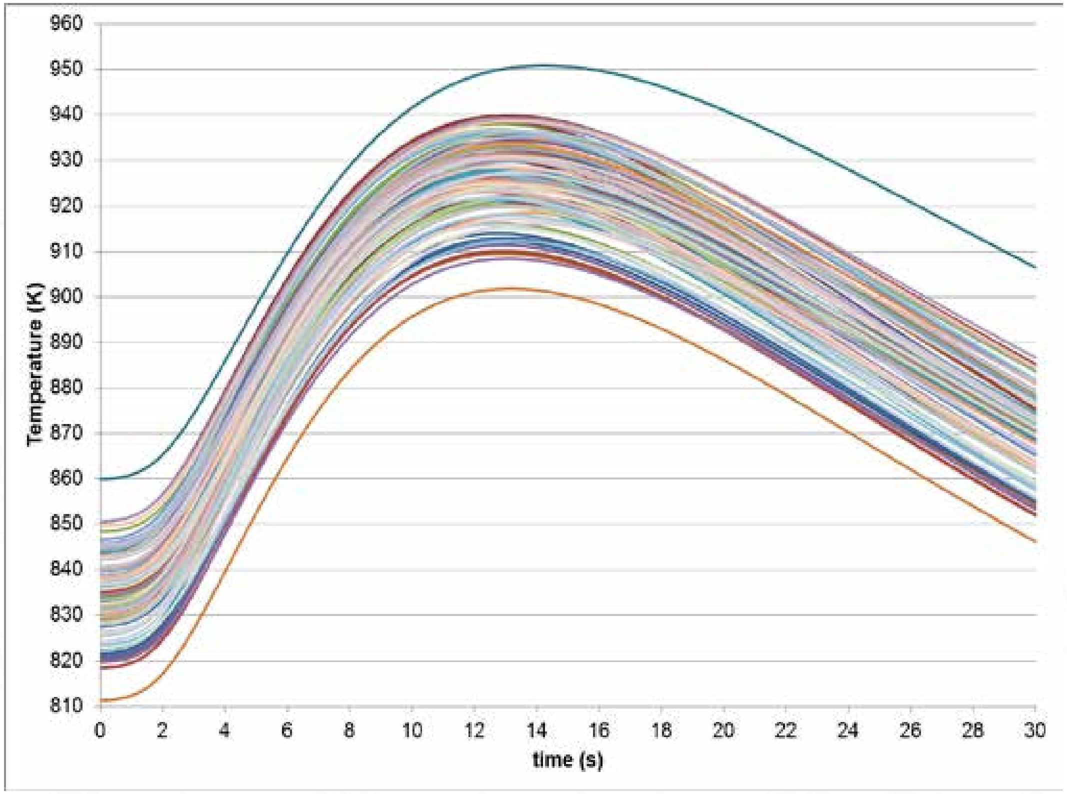
Facility	Temp. Defect	C. R. Worth	Power Distr.	K_{eff}	Water Ingress	Decay Heat
HEU-CORES						
Peach Bottom Critical	±14%	-11%	±10%	±0.7%	NA	-
Peach Bottom	-11% to +4%	-6% to +10%	±10%	±0.7%	-	NA
HTGR Critical	+6%	+4% to 13%	-	-0.1% to +1.0%	-	-
Fort St. Vrain	-9% to +12%	±10%	±15%	±0.5%	-	NA
HTLTR	±8%	-	-	-	-	-
KAHTER	-	NA	NA	-0.3% to +6%	±13%	-
DRAGON	NA	-11%	NA	-	-	NA
HEU/LEU CORES						
AVR	-25%	-5% to +15%	-	±11%	-	NA
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.

