Present Status and Future Plan of HTTR Project

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The High Temperature Gas-cooled Reactor (HTGR) is particularly attractive due to its capability of producing high temperature helium gas as well as its inherent safety characteristics. Hence, perspective of HTGR as possible future nuclear energy source was discussed in the review of “Long-term Program for Research, Development and Utilization of Nuclear Energy” by the Atomic Energy Commission of Japan, and the High Temperature Engineering Test Reactor (HTTR), which is the first HTGR in Japan, was successfully constructed at the Oarai Research Establishment of the Japan Atomic Energy Research Institute.

The HTTR attained the first criticality on November 10, 1998 and achieved the full power of 30MW and the reactor outlet coolant temperature of about 850 °C on December 7, 2001. The purpose of the HTTR project is to establish and upgrade HTGR technologies. It is widely recognized to the nuclear community that the timely and successful operation and tests of the HTTR are major milestones in development of the HTGR and high temperature nuclear process heat application. Extensive tests such as safety demonstration tests and irradiation tests are planned in the HTTR and a process heat application system will be coupled to the HTTR, where hydrogen will be produced directly from the nuclear energy.

KEYWORDS: HTGR, HTTR, Full power operation, Rise-to-power test, Safety demonstration test, Nuclear heat utilization, Hydrogen production

I. Introduction

The High Temperature Gas-cooled Reactor (HTGR), which is a graphite moderated, helium gas cooled reactor, is particularly attractive with its capability of producing high temperature helium gas as well as its inherent safety characteristics. The HTGR is also appealing as an option to efficiently burn weapons-grade plutonium for energy production. These interesting aspects make the HTGR worthy of further discussion on the future advanced reactors, along with advanced light water reactor (LWR). The HTGR is also expected to contribute to solving the current global environmental issue of CO₂ emission, since it can be alternative or supplemental to the fossil-fuel energy sources for process heat application. Under this understanding, perspective of HTGR as a possible future nuclear energy source was discussed in the review of “Long-term Program for Research, Development and Utilization of Nuclear Energy” by the Atomic Energy Commission of Japan, and the High Temperature Engineering Test Reactor (HTTR), which is the first HTGR in Japan, was successfully constructed at the Oarai Research Establishment of the Japan Atomic Energy Research Institute (JAERI). The commission recommended that a feasibility study should be done first to decide the commercial development for the gas turbine power generation HTGR, and the development study of high temperature heat application using HTGR should be continued to enhance the application field of the nuclear energy.

On the other hand, increasing interest has been given to the HTGRs in the world as represented by the South African PBMR project and US/Russian GT-MHR project. Under these circumstances, the HTGR development activity in Japan is becoming more active than before with the progress of the HTTR project. It is widely recognized to the nuclear community that the timely and successful operation and tests of the HTTR are major milestones in development of the HTGR and high temperature nuclear process heat application.

This paper gives an overview of the status of the HTTR project, typical rise-to-power test results and the future test plan using the HTTR.

II. HTTR Project

1. History in brief

The history of the HTTR development dates from 1960s. At that time, the possibility of direct steel making was sought by utilizing the heat from HTGR. Then, the Very High Temperature Reactor (VHTR) project with reactor outlet temperature of 1000 °C was initiated in 1969 at JAERI, including research and development (R&D) which covers all fields necessary for the reactor design and construction of the VHTR. However, there was no urgent or strong commercial demand coming up afterwards, although the essential needs of the HTGR were well understood for

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the future. Thus, the project was reviewed by the government to shift to more basic research for the future rather than immediate development of commercial reactors. In accordance with this review, the Atomic Energy Commission of Japan issued in 1987 the revision of Long-term Program for Research, Development and Utilization of Nuclear Energy, recommending that Japan should proceed with the development of more advanced technologies for the future, in parallel with existing nuclear systems. The Long-term Program emphasized that the HTGR is considered as one of the most promising nuclear reactors to improve the economy and enhance the application of nuclear energy. In conclusion, the construction of the HTTR was decided to establish and upgrade HTGR technologies as well as to be used as a tool for innovative basic research in the field of high temperature engineering. The construction of the HTTR was initiated in 1991, the rise-to-power test started in 1999 and the operation permit of the HTTR was issued on March 6, 2002 from the Ministry of Education, Culture, Sports, Science and Technology (MEXT). We have just started extensive tests using the HTTR. The maximum reactor outlet coolant temperature of 950 °C will be attained in near future.

2. Outline of HTTR Design

The major specification of the HTTR is summarized in Table 1. The reactor core is designed to generate 30 MW of thermal power and consists of array of hexagonal graphite fuel assemblies so-called “pin-in-block type fuel”, control rods, graphite reflectors etc.

