Status report 96 - High Temperature Gas Cooled Reactor - Pebble-Bed Module (HTR-PM)

Overview

Full name	High Temperature Gas Cooled Reactor - Pebble-Bed Module
Acronym	HTR-PM
Reactor type	Pebble Bed Type Reactor
Coolant	Helium
Moderator	Graphite
Neutron spectrum	Thermal Neutrons
Thermal capacity	500.00 MWth
Gross Electrical capacity	211.00 MWe
Design status	Detailed Design
Designers	Tsinghua University
Last update	10-08-2011

Description

Introduction

1.1 Development history

The High Temperature Gas Cooled Reactor - Pebble-bed Module (HTR-PM) is a modular High Temperature Gas Cooled Reactor (HTGR) demonstration power plant which is designed by the Institute of Nuclear and New Energy Technology (INET), Tsinghua University of China. The current HTR-PM design falls into the category of innovative small sized reactors, featuring 200MW electrical output by two reactor modules. In January 2006 the project named "Large Advanced Pressurized Water Reactor and High-Temperature Gas-cooled Reactor Nuclear Power Plants" became one of the 16 top priority projects of the "Chinese Science and Technology Plan" for the period 2006 - 2020. By February 2008 the implementation plan and the budget for the HTR-PM project was approved by the State Council of China. The demonstration nuclear power plant is being constructed in Rongcheng, Shandong Province, China. The construction of the plant is scheduled to be completed by the end of 2013.

Research and development work in China on high temperature gas cooled reactors started in the 1970s[1-3]. In the 1970s, the research and development work was focused on gas-cooled breeders using thorium fuel cycle. The R&D and design work was carried out for a helium cooled thorium breeder of 100MW output, using spherical fuel elements. The research and development work included:

- Neutronic and thermo-hydraulic code development and reactor conceptual design,

- Tests and design for graphite reactor core internals, pressure vessels, reactor shut-down components, helium circulators, steam generators and other key components,
- Process system engineering,
- Helium technology,
- Coated particle and fuel element technology as well as spent fuel reprocessing.

Though the thorium reactor project was terminated in the late 1970s, the work in this time period resulted in a lot of R&D achievements. It formed good basis for further development work from the technological point of view.

In the early 1980s, research and development work of gas-cooled reactor technology continued in China, and international cooperation was introduced into the R&D work. INET started to cooperate with the Nuclear Research Centre, Juelich, Germany to perform design and safety studies of modular HTGR designs. Much work was done on code development for reactor physics, thermohydraulics and safety analysis. During this time period, the research and development work on gas-cooled reactors was supported by the State Science and Technology Commission of China. Research projects included code development, safety analysis, fuel and fuel cycle technology as well as HTGR application. Conceptual designs of modular high-temperature gas-cooled reactors were made for different fuelling schemes and for different application purposes. At this time, the conceptual design of a two-zone core modular reactor was proposed, with a centre moving zone of graphite pebbles which are fuel-free. This two-zone reactor could generate thermal power of up to 500MW and the maximum fuel temperature could still be limited to allowable values at loss of coolant accident which is the most important safety feature of modular HTGR designs.

Research work on HTGR applications was also performed in the early 1980s. Investigations were made into the oil recovery and petrochemical industries and into the coal industry to study the application potential of HTGR on power-heat co-generation basis. The purpose was to use HTGR nuclear heat for the substitution of fossil fuels to generate the great amount of process heat which is used in these industries, thus saving a lot of fossil fuels and resulting in less environmental pollution. Under cooperation with German industries and research centres, feasibility studies were made together with the Shengli Oil Field and Yanshan Petrochemical Complex to use HTGR process heat for the heavy oil recovery at the oil field and for the petrochemical plant.

In 1986, the HTGR R&D projects were included into the national high technology program. Research and development activities on modular HTGR technologies started to be carried out systematically. R&D projects covered the following areas, with several research and design institutions in China and international cooperation being involved:

- Modular HTGR design,
- Fuel manufacture technology,
- Reprocessing technology of thorium fuel cycle,
- Graphite and other structural materials,
- Helium technology and components,
- Pressure vessel,
- Pebble bed flow and fuel handling technology.

From the above research work, the following main achievements were made:

- Methodologies and capabilities for designing HTGRs were established, and the conceptual design of a 10MW pebble bed test reactor was worked out.

- Remarkable progress was made in the manufacturing technology of spherical coated particle fuels, including progresses in manufacturing fuel kernel, coating, matrix graphite, fuel element, and in the measurement techniques. Equipment for fuel manufacturing was installed, sample coated particles were manufactured which is proved to have good performance.

- A number of test facilities were set up, for example helium test loops and fuel handling test rigs. Experiments were performed to test helium sealing and helium purification technologies and fuel handling components. Research was also made on fuel burn-up measurement. Progress was also made in research of metallic alloy and insulation materials.

- HTGR development strategies up to the year 2000 were formulated. It was made recognized that modular high temperature gas-cooled reactors have advanced inherent safety features, and they have the potential to be used for power generation with high efficiency and for process heat applications. Considering the conditions in China, it was proposed, as the first major strategic step of HTGR development, to build a 10MW pebble bed test reactor by the year 2000. The purpose of the erection of the test reactor is to acquire the know-how in designing, constructing and operating modular helium cooled reactors.

