The HTR-10 project and its further development

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1. Overview of the HTR-10 project

1.1 Objectives[1]

The 10MW High Temperature Gas-cooled Reactor-Test Module (termed as HTR-10) is one of key project in the National High Technology Research and Development Program (1986-2000). Main objectives for the HTR-10 are: (1). To acquire know-how to design, construct and operate the HTGRs, (2). To establish an experimental facility, (3). To demonstrate the inherent safety features of the Modular HTGR, (4). To test electricity and heat co-generation and closed cycle gas turbine technology and (5). To do research and development work for high temperature process heat application. The Institute of Nuclear Energy Technology (INET) of Tsinghua University was appointed as the leading institute to be responsible for design, license applications, construction and operation of the HTR-10.

1.2 Design features[2]

The HTR-10 technical design represents the features of Modular HTGR design. The HTR-10 is a pebble bed high temperature gas-cooled reactor; it uses the spherical fuel elements. The reactor core and the steam generator are housed in two steel pressure vessels, which are arranged side by side. These two vessels are connected to each other by the vessel of the hot gas duct. These three vessels consist of primary pressure boundary. All these steel pressure vessels are in touch with the cold helium of about 250 ^{o}C coming out from the circulator, which sits the top of the steam generator vessel.

The helium at temperature of 250 ^{o}C enters the main circulator and is pressured, then flows into the outer coaxial pipe of the hot gas duct and enters the channels in the side reflector, then flows through these channels from the bottom to the top. The cold gas directly enters reactor core and flows through the pebble bed from the top to the bottom, while it is heated up to temperature of 700 ^{o}C. The hot helium coming out from a hot gas chamber in the bottom reflector flows through the hot gas duct and then enters the steam generator. The heat is transferred to water in secondary circuit while the helium temperature is cooled down to 250 ^{o}C. The cross section of the HTR-10 primary circuit is shown in Fig.1. The main design parameters are listed in Table 1.

The reactor core has a diameter of 1.8m and mean height of 1.97 m and the volume of 5.0 m^3 which is surrounded by the graphite reflectors. The reactor thermal power is 10MW and the mean power density of the core is 2 MW/m^3. The helium coolant in the primary system is designed at the pressure of 3.0MPa and mean core inlet and outlet temperature of 250 ^{o}C and 700 ^{o}C respectively.

The core structure consists of the ceramic and metallic internal structures. Graphite serves as the main material of the core structure, which consists of the top, bottom and side reflectors. The thickness of the side reflector is 100cm, including a 22.5 cm think layer of boronated carbon bricks at the outer periphery. The boronated carbon bricks provide thermal and neutron shielding to the metallic internal structure and the reactor pressure vessel. All graphite and carbon bricks are connected by key-keyway system to keep their relative position. The ceramic core structures are housed in a metallic core vessel, which is supported on the reactor pressure vessel.

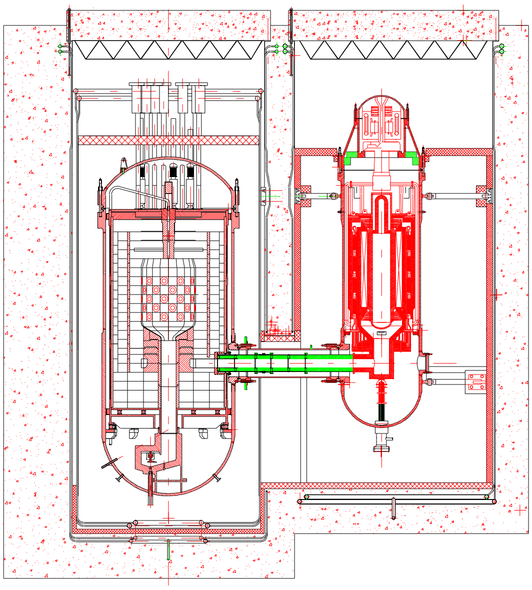

Figure 1 Cross section of the primary circuit

Items	Unit	Value
Reactor thermal power	MW	10
Active core volume	m^3	5
Average power density	MW/m^3	2
Primary helium pressure	MPa	3
Helium inlet temperature	^oC	250/300
Helium outlet temperature	^oC	700/900
Helium mass flow rate	kg/s	4.3/3.2
Fuel		UO_2
U-235 enrichment of fresh fuel elements	%	17
Diameter of spherical fuel elements	mm	60
Number of spherical fuel elements		27,000
Refueling mode		Multi-pass, continuously
Average discharge burnup	MWd/t	80,000

