# HTGR Technology Course for the Nuclear Regulatory Commission

**May 24 – 27, 2010**

**Module 15**

**HTGR Accident Analysis Tools**

**John Bolin**

**General Atomics**

# 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

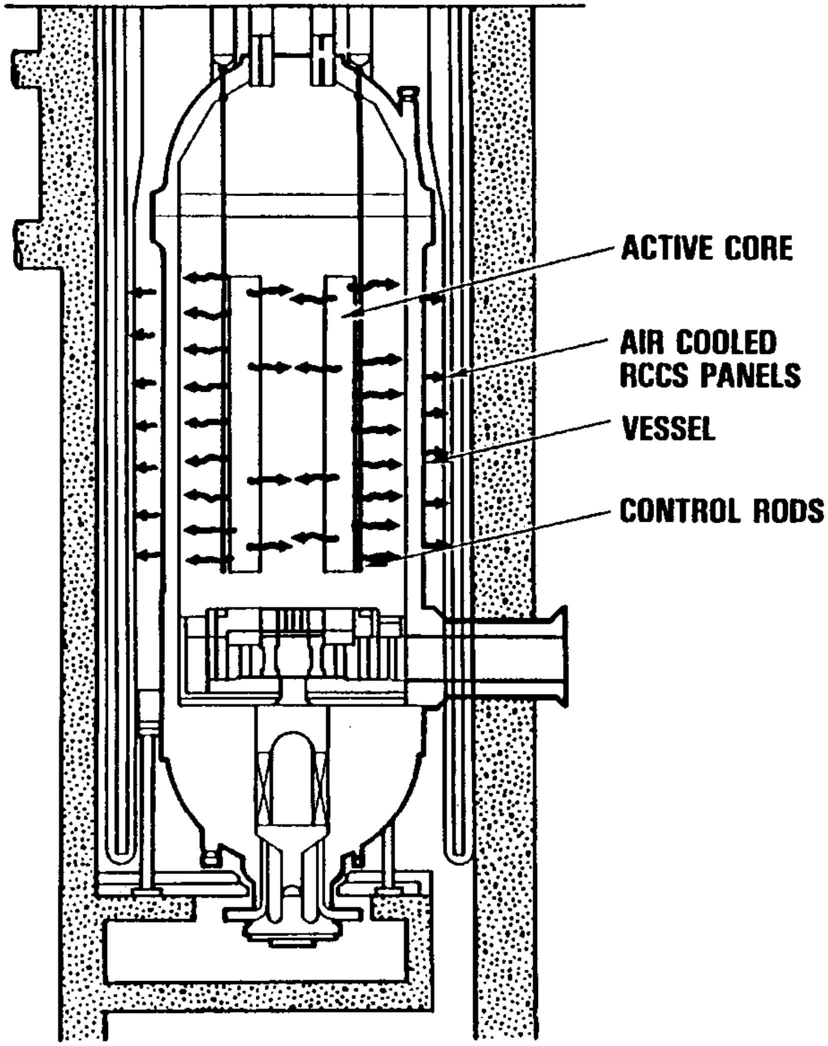
# Thermal Hydraulic Methods Used for Pebble Bed Core Heat Removal

- **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)

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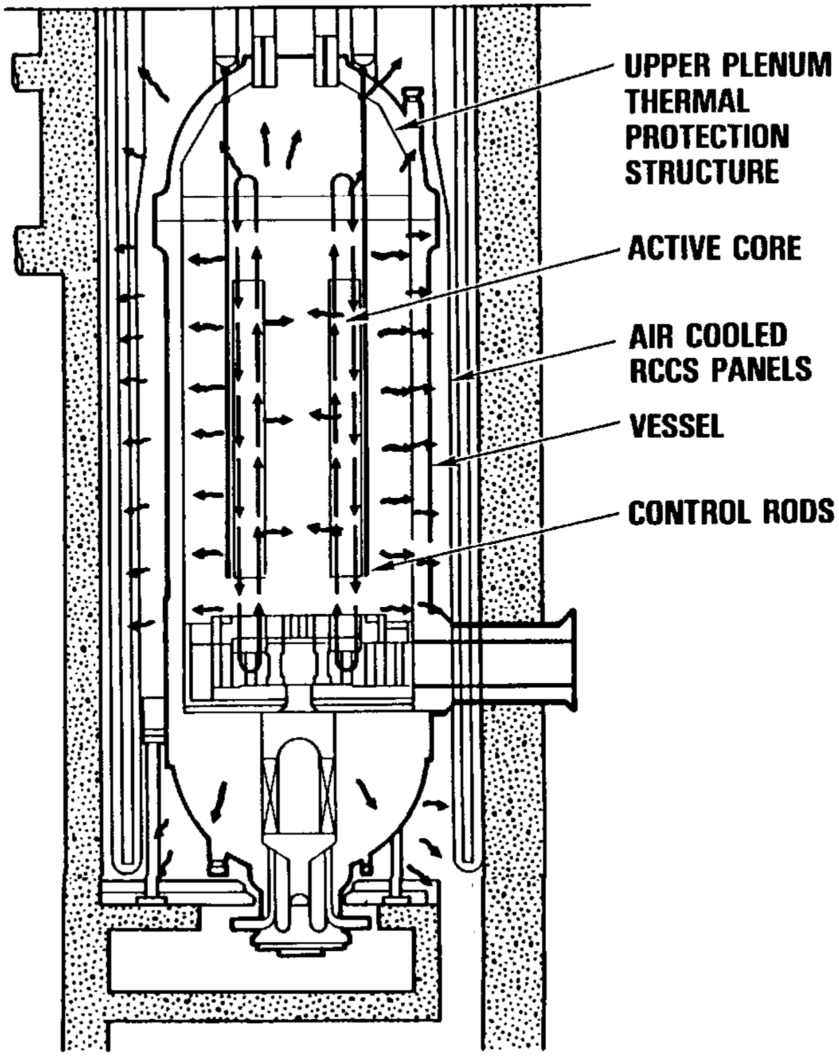
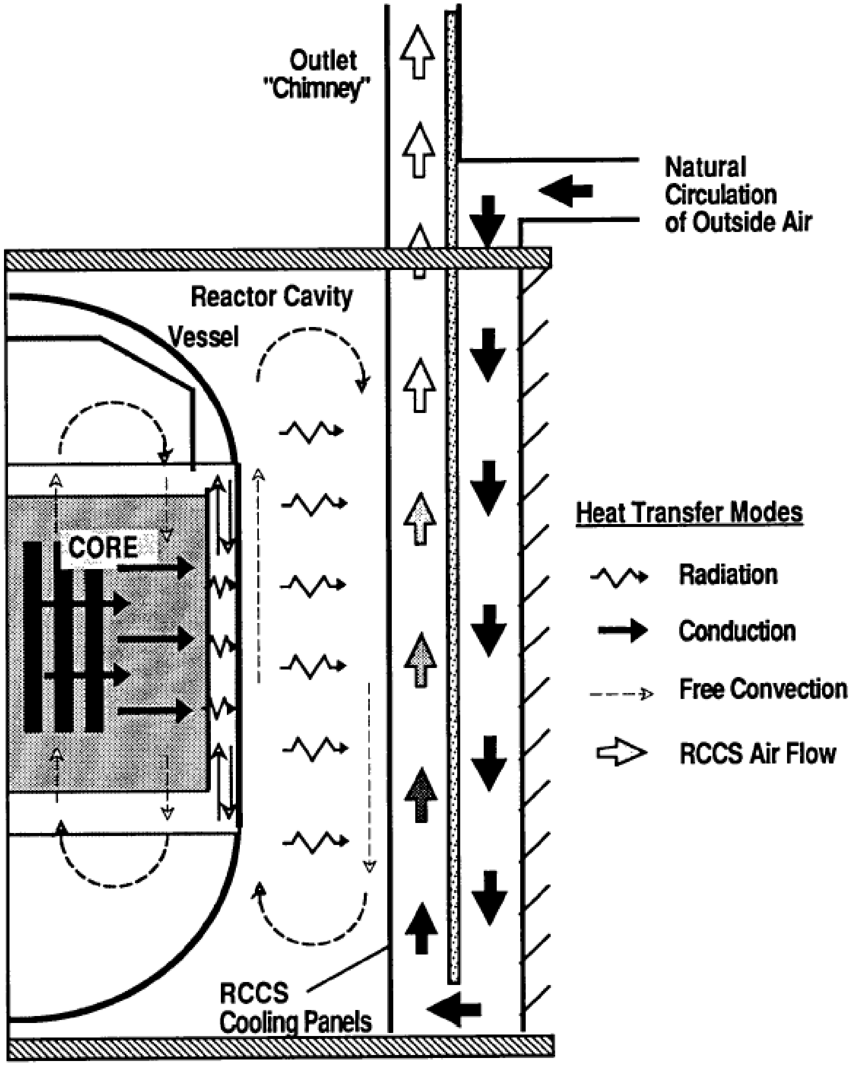
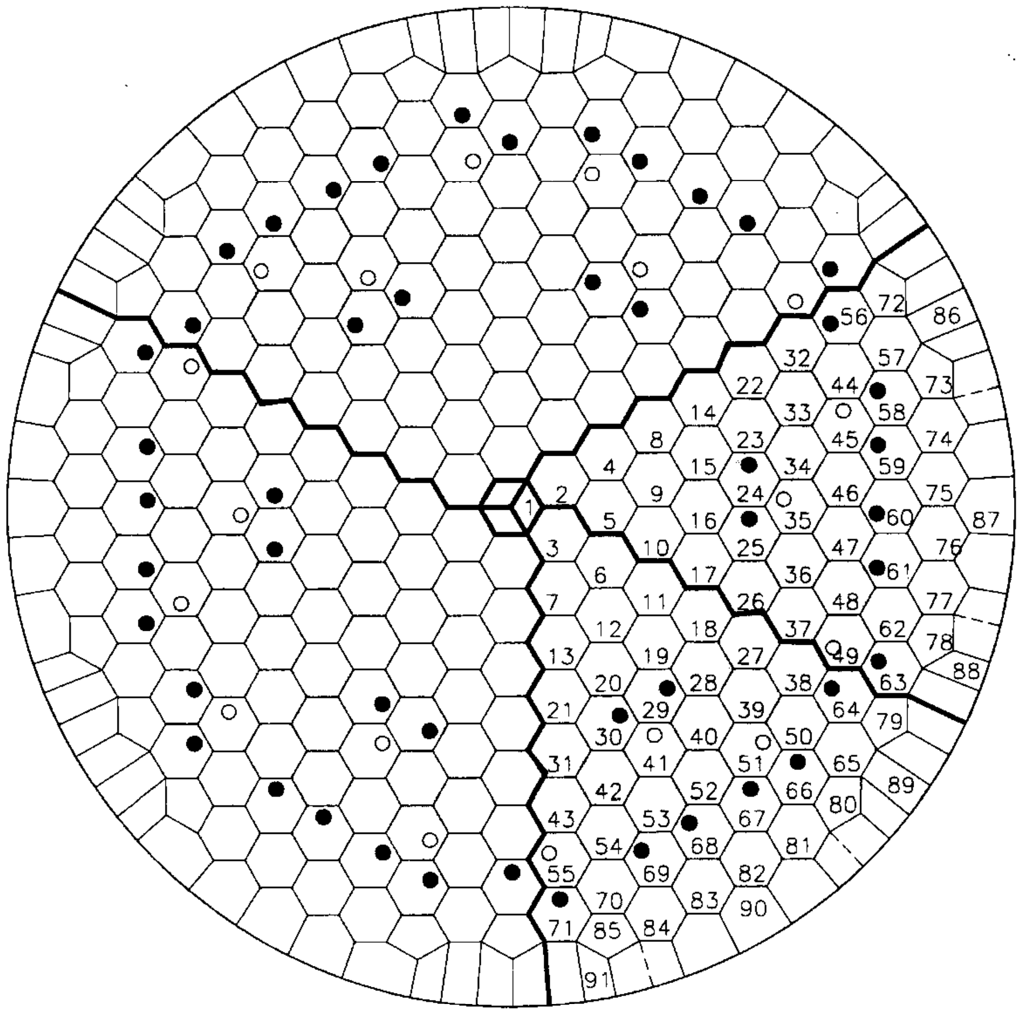
- VESSEL RADIATION
- RCCS CONVECTION