In March 1992, the state government approved the project of constructing the 10MW pebble bed high temperature gas-cooled test reactor (referred to as HTR-10). Basic design of HTR-10 was completed in 1994[4], and the Construction Permit was issued after licensing review of the Preliminary Safety Analysis Report and other relevant application documents. The construction of HTR-10 formally started with the first concrete being poured in June 1995. In parallel to the engineering design and construction work, a number of engineering tests were made, including tests for the fuel handling system, reactor shutdown system, once-through steam generator, etc. By the end of 1998, key components like pressure vessels, the steam generator and helium circulator were manufactured, and installation and commissioning work started. In the meantime, spherical fuel elements were manufactured and qualified through irradiation experiments. In December 2000, the HTR-10 reactor was loaded with nuclear fuel and went critical.

In January 2003, the reactor reached full power of 10MW. The HTR-10 reactor is currently under operation, and safety experiments are being performed with the test reactor, and HTR-10 is supposed to operate for long term. At the same time, development work of helium turbine technologies is being performed.

From April 2003 to September 2006 INET completed four experiments to confirm and verify claimed crucial inherent safety features of modular HTRs:

- Loss of offsite power without any counter-measures,
- Main helium blower shutdown without any counter-measures,
- Loss of main heat sink without any counter-measures,

and especially

- Withdrawal of all rods without any counter-measures.

All these experiments were authorized, guided and supervised by the National Nuclear Safety Authority.

During this R&D-period of the HTR-10, five significant achievements were obtained:

(1) *Manufacture of spherical coated particle fuel element*: the know-how of fabricating fuel elements for the HTR-10 was mastered. The free Uranium fraction could steadily be decreased to a value as low as 3×10^{-5}.

(2) *Corresponding technologies for pebble-bed HTGRs*: the technology of fuel element handling and spherical fuel element transportation by pulse pneumatic mechanism,

(3) *Helium process technologies*: such as helium sealing and purification, the lubrication for rotating equipments in a helium atmosphere, electrical insulation and rotor dynamics,

(4) *Domestic manufacture of key equipments for HTGRs*: this mainly covers the reactor pressure vessel, the steam generator pressure vessel, the hot gas duct, the once-through steam generator with helical tubes, the helium blower, the fuel handling equipments and reflector graphite components.

(5) Successful development of fully digital reactor protection systems.

A second step of HTGR-application in China was started in 2001 [8-11] when the High Temperature Gas-cooled Reactor - Pebble-bed Module (HTR-PM) project was launched. The preliminary investment agreement was signed in December 2004 by “China Huaneng Group”, by “China Nuclear Engineering Corporation” and by “Tsinghua Holding Corporation”. In January 2006 the project named “Large Advanced Pressurized Water Reactor and High-Temperature Gas-cooled Reactor Nuclear Power Plants” became one of the 16 top priority projects of the “Chinese Science and Technology Plan” for the period 2006 – 2020. By February 2008 the implementation plan and the budget for the HTR-PM project was approved by the State Council of China. In November 2003, the “Chinergy

Company" has been established and was designated to be the main contractor of the HTR-PM nuclear island, while in January 2007 the "Huaneng Shandong Shidao Bay Nuclear Power Company" was founded being the owner of the HTR-PM demonstration plant.

Compared with HTR-10, the components of HTR-PM are scaled up, for example the helium circulator and the fuel handling equipment. From the HTR-10, certain research and development work on some special technologies or equipment are ongoing supported by the central government under the project "Large Advanced Pressurized Water Reactor and High-Temperature Gas-cooled Reactor Nuclear Power Plants". Significant progress has also been made in Rongcheng, Shandong Province, the site for demonstration power plant. Currently, both the Preliminary Safety Analysis Report and the Environmental Impact Assessment Report have been approved by the National Nuclear Safety Administration. The first tank of concrete will be poured soon.

1.2 HTR-PM technological goals and application

The HTR-PM should achieve the following technical goals:

(1) Demonstration of inherent safety-features

The inherent safety features of modular HTGR power plants guarantees and requires that under all conceivable accident scenarios the maximum fuel element temperatures will never surpass its design limit temperature without employing any dedicated and special emergency systems [e.g. core cooling systems or special shut-down systems, etc.]. This ensures that accidents [e.g. similar to LWR core melting] are not possible so that unacceptable large releases of radioactive fission products into the environment will never occur.

(2) Demonstration of economic competitiveness

The first HTR-PM demonstration power plant will be supported by the Chinese government, so that the owner can always maintain the plant operation and obtain investment recovery. However, this government supported demonstration plant has to demonstrate that a cost overrun during the construction period will be avoided and that the predicted smooth operation and performance will be kept. Hence, the demonstration plant must clearly demonstrate that follow-on HTR-PM plants will be competitive to LWR plants without any government support.

(3) Confirmation of proven technologies

In order to minimize the technical risks, the successful experiences gained from the HTR-10 and from other international HTGR plants will be fully utilized in the HTR-PM project. The HTR-PM reactor design is very similar to the HTR-10. The turbine plant design will use the mature technology of super-heated steam turbines which is widely used in other thermal power plants. Besides, the manufacture of fuel elements will be based on the technology verified and proven during the HTR-10 project. In addition, the key systems and equipments of the plant will be rigorously tested in large scale experimental rigs in order to guarantee the safety and reliability of all components. Furthermore, international mature technologies and successful experiences will be absorbed through international technical consultations.

(4) Standardization and modularization

The HTR-PM demonstration plant, consisting of two pebble-bed module reactors of combined 2**x**250 MW_{th} power, adopts the operation-mode of two modules connected to only one steam turbine/generator set. This design allows to demonstrate the advantages and key benefits of employing and implementing a design of standardization and modularization. If the construction and operation of the HTR-PM demonstration plant proves to be successful, larger scale HTR-PM plants - using multiple-modules feeding one steam turbine only - will become a reality.