Table 1 Main parameters of the HTR-10

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Fuel elements are the spherical fuel elements (6cm in diameter) with coated particles of TRISO type. The coated particle consists of a UO_2 kernel with diameter of 0.5mm, which is coated with low-density pyrolytical carbon, inner high-density pyrolytical carbon, silicon carbide and outer high-density pyrolytical carbon layer. Each fuel element has about 8,300 coated particles and contains 5g uranium with U-235 enrichment of 17%. The reactor core contains about 27,000 fuel elements. Average discharge burn up is 80,000 MWD/t. Spherical fuel elements go through the reactor core in a "multi-pass" pattern. A pulse pneumatic fuel handling system is used for continually charging and discharging fuel elements. The burn up of the discharged fuel elements is measured individually and those fuel elements which have not reached the designed burn up value will be sent back pneumatically to the reactor core.

There are two independent reactor shutdown systems. The first shutdown system consists of 10 control rods and the second shutdown system consists of seven slotted holes with dimension of 160×60 mm to contain the small absorber balls. Both of them are placed in the side reflector and are able to bring the reactor to cold shutdown conditions and to keep it there. Since the reactor has strong negative temperature coefficients and decay heat removal does not require any circulation of helium coolant, turning-off the helium circulator can also shut down the reactor from power operating conditions. The digital reactor control and protection system is used in the HTR-10. All of data are managed and recorded by computers.

The integrated steam generator and intermediate heat exchanger (IHX) are designed for the HTR-10. The steam generator is once through type, it is composed of 30 modular helical tubes, which are arranged in a circle between two insulation barrels inside the steam generator pressure vessel. The place inside the inner barrel is designed for an intermediate heat exchanger, which is to be installed in the second phase of the project.

Decay heat removal of the HTR-10 is designed on a completely passive basis. At a loss of pressure accident, against which no core cooling is envisaged, decay heat will dissipate through the core structures by means of heat conduction and radiation to the outside of the reactor pressure vessel, where, on the wall of the concrete cavity, a surface cooling system is designed. This system works on the principle of natural circulation of water and it takes the decay heat via air coolers to the atmosphere. In fact, this surface cooling system is designed to protect the vessel and concrete cavity structures more than the ceramic reactor core from being overheated by decay power.

No pressure-retaining and leak-tight containment is designed. The concrete cavities which house the reactor and the steam generator as well as other parts of the primary pressure boundary and which are called as “confinement”, together with the accident ventilation system, serve as the last barrier to the radioactivity release into the environment.

At the secondary circuit, a steam turbine cycle for electricity and heat co-generation is designed. A 3000kW standard superheated steam turbine-generator unit is used in the HTR-10. The steam generator produces the steam at temperature of 440 ^{o}C and the pressure of 4.0MPa to provide the steam turbine-generator unit, which can generate electricity of about 2500kW at the full power operation condition. The co-generation operation mode can produce electricity and supply hot water for district heating use.

1.3 Design, construction and licensing(1)(2)

During 1991-1992 INET completed a pre-feasibility study for the construction phase of the HTR-10 project and the necessary administrative procedures to initiate a project in China were completed. Permission for building the HTR-10 was granted by the State Council in March of 1992. After the building permit was issued INET compiled an Environmental Impact Report for the HTR-10, and submitted this report to the National Environmental Protection Administration in mid-1992. This Environmental Impact Report was reviewed by an environmental committee with expertise in the field of nuclear energy and this report was approved in December of 1992. Approval for the site of a nuclear reactor is granted only after the submission and examination of the “Environmental Impact Report” and the “Sitting and Seismic Report”; the latter report was also prepared and then submitted to the National Nuclear Safety Administration, and after an exhaustive examination of this report the reactor site in the northwest suburb of Beijing was approved in December of 1992.