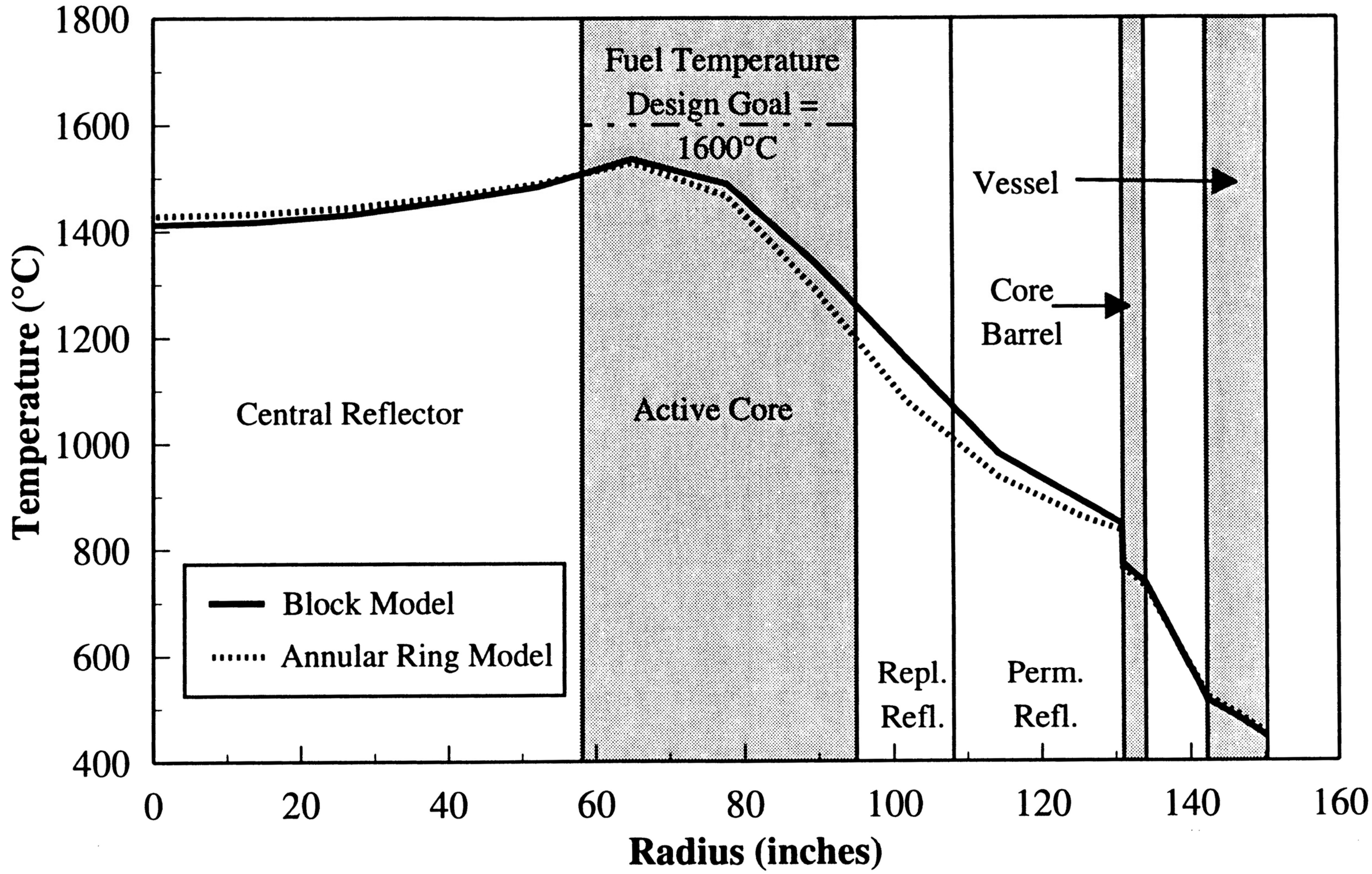
# Heat Transfer to the RCCS



# One-Third Core SINDA/FLUINT Model



# Comparison of Radial Temperature Distributions from Two Models During DLOFC




Idaho National Laboratory



GENERAL ATOMICS

# 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

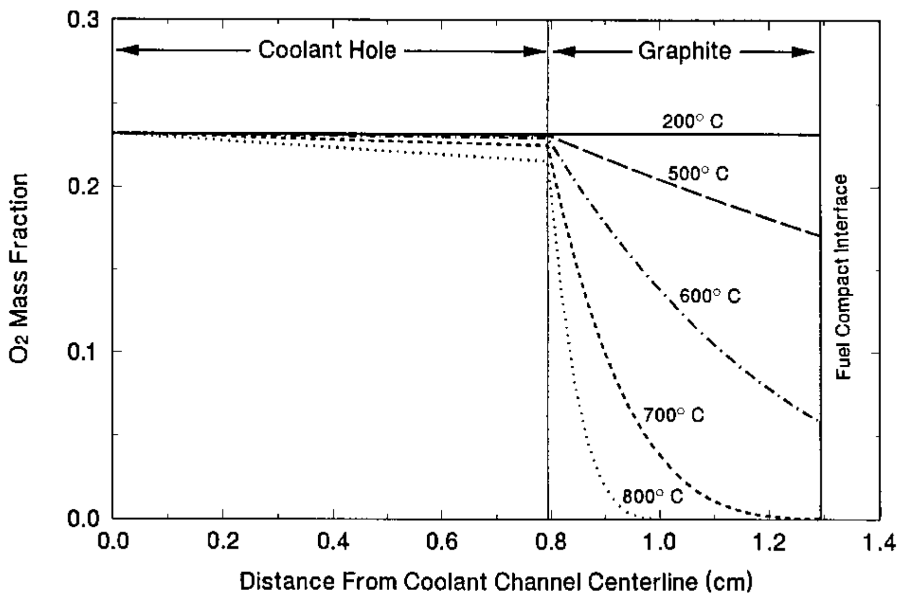
- **OXIDE-4 code used for graphite oxidation due to air and steam**
- **AIP – Air Ingress Program – used to model graphite oxidation from air**
  - Models $O_2$ and $CO_2$ reactions with graphite
  - Models CO combustion in flow channel
  - Models natural convection flows by balancing buoyancy against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

# Progression of Air Ingress Events

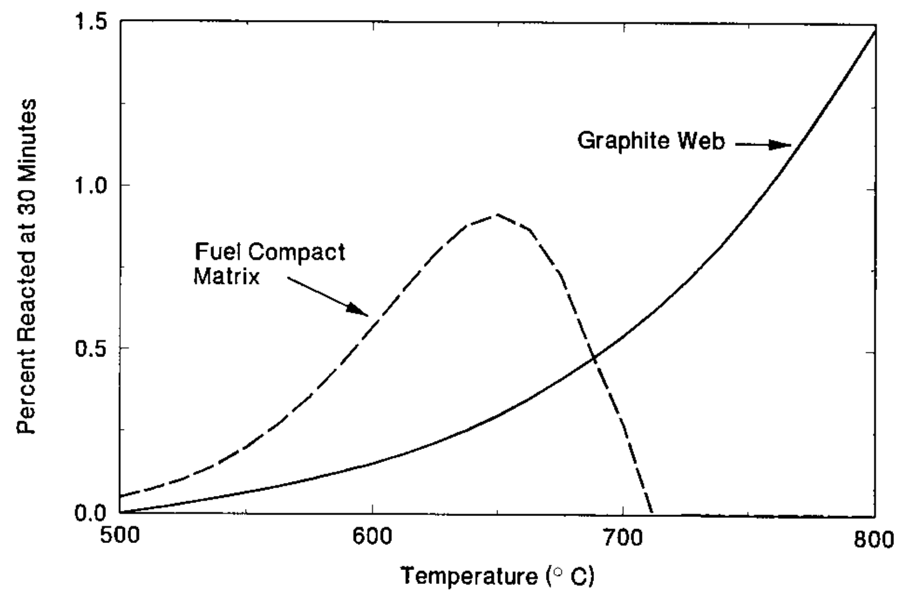
- **Overall oxidation rate determined by rate of air supply**
  - Friction between core and fluid greatly limits flow rate
  - Flow rate further limited as core heats up because viscosity increases with temperature
  - Core cool down reduces oxidation to negligible level
  - Graphite mass loss is a few 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

