HTGR Technology Course for the Nuclear Regulatory Commission

May 24 – 27, 2010

Module 15

HTGR Accident Analysis Tools

Outline

- **Accident analysis codes, modeling and phenomena**
- **Applications of computational fluid dynamics (CFD) modeling**
- **Accident simulation**
  - Depressurization (DLOFC)
  - Pressurized loss of forced cooling (PLOFC)
  - ATWS and other reactivity accidents
  - Air and water ingress
- **Uncertainty analysis and sensitivity studies**
- **Code benchmarking, verification, and validation**

Thermal Methods Used for Prismatic Core Heat Removal

- **SINDA/FLUINT**
  - 3D thermal/fluid network for pressurized and depressurized conduction cooldown (PLOFC and DLOFC)
- **TAC2D**
  - R-Z finite element model for depressurized conduction cooldown (DLOFC) and uncertainty/sensitivity analysis

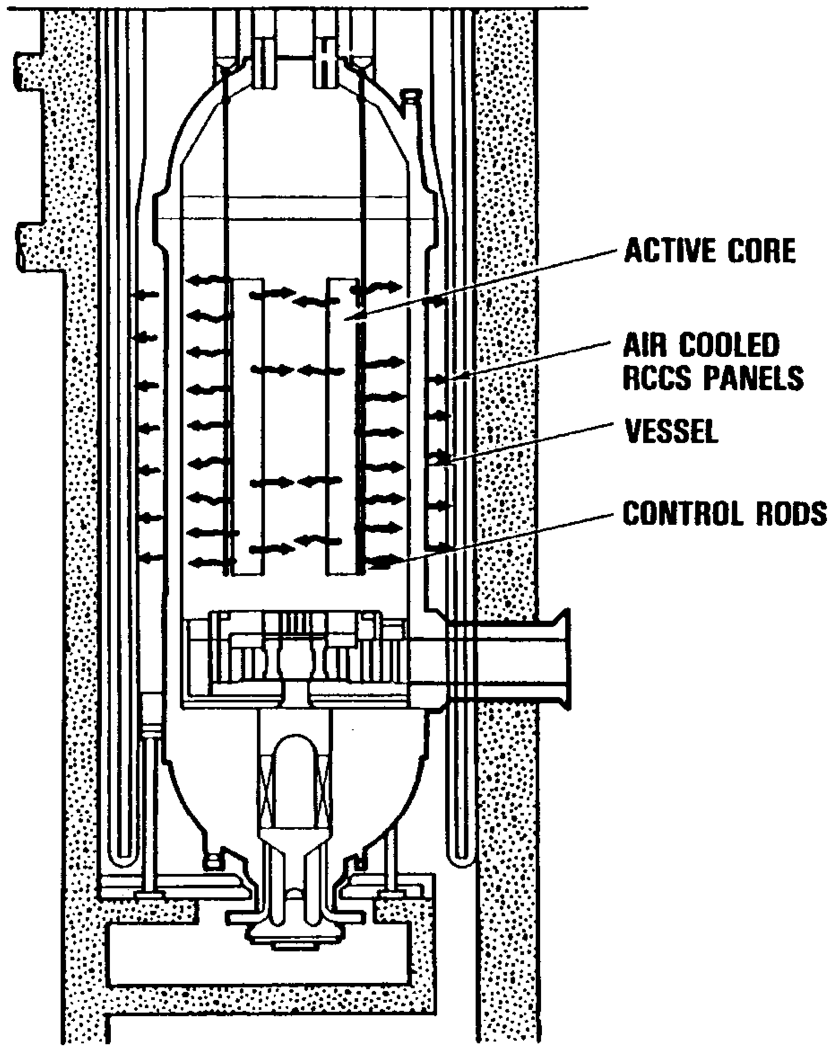
Thermal Hydraulic Methods Used for Pebble Bed Core Heat Removal

- **VSOP99 –** burnup and isotopic distribution for thermal analysis
- **FLOWNEX –** flow network with heat transfer and reactor kinetics
- **TINTE –** detailed thermal analysis and neutronics
  - PLOFC
  - DLOFC
- **STAR CD –** computational fluid dynamics

Heat Removal During Depressurized Conduction Cooldown (DLOFC)

HEAT REMOVED BY:

- CORE CONDUCTION
- CORE INTERNAL RADIATION
- VESSEL RADIATION
- RCCS CONVECTION



Heat Removal During Pressurized Conduction Cooldown (PLOFC)

HEAT REMOVED BY:

- CORE CONVECTION
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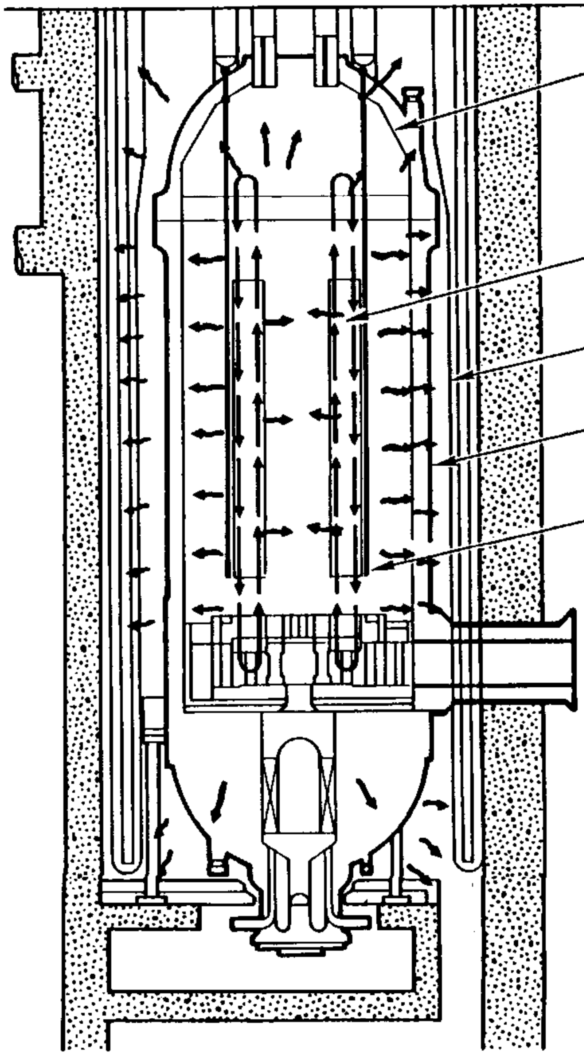
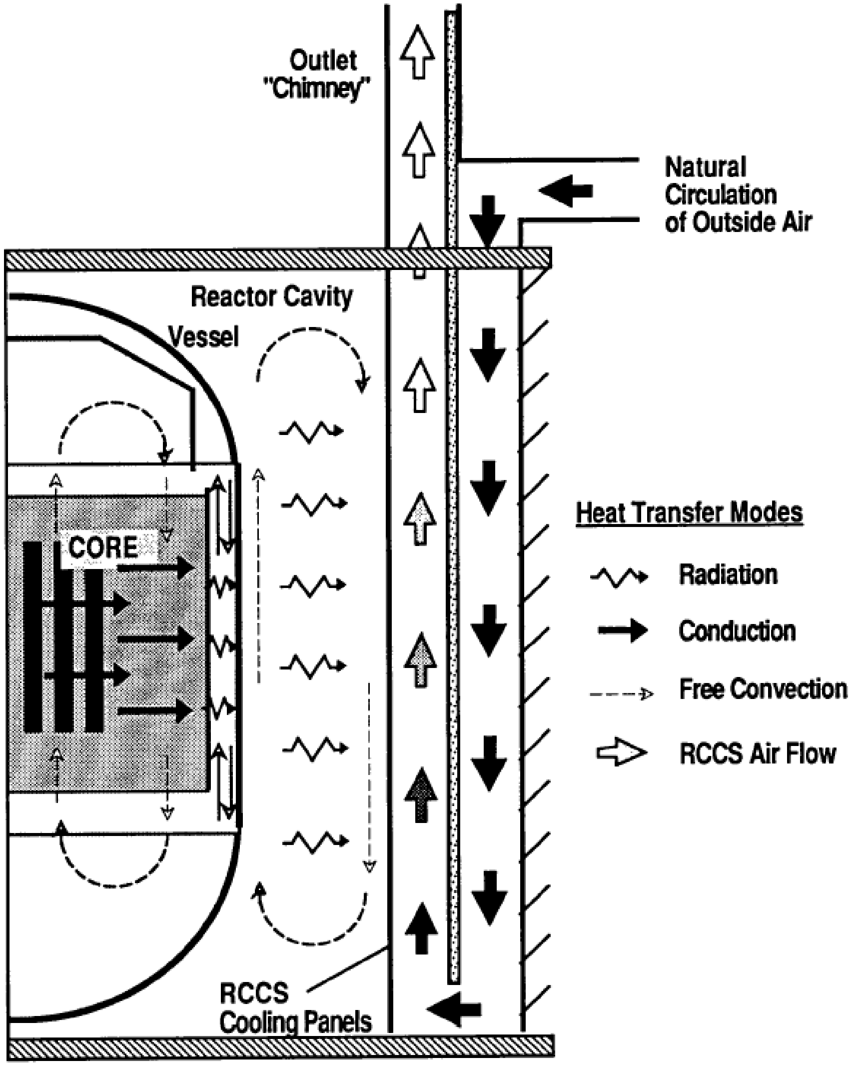
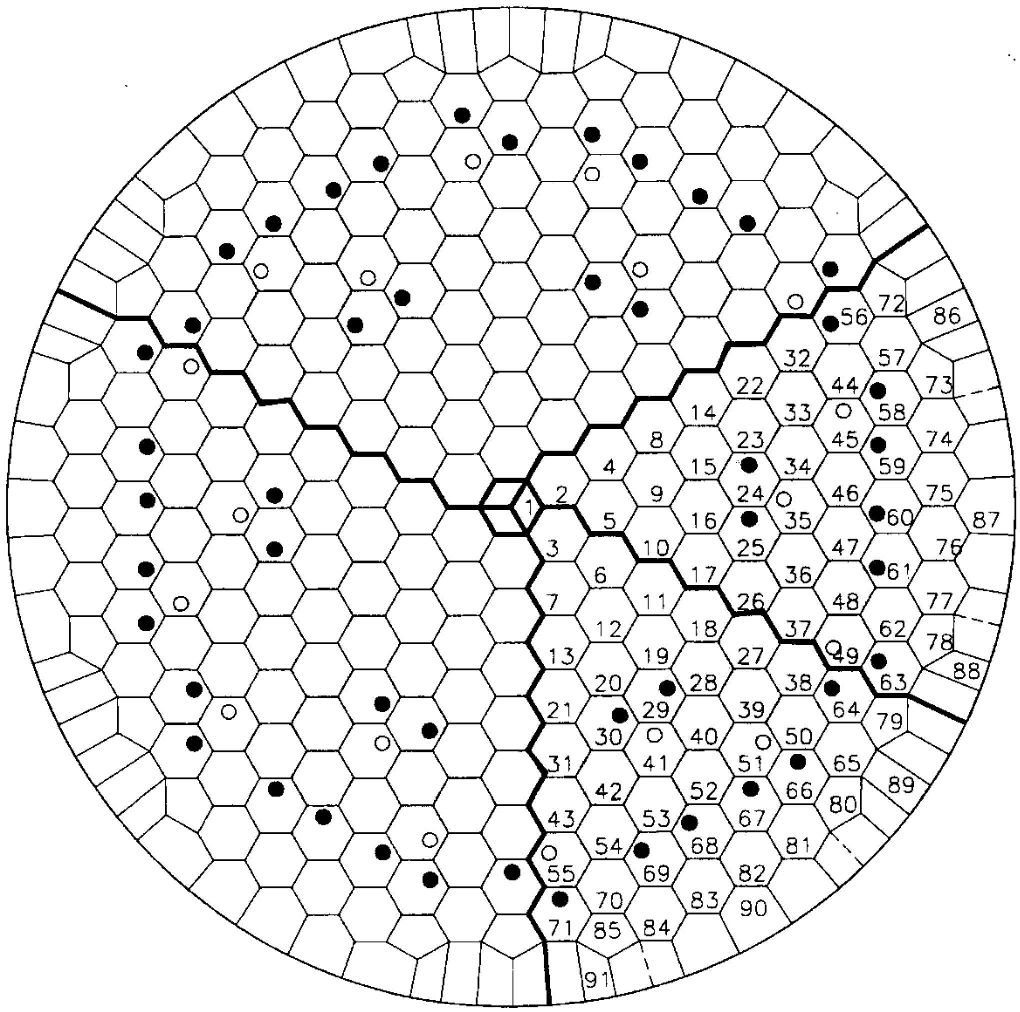
- CORE INTERNAL RADIATION
- VESSEL RADIATION
- RCCS CONVECTION