Meanwhile the basic design was really started. To have a uniform standard for creating and reviewing the design, INET complied with the “Design Criteria for the HTR-10” and the “Standard Content and Format of the Safety Analysis Report of the HTR-10”, both of these standards having been approved by the National Nuclear Safety Administration (in September of 1992 and March of 1993, respectively). After completing the basic design of the HTR-10, INET asked Siemens/Interatom to review the design and this review was completed in August of 1994. This review was aided by the experience available in Germany from the construction and operation of their HTGRs and we also gained access to the knowledge that the Siemens/Interatom team acquired during their advanced design work on the HTR-Module. The many productive comments and suggestions of the Siemens/Interatom team were to be particularly valuable for the detailed design and construction of the HTR-10.

INET then cooperated with other academic institutes in China and began the detailed design of the HTR-10. The Architecture Institute of Tsinghua University was responsible for designing the building that would house the reactor, the Nuclear Power Institute of China created the design for the helium purification system, and China's Electric Power Technology Import & Export Corporation was responsible for the design of the Power Conversion Unit. The design of all of the remaining systems and their general implementation was accomplished by various teams at INET.

The administrative procedure necessary for licensing the HTR-10 was the same two-step procedure that is generally used for other nuclear reactors in China: (1) obtaining a construction permit, and (2) obtaining a commissioning permit. As required, INET compiled a "Preliminary Safety Analysis Report" to support the application for a construction permit and submitted this report to the National Nuclear Safety Administration in December of 1993. The work involved in licensing the HTR-10 continued for one year. After the National Environmental Protection Administration approved the Environmental Impact Report for the HTR-10 (for application for a construction permit) in September of 1994, the National Nuclear Safety Administration formally issued the construction permit in December of 1994.

The first tank of concrete for the reactor building was poured on 14 June 1995, the construction of the nuclear island building was finished in October of 1997, and the conventional island building was finished in 1999.

Main components of the HTR-10 include the reactor pressure vessel, the steam generator pressure vessel, the steam generator, the hot gas duct pressure vessel, the reactor internal structure, the control rod and the helium circulator. These components were all manufactured by various industries in Shanghai. Three pressure vessels and the steam generator were installed in the primary cavity of the HTR-10 in November of 1998. The installation of the metallic and ceramic reactor internal structure was completed in December of 1999; and the installation of the control rod system, the small absorber ball system, the fuel handling system, the helium circulator, and the power conversion unit were completed in May of 2000.

The "Final Safety Analysis Report" and the Environmental Impact Report (for application for a commissioning permit) were submitted to the National Nuclear Safety Administration and the National Environmental Protection Administration, respectively, in October of 1999. The Final Safety Analysis Report was approved and the commissioning permit was issued by the National Nuclear Safety Administration in November of 2000.

1.4 Commissioning[1]

Commissioning of the HTR-10 is divided into three phases, that is, Phase A, pre-operational tests for the components and systems; Phase B, first core loading, first criticality, zero power tests, hot tests of related components and systems, and low power tests; Phase C, raising power tests and full power operation test.

Pre-operation test was carried out after finishing the installation for each system or component. The testing for air leakage of the primary circuit was carried out in August of 2000 with the leakage rate being determined to be within the technical specifications. Preoperational testing of the primary circuit system and auxiliary systems (under cold conditions), a performance test of the power supply systems, and testing of the instrumentation and control systems were completed in October of 2000.

The initial core loading was started on 21 November 2000. For the initial core loading, graphite balls without nuclear fuel called the dummy balls are firstly placed into the discharge tube and the bottom cones region of the core. Then a mixture of fuel and dummy balls are loaded gradually to approach the first criticality. The percentages of fuel balls and dummy balls are envisaged to be 57% and 43% respectively. It last about 11 days to finish the loading and the HTR-10 reached the pre-criticality on 1 December of 2000. On 21 December 2000 the HTR-10 officially reached the first criticality.