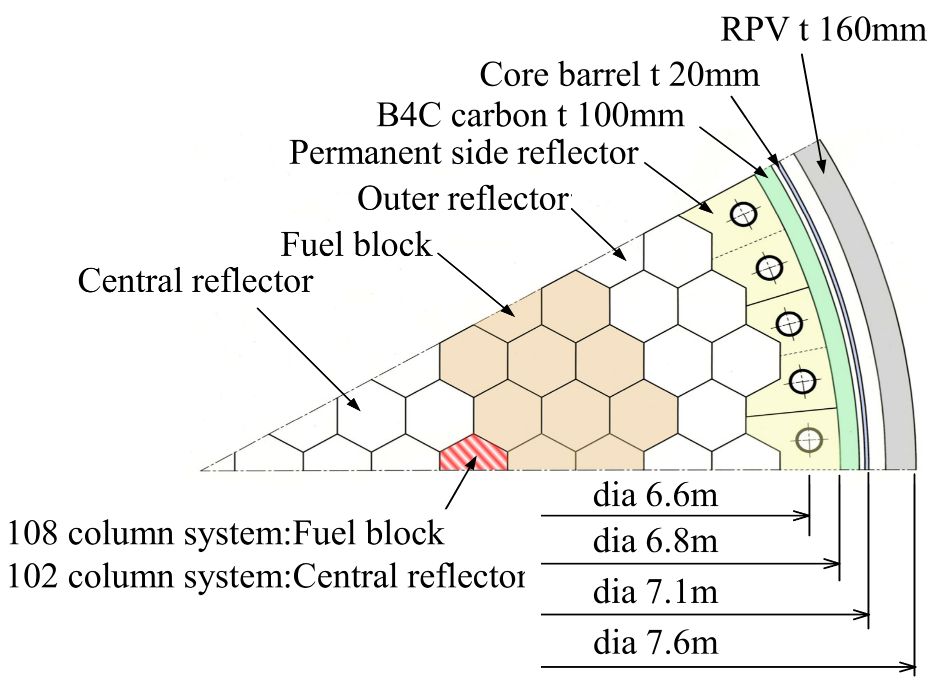
Oxygen Profile



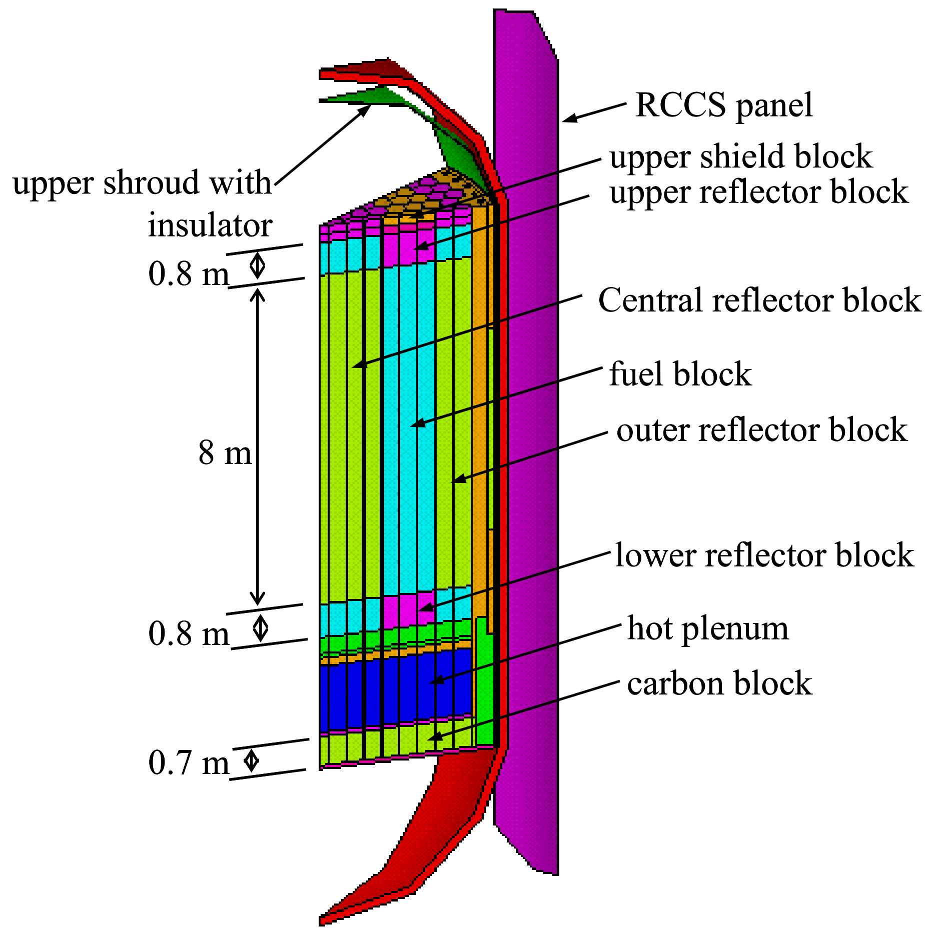

Reaction with Graphite and Compact



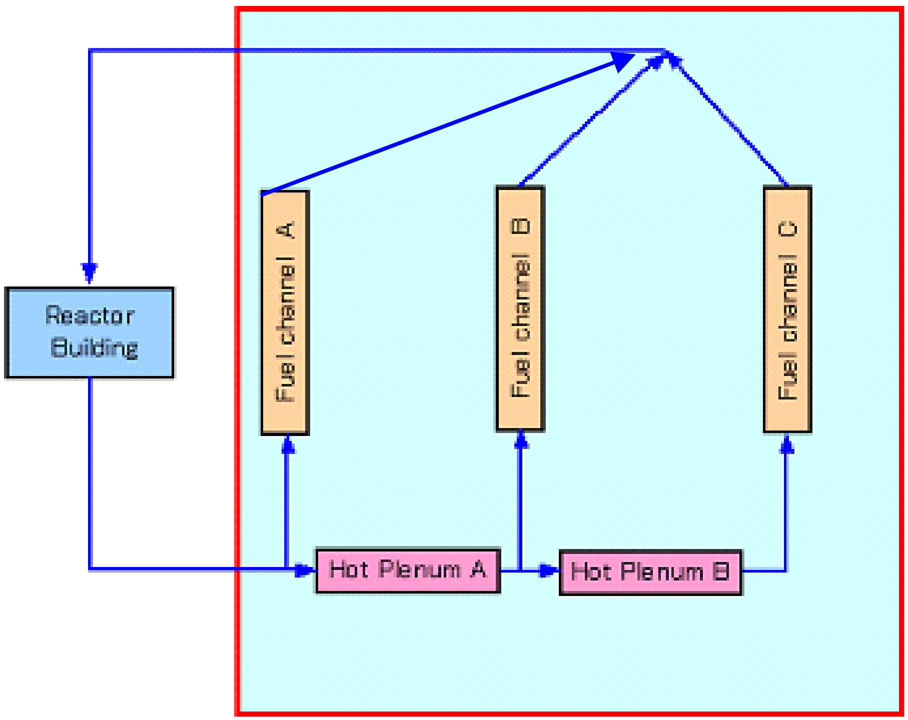
# ANSYS Model Used to Simulate Air Ingress