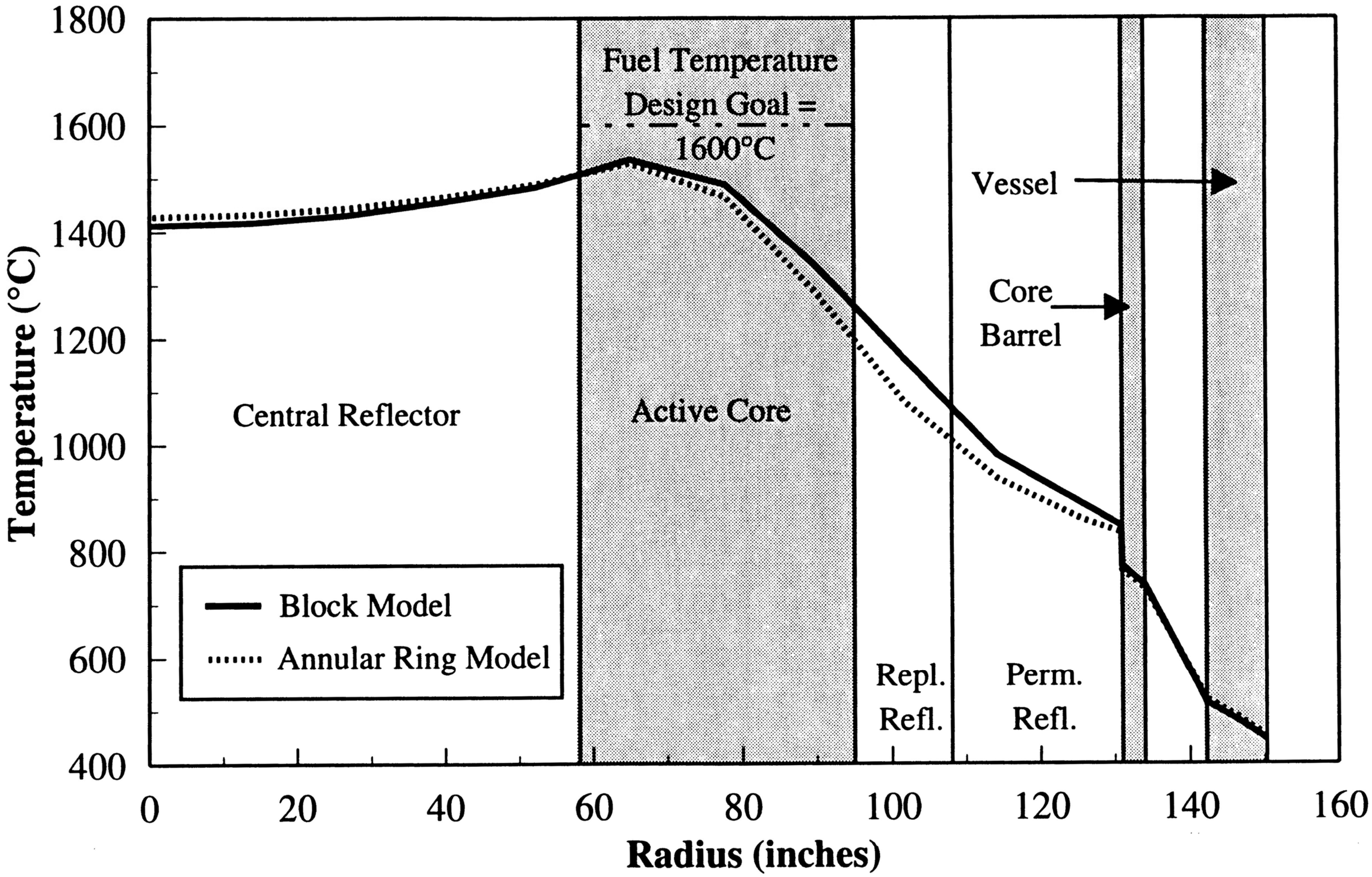
Heat Transfer to the RCCS



One-Third Core SINDA/FLUINT Model



Comparison of Radial Temperature Distributions from Two Models During DLOFC



PBMR Flownex Model

- **Flownex model includes**
  - Fluid volume and inventory
  - Metal mass for thermal capacitance
  - Area available for heat transfer
  - Reactor modeled using point kinetics
  - Pressure drop using loss factors or friction factor correlations
  - Heat transfer based on Nusselt number correlations

PBMR TINTE Code

- **TINTE (Time Dependent Neutronics and Temperatures) used to model time-dependent transients**
- **Events can be slow (DLOFC) or fast (control rod withdrawal)**
- **TINTE provides transient temperatures for design as well as fission / total power**
- **Fuel element burn-up and isotopic distribution supplied by VSOP99**

Fuel Performance Methods Used for Prismatic and Pebble Bed Safety Analysis

- SORS/NP1 calculates fuel particle coating performance and radionuclide release using temperature results from SINDA/FLUINT or TAC2D for prismatic HTGR
- GETTER / NOBLEG calculates transient radionuclide release using temperature results from TINTE for pebble bed HTGR

SORS Fuel Particle Performance Models for Prismatic HTGR

- **Pressure vessel failure models for:**
  - Standard intact particles
  - Particles with failed OPyC layer
  - Particles with missing buffer layer
- **SiC failure by:**
  - Kernel migration
  - Heavy metal dispersion
  - Fission product corrosion
  - Thermal decomposition
- **OPyC failure by fast neutron irradiation**

SORS Radionuclide Transport From Fuel for Prismatic HTGR

- Release model for volatile radionuclides from exposed fuel kernels
- Diffusion model for metallic radionuclides from fuel kernels
- Diffusion model for radionuclides through SiC and OPyC layers
- Release model for heavy metal contamination

SORS Graphite – Coolant Radionuclide Transport Models for Prismatic HTGR

- Graphite – coolant transport based on vapor pressure / concentration equations
- Diffusion of radionuclides into reflector graphite
- Diffusion in active core graphite is ignored

NOBLEG Code for Pebble Bed HTGR

- NOBLEG calculates steady state radionuclide release to solve short lived gaseous radionuclide diffusion behavior under normal operating conditions
- NOBLEG contains thermal hydraulic and mass transfer subroutines to determine temperatures and radionuclide production and transport in spherical fuel
- NOBLEG has been extensively verified and validated with German irradiation test data

GETTER Code for Pebble Bed HTGR

- GETTER predicts long-lived metallic radionuclide behavior under normal conditions and metallic and halogen radionuclide release during temperature transients
- GETTER contains neutronic, thermal hydraulic and mass transfer subroutines to determine burn-up, temperatures and radionuclide production and transport in spherical fuel
- GETTER has been extensively verified and validated with German irradiation test data

Pressure Boundary and Reactor Building Radionuclide Transport Codes

- **Past prismatic HTGR assessments at GA used simplified models in TDAC and POLO**
  - Depressurization, hydrostatic displacement, thermal expansion and contraction
  - Liftoff, washoff, steam-induced vaporization of plated-out radionuclides
  - Venting of reactor building along with gravitational settling and plateout
- **HTGR version of MELCOR expected to be used in future radionuclide transport assessments**
- **ASTEC used by PBMR to calculate radionuclide releases from reactor building**

ASTEC
Accident Source Term Evaluation Code

- **ASTEC simulates all phenomena during severe accident in LWR from initiating event to release of radionuclides from reactor building**
- **ASTEC is a multi-module, integral code similar to MELCOR**
- **ASTEC can model the following aerosol and fission product behavior:**
  - Coagulation, thermophoresis and diffusiophoresis
  - Filters
  - Steam condensation onto aerosols
  - Washing
  - Aerosols removal by spray

Methods for Analyzing Reactivity Transients

- **BLOOST code developed by GA and RELAP5 code developed by NRC contractors**
  - Both use point kinetics model and fuel and moderator temperature reactivity feedback
- **VSOP99 and TINTE codes used for pebble bed reactivity transients**
- **3-D kinetics may not be needed due to longer neutron migration distances**

Oxidation Programs for Prismatic HTGRs

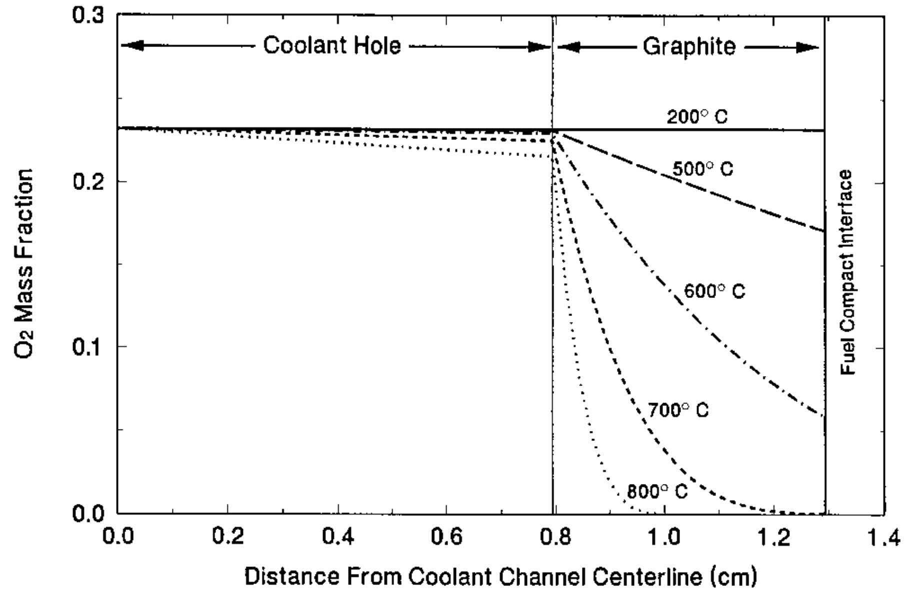
- **OXIDE-4 code used for graphite oxidation due to air and steam**
- **AIP – Air Ingress Program – used to model graphite oxidation from air**
  - Models O_2 and CO_2 reactions with graphite
  - Models CO combustion in flow channel
  - Models natural convection flows by balancing buoyancy against frictional losses
- **ANSYS and GRACE codes used by Fuji Electric for HTTR**
- **GAMMA+ code developed by KAERI**

