DEVELOPMENT OF CORE DYNAMICS ANALYSIS OF COOLANT FLOW REDUCTION TESTS OF HTTR

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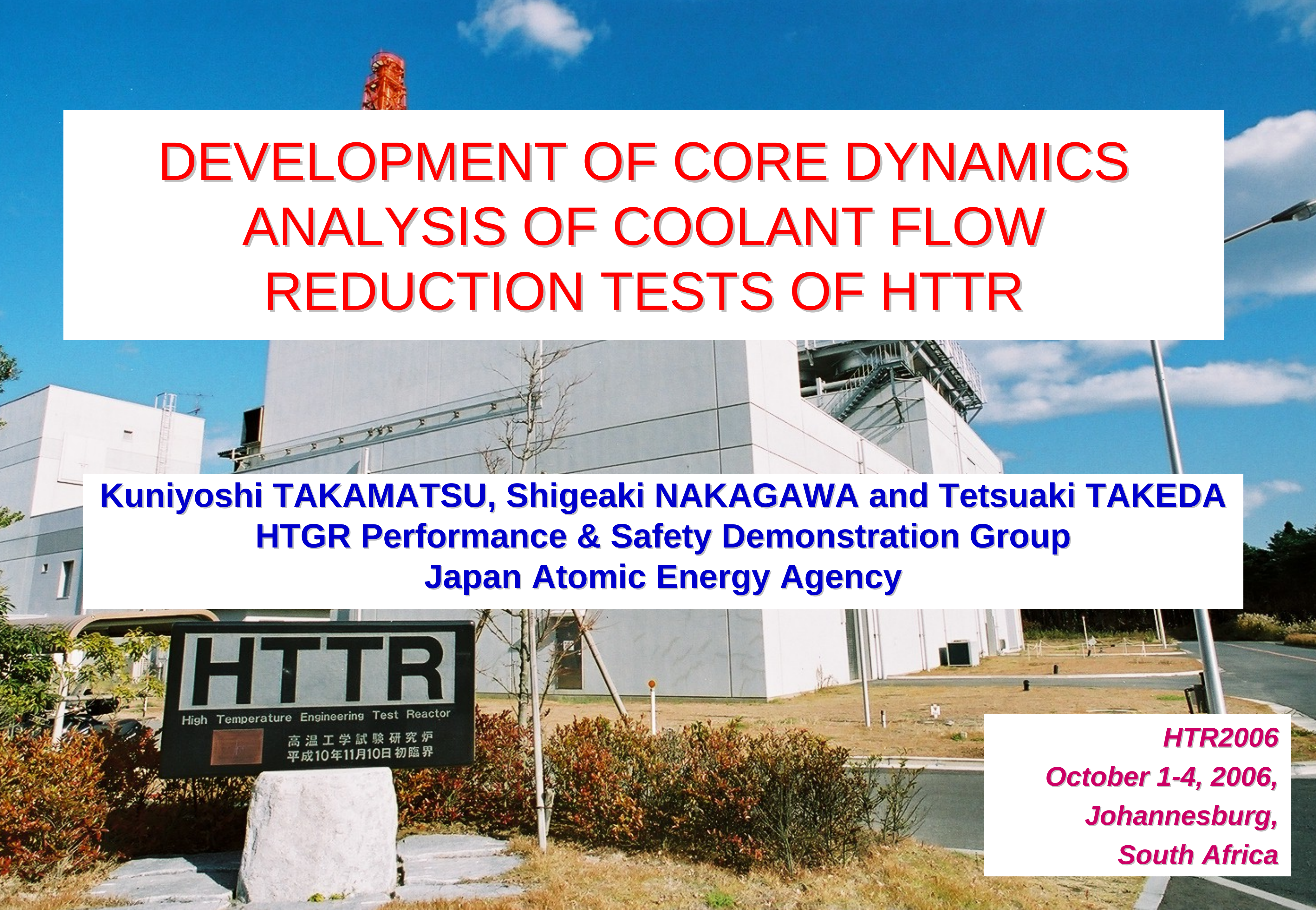
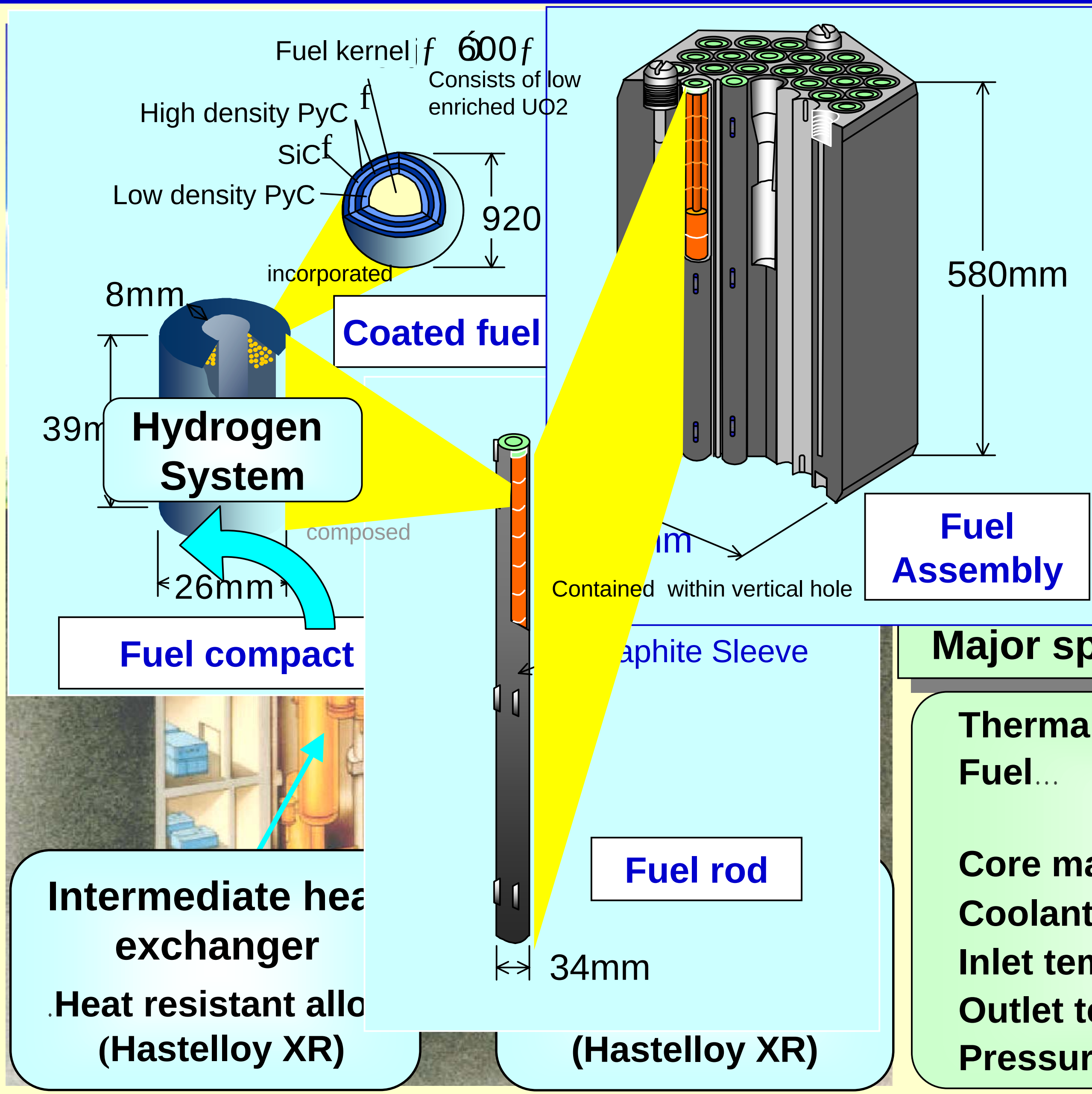
HTR2006
October 1-4, 2006,
Johannesburg,
South Africa



