Safety Demonstration Tests using High Temperature Engineering Test Reactor (HTTR)

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Safety demonstration tests using the High Temperature Engineering Test Reactor (HTTR) are conducted for the purpose of demonstrating inherent safety features of High Temperature Gas-cooled Reactors (HTGRs) quantitatively as well as providing the core and plant transient data for validation of HTGR analysis codes for safety evaluation. The safety demonstration tests are divided to the first phase and second phase tests. In the first phase tests, simulation tests of anticipated operational occurrences and anticipated transients without scram (ATWS) are conducted. The second phase tests will simulate accidents such as a depressurization accident (loss of coolant accident). The first phase tests simulating reactivity insertion events and coolant flow reduction events started in FY 2002. Pre-test analysis of the first phase tests was conducted using the core and plant dynamics analysis code, ACCORD and Monte Carlo code, MVP. The pre-test analysis results agreed fairly well with the test results of a reactivity insertion test, which is a control rods withdrawal test, and a coolant flow reduction test, which is a gas circulator trip test, at power levels of 50% and 30% of the rated power, respectively. The first phase safety demonstration tests will continue until FY 2005, and the second phase tests are planned to start in FY 2006.

KEYWORDS: HTTR, HTGR, safety demonstration test, reactivity insertion test, coolant flow reduction test

I. Introduction

The High Temperature Engineering Test Reactor (HTTR), which is the first High Temperature Gas-cooled Reactor (HTGR) in Japan with the thermal power of 30 MW and the reactor outlet coolant temperature of 950 °C maximum, was constructed at the Oarai Research Establishment of the Japan Atomic Energy Research Institute (JAERI) for the purpose of establishing and upgrading technologies of HTGRs as well as nuclear heat utilization.1)

The HTTR attained the first criticality on November 10, 1998. The rise to power test of the HTTR started in September 1999 and the HTTR reached the full power of 30 MW and the reactor outlet coolant temperature of 850 °C on December 7, 2001. Then, on March 6, 2002 JAERI received a certificate of the pre-operation test from the government, that is, an operation permit of the HTTR at the rated operation mode (operation at the reactor outlet coolant temperature of 850 °C), completing the rise to power test at rated operation mode. The HTTR will accomplish the reactor outlet coolant temperature of 950 °C at high temperature test operation mode in FY 2003 after accumulating operation experiences.

Various tests utilizing the HTTR has been started. The HTTR tests can be categorized to four subjects:

- Safety demonstration test,
- Basic technological test, and
- Nuclear heat utilization test

In the safety demonstration test, inherent safety features of the HTGR are demonstrated. In the basic technological test, operational performance of the HTGR is to be quantified through operation and maintenance experiences of the HTTR. Nuclear heat utilization technology will be demonstrated by connecting a hydrogen production system to the secondary helium cooling system of the HTTR.

This paper describes an outline of the safety demonstration test plan as well as the detailed test methods and conditions of the first phase tests:

- Reactivity insertion test and
- Coolant flow reduction test,
which started in FY 2002. In addition, pre-test analysis results and test results obtained so far are shown also.

II. Outline of HTTR design

The cutaway and cross section views of the reactor are shown in Fig. 1. The reactor consists of a reactor pressure vessel, fuel elements, replaceable and permanent reflector blocks, core support structures, control rods, etc. Thirty columns of fuel elements and seven columns of control rod guide blocks form the reactor core called fuel region, which is surrounded by replaceable reflector blocks and large-scale permanent reflector blocks. The fuel element of the HTTR is a so-called pin-in-block type. Enrichment of U-235 is 3–10 (Average 6) wt%.

Sixteen pairs of control rods in the fuel and replaceable reflector regions of the core control reactivity of the HTTR. A control rod drive mechanism drives each pair of control rods using an AC motor. At a reactor scram, electromagnetic
clutches of the control rod drive mechanisms are separated and the control rods fall into holes of the control rod guide blocks by gravity at a constant speed, shutting down the reactor safely. In an unlikely event that the control rods insertion fails, reserved shutdown pellets made of B4C/C are dropped into the core.

As shown in Fig. 2, the cooling system of the HTTR consists of a main cooling system (MCS), which operates at normal operation, and an auxiliary cooling system (ACS) and a vessel cooling system (VCS), which operate to remove residual heat of the core after a reactor scram. The ACS and the VCS are engineered safety features. In commercialization of economically competitive HTGR, it is necessary to eliminate the engineered safety features by establishing new safety evaluation philosophy making the most of results of the HTTR safety demonstration tests.