Progression of Air Ingress Events

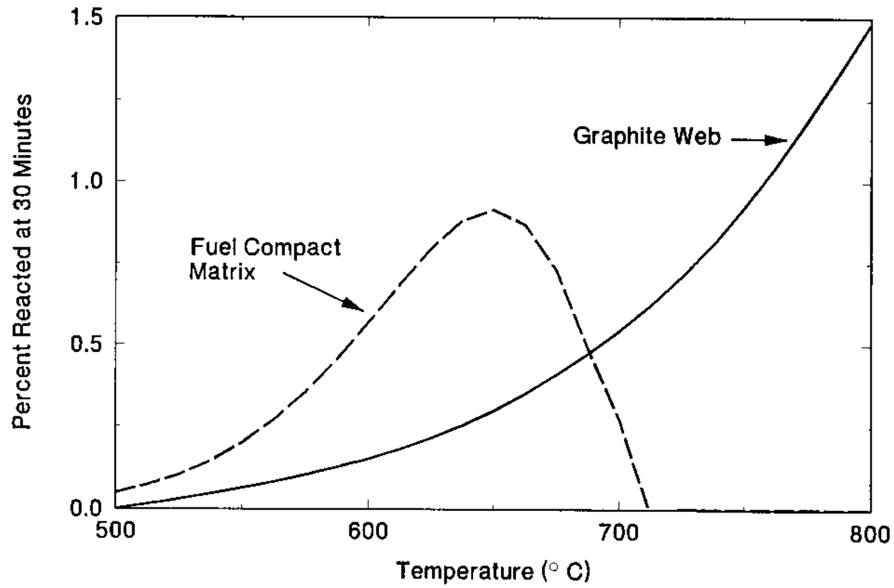
- **Overall oxidation rate determined by rate of air supply**
  - Friction between core and fluid greatly limits flow rate
  - Flow rate further limited as core heats up because viscosity increases with temperature
  - Core cool down reduces oxidation to negligible level
  - Graphite mass loss is a few percent at most and limited to lower plenum and reflectors
- **Radioactivity released by graphite oxidation is small**
  - Relatively low levels of radioactivity in graphite
  - Radiological consequences only marginally greater than conduction cooldown w/o air ingress

Typical Oxidation Behavior in Prismatic HTGRs

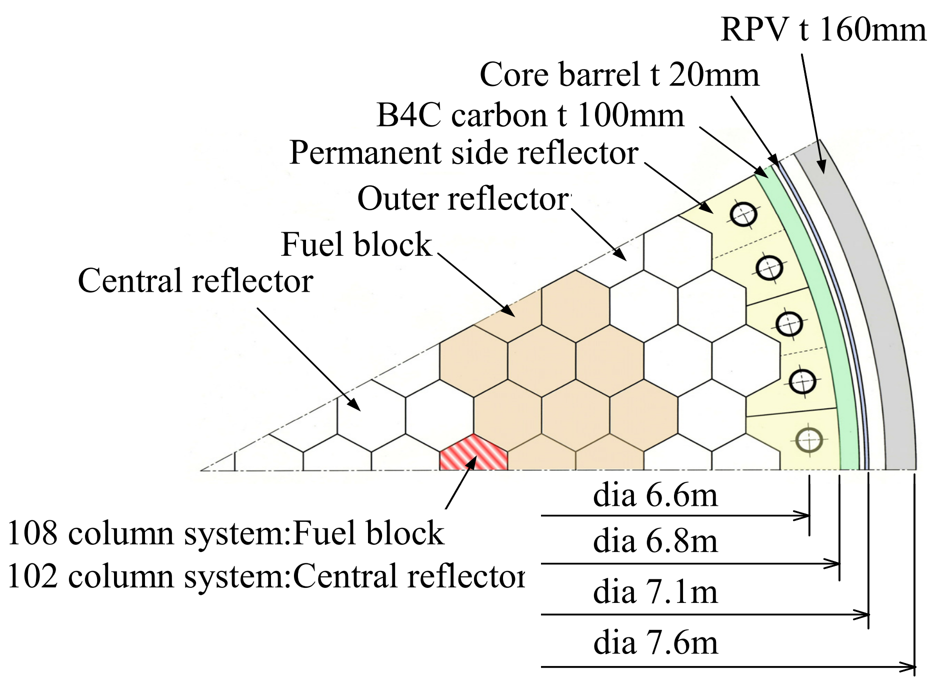
Oxygen Profile



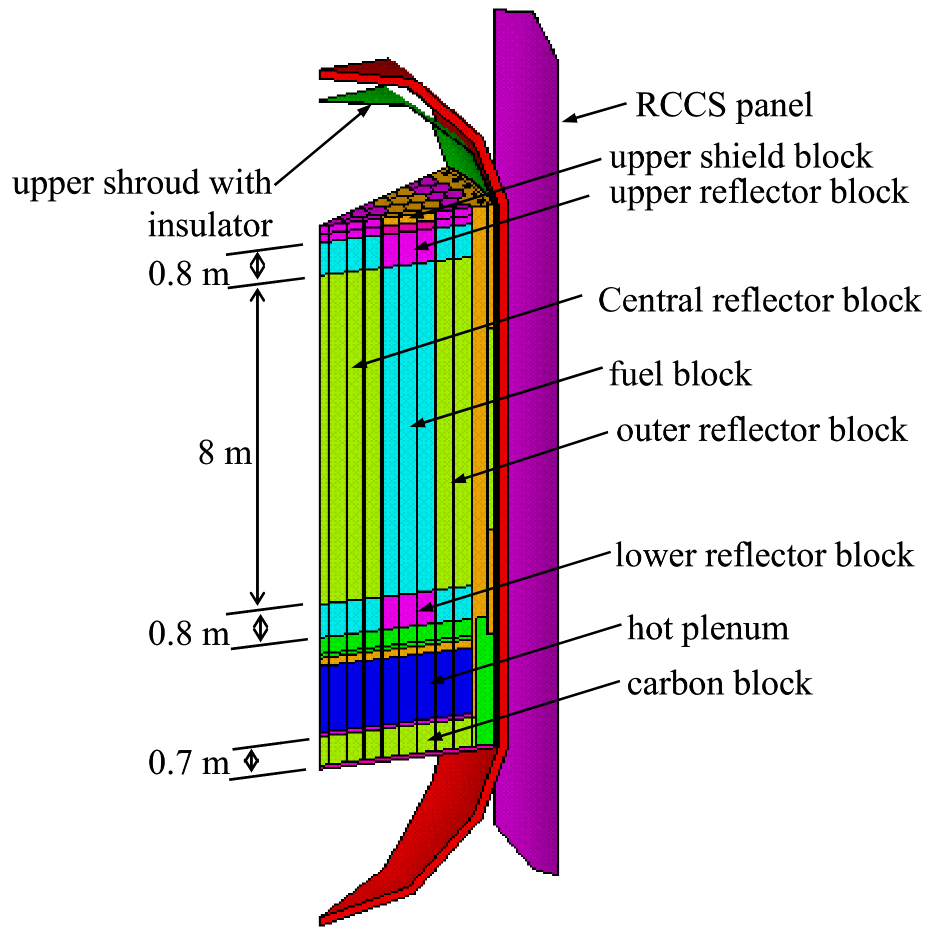
Reaction with Graphite and Compact



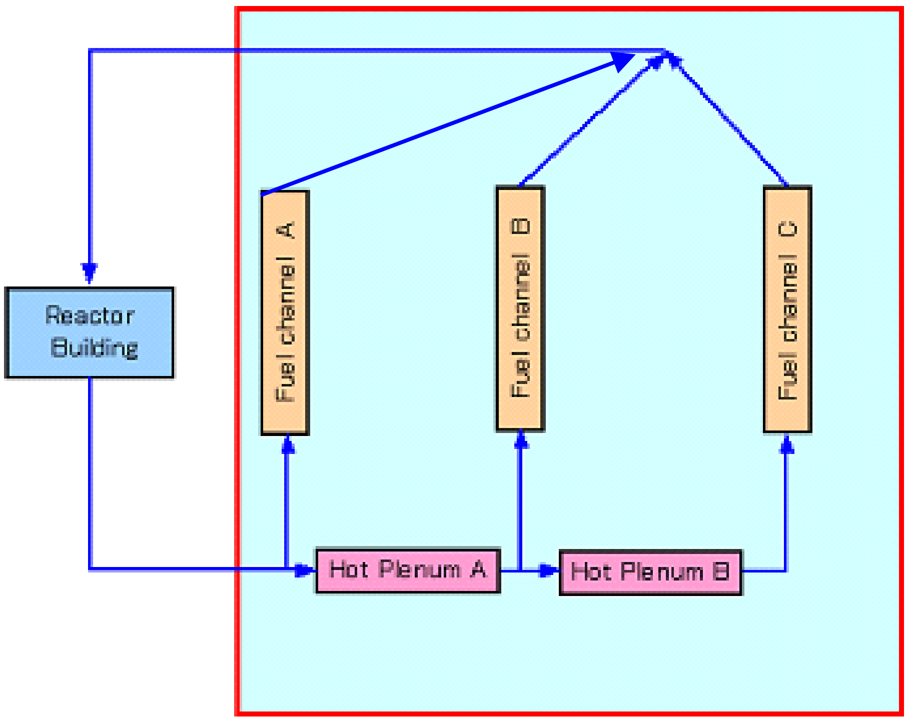
ANSYS Model Used to Simulate Air Ingress



