The Benefits of Nuclear Technology for Social Inclusion

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"Very High Temperature Reactors. Current Status and Challenges"

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Nuclear Energy

Advantages

- *No emissions of green house gases (GHG).*
- *Low volume of waste in relation to the large energy production.*
- *Economic competitiveness.*

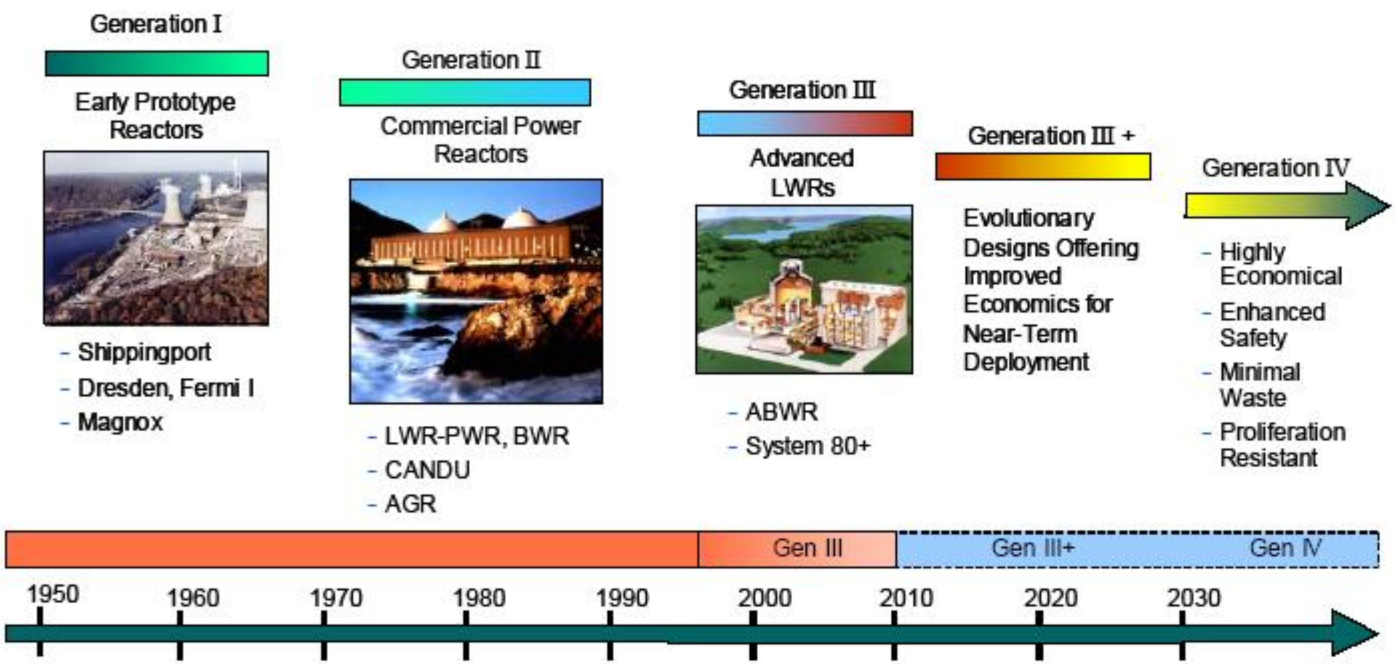
Nuclear Energy

Disadvantages

- *Long-term radiotoxicity of nuclear waste.*
- *Need for high safety requirements.*
- *High investment cost.*
- *Plutonium proliferation risk.*

From the point of view of sustainability of nuclear energy, nuclear waste management and its inventory minimization are the most important issues that should be addressed.

Various reactor concepts are being studied to comply requirements of future nuclear fuel cycles such as safety, sustainability, cost-effectiveness and proliferation risk reduction



These generation IV reactor concepts include both thermal and fast reactors and are being developed in conjunction with advanced reprocessing technologies.

Generation IV Nuclear Energy Systems

The GIV roadmap process culminated in the selection of six systems. The motivation for the selection of six systems is to:

- *Identify systems that make significant advances toward the technology goals.*
- *Ensure that the important missions of electricity generation, hydrogen and process heat production, and actinide management may be adequately addressed*
- *Accommodate the range of national priorities and interests of the GIV countries.*

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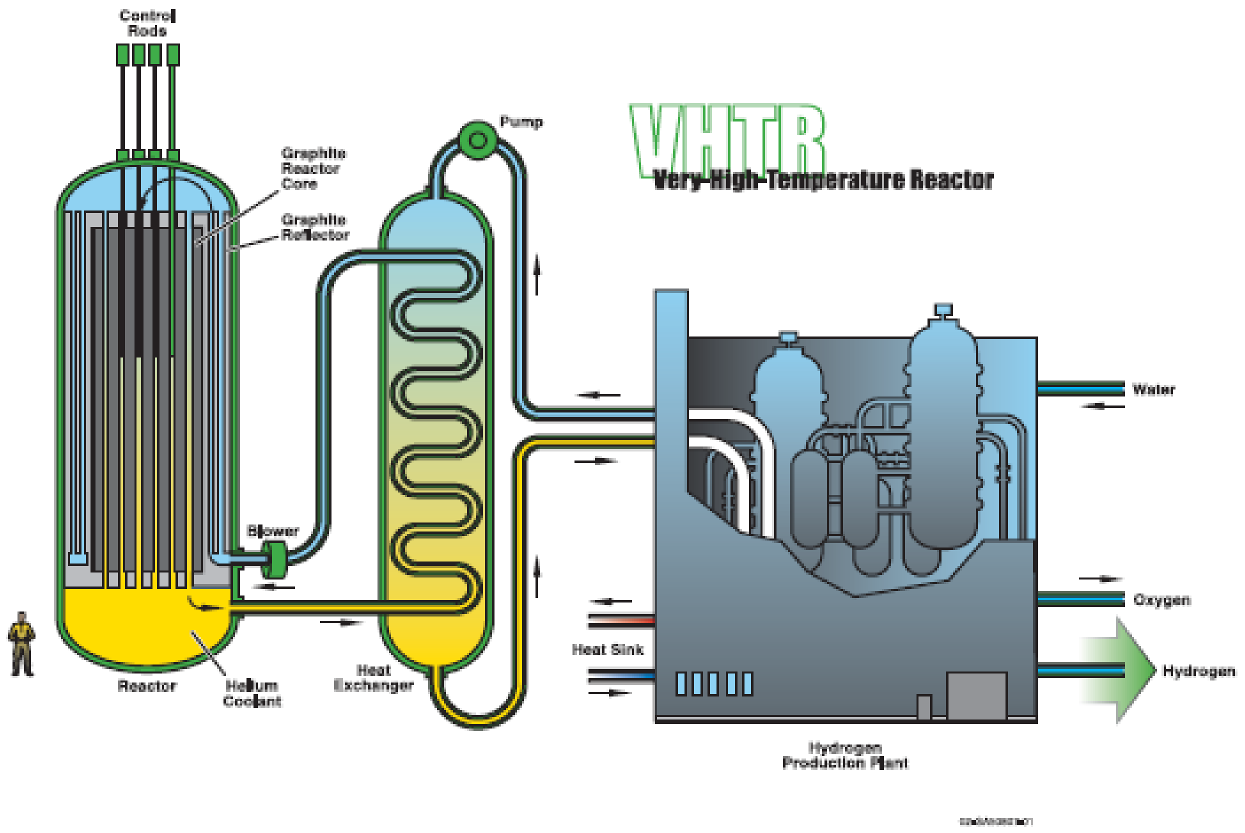
Generation IV of Nuclear Energy Systems

Generation IV System	Acronym
Gas-Cooled Fast Reactor System	GFR
Lead-Cooled Fast Reactor System	LFR
Molten Salt Reactor System	MSR
Sodium-Cooled Fast Reactor System	SFR
Supercritical-Water-Cooled Reactor System	SCWR
Very-High-Temperature Reactor System	VHTR

The VHTR is a graphite-moderated helium-cooled reactor with thermal neutron spectrum and a once-through uranium cycle



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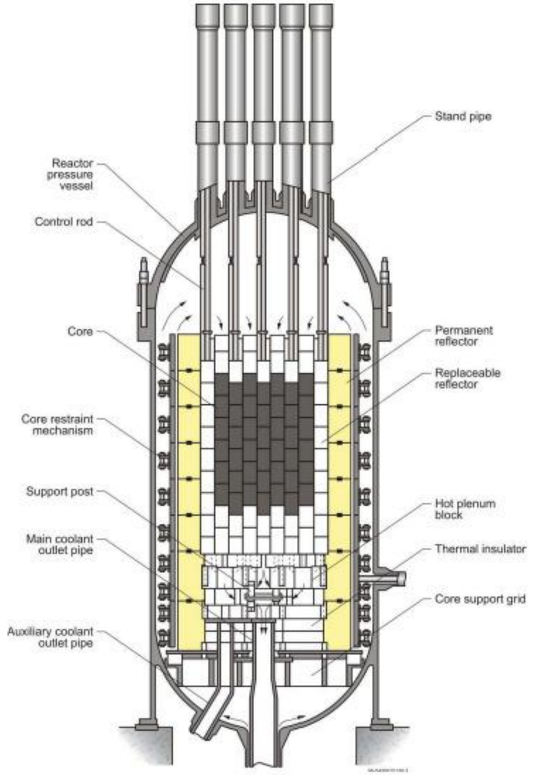
The reactor core type can be a prismatic block core such as the operating Japanese HTTR, or a pebble-bed core such as the Chinese HTR-10



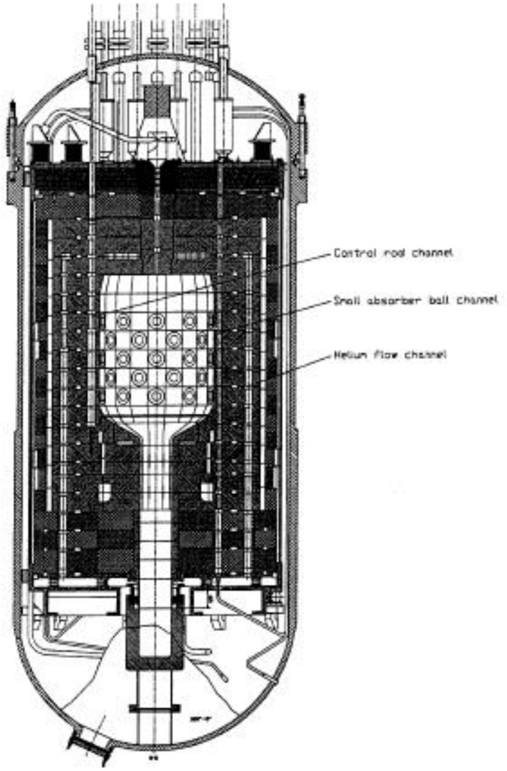


FIG. 4.6. HTR-10 reactor vertical cross-section.

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The development of gas-cooled nuclear reactor technology promises improved performance in:

- ***sustainability***
- ***economics***
- ***proliferation resistance***

*As a result of these efforts, **nuclear energy** will increase its contribution **to the reduction of CO_2 emissions** when it is used **to replace conventional sources of process heat.***
***By investing in gas-cooled reactor technology**, is endeavouring **to jump-start** a new application for nuclear energy with potential benefits to the environment.*

Advantages of Very High Temperature Reactors

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Clear passive safety mechanism

1. The high graphite inventory provides ***significant thermal inertia***.
2. **Graphite** also has a ***high thermal conductivity***, which facilitates the transfer of heat to the reflector, and it can withstand high temperatures.
3. The ***strongly negative power reactivity coefficient*** gives a negative feedback, such that the reactor shuts down by itself in case of a loss of coolant accident.

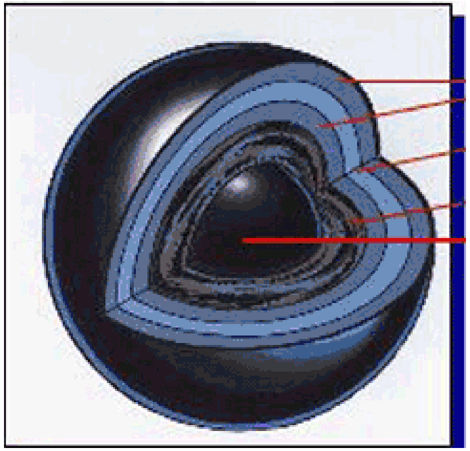
Clear passive safety mechanism

- The high quality of fuel elements – tri-isotropic (TRISO) coated particles – **minimizes operational and accidental fission gas release**. The materials selected have resistance to high temperatures.

- The **low power density** enables stabilization of core temperature significantly below the maximum allowable, even in case of incidents **such as loss-of-coolant accident**.

- No possible **buildup** of explosive hydrogen mixtures.

- **Helium** is inert and is not activated.