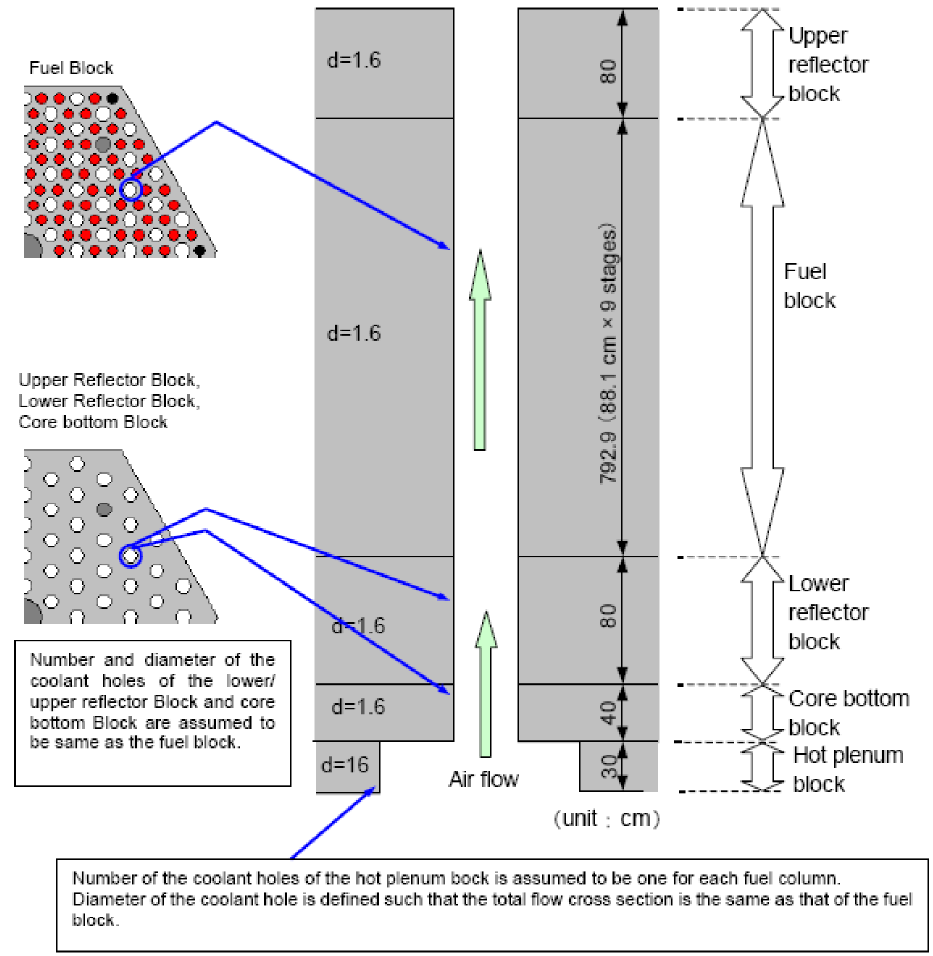

Horizontal view of 30 deg. Sector model



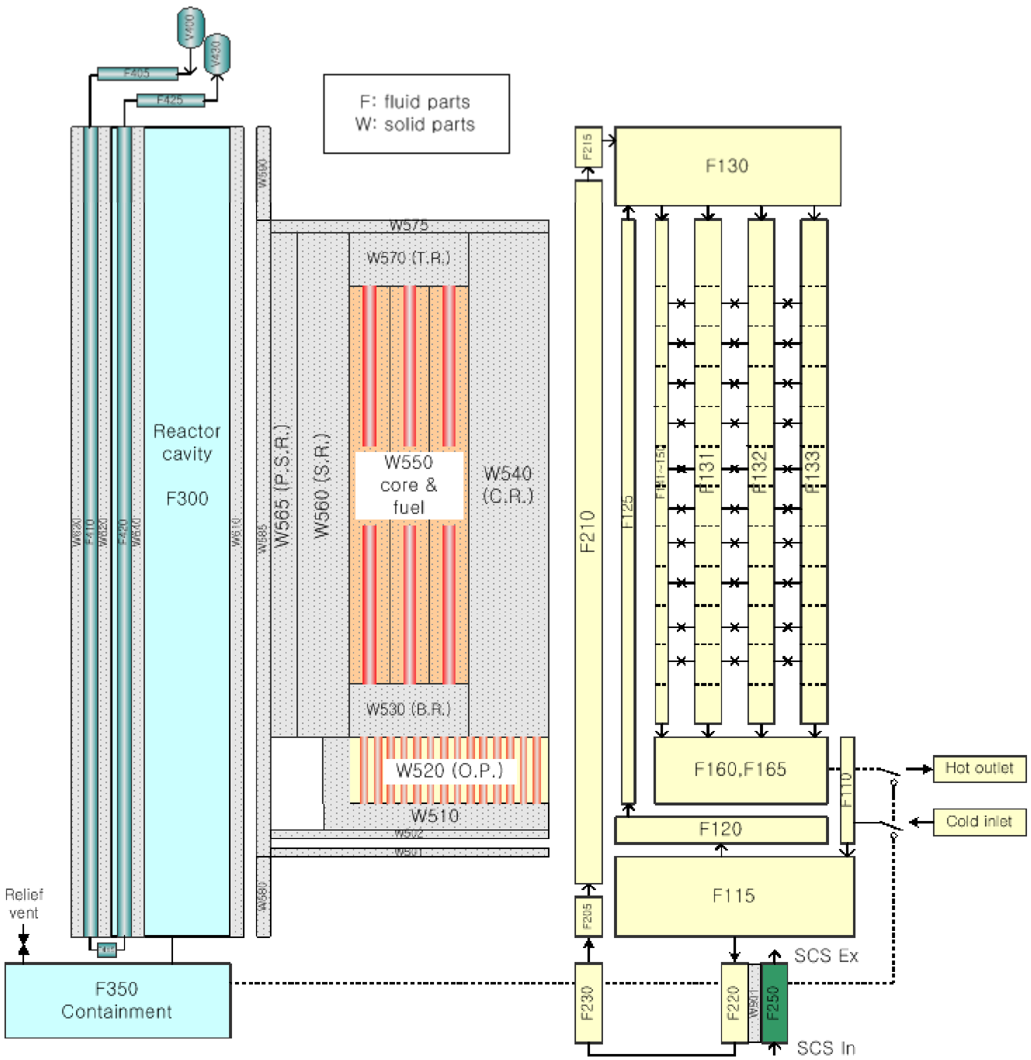
# GRACE Model Used to Simulate Air Ingress




## 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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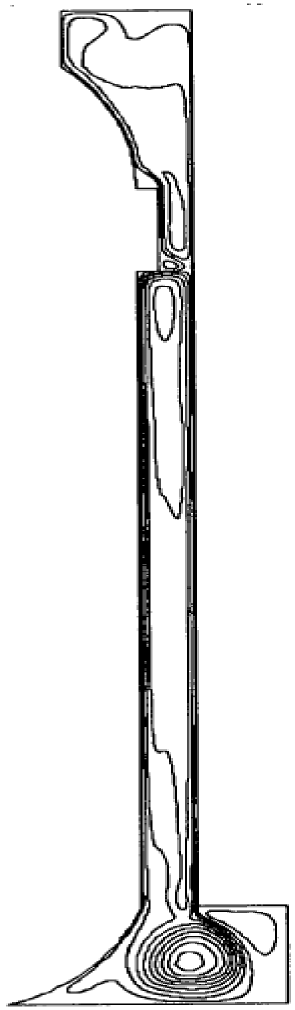
# CFD Applications of Interest to Designers

- **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

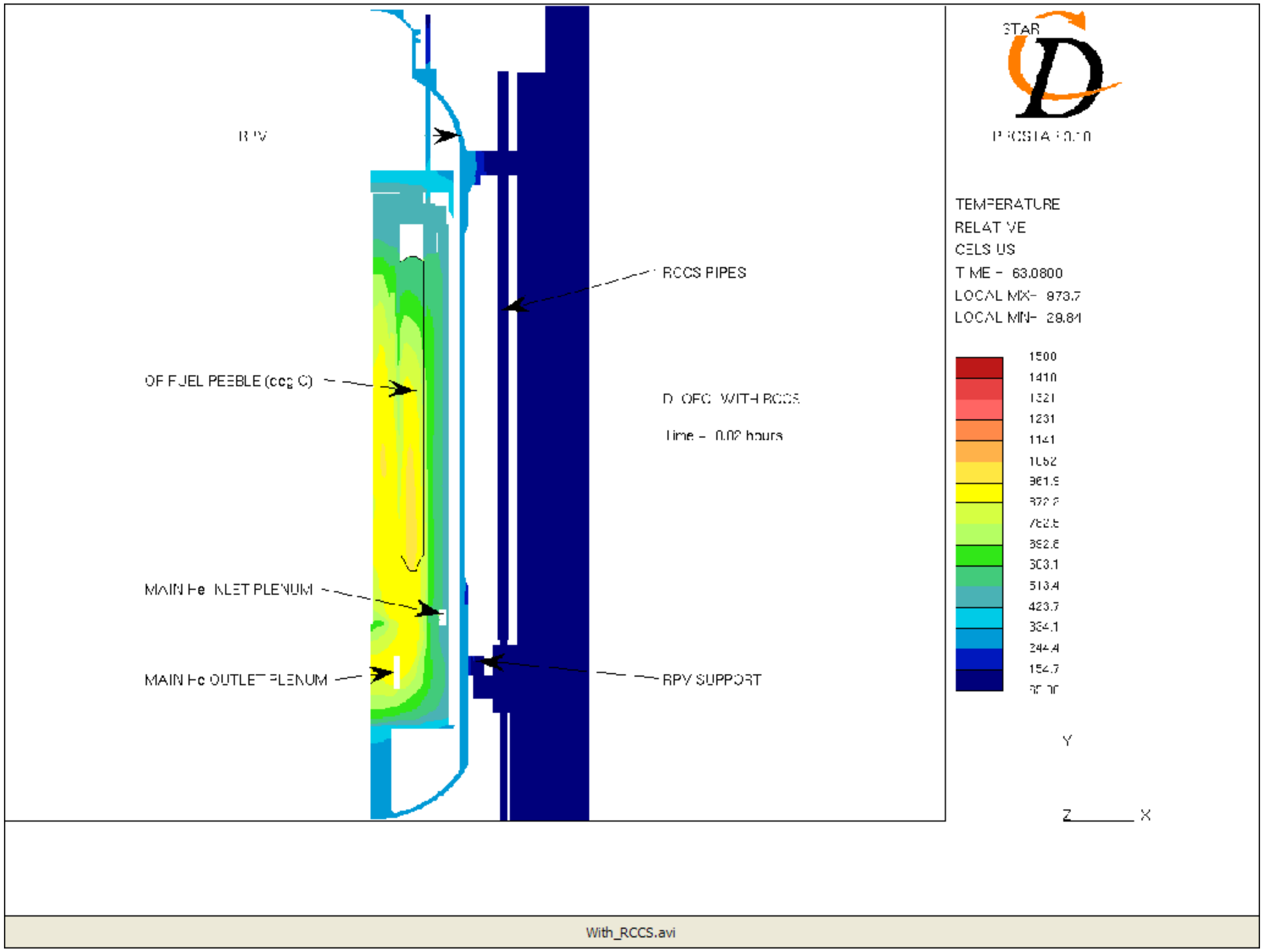
# Typical Reactor Cavity Natural Circulation Flow Field (MHTGR)