Recent Uncertainty Assessment

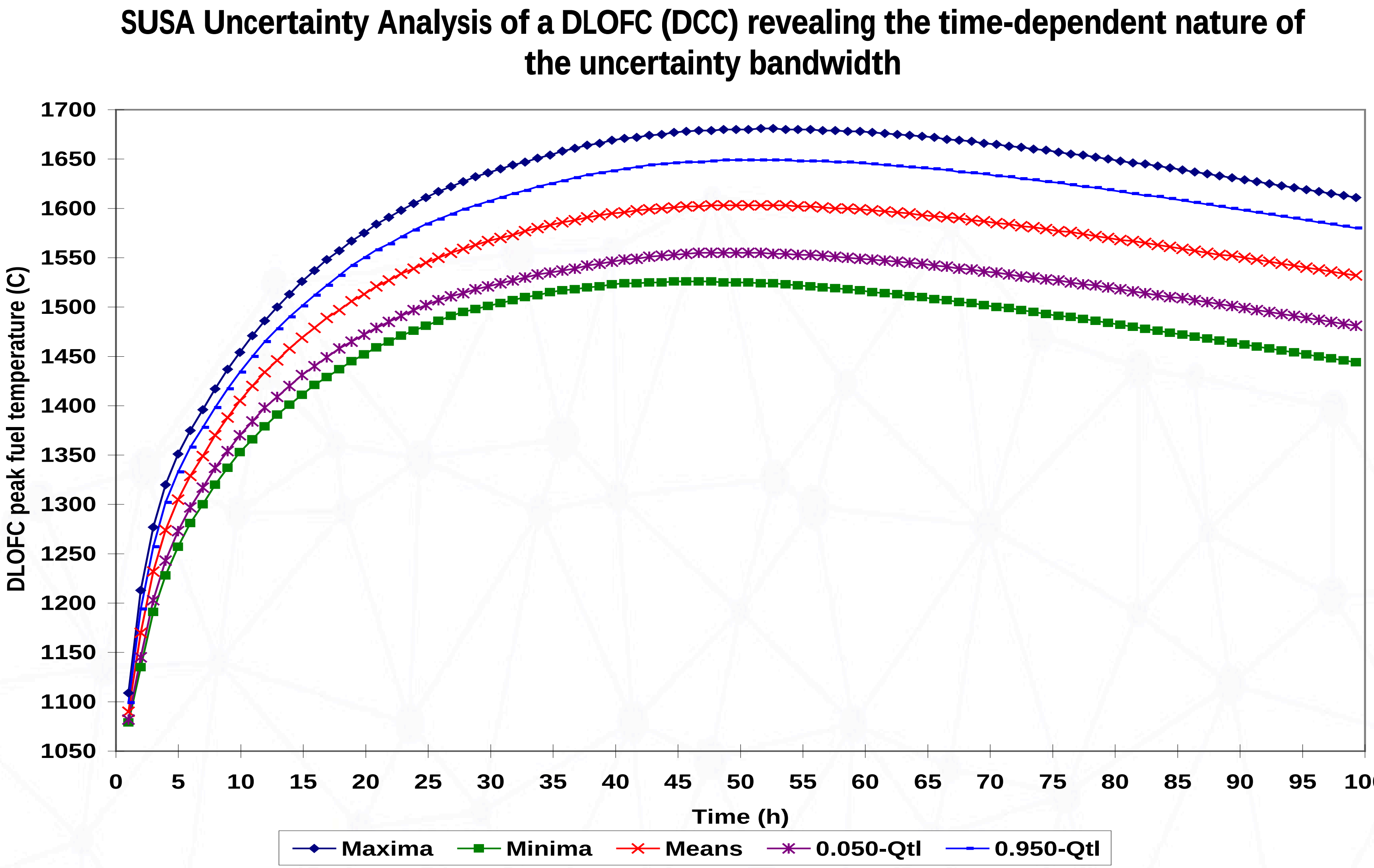
Input Parameter	2σ Uncertainty
Core Bypass Flow (or gap width)	± 5.9%
Reactor Inlet Temperature	± 2%
Helium Mass Flow	± 2%
Pebble Bed Thermal Conductivity	± 8%
Fuel Sphere Graphite Thermal Conductivity	± 14%
Fuel Sphere Graphite Specific Heat	± 6%
Reflector Thermal Conductivity	± 10%
Reflector Specific Heat	± 10%
Reflector Emissivity	± 7%
Core Barrel Emissivity	± 5%
Reactor Pressure Vessel (RPV) Emissivity	± 7%
Core Barrel Thermal Conductivity	± 5%
Helium Thermal Conductivity	± 5%
RPV Thermal Conductivity	± 5%
Core Barrel Specific Heat	± 5%
Helium Specific Heat	± 5%

Fuel temperature response for 1,000 perturbed CFX calculations (slow power ramp transient)



Strydom G., Bostelmann, F., and Yoon, S. J., 2015, Results for Phase 1 of the IAEA Coordinated Research Project on HTGR Uncertainties, INL/EXT-14-32944, Rev. 2.

