Modular High Temperature Gas-cooled Reactor: Accident Analysis

April 2021

Jason Albert Christensen
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Modular High Temperature Gas-cooled Reactor: Accident Analysis

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Modular High Temperature Gas-cooled Reactor: Accident Analysis

CNSC Seminar
HTGR Accident Analysis – Overview

- Types of Potential Accidents and Reactor Response
- Codes and Tools
- Experimental Validation
- Safety Analysis Approach
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.
Depressurized Loss of Forced Cooling (DLOFC)

aka Depressurized Conduction Cooldown

- 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.

- 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.
DLOFC vs PLOFC: HTR-PM

DLOFC (65 mm pipe break): Total loss of coolant within 60 s. Peak fuel temperature reaches ~1480°C @ 30 hours.

PLOFC (turbine trip): Core pressure settles at ~6 MPa. Peak fuel temperature reaches ~1140°C @ 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.

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!

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

Air Ingress (2)

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

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

HTGR Accident Analysis – Overview

• Types of Potential Accidents and Reactor Response
• Codes and Tools
• Experimental Validation
• Safety Analysis Approach
# Codes and Methods used for Past and Current HTGR Analysis – Prismatic

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

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

<table>
<thead>
<tr>
<th>Facility</th>
<th>Temp. Defect</th>
<th>C. R. Worth</th>
<th>Power Distr.</th>
<th>$K_{\text{eff}}$</th>
<th>Water Ingress</th>
<th>Decay Heat</th>
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<tbody>
<tr>
<td><strong>HEU-CORES</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>Peach Bottom Critical</td>
<td>±14%</td>
<td>-11%</td>
<td>±10%</td>
<td>±0.7%</td>
<td>-</td>
<td>-</td>
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<tr>
<td>Peach Bottom</td>
<td>-11% to +4%</td>
<td>-6% to +10%</td>
<td>±10%</td>
<td>±0.7%</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>HTGR Critical</td>
<td>+6%</td>
<td>+4% to 13%</td>
<td>-</td>
<td>-0.1% to +1.0%</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Fort St. Vrain</td>
<td>-9% to +12%</td>
<td>±10%</td>
<td>±15%</td>
<td>±0.5%</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td><strong>HTLTR</strong></td>
<td>±8%</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
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<tr>
<td><strong>KAHTER</strong></td>
<td>-</td>
<td>DA</td>
<td>-</td>
<td>-0.3% to +6%</td>
<td>±13%</td>
<td>-</td>
</tr>
<tr>
<td>DRAGON</td>
<td>DA</td>
<td>-11%</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
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<tr>
<td><strong>HEU/LEU CORES</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>AVR</td>
<td>-25%</td>
<td>-5% to +15%</td>
<td>-</td>
<td>±11%</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td><strong>LEU CORES</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>HITREX-2</td>
<td>-</td>
<td>-</td>
<td>±10%</td>
<td>±0.5%</td>
<td>-</td>
<td></td>
</tr>
<tr>
<td>HITREX-2</td>
<td>-</td>
<td>-</td>
<td>±10%</td>
<td>±0.5%</td>
<td>-</td>
<td></td>
</tr>
</tbody>
</table>

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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
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.
• 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.*

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

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

Phenomena Characterized in:


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)
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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

Suggested Reading List

- 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
- 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