The HTR-PM is a land-based modular nuclear power generating unit. As the first industrial HTGR demonstration plant in China, it is currently designed for the purpose of power generation. When market requires, more modules can be constructed in series to form a larger nuclear power plant with appropriate power output. The plant design is aimed at standardization and modularization.

1.3 Summary level Technical Data

Items	Characteristics
Installed capacity (thermal and electrical)	2**x**250 MW_{th}, 211 MW_e
Mode of operation (basic, load follow)	Basically for base load, potential for load following
Availability target	Availability factor target: 90% Load factor target: 85%
Summary of major design characteristics (e.g. fuel types, enrichment, coolant and moderator materials, structural materials, core dimensions, fuel dimensions, core material dimensions, vessel material and dimensions, number of circuits/loops)	Type of fuel: UO_2 enriched to 8.5% in coated particles dispersed in graphite matrix, spherical fuel elements of 6cm diameter. Type of coolant/moderator: Helium/Graphite. Type of structural material: Graphite and carbon bricks as core structural materials, steel vessels. Core type/characteristic dimensions: Pebble bed core, 3.0m in diameter, 11.0m in height. Vessel type/characteristic dimensions: Carbon steel reactor vessel, 5.7m of inner diameter and 25 m of overall height. Cycle type: Indirect cycle with steam generator Number of circuits: Three circuits, with the first one being the primary system cooling the reactor, the second one being the steam turbine circuit and the third one being the condenser cooling water circuit.
Simplified schematic diagram of the nuclear installation	see Figure 1
Neutron physical characteristics and power peaking factors	Strong negative temperature reactivity feedback over the whole temperature range and for all operational states Average power density of fuel zone: 3.2 MW/m^3. Maximum power density of fuel zone: 6.6 MW/m^3. Average output power per fuel ball: 0.6kW. Maximum power per fuel ball: 1.8kW.
Reactivity control mechanisms	Control rod movement is the main reactivity control mechanism. Application of small absorber ball system is mainly for cold shutdown.

Cycle type and thermodynamic efficiency	Indirect steam turbine cycle with reheating. Turbine generator thermodynamic efficiency: 40%.
Thermal hydraulic characteristics	See Table 1
Maximum/average discharge burn up of fuel	90,000MWd/tU
Fuel lifetime or residence time in the core	1057 days
Design lifetime for reactor core, vessel and structures	40 years
Design and operating characteristics of systems for non-electric applications, including type, ranges for sharing energy production between different applications and specific production rate per unit of thermal and equivalent electric energy	Currently not applicable

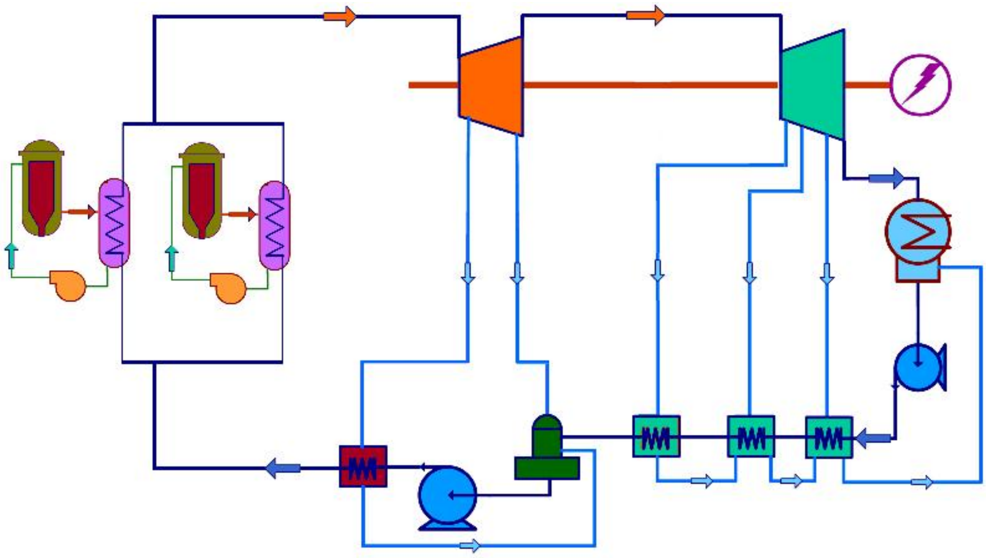


Fig. 1 - Simplified Schematic Diagram of the HTR-PM Demonstration Power Plant

Description of the nuclear systems

2.1 Main Characteristics of the Primary Circuit

HTGR uses helium as coolant and graphite as moderator as well as structural material. A single-zone core design was adopted, in which the spherical fuel elements are placed. The cylindrical active reactor core has an outer diameter of 3.0 meters and effective height of 11.0 meters. The effective core volume is $77.8m^3$. In equilibrium core, the reactor core contains 420,000 fuel elements.

The reflectors include top, side and bottom graphite reflectors. Graphite reflectors are made from graphite blocks, which are constructed layers by layers. In the circumferential direction, every lay of graphite reflectors consists of 30 graphite blocks. Inside the side graphite reflector blocks, corresponding numbers of channels are designed for reactor shutdown systems and for helium flow. The bottom reflector takes a cone shape at the upper surface to facilitate the pebble flow. Inside the bottom reflector, channels are designed for the flow of hot helium. The hot helium chamber is designed in the bottom reflector area, where hot helium of different outlet temperatures is mixed and then directed to the hot gas duct in which the hot helium flows to the Steam Generator. In the centre of the bottom reflector is the fuel discharge tube.