Outline of HTTR (High Temperature Engineering Test Reactor)



hment of HTGR technology

lation of long-term operation data

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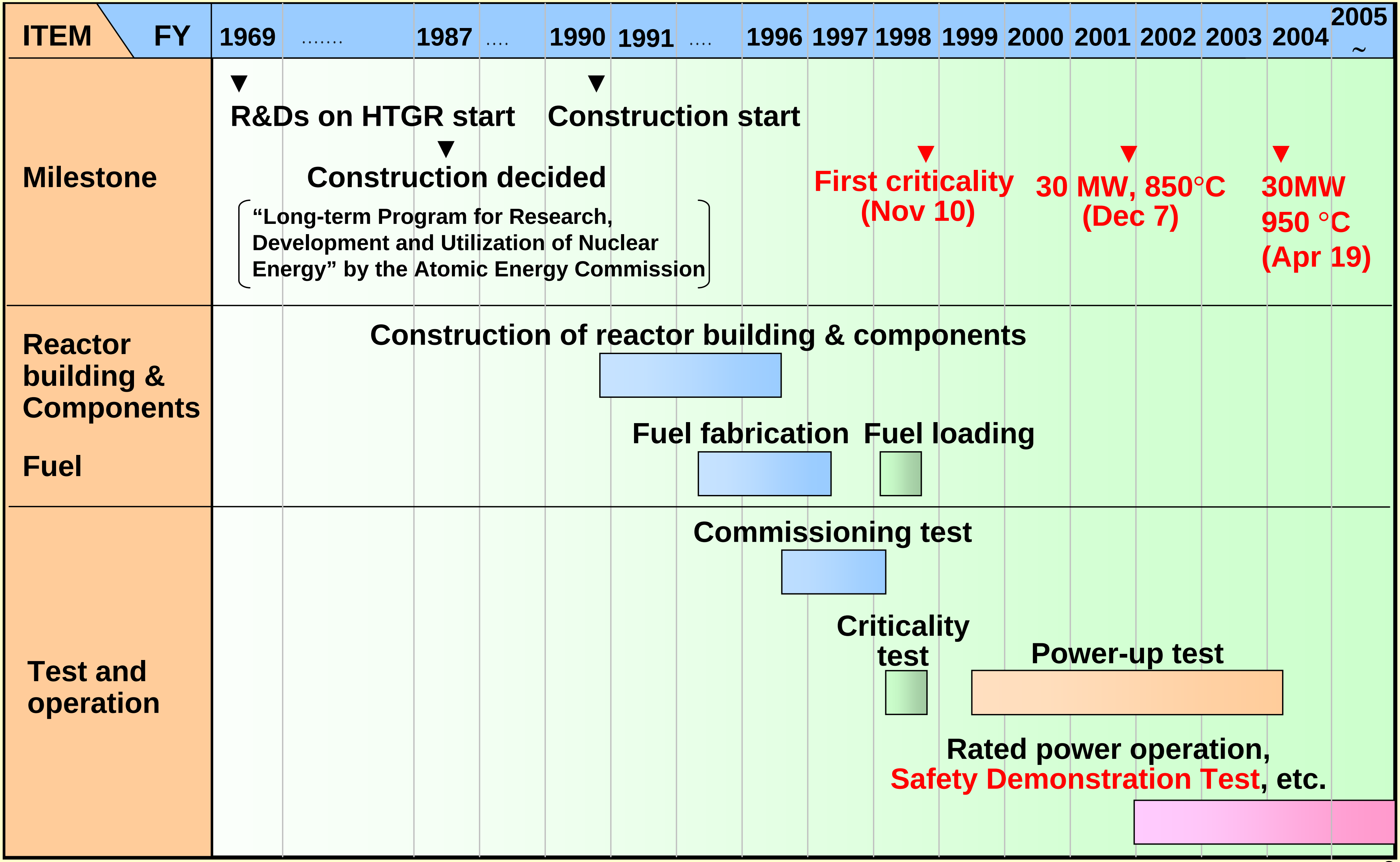
stration of hydrogen production

