INL/MIS-21-62420-Revision-0

Modular High Temperature Gas-Cooled Reactor: Accident Analysis

April 2021

Changing the World's Energy Future

Jason Albert Christensen

Idaho National Laboratory

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Modular High Temperature Gas-cooled Reactor: Accident Analysis

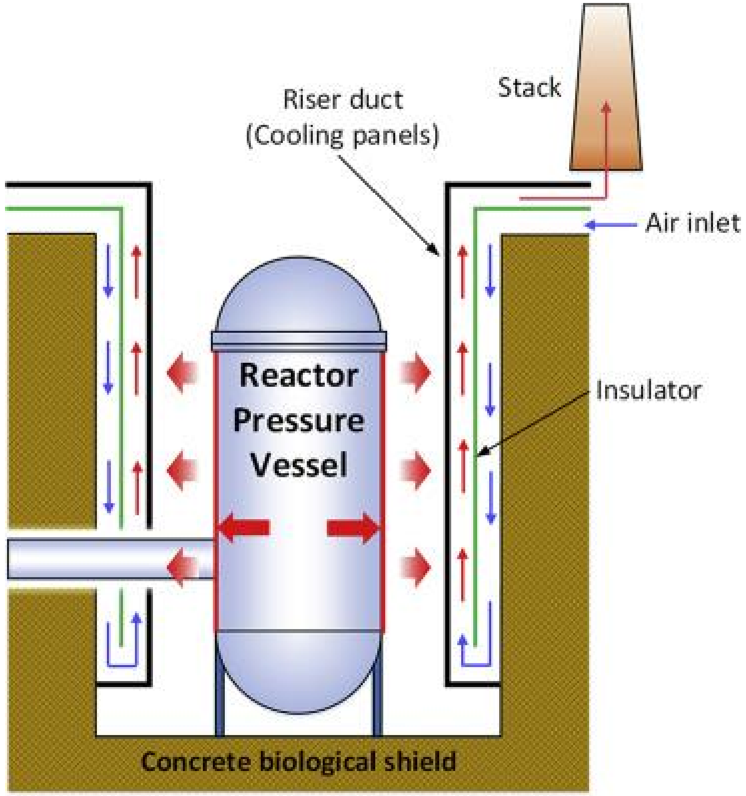
CNSC Seminar

Idaho National Laboratory

HTGR Accident Analysis – Overview

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

Reactor Cavity Cooling

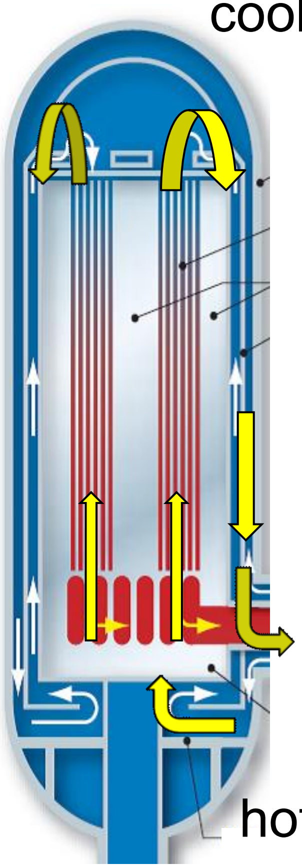


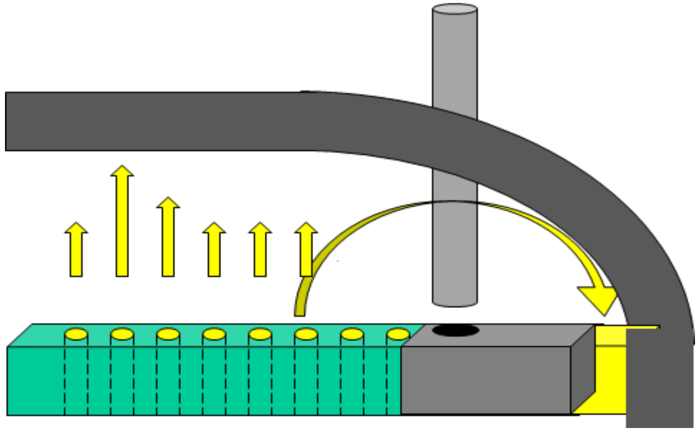
Dong-Ho Shin, Chan Soo Kim, Goon-Cherl Park, Hyoung Kyu Cho,Experimental analysis on mixed convection in reactor cavity cooling system of HTGR for hydrogen production, International Journal of Hydrogen Energy, Volume 42, Issue 34, 2017.

Pressurized Loss of Forced Cooling (PLOFC)

aka Pressurized Conduction Cooldown

- Blower trip leads to loss of forced flow through core. Doppler feedback shuts down fission within first few seconds.
- Forced downflow quickly yields to buoyancy-driven upflow through channels (or bed) - the transition flow is complex.
- Core temperature increases over many hours, then cools down slowly.
- Hotter lower vessel structures drive 'plenum-to-plenum' currents and complex recirculation patterns.
- RCCS provides heat sink capability.
- If unmitigated (e.g., shutdown cooler not used), hot plumes impinging on upper plenum structures may damage CR guide tubes and the RPV head for some of the higher power designs.

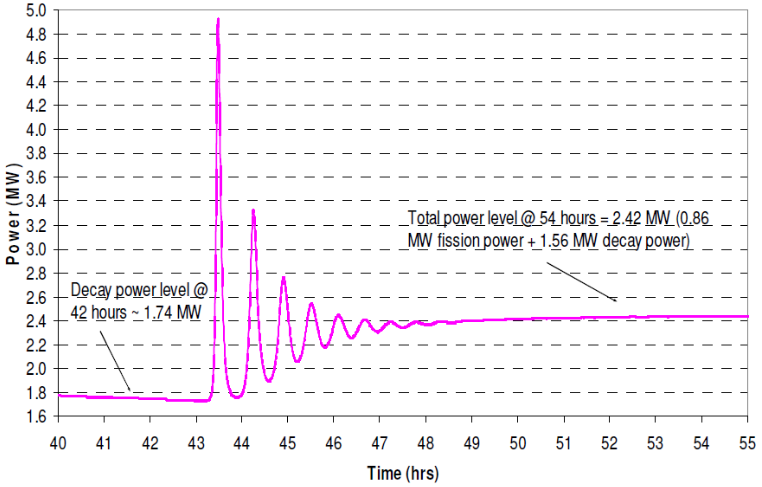




Valentin, F. I., N. Artoun, M. KawaJI and D. M. McEligot, 2018. Forced and mixed convection heat transfer at high pressure and high temperature in a graphite flow channel. J. Heat Transfer, 140, pp. 122502-1 to -10

Depressurized Loss of Forced Cooling (DLOFC)