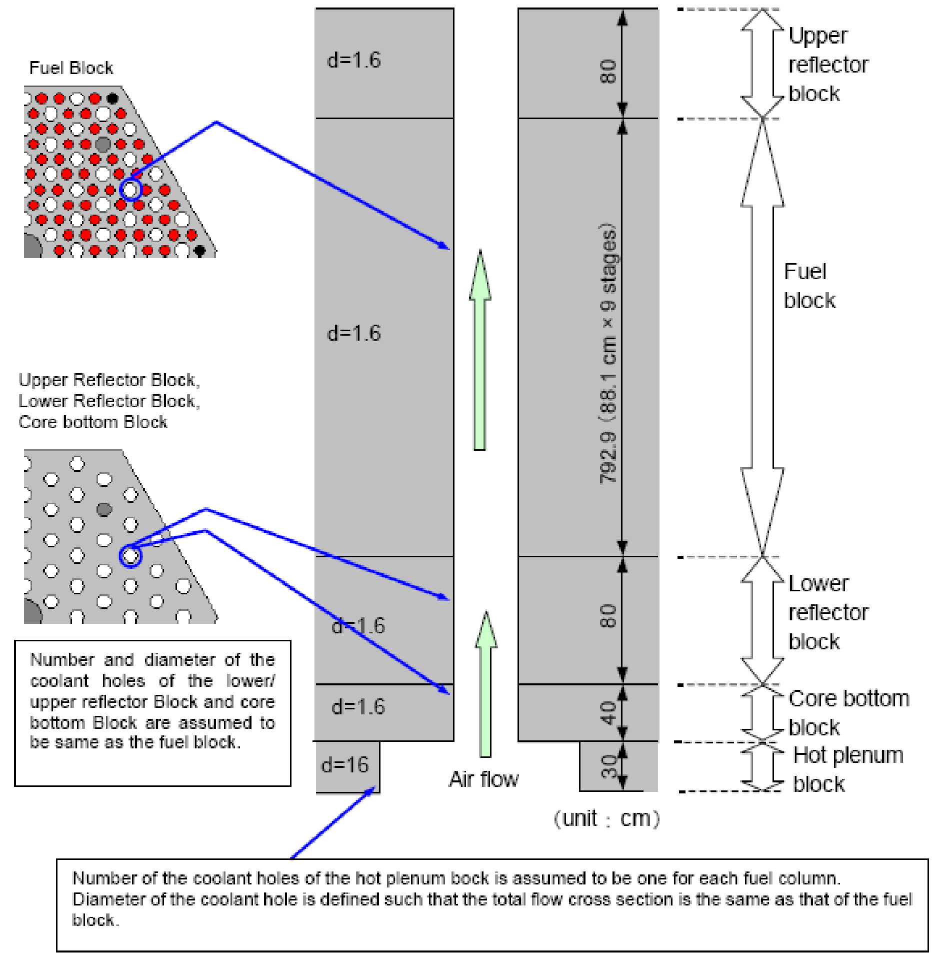
Horizontal view of 30 deg. Sector model



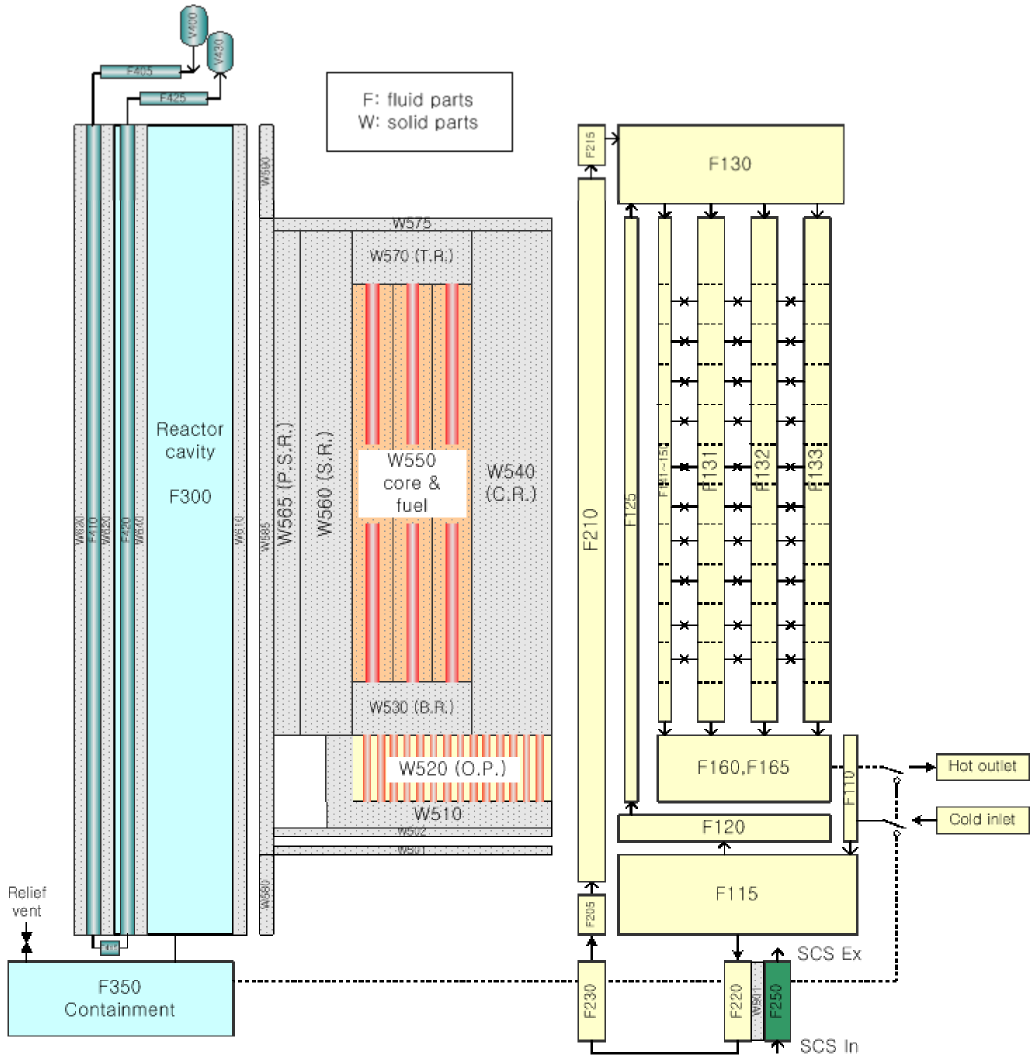
GRACE Model Used to Simulate Air Ingress



Fuel Channel Model



GAMMA+ Model to Simulate Air Ingress



Water Ingress Analysis Methods

- **OXIDE-4 code used to model graphite oxidation by air and steam in prismatic HTGR**
- **FLOWNEX and TINTE used to model water ingress in pebble bed HTGR**
- **Water-graphite reaction**
  - Endothermic producing H_2 and CO
  - Requires temperatures >700°C
  - Slow reaction rate

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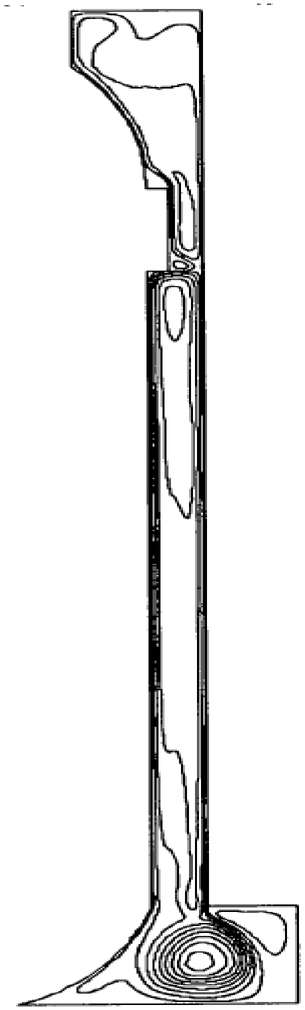
CFD Applications of Interest to Designers

- **Validate engineering assumptions**
- **Assess mixing and flow distribution**
- **Assess gap and cross flows**
- **Assess natural circulation**

HTGR Specific Candidate CFD Applications

- Lower plenum mixing during normal operation
- Flow distribution from cold duct to upper plenum
- Core gap flow and cross flow
- Natural circulation in reactor cavity
- Natural circulation in RCCS
- Natural circulation within reactor vessel
- Startup of shutdown cooling and transition from natural circulation to forced convection cooling
- Air and water ingress

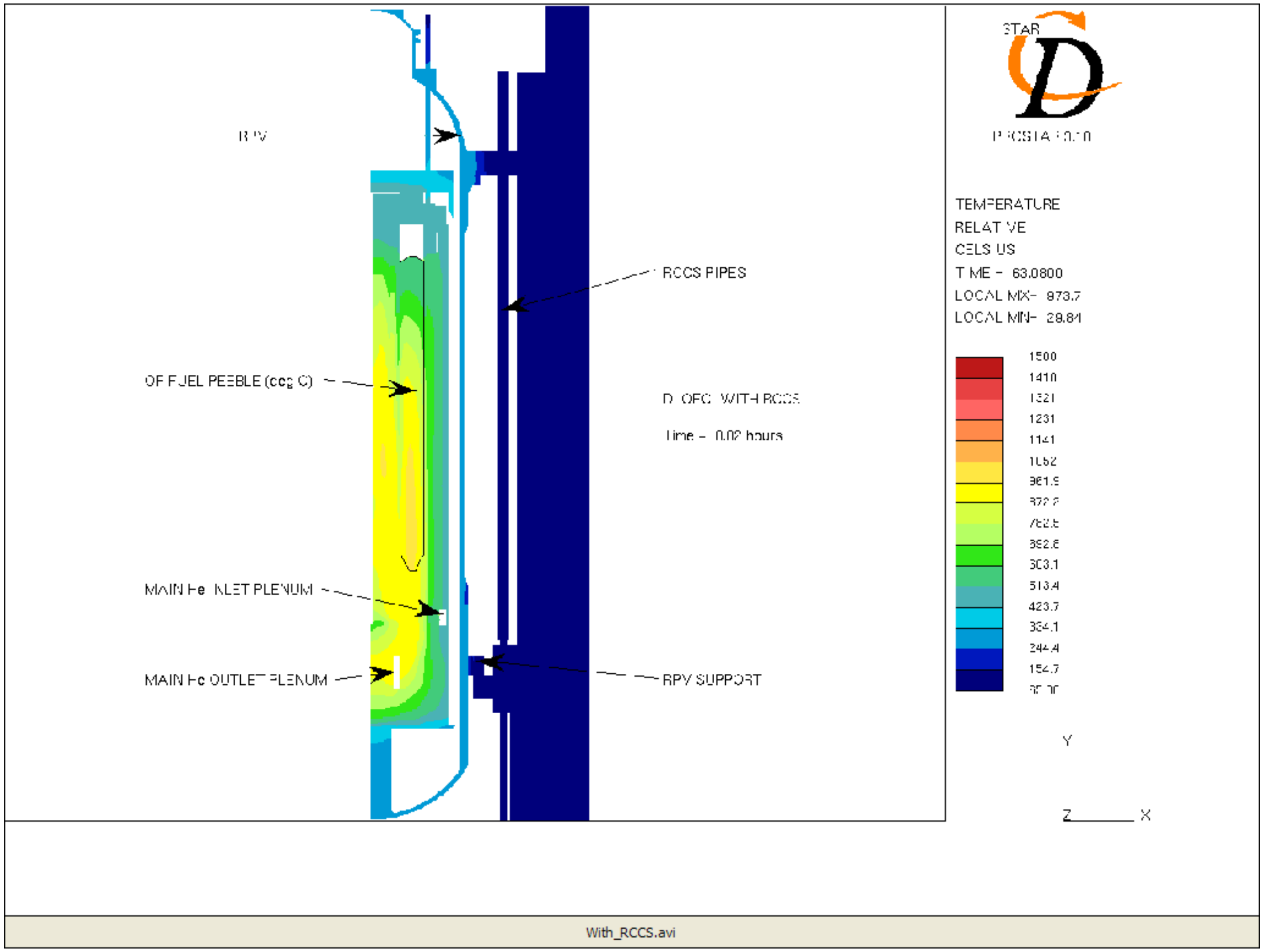
Typical Reactor Cavity Natural Circulation Flow Field (MHTGR)