Time-dependent Uncertainties



Strydom, G., Uncertainty and Sensitivity Analysis of a Pebble Bed HTGR Loss of Cooling Event, Science and Technology of Nuclear Installations, Volume 2013, Article ID 426356.

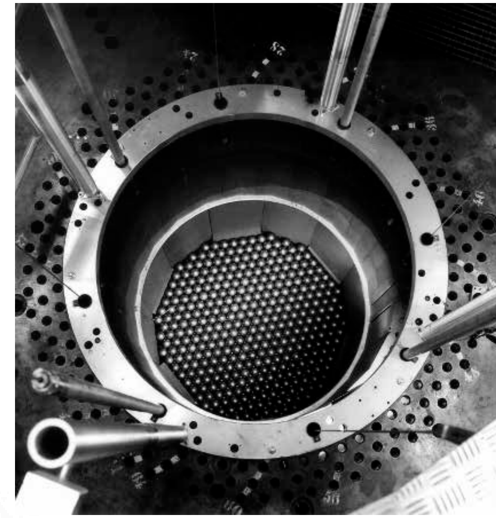
HTGR Accident Analysis – Overview

- Types of Potential Accidents and Reactor Response
- Codes and Tools
- Experimental Validation
- Safety Analysis Approach
- Licensing Modernization Project
- Use of PRA in LMP, ASME/ANS, Non-LWR PRA Standard
- Methods for Incorporating Passive System Reliability into a PRA

Critical Experiments for Neutronics

PEBBLE BED

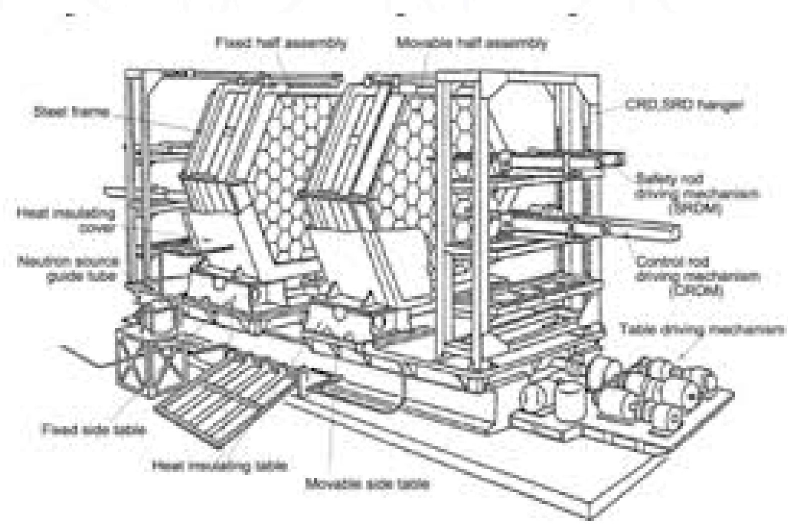
- HTR-Proteus critical experiments
  - 1980's, Paul Scherrer Institute, Switzerland
  - Bess 2014
- HTR-10 Initial Criticality
  - ~2000, INET, China
  - IAEA 2003, 2013
- ASTRA
  - Mid 1990's , Kurchatov Institute, Russia
  - IAEA 2013
- HTR-PM – scheduled to go critical within a year. INET has offered up physics test results to support a GIF benchmark



The HTR Proteus experiment from above.

PRISMATIC

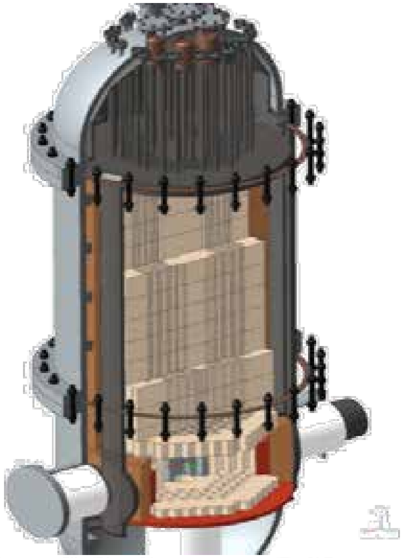
- VHTRC
  - Mid-1980s, Japan
  - Ref: Bostelmann 2016
- HTTR
  - Ref: IAEA 2003, 2013
- Fort St. Vrain
  - Ref: Martin, 2016