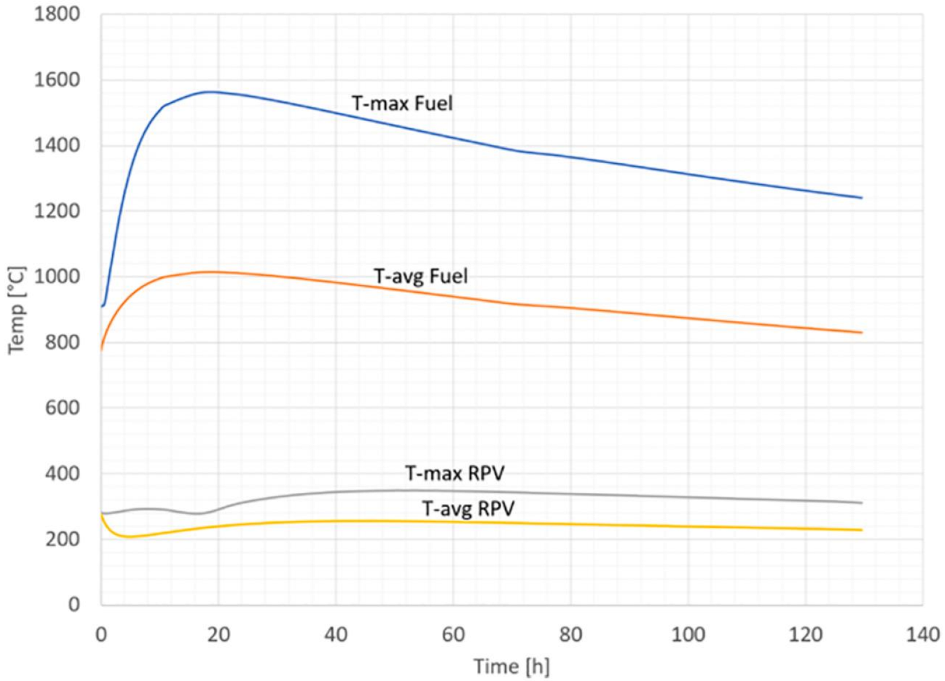
aka Depressurized Conduction Cooldown



PBMR-400 DLOFC with no SCRAM: re-critical power phase

- Postulated break in pressure boundary that leads to bounding safety case for fuel temperature.
- Break can vary in size from small leak (e.g., 5 mm) to large pipe break (65 mm).
- Depressurization of system can be rapid (within seconds) or over hours for small leaks.

Xe-100 (X-energy) DLOFC: fuel and RPV maximum and average temperatures



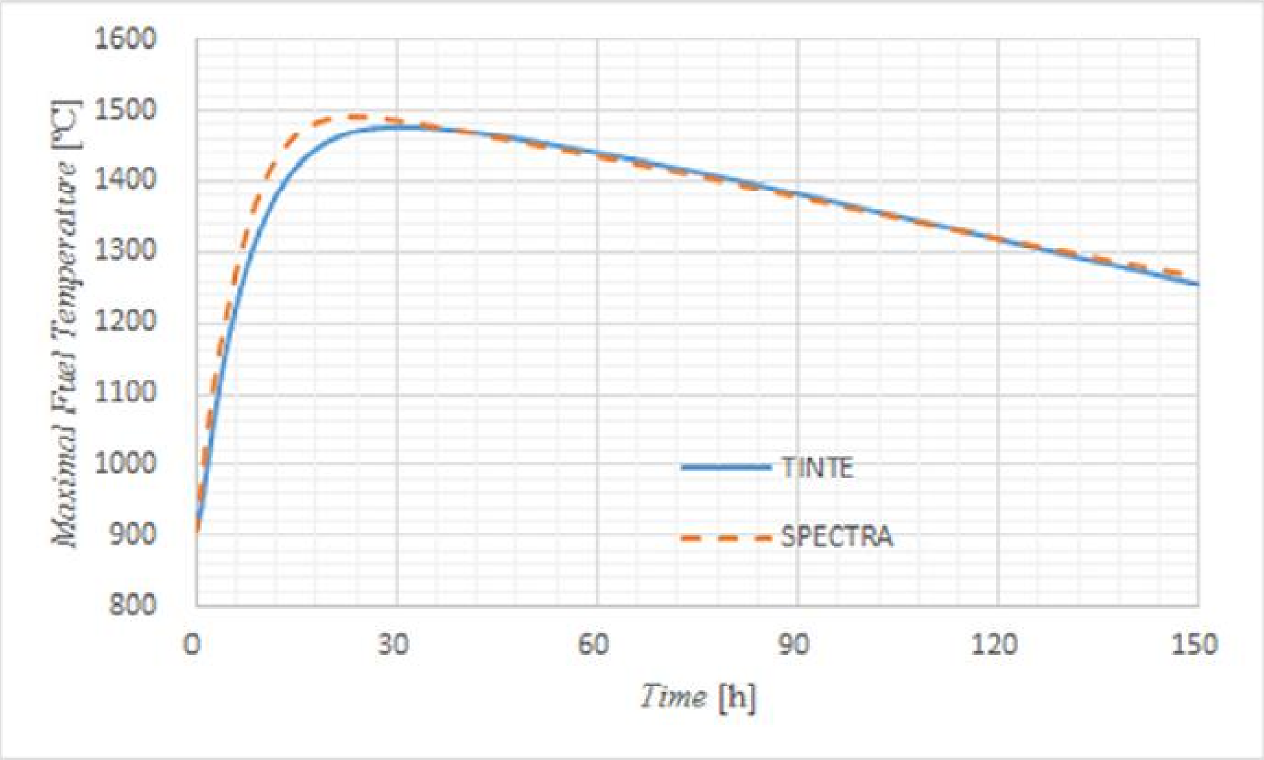
- Same phenomena as PLOFC, but natural convection plays a lesser role.
- Air/helium mixture (building atmosphere) can ingress into break location once blow-down phase is completed
- Graphite oxidation starts in the hot bottom structures first, then moves upwards in the core through the flow channels/bed.