Depicts localized natural circulation cells

PBMR CFD Analysis Capabilities

- **D-LOFC with STAR CD**



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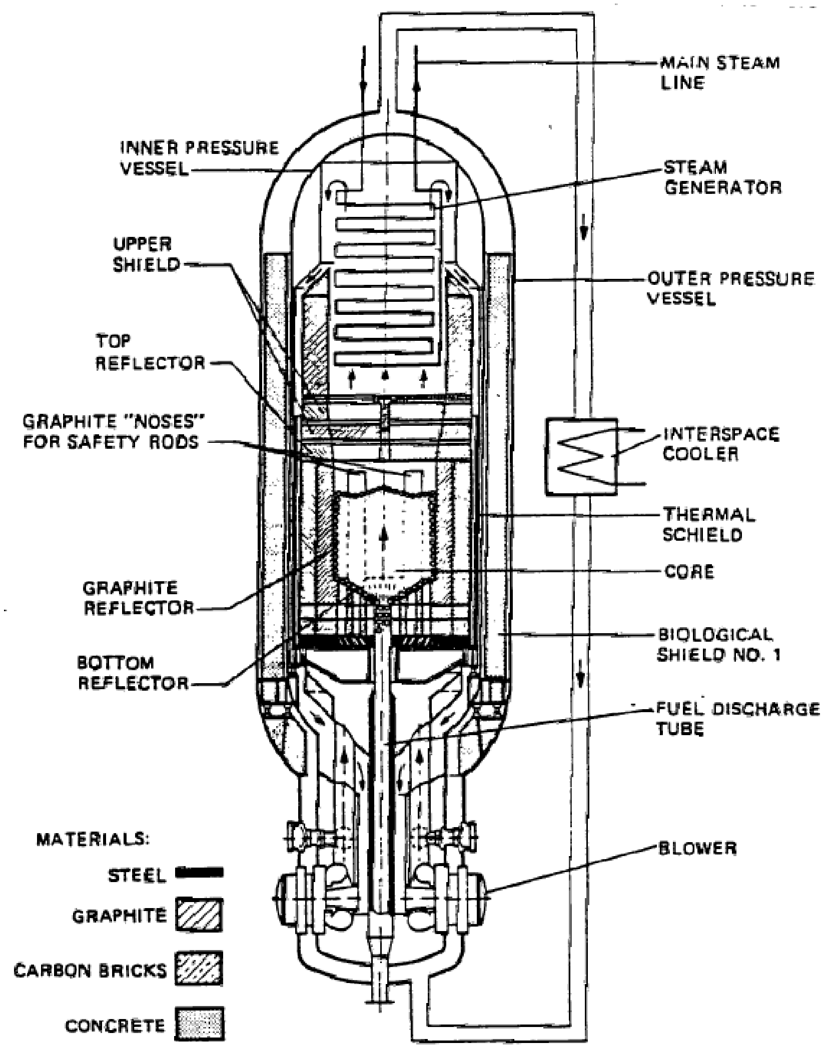
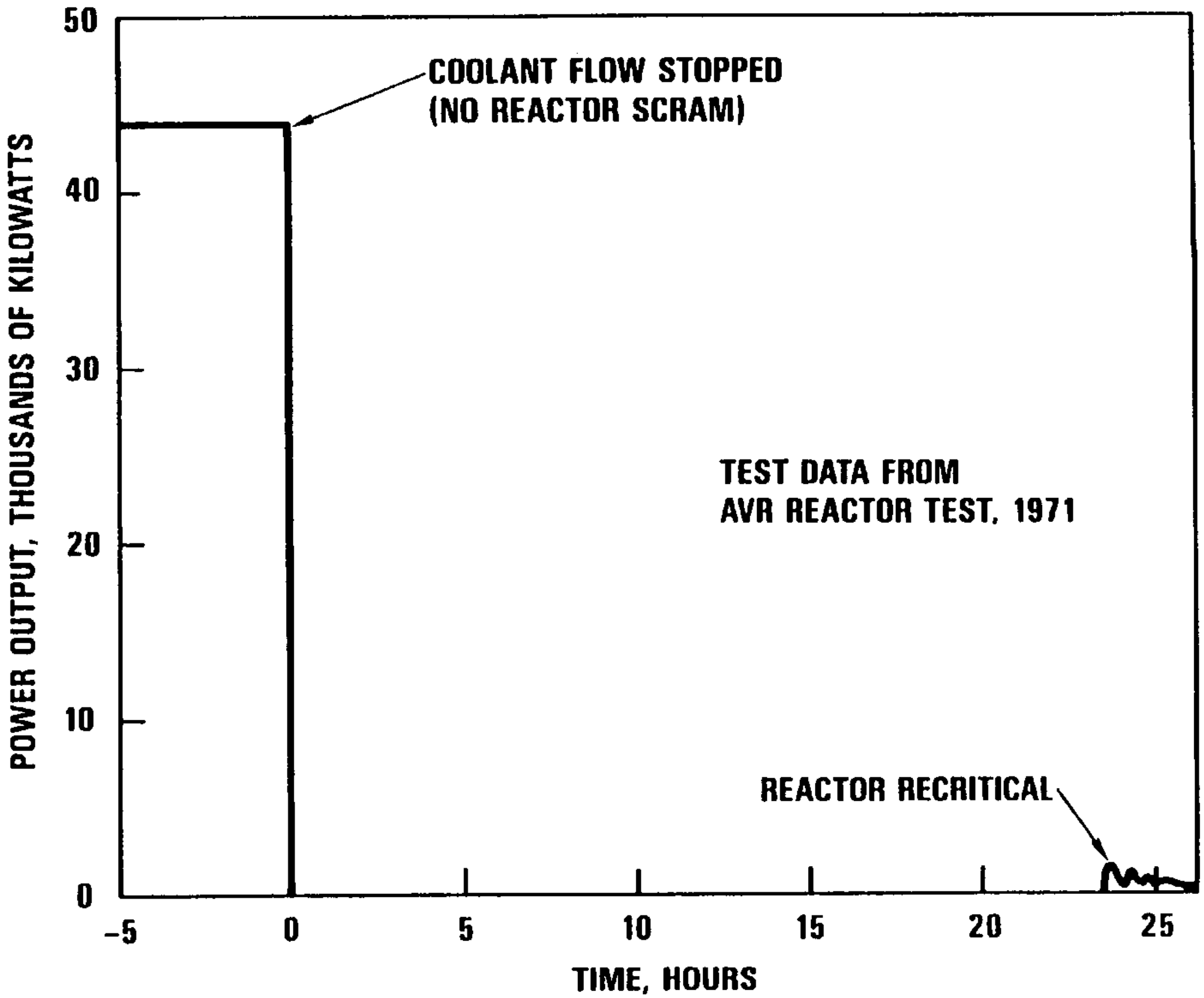
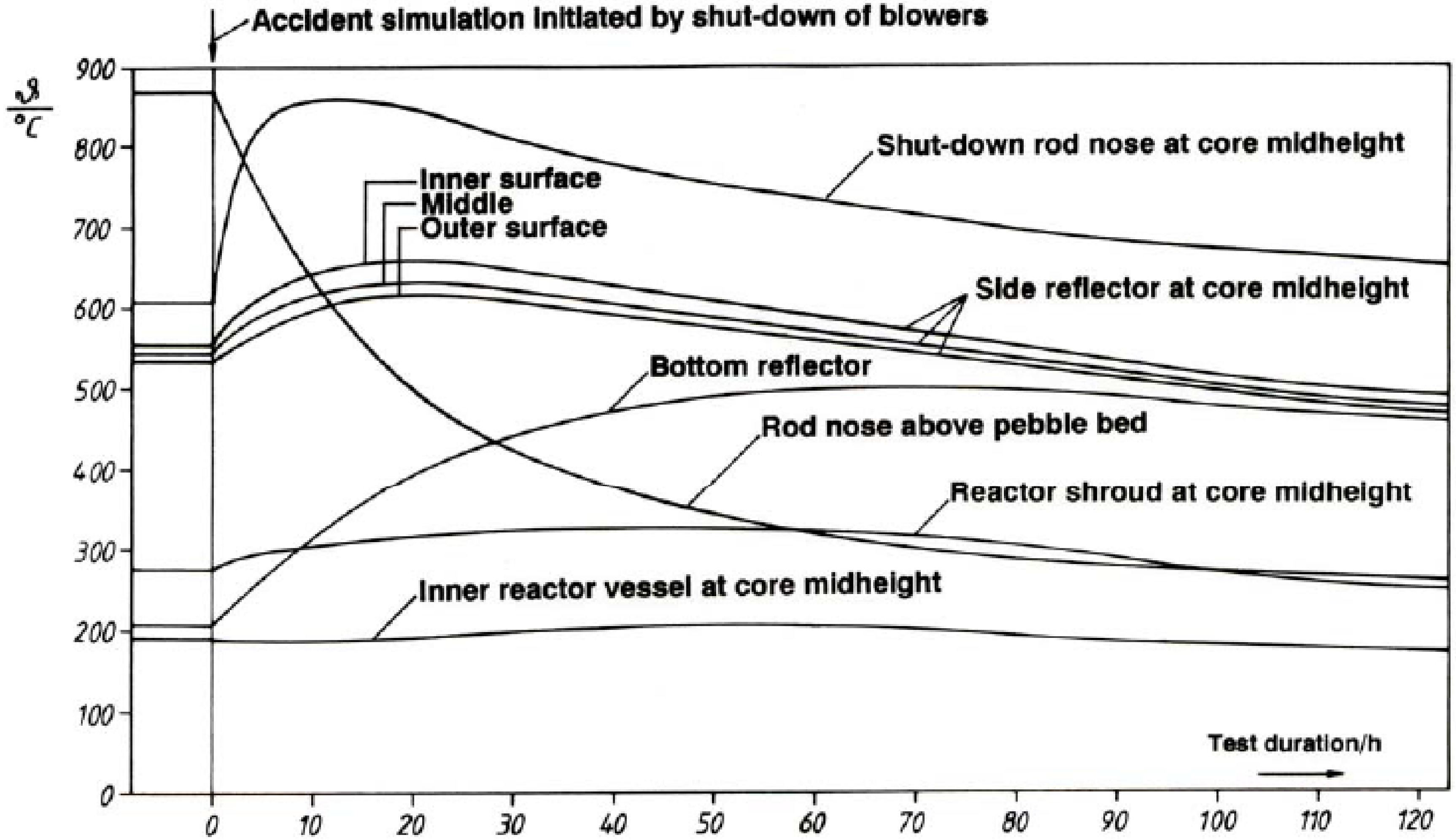
German AVR Arrangement



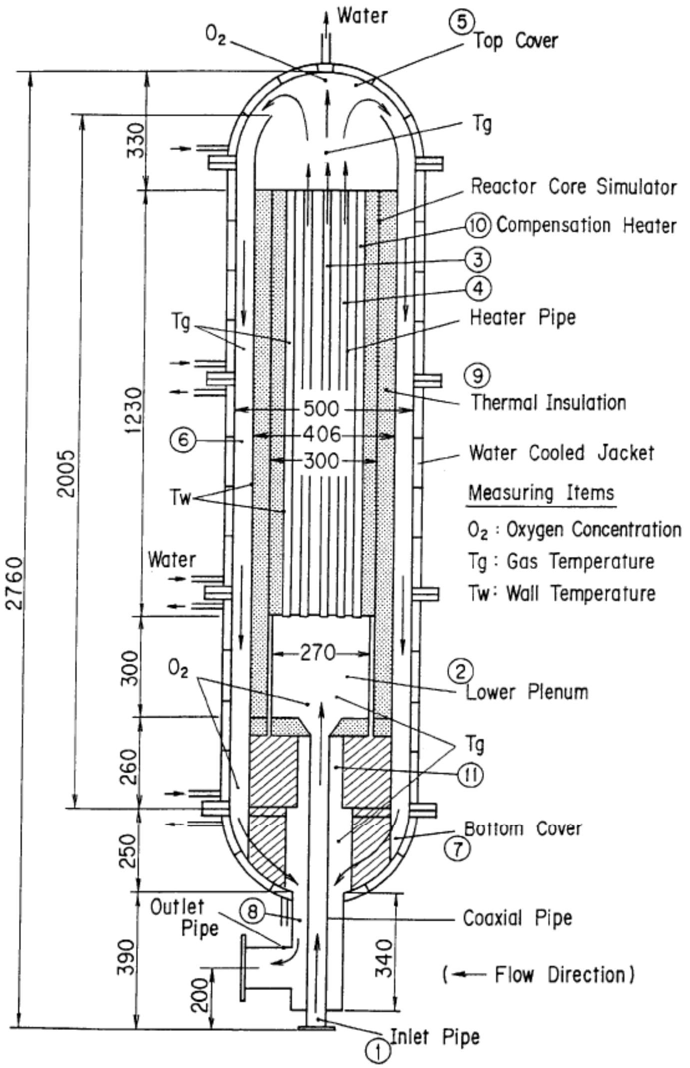
ATWS Test in German AVR Demonstrated Termination of Nuclear Reaction