The main cooling system, which consists of a primary cooling system, a secondary helium cooling system, and a pressurized water cooling system, removes heat generated in the core and dissipates it to atmosphere by a pressurized water air cooler in the pressurized water cooling system.

The primary cooling system consists of an intermediate heat exchanger (IHX), a primary pressurized water cooler (PPWC), a primary concentric hot gas duct, etc. Primary coolant of helium gas from the reactor at 950 °C maximum flows inside of an inner pipe of the primary concentric hot gas duct to the IHX and the PPWC. The primary coolant is cooled to about 400 °C by the IHX and the PPWC and returns to the reactor flowing through an annulus between the inner and outer pipes of the primary concentric hot gas duct. The HTTR has two operation modes regarding use of heat exchangers for both of the rated operation mode and the high temperature test operation mode. At the single loaded operation mode only the PPWC is operated in the primary cooling system, whereas at the parallel loaded operation mode both the IHX and PPWC are operated, and the IHX and the PPWC remove heat of 10 MW and 20 MW, respectively.

The auxiliary cooling system, consisting of an auxiliary helium cooling system, an auxiliary water cooling system, a concentric hot gas duct, etc. is in standby during normal operation and starts up to remove residual heat after a reactor scram.

The vessel cooling system cools the biological concrete shield surrounding the reactor pressure vessel at normal operation, and removes heat of the core by natural convection and radiation outside of the reactor pressure vessel under accidents of no forced-cooling condition such as rupture of the primary concentric hot gas duct, when neither the main cooling system nor the auxiliary cooling system can cool the core effectively.

The reactor power control device consists of a reactor power control system and a reactor outlet coolant temperature control system. The reactor power and reactor outlet coolant temperature control systems are cascade-connected: the latter control system ranks higher to give demand to the reactor power control system as shown in Fig. 3. The signals from each channel of the power range monitoring system are transferred to three controllers using the microprocessors. In the case that there is a deviation between the process and set values, a pair of control rods is inserted or withdrawn at the control rod speed from 1 to 10 mm/s according to the deviation. The relative position of 13 pairs of control rods, except for three pairs of control rods
used only for the scram, are controlled within 20 mm one another by the control rod pattern interlock to prevent any abnormal power distribution.

The plant control device controls plant parameters such as the reactor inlet coolant temperature, primary coolant flow rate, primary coolant pressure, and differential pressure between the primary cooling system and the pressurized water cooling system. The schematic diagram of the plant control device is shown in Fig. 4.

III. Safety demonstration test plan
In safety demonstration tests using the HTTR, anticipated operational occurrences (AOOs) and accidents are simulated mostly without scram, though most of the postulated AOOs and accidents for the HTTR safety evaluation initiate scrams. The postulated events, which were considered in the safety evaluation of the HTTR as AOOs and accidents, are listed in Tables 1 and 2, respectively.

The safety demonstration tests are conducted to demonstrate inherent safety features of the HTGRs as well as to obtain the core and plant transient data for validation of safety analysis codes and for establishment of safety design and evaluation technologies of the HTGRs. As for the scram tests, loss of off site electric power simulation tests of the HTTR were conducted at 50% and 100% of rated power during the rise to power test in March 2001 and in March 2002, respectively. Safe shutdown and cooling of the reactor was confirmed in the tests.

The safety demonstration tests are divided into the first phase (phase I) and the second phase (phase II). In the phase I safety demonstration tests, AOO simulation tests without a reactor scram will be conducted. The phase I tests consist of “Reactivity insertion test – control rod withdrawal test,” and “Coolant flow reduction test.” In the “Reactivity insertion test,” a central pair of control rods is withdrawn and a reactivity insertion event is simulated. The “Coolant flow reduction test” is composed of “Partial loss of coolant flow test,” and “Gas circulators trip test.” In the “Partial loss of coolant flow test,” primary coolant flow rate is slightly reduced by the control system of the primary coolant flow rate with the reactor outlet coolant temperature control system being operated. In the “Gas circulators trip test,” primary coolant flow rate is reduced to 67% and 33% of rated flow rate by running down one and two out of three gas circulators at the PPWC without a reactor scram, respectively. The phase I tests have already been licensed and are planned to be conducted from FY 2002 to 2005. Test methods and conditions as well as pre-test analysis and test results of the phase I tests are shown in Chapters V to VII.
Table 1  Postulated events classified as AOOs