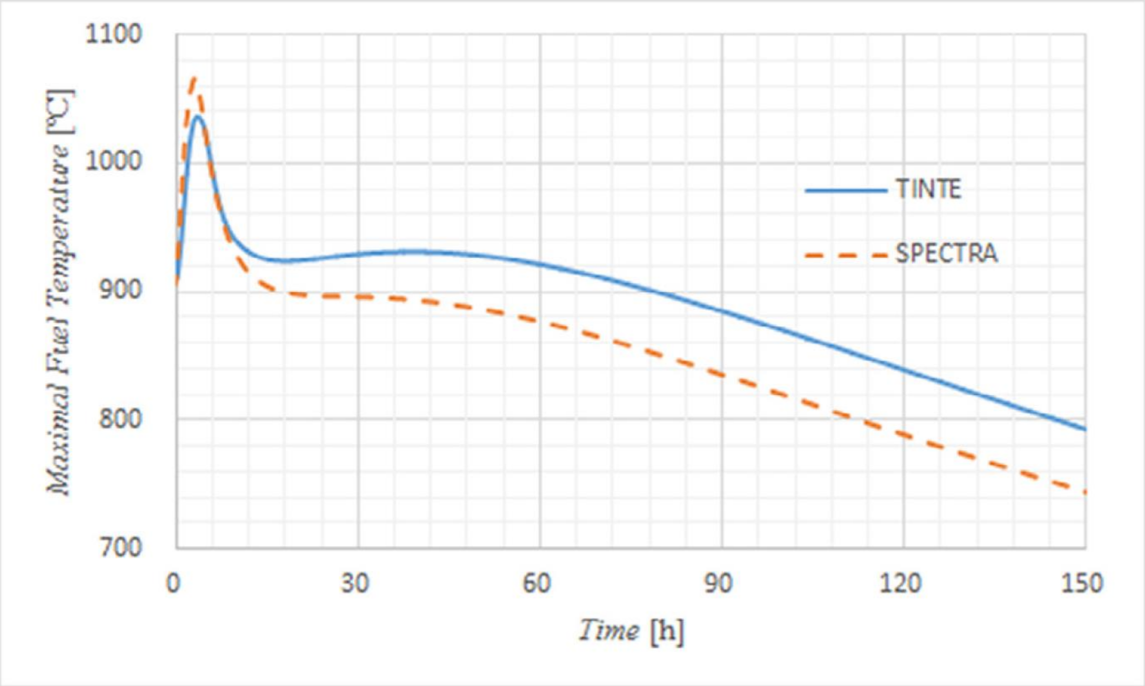
G. Strydom, 2008. TINTE Transient Results for the OECD 400 MW PBMR Benchmark. Proc. of ICAPP 2008.
E. Mulder, W. Boys, 2020. Neutronics characteristics of a 165MWth Xe-100 reactor. Nuclear Engineering and Design 357 (2020) 110415

DLOFC vs PLOFC: HTR-PM

DLOFC (65 mm pipe break): Total loss of coolant within 60 s.

Peak fuel temperature reaches ~1480C @ 30 hours.





PLOFC (turbine trip): Core pressure settles at ~6 MPa.

Peak fuel temperature reaches ~1140C @ 5 hours.

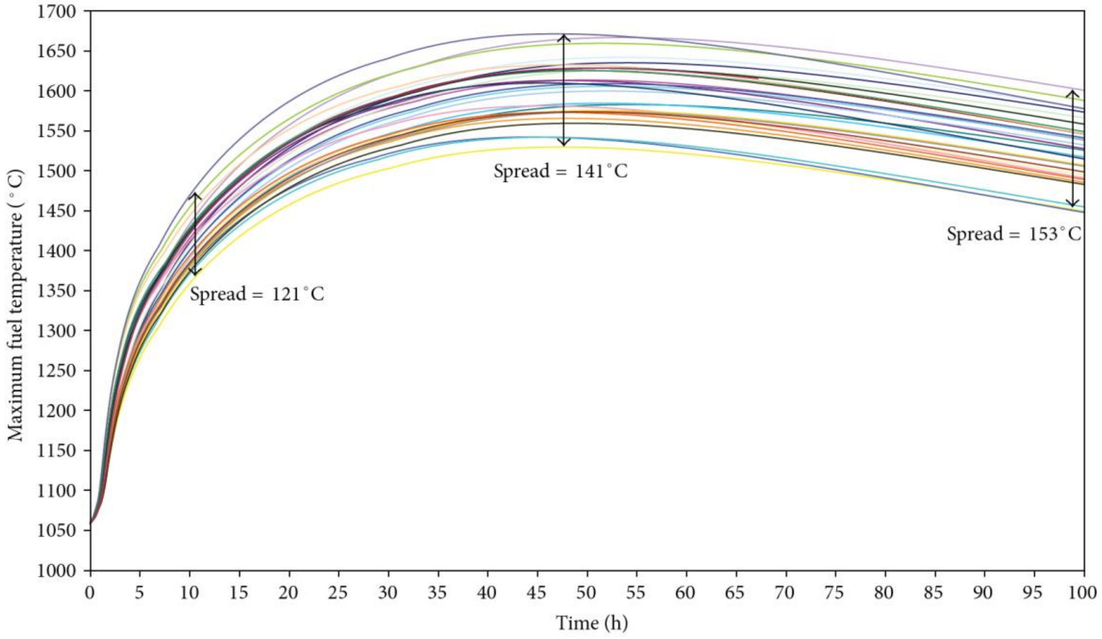
Natural convection at 6 MPa in core transport heat from hot bottom core region to upper cold core region.

Effect is also present during DLOFC event, but at atmospheric pressure not as effective.

Z. Yanhua et al., 2018. Study on the DLOFC and PLOFC accidents of the 200 MWe pebble-bed modular high temperature gas-cooled reactor with TINTE and SPECTRA codes. Annals of Nuclear Energy 120 (2018) 763–777

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!