After criticality of the HTR-10 relative zero power tests were and are being carried out. Up to now following tests were done: determination of the highest allowable operation power level for HTR-10 physical experiments; reactivity calibration of single ball; reactivity calibration of the small absorber ball system; reactivity calibration of the control rod, the measurement of the characteristic curve of the helium circulator at various revolution, pre-test of heat up for whole primary circuit by the helium circulator and dehumidifying graphite reflectors and carbon bricks using the helium circulator to heat them up to the temperature of 120 °C. The water of about 103kg was released from graphite reflectors and carbon bricks.

2. Fuel elements irradiation test

Fuel elements for the HTR-10 are being irradiated on Russian IVV-2M reactor to test thier irradiation behaviors. There are 4 fuel elements and some coated particles being irradiated. The irradiation test began on 3rd of July 2000. Up to 9th of April 2001 effective irradiation time reached 5976.5 hours and burnup reached 38,100-43,400MWd/t(U). Recently the report on burnup up to about 68000MWd/t(U) was received from Russian Institute. The result is still fine.

The irradiation temperature is 1000 ± 50 °C. The fuel element of Cap.3 was irradiated at the temperature of 1200 °C for 200 hours after its burnup reached 38,700 MWd/t(U) to simulate the accident condition of the HTR-10. The release of the fusion gas ^{88}Kr (R/B) was increased from (3-4) $X10^{-7}$ to $1X10^{-6}$, but the release of the fusion gas ^{88}Kr (R/B) was returned to original value when the irradiation temperature was reduced to 1200 °C. It means that there is no coated particle failure during the test.

3. Further development[3]

3.1 Rising power and synchronism

It is planned to raise power and synchronize by the end of this year. Before raising power some tests have to be carried out, including dehumidifying the graphite and carbon bricks; exhaust of the primary circuit; filling helium; cold criticality at helium atmosphere and relative calibration experiments; measurement of the reactivity temperature coefficient etc..

3.2. Safety experiments

The most important advantage of modular high temperature gas-cooled reactors is their inherent safety features. These safety features include the passive decay heat removal and reactor self-shutdown under accident conditions through a negative reactivity temperature coefficient. Engineered safety features are usually not necessary to protect the reactor when accident conditions occur. The unique safety performance of the modular HTGR is an important selection factor for the government, the public and the utilities when they consider this reactor technology. The HTR-10 test reactor can be used to perform safety demonstration experiments to verify the safety features and to promote better understanding of the features. After raising power and synchronization it is planned to make some safety experiments, if possible, to demonstrate HTGR inherent safety in a real nuclear island. Some of the safety demonstration experiments under consideration for the HTR-10 are:

- Loss of feed water supply
- Loss of helium flow
- Failure of primary isolation valve
- Loss of off-site power supply
- Failure of cavity cooling system
- Fault withdrawal of control rods
- Turbine trip
- ATWS
- Mass fuel elements test at high temperature

3.3. HTR-10 coupling with gas turbine cycle

Recent developments in gas turbine equipment, high-efficiency heat exchangers and electromagnetic bearings have enabled the development and design of reactor plants combining a safe modular gas-cooled reactor and a power conversion system based on the high-efficiency gas-turbine Brayton cycle. The components of a gas turbine cycle, such as the turbine, compressor, magnet bearings, and plate-and-fin recuperator could be found in other industries. The major challenge for a commercial direct helium gas turbine cycle for HTGR would be an integrated test with a real HTGR primary system environment. The questions concerning radioactivity deposition on the turbine blades, the influence of turbine shaft over-speed, as well as the system configuration, could be answered in the tests. HTR-10 will be a very attractive reactor to perform these tests.

Therefore after successful operation of the HTR-10 it is planned to install the gas turbine into the HTR-10 instead of or combined with existed steam turbine to realize the gas turbine cycle and get more experience on gas turbine cycle operation. The jointly conceptual design of the gas turbine cycle of the HTR-10 with OKBM, Russia is under way. Meanwhile Nissho Iwai, Fuji Electric Corporation and INET jointly did preliminary economic evaluation on the HTGR with the steam turbine cycle and will enhance their cooperation on the HTGR with the gas turbine cycle in the future.