<table>
<thead>
<tr>
<th>Table 1 Main specification of HTTR</th>
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<tbody>
<tr>
<td>Thermal power</td>
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<tr>
<td>Outlet coolant temperature</td>
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<tr>
<td>Inlet coolant temperature</td>
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<tr>
<td>Fuel type</td>
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<tr>
<td>Fuel element type</td>
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<tr>
<td>Direction of coolant flow</td>
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<tr>
<td>Pressure vessel</td>
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<tr>
<td>Number of cooling loop</td>
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<tr>
<td>Heat removal</td>
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<tr>
<td>Primary coolant pressure</td>
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<td>Containment type</td>
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(1) Reactor

A fuel assembly consists of fuel blocks and a hexagonal graphite block, 360mm across flats and 580mm in height, as shown in Fig.1. The fuel assembly has three dowel pins on the top and three mating sockets at the bottom to align the fuel assemblies. Tri-isotropic (TRISO)-coated fuel particles with UO₂ kernel, about 6wt% average enrichment and 600 μm in diameter, are dispersed in the graphite matrix and sintered to form a fuel compact. Fuel compacts are contained in a fuel rod, 34mm in outer diameter and 577mm in length. Fuel rods are inserted into vertical holes in the graphite block. Helium gas coolant flows through gaps between the holes and the rods.

In the safety design of the HTGR fuels, it is important to retain fission products within particles so that their release to the primary coolant may not exceed an acceptance level. From this point of view, the basic design criteria for the fuel are to minimize the failure fraction of as-fabricated fuel coating layers and to avoid significant additional fuel failures during operation. To meet the latter criteria for the first loading fuel, the fuel temperature is limited to below 1495 °C under operating conditions and below 1600 °C under abnormal transient conditions.

The arrangement of core components and core internals is shown in Fig.2. One column is the row of prismatic hexagonal blocks piled up axially. The active core, which is 2.9m in height and 2.3m in diameter, consists of 30 fuel columns and 7 control rod (CR) guide columns and is surrounded by 12 replaceable reflector columns, 9 reflector region CR guide columns and 3 irradiation test columns. The core internals consist of graphite and metallic core support structures and shielding blocks. They support and arrange the core components, such as fuel assembly and replaceable reflector blocks, within the reactor pressure vessel (RPV). The graphite core support structure consists of permanent reflector blocks, hot plenum blocks, support posts, core bottom structures and so on. The metallic core support structures consist of support plates, a support grid, core restraint mechanism and so on.

Reactivity is controlled by CRs as shown in Fig.2. The CRs are individually supported by control rod drive mechanism located in stand pipes connected to the hemispherical top head of the RPV. Reactor shutdown is made at first by inserting 9 pairs of CRs into the reflector region, and then by inserting the other 7 pairs of the CRs into the active core region after the temperature is reduced, so that the CRs should not exceed their design temperature limit.
(2) Reactor Cooling System

The reactor cooling system is composed of a single main cooling system (MCS), an auxiliary cooling system (ACS) and a vessel cooling system (VCS) as schematically shown in Fig. 3.

The MCS is operated in normal operation to remove heat from the core and send it to the environment via intermediate heat exchanger (IHX) of 10 MW and primary pressurized water cooler (PPWC) of 20 MW in parallel, called parallel loaded operation, or via only the PPWC of 30 MW, called single loaded operation. The primary coolant gas comes into the reactor pressure vessel (RPV) at 395 °C, and heated up to 850 °C at the reactor outlet at the rated operation mode and 950°C at high temperature test operation mode.

The ACS is designed as an engineered safety feature to operate upon a reactor scram and cool down the core and the core support structure. On the other hand, the VCS cools the biological shield of concrete in normal operation and acts as a cooling system upon postulated accidents such as loss of forced convection cooling, e.g. pipe rupture of primary cooling circuit. The decay and residual heat is removed by the heat transfer (largely by radiation) from the RPV to the cooling panel of the VCS.

(3) Instrumentation and Control System

Instrumentation and control system consists of the instrumentation, control, safety protection systems as well as a control room. Instrumentation system consists of nuclear instrumentation, in-core temperature monitoring and fuel failure detection reactor systems.

The nuclear instrumentation system of the HTTR is composed of the wide range monitoring system (WRMS) and the power range monitoring system (PRMS) as shown in Fig. 4. The WRMS is used to measure the neutron flux from 10⁻⁸% to 30% of the rated power. Three fission chambers are installed in the permanent reflector blocks through the stand-pipe. The PRMS is used to measure the neutron flux from 0.1% to 120% of the rated power. The PRMS is also used as the sensor for the reactor power control system. It is difficult to monitor the reactor core whole because the temperature and neutron flux level in the RPV become very high in full power operation. Accordingly the detector of the PRMS is required to be located outside the RPV.