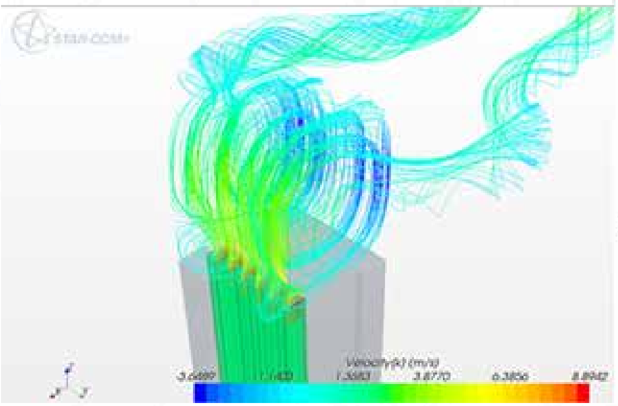


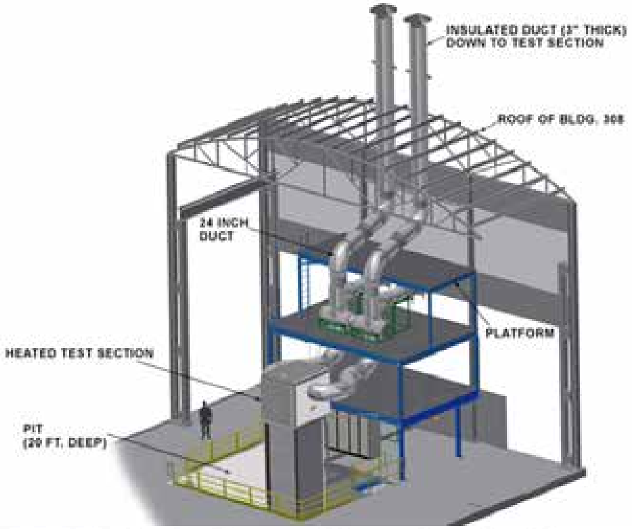
Sketch of the VHTRC Experiment

Thermal Fluid Integral Experiments Sponsored by DOE

- High Temperature Test Facility at Oregon State University
- Natural Circulation Shutdown Heat Removal Facility at Argonne National Lab
- Vendors participated in the design and test matrix planning for the HTTR and NSTF experiments.
- Framatome and X-Energy facilitated the conversion of NSTF to a water-cooled configuration.
- The NRC sponsored the design and construction of HTTF