3.4. HTR-PM project

For commercial development of the HTGR State Power Corporation and INET signed the agreement on doing pre-feasibility study on selecting HTGR system suitable for Chinese condition for future development, as the Modular HTGR has be recognized as one of candidates for forth generation

advanced energy system. State Planning and Development Commission also expressed their support on development of the HTGR in China. Therefore it is hoped to set up the prototype HTGR project around 2005 and start to construct it in 2005-2006. The prototype HTGR project will probably be a pebble bed type HTGR with gas turbine cycle and electric output of 100MW, it termed as HTR-PM.
The proposed HTR-PM has the following characteristics:

- Generates electricity at lower cost and with less environment impact. The design does not have sophisticated steam and water systems used in traditional fossil fuel plants and existing nuclear plants, but generates more energy than existing nuclear plants with the same fuel consumption and radioactive waste.
- High level of safety in comparison with existing and upgraded nuclear plants. The unique safety characteristics of modular high-temperature reactors allows them to be located near towns, chemical factories and petroleum refineries to reduce the consumption of fossil fuel so the fossil fuels can be used for other applications, which will contribute to future economy growth.
- Provides the ability to practically eliminate the use of fossil fuels to generate high-temperature heat required for petroleum refining and the chemical industry.

Development of the HTR-PM has already begun. The focus is on the power conversion unit. Three options, a steam turbine cycle, a direct helium gas turbine cycle and an indirect gas turbine cycle are under consideration. The major parameters being considered are:

(1) Core inlet temperature: There are several core inlet temperature options. The core inlet temperature would determine the steel used in the reactor structure. Temperatures in the range of 250-300 ^{o}C would allow use of conventional LWR material in the reactor pressure vessel which doesn’t need a further verification program. HTR-10 uses such materials. A higher core inlet temperature would require new materials for the reactor pressure vessel or a dedicated cooling system, which will influence plant efficiency and system configuration.

(2) Technology availability: Use of the steam turbine cycle in HTR-PM will use the available technology tested in the HTR-10 project. However, steam turbines having powers of 80-100 MWe have not been optimized for the plant efficiency due to the small blade height. Therefore 2-4 pebble-bed reactors will be connected to a steam turbine. This increases the system scale and complexity. The direct helium turbine will require an integrated test of the whole system, which would depend on the results of the HTR-10 second phase. The materials used in the HTR-10 may have to be changed which would require further verification. The use of an indirect gas turbine would relax the pressure requirements of the gas turbine systems. However, the IHX is expensive and complex. INET, together with the State Power Corporation, has formed a joint program to evaluate the above three options.

A preliminary time schedule, as indicated in the joint proposal to the Chinese government by INET and the State Power Corporation, for these activities is given in Table 2

Table 2. Future HTGR development schedule in China

Duration	Activities
~2000.12	HTR-10 first criticality
2001.1~2002.12	HTR-10 hot commissioning, power operation, design performance verification
2001.1~2003.12	HTR-10 safety demonstration experiments
2001.1~2005.12	HTR-10 gas turbine cycle realization
2001.1~2004.12	HTR-PM feasibility study
2004.1~2005.12	HTR-PM project preparation & licensing
2006.1~2009.12	HTR-PM demo-plant construction

4. Conclusion

The HTR-10 technical design represents the features of HTR-Module design. After five years construction, installation and pre-operation the HTR-10 reached the criticality in December 2000. Up to now all of results on zero point experiments and fuel elements irradiation test are fine. China will continue to develop the high temperature gas-cooled reactor in the future using the HTR-10 base.

Reference

1. Xu Yuanhui HTR-10 Commissioning and its Development The 6th Nuclear Energy Symposium, July 6-9 2001, Puspiptek, Serpong, Indonesia

2. Sun Yuliang, Xu Yuanhui, Licensing of the HTR-10 Test Reactor Workshop on "Safety and Licensing Aspects of Modular High Temperature Gas Reactors" July 3-5, 2000, Aix-en-Provence, France

3. Zuoyi Zhang, Suyuan Yu, Yuliang Sun, Future HTGR Developments in China after the Criticality of HTR-10 to be published in Nuclear Engineering and Design