In order to monitor the core outlet temperature of the primary coolant, seven thermocouples (T/Cs) are arranged in the hot plenum blocks below the reactor core as shown in Fig. 4. The N-type T/C (Nicrosil-Nisil) is selected because...
the temperature of the primary coolant in the hot plenum reaches about 1000 ℃ at the rated power operation.

The fuel failure detection system is composed of precipitators, a pre-amp, and a control box and so on. The precipitator is used in order to detect β rays radiated from the short-lived gaseous fission products such as Kr-88 and Xe-138 from the failed fuel particles.

The reactor control system is designed to assure high stability and reasonably damped characteristics against the various disturbances during the operation. The main control system of the HTTR consists of the operational mode selector, the reactor power control and plant control systems. The operational mode selector is designed to select several mode operations such as the rated power operation, the high temperature test operation, the safety demonstration test operation and so on. The reactor power control system consists of the power control and reactor outlet coolant temperature control devices. The reactor outlet coolant temperature control device is an upper one to give demand of reactor power to the power control device.

The safety protection system consists of the reactor protection and engineered safety features actuating systems. It is designed with 2-out-of-3 circuit logic 2-trains. The reactor protection system of the HTTR automatically initiates a reactor scram by inserting the CRs. The engineered safety features actuating system of the HTTR is designed to arrest the release of the fission product and to ensure the integrity of the core, the reactor coolant pressure vessel boundary against unexpected conditions during abnormal operational transients and accidents.

3. Rise-to-power Test

(1) Outline

For safe and steady execution, the rise-to-power test was conducted step by step, that is, it was divided into three phases of the power levels of 10, 20 and 30 MW.  

Test items of the rise-to-power test can be categorized to tests for commissioning and for evaluating performance of the HTTR. The former test items are for example measurement of CR reactivity worth, performance at abnormal transient (loss of off-site electric power test), radiation shielding performance, and measurement of radioactive material concentration in reactor building. The latter test items include performance of reactor control system, calibration of nuclear instrumentation system (NIS) to thermal power, performance of heat exchangers in the MCS, thermal expansion of high temperature components, measurement of impurity in main cooling system, behavior of fuel and fission product and so on. Major test items are shown in Table 2.

In the first phase test to 10 MW, core physics, radiation shielding performance and performance of reactor control system was mainly investigated.

Major objective of the second phase test to 20 MW was to investigate thermal properties such as performance of heat exchangers, thermal expansion of high temperature components, and thermal-hydraulics in the reactor core.

<table>
<thead>
<tr>
<th>Table 2 Major items of rise-to-power test</th>
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<tr>
<td>Test item</td>
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<tr>
<td>Commissioning test</td>
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<tr>
<td>Control rod reactivity worth measurement</td>
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<tr>
<td>Loss of off site power test</td>
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<tr>
<td>Coolant pressure/temperature measurement</td>
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<tr>
<td>Radiation shielding performance</td>
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<tr>
<td>Radioactive material concentration in reactor building</td>
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<tr>
<td>Reactor &amp; plant performance test</td>
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<tr>
<td>Reactor control system performance</td>
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<tr>
<td>Calibration of NIS to thermal power</td>
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<tr>
<td>Thermal hydraulics in the core</td>
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<tr>
<td>Performance of heat exchangers in MCS</td>
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<tr>
<td>Performance of vessel cooling system</td>
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<tr>
<td>Behavior of fuel and fission product</td>
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<tr>
<td>Measurement of impurity in MCS</td>
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<tr>
<td>Thermal expansion of high temp. components</td>
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NIS : Nuclear Instrumentation system.
MCS : Main cooling system.

The third phase test to 30MW was performed to confirm the overall function of the HTTR and to get the operation permit by undergoing the commissioning tests by MEXT. The rise-to-power test at the rated operation mode was successfully completed in March 2002. The following gives typical test results of the rise-to-power test.

(2) Test Result

The reactor control system, pressurized water temperature control system, reactor inlet temperature control system and reactor power control system were adjusted at the thermal power of 9 MW by giving some perturbations to the controllers. It was confirmed that stable control of the abovementioned control system is possible within the range of control constants (P, I values) predicted in pre-test calculations. Figure 5 shows a test result of the reactor power control system.

Figure 6 shows the thermal power and reactor outlet coolant temperature up to full power. The rated power and 850 ℃ was reached on December 7, 2001.

Using the test results to 30MW, we evaluated overall heat transfer coefficients of the PPWC and IHX, which were about 8% and 6% lower than the design center values. However, they are within the predicted range of variance, and so we consider that the MCS has enough capacity to remove the power of 30MW.