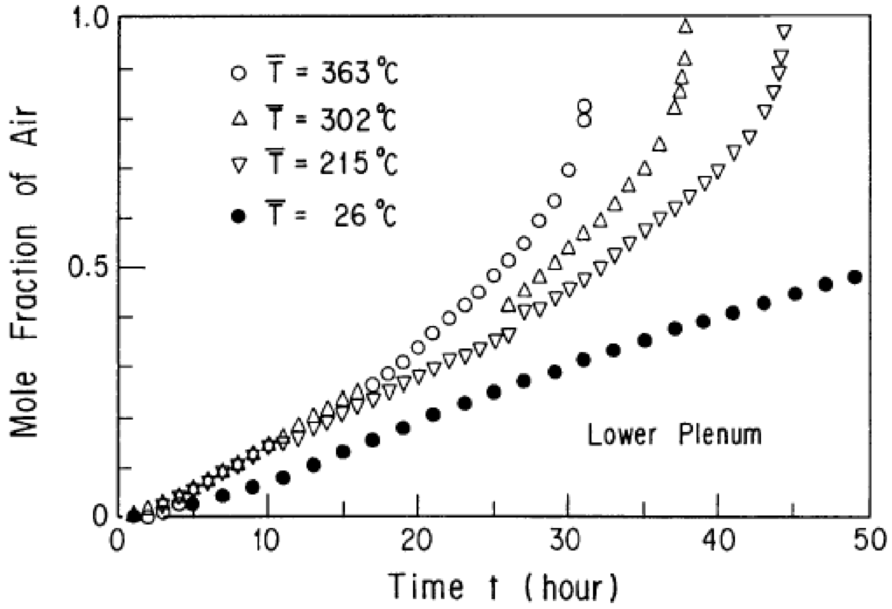
AVR LOCA (DLOFC) Simulation

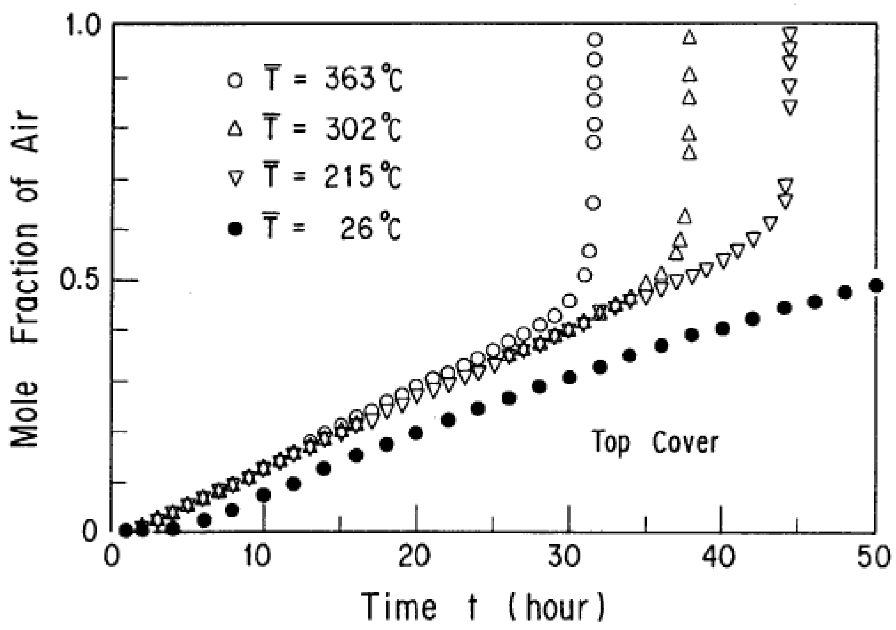


JAERI Air Ingress Test Rig

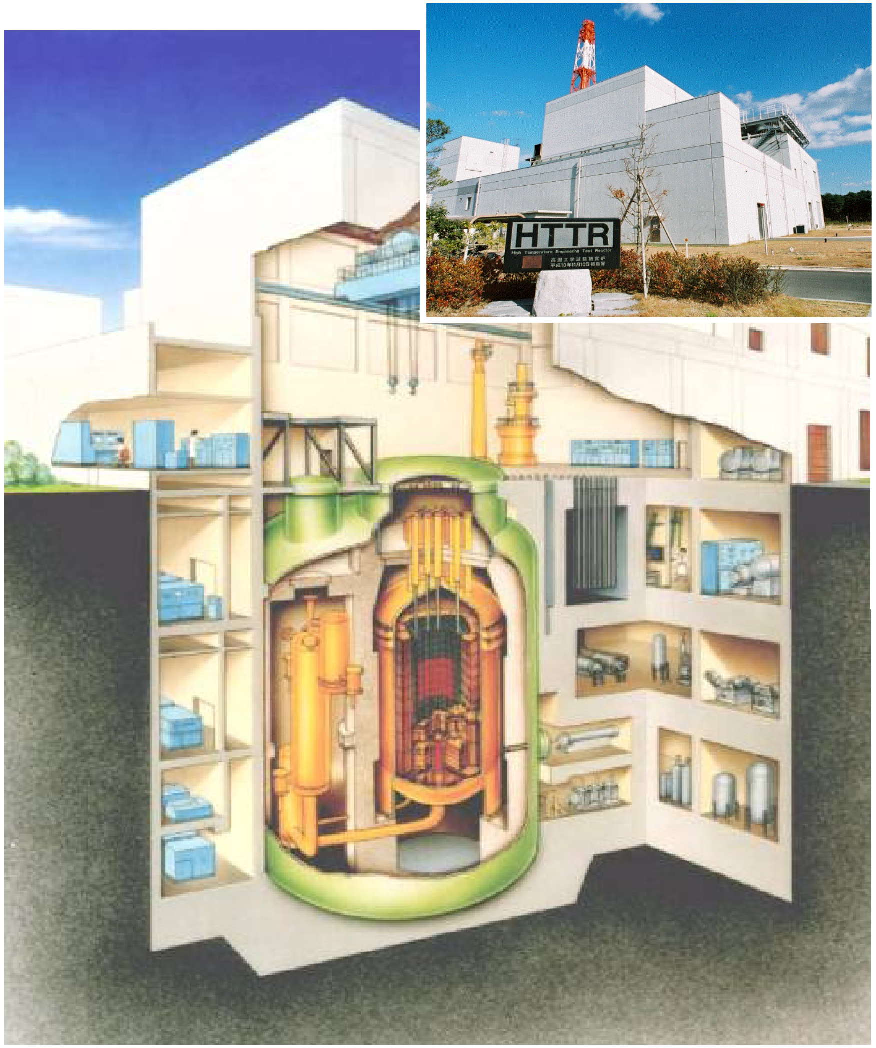


JAERI Air Ingress Test Results





Japan's High Temperature Engineering Test Reactor (HTTR)



Major specification

Thermal power	**30 MW**
Fuel	**Coated fuel particle / Prismatic block type**
Core material	**Graphite**
Coolant	**Helium**
Inlet temperature	**395 °C**
Outlet temperature	**950 °C (Max.)**
Pressure	**4 MPa**

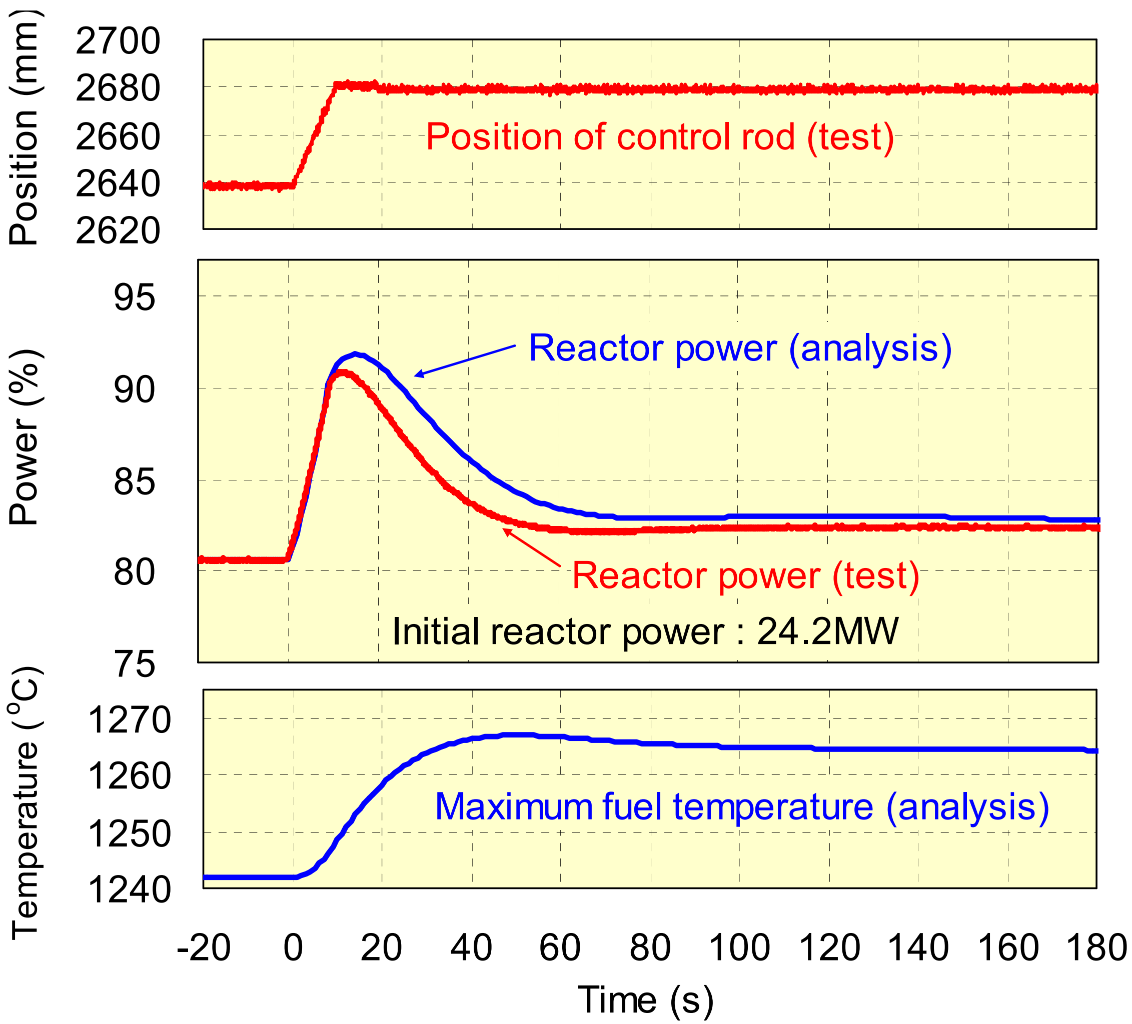
History

- First criticality : 1998
- Full power operation : 2001
- Safety demonstration test : 2002
- High temperature operation (950℃) : 2004

Reactivity Insertion Test in Japan's HTTR

- **Test conditions**
  - Reactor power: 30% - 80%
  - Central pair of control rods are withdrawn
  - Withdrawal rate: 1 or 5 mm/s
  - Withdrawal distance: 50 mm (max)
- **Data to be obtained**
  - Reactor power
  - Reactivity
  - Primary coolant temperatures
  - Temperatures of reactor internals, etc.

