Present status of HTTR and its Operational Experience

**Key words:** HTGR; HTTR; Outlet coolant temperature; Full power operation; Safety demonstration test; Process heat application; Hydrogen production

**ABSTRACT**

A High Temperature Gas-cooled Reactor (HTGR) is particularly attractive because of its capability of producing high temperature helium gas and its inherent safety characteristics. Hence, the High Temperature Engineering Test Reactor (HTTR) was successfully constructed at the Oarai Research Establishment of the Japan Atomic Energy Agency. The HTTR achieved the full power of 30MW and reactor outlet coolant temperature of about 850ºC on December 7, 2001. After several operation cycles, the HTTR achieved the reactor outlet coolant temperature of 950ºC on April 19, 2004. It is the highest coolant temperature outside reactor pressure vessel in the world. This is one of the major milestones in HTGR development of high temperature nuclear process heat application. Extensive 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. This paper gives an overview of the HTTR Project focusing on the latest results from the HTTR test and the future test plan using the HTTR.

**1. INTRODUCTION**

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. HTGR is also appealing as an option to efficiently burn weapons-grade plutonium for energy production. These interesting aspects make HTGR worthy of further discussion on the future advanced reactors, along with Advanced Light Water Reactor (ALWR). 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 "Framework for Nuclear Energy" issued by the Japan Atomic Energy Commission, it is stated that HTGR can be a high temperature heat source for power generation with excellent efficiency and hydrogen production. It is important to continue to promote research and development.

On the other hand, increasing interest has been given to HTGRs in the world as represented by the Very High Temperature Reactor (VHTR) in Gen-IV concept, 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 HTGR and high temperature nuclear process heat application.

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

2. HTTR Project

2.1 History in Brief

The history of the HTTR development dates from 1960s. At that time, the possibility of direct steel manufacturing was sought by utilizing the heat from HTGR. Then, VHTR project with reactor outlet temperature of 1000°C was initiated in 1969 at the Japan Atomic Energy Research Institute (JAERI, the predecessor of the present JAEA), including research and development (R&D) which covers all fields necessary for the reactor design and construction of VHTR. However, there was no urgent or strong commercial demand coming up afterwards, although the essential needs of HTGR were well understood for 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 Japan Atomic Energy Commission 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 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 power-up test started in 1999 and the operation licensing of the HTTR was issued on March 6, 2002 from the Ministry of Education, Culture, Sports, Science and Technology (MEXT). After several operation cycles, we started the power-up test to achieve 950°C of outlet coolant temperature in 2004. The maximum reactor outlet coolant temperature of 950°C has been attained in April 2004.

2.2. Outline of HTTR Design

![Cooling system of the HTTR](https://example.com/httr_diagram.png)

**Fig. 1** Cooling system of the HTTR
The major specification of the HTTR is summarized in Table 1.

<table>
<thead>
<tr>
<th>Specification</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power</td>
<td>30MW</td>
</tr>
<tr>
<td>Outlet coolant temperature</td>
<td>850°C/950°C</td>
</tr>
<tr>
<td>Inlet coolant temperature</td>
<td>395°C</td>
</tr>
<tr>
<td>Fuel</td>
<td>Low enriched UO₂</td>
</tr>
<tr>
<td>Fuel element type</td>
<td>Prismatic block</td>
</tr>
<tr>
<td>Direction of coolant flow</td>
<td>Downward</td>
</tr>
<tr>
<td>Pressure vessel</td>
<td>Steel</td>
</tr>
<tr>
<td>Number of cooling loop</td>
<td>1</td>
</tr>
<tr>
<td>Heat removal</td>
<td>IHX and PWC (parallel loaded)</td>
</tr>
<tr>
<td>Primary coolant pressure</td>
<td>4MPa</td>
</tr>
<tr>
<td>Containment type</td>
<td>Steel containment</td>
</tr>
</tbody>
</table>

The reactor cooling system is composed of a main cooling system (MCS), an auxiliary cooling system (ACS) and a vessel cooling system (VCS) as schematically shown in Fig. 1. 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 10MW and primary pressurized water cooler (PPWC) of 20MW in parallel, called parallel loaded operation, or via only the PPWC of 30MW, 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 depressurization accident, 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.