High temperature components such as the IHX, PPWC, etc are floatingly supported by hangers and oil snubbers, that is displacement of the heat exchangers by thermal expansion of themselves and concentric hot gas duct, which connect them with the RPV, is not restricted rigidly. To confirm performance of this floating support system, as shown in Fig.7 displacement of the heat exchangers and concentric hot gas duct was measured at the thermal power levels of 9MW, 20MW and 30MW, and compared with analytical results. Figure 8 shows typical test result as comparison with analytical result. The analytical results agreed well with the measure values.
PRMS : Power Range Monitoring System.

Fig.5 Characteristics of reactor control system

Fig.6 Reactor operational trend up to full power
Fuel and fission product behavior of the HTTR at several thermal power levels up to 30 MW was evaluated based on the measured data by fuel failure detection system and primary coolant sampling system. Figure 9 shows primary coolant activity of Kr-88 vs. reactor thermal power. The concentration was less than $5 \times 10^{-2}$ Bq/cm$^3$, which is much smaller than the alarm level of $10^4$ Bq/cm$^3$. In the fabrication of the first-loading fuel of the HTTR, as-fabricated fuel compacts contained almost no wall-through failed particles and the average wall-through failure fraction was as low as $8 \times 10^{-5}$.  

4. Future Plan of the HTTR

The HTTR project is the overall national project for HTGR technologies and its application. This section introduces the future plan of the HTTR project. Figure 10 shows the total development schedule using the HTTR.  

(1) Evaluation of Reactor Performance

Based on the HTTR operational data, the HTGR reactor performance is to be evaluated and analytical computer codes will be verified or modified for predicting realistic reactor performance under steady state and operational transient conditions. The evaluation is focused on: (a) Core physics in relation with thermal response and control system, (b) Thermal analysis for fuel, reactor internals and high temperature components, (c) Fuel performance on fission product release and degradation of the coating layers to contain the fission products, (d) Structural integrity of reactor internals and high temperature components, (e) Decay heat and residual heat removal characteristics, and so forth. The fruits from the HTTR operational data and their evaluation are expected to be utilized for the commercial HTGR designs underway in South Africa, Russia, United States, etc. as well as for the design of the future Japanese advanced HTGR.
Fig.10  Total development schedule using the HTTR\(^4\)

(2) Safety Demonstration Test\(^5\)

It is well known that the HTGR has inherent safety features, characterized by no risk of reactor core meltdown even in the case of no forced cooling systems functioned with failure of reactor shutdown. It is of great importance and one of the best ways for the wide public acceptance to demonstrate such inherent safety of the HTGR using an actual HTGR. It is, therefore, planned in the HTTR to conduct a safety demonstration test. The safety demonstration test is divided into two phases.

The first phase test, which simulates the transient or accident events, includes primary coolant flow reduction test and a control rod withdrawal test at power operation. In the primary coolant flow reduction test, coolant flow rate is reduced by running down one or two gas circulators out of three as shown in Fig.11. In the test, scram set values of primary flow rate, core differential pressure and reactor outlet coolant temperature are altered so that the reactor scram should not occur. In the reactivity insertion test, the central pair of control rods in the core is withdrawn with the reactor power control system being disabled as shown in Fig.12. This test demonstrates that rapid increase of reactor power by withdrawing the control rods is restrained by only the negative reactivity feedback of the core without operating the reactor power control system.

On the contrary, the second phase test simulating the severe accident conditions will be conducted after completion of the first phase test. The second phase test contains loss of forced cooling test, vessel cooling system stop test and so on.

We have now performing the first phase test.
(3) Development of Process Heat Application System

To enhance the nuclear energy application to heat process industries, JAERI has continued extensive efforts for development of hydrogen production systems using the nuclear heat from HTGR. In the hydrogen production system by steam reforming of natural gas mentioned above, the emission of CO$_2$ is unavoidable, because natural gas of methane is used as feed gas. Therefore, the final goal of the hydrogen production system using HTGR is to produce hydrogen from water without emission of CO$_2$. For this purpose, the thermochemical iodine-sulfur (IS) process is under development in a small-scale laboratory experiment in parallel. Figure 13 shows basic scheme of IS process. In the experiment, a closed-cycle continuous operation in a steady state for 48 hours was successfully achieved at JAERI, and then the development activity was shifted to engineering system development using a large-scale facility.

A process heat application system will be coupled to the HTTR, where hydrogen will be produced directly from the nuclear energy.

Fig.13 Basic scheme of IS process

- Concluding Remarks

The HTGR is particularly attractive due to its capability of producing high temperature helium gas as well as its inherent safety characteristics. The HTGR is also expected to contribute to solving the current global environmental issue of CO$_2$ emission, since it can be alternative or supplemental to the fossil-fuel energy sources for process heat application. Under this understanding the HTTR project is ongoing for the HTGR development.

A further effort is to be continued at JAERI. Finally, it should be emphasized that support and understanding from people involved in the nuclear development are needed and wished for the success of the HTTR project.

Acknowledgement

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References
   (in Japanese)