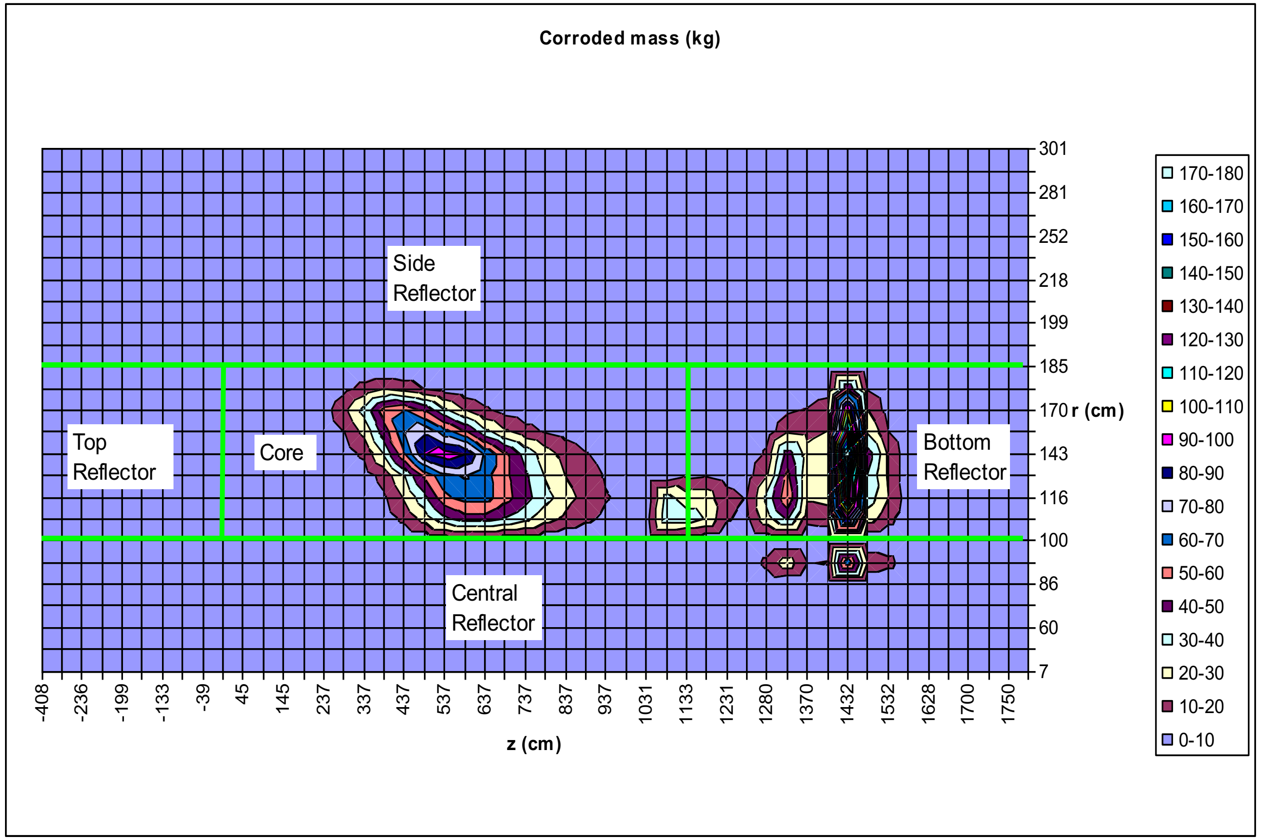


Parameter	Mean value	2 Standard deviations (2σ) value	PDF Type
Reactor power	400 MW	±8 MW (2%)	Normal & Uniform
Reactor inlet gas temperature (RIT)	500°C	±10°C (2%)	Normal & Uniform
Decay heat multiplication factor	1.0	±0.057 (5.7%)	Normal & Uniform
Fuel specific heat multiplication factor	1.0	±0.06 (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

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.

Air Ingress

- The amount of air that re-enters the primary system is a function of building mixture composition 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 and CO2 generation is temperature dependent.
- Much of the oxygen is consumed by the lower graphite structures for a bottom break location.



Oxidation/Degradation of graphite samples

E. Sikik, 2008. SIMULATIONS OF THE AIR AND WATER INGRESS TRANSIENTS FOR THE PEBBLE BED MODULAR REACTOR (PBMR) BY MEANS OF THE TINTE CODE, HTR2008.
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.

Air Ingress (2)

- Nuclear grade graphite does not burn (Windes, 2017) – but it does oxidize.
- 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?

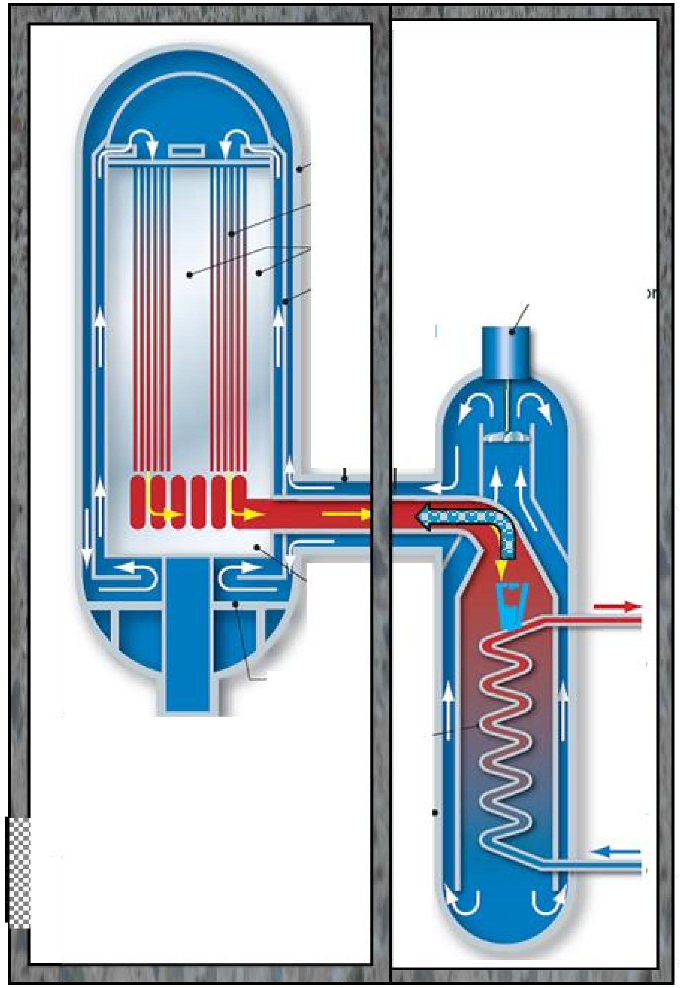
Issue: How much oxygen can actually get back in? Sensitive to building air inventory and engineered vent pathways.

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

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.

Steam Generator 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



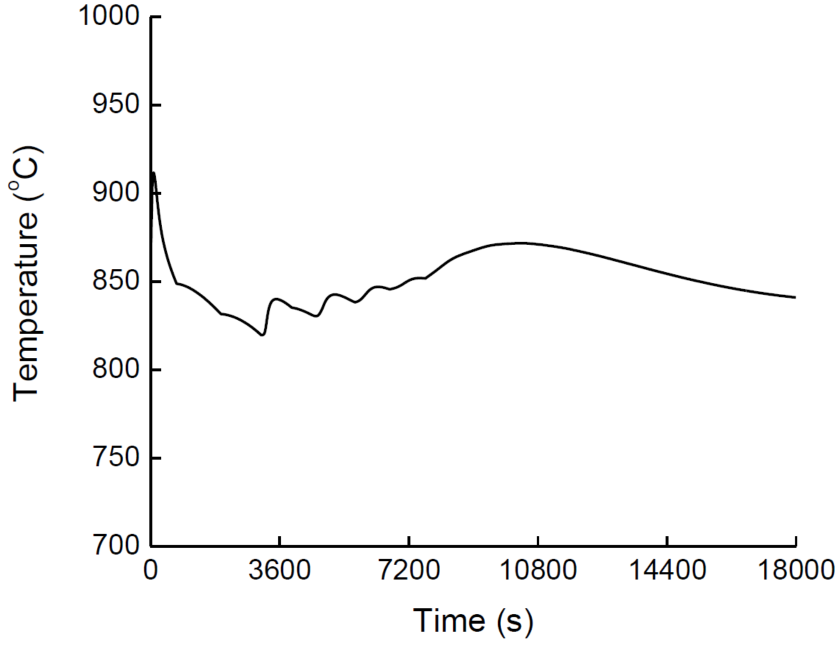
Issue: Amount (and phase) of water entering the core depends upon location of break – challenging multiphysics problem.