Clear passive safety mechanism

- **Emergency cooling systems** are less necessary.
- Decay heat can be removed **through natural mechanism**, such as heat conduction, heat radiation, etc. (Inherent safety)

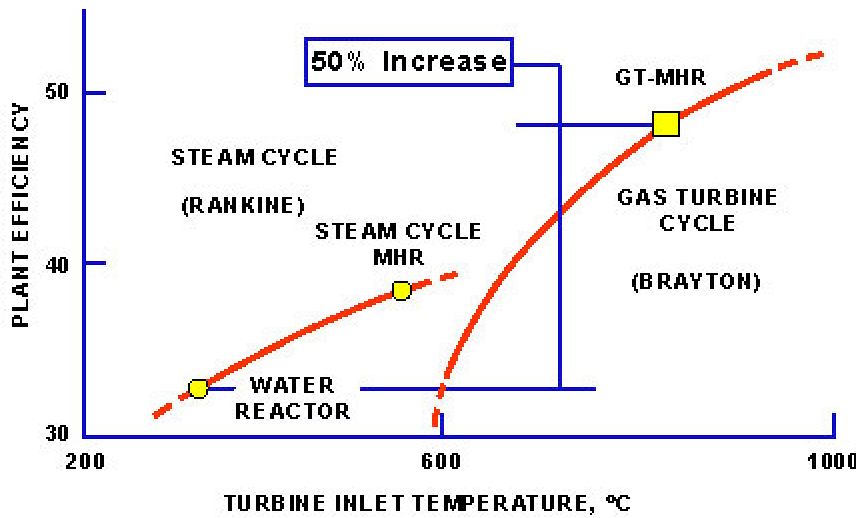
Together, the aforementioned aspects **prevent the massive release of fission products in the case of an accident.**

Advantages of Very High Temperature Reactors

High conversion energy efficiency

- Electricity generation with high efficiency due to **high core outlet temperature.**
- Possible use of **Brayton** vs. Rankine Cycle. (LWR)
- Steam turbine. (Plant efficiency near 42 %)
- **Gas-turbine.** (Plant efficiency increase about 50%)
- **Cogeneration.** (The high efficiency power conversion of HTRs is pointed out in particular in cogeneration mode)

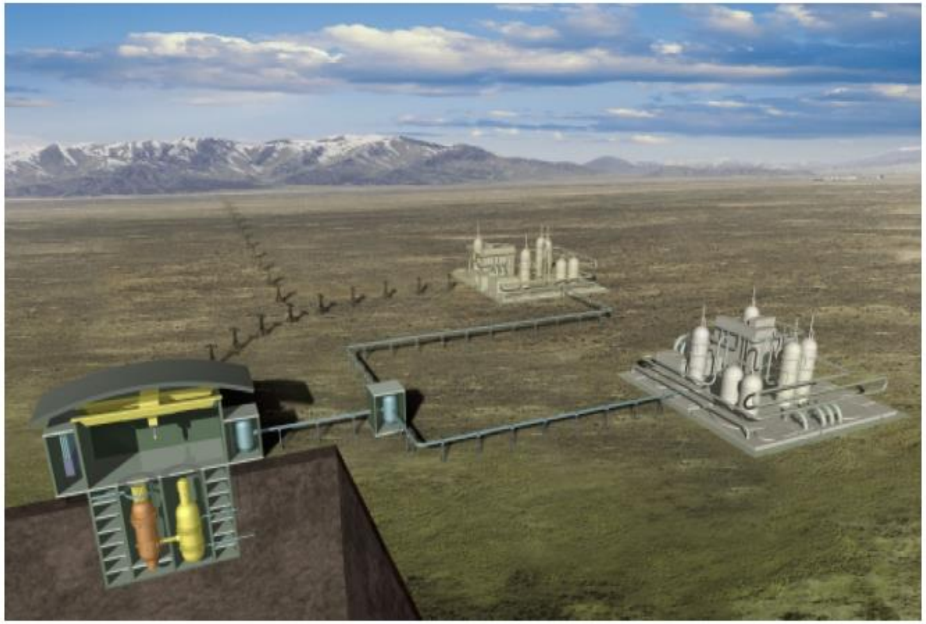
High conversion energy efficiency



High conversion energy efficiency also reduces the thermal impact on the natural environment.

High core outlet temperature

- ***Hydrogen generation methods from nuclear energy:***
  - High Temperature Electrolysis
  - Thermo Chemical Process I-S



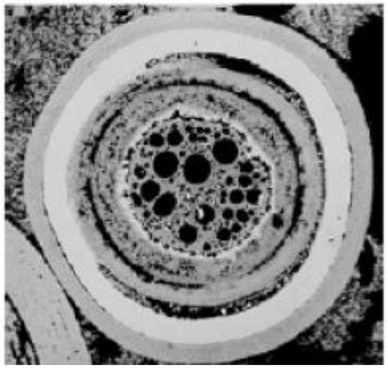
Cogeneration and heat supply

VHTRs can extend the benefits of nuclear energy beyond the electrical grid by providing industry with carbon-free, high-temperature process heat for a variety of applications.

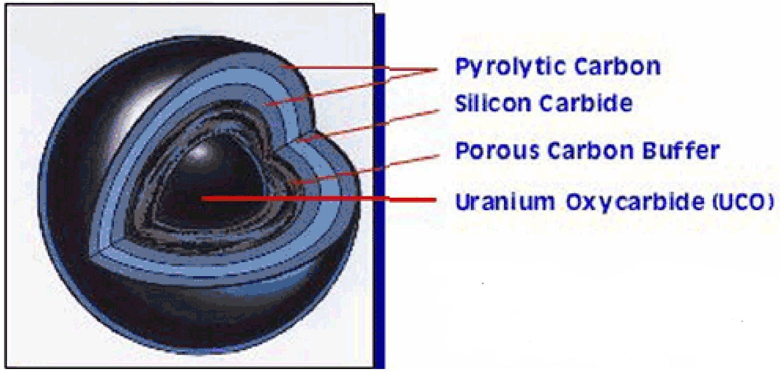
- petroleum refining, biofuels production and production of chemicals fertilizers.
- district heating, desalination, pulp and paper.
- lime, non-ferrous metals, iron and steel, glass, cement and ceramics.
- industrial gases and hydrogen.

High fuel burn up

Deep burnup concept, (TRISO fuel) up to 700 GWd/t.
High waste actinide burnup capabilities, more than 90 % destruction of minor actinides and more than 94 % of ^{239}**Pu** makes this reactor a valid proposal for the **reduction of nuclear waste** and the **prevention of proliferation.**
Great flexibility in the choice of the fuel type: fertile and non-fertile cores (e.g.Th, U or Pu).
Once-through nuclear fuel cycle with minimal wastes.

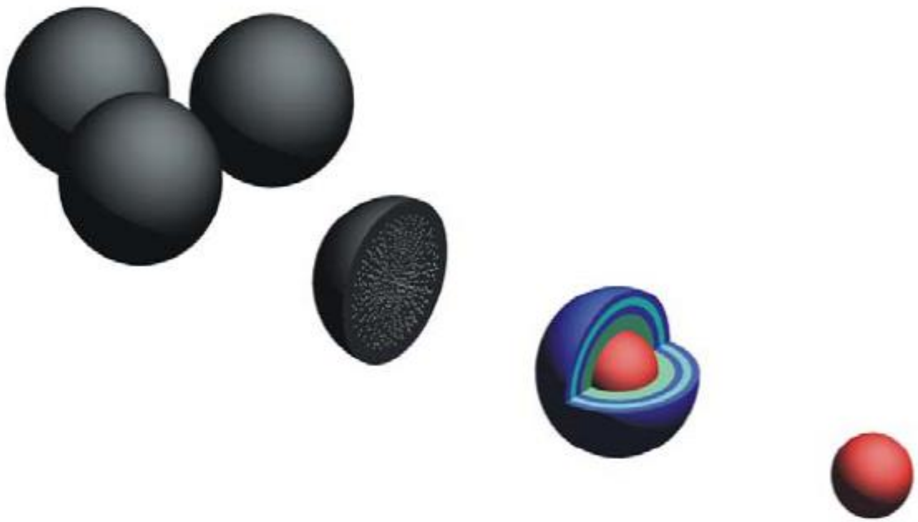


TRISO fuel



- ***Is stable** at high temperatures and **has very high melting points.***
- *Provides **large thermal margins** to ensure reactor integrity during loss of cooling events.*
- ***CFP** are nearly spherical and **include gas expansion volumes**, which will accommodate fission gas products and result in lower internal pressures.*
- *The spherical shape easily **withstands mechanical stresses** due to internal pressures.*

Ceramic coated particle fuel-TRISO



Fuel kernel
200-600 µm
UO_2, UC_2, UCO (Thorium and Plutonium)
Buffer
Porous carbon
Inner pyrolytic carbon (IPyC)
Dense, isotropic layer of carbon
Silicon carbide (SiC)
Dense, isotropic layer of SiC
Outer pyrolytic carbon (OPyC)
Dense, isotropic layer of carbon

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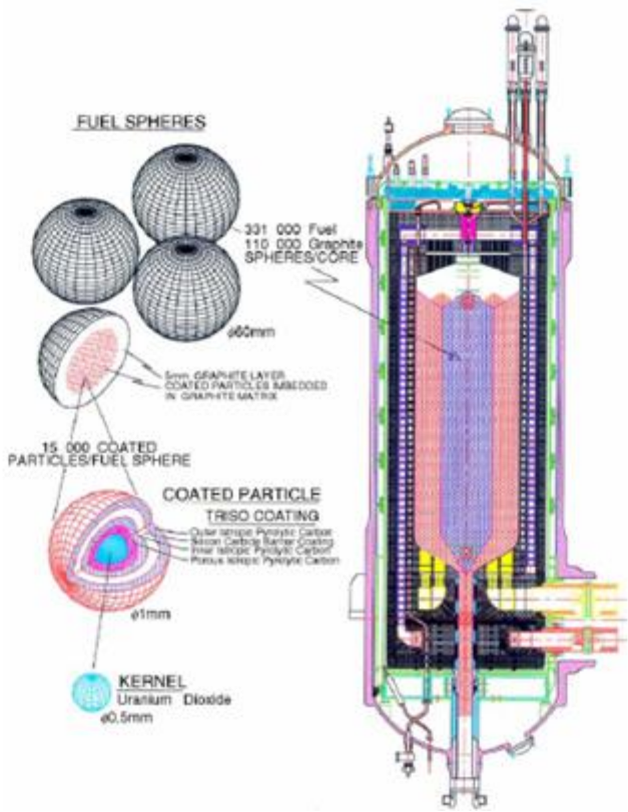
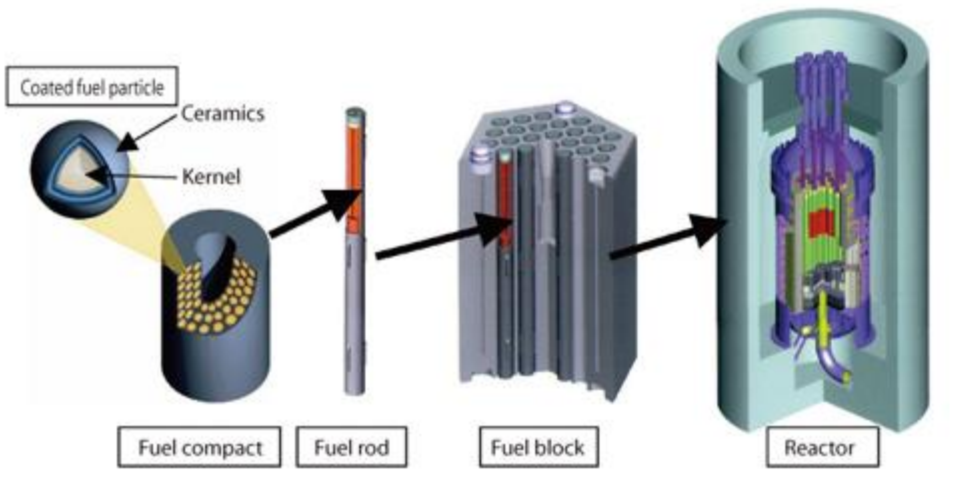
Pebble-Bed and Prismatic Reactor Design





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History of High Temperature Gas Cooled Reactors

Invented by Professor Dr. Rudolf Schulten in the 1950s and developed in Germany in the 1970s

- *In 1966, **Philadelphia Electric** has put into operation the **Peach Bottom I** nuclear reactor, **it was the first HTGR**.*

(115 MWth and a peak temperature of 1000 ° C, 40 MWe)

-First Commercial (U/Thorium Cycle)

-Generally Good Performance (75% CF)

- *A decade after, **General Atomics** put into operation the **Fort St. Vrain** reactor (in Colorado) which had a much lager power, 842 MWth.* (330 MWe) 1979-1989 (U/Th)

Poor Performance

-Mechanical Problems

-Decommissioned

History of High Temperature Gas Cooled Reactors

- ***Two research HTGRs in Europe***.
  - The British **Dragon**, 1966-1975 (test reactor of 20 MWth)
  - The German Arbeitsgeminshaft Versuchsreaktor **(AVR),** 1967-1989 (46 MWth).
- **THTR (**1986-1989) Hochtemperatur Kernkraftwerk launched the **Thorium High Temperature Reactor** in Germany. (750 *MWth.)*

History of High Temperature Gas Cooled Reactors

Technical data of the power (normal text) and research (italics text) high temperature gas reactors

	Peach Bottom I	Fort St. Vrain	THTR	*Dragon*	*AVR*
Operation	1966–1974	1966–1989	1986–1989	1966–1975	1967–1988
Power (MW_{th})	115	842	750	20	46
Coolant					
Pressure (MPa)	2.5	4.8	4	2	1.1
Inlet/outlet temperature (°C)	344/750	406/785	250/750	350/750	270/950
Fuel					
Type	$(U\text{-}Th)C_2$ PyC coated particles	$(U\text{-}Th)C_2$ TRISO	$(U\text{-}Th)O_2$BISO	$(U\text{-}Th)C_2$ PyC coated particles	UO_2 BISO
Peak temperature (°C)	1000	1260	1350	1000	1350
Form	Graphite compacts in hollow rods	Graphite compacts in hexagonal blocks	Graphite pebbles	Graphite hexagonal blocks	Graphite pebbles

History of High Temperature Gas Cooled Reactors

Features in common

- Use of coated fuel particles (CFPs), BISO or TRISO.
- Graphite as moderator.
- Shaped fuel of pebbles, rods or blocks.
- Helium as coolant.

AVR (German Arbeitsgeminshaft Versuchsreaktor)

- Experimental HTR, built as experiment at industrial scale in Jülich, Germany.
- Operated successfully during 21 years (1967-88) and was used as a test bed for various fuels and refueling strategies.
- Many important experiments to demonstrate the safety features of HTRs were carried out, such as a simulation of a depressurized loss of forced cooling (DLOFC) experiment.
- These experiments provided valuable data for validation of computer code systems.