Depicts localized natural circulation cells

# PBMR CFD Analysis Capabilities

- **D-LOFC with STAR CD**



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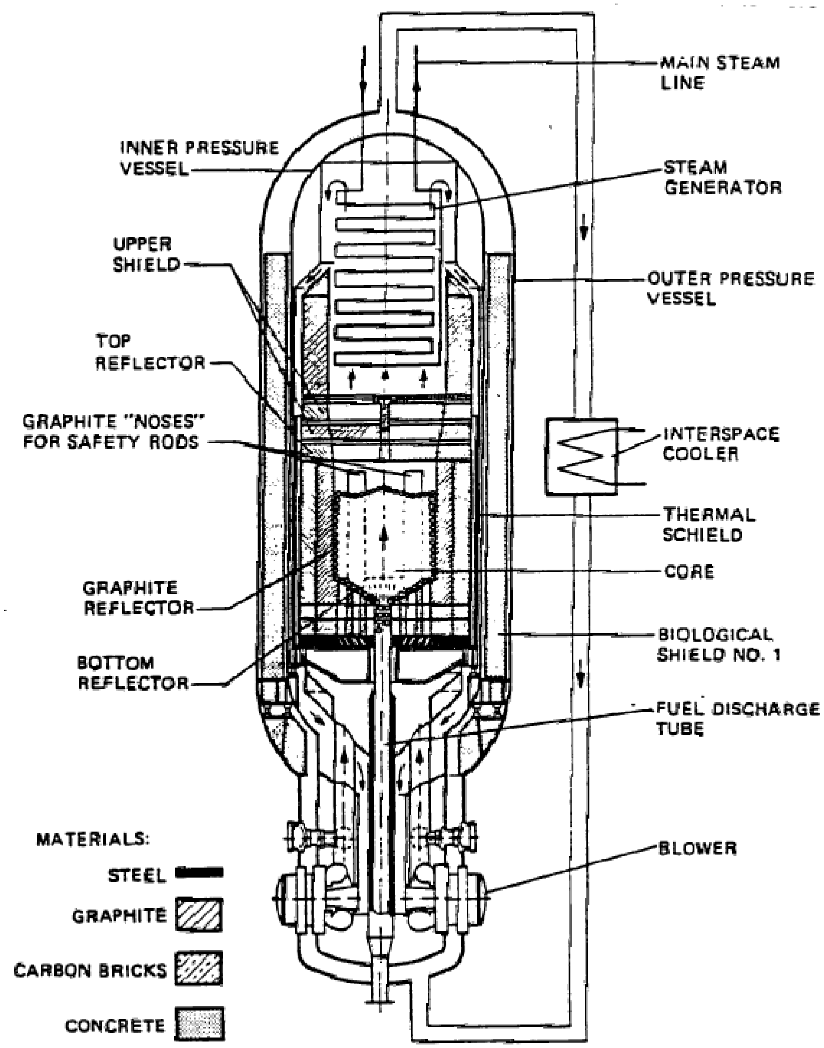
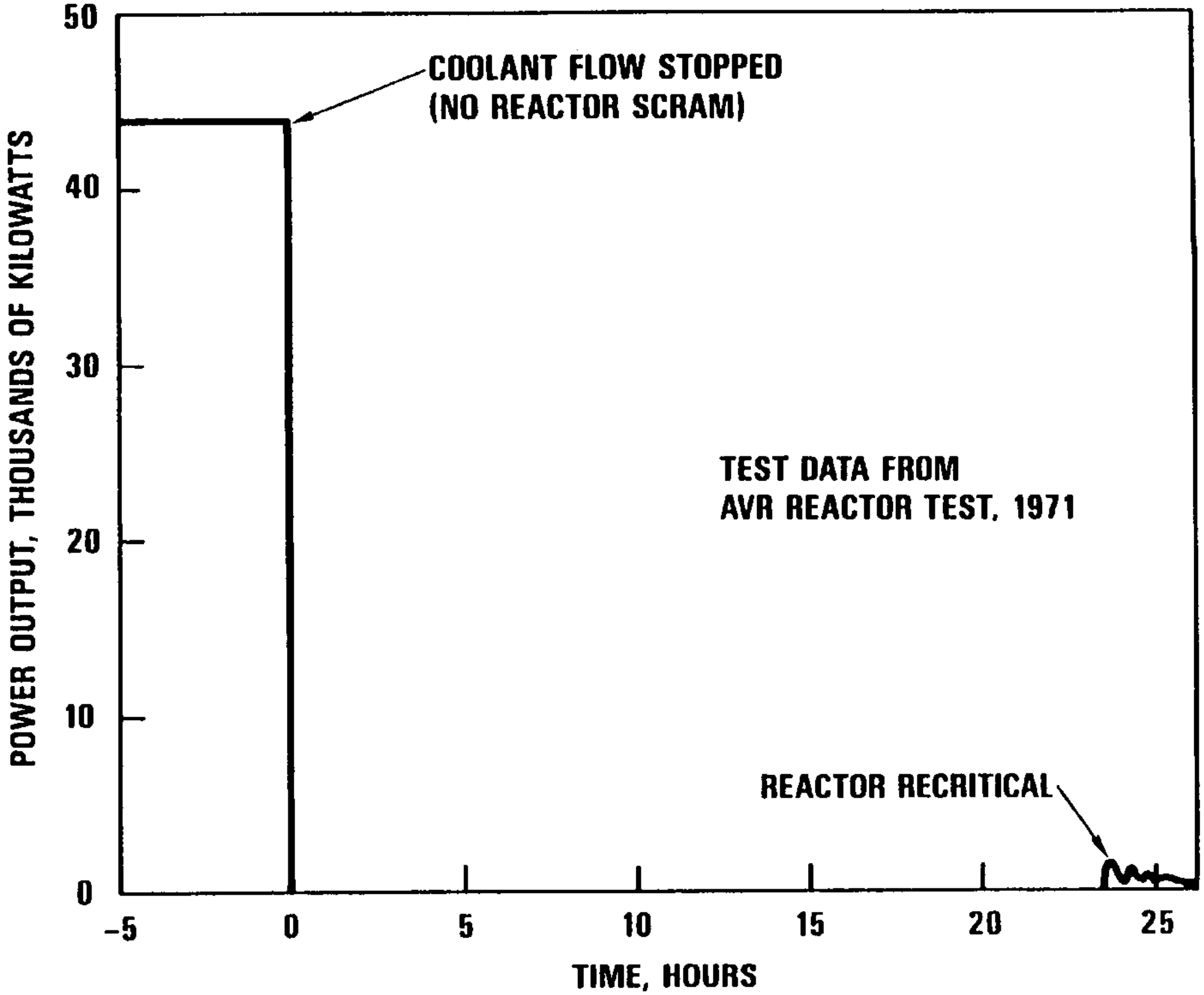
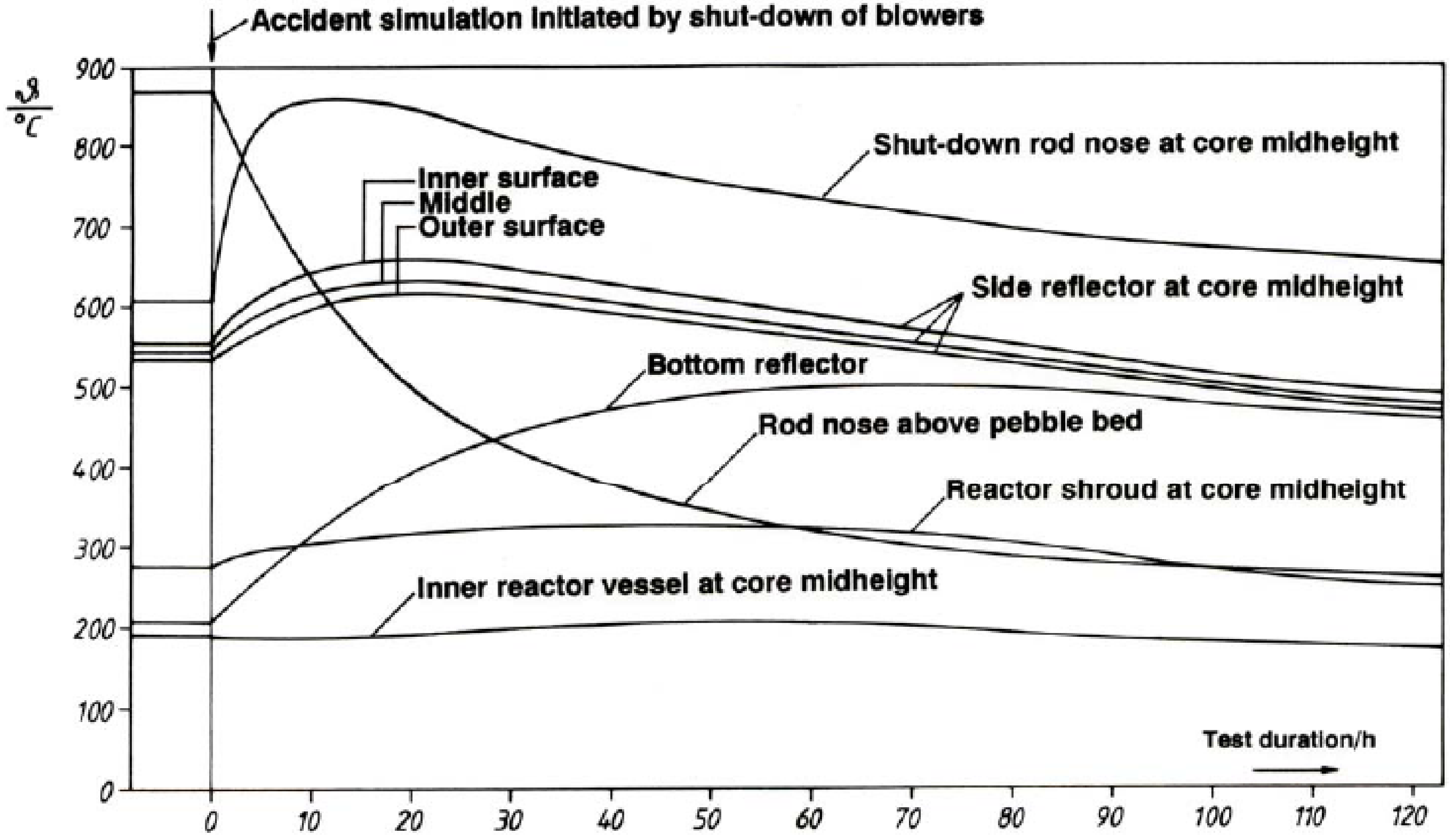
# German AVR Arrangement



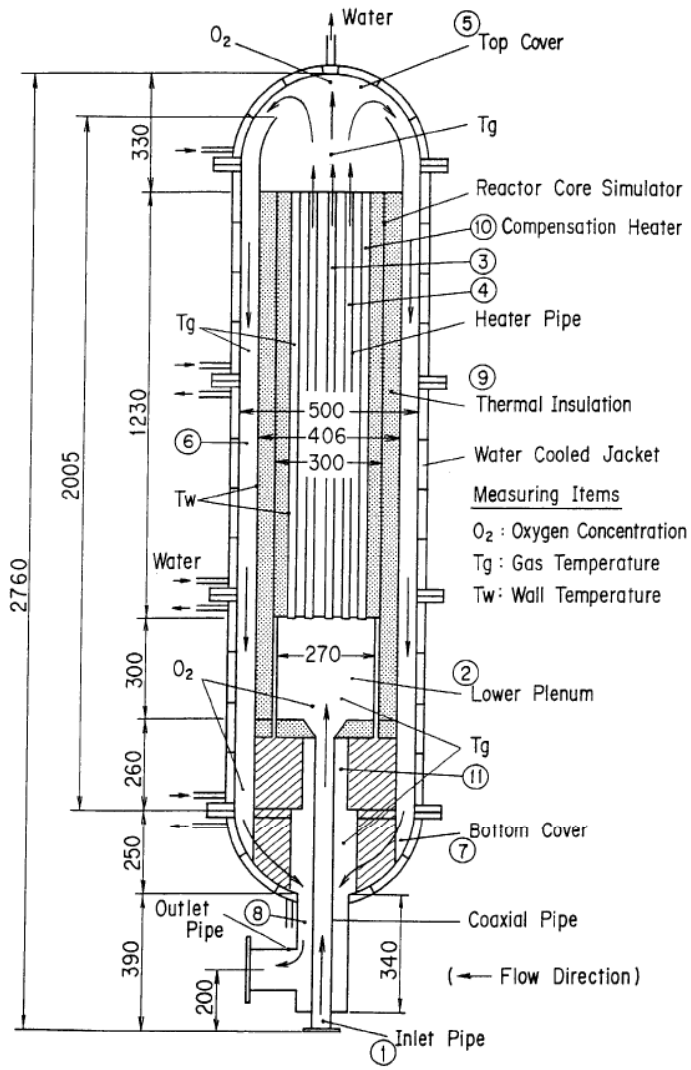
# ATWS Test in German AVR Demonstrated Termination of Nuclear Reaction