The primary helium coolant works at the pressure of 7.0MPa. The rated mass flow rate is 96kg/s. Helium coolant enters the reactor in the bottom area inside the pressure vessel with an inlet temperature of 250ºC. Helium coolant flows upward in the side reflector channels to the top reflector level where it reverses the flow direction and flow into the pebble bed in a downward flow pattern. Bypass flows are introduced into the fuel discharge tubes to cool the fuel elements there and into the control rod channels for control rods cooling. Helium is heated up in the active reactor core and then is mixed to the average outlet temperature of 750ºC and then flows to the Steam Generator.

The reactor core and the steam generator are housed in two steel pressure vessels which are connected by a connecting vessel. Inside the connecting vessel, the hot gas duct is designed. All the pressure retaining components, which comprise the primary pressure boundary, are in touch with the cold helium of the reactor inlet temperature. The primary pressure boundary consists of the reactor pressure vessel (RPV), the steam generator pressure vessel (SGPV) and the hot gas duct pressure vessel (HDPV), which all are housed in a concrete shielding cavity as shown in the figure.

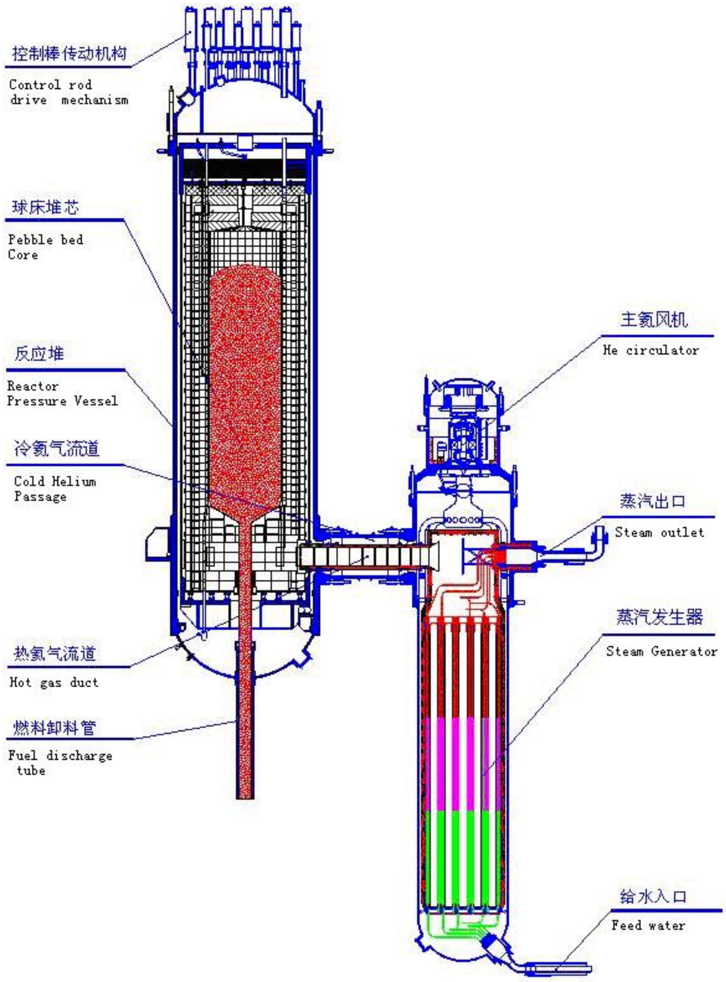


Fig. 2 - HTR-PM primary loop

2.2 Fuel Design

Fuel elements are spherical ones. Every fuel element contains 7g heavy metal. The enrichment of U-235 is 8.5%. Uranium kernels of about 0.5mm diameter are coated by three layers of pyro-carbon and one layer of silicon carbon. Coated fuel particles are dispersed in matrix graphite which is 5cm in diameter. Surrounding the fuel containing graphite matrix is a 5mm thick graphite layer. Figure 3 shows the fuel cross sections. The key design data of fuel elements are given below:

Fuel Element:

Diameter of ball	6.0 cm
Diameter of fuel zone	5.0 cm
Heavy metal (uranium) loading (weight) per ball	7.0 g
Enrichment of ^{235}U (weight)	8.5 %

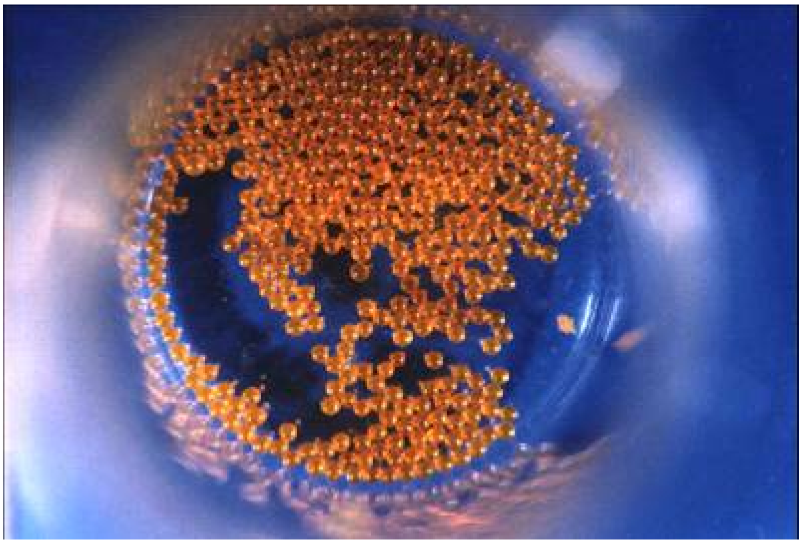


Fig. 3 - UO2 kernels

Fig. 4 - Spherical Fuel Elements

2.3 Fuel handling system

The operation mode adopts continuous fuel loading and discharging: the fuel elements drop into the reactor core from the central fuel loading tube and are discharge through a fuel extraction pipe at the core bottom. Subsequently, the discharged fuel elements pass the burn-up measurement facility one by one. Depending on their state of burn-up they will either be discharged and transported into the spent fuel storage tank when having reached their design burn-up, or they will be re-inserted into the reactor to pass the core once again. The power distribution of the core depends on the number of passes one chooses. Obviously, the higher the number of passes is chosen the flatter will be the power distribution. This is favorable when regarding fuel element temperatures during accident conditions. On the other hand, a high number of fuel passes complicates the fuel handling devices as well as the complexity of the burn-up measurement facility.