The reactor core is designed to generate 30MW of thermal power and consists of array of hexagonal graphite fuel assemblies so-called “pin-in-block type fuel”, control rods, graphite reflectors etc. The core is supported by a graphite support structure and is tightened by a core restraint mechanism. The vertical cut-away view of the HTTR reactor pressure vessel (RPV) is shown in Fig.2. (Saito 1994)

Fig.2 Vertical cut-away view of the HTTR RPV.

2.3 Power-up Test and achievement of 950°C

(1) Power-up test at rated operation mode
For safe and steady execution, the power-up test was conducted step by step. It was divided into three phases of the power levels of 10, 20 and 30MW at the rated operation mode. Test items of the power-up test can be categorized to tests for commissioning and for evaluating performance of the HTTR. The former test items are measurement of control rod reactivity worth, performance at abnormal transient (loss of off-site electric power test), radiation shielding performance, measurement of radioactive material concentration in reactor building and so on. 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, thermal-hydraulics in reactor core, measurement of impurity in main cooling system, behavior of fuel and fission product and so on. The rated power and 850°C was reached on December 7, 2001 in single loaded operation. The power-up test at the rated operation mode was successfully completed in March 2002.
(2) Power-up test at high temperature test operation mode and achievement of 950°C
After the power-up test at the rated operation mode and several operation cycles, the power up test at the high temperature operation mode was started in March 2004. In the test, measured data were compared to the data of the rated operation mode to confirm the characteristics of the reactor at the high temperature test operation mode. The reactor power, reactor inlet and outlet coolant temperature at the single loaded operation which achieved 950°C is shown in Fig. 3. The reactor power was raised step by step. The characteristics of the core, e.g. thermal-hydraulics in reactor core, performance of heat exchangers, etc. are compared to the data at the rated operation mode in each power step. The full power and 950°C was achieved on April 19, 2004. In June 2004, the power-up test in parallel loaded operation was conducted. The power-up test at the high temperature test operation was successfully completed.

(3) Fuel performance
Fuel and fission product behavior of the HTTR at several power levels up to 30MW was evaluated based on the measured data by a fuel failure detection system and a primary coolant sampling system. Figure 4 shows the release to birth ratio (R/B) of $^{88}$Kr vs. reactor power. The R/B below 50% of the reactor power shows similar results to the data at the rated operation mode. The R/B above 50% of the reactor power increase exponentially to $1.0 \times 10^8$ at the full power operation, which was slightly larger than the results at the rated operation mode of $7 \times 10^{-9}$ (Ueta 2003).
At the high temperature test operation mode, the fuel temperature becomes higher than the rated operation mode. It is considered that the higher fuel temperature shows higher R/B. It suggests that the R/B was within the release level by diffusion of the generated fission gas from the contaminated uranium in the fuel compact matrix, and no significant failure was occurred during power-up test with high temperature test operation mode. Finally, the R/B at full power resulted three orders lower than the limitation of $5.35 \times 10^{-4}$, which corresponds to 1% fuel failure. 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 $8 \times 10^{-5}$ (Sawa 1999).

(4) Core performance
1) Nuclear characteristics
During the power-up tests, many kinds of nuclear characteristics, such as control rod worth, isothermal temperature coefficients, neutron flux distribution and so on, were measured.
Isothermal temperature coefficients and power coefficients are also measured during power up tests. Isothermal coefficients were measured by the control rod position change when core inlet temperature was increased. The measured isothermal temperature shows good agreement with calculated value. Power coefficients were evaluated by the change in control rod position at each power level and control rod worth curve. Measured power coefficients are negative in all power range. (Nojiri 2004)

The power distribution in the core should be optimized and the power distribution should be kept during burnup to achieve fuel temperature low. Therefore, change in excess reactivity with burnup should be kept small using burnable absorber. The change in excess reactivity was evaluated by the control rod position at criticality. The excess reactivity is evaluated by the control rod position at zero power criticality and control rod worth curve. Evaluated excess reactivity change is shown in Fig. 5. The evaluated excess reactivity shows good agreement with design results. It shows that burnable absorber could compensate decrease in reactivity due to burnup of fuel.