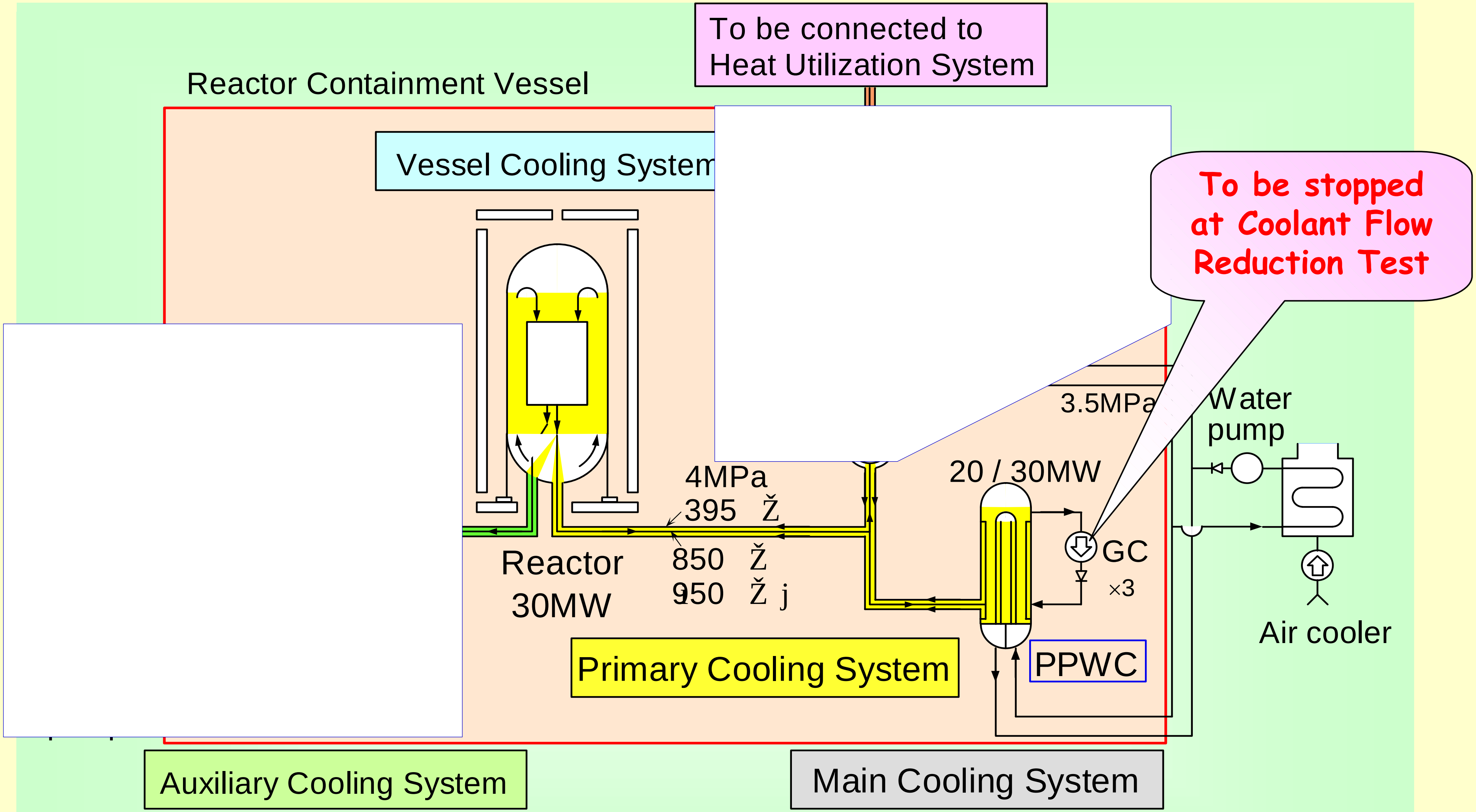
Major specification

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 of the HTTR Project



Schematic Diagram of HTTR Cooling System



IHX : Intermediate heat exchanger
PPWC: Primary pressurized water cooler

SPWC: Secondary pressurized water cooler
AHX : Auxiliary heat exchanger
GC : Gas circulator

Safety Demonstration Test - Objectives -

- **Demonstration of Inherent Safety Features of HTGRs using the Actual Reactor, HTTR**

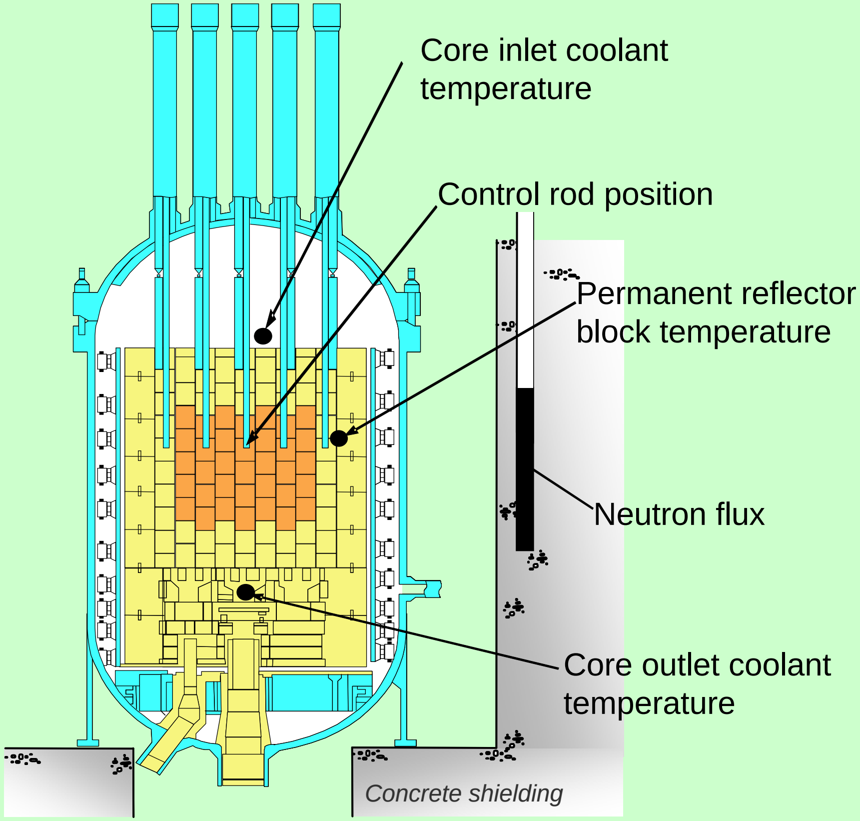
- **Improvement of Prediction Accuracy of Reactor Behaviour during Accidents**
  - **Code development and verification using transient data**