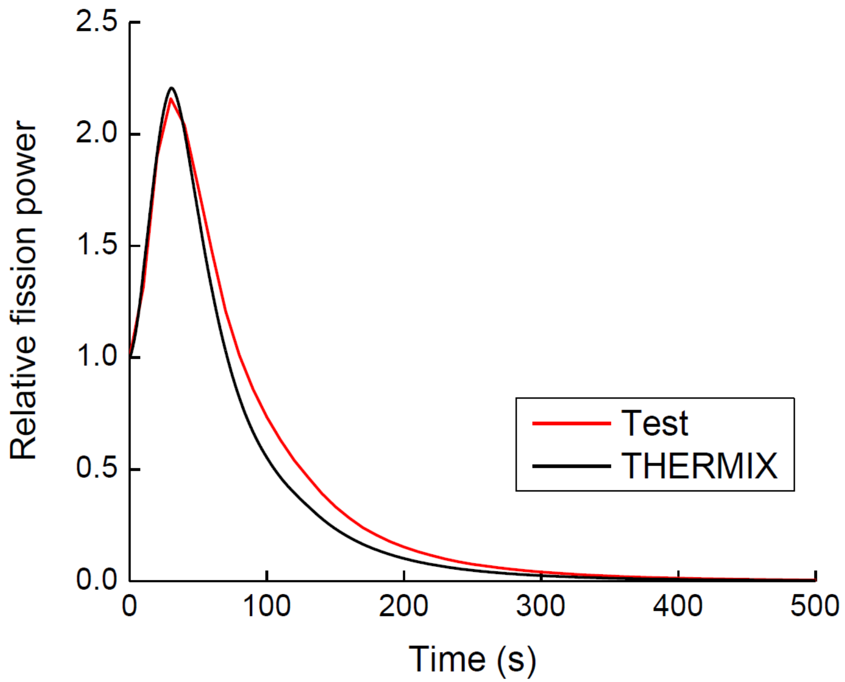
Rod Bank Withdrawal and Seismic Events

- 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 computational demanding.
- 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).
- 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 relatively small but positive reactivity insertion.
- Earthquake effects on other plant structures would need to be evaluated.

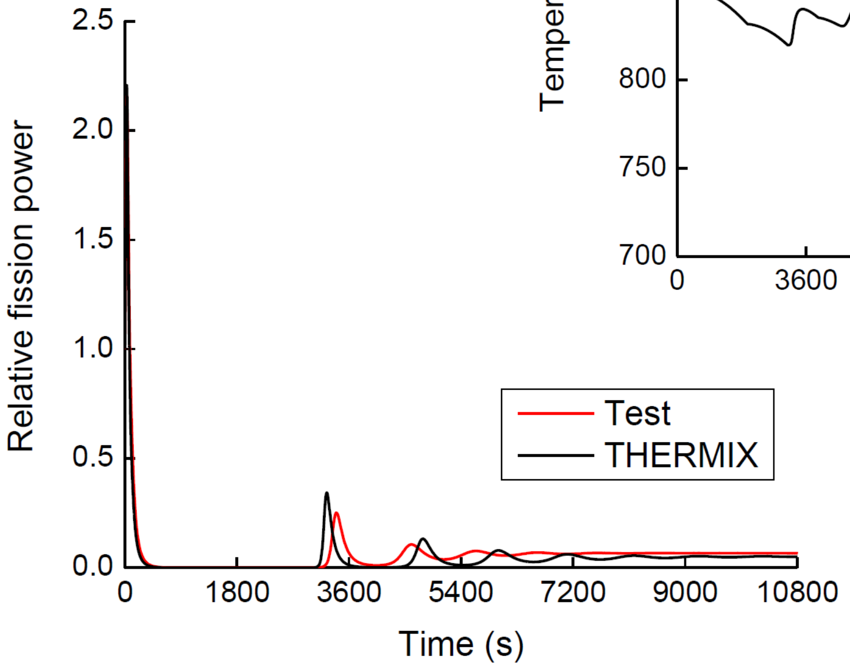
Rod Bank Withdrawal: HTR-10

Full rod bank withdrawal at 1 cm/s at HTR-10 pebble bed reactor





(a) Power in short term



(b) Power in long term

F. Gou, 2016. Dynamic response of the HTR-10 under the control rod withdrawal test without scram. International Youth Nuclear Congress 2016, IYNC2016, 24-30 July 2016, Hangzhou, China

HTGR Accident Analysis – Overview

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

Shenoy, A. and McEachern, D., HTGR Core Thermal Design Methods and Analysis, GA-A12985, 1974.