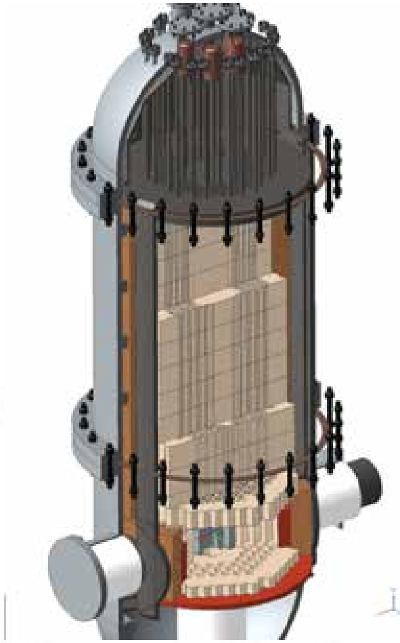


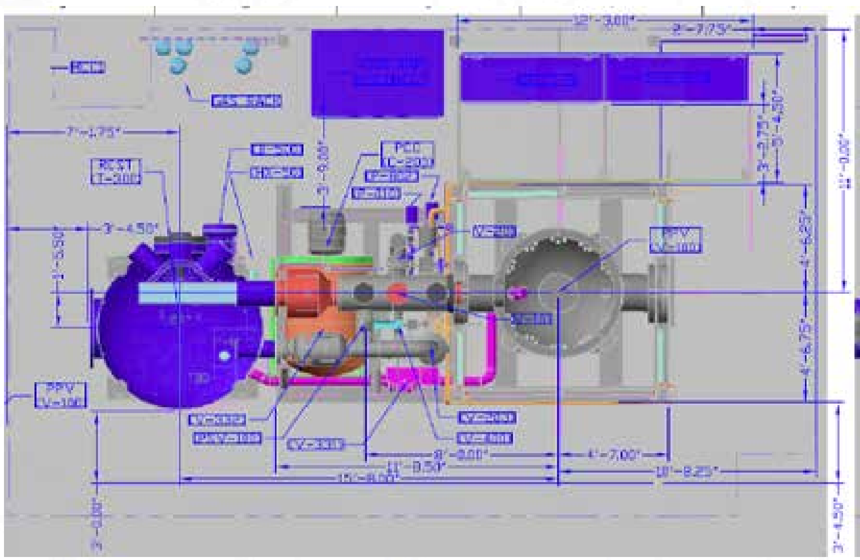




High Temperature Test Facility (HTTF) – Oregon State University

- Designed to simulate core behavior during a depressurized loss of forced cooling accident
- ¼-scale MHTGR
- Design allows different pipe break configurations to characterize the exchange of helium and air between the primary loop and building
- 428 experiment measurements (362 thermocouples, 48 gas sensors, 18 others) and 31 process instruments
- Primary focus is on depressurized conduction cooldown transient, but other experiments are planned as well.
- Matrix testing resumed in April 2019



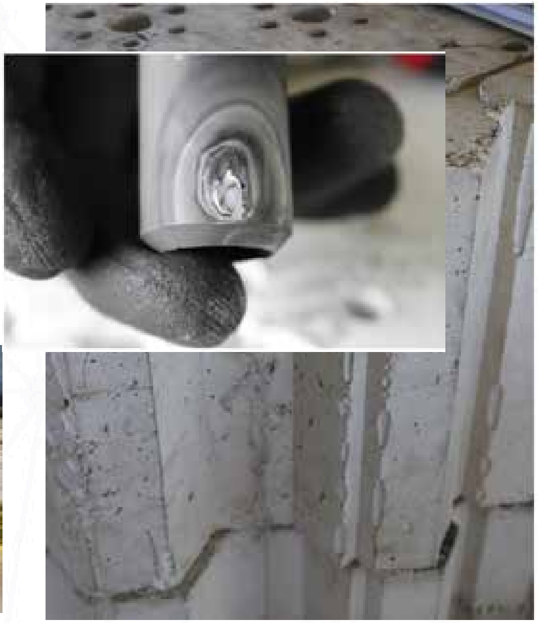


High Temperature Test Facility – Oregon State University (cont.)

- HTTR encountered local over-heating during initial testing. The heaters and instrumentation have been re-designed and rebuilt. Four ceramic blocks were replaced

New ‘dogbone’ heater rod

Current heater rod stack



Damaged ceramic core block resulting from block shifting that degraded heater rod electrical continuity causing localized hot spots - 4 need to be replaced

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

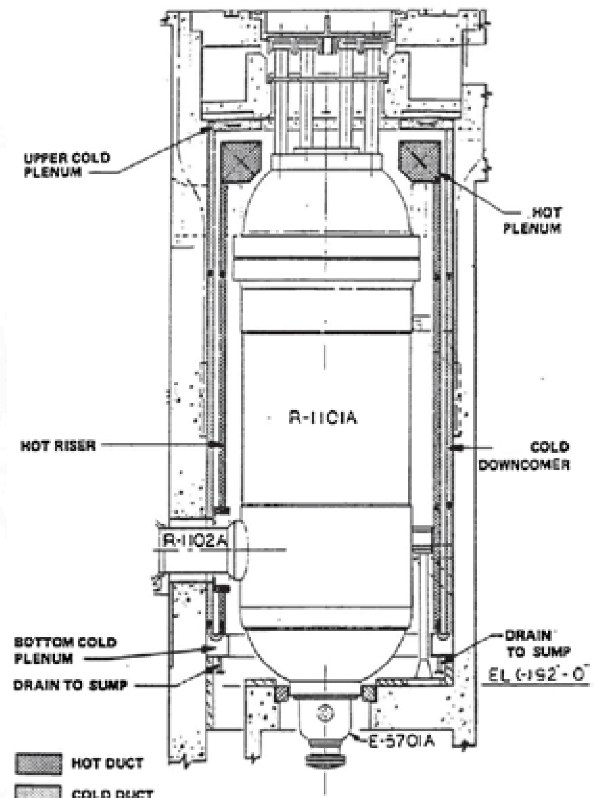
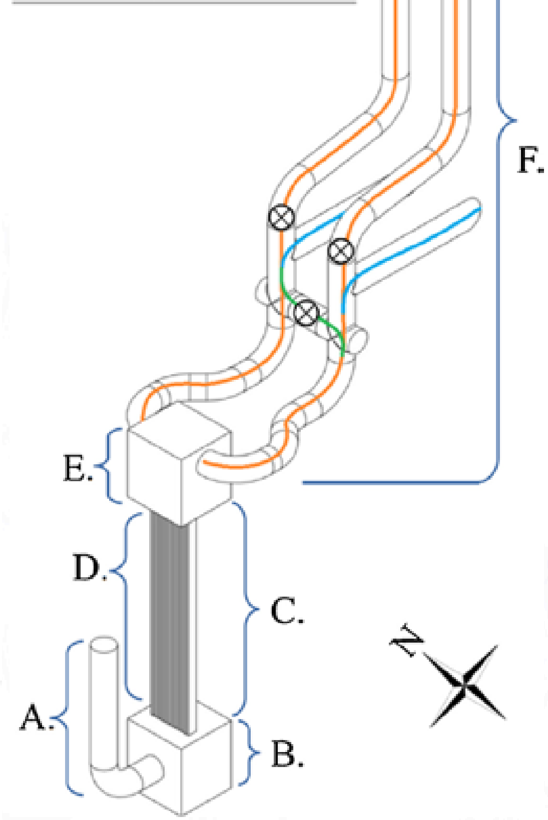