<table>
<thead>
<tr>
<th>AOO-1</th>
<th>Abnormal control rod withdrawal under subcritical condition</th>
</tr>
</thead>
<tbody>
<tr>
<td>AOO-2</td>
<td>Abnormal control rod withdrawal during rated operation</td>
</tr>
<tr>
<td>AOO-3</td>
<td>Decrease in primary coolant flow rate</td>
</tr>
<tr>
<td>AOO-4</td>
<td>Increase in primary coolant flow rate</td>
</tr>
<tr>
<td>AOO-5</td>
<td>Decrease in heat removal by secondary cooling system</td>
</tr>
<tr>
<td>AOO-6</td>
<td>Increase in heat removal by secondary cooling system</td>
</tr>
<tr>
<td>AOO-7</td>
<td>Loss of off-site electric power</td>
</tr>
<tr>
<td>AOO-8</td>
<td>Abnormality of irradiation specimens and experimental equipment</td>
</tr>
<tr>
<td>AOO-9</td>
<td>Abnormality during safety demonstration tests</td>
</tr>
</tbody>
</table>

Table 2  Postulated events classified as accidents (ACDs)

<table>
<thead>
<tr>
<th>ACD-1</th>
<th>Channel blockage in fuel block</th>
</tr>
</thead>
<tbody>
<tr>
<td>ACD-2</td>
<td>Failure of inner pipe in primary concentric hot gas duct</td>
</tr>
<tr>
<td>ACD-3</td>
<td>Failure of inner pipe in secondary concentric hot gas duct</td>
</tr>
<tr>
<td>ACD-4</td>
<td>Rupture of secondary concentric hot gas duct</td>
</tr>
<tr>
<td>ACD-5</td>
<td>Rupture of pipe in pressurized water cooling system</td>
</tr>
<tr>
<td>ACD-6</td>
<td>Rupture of primary concentric hot gas duct (Depressurization accident)</td>
</tr>
<tr>
<td>ACD-7</td>
<td>Failure of PPWC heat transfer tube</td>
</tr>
<tr>
<td>ACD-8</td>
<td>Failure of primary helium purification system</td>
</tr>
<tr>
<td>ACD-9</td>
<td>Failure of gaseous radwaste treatment system</td>
</tr>
<tr>
<td>ACD-10</td>
<td>Failure of sweep gas pipe in irradiation test equipment</td>
</tr>
<tr>
<td>ACD-11</td>
<td>Failure of standpipe</td>
</tr>
</tbody>
</table>

In the phase II safety demonstration tests, accident simulation tests with or without scram will be mainly performed. The phase II tests, which are advanced tests of the phase I tests, will be conducted after confirming safety features of the HTTR by the phase I tests and obtaining new licenses. The phase II tests include:

i) Loss of forced cooling test (All gas circulators trip test),

ii) All black out test (Vessel cooling system stop test), and

iii) Depressurization test (Simulation of loss of coolant accident (LOCA)).

In the “Loss of forced cooling test,” all the three gas circulators at the PPWC are run down without a reactor scram, and loss of forced cooling will be simulated. In the “All blackout test,” the vessel cooling system, which is an engineered safety feature and has two independent systems, is shut down in addition to the stop of all the gas circulators. This test simulates an accident in which all the cooling system is run down without a reactor scram. In the “Depressurization test,” primary coolant pressure is reduced by removing primary coolant of helium gas to storage tanks in addition to the stop of all the gas circulators, simulating a loss of coolant accident.

IV. Analysis code and model

Pre-test analysis of the safety demonstration tests shown in Chapters V to VII was conducted using the core and plant dynamics analysis code ACCORD developed by JAERI. Steady state and transient behaviors of the reactor and plant of the HTGR can be evaluated using the ACCORD code. The followings are the major characteristics of this code

i) Plant system can be analyzed for over 1hr after an event occurrence by modeling the heat capacity of the reactor core.

ii) Thermal hydraulics for each component such as a reactor, heat exchangers, and concentric hot gas ducts can be analyzed by separating heat transfer calculation for a component from fluid