- **Improvement of Safety Evaluation Technologies of HTGRs**

- **Contribute to R&D for the Generation IV reactor VHTR**

Safety Demonstration Test
- Major Items of Measurement -

Parameter	Element
Core inlet coolant temperature	9 K-type thermo-couples
Core outlet coolant temperature	7 N-type thermo-couples
Permanent reflector block temperature	9 K-type thermo-couples
Neutron flux	3 uncompensated ionization chamber
Control rod position	16 encoder sensors

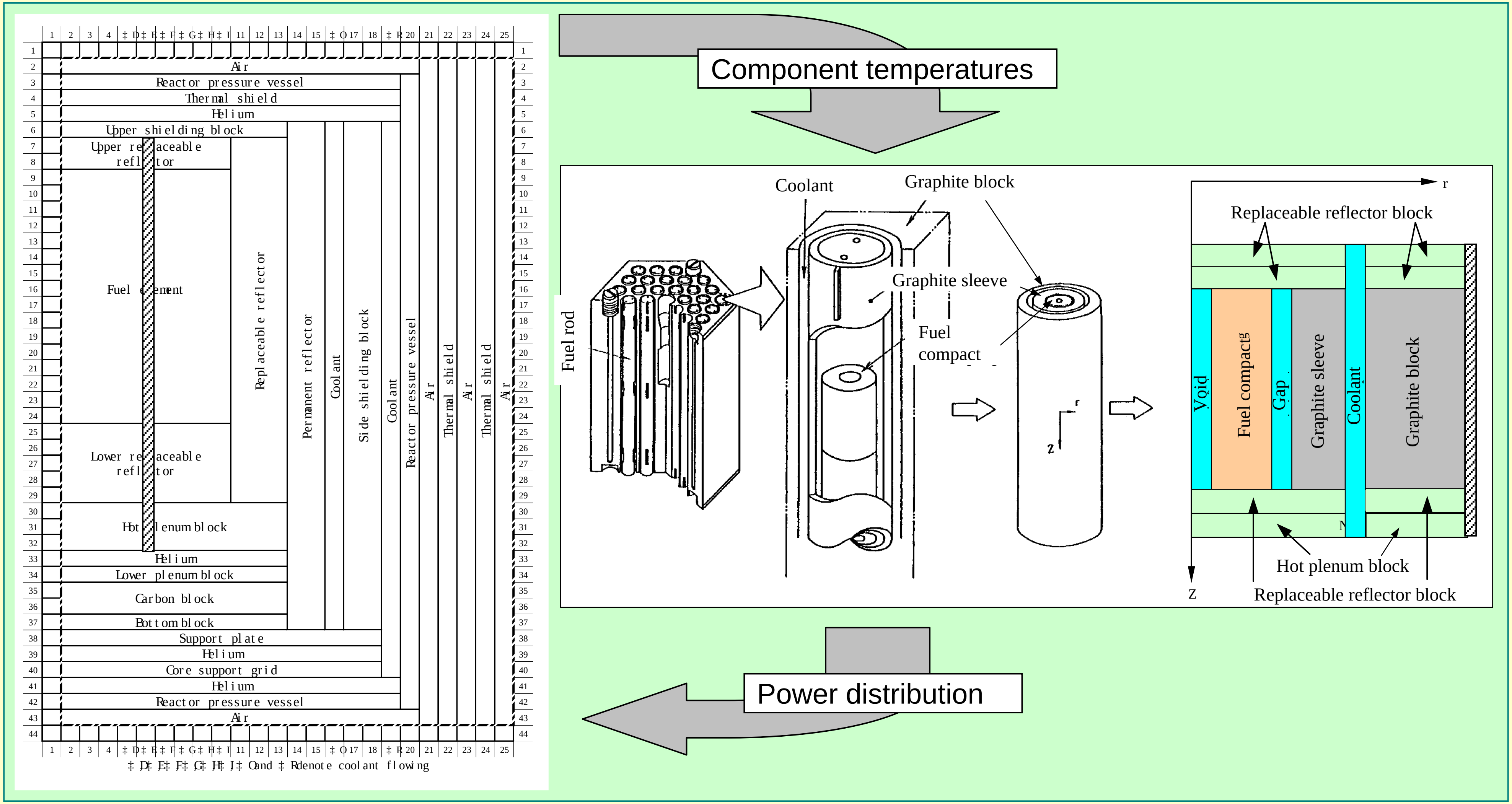


Safety Demonstration Test
- Major points for modeling -

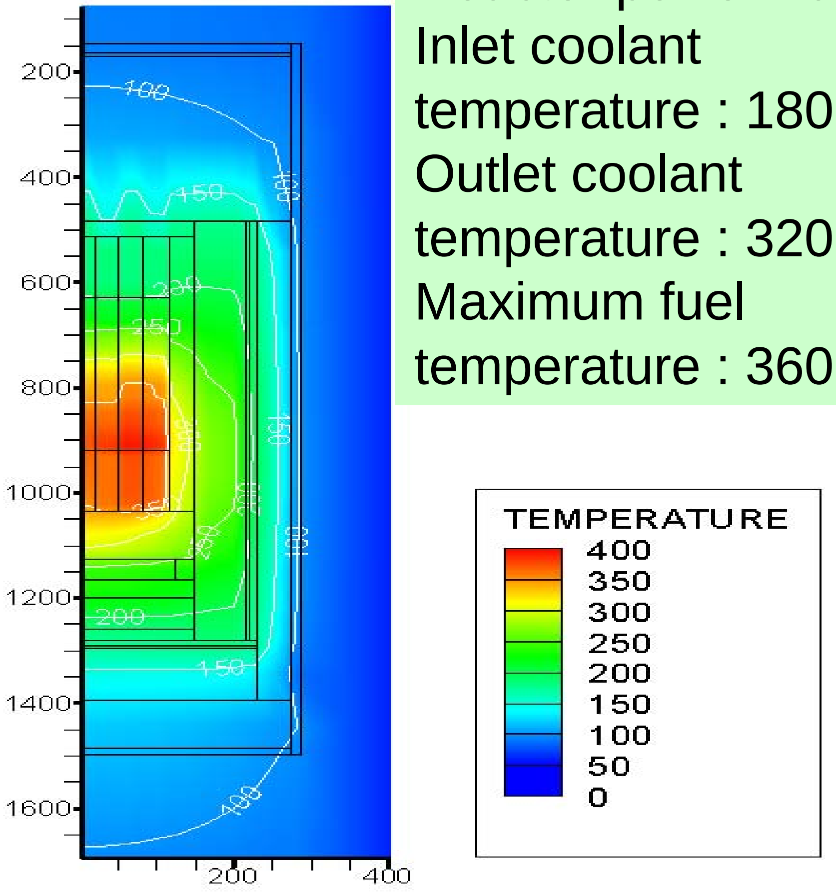
- The range of modeling: RPV, RPV internals and the structure of the vessel cooling system (VCS).
- A two-dimensional cylindrical model is adopted for calculating the temperature distribution in the core.
- The fuel element in the core is assumed to be homogeneous.