FIGURE 5.5-6
RCCS COOLING PANEL CONFIGURATION – ELEVATION
HIGH TEMPERATURE GAS-COOLED REACTOR PRELIMINARY SAFETY INFORMATION DOCUMENT HTGR-86-024

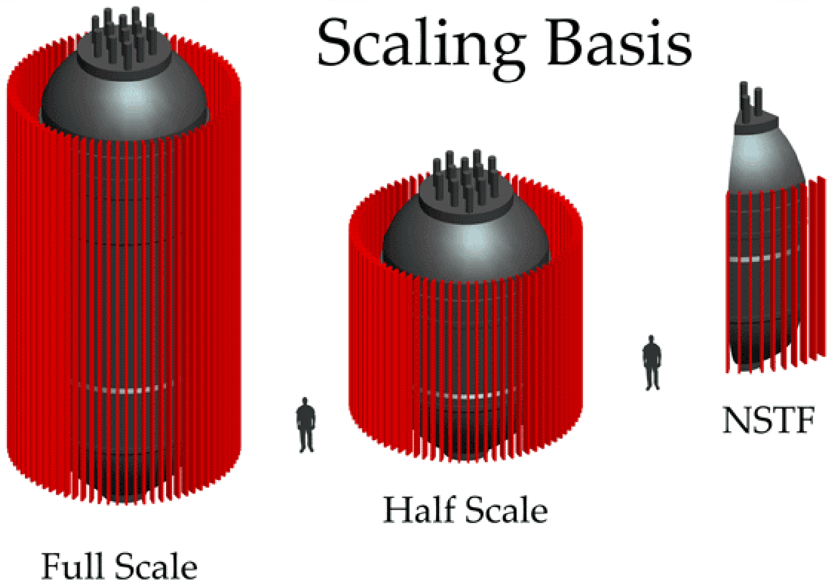
A.	Inlet downcomer	3.35-m
B.	Inlet plenum	1.12-m
C.	Heated cavity	6.92-m
D.	Riser channels	7.50-m
E.	Outlet plenum	1.88-m
F.	Chimney ducts	18.0-m



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

Reactor Vessel (Cavity) Cooling System Experiments at ANL's Natural Circulation Shutdown Heat Removal Facility (NSTF)

- Originally constructed to support General Electric PRISM (Power Reactor Inherently Safe Module) development refurbished to half-MHTGR scale under the Next Generation Nuclear Plant project
- Air-cooled experiments completed in 2016
- Conversion to water-cooled configuration (Framatome SC-HTGR). Experiments have commenced.