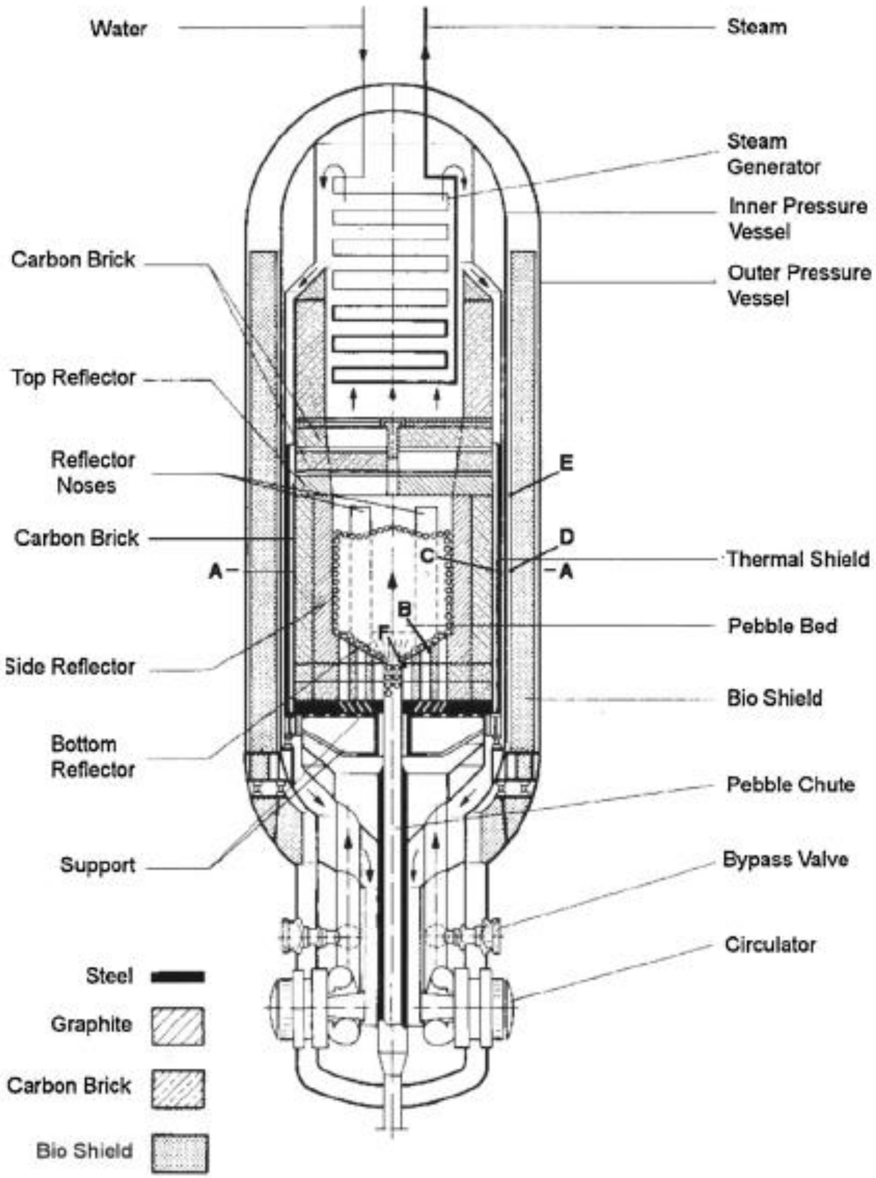

Fig. 1. Schematic overview of the AVR reactor core, adopted from Krüger (1989).

AVR

The steam generator, the reactor and the blowers are **all in the same reactor pressure vessel.**

The reactor operated in two full load modes: at **high outlet temperature (950 ◦C)** and at **low outlet temperature (810 ◦C).**

The helium flows through the reactor core from the bottom to the top, and is cooled down in the steam generator.

Technical data of AVR reactor

Power / Average power density	46 MW_{th} (15 MW_{el}) / 2.5 MW/m^3
Cycle	Steam cycle with the steam generator (73 bar) inside the reactor vessel
Core height / diameter	2.8 m / 3 m
Coolant / Pressure	He / 10.8 bar
He outlet / inlet temperature	1st phase until 02/1974: $\leq$ 850°C; 2nd phase from 02/1974: $\leq$ 950 (990)°C / 275°C
He-flow	13 – 15.5 kg/s (depending on desired gas outlet temperature) in up flow direction
Fuel	Core: 100000 matrix graphite pebbles (6 cm diam.) containing coated fuel particles. Diverse fuel types, at end mainly improved TRISO fuel

Table I: Main design data of AVR pebble bed reactor

AVR Fuel Elements behavior

- **FE were recycled continuously** from the bottom of the core.
- Daily, **600 FE were recycled** through with **60 elements being removed and 60 fresh fuel balls being added**.
- The FE **endured high temperatures**, **neutron irradiation**, **corrosion** due to coolant's contamination and the **mechanical forces** in the transport system.
- **The retention of gaseous fission products was excellent**.
- **Maximum fuel temperatures** in fresh fuel elements were **1150°C**, **release of solid fission products was low**.
- **Only early FE BISO particles released strontium** when the reactor operated at coolant temperatures **above 900°C**.

Study about TRISO-coated particle fuel performance

The American authors recognized that historically, **the irradiation performance of TRISO fuel in Germany (AVR) had been superior to that in the US.**
German fuel generally **had displayed gas release values** during irradiation **three orders of magnitude lower** than **US fuel**.
The Germans demonstrated **high quality production of TRISO-coated fuel** and **excellent irradiation and safety test behaviour** under reactor relevant conditions.
The studio compared the German and US fuel fabrication processes and the corresponding irradiation **databases** to identify the **technical reasons for the differences** in reactor behaviour.

The Review Conclusions

1. **There have been historical differences** in fabrication process of TRISO fuel used in Germany and the US.
2. **Different philosophies were used** to implement the irradiation and testing programs **(industrial/production scale vs. mixture of lab scale and larger scale fabrication).**
3. **The temperature gradient is a strong function of the power density** in the fuel element.
4. US fuel pebbles **have a higher packing fraction** of particles (up to 50%) than German pebbles (~10%)
5. German researches **recommended that the acceleration of any CFP irradiation should be no greater than three times the real time acceleration** (The US irradiations were accelerated 3–10 time the real time compared to 2-3 times the real time for most of the German irradiations).

A safety re-evaluation of the AVR pebble bed reactor operation and its consequences for future HTR concepts.

Rainer Moormann. (Know as Jülich report, June 2008)

- *The report deals mainly with* ***some insufficiently published unresolved safety problems of AVR operation.***
- ***AVR primary circuit was heavily contaminated with metallic fission products*** *(Sr-90, Cs-137) which created problems in* ***current dismantling.***
- ***The amount of this contamination is not exactly known****, but the evaluation of fission product deposition experiments indicates* ***that the end of life contamination reached some orders of magnitude more than pre-calculated*** *and far more than in large LWRs.*
- *A major fraction of this contamination is bound on* ***graphitic dust*** *and thus partly mobile in depressurization accidents.*
- *The AVR contamination* ***was mainly caused by inadmissible high core temperatures****, increasing fission product release rates, and not – as presumed in the past - by inadequate fuel quality only.*

Interpretations of unintentional high temperatures in AVR core.

The pronounced fission product release suggests a large fraction of FE with inadmissible high temperatures in the AVR core due to:

- coolant bypass flows inside and/or outside of the active core.
- power peaks near the reflector.
- local variations of density of pebble bed.
- human errors in fuelling procedure.
- uncertainties in pebble flow behaviour.
- uncertain burn-up measurements particularly until 1981.
- power asymmetry in the core.
- flow anomalies due to breakage in the bottom reflector

Rainer Moormann. Jülich report, Juny 2008

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Re-evaluation of fission product release from AVR core into the coolant circuit

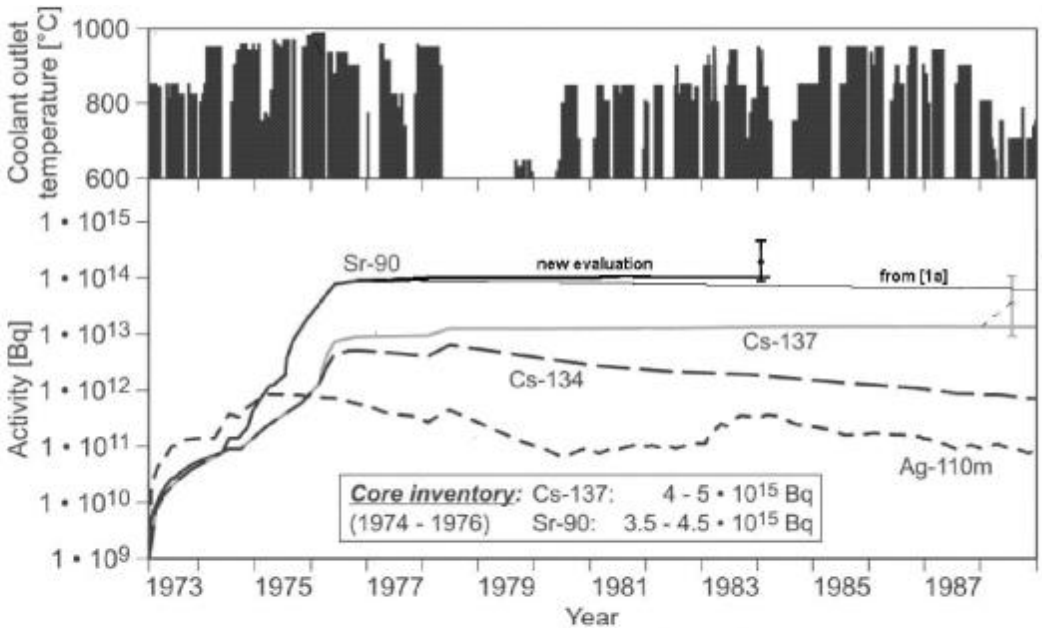

Fig. 2: Time dependent average hot gas temperatures (above) and accumulated activity release into the AVR primary circuit estimated from VAMPYR-I hot gas filter results for Cs-137 and Sr-90 with uncertainty scatter (below). Core inventories are from [49].

Measurements of Cs, Sr and Ag release from the AVR core by a deposition tube in the hot gas region **revealed that there was a pronounced fission products release into the primary circuit between 1974 and 1976**, what coincides **with the hot gas temperature increase** (from 850 to 950 °C) in February of 1974.

The Principal Facilities. HTTR, HTR-10, PROTEUS...

Several past and present experimental and prototypic facilities based on HTGR concepts that could be used as the V&V basis of codes employed in the design and analysis of the VHTR cores are:

- **Pebble-bed type cores**: **ASTRA**, **AVR,** CESAR II, GROG, **HTR-10, HTR-PROTEUS**, KAHTER, SAR and THTR
- **Prismatic block-type cores:** CNPS, DRAGON, FSV, GGA HTGR Criticals, HITREX-1, HTLTR, **HTTR**, MARIUS-IV, Peach Bottom HTGR, Peach Bottom Criticals, SHE, NESTOR/HECTOR and VHTRC.

Facilities for pebble-bed type cores:

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Facility (Country)	Geometry	Size	Fuel type	Asymptotic state or zero-power startup	Availability of data	Priority
ASTRA (Russia)	Annular, but not azimuthally symmetric	Small	As desired	Zero-power startup	Existing facility – data can be obtained	High
AVR (Germany)	Cylindrical	Short; radial extent appropriate	Various; some low-enrichment TRISO	Both	Limited	High
CESAR II (France)	Hexagonal	Small	Low-enriched UO2	Zero-power startup	Neutronics data exist	Medium
GROG (Russia)	Cylindrical or annular	Short; radial extent appropriate	As desired, but very low packing fraction	Zero-power startup	Existing facility – data can be obtained	Medium
HTR-10 (China)	Cylindrical	Small	Low-enriched TRISO	Both	Existing facility- data can be obtained	Highest
HTR-PROTEUS (Switz.)	Cylindrical	Small	LEU pebble-bed fuel	Zero-power	PSI and IAEA would need to be contacted	High
KAHTER (Germany)	Cylindrical	Small	Uncertain	Zero-power startup	Uncertain	High
SAR (Austria)	Cylindrical	Small	Probably low-enrichment TRISO	Zero-power startup	Limited data were obtained for this special-purpose test	Low
THTR (Germany)	Cylindrical	Large	Thorium-uranium	Most data for zero power; reactor presumably achieved steady state	More data available for zero-power startup than operating conditions	Medium

Facilities for prismatic block-type cores:

Facility (Country)	Geometry	Size	Fuel type	Asymptotic state or zero-power startup	Availability of data	Priority
CNPS (USA)	Cylindrical	Small	LEU	Zero	LANL data	High
DRAGON (England)	Hexagonal	Small	HEU/Th	Both	Data must be retrieved from U.K./OECD	Low
Fort St. Vrain (USA)	Cylindrical	Large	HEU/Th	Both	Data is GA proprietary	Medium/ High
GGA HTGR criticals (USA)	Cylindrical	Small	HEU	Zero	Data is GA proprietary	Medium/ High
HITREX-1 (USA)	Hexagonal	Small	LEU fuel	Zero	U.K. nuclear data	Medium/ High
HTLTR (USA)	Block	Small	Pu-Th fuel	Zero	PNNL data	Low
HTTR (Japan)	Cylindrical/ Annular	Small	LEU fuel	Both	Existing facility- data can be obtained	High
MARIUS-IV (France)	Unknown	Small	HEU-Th	Zero	Unknown	Low
Peach Bottom HTGR (USA)	Cylindrical	Small	HEU/Th	Both	Data is GA proprietary	Low
Peach Bottom Criticals (USA)	Cylindrical	Small	LEU/Th	Zero	Data is GA proprietary	Low
SHE (Japan)	Hexagonal	Small	LEU fuel	Zero	JAEA data	Medium/ High
NESTOR/ HECTOR (England)	Square and cylindrical	Small	LEU fuel	Zero and elevated temperatures	U. K. nuclear data	Medium/ High
VHTRC (Japan)	Hexagonal	Small	LEU fuel	Zero	JAEA data	High

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The HTTR objectives:

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- *Establish and **upgrade the technology base** of the HTGR.*
- *Perform innovative basic **research in the field of high temperature engineering.***
- ***Demonstrate high temperature heat applications** and utilization of nuclear heat.*

Table 2-1. Specification of the HTTR

Thermal power	30 MW
Outlet coolant temperature	950°C
Inlet coolant temperature	395°C
Primary coolant pressure	4 MPa
Core structure	Graphite
Equivalent core diameter	2.3 m
Effective core height	2.9 m
Average power density	2.5 W/cm^3
Fuel	UO_2
Uranium enrichment	3 to 10 wt%
Type of fuel	Pin-in-block
Burn-up period (efpd)	660 days
Coolant material	Helium gas
Flow direction in core	Downward
Reflector thickness	
Top	1.16 m
Side	0.99 m
Bottom	1.16 m
Number of fuel assemblies	150
Number of fuel columns	30
Number of pairs of control rods	
In core	7
In reflector	9

Construction was completed on May 1996.
Fuel loading on July 1998.
The first criticality was attained in annular type core of 19 columns on Nov of 1998.
The first full power operation with an average core outlet temperature of 850°C was completed in 7 December 2001.
Operational licensing of the HTTR was approved on 6 March 2002.