- The flow paths in the core: the cross section, the velocity and the pressure drop are equivalent to the real ones.
- Heat removal from the surface of the RPV and thermal radiation to the cooling panel of the VCS are considered, although heat removal by convection is disregarded.
- The amount of heat removal by the VCS is designed to be from 0.3 MW to 0.6 MW in order to achieve a reactor outlet gas temperature of 850 - 950°C.
- The analytical model of the core and the fuel channel are coupled for the data transfer, such as the component temperature and the power distribution.

Safety Demonstration Test

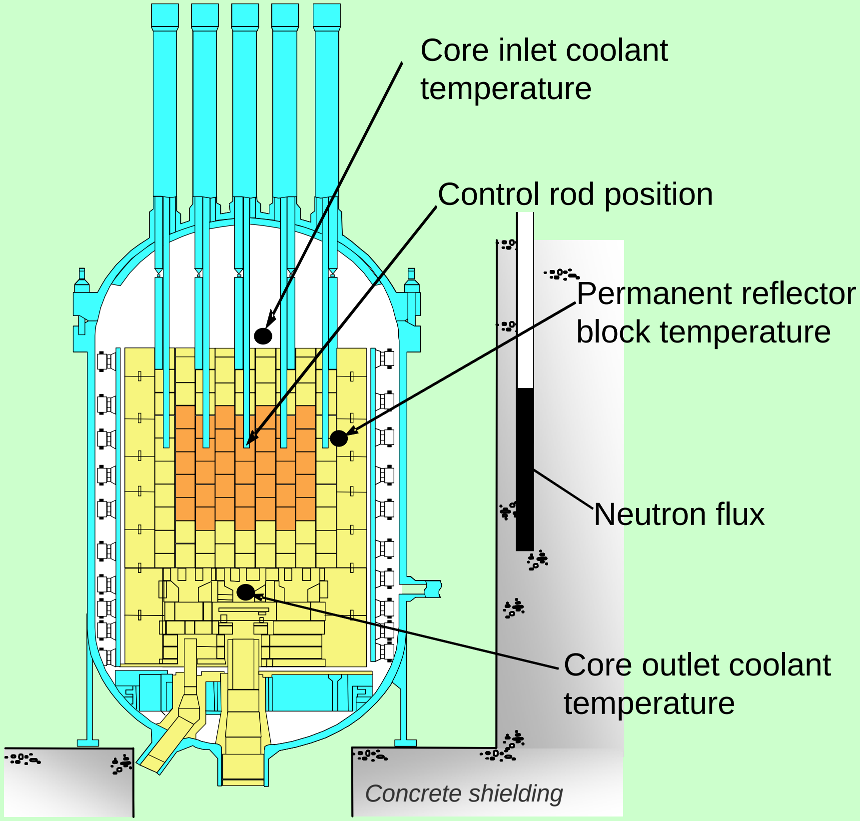
- Analytical model -



Safety Demonstration Test
- Steady state temperature distribution -



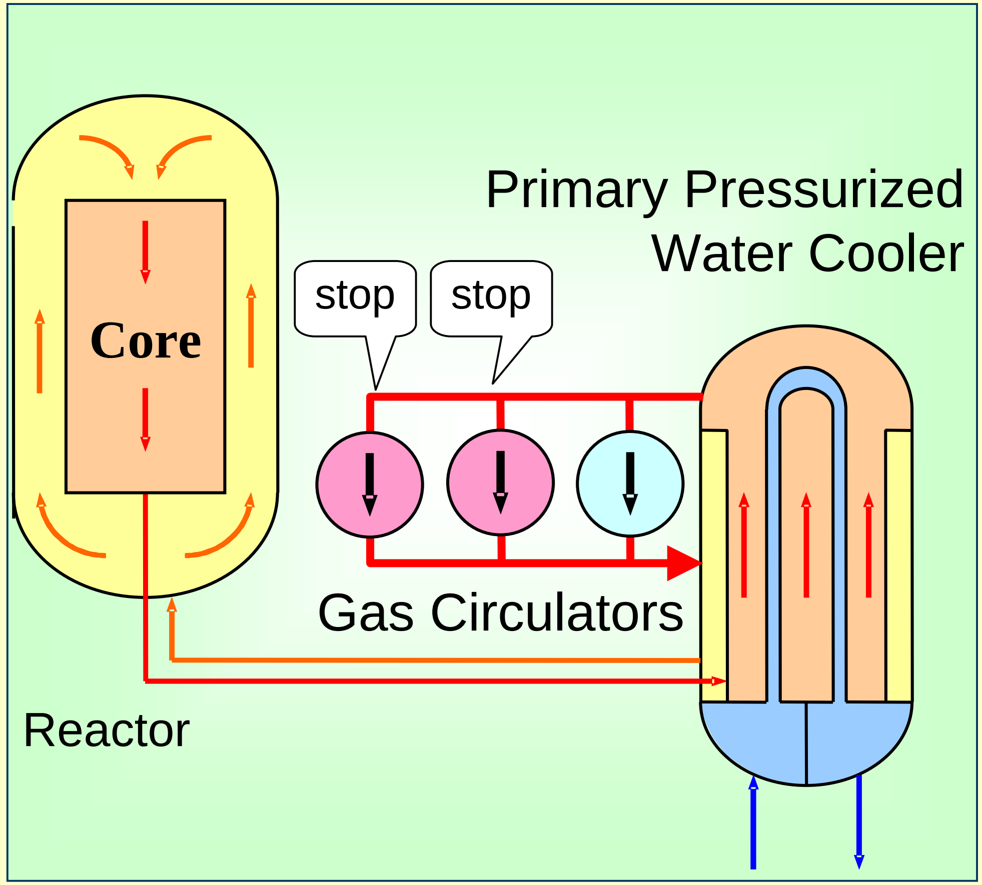
Reactor power : 9MW
Inlet coolant temperature : 180°C
Outlet coolant temperature : 320°C
Maximum fuel temperature : 360°