2) Fuel temperature
For the HTTR, fuel temperature is the most important parameter to ensure the integrity of the fuel. However, the fuel temperature could not be measured directly. The fuel temperature is evaluated by the measured thermal power, primary coolant flow rate, coolant temperatures. In the design, fuel temperature was evaluated considering many uncertainties such as error of reactor power, core flow rate and so on. Some of uncertainties became clear through power-up tests. The maximum fuel temperature at the high temperature test operation was evaluated by the measured results and revised uncertainties. Evaluated maximum fuel temperature was 1435°C which is lower than the limit of 1495°C. (Tochio 2006)

(5) Coolant chemistry
Chemistry control is important for the helium coolant because impurities caused oxidation of the graphite used in the core and corrosion of high-temperature materials used in the heat exchanger and so on. The coolant chemistry was monitored by the helium sampling system continuously between the reactor start-up and shutdown. The actual chemical impurities of CO, H₂, CO₂, CH₄, O₂ water vapor etc. were removed by the helium purification system the
concentration of each impurity was extremely limited by the operating procedure during the operation. Water vapor, carbon monoxide and carbon dioxide concentration at the reactor inlet during the high temperature test operation in single loaded operation are shown in Fig. 6. At the beginning of the operation, there are very few CO and CO$_2$ in the primary circuit. Water vapor is removed by the helium purification system with reactor operation. In the operation below 80% of reactor power, impurities did not increased rapidly. However, over 80% of reactor power CO and CO$_2$ are increased rapidly. Two reasons are considered. One is the impurity emission from the graphite material used in the core and as an insulator in the concentric hot gas duct. Over 80%, reactor outlet temperature became higher than 850°C. It is the first time to be above 850°C. Therefore, impurities are emitted from graphite and insulator. The other is the chemical equilibrium in the core. The water vapor which was emitted from the graphite converted to H$_2$ and CO by an immediate reaction in the high-temperature conditions in the core. Therefore, the behaviour of H$_2$ and CO were very similar to that of water vapor especially after the power up from 70%. (Fujikawa 2004)

2.4 Maintenance experience

The HTTR has been operated about eight years since the first criticality achieved. Many maintenance works have been carried out so far. (Tochio 2007)

(1) Filter exchange of primary gas circulator

There are four Primary Gas Circulators (PGCs) in the primary helium circuit, three are arranged around the PPWC and the rest is arranged around the IHX. The PGC is a centrifugal, dynamic gas bearing type. The gas circulator consists of the electric stator and rotor assembly, the internal structure, the bearing, the impeller and the filter.

The filter unit, which is on the top of the circulator, protects the impeller and rotating shaft from dust, and it is made of sintered metal SUS316. The differential pressure rise at the filters in the PGCs at the PPWC was observed. It was observed from 1999 and reached to 26.4kPa. This value corresponded to 33.7kPa for rated power, which is over the alarm level of 30kPa.

The filter exchange was carried out in a temporary greenhouse build on the filter. The used filters were picked-up in the storage tank to move out the containment vessel. The picked-up filter surface was observed in order to investigate the cause of differential pressure rise. The foreign adhesion material was the thermal insulator and the carbon. This foreign material was monitored to identify cobalt-60 ($^{60}$Co) and Antimon ($^{124}$Sb) that were corrosion products. The evaluated
personal exposure was about 0.42mSv less than the planned exposure about 1.8mSv.

The filter exchange was completed successfully without any contamination. The differential pressure after the filter exchange was measured as 2.5kPa at full-power of 30MW, which was much lower than the predicted one.

(2) Maintenance for reserved shutdown system
The annual inspection of the HTTR had progressed from Jul. 26, 2004 to Mar. 4, 2005. During the for reserved shutdown system (RSS) movement inspection before reactor start-up, it had found that the position lamp of one of the reserved shutdown system (RSS) did not actuate at Feb. 21, 2005. The RSS is a backup reactor scram system and the HTTR has 16 sets of the RSS. Suddenly, the RSS actuated with smoothly at Feb. 27, 2005. By the comparison between the RSS and another normal RSS, the causes were expected that a brake of the driving motor of the RSS had some trouble.

Through the investigations, it was cleared that the distortion of oil seal in the driving motor caused the oil intrusion in the break which is located below the driving motor. The mixture of intruded oil and abrasion power adhered the break part each other. The following corrective actions were taken.

i) The motor of the troubled RSS was changed with new one.

ii) It was decided to perform the RSS actuation inspection every month, which include measurement of both of the brake current changing time and the continuous time of the motor start-up current.

iii) All of the motors used in the RSS shall be changed

(3) Neutron detector replacement
The three developed neutron sensor for the Wide Range Monitoring (WRM) system had been installed inside the pressure vessel of the HTTR, the WRM environment temperature is 450°C under normal operation and 550°C under an accident The WRM is designed to satisfy the requirement for the safety protection against an accident during start-up and for post-accident monitor. The detector of the WRM is a fission chamber.