HTTR-30 features

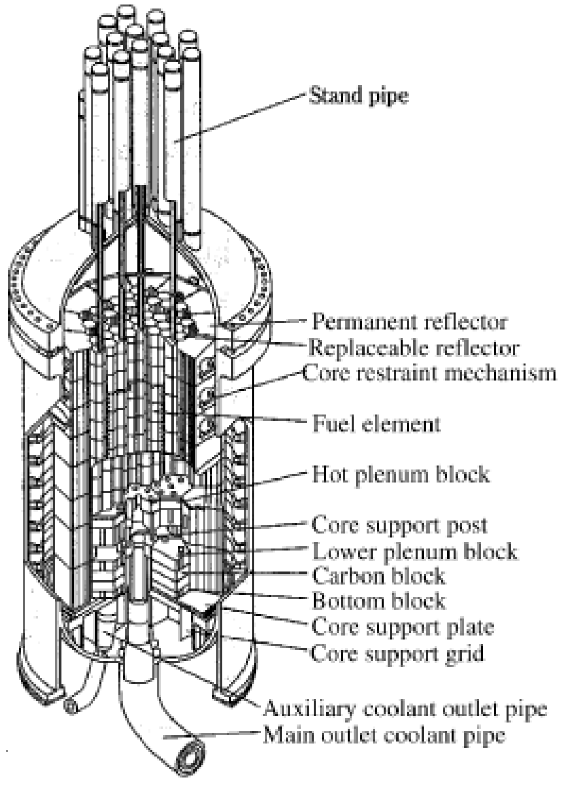
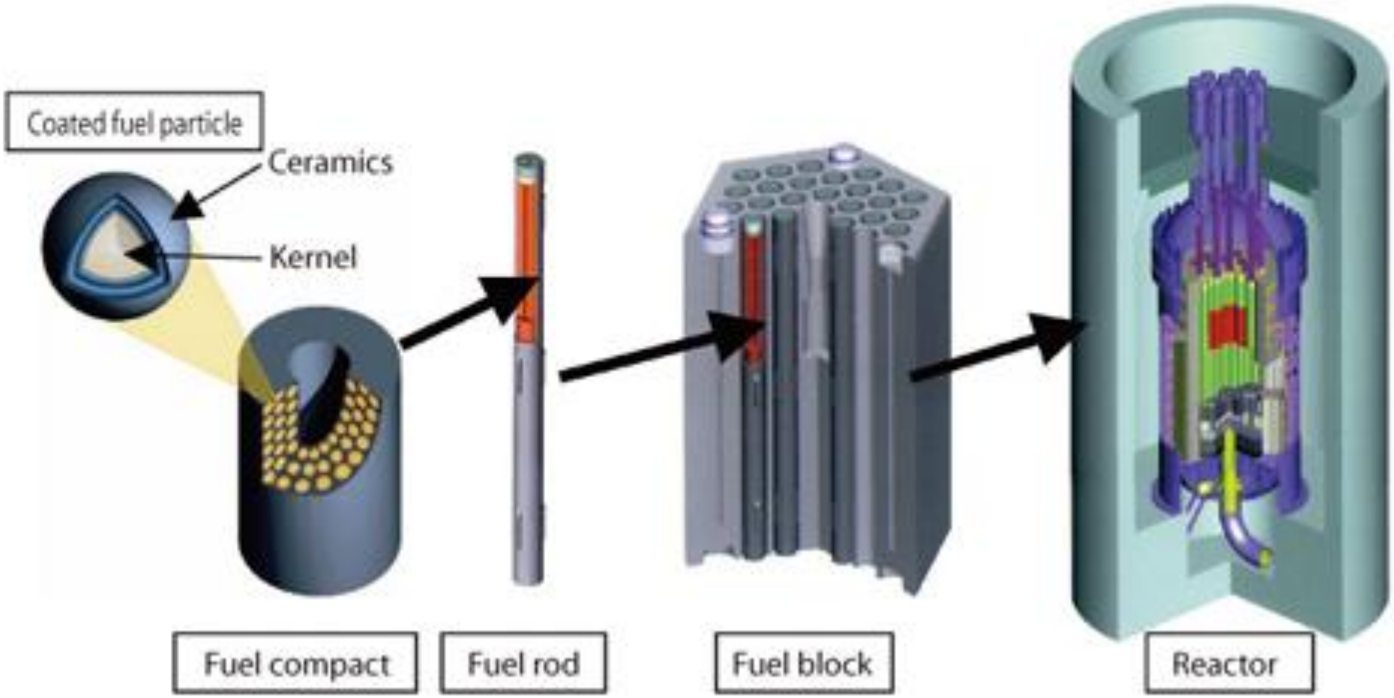


Fig. 3. Reactor core structure of the HTTR.

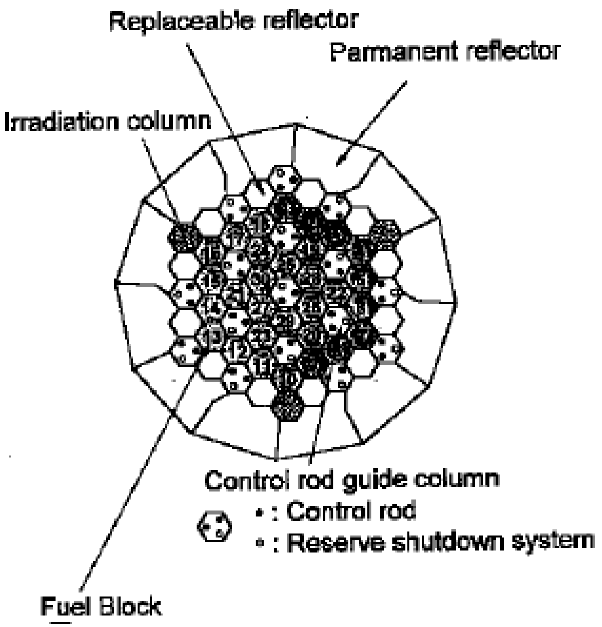
Annular Core. (High inherent safety characteristics for loss of coolant accidents).
The reactor vessel is 13.2 m tall and has 5.5 m inner diameter.
The reactor core consists mainly of:
•hexagonal fuel blocks,
•control rod guide blocks,
•replaceable reflector blocks.
(30 fuel columns and 7 control rod guide columns). One column is made up of five fuel blocks and four replaceable reflector blocks.
The active core is surrounded by replaceable reflector blocks and permanent reflector blocks.

A fuel assembly consists of fuel rods and a hexagonal graphite block, 360 mm across flats and 580 mm in height.



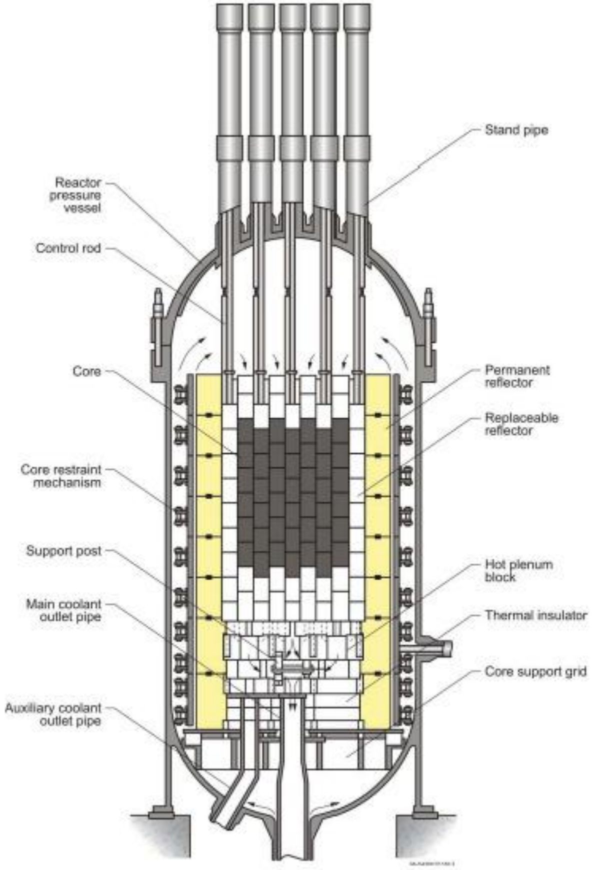
HTTR reactivity control system

- *The control rods are **individually supported by control rod drive mechanisms.***
- *The control rods are inserted **into the channels in the active core and into replaceable reflector regions** around the active core.*
- ***Reactor shutdown is made at first by inserting nine pairs of control rods into the reflector region, and then by inserting the other seven pairs of the control rods into the active core region 40 min later or after the outlet coolant temperature decreases to 750 °C,** so that the control rods should not exceed their design temperature limit.*



HTTR operational mode.

- The reactor **outlet coolant temperature at the full power is set at both 850° and 950°C.**
- The reactor operational mode at 850°C is defined as "rated operation" and **at 950°C is "high temperature test operation"** because operation of the HTTR is not allowed at 950°C for full life of the initial core.
- **A steel reactor containment vessel** of 18.5 m in diameter and 30 m in height **is installed in the center of the reactor building.**



Reactor cooling system

The main cooling system is composed of a primary cooling system, a secondary helium cooling system and a pressurized water cooling system.

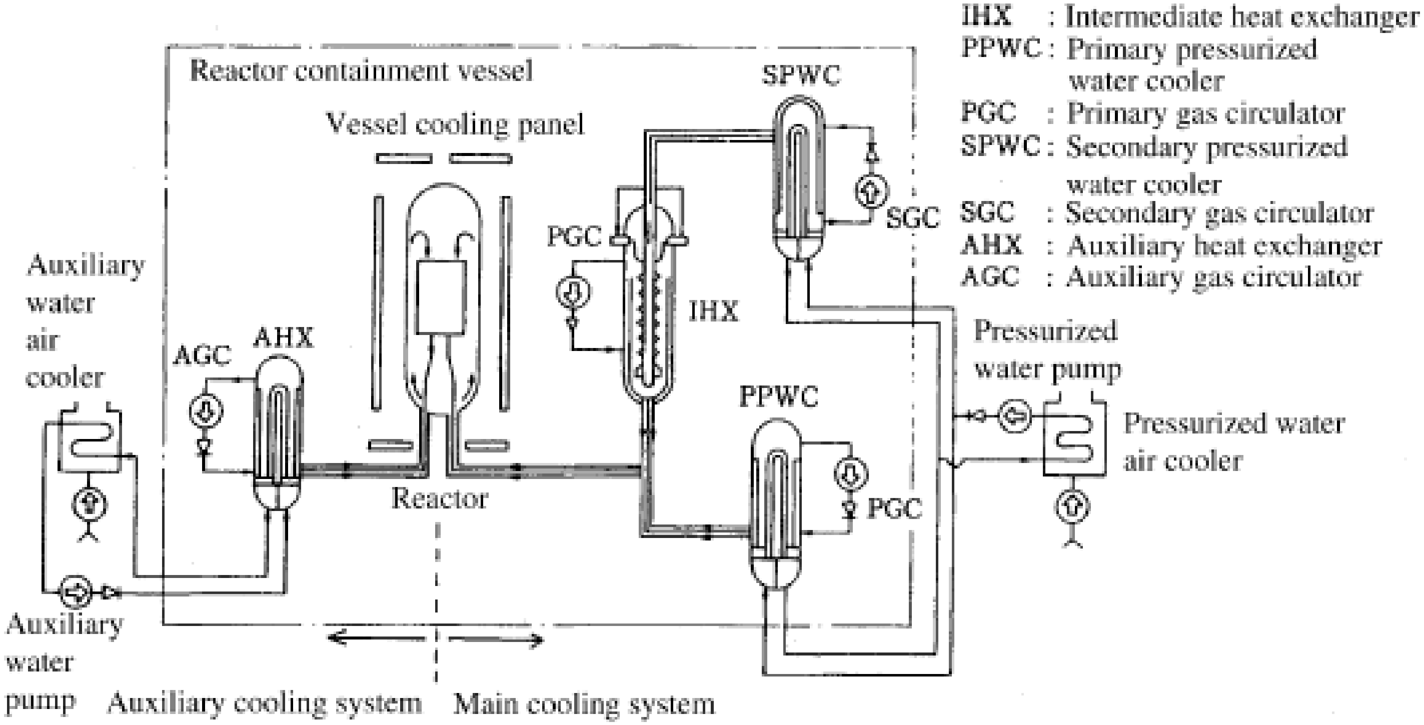


Fig. 4. Cooling system of the HTTR.