Coolant Flow Reduction Test - Test Method -

Test Conditions

- Reactor Power 30%(9MW)
- Reactor Outlet Coolant Temperature
  - Initial Below 850°C
  - During Test Below 950°C
- Gas Circulators to be Stopped 1 or 2 (out of 3)
- Reactor Power Control System Disabled*



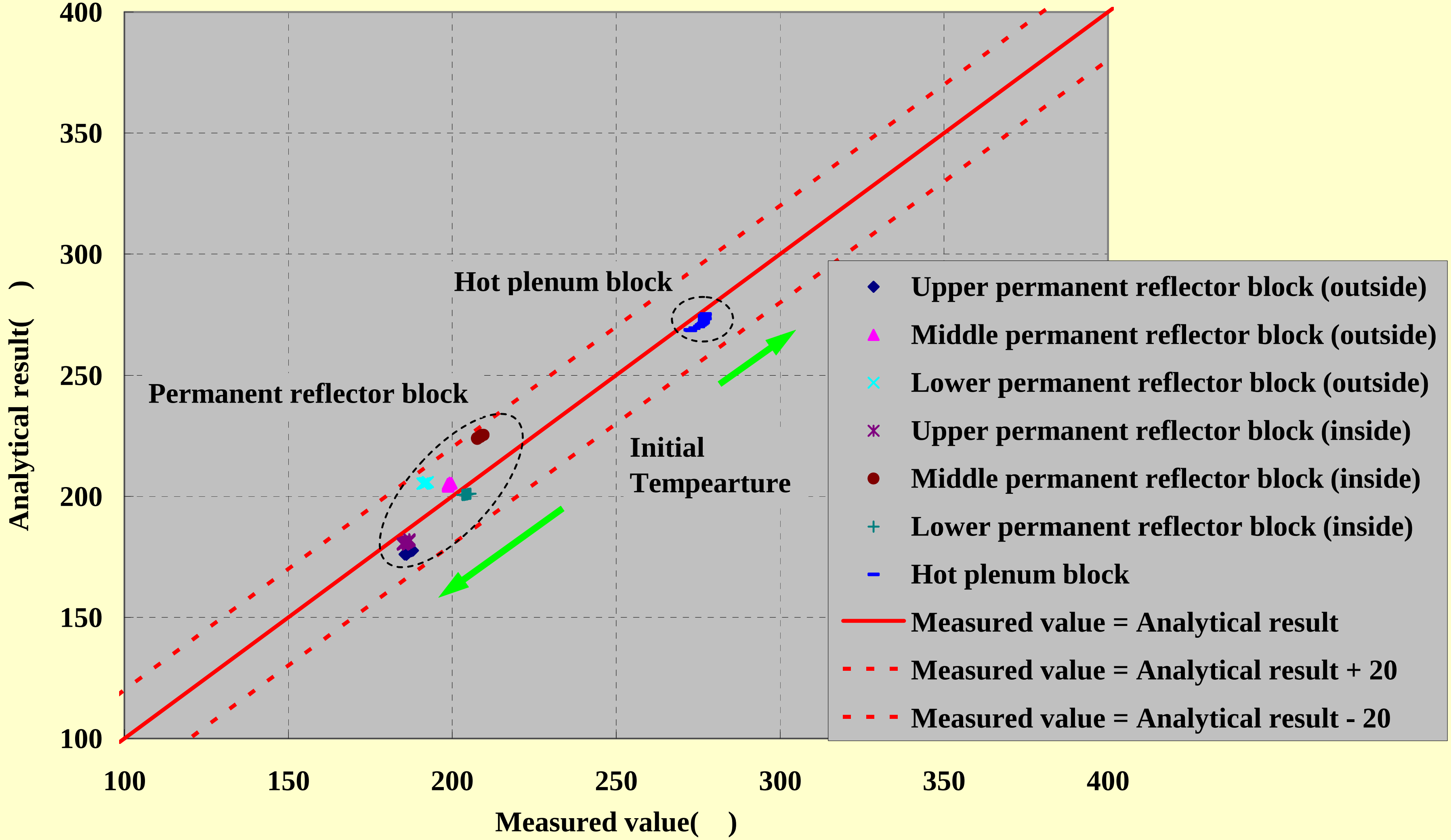
* Power supply for driving all control-rods is cut off.

** Scram set-values of primary coolant flow rate (Low), etc. are modified to prevent a reactor scram.

Transient Data of Reactor Power, Reactor Inlet & Outlet Coolant Temperature, etc. are obtained.