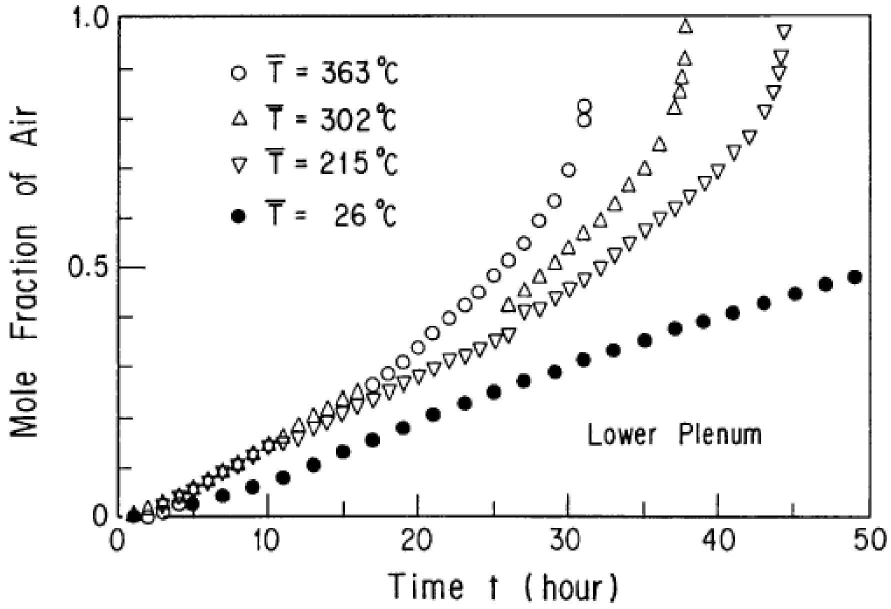
# AVR LOCA (DLOFC) Simulation

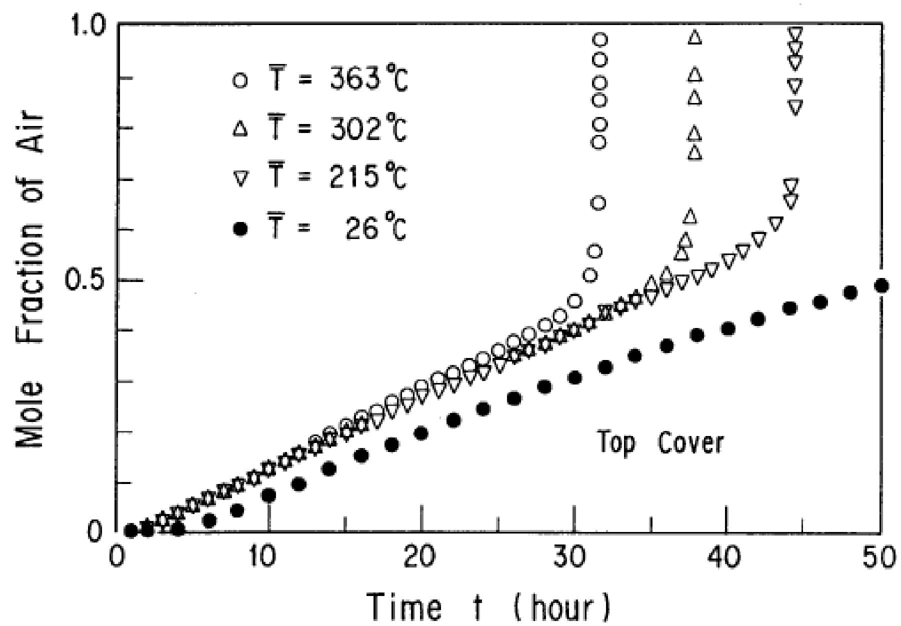


# JAERI Air Ingress Test Rig

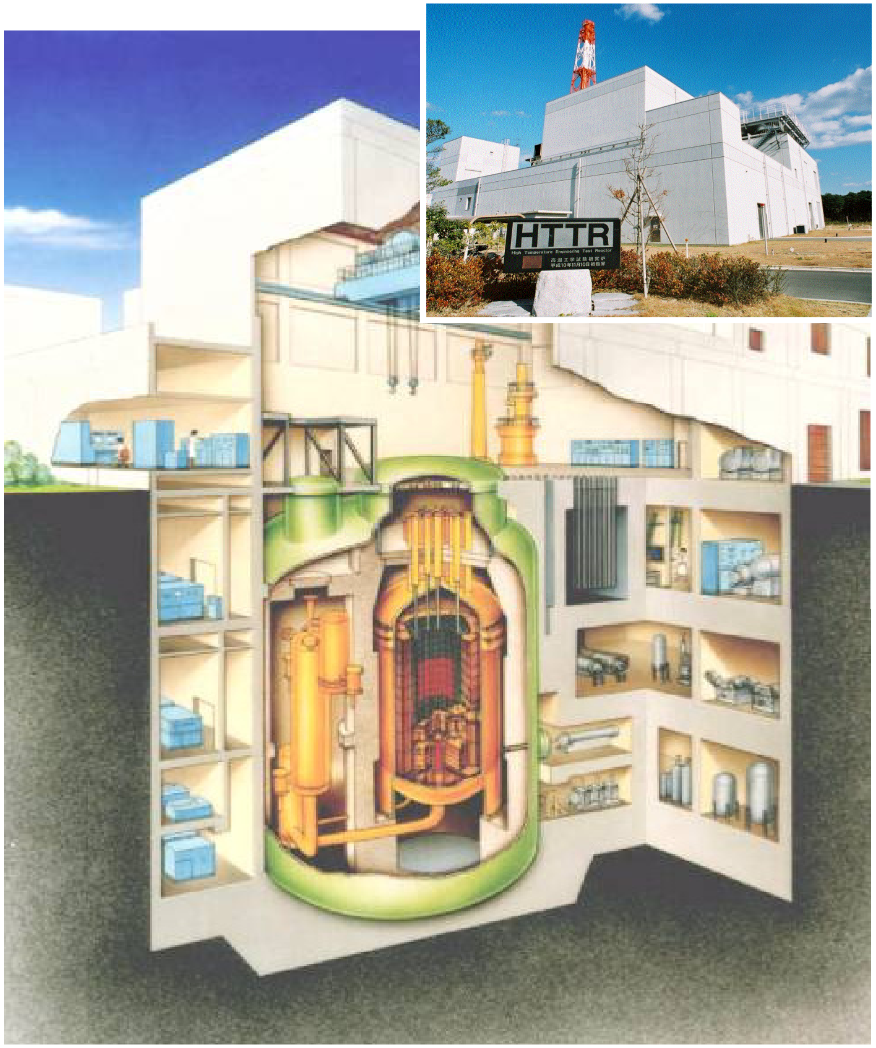


# JAERI Air Ingress Test Results



# 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** |

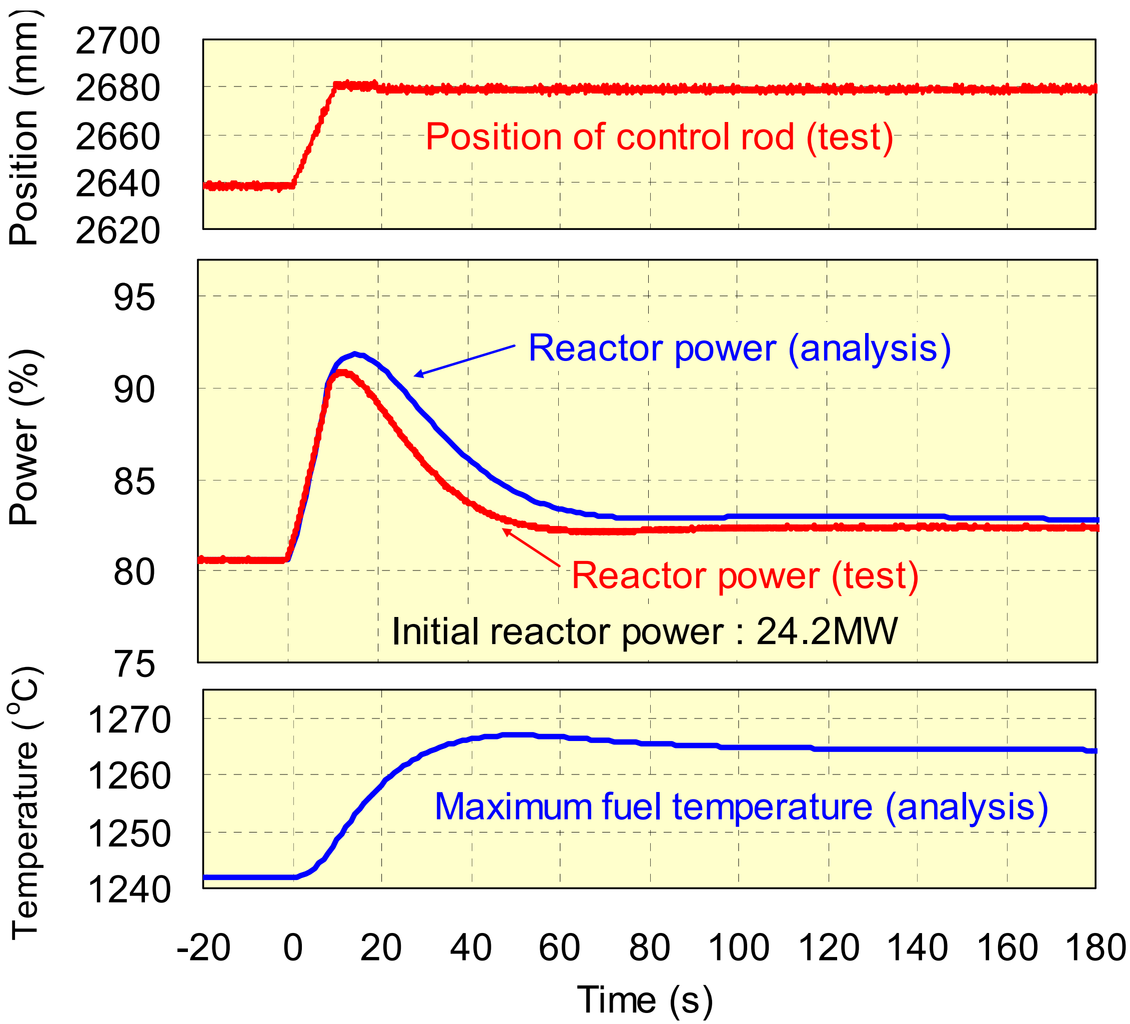