The HTTR hydrogen production system is the first demonstration of hydrogen production using heat supplied directly from the HTGR in the world

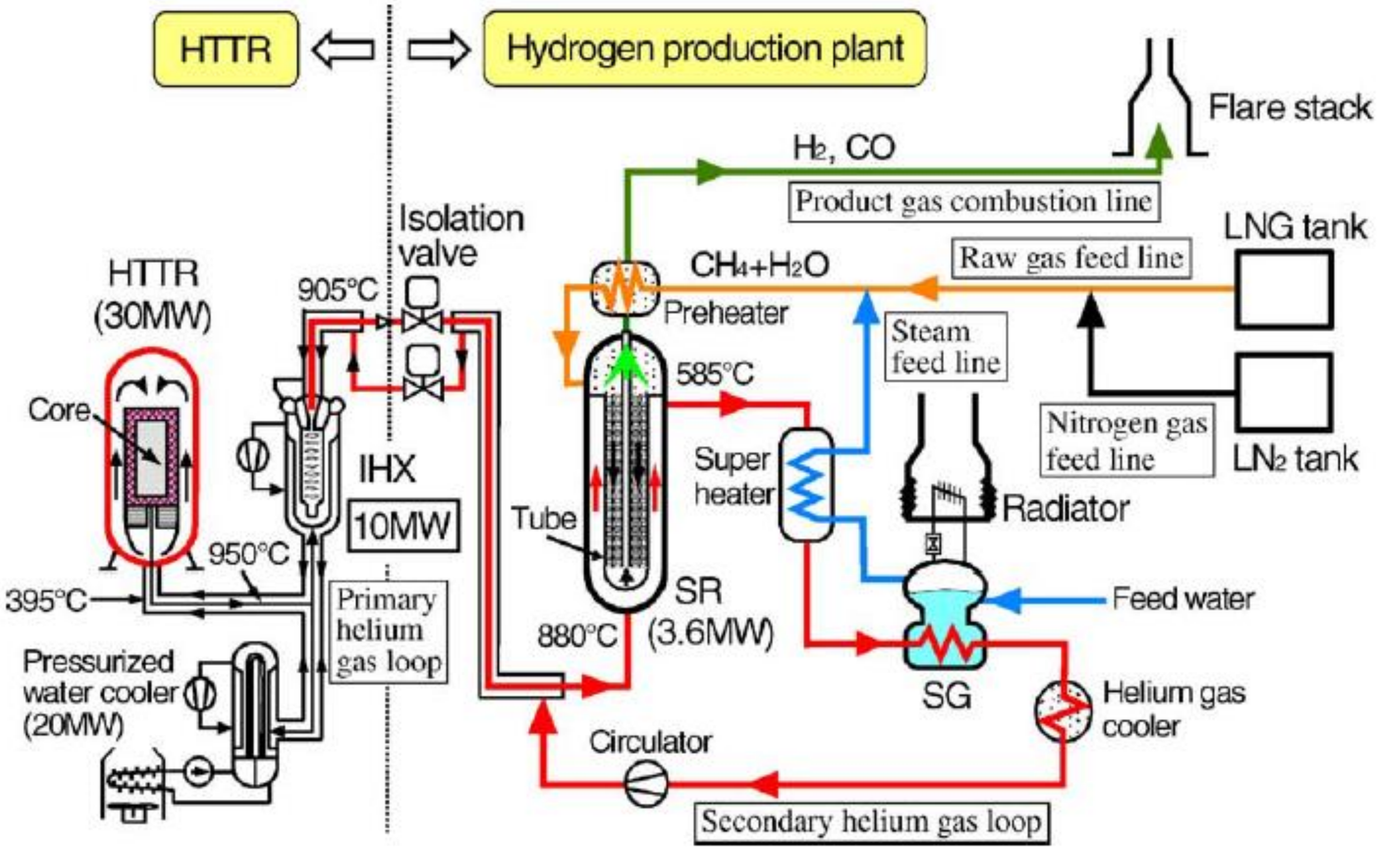
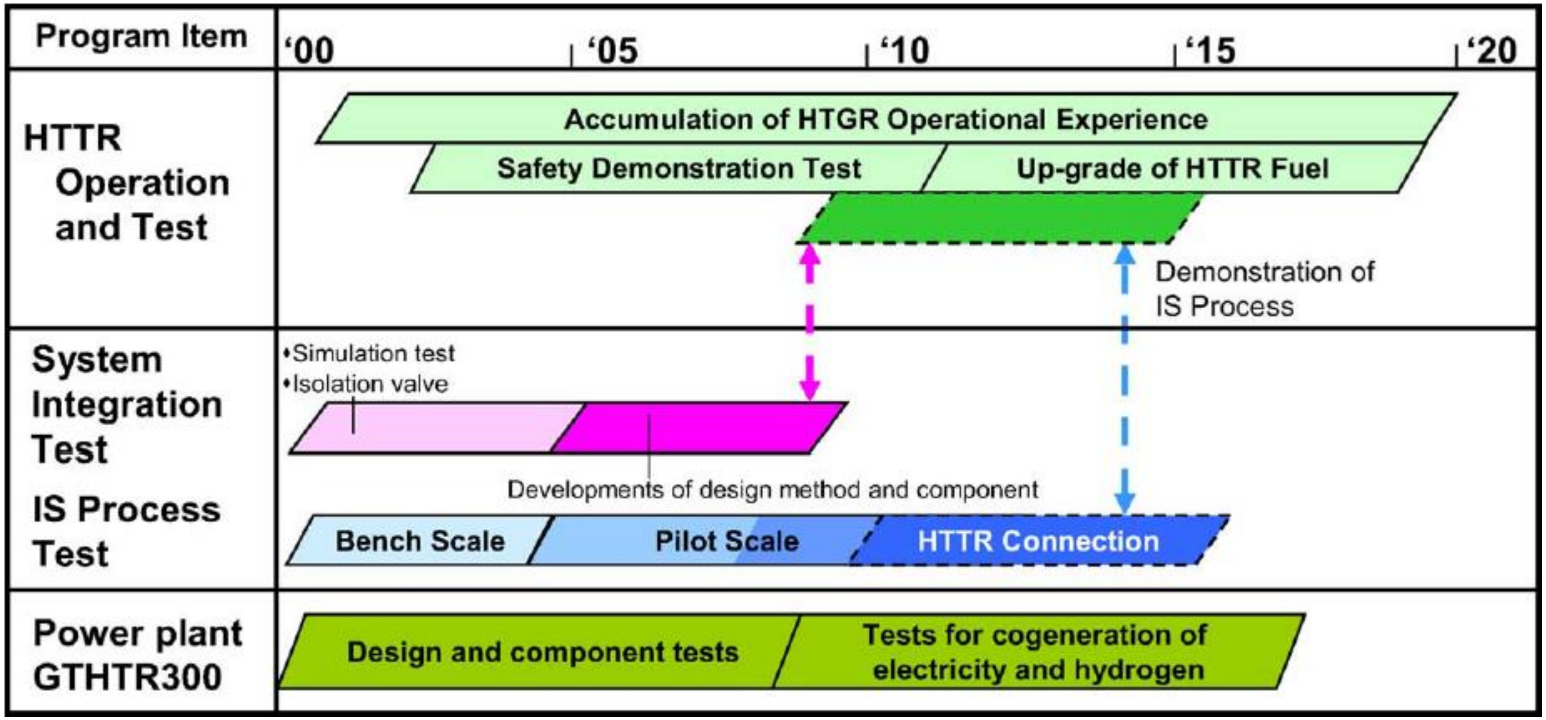


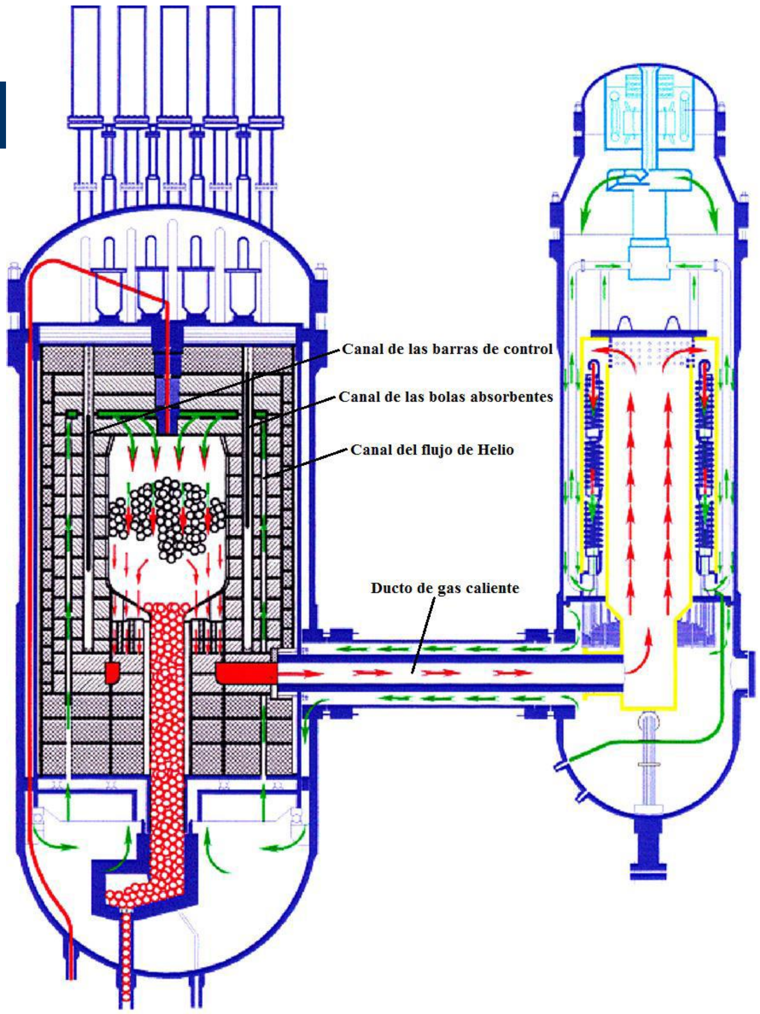
Fig. 1. Flow scheme of HTTR hydrogen production system.

Operation of the test plant will demonstrate the technical feasibility of IS process.
The test data will be used to verify the analytical codes to be developed.
After completion of the pilot test of IS cycle, it is planned to proceed to the demonstration test using HTTR.



In 1992, China decided to construct the 10 MWth High-Temperature Gas-Cooled Pebbled Bed Test Reactor (HTR-10)

China started R&D of HTGRs in 1970s, at "**Institute of Nuclear and New Energy Technology**" (INET) of the Tsinghua University.
In December 2000, **the HTR-10 reached its first criticality.**
In January 2003 the HTR-10 was **successfully connected to the electric grid.**



Objectives of the Pebbled Bed HTR-10

Are aimed to verify and demonstrate the technical and safety features of the modular PBR and to establish an experimental base for developing nuclear process heat applications.

The specific aims of the HTR-10 have been defined as follows:

- ***To acquire the experience*** *of HTGR design, construction and operation.*
- *To carry out the* ***irradiation tests*** *of fuel elements.*
- ***To verify*** *the* ***inherent safety features*** *of the modular HTGR.*
- ***To demonstrate*** *the* ***electricity/heat co-generation*** *and* ***steam/gas turbine combine cycle.***
- ***To develop*** *the* ***high temperature process heat*** *applications.*