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

USED FOR LICENSING BENCHMARKED NEAMS

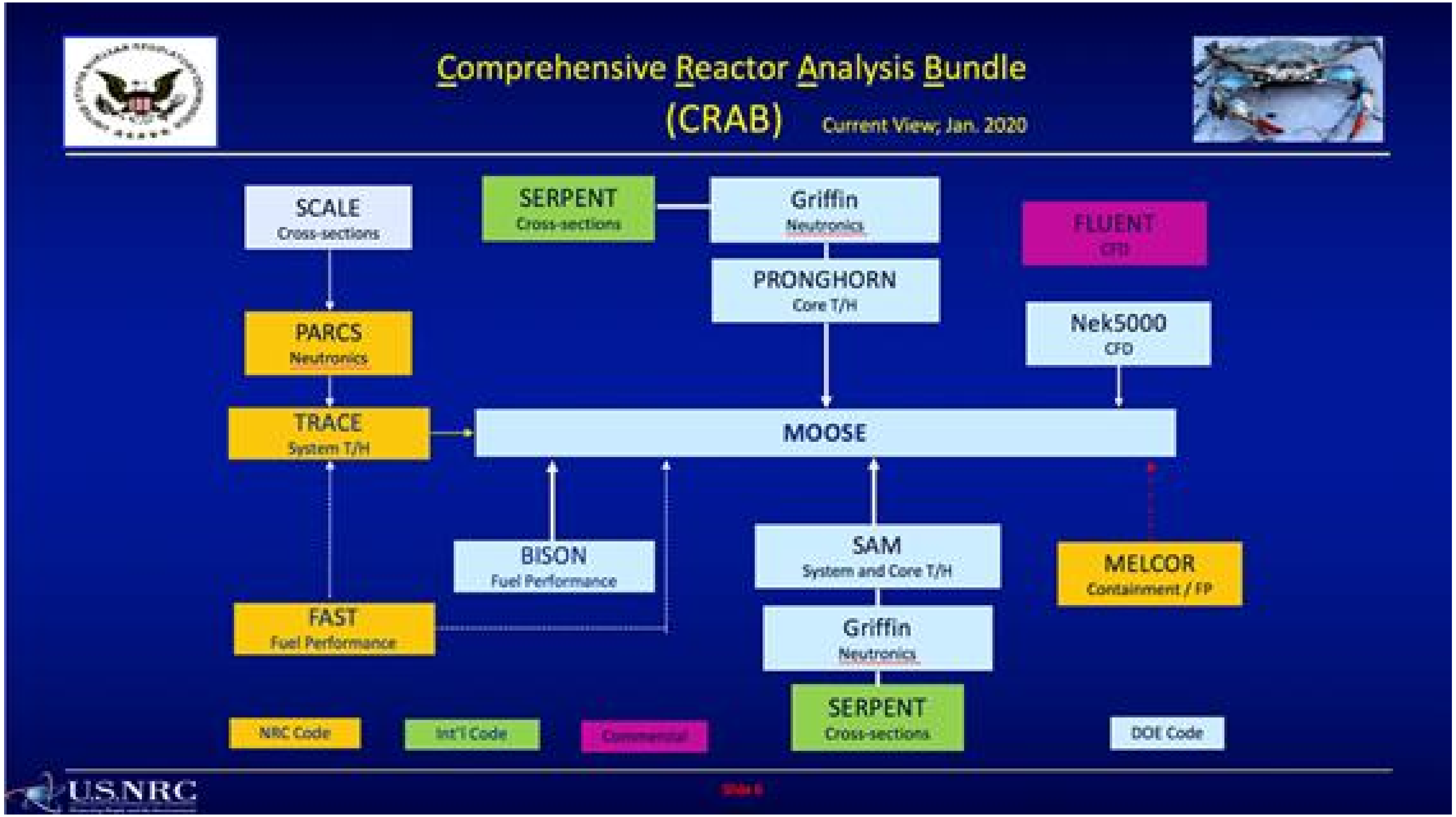
Purpose	Previously Used Codes	Codes for Today and tomorrow	Remark
Cross Section Generation	MICROX MICROR	SCALE/MCNP SERPENT	Slowing-down in graphite, heterogeneity, leakage, control rods
Criticality/Rod Worth Steady State Verification	DIF3D	Monte Carlo	
Steady State Design and Fuel Management	DIF3D/BURP	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 Bed

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	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.

NEAMS/NRC Codes



Uncertainties 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%	-	-
Peach Bottom	-11% to +4%	-6% to +10%	±10%	±0.7%	-	-
HTGR Critical	+6%	+4% to 13%	-	-0.1% to +1.0%	-	-
Fort St. Vrain	-9% to +12%	±10%	±15%	±0.5%	-	-
HTLTR	±8%	-	-	-	-	-
KAHTER	-	DA	-	-0.3% to +6%	±13%	-
DRAGON	DA	-11%	-	-	-	-
HEU/LEU CORES						
AVR	-25%	-5% to +15%	-	±11%	-	-
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.

HTGR Accident Analysis – Overview

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

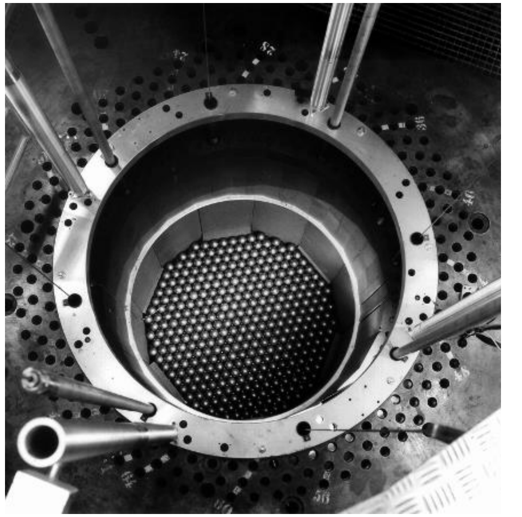
Critical Experiments for Neutronics

PEBBLE BED

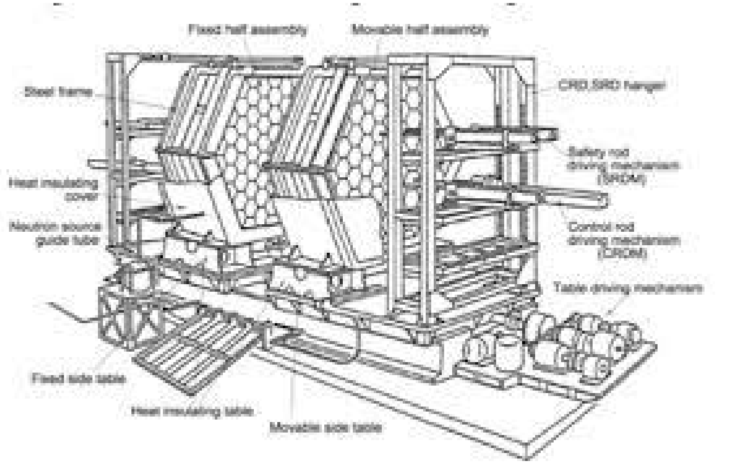
- HTR-Proteus critical experiments
  - 1980's, Paul Scherrer Institute, Switzerland
  - Bess 2014
- HTR-10 Initial Criticality
  - ~2000, INET, China
  - IAEA 2003, 2013
- ASTRA
  - Mid 1990's , Kurchatov Institute, Russia
  - IAEA 2013
- HTR-PM – scheduled to go critical by end of 2021. INET has offered up physics test results to support a GIF benchmark

PRISMATIC

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



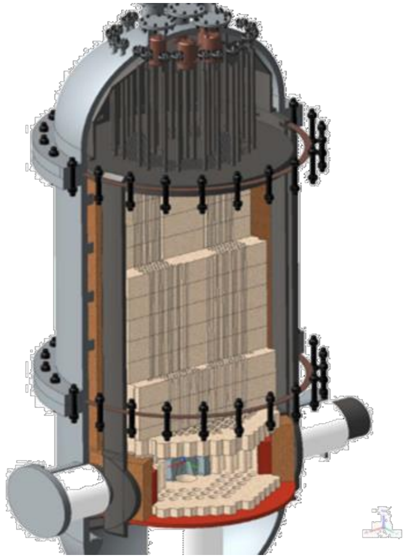
The HTR Proteus experiment from above.