## History

- 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

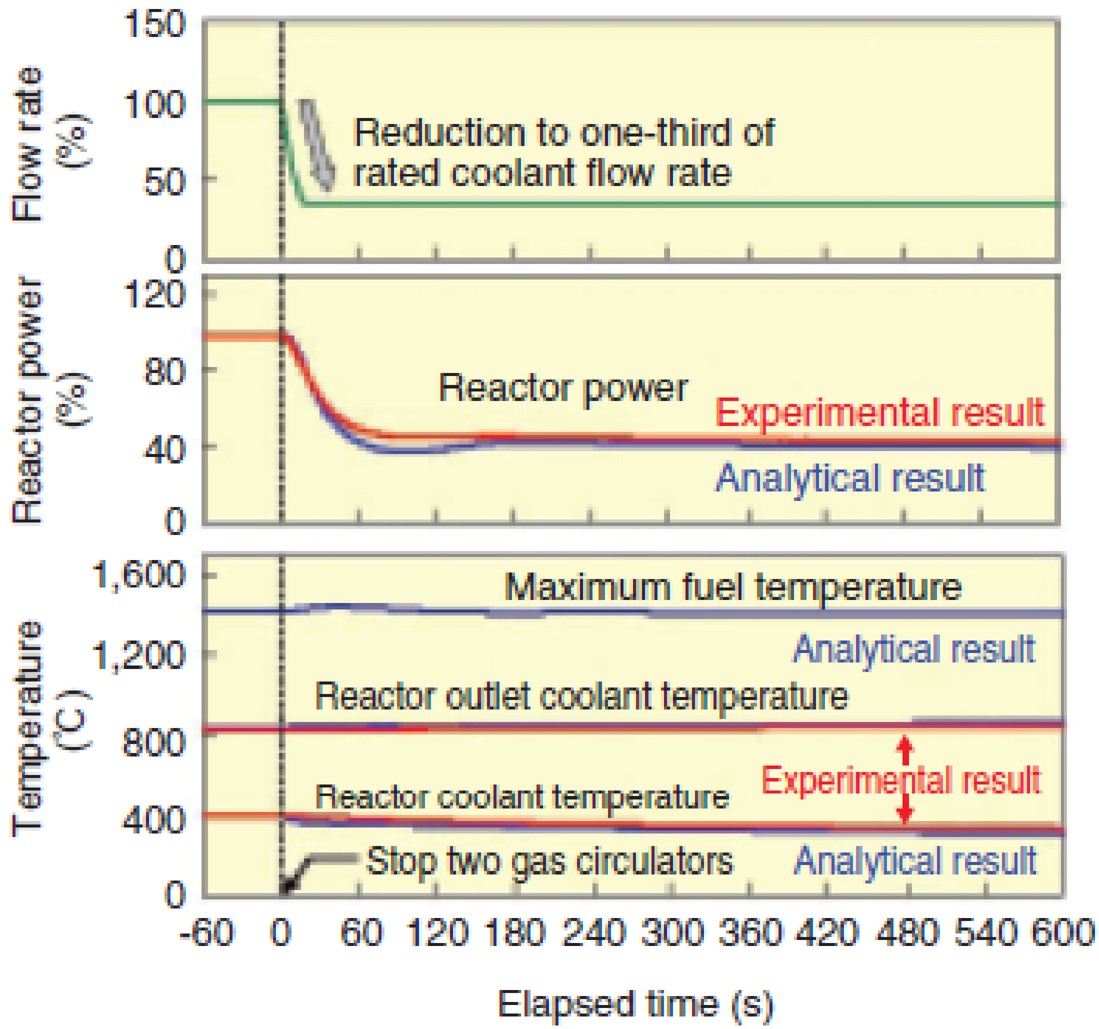



*K. Takamatsu, et al.,*
*Journal of Power and Energy Systems*
*Vol. 2 (2008) , No. 2, p.790-803*

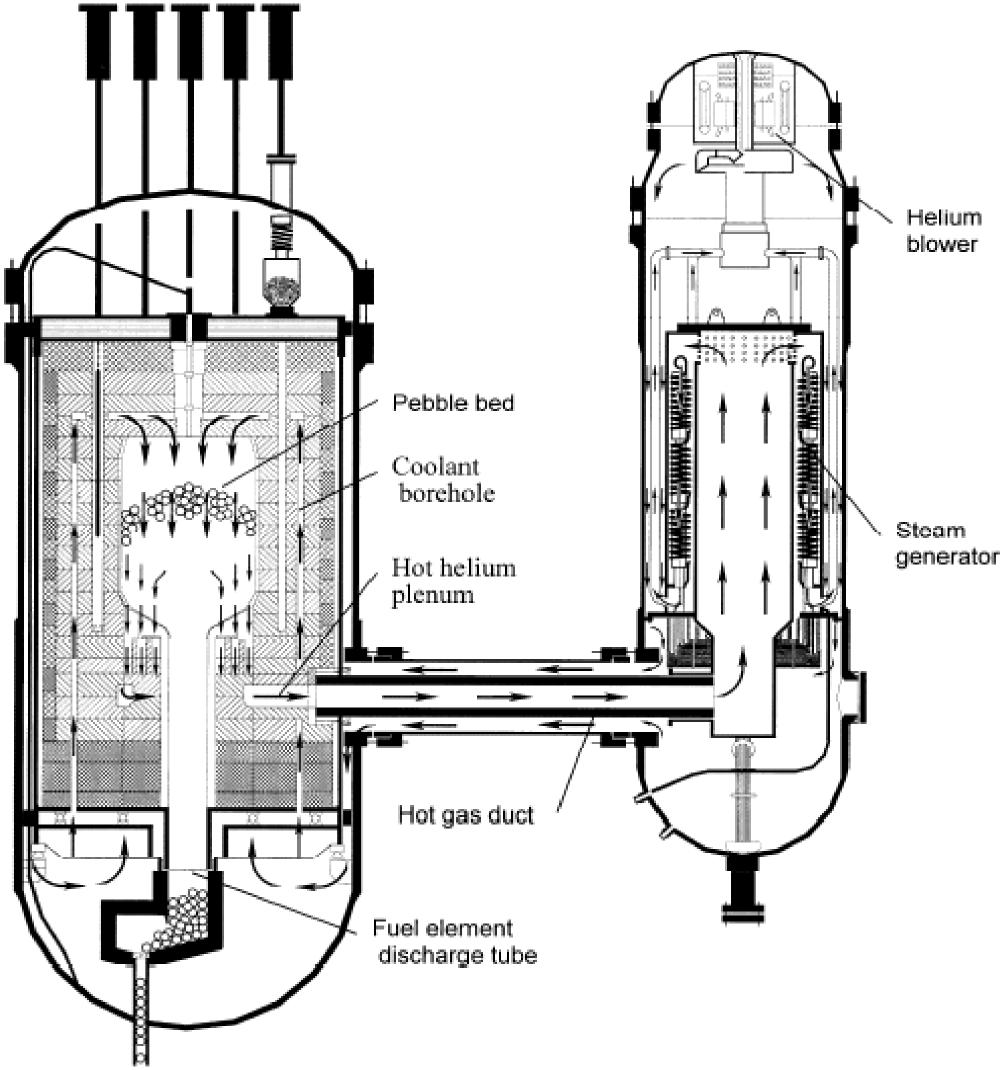
# Coolant Flow Reduction Test in Japan's HTTR