Finned water-cooled riser channels in the NSTF

Numerous NEUP-funded Experiments

- Separate and Mixed Effects studies in:
  - Bypass Flow
  - Core Heat Transfer
  - Air Ingress
  - Plenum-to-Plenum Heat Transfer
  - Lower plenum flow
  - Building Response to depressurization

High Pressure, High Temperature Facility for Natural Circulation Experiments, City College of New York , NEUP Project 11-3218,Kawaji)

Phenomena Characterized in:

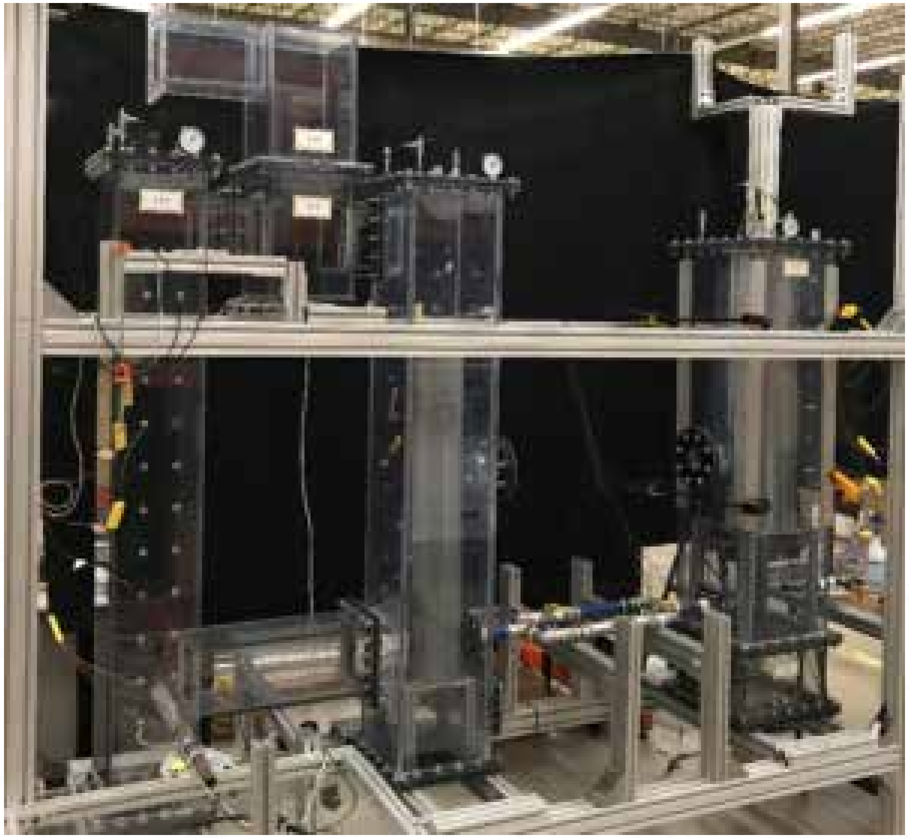
Schultz, R.R., Gougar, H., Lommers, L., Identification and Characterization of Thermal Fluid Phenomena Associated with Selected Operating/Accident Scenarios in Modular High Temperature Gas-Cooled Reactors, Paper 2018-0177, Proceedings of HTR 2018, Warsaw, Poland, October 8-10, 2018.

Final Reports downloadable from https://neup.inl.gov

Building Response

- Advanced Reactor Concepts grant to Texas A&M with cost-share with AREVA
- Designed to look at flow in the reactor building subsequent to pipebreak and depressurization
- Initial tests were completed. Further experiments solicited in the 2019 NEUP call

1/32-scale Building Response Experiment at Texas A&M



Se Ro Yang, Ethan Kappes, Thien Nguyen, Rodolfo Vaghetto, Yassin Hassan, Experimental study on 1/28 scaled NGNP HTGR reactor building test facility response to depressurization event, Annals of Nuclear Energy, Volume 114, 2018.

HTGR Accident Analysis – Overview

- Types of Potential Accidents and Reactor Response
- Codes and Tools
- Experimental Validation
- Safety Analysis Approach

Safety Analysis Approach

Each scenario must be evaluated in the context of:

- Phenomenology and sequence timing (what happens and when)
- Break size, break location, orientation
- Graphite structural material (nuclear or non-nuclear)
- Building response

Design-dependent

Design implications

- Mitigation systems?
- Accident management procedures?

Design-dependent: redundancies, diversities, etc.