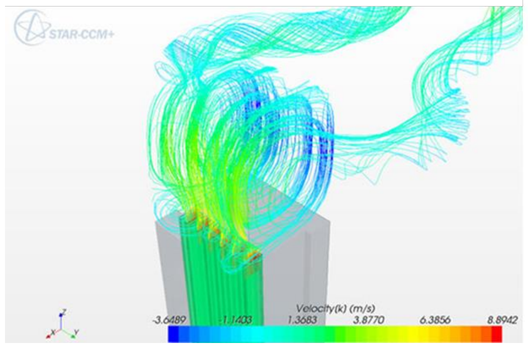


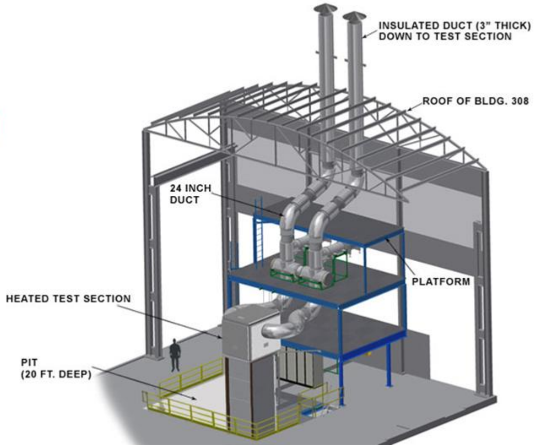
Sketch of the VHTRC Experiment

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.
- The NRC sponsored the design and construction of HTTF
- Framatome and X-Energy facilitated the conversion of NSTF to a water-cooled configuration in 2018.

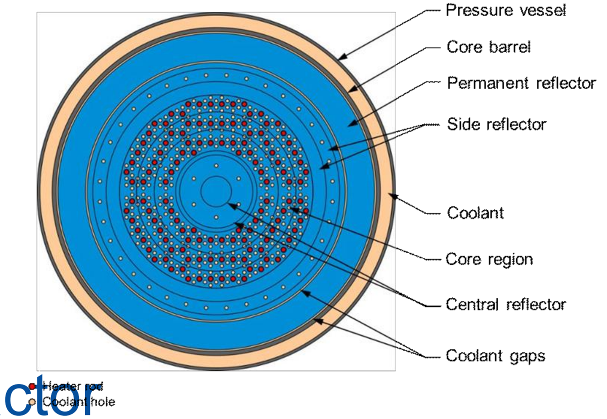


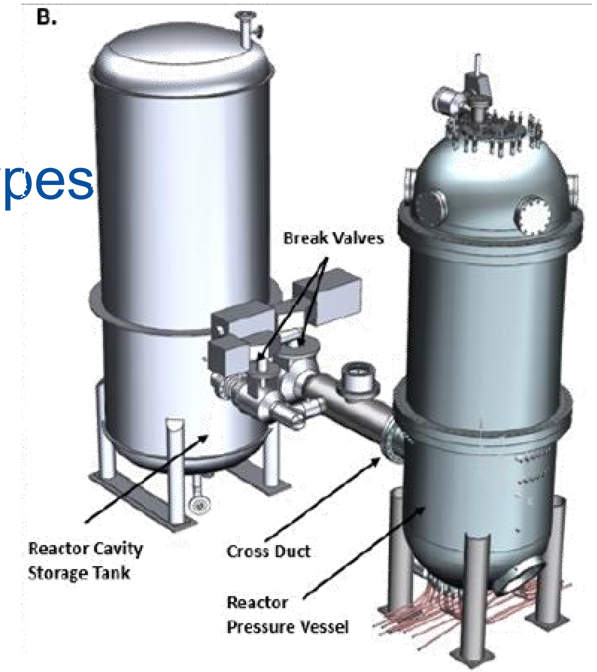




HTTF at Oregon State University

- High Temperature Test Facility (HTTF) at Oregon State University
  - Provide data for system code validation.
  - Primarily designed to model the depressurized conduction cooldown transient.
    - Variety of break size and location.
    - Reactor Cavity Cooling System as boundary condition.
    - Modular design to allow for the examination of different core types
  - Facility Scaling.
    - ¼ length scale. ¼ diameter scale.
    - Reduced pressure.
    - Prototypical temperature.
  - Reference design
    - Modular High Temperature Gas Reactor
  - INL will complete simulation of LOFC test PG-27 with RELAP5-3D by September 2021.
  - ANL also modeling HTTF transients with NEAMS-supported SAM code.*

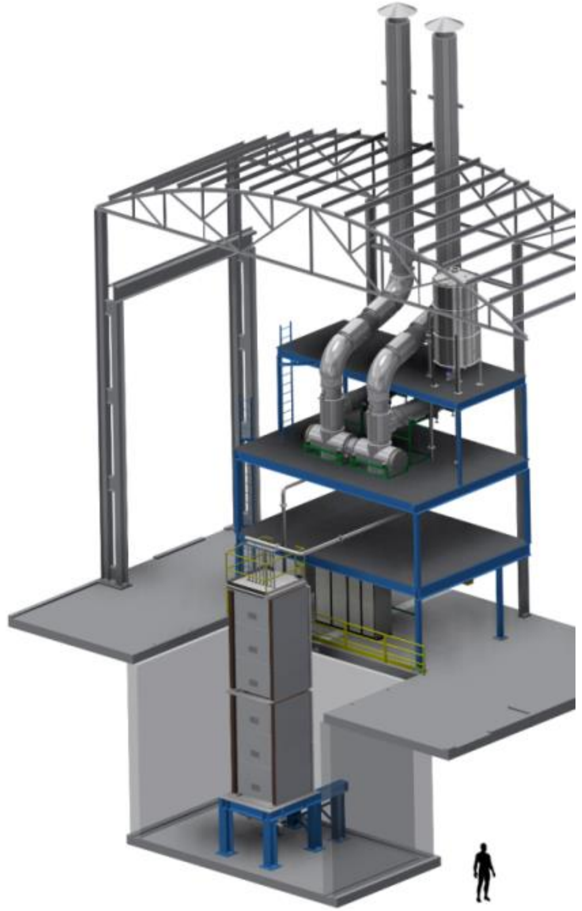




* See forthcoming 2021 ANS paper "*Modeling of HTTF test PG-26 using RELAP5-3D and SAM*", Aaron Epiney, et al.