On Sep. 14, 2005, one of the WRMs showed the count rate zero, which continued for 2 days. After that, the count rate recovered and it showed 20cps. During 2 days, it cleared that the pulse signal from the chamber was not observed and measured capacitance showed a small deviation compared to that in an annual inspection. The cause is expected on the detector.

The cause reason was cleared as the followings:
The lead wire/MI-cable and an inner chamber/signal lead wire were welded with a DC welding method. The lead wire was melted and the diameter of the lead wire had become small because the welding time of this method was long. Using pulse-welding method, the welding time was short and the diameter was not changed.

Corrective actions were taken as follows:
The welding method was changed from a DC welding to a pulse welding. The strength of the welding signal lead was not so much changed from no-welding one.

2.5 Future Plan

The HTGR technologies and its application development schedule using the HTTR is shown in Table 2.

(1) Long-term high temperature test operation
For commercialization of HTGRs, it is necessary to demonstrate stable operation in long term with outlet coolant temperature of 950°C. However, the outlet coolant temperature of 950°C had been achieved only about a week in the HTTR. To carry out the safety demonstration tests, the HTTR should start and stop in a short term. It is necessary to carry out long-term operation to obtain the HTTR performance data. So far, operational period is not enough to obtain saturated data such as leak rate of the primary cooling system, leak rate of the containment vessel, temperature of concrete structure such as biological shielding and so on.

The long-term high temperature test operation is planned in 2009. In the test, 50 day operation with outlet coolant temperature of 950°C is planned. In advance the test, 30day long-term operation with outlet coolant temperature of 850°C is also planned in 2007.

(2) Evaluation of Reactor Performance
Based on the HTTR operational data, the HTGR reactor performance is to be evaluated and computer codes for analyses will be verified or modified for predicting realistic reactor performance under steady state and operational transient conditions. The evaluation is focused on: (a) Reactor 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.

These data will be summarized and compiled for database.

(3) Safety Demonstration Test
To demonstrate inherent safety features of HTGR
using an actual HTGR, it is planned in the HTTR to conduct a safety demonstration test. The safety demonstration test is divided into two phases. The first phase tests, which simulate the transient or accident events, include a control rod (CR) withdrawal test, in which the central pair of CRs is withdrawn, and a primary coolant flow reduction test, in which one or two out of three gas circulators are run down with position of CRs kept unchanged, at power operation. So far, the CR withdrawal tests at 30%, 50%, 60% and 80% of the power have been carried out. Also the primary coolant flow reduction tests at 30%, 60%, 80% and 100% of the power have been carried out.

In the second phase, more severe tests are planned such as all loss of forced cooling test by trip of all the three gas circulator, all-blackout test including stopping the vessel cooling system, etc. These tests will be conducted after completion of the first phase tests. It will be carried out in 2007 and later.

(3) Development of Process Heat Application System
To enhance the nuclear energy application to heat process industries, JAEA has continued extensive efforts for development of hydrogen production systems using the nuclear heat from HTGR.

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 at JAEA. In the experiment, a closed-cycle continuous operation in a steady state for one week was successfully achieved, and then the development activity was shifted to engineering system development using a large-scale facility.

4. Concluding Remarks
The HTGR has salient features concerning reactor safety as well as supply of high temperature heat. Under this understanding the HTTR have been achieved the outlet coolant temperature of 950°C and the HTTR project is ongoing for HTGR development. Global eyes are kept by not only nuclear people of interest but also the public upon the HTTR project, since its successful achievement of 950°C may enhance the possibility to solve the environmental issues of CO$_2$ emission as well as a possible energy crisis which might happen in the future. A further effort is to be continued at JAEA. 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.

References

Table 2 Operation schedule of the HTTR

<table>
<thead>
<tr>
<th>Year</th>
<th>2006</th>
<th>2007</th>
<th>2008</th>
<th>2009</th>
<th>2010</th>
</tr>
</thead>
<tbody>
<tr>
<td>Safety demonstration tests</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>850°C 30days operation</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>950°C 50days operation</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Evaluation of Reactor Performance</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Development of IS Process</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Copyright © 2007 by JSME