Reactivity Insertion Test Results

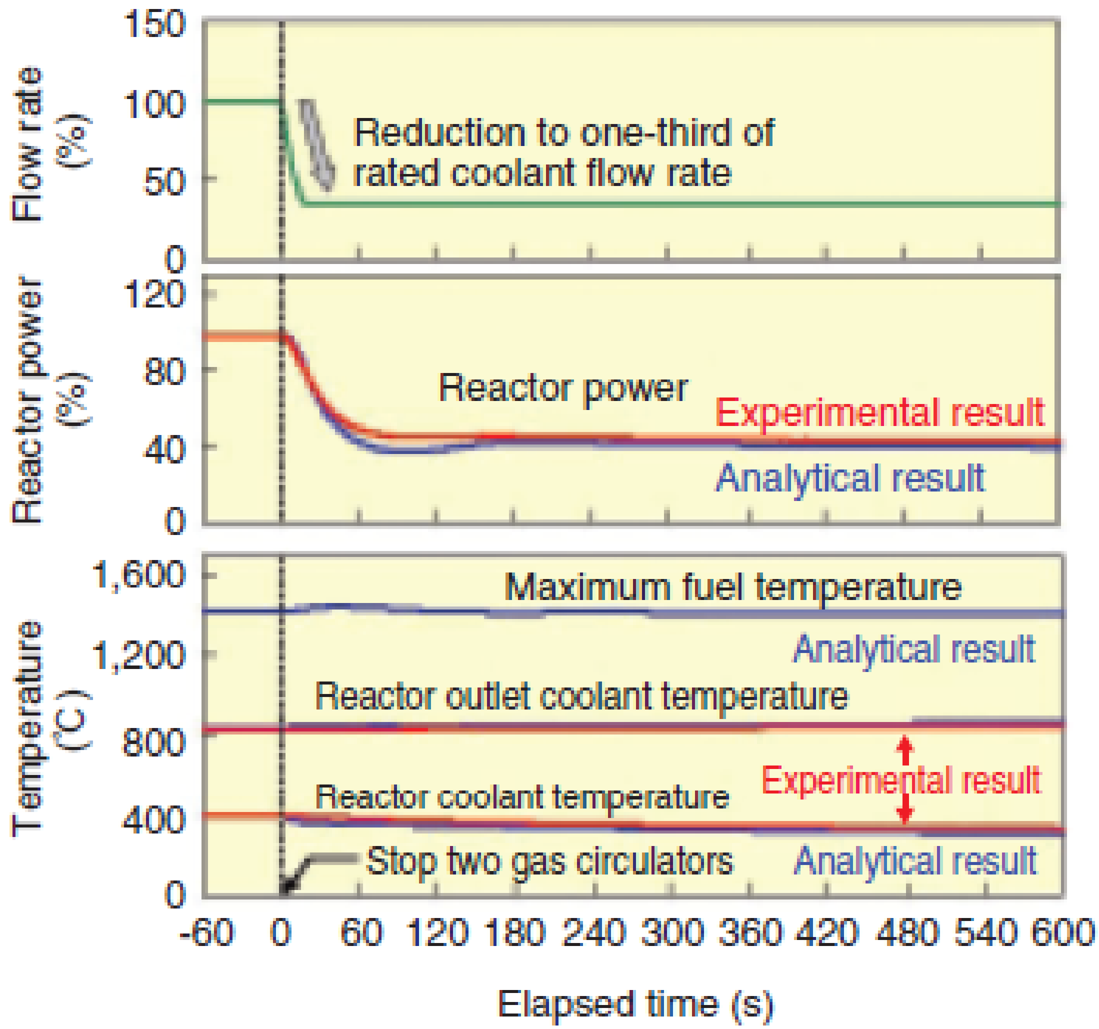


K. Takamatsu, et al.,
Journal of Power and Energy Systems
Vol. 2 (2008) , No. 2, p.790-803

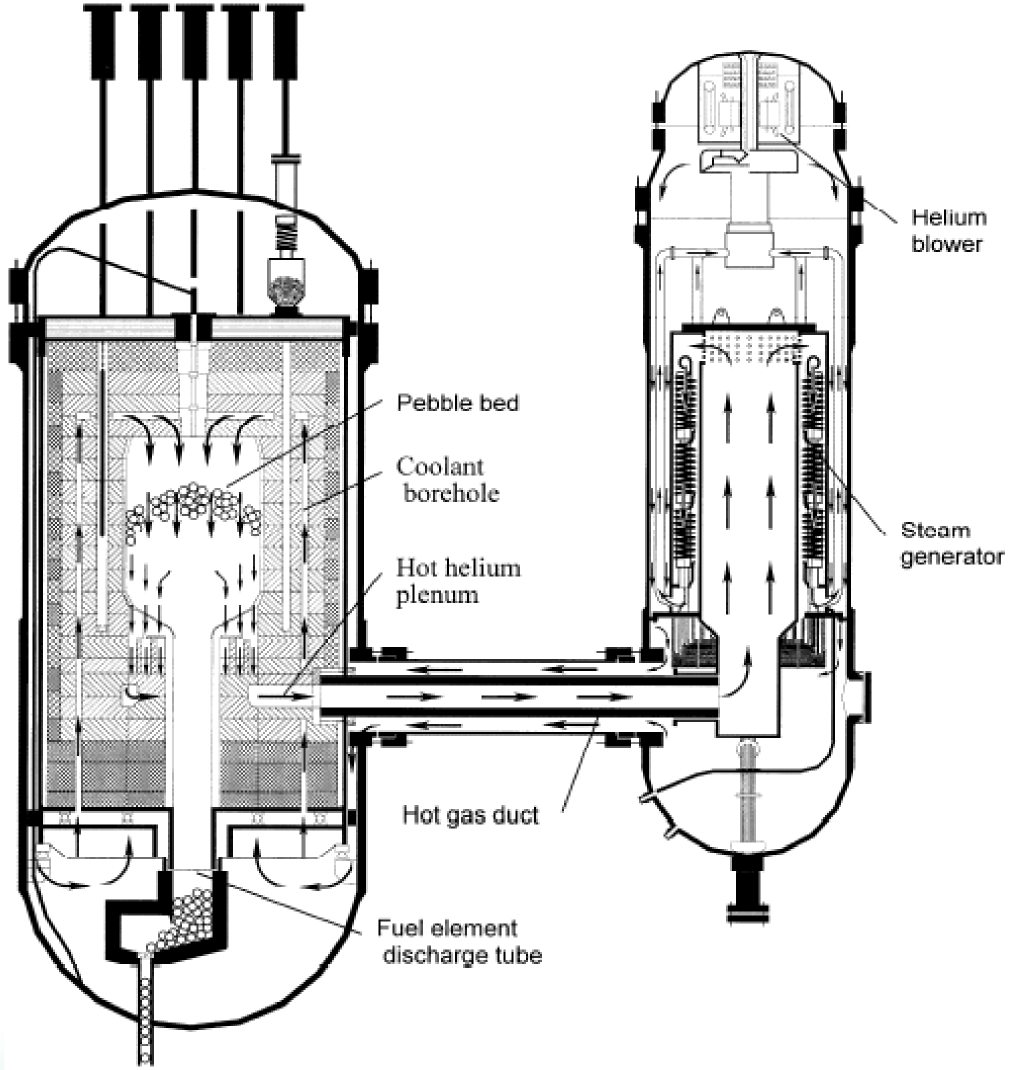
Coolant Flow Reduction Test in Japan's HTTR

- **Test conditions**
  - Reactor power: 30% - 100%
  - Parameters: change of primary coolant flow rate and rate of change
  - All of the control systems are operating
- **Data to be obtained**
  - Reactor power
  - Reactivity
  - Primary coolant temperatures
  - Primary coolant flow, etc.

Coolant Flow Reduction Test Results



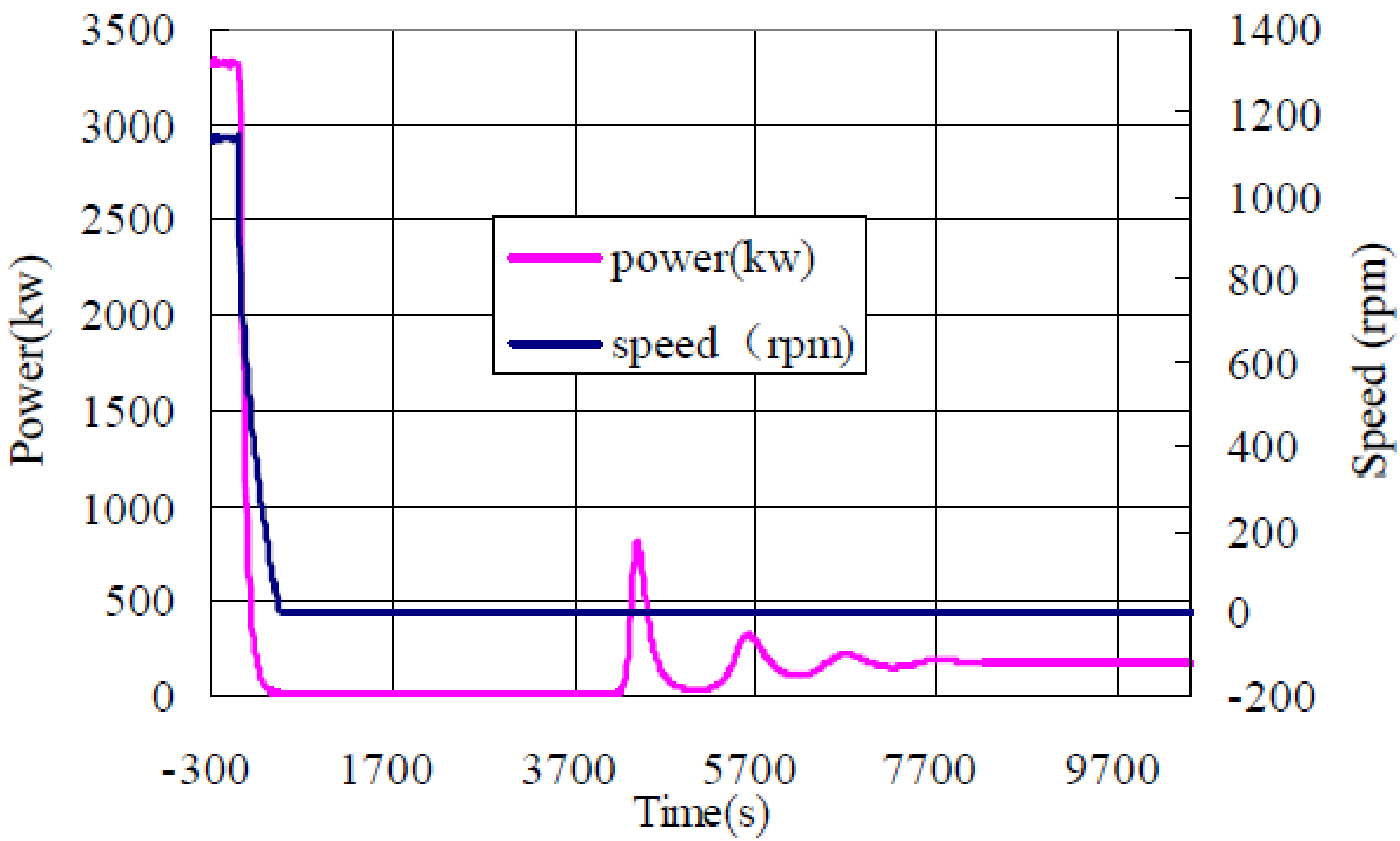
China's HTR-10



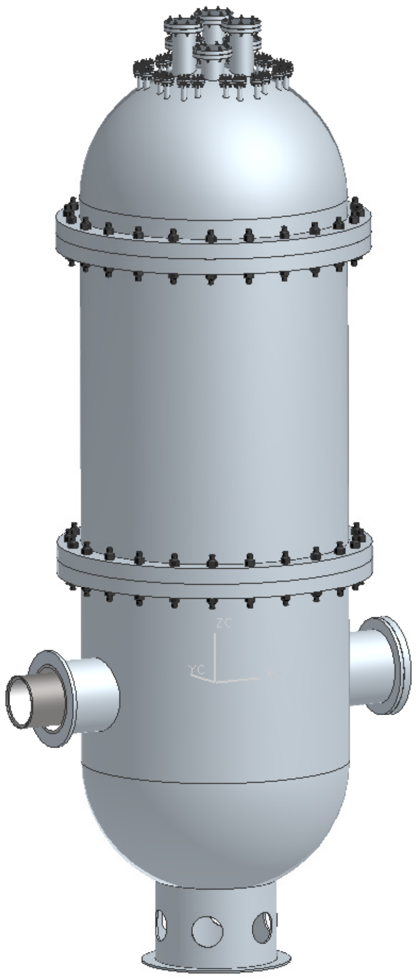
HTR-10 Safety Demonstration Tests

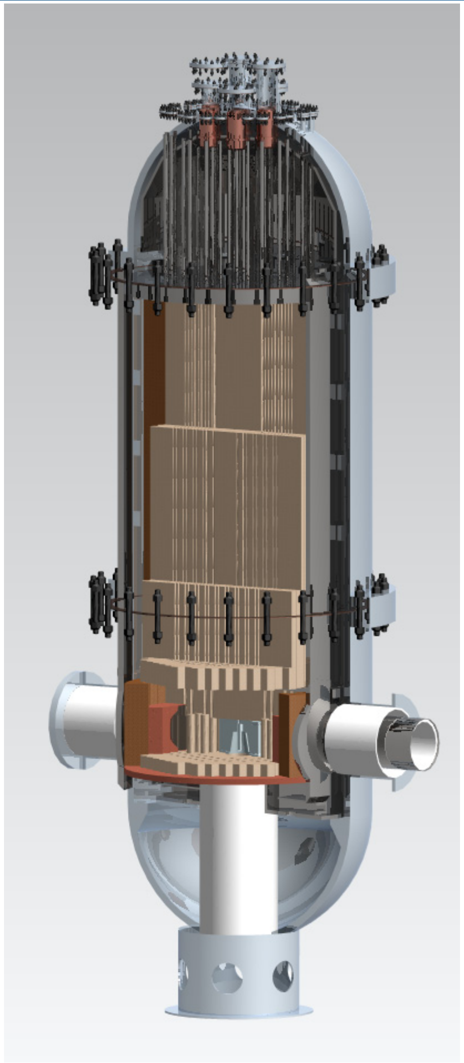
- Loss of offsite power without counter-measures
- Main helium blower shutdown (LOFC) with ATWS
- LOFC-ATWS with control rod withdrawal
- Loss of main heat sink without counter-measures