Natural Convection Shutdown Heat Removal Test Facility (ANL)

- Natural Convection Shutdown Heat Removal Test Facility (NSTF) was initiated in 2010 in support of DOE NGNP, SMR, and ART programs.
  - Air-based testing program (completed, 2013 - 2016)
  - Water-based testing program (on-going, 2018 to present)
- Main objectives of NSTF program:
  - passive safety and decay heat removal for advanced concepts
  - generate NQA-1 qualified licensing data for industry
  - provide benchmark data for code V&V
- Concurrent with a broader scope and multiple collaborators
  - Experimental facilities at scales (½, ¼, etc.) for both air and water
  - Complimenting CFD modeling and 1D systems level analysis
  - Collaborating towards development of a central data base



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

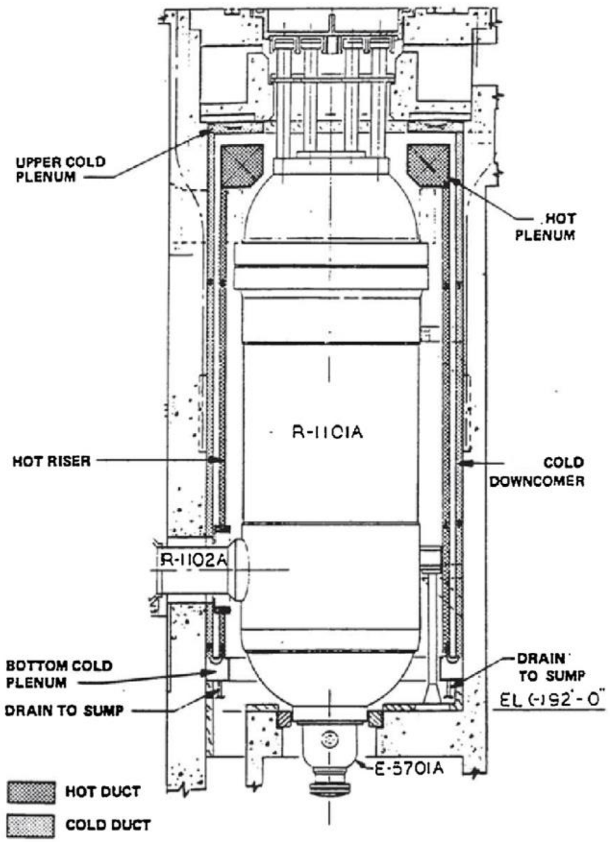
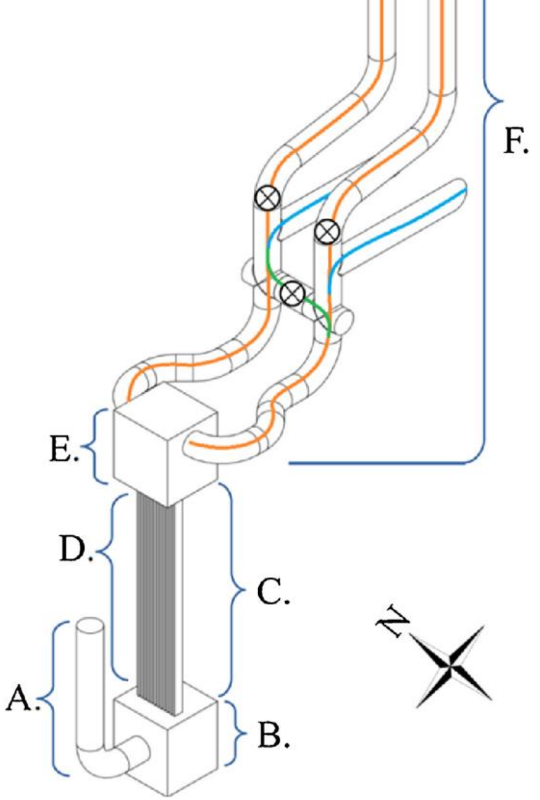


FIGURE 5.5-6
RCCS COOLING PANEL CONFIGURATION – ELEVATION
HIGH TEMPERATURE GAS-COOLED REACTOR PRELIMINARY SAFETY INFORMATION DOCUMENT HTGR-86-024

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.

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)

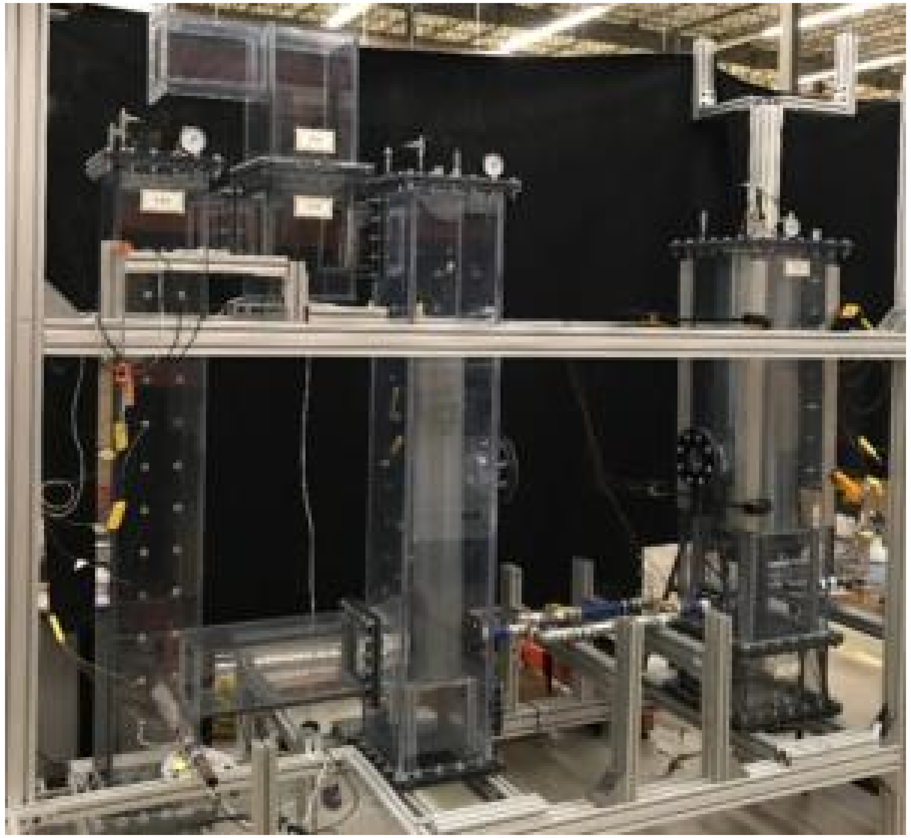
Phenomena Characterized in:

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 pipe break and depressurization
- Initial tests were completed. Further experiments solicited in the 2019 NEUP call (www.neup.inl.gov)



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
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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 large** 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 considered during design or mitigated. Graphite does not burn.
- **Moisture ingress** (steam generator tube rupture) may be the limiting case with respect to fission product release.
- Code systems designed for HTGRs have improved since the first HTGRs were licensed. Computational power is driving **more extensive use of high-fidelity tools**. Large margins, however, still allow **approximate methods** to be used effectively.

Summary (2)

- Uncertainties can be **significant and time-dependent.** These can mostly be attributed to uncertainties in material properties, thermal fluids, and neutron cross-sections.
- HTGR 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 IET experiments have been conducted.
- Safety Analysis must factor individual design features, but the general approach applies to all modular HTGRs. “Cliff-edges” 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

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