2.4 Primary circuit component description

The reactor internals consist of graphitic, carbonic and metallic components. The graphitic internals act primarily as the neutron reflector; in addition they provide the means to be able to arrange the helium flow channels and the absorber borings. The main function of the metallic internals is to support the graphite and carbon internals along with the ceramic structure of the pebble-bed core, and to pass various loads and forces to the reactor pressure vessel.

The main helium blower, designed as a vertical structure, is installed on the top of the steam generator inside the steam generator pressure vessel [SGPV]. The electric motor is mounted on an insertable assembly; the motor is driven by the converter outside of the pressure vessel. A magnetic bearing system is envisaged.

The steam generator consists of 19 separate helical tube assemblies; each assembly has 5 layers and includes 35 helical tubes. To ensure two-phase flow stability, throttling apertures are installed at the entrance of all helical tubes. The assembly-type design of the steam generator uses the experiences from the steam generator employed in the HTR-10. In-service inspection is possible. For full verification of the steam generator assembly full scale testing will be performed.

2.5 Auxiliary systems

As the main auxiliary system to the NSSS (Nuclear Steam Supply System), the helium purification system aims to purify the primary coolant and to eliminate impurities such as H_2O, O_2, CO, CO_2, H_2, N_2, CH_4, etc. The system comprises sub-systems and components with functions respectively of helium purification, regeneration of purifier, oxygen supply and storage, liquid nitrogen supply and storage, helium supply and storage, primary circuit vacuum, etc.

2.6 Operating modes

The HTR-PM plant is designed primarily for base load operation, but it can also be operated for load following.

2.7 Standard Fuel cycle

The current HTR-PM design uses Low Enriched Uranium (LEU) as nuclear fuel. Fissile material (U-235) is enriched to 8.5% for fresh fuel elements. The fuel cycle, which is currently foreseen, is the Once-Through cycle. In the reactor operation, spherical fuel elements go through the reactor core in a multi-pass mode before they reach the discharge burn-up and are discharged from the core into storage tanks. The average discharge burn-up is 90,000MWd/tU.

2.8 Alternative Fuel options

The currently foreseen fuel cycle for the HTR-PM technology is LEU uranium-plutonium Once-Through cycle. In comparison to the existing water cooled reactor technologies, it uses the uranium resources more efficiently. If necessary, the spent fuel can be reprocessed to recover the remaining fissile materials in the spent fuel in order to better use the fissile material resources. The HTR-PM technology, which is a graphite moderated, helium cooled reactor system, is flexible in fuel cycle technologies. It is also able to use the thorium-uranium fuel cycle. Considering the large amount of the thorium resources in China, the HTR-PM technology has good features in terms of sustainability.

2.9 Spent nuclear fuel and disposal plans

As was stated, the currently foreseen fuel cycle is the once through cycle. The spent fuel can eventually be reprocessed in case fuel cycle strategies would change to recover the uranium and plutonium fissile material in the spend fuel. There has been much development on the reprocessing technology. The first steps would be to remove the matrix graphite in the fuel elements and the pyro-carbon and silicon carbon coating layers. The remaining spent fuel materials can be reprocessed using the existing technologies for current water reactor fuels.

The proposed option for spent fuel storage and disposal is to store the spent fuels at the plant site for certain period, and then to store them at a centralized storage before they go to final disposal. For the moment, national strategies for the management of spent fuel elements with regards to storage and final disposal need to be formulated yet. These strategies and the implementation will depend heavily on the scale of eventual capacities of the constructed plants. For the first demonstration plant, the spent fuel elements of the whole plant life are planned to be interim stored at the plant site.

2.10 Examples of energy systems with NPPs of this kind

There is currently a lot of interest worldwide in the development and deployment of modular high temperature gas cooled reactors. Besides in China, similar development and design activities are also underway in France, Japan, Russia, South Africa, USA, etc[5-7]. The main differences among different programmes lie in the fuel technology (pebble fuel versus prismatic fuel, uranium versus plutonium) and power conversion technology (steam turbine versus gas turbine).

Description of safety concept

The safety design philosophy of the HTR-PM is to realize required high level of safety while at the same time to maximum simplify the design for systems that are required only for safety purposes. Emergency measures outside the plant boundary should be made technically not necessary or to a minimum level. The HTR-PM safety design is heavily based on inherent features and passive safety systems, while still adhering to the defense-in-depth principles. Following three main features characterize the basic safety concept of the HTR-PM design.

- Retention of radioactivity materials is achieved through multi-barriers with heavy emphasis on the fuel elements (the first barrier) also for accidental conditions. The fuel elements with coated particles serve as the first barrier. Every fuel kernel of about 0.5mm diameter is coated with three layers of pyro-carbon and one silicon carbon (SiC) layer. A large number of coated particles are dispersed in matrix graphite of 5cm in diameter to form the fuel containing part of a fuel element, which in turn is protected by a 0.5cm thick fuel-free graphite layer. The fuel elements used for HTR-PM have been demonstrated to be capable of retaining fission products within the coated particles under temperatures of 1,620ºC which is not expected for any plausible accident scenarios. The second barrier is the primary pressure boundary which consists of the pressure vessel units of the primary components. The third barrier is the reactor building and some additional auxiliary buildings which house primary helium containing components.