Technical features incorporated in the design of the HTR-10

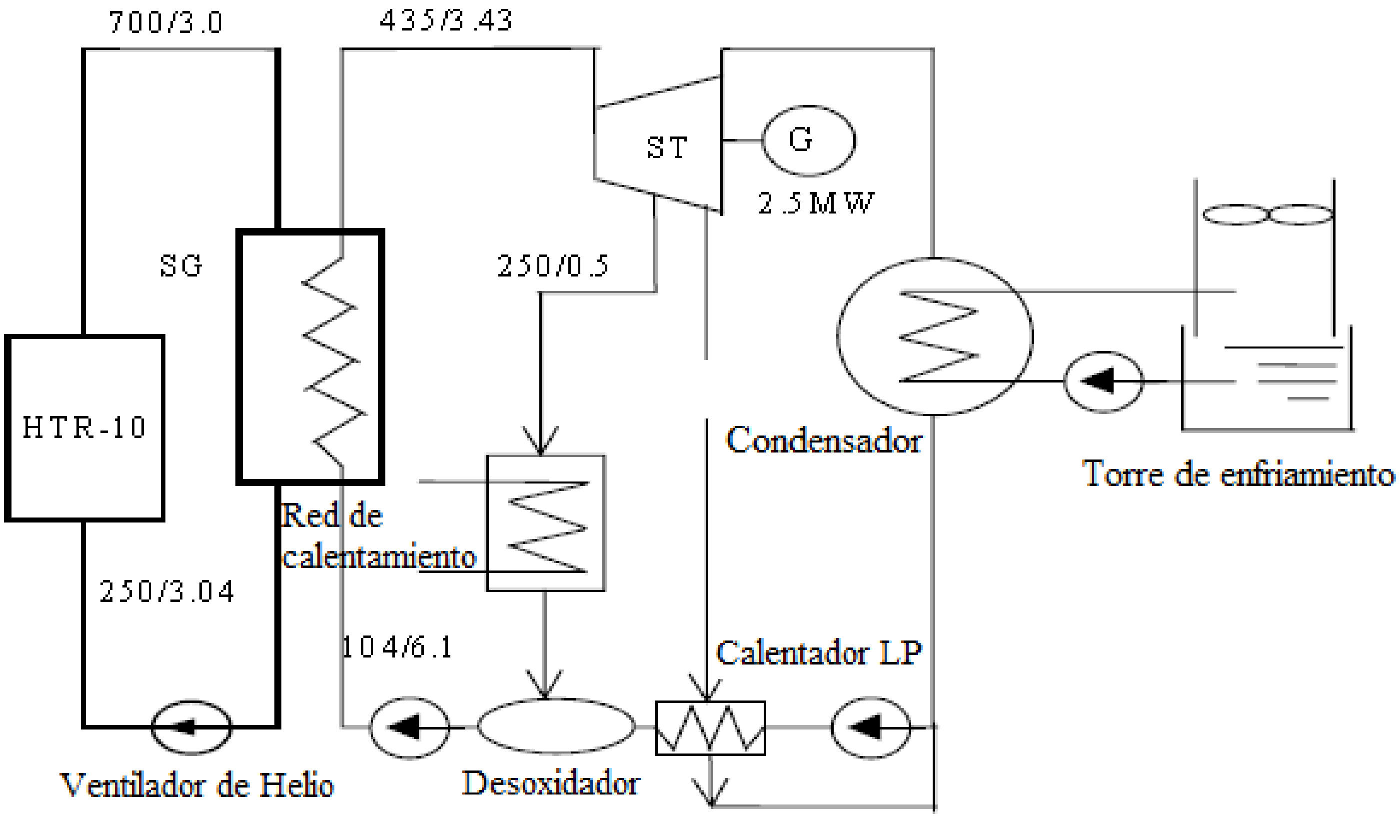
1. Use of spherical fuel elements, **pebbled bed reactor**, **CPF**.
2. The **maximum fuel element temperature limit** cannot be exceeded in any accident.
3. The reactor and the steam generator are housed in **two separate steel pressure vessels**.
4. **An active core cooling system is not required for residual heat removal** in case of accident.
5. **The reactor core is entirely constructed by graphite materials**, no metallic component are used in the region of the core.
6. The **two reactor shut down systems**, i.e. ten control rods and seven small absorber ball systems, **are all positioned in the side reflector**.
7. Spherical fuel elements go through the reactor core in **a “multi-pass" pattern**.

Key Design Parameters of the HTR-10

Table 4-1. Key Design Parameters of the HTR-10 [4-1]

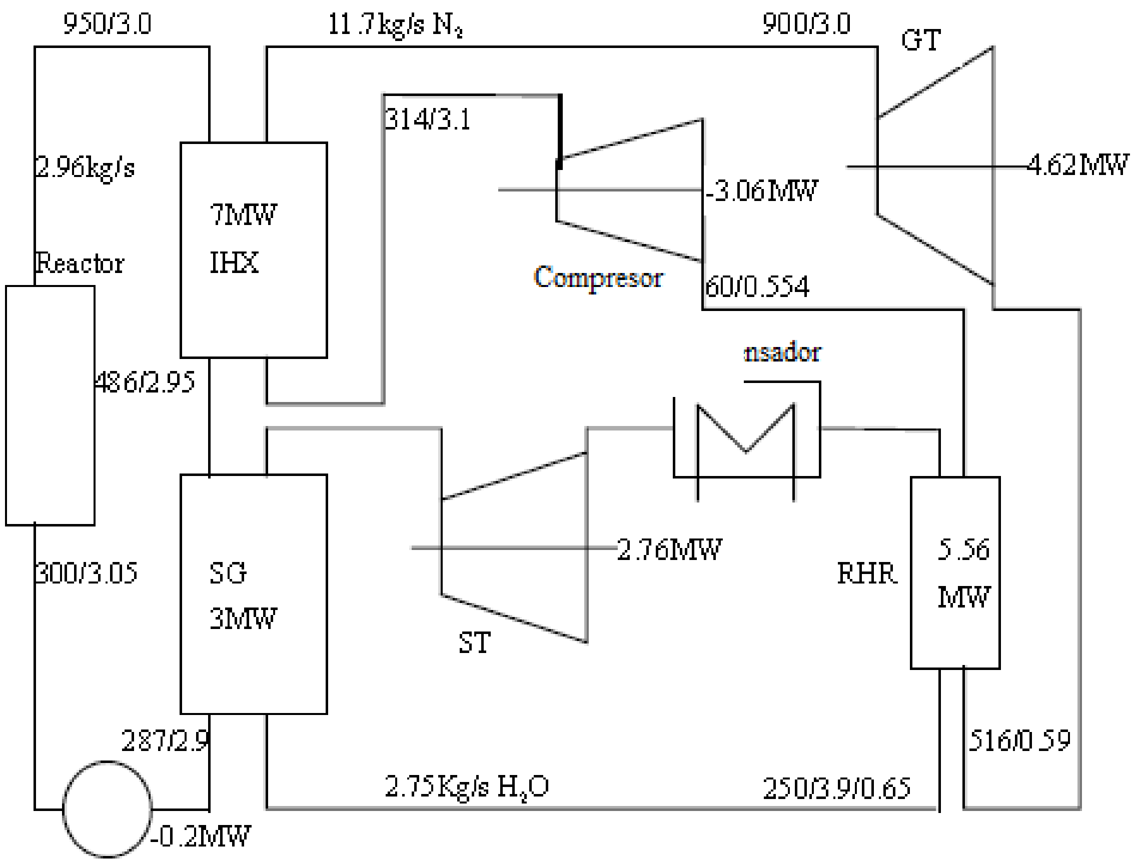
Reactor thermal power	MW	10
Primary helium pressure	MPa	3.0
Reactor core diameter	cm.	180
Average core height	cm.	197
Average helium temperature at reactor outlet	°C	700
Average helium temperature at reactor inlet	°C	250
Helium mass flow rate at full power	kg/s	4.3
Main steam pressure at steam generator outlet	MPa	4.0
Main steam temperature at steam generator outlet	°C	440
Feed water temperature	°C	104
Main steam flow rate	t/hr	12.5
Number of control rods in side reflector		10
Number of absorber ball units in side reflector		7
Nuclear fuel		UO_2
Heavy metal loading per fuel element	g	5
Enrichment of fresh fuel element	%	17
Number of fuel elements in equilibrium core		27,000
Fuel loading mode		multi-pass

First phase, core **outlet temperature of 700°C** and inlet of 250°C.The secondary circuit included a **steam turbine cycle** for electricity generation with the capability for district heating.



Flow diagram of the first phase of HTR-10 operation

Second phase, core **outlet temperature of 900°C**. **A Gas turbine and steam turbine combined cycle** for electricity generation. The intermediate heat exchanger provides **high temperature nitrogen gas of 850°C** for the **GT cycle**. The SG **produces steam** at 435 **°**C for the ST cycle with the remaining thermal power.

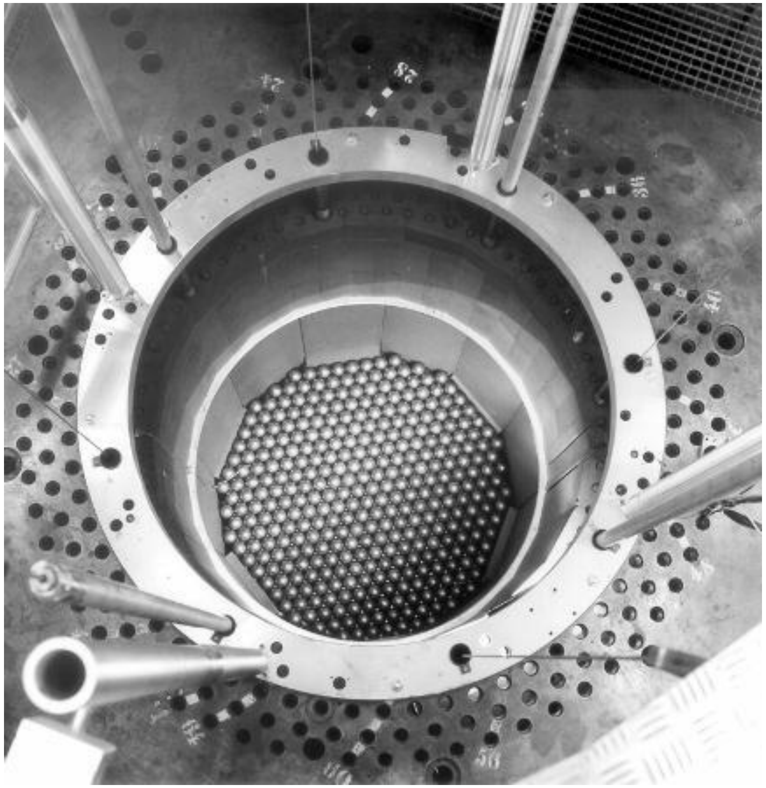


Flow scheme of steam turbine/gas turbine combined cycle

R&D on the HTR-10 lead to five significant achievements:

- **The know-how of fabricating FE for the HTR-10 was mastered**. (Spherical coated particle fuel element)
- Technology of **spherical FE handling and transportation by pulse pneumatic mechanism** was mastered.
- **Helium process technologies**: such as helium **sealing and purification**, the **lubrication equipments** in a helium atmosphere, electrical insulation were developed.
- **Domestic manufacture of key equipments** for HTGRs: the reactor pressure vessel, the steam generator pressure vessel, the hot gas duct, the steam generator with helical tubes, the helium blower, the fuel handling equipments and reflector graphite components.
- Successful development of **fully digital reactor protection systems.**

PROTEUS
Critical experiment facility at the Paul Scherrer Institute, Villigen, Switzerland



PROTEUS has been configured as a multi-zone system for the purpose of reactor physics investigations of both **gas-cooled fast breeder** and **high conversion reactors.**
For the LEU-HTR experiments **PROTEUS** was for the first time configured as a single zone, **pebble bed system**, surrounded radially and axially by a thick graphite reflector.
IAEA established a CRP on the validation of safety related physics calculations for **Low-Enriched High Temperature Gas Cooled Reactors** (HTGRs) in 1990.

Main components of HTR Proteus facility

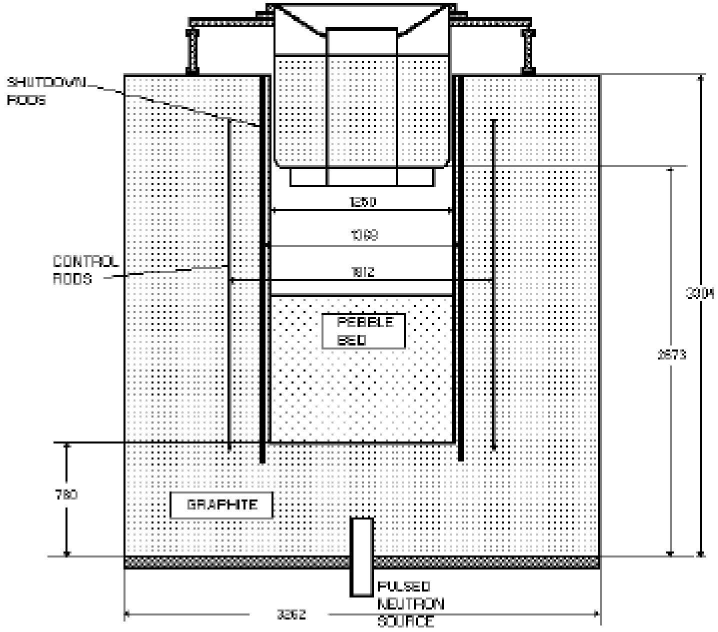


Figure 1. A schematic side view of the HTR-PROTEUS facility

1. **Fuel and moderator pebbles.**
2. **Graphite — radial, upper and lower. axial reflectors and filler pieces.**
3. **Aluminum structures.**
4. **Shutdown rods.**
5. **Fine control rods.**
6. **Automatic control rod.**
7. **Static "measurement rods“.**
8. **Polyethylene rods used to simulate water ingress.**

The following nine institutes from seven countries participated in the benchmark of CRP

- The Institute of Nuclear Energy Technology (INET) in China
- The KFA Research Center Jülich (KFA) in Germany
- The Japan Atomic Energy Research Institute HTTR Group (JAERI-HTTR) in Japan
- The Japan Atomic Energy Research Institute VHTRC Group (JAERI-VHTRC) in Japan
- The Netherlands Energy Research Foundation (ECN) in the Netherlands
- The Interfaculty Reactor Institute, Delft University of Technology (IRI) in the Netherlands
- The Kurchatov Institute (KI) in the Russian Federation
- The Paul Scherrer Institute (PSI) in Switzerland
- The Oak Ridge National Laboratory (ORNL) in the USA

The benchmarks consisted of six graphite reflected 16.76% enriched-uranium pebble-bed systems of three different lattice geometries and two different moderator-to-fuel pebble ratios. (2:1 and 1:2)

Table 3.1.1: Benchmark Model Summary

Problem Name	M/F Pebble Ratio	C/U Ratio	C/^{235}U Ratio	Void Fraction	Core Radius (cm)	Core Height (cm)	Number of Fuel Pebbles
LEUPRO-1	1/2	954	5630	0.2595	58	99	4567
LEUPRO-2	2/1	1894	11181	0.2595	58	138	3183
LEUPRO-3	1/2	954	5630	0.3954	59.85	132	5294
LEUPRO-4	2/1	1894	11181	0.3954	59.85	173	3469
LEUPRO-5	1/2	954	5630	0.38	62.50	120	5382
LEUPRO-6	2/1	1894	11181	0.38	62.50	173	3879

Calculated results were obtained for both unit cells and for the whole reactor

For the unit cells the following parameters were calculated:

1. **Kinf** (0) for $B^2=0$, i.e. production/absorption for $B^2=0$.
2. the **critical buckling** B^2 and Kinf (B^2cr).
3. the **migration area** $M^{2.}$
4. the **spectral indices.**

For the whole reactor the following results were requested:

1. **Keff** for the specified dimensions and specified atomic densities.
2. the **critical pebble-bed core height Hcr.**
3. the **spectral indices** at core center and core averaged.