- **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



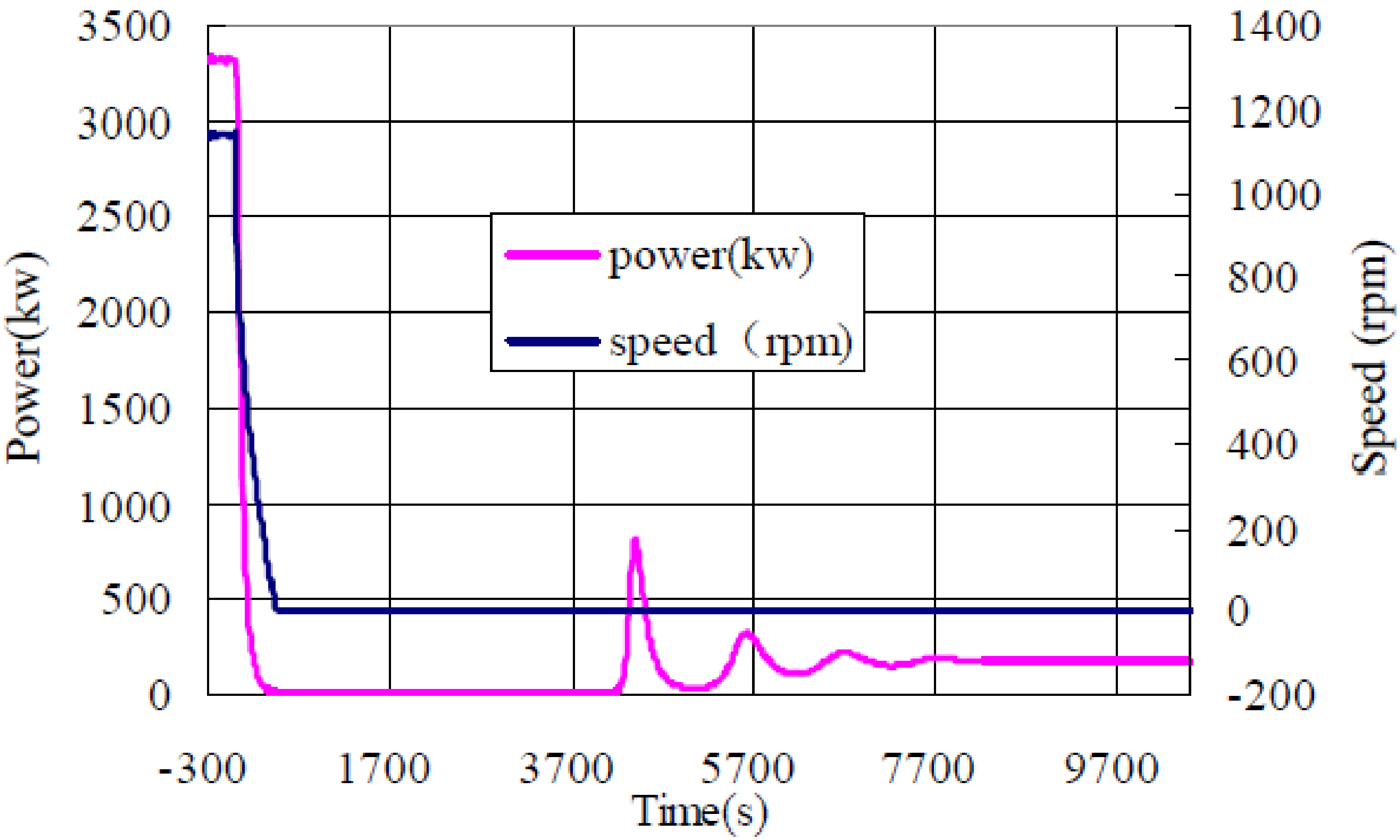
# China's HTR-10



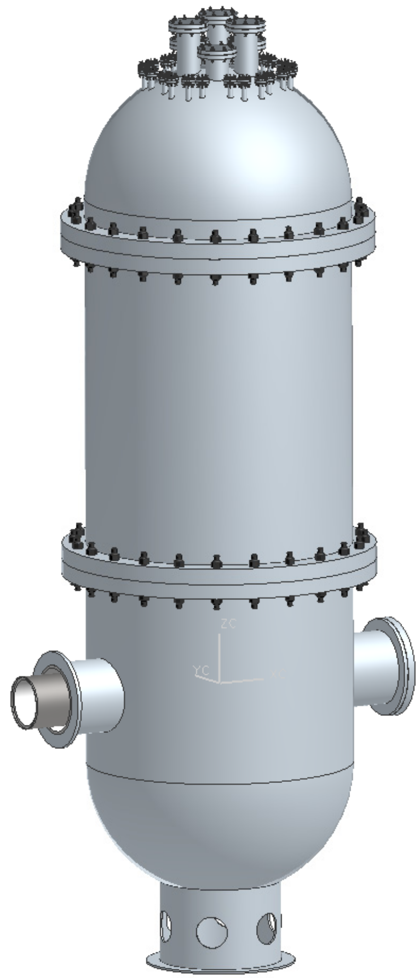
# HTR-10 Safety Demonstration Tests

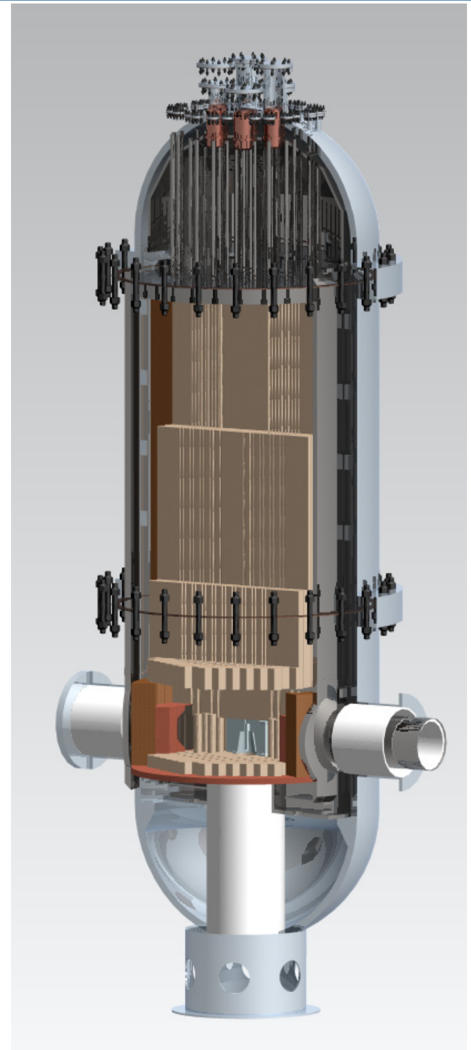
- Loss of offsite power without counter-measures
- Main helium blower shutdown (LOFC) with ATWS
- LOFC-ATWS with control rod withdrawal
- Loss of main heat sink without counter-measures

# HTR-10 LOFC and ATWS Test



# High Temperature Test Facility Planned at Oregon State University





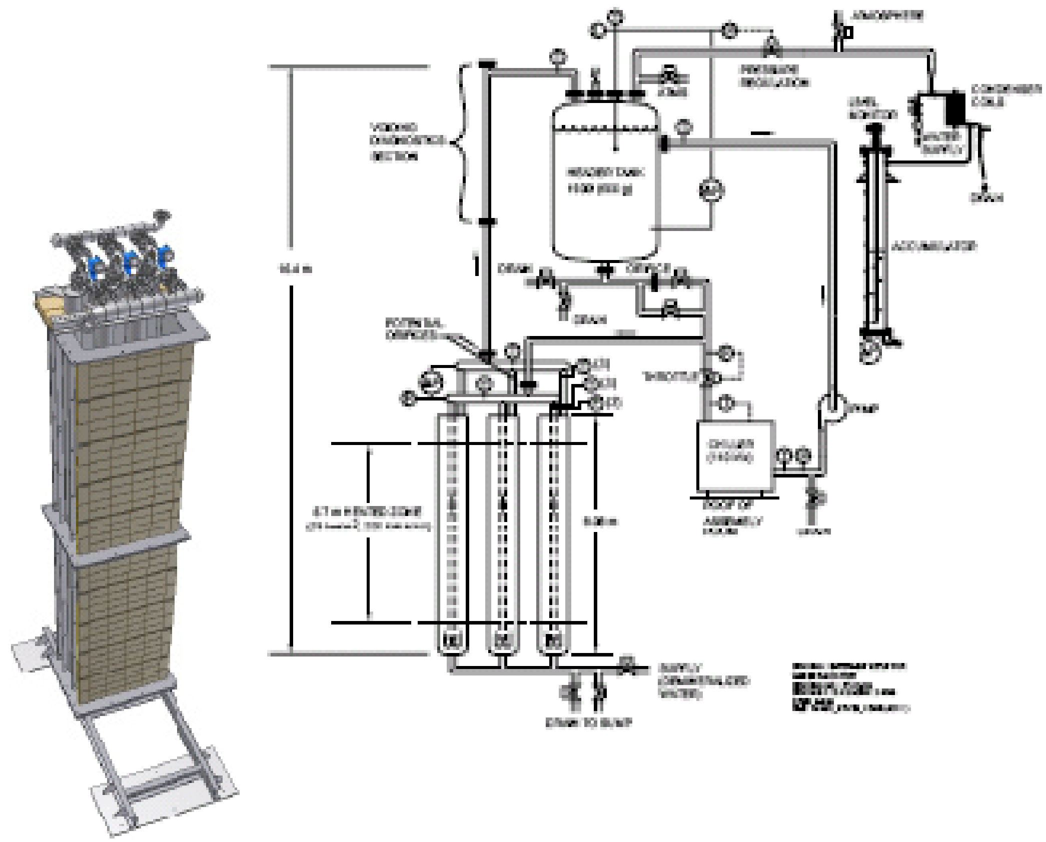
Oregon State UNIVERSITY

# 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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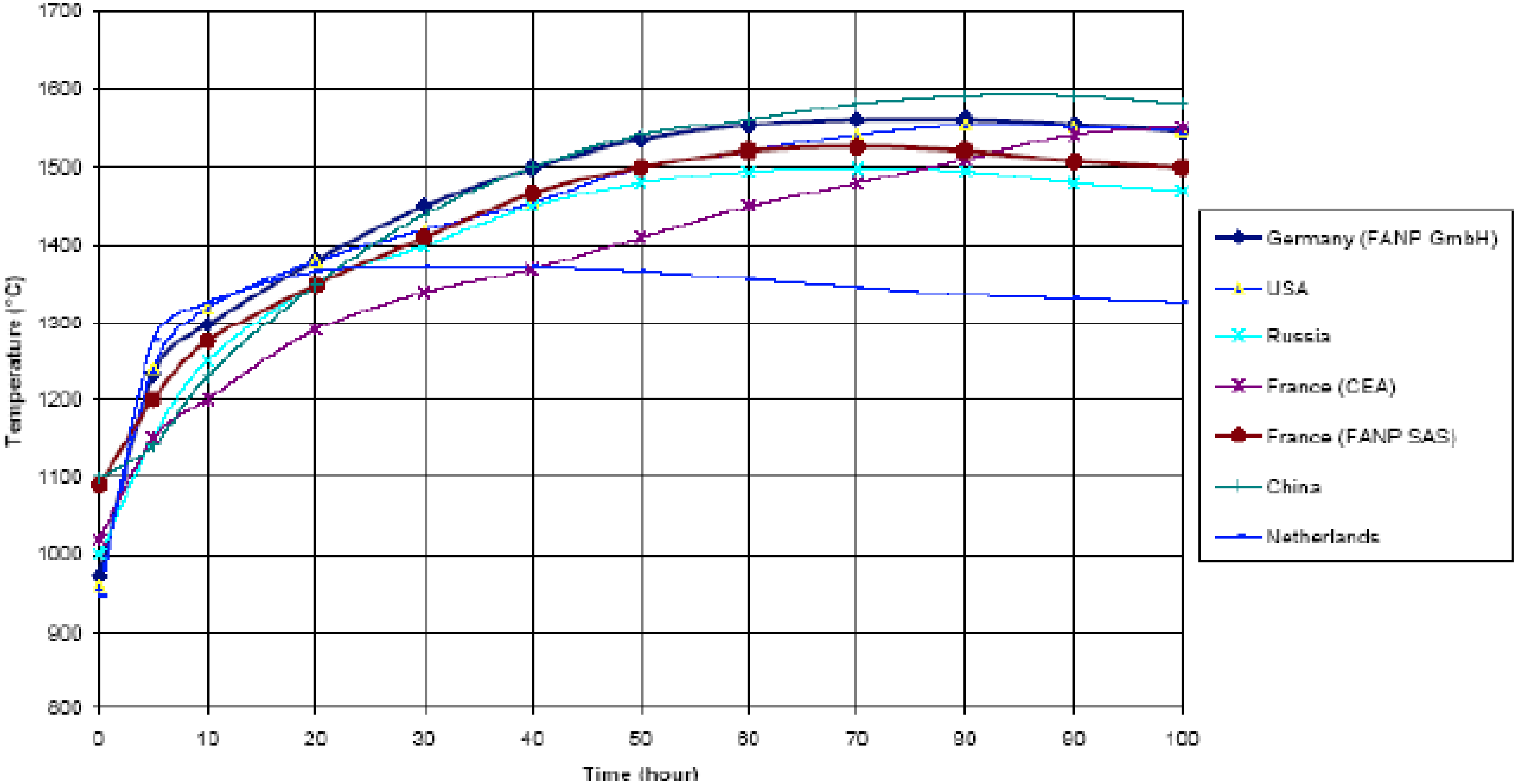
# Code Benchmarking, Verification, and Validation

- **"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

No

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

Yes

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.