Transient temperature distribution

- One gas circulator trip test -

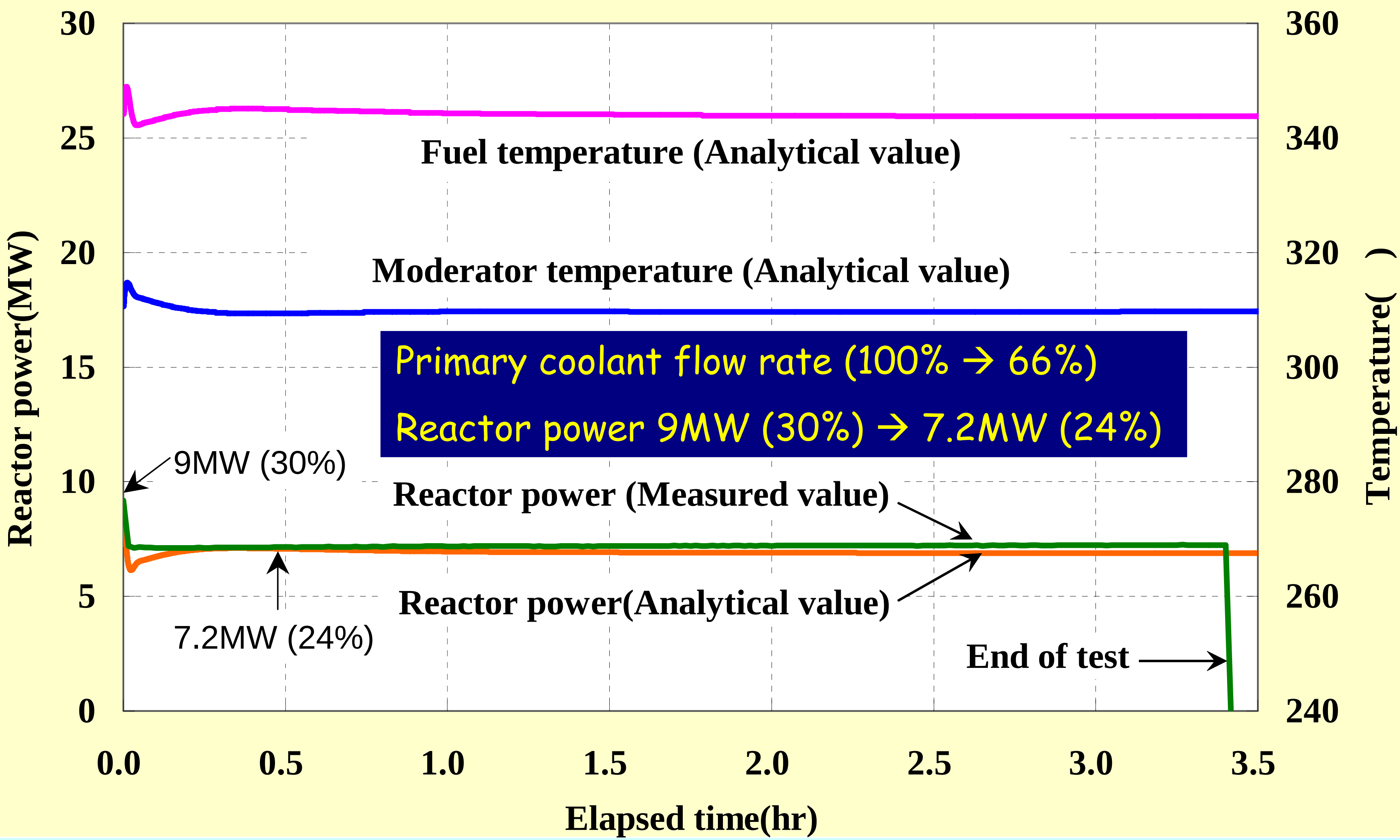


Inherent Safety Features: Large Heat Capacity of Core to Power Density
Negative Reactivity Feedback Effect of Core

-**Temperature Transient of the Reactor Core is Slow**.