- Decay heat shall be removed inherently under accident conditions. Under accident conditions, the primary helium circulator shall be stopped. Because of the low power density and the large heat capacity of the graphite structures, the decay heat in the fuel elements can dissipate to the outside of the reactor pressure vessel by means of heat conduction and radiation within the core internal structures, without leading to unacceptable fuel temperature limits.

- Overall negative temperature reactivity feedback is assured under all conditions. The reactor nuclear design assures that the temperature reactivity coefficients of fuel and moderator are always negative under all operating and accident conditions. This, together with the protection action of stopping the primary helium blower under accidents, will lead to reactor automatic shutdown in accident cases.

The HTR-PM design makes good use of the above safety features. Because the reactor safety is very much characterized by the inherent safety features, the plant engineered safety systems are kept minimum, so that the plant safety and safety management is simplified to very high extent. It is also clear and transparent. When accidents occur, a very limited number of reactor protection actions shall be called upon by the reactor protection system. No or very limited actions through any systems or human interventions are foreseen after the limited reactor protection actions are activated. The limited reactor protection actions shall be to trip the reactor and the helium circulator, to isolate the primary and secondary systems. When there is large leak or rapture of steam generator heat exchange tubes, a water discharge system is designed to minimize the amount of water ingress into the reactor core.

Another feature of the HTR-PM design is the long time period of accident progression due to the large heat capacity of fuel elements and graphite reactor internal structures. It requires days for the fuel elements to reach the maximum temperature when the coolant is completely lost.

The safety design and operation of an HTR-PM plant follows the defence-in-depth principles which are well formulated in IAEA safety standards. The first three levels of defence-in-depth apply for the HTR-PM design and operation, namely, Level 1 - Prevention of abnormal operation and failure, Level 2 - Control of abnormal operation and detection of failure, and Level 3 - Control of accidents within the design basis. For Level 4 of defence-in-depth (Mitigation of radiological consequences of significant release of radioactive materials), the mitigation measures are made minimum, as accident consequences are limited with extremely high level of confidence at the first three levels of defence-in-depth. For Level 5 of defence-in-depth (off-site emergency responses), the HTR-PM design is aimed at making such off-site emergency response measures technically not necessary.

Accidents classification and safety analysis of the HTR-PM design will be performed according to requirements of standards, and the design, licensing and operation experience of the HTR-10 test reactor. For the HTR-PM plant, the most significant design basis accident is the complete loss of primary helium coolant. This accident has been analysed. In such an accident scenario, no any additional cooling systems are designed for cooling the reactor core. The decay heat in fuel elements shall dissipate to the outside of the pressure vessel primarily by means of heat conduction and radiation within the reactor core internals. The maximum fuel temperature during this accident scenario is the highest and most challenging for the fuel, therefore this accident scenario is the most critical one for the HTR-PM design. In the case of HTR-PM, the maximum fuel temperature will be less than 1620°C for any conceivable accident.

Proliferation resistance

Some of the important technical features of HTR-PM, which reduce the attractiveness of its spent nuclear fuel material for use in nuclear weapon programmes, include:

- The volume of new and spent fuel is rather large, and the inspection and verification should be easy.
- The burn-up level of the discharged fuel is as high as 90,000MWd/tU, less fissile materials are left in spent fuel and they are of much less interest to weapon grade fissile materials.
- The currently adopted fuel cycle is the once-through cycle using low enriched uranium fuel, no reprocessing is foreseen.
- It requires sophisticated and costly reprocessing to get any fissile materials from the spent fuel elements.

Safety and security (physical protection)

The safety of an HTR-PM plant is realized in such a way that the nuclear safety does not heavily rely on peripheral engineered safety systems. Radioactive materials in the reactor core will remain retained in the fuel elements as long as the reactor structures remain physically protected. The reactor itself is housed in a very thick concrete cavity inside the reactor building. The reactor building and the primary concrete cavity form robust barriers against sabotage attacks. The reactor safety features also make the reactor itself less sensitive to conceivable internal sabotage actions. Such robust technical features make the safety of an HTR-PM plant much less vulnerable to sabotage actions. Notwithstanding these, routine technical measures required by applicable standards are foreseen for physical protection purposes.

Description of turbine-generator systems

Steam turbine

Type	N211-13.24/566
Rated output power	211MWe
Rated speed	3000rpm
Heat input rate from NSSS	500MW
Rated inlet steam pressure	13.24MWe
Steam flow rate	671t/h
Rated inlet steam temperature	566°C
Rated discharge pressure	4.5kPa(a)
Discharge pressure in summer	9.6kPa(a)
Rated cooling water temperature	16°C

Generator

Rated output power	211MWe
Rated voltage	18kV
Power factor	0.85
Rated frequency	50Hz
Efficiency	98.7%
Cooling method	Air
Excitation method	Static