HTR-10 LOFC and ATWS Test



High Temperature Test Facility Planned at Oregon State University





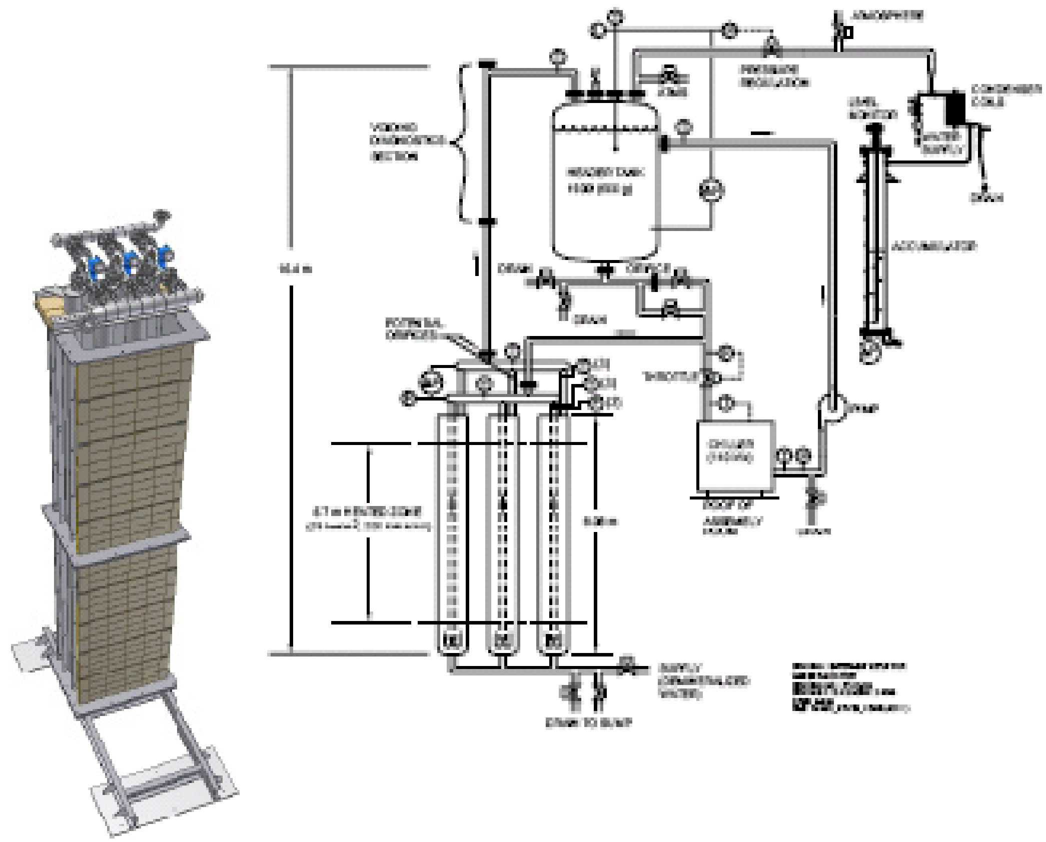
OSU Oregon State UNIVERSITY

OSU HTTF Objective and Approach

- **Objective: Generate validation data for both systems analysis and CFD software**
- **High Temperature Test Facility (HTTF):**
  - Designed in a scaled manner to be capable of simulating flow and heat transfer behavior during DLOFC transient
  - Other scenarios examined for applicability of facility: PLOFC and normal operations

RCCS Experiments Planned at ANL

- Empty cavity, single and multiple tubes
- Constant wall temperature and constant heat flux
- Steady state and transient
- Air- and water-cooled RCCS tests



Natural Convection Shutdown Heat Removal Test Facility (NSTF) at Argonne Nat. Lab

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Prismatic Sensitivity Analyses for Depressurized Conduction Cooldown (DLOFC)

	Peak Fuel	Upper Core Restraint	Vessel Midwall
Decay Heat*1.1	1.045	1.021	1.029
Kf*1.28, Kg*1.2	0.983	1.013	1.003
Kf*0.72, Kg*0.8	1.073	0.984	0.993
Cpf*0.91, Cpg*0.91	1.012	1.006	1.008
εvo=.7, εvi=.6, ε800H=.45	1.015	1.059	1.007

600 MWt Prismatic HTGR

Uncertainty Analysis Approach

- **Monte Carlo evaluation coupled to simple TAC2D thermal model**

- **Uncertainty for each model parameter is sampled with a specified distribution to obtain statistical temperature distribution for peak components of interest**

Prismatic Uncertainty Analysis for Depressurized Conduction Cooldown (DLOFC)

	Lower Uncertainty (°C)	Best Estimate (°C)	Upper Uncertainty (°C)	Limits (°C)
Fuel	1316	1417	1538	1600
Control Rods	1100	1181	1275	>1315
Core Barrel	685	734	801	760
Reactor Vessel Midwall	490	490	541	565
Upper Core Restraint	623	690	773	1095
Upper Plenum Shroud	490	490	537	900

600 MWt Prismatic HTGR

Importance of Uncertainty to Peak Core Temperature During DLOFC

	Fraction of Total Uncertainty
Decay Heat	0.568
Radial Conductivity	0.391
Heat Capacity	0.009
Emissivity	0.006
Vessel Temperature	0.002
Initial Temperatures	0.006
Axial Heat Rate	0.016

600 MWt Prismatic HTGR

Prismatic Uncertainty Analysis for Pressurized Conduction Cooldown (PLOFC)

	Lower Uncertainty (°C)	Best Estimate (°C)	Upper Uncertainty (°C)	Limits (°C)
Fuel	1045	1140	1240	1600
Control Rods	840	910	985	>1315
Core Barrel	645	680	715	760
Reactor Vessel Midwall	490	490	490	565
Upper Core Restraint	865	920	970	1095
Upper Plenum Shroud	685	725	760	900

600 MWt Prismatic HTGR

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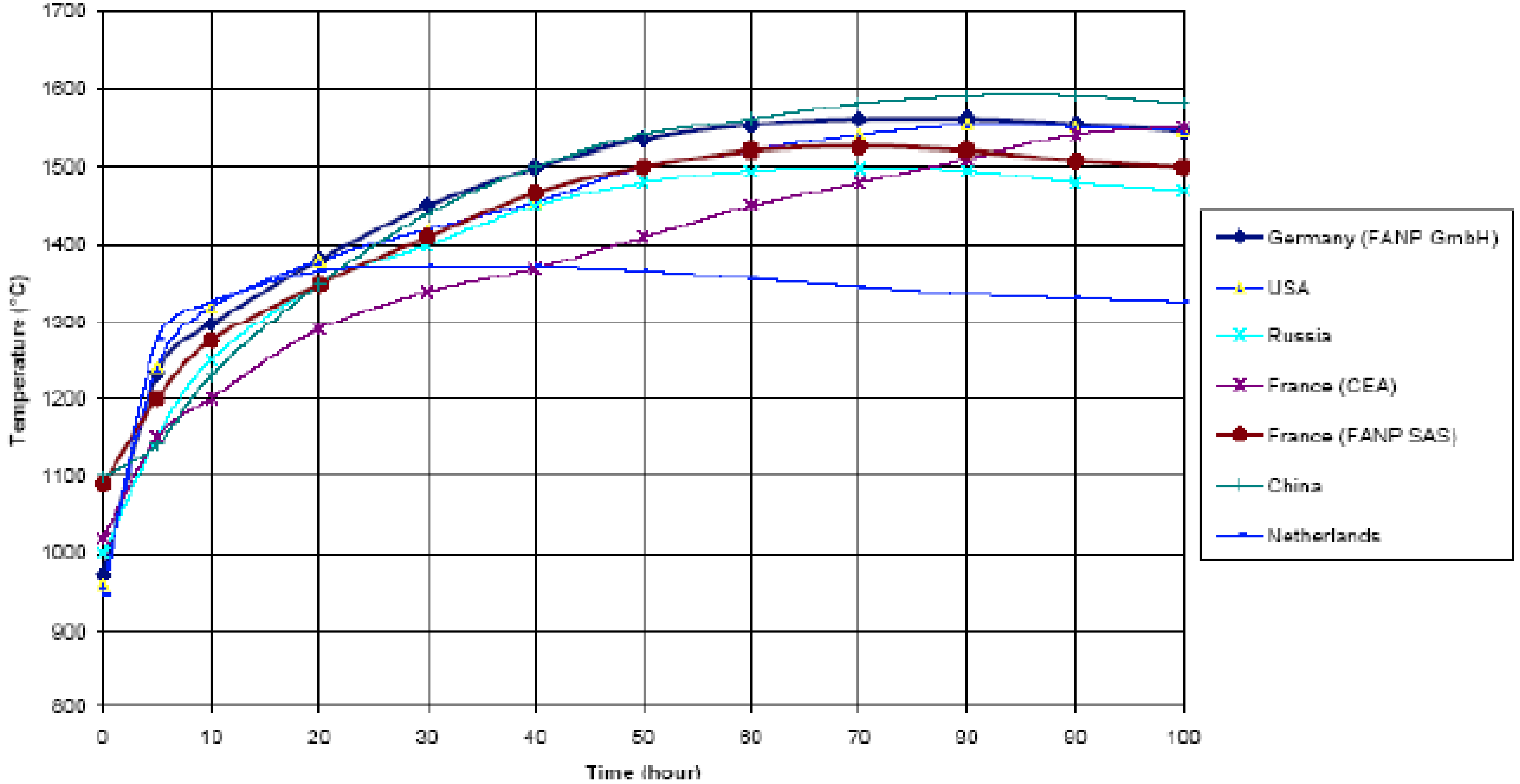
Code Benchmarking, Verification, and Validation

- **"Evaluation of High Temperature Gas Cooled Reactor Performance: Benchmark Analysis Related to Initial Testing of the HTTR and HTR-10," IAEA-TECDOC-1382, November 2003.**
  - Reactor physics and thermal hydraulics benchmark problems
  - Analyses performed by China, Japan, France, Germany, Indonesia, Netherlands, Russian Federation, Turkey, South Africa and USA

IAEA CRP-3 Benchmarking DLOFC Peak Fuel Temperatures



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 → Analysis

No → Development

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

Yes → 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?

Summary

- Accident analysis tools have been developed for both prismatic and pebble bed HTGRs over their long history
- Test reactors have been used to demonstrate the safety characteristics of the HTGR
- Modern analytical tools such as computational fluid dynamics have been and will be used
- Benchmarking, verification, and validation efforts are underway

Suggested Reading

- "Preliminary Safety Information Document for the Standard MHTGR," HTGR-86024, Rev. 13, September 1992, ML093560560.
- "Heat Transport and Afterheat Removal for Gas Cooled Reactors Under Accident Conditions," IAEA-TECDOC-1163, 2001.
- "Evaluation of High Temperature Gas Cooled Reactor Performance: Benchmark Analysis Related to Initial Testing of the HTTR and HTR-10," IAEA-TECDOC-1382, 2003.
- "Accident Analysis for Nuclear Power Plants with Modular High Temperature Gas Cooled Reactors," IAEA Safety Report Series No. 54, 2008.