Transient reactor power and temperature distribution

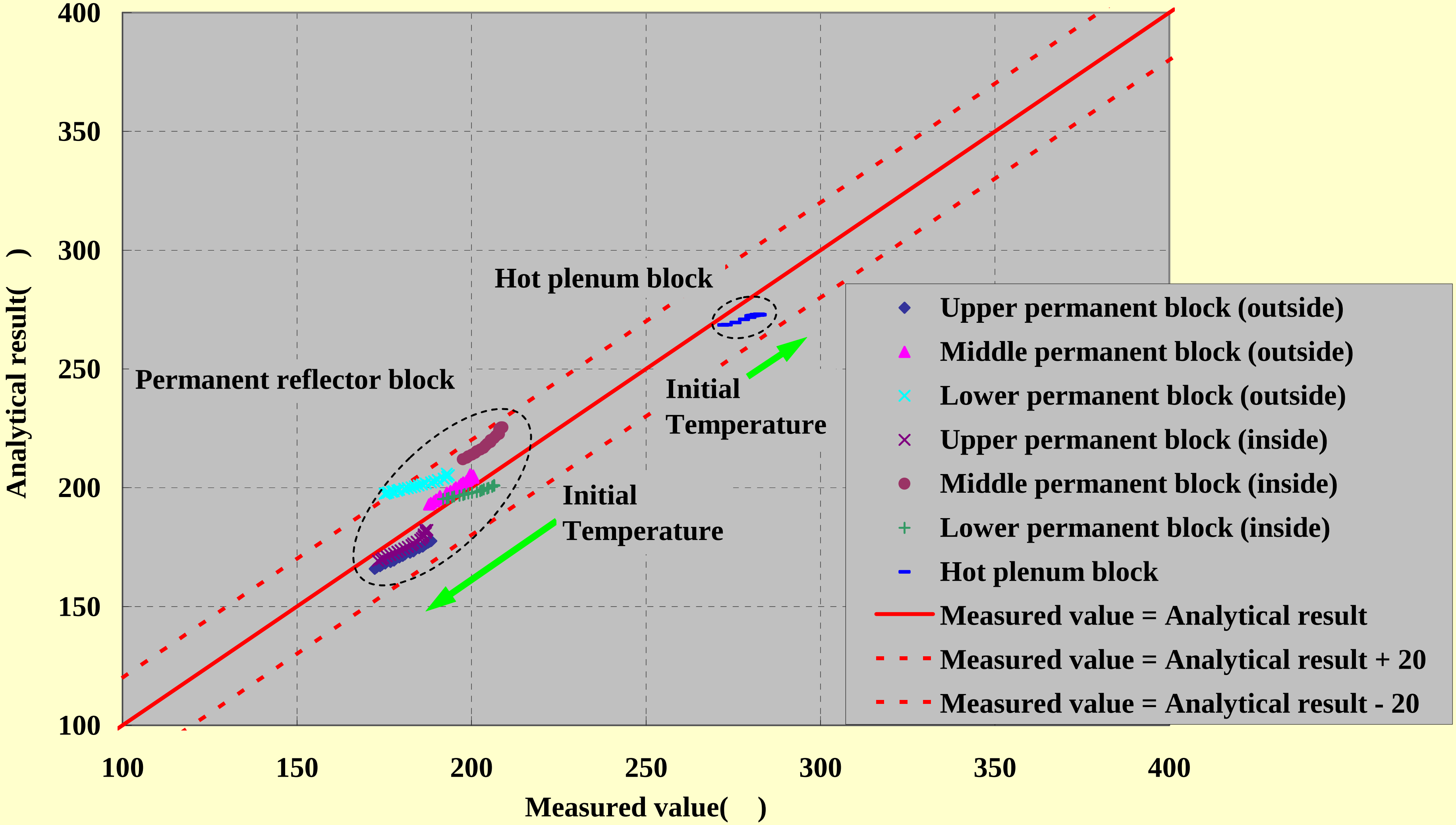
- One gas circulator trip test -



Coolant Flow Reduction → Core Temperature Raise → Negative Reactivity Feedback Effect of the Core → Reactor Power Decreased to a Stable Level without a Scram

Transient temperature distribution

- Two gas circulators trip test -



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Transient reactor power and temperature distribution

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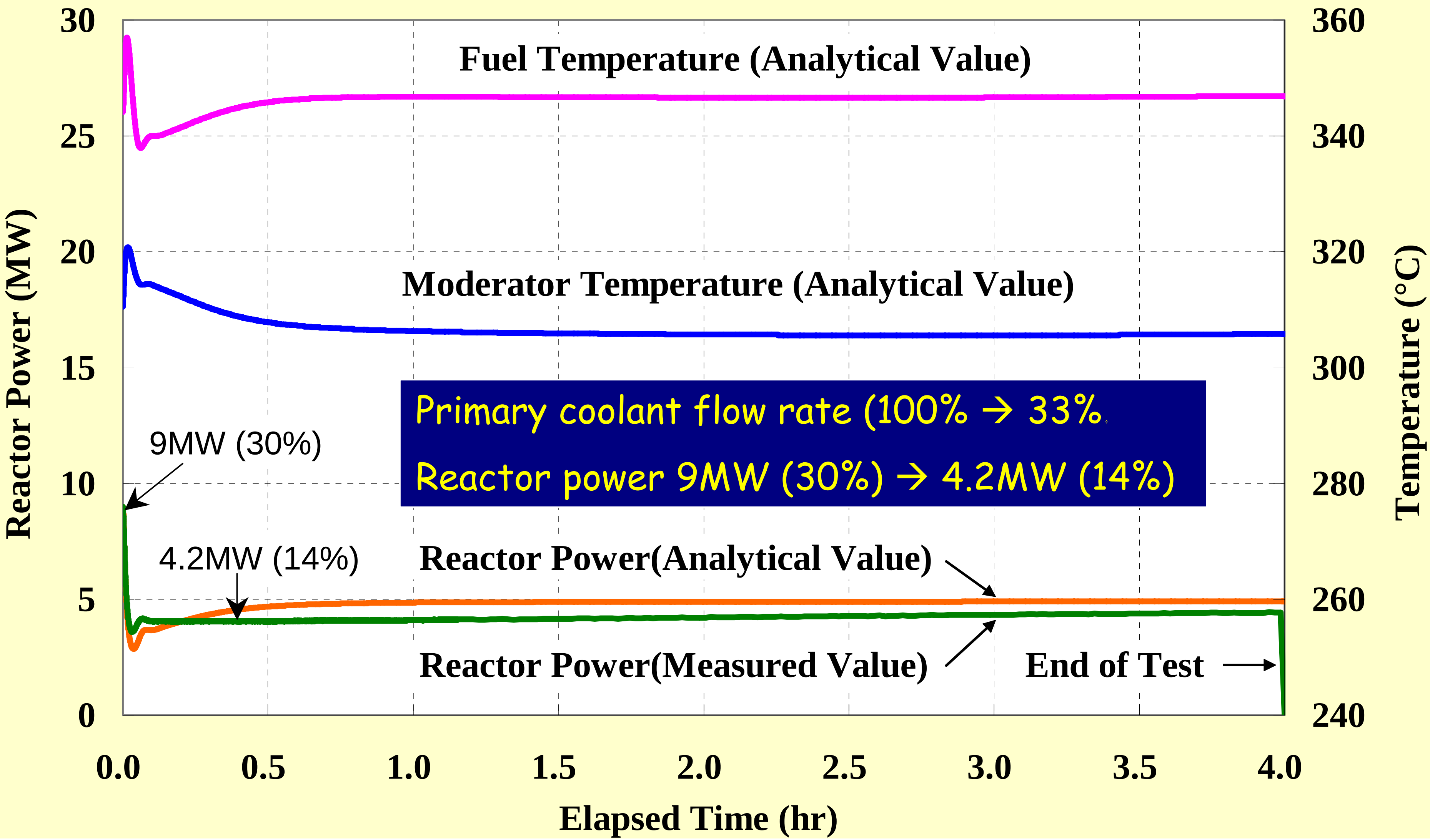

Coolant Flow Reduction → Core Temperature Raise → Negative Reactivity Feedback Effect of the Core → Reactor Power Decreased to a Stable Level without a Scram

Concluding Remarks

- Safety demonstration tests using the HTTR started in March 2003 to verify the inherent safety features and to improve the safety evaluation technologies of HTGRs.

- Coolant flow reduction test by tripping one and two gas circulators have been conducted. The reactor power was decreased to a stable level without a scram (ATWS).

- A comparison of the measured values and the analytical results showed a good agreement.

- Obtained results are very useful for validation of codes for safety evaluation and expected to contribute to commercialization of HTGRs such as the VHTR system selected as the Generation IV system.