Condenser

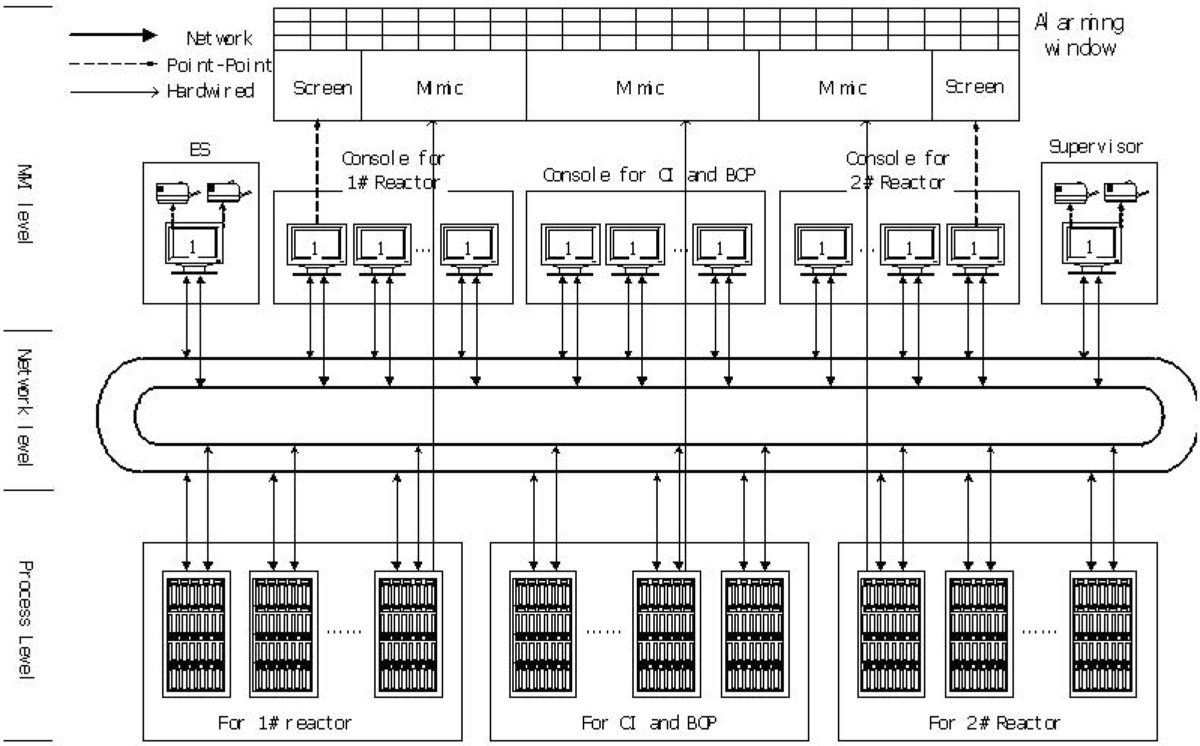
Type	Shell and tube condenser
Cooling area	$12000m^2$
Design pressure for shell	0.098MPa(a)
Design pressure for tube	0.4MPa(a)
Exhaust back pressure	4.5kPa
Tube material	Titanium
Water recirculation rate	27711.4t/h
Design inlet water temperature	16°C
Maximum inlet water temperature	33°C

Electrical and I&C systems

Overall monitoring and controlling function is performed via one integrated distributed control system (DCS) covering the operations of two reactors, two steam generators and pertinent BOP components. One main control room is responsible for control of reactors, electricity and mechanical operations. Visual display units (VDU) are adopted for monitoring of the unit. Both control console and graphic panel are designed in the main control room, with the first responsible for display via VDU and some emergence operation, and the last in charge of report on the basic safety and operation status as the supplementary to the control console. Both reactors and the steam generator are controlled independently, with a coordinating supervisory control module placed on the higher hierarchical level, for dynamic distribution of power outputs of the two reactors to keep the energy balance, according to the grid demand and the operation status of the reactors.

Independent reactor protection system and post-accident monitoring system are designed respectively for the two reactors. The two systems are to ensure the protection and safety monitoring performance upon the failure of the DCS. Redundant protection measures are designed for multi-stage protection of the main components of the plant.

The figure shows the layout of the control system.



Spent fuel and waste management

The proposed option for spent fuel storage and disposal is to store the spent fuels at the plant site for certain period, and then to store them at a centralized storage before they go to final disposal.

The HTR-PM fuel elements are of good capability of retaining radioactive materials generated from the fission process. The graphite-helium system does not have much activation problem. Primary coolant of helium is of low radioactivity level and is constantly purified. During normal operation, doses to operational staff and radioactive releases to environment are very low and are minimized. The amount of liquid and solid radioactive wastes generated during plant operation is extremely small and are well managed. Therefore, the features of the HTR-PM technology with regard to adverse environmental impact are remarkably favorable.