Credible break size:

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

"Cliff-edges" have been largely eliminated but 'knife-edge" transitions can effect source terms, structural integrity

Summary

- **Safety margins are huge** in terms of radiological release. Accident scenarios develop slowly and have few consequences for the fuel. Other structures may be vulnerable
- **Graphite can oxidize** and air ingress can degrade structural integrity if not designed away or mitigated. Graphite does not burn
- **Moisture ingress** (steam generator tube rupture) may be the limiting case with respect to fission product release
- Codes system designed for HTGRs exist and have improved since the first HTGRs were licensed (but they were adequate for the purpose). Computational power is driving **more extensive use of high fidelity tools**. Margins, however, still allow **approximate methods** to be used effectively
- Uncertainties can be **large, time-dependent** and are mostly attributable to uncertainties in material properties and tolerances, not so much to neutronic uncertainties
- Critical experiment data are limited but probably adequate. Integral experiments are underway at ANL and Oregon State University to **confirm gross thermal-fluid behavior**. Numerous SET and MET experiments have been conducted. (International integral tests and engineering reactors were not discussed but may be useful.)
- Safety Analysis must factor individual design features but the general approach applies to all modular HTGRs. “Cliff-edges” really do not appear in existing design concepts but **“knife-edge” phenomena should be identified and understood to characterize margins to FP release**

Suggested Reading List

- John D. Bess, Leland M. Montierth, Oliver Köberl & 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
- Bostelmann,F., Hammer, H. Ortensi, J. Strydom, G. Velkov, K., Zwermann, W., Criticality calculations of the Very High Temperature Reactor Critical Assembly benchmark with Serpent and SCALE/KENO-VI, Annals of Nuclear Energy, Volume 90, 2016,
- Dong-Ho Shin, Chan Soo Kim, Goon-Cherl Park, Hyoung Kyu Cho, Experimental analysis on mixed convection in reactor cavity cooling system of HTGR for hydrogen production, International Journal of Hydrogen Energy, Volume 42, Issue 34, 2017.
- HTGR Technology Course for the Nuclear Regulatory Commission, 2010.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of High Temperature Gas Cooled Reactor Performance: Benchmark Analysis Related to Initial Testing of the HTTR and HTR-10, IAEA-TECDOC-1382, IAEA, Vienna (2003).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of High Temperature Gas Cooled Reactor Performance: Benchmark Analysis Related to the PBMR-400, PBMM, GT-MHR, HTR-10 and the ASTRA Critical Facility, IAEA-TECDOC-1694, IAEA, Vienna (2013).
- Lisowski, D.D. et al, Experimental Observations of Natural Circulation Flow in the NSTF, Nuclear Engineering and Design 306, (2016) 124-132.
- Martin, W, Creation of a Full-Core HTR Benchmark with the Fort St. Vrain Initial Core and Assessment of Uncertainties in the FSV Fuel Composition and Geometry, Battelle Memorial Institute United States. Department of Energy. Office of Scientific and Technical Information.
- Moorman, R., “Phenomenology of Graphite Burning in Air Ingress Accidents of HTRs”, 2011.
- Schultz, R.R., Gougar, H., Lommers, L., Identification and Characterization of Thermal Fluid Phenomena Associated with Selected Operating/Accident Scenarios in Modular High Temperature Gas-Cooled Reactors, Paper 2018-0177, Proceedings of HTR 2018, Warsaw, Poland, October 8-10, 2018.
- Strydom G., Bostelmann, F., and Yoon, S. J., 2015, Results for Phase 1 of the IAEA Coordinated Research Project on HTGR Uncertainties, INL/EXT-14-32944, Rev. 2.
- Strydom, G., Uncertainty and Sensitivity Analysis of a Pebble Bed HTGR Loss of Cooling Event, Science and Technology of Nuclear Installations, Volume 2013, Article ID 426356
- Se Ro Yang, Ethan Kappes, Thien Nguyen, Rodolfo Vaghetto, Yassin Hassan, Experimental study on 1/28 scaled NGNP HTGR reactor building test facility response to depressurization event, Annals of Nuclear Energy, Volume 114, 2018.
- Valentin, F. I., N.Artoun, M. KawaJI and D. M. McEligot, 2018. Forced and mixed convection heat transfer at high pressure and high temperature in a graphite flow channel. J. Heat Transfer, 140, pp. 122502-1 to -10
- Windes, W. et al, “Discussion of Nuclear-Grade Graphite Oxidation in Modular High Temperature Gas-Cooled Reactors, 2017.

INL

Idaho National Laboratory