Evaluation of high temperature gas cooled reactor performance

This CRP complements other projects in **validating safety and performance capabilities** of the HTGR. **Computer codes and models** are verified through **actual test results** from operating reactor facilities.

- **Participant in this CRP:**

1. Institute of Energy Technology (INET), Beijing, China
2. • SACLAY (CEA), Gif-sur-Yvette, France
3. • Research Centre Juelich (ISR), Juelich, Germany
4. • National Nuclear Energy Agency (BATAN), Serpong, Indonesia
5. • Japan Atomic Energy Research Institute (JAERI), Oarai, Japan
6. • Nuclear Research and Consultancy (NRG), Petten, Netherlands
7. • OKBM/Kurchatov Institute, Nizhny Novgorod, Russian Federation
8. • Pebble Bed Modular Reactor (PBMR), Centurion, South Africa
9. • Department of Nuclear Engineering, Hacettepe University, Ankara, Turkey
10. • Oak Ridge National Laboratory (ORNL), Oak Ridge, TN, United States of America

HTTR start-up core physics tests and thermal hydraulics

- **Initial Criticality**.
- **Control Rod Position at Criticality.** The control rod insertion depths are evaluated at the critical condition for the following three cases.

1) **18 columns.** (thin annular core)
2) **24 columns.** (thick annular core)
3) **30 columns.** (fully-loaded core)

- **Excess Reactivity** The excess reactivity is evaluated for the three cases mentioned above.
- **Isothermal Temperature Coefficient.**
- **Scram Reactivity.** The Scram reactivity is to be evaluated for the following two cases:

1) **All reflector CRs** are inserted at the critical condition.
2) **All CRs in reflector and core** are inserted at the critical condition.

HTTR Thermal Hydraulic Benchmark Problem

Two sets of thermal hydraulic benchmark problems associated with the HTTR were investigated.

1. *The prediction of the amount of **heat removed by the Vessel Cooling System (VCS)** at **30 MW power** and the associated temperature profile on the surface of the side panel.*

2. *The analytical simulation on **transient behaviour of the reactor and plant during the loss of off-site electric power for HTTR** operation at 15 and 30 MW.*

Neutronic benchmark for the HTR–10

The benchmark is divided in four different sub-problems:

- Evaluation of the **core critical height** for the initial load. ***Involve calculating of loading height, starting from the upper surface for the first critically.***
- Calculations of the **fuel and moderator temperature coefficients**.
- Calculations of the **control rods worth** for the full core.
- Calculations of the **control rods worth** for the initial core

VSOP is a code system used as the main tool for design studies of HTR. (*Research Centre Juelich (ISR), Juelich, Germany*)

The benchmark problems were calculated using the following parts of **VSOP** code system.

1. **ZUT** (*Self-shielded cross sections*)
2. **GAM** (*fast and epithermal spectral code*)
3. **THERMOS** (*thermal cell code*)
4. **CITACION (***The eigenvalues and flux distributions of the whole reactor*)

VSOP considers the following main features of **PBR**:

1. The **double heterogeneous nature of spherical FE** with the coated particles.
2. The **streaming correction** of the diffusion constant in the pebble bed.
3. Use of **anisotropic diffusion constants** (channels of the control rod and of the small absorber balls).

Comparison between the calculated and experimental results showed that **VSOP and MCNP** calculations of critical loading height had **very good agreement.**

Table 4-10. VSOP calculation for HTR-10 initial criticality under air

Loading height (cm)	Number of fuel balls	Number of dummy balls	K_{eff} (27°C)
126	9858	7436	1.010562
120	9388	7082	0.992149

By linear interpolation of the calculated values, the critical loading height predicted by VSOP is 122.558 cm, which corresponds to a total loading of 16821 balls.

Table 4-11. MCNP calculation for HTR-10 initial criticality under air

Loading height (cm)	Number of fuel balls	Number of dummy balls	K_{eff} (27°C)	Standard deviation	99% confidence intervals
126	9858	7436	1.01002	0.00087	1.00772-1.01232
120	9388	7082	0.99079	0.00080	0.98869-0.99290

By linear interpolation of the calculated values, the critical loading height predicted by MCNP is 122.874 cm, which corresponds to the loading of 16864 balls in total and agrees well with the VSOP calculation.

- Density of dummy balls: 1.73 → 1.84 g/cm^3
- Boron equivalent of impurities in dummy ball: 1.3 → 0.125 ppm
- Core atmosphere at initial criticality: Helium → Air

XVIII ENFIR

After the **Original Benchmark** was defined, and before of the HTR-10 first load, three conditions of the benchmark definition were changed, because there were **Original and Deviated Benchmark.**

Participant Countries	Original benchmark		Deviated benchmark	
	Diffusion/Transport	Monte Carlo	Diffusion/Transport	Monte Carlo
China	125.8	126.1	122.558	122.874
France[1]	-	-	-	115.36 117.37
Germany[2]	124.2 126.8	-	121.0 123.3	-
Indonesia[3]	107 120	-	-	-
Japan	113	-	-	-
Netherlands	125.3	-	122.1	-
Russia	136	137.3	-	-
South Africa	-	-	122.537	-
Turkey[4]	119.27	129.7 135.3	-	-
United States[5]	-	127.5 128	-	-
Our study	-	124.15	-	121.21
Experimental result for the critical height: 123.06				

Critical height (cm) results for both, Original and Deviated Benchmark

Main Designs

The South African Pebble Bed Modular Reactor (PBMR) project

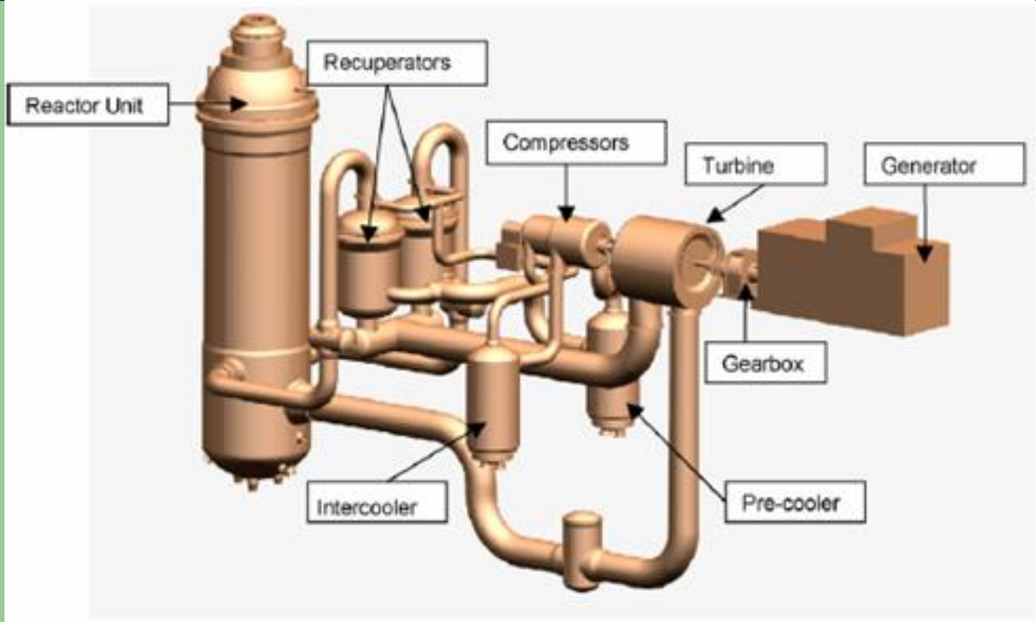


Fig. 2. Main power system of PBMR.

Planning research and engineering projects:
South African PBMR project,
USA-Russia GT-MHR project,
USA Next Generation Nuclear Power Plans,
HTTR in Japan,
Korean HTGR plan,
and Chinese High Temperature Gas-Cooled Reactor-Pebble bed Module (HTR-PM) demonstration project.

Design features of PBMR-400

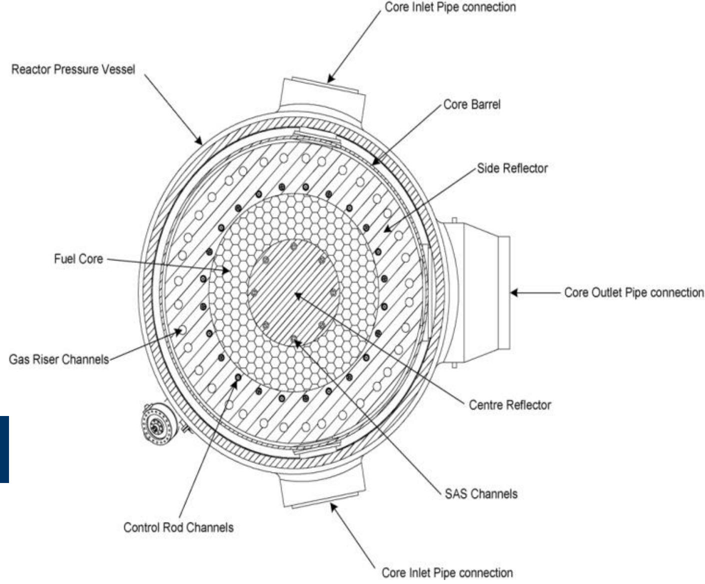


Table 1
Reactor unit parameters

Parameter	PBMR
Reactor thermal power	400 MW
Reactor inlet temperature	500 °C
Reactor outlet temperature	900 °C
Mass flow rate	192 kg/s
System operating pressure	9 MPa
Pressure vessel	Steel
Reactivity control system	Twenty-four control rods in the side reflector
Reserve shutdown system	Eight channels in the centre reflector filled with absorber spheres
Coolant flow direction	Downwards
Pebble bed inner diameter	2.0 m
Pebble bed outer diameter	3.7 m
Pebble bed height	11.0 m
Volume of pebble bed	~84 m^3
Number of fuel spheres	~452 000
Fuel burnup target	92 000 MWd/tU

- **Heat input to the Brayton cycle.**
- **The PBMR is conceived as modular.** (Will be optimal if built in a group of 8-10 units, sharing some facilities such as the control room)
- **Safe shutdown** and the ability to remove the decay heat.
- **Safe requirements have a natural and passive way** by relying only on gravity, conduction, convection and thermal radiation.

The Pebble Bed Modular Reactor (PBMR) was abandoned in 2010 after 12 years of effort and the expenditure of a large amount of money

South Africa government decided to no longer invest in the PBMR project. (Was taken into account the significant investment already made and the impressive scientific advances already achieved in pioneering this particular form of nuclear technology**)**

Some reasons:

1. **No customers had been won.**
2. **There were important delays in project.** (The costs of the demonstration plant have escalated by a factor of more than seven)
3. **No foreign investors and partners were achieved.** (A demonstration plant will not be economically viable by itself)
4. **The opportunity to participate in the USA's Next Generation Nuclear Plant (NGNP) program part of the Westinghouse consortium was lost when Westinghouse withdrew.**
5. **Valuation of economic parameters,** such as operating performance, operating cost and decommissioning cost and others **were made in optimistic form.**

The Next Generation Nuclear Plant Demonstration Project

- ***The NGNP Demonstration Project*** *was formally established to demonstrate the generation of electricity and/or hydrogen with a* ***high-temperature nuclear energy source .***
- *Through scientific and international collaboration, NGNP supports the development of* ***gas-cooled nuclear reactor technology*** *that promises improved performance in* ***sustainability, economics, and proliferation resistance.***
- *The Project is executed in* ***collaboration with industry, US DOE national laboratories, universities****, and the international community.*
- *The NGNP Demonstration Project includes* ***design, licensing, construction, and R&D.***
- *Program* ***Budget 2012 of 49.6 millions USD****.*

The deep burn modular helium reactor (DB-MHR) has been conceived by General Atomics (GA) as a new generation of nuclear power plants, graphite-moderated and helium-cooled

Uses **four modules of 600 MW to destroy Pu** from LWRs and, at the same time, to produce electricity or generate high temperature process heat (hydrogen) or other purposes. **Great flexibility in the choice of the fuel** type: fertile and non-fertile cores (e.g.Th, U or Pu).

Low power density over the whole reactor core about **2.1 MW/m3** (the average power density in the power generating rings is **6.23 MW/m3**).

Use a special type of graphite (The H451 nuclear-grade graphite) :

•Inhibits exothermic oxidation reactions (graphite fires).
•Contains a low concentration of impurities.
•High density.
•High thermal conductivity and specific heat.

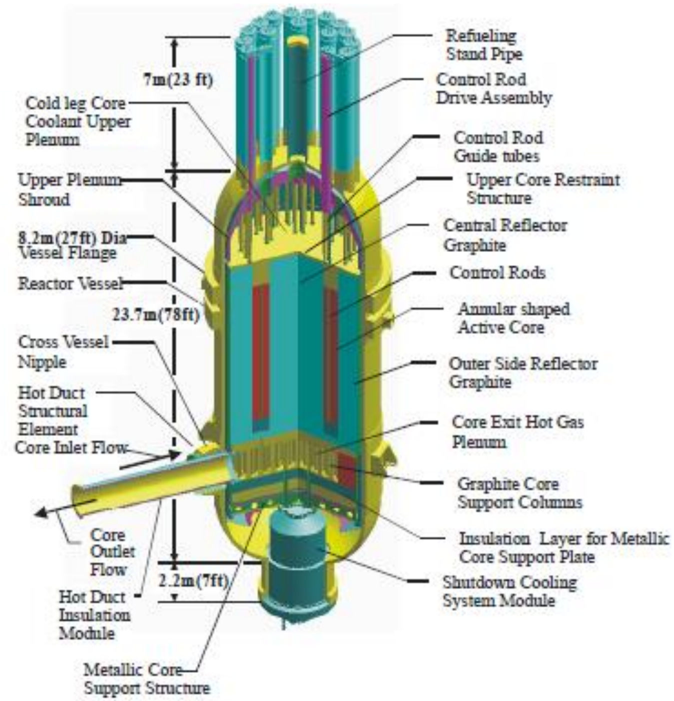


Fig. 1. GT-MHR reactor vessel cutaway showing the arrangement of the reactor components

The MIT pebble bed project is developing a conceptual design of a 250 Mw_{th}–120Mw_e modular pebble bed reactor using an indirect helium to helium heat exchanger gas turbine cycle power plant.