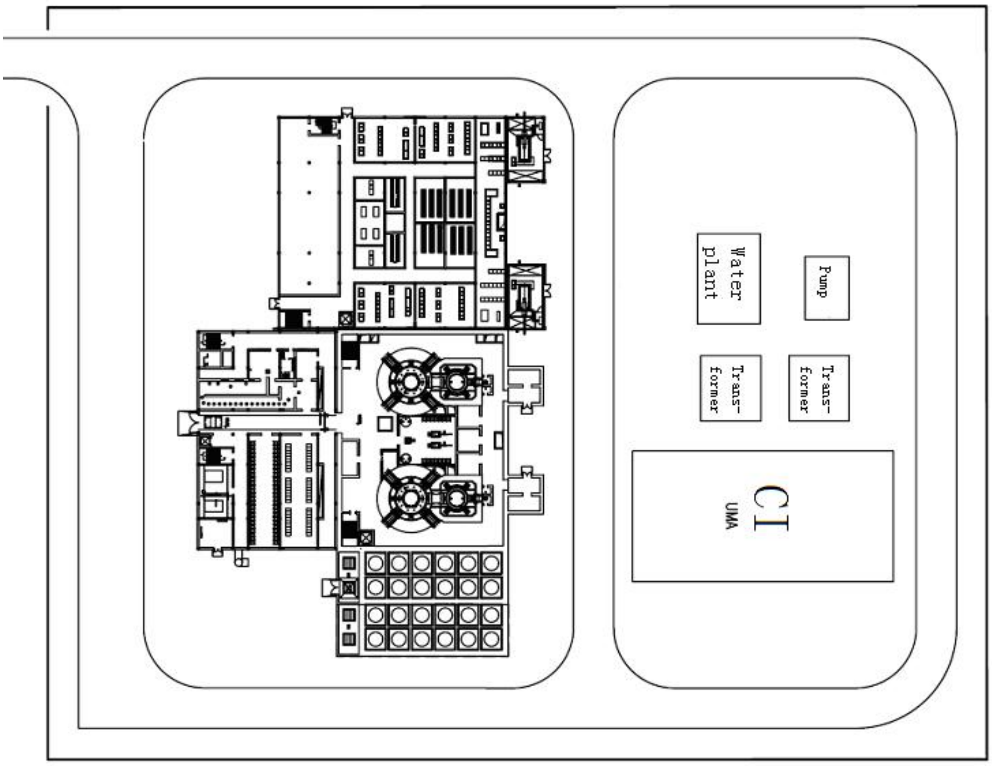
Plant layout

The HTR-PM adopts a layout mode of two-module reactors coupled to one steam turbine in order to verify the feasibility and rationality of coupling a multitude of modules to one steam turbine. In addition, this kind of arrangement enables to test the sharing of the auxiliary systems.

General philosophy governing plant layout:

- The plant layout of the current design deals with the two modules which serve as the demonstration purpose.
- The civil buildings include: reactor building, turbine-generator building, helium storage and supply building, spent fuel storage building, waste treatment building, waste storage building and evaporation pool, fire fighting water supply tank.

The figure shows the arrangement of the reactor building and the turbine generator building.



Plant performance

The HTR-PM will realize the following safety features:

1. The radioactive inventory in the primary helium coolant is very small when the reactors are working at normal operation conditions. Even if this limited amount of radioactivity would be released into the environment following an accident, there is no need to take any emergency measures.
2. For any reactivity accident or for a loss of coolant accident the rise of the fuel element temperature will not cause a significant additional release of radioactive substances from the fuel elements.
3. The consequences of water or air ingress accidents depend on the quantity of such ingresses. The ingress processes and the associated chemical reactions are slow, and can readily be terminated within several dozens of hours (or even some days) by taking very simple actions.

The nuclear safety goal of the HTR-PM can be summarized follows: The consequences of all conceivable accidents will not result in significant offsite radioactive impacts. The plant is designed to meet the safety target of Generation-IV nuclear energy systems which stipulates: "eliminate the need for offsite emergency measures". The same viewpoint is put down by IAEA in its report No.NS-R-1 “SAFETY OF NUCLEAR POWER PLANTS: DESIGN”. Here it is expressively stated that “An essential objective is that the need for external intervention measures may be limited or even eliminated in technical terms, although such measures may still be required by national authorities.”

Development status of technologies relevant to the NPP

The technical research for the HTR-PM began in 2001. The main technical scheme of the nuclear island was finally fixed in 2006.

HTR-PM adopts the proven and mature technology as much as possible. Anyway, in order to ensure the operation reliability and to demonstrate the safety, a large scale of research and experiment are carried out.

The key technology research and engineering verifications are carried out according to elaborate plans. An HTGR engineering laboratory and a large helium engineering testing loop are under construction at INET. Here, the engineering verification experiments for the main component prototypes will be performed on large test rigs in full scale.

According to the requirements of the HTR-PM project, a fuel production line will be built soon having a capacity of producing 300,000 spherical fuel elements per year.

Deployment status and planned schedule

The expected project construction period from pouring the first tank of concrete to generating electricity for the grid is scheduled to be 50 months. Although the workload of building, construction and installation is relatively clear and straight forward, the project schedule, nevertheless, leaves certain time margins allowing for possible uncertainties. The current plan aims for feeding electricity to the national power grid in 2013.

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Technical data

General plant data

Reactor thermal output	500 MWth
Power plant output, gross	211 MWe
Mode of operation	Baseload
Plant design life	40 Years
Primary coolant material	Helium
Moderator material	Graphite
Thermodynamic cycle	Rankine
Type of cycle	Indirect

Reactor core

Active core height	11.0 m
Fuel column height	11.0 m
Average fuel power density	85.7 KW/KgU
Fuel material	UO2
Fuel element type	Spherical
Outer diameter of elements	60.0 mm
Average discharge burnup of fuel	90 MWd/Kg

Primary coolant system

Primary coolant flow rate	96 Kg/s
Reactor operating pressure	7.0 MPa
Core coolant inlet temperature	250 °C
Core coolant outlet temperature	750 °C

Power conversion system

Working medium	Steam
Working medium flow rate at nominal conditions	99.4 Kg/s
Working medium pressure (SG outlet)	14.1 MPa
Working medium temperature (SG outlet)	570 °C

Reactor pressure vessel

Inner diameter of cylindrical shell	3000 mm

Steam generator or Heat Exchanger

Type	Large helical
Number of heat exchanger tubes	665

Residual heat removal systems

Active/passive systems	Passive

Turbine

Turbine speed	3000 rpm
HP turbine inlet temperature	566 °C

Generator

Voltage	18 kV
Frequency	50 Hz

Condenser

Type	Single pass, shell and tube
Condenser pressure	4.5 kPa

Plant configuration and layout

Plant configuration options	Ground-based

Fuel Characteristics

Enrichment	8.5
Diameter of kernels	0.5 mm
Diameter of spheres	60.0 mm
Diameter of Fuel Zone in Sphere	50.0 mm