Table 1. Nuclear Specifications for the MIT Pebble Bed Reactor

Thermal Power	250 MWth - 120 MWe
Target Thermal Efficiency	45 %
Core Height	10.0 m
Core Diameter	3.5 m
Pressure Vessel Height	16 m
Pressure Vessel Radius	5.6 m
Number of Fuel Pebbles	360,000
Microspheres/Fuel Pebble	11,000
Fuel	UO_2
Fuel Pebble Diameter	60 mm
Fuel Pebble enrichment	8%
Uranium Mass/Fuel Pebble	7 g
Coolant	Helium
Helium mass flow rate	120 kg/s (100% power)
Helium entry/exit temperatures	520/900 C
Helium pressure	80 bar
Mean Power Density	3.54 MW/m^3
Number of Control Rods	6

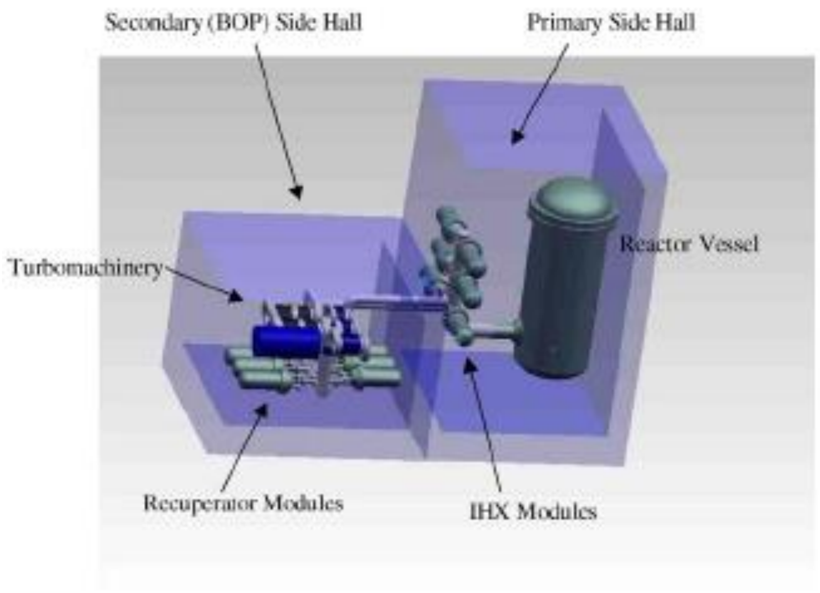
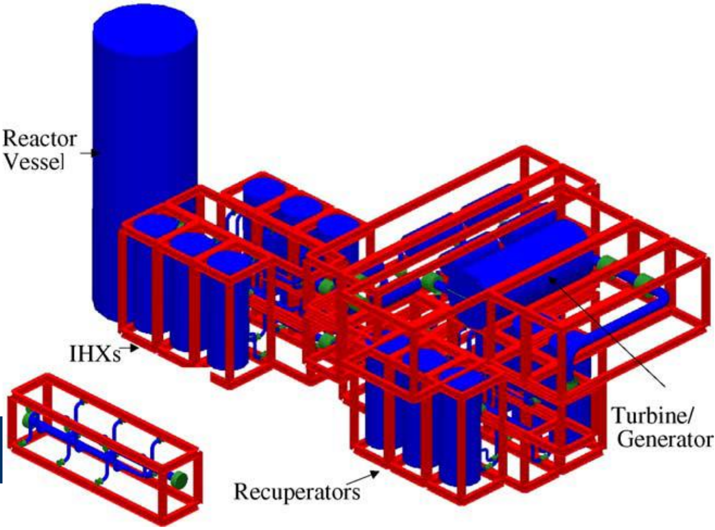


Fig. 3. Preliminary conceptual layout of the MIT PBR.

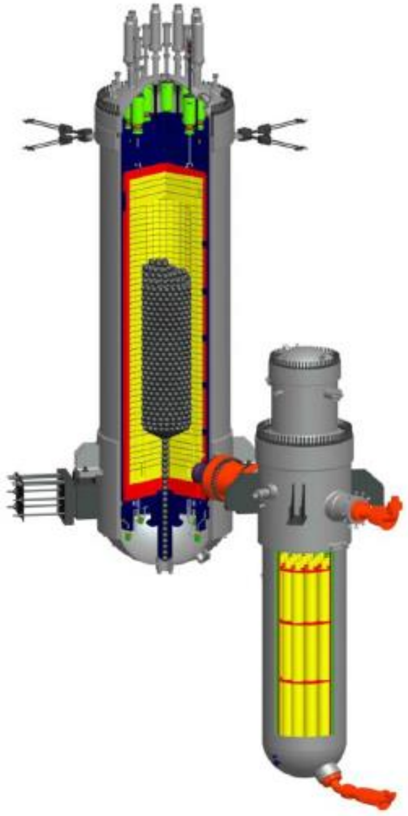
Main features of MIT design



1. **An intermediate cycle** allows **for design flexibility** in the secondary side to produce **either electric power** or **very high grade heat to hydrogen** production plants.
2. **The isolation from the primary system is a safety measure** to avoid contamination on the secondary side.
3. **Modularity in manufacturing requires** that all the components be able to be transported **by truck or train**. It offers potentially large advantages in terms of **shorter construction times**, **lower costs** of power and **less financial risk**.
4. It is envisioned as being a **heat source** for **many other applications** such **as hydrogen production and oil sands bitumen extraction.**

Chinese 2x250MWth HTR-PM demonstration plant

1. **The Institute of Nuclear and New Energy Technology** (INET) of Tsinghua University developed and designed an HTR demonstration plant, called the HTR-PM ***(High-Temperature-Reactor Pebble-Bed Module)*** based on HTR-10.
2. The HTR-PM plant consist of **2 nuclear steam supply system.**
3. Each system has a **PBMR of 250 MWth and** a Steam Generator.
4. The two modules feed **one steam turbine** and generate an electric power of **210 MW.**
5. A pilot fuel production line is being built to fabricate **300,000 pebble fuel elements per year**.

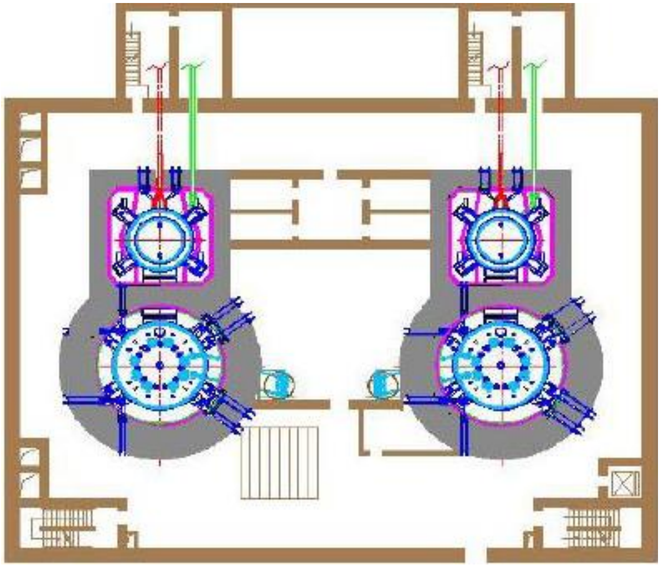


HTR-PM Designs Parameters

Plant electrical power, MWe	211
Core thermal power, MW	250
Number of NSSS Modules	2
Core diameter, m	3
Core height, m	11
Primary helium pressure, MPa	7
Core outlet temperature, ℃	750
Core inlet temperature, ℃	250
Fuel enrichment, %	8.9
Steam pressure, MPa	13.25
Steam temperature, ℃	567

HTR-PM is an up-scaling of the HTR-10.

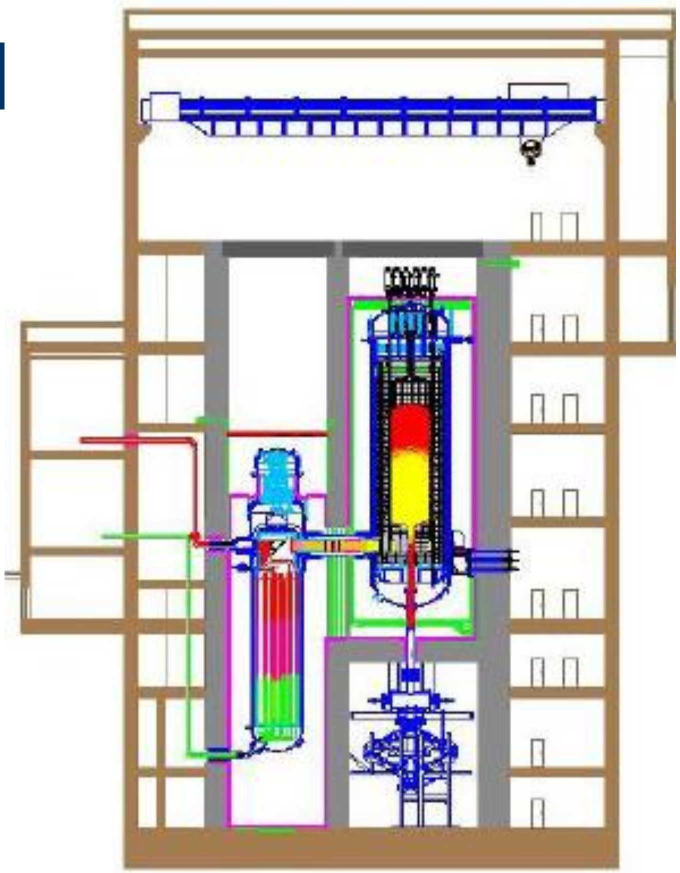
Two NSSS modules in one Building



The main technical objectives of the HTR-PM project are:

a) Demonstrate the **claimed inherent safety features.**
b) Help reveal the **potential economic competitiveness.**
c) Reduce **technical risks**. (Experiences made with the HTR-10 and other mature industrial technologies).
d) Provide a sound basis for achieving **modularized design and construction.**
e) **The most difficult key-issue** of the HTR-PM demonstration plant will be **to show** that an Nth of-its-kind HTR-PM plant **will be economically viable**

Reactor Building



HTR-PM construction

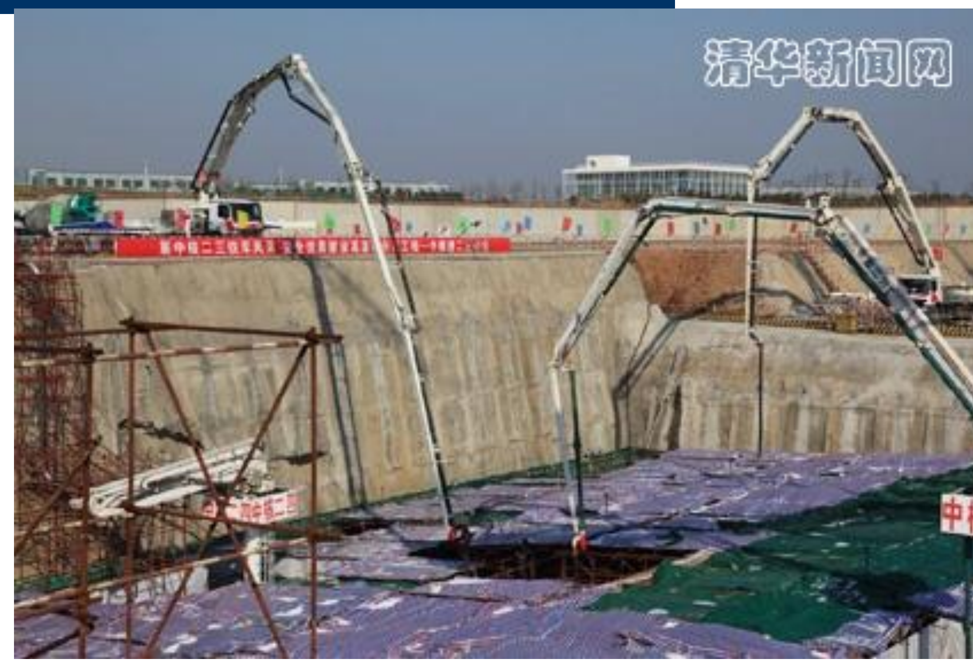
HTR-PM Demonstration Plan site

Future of HTR Development

- **Commercialization:**
Duplication, mass production

- **Next step project:**
Super critical steam turbine,
co-generation

- **R&D on future technologies:**
Higher temperature,
Hydrogen Production,
Process heat application,
Gas turbine



First Concrete in December 2012

Conclusions

The attractive advantages of VHTR justify the big effort made by the nuclear international community to develop this technology.

High core outlet temperature.
hydrogen
cogeneration.
heat supply.
High conversion energy efficiency.
Clear passive safety mechanism.
Deep burn up.
Modularity and standardization.
Minimal waste (Once-Through Fuel Cycle)

Conclusions

The test reactors, the prismatic HTTR-30 and the pebbled bed HTR-10, have demonstrated good performances and they assure the future possibilities of VHTR.

The Chinese HTR-PM demonstration plant, in construction now, should show that an Nth of-its-kind HTR-PM plant will be economically viable, and this will be an important step in the way to VHTRs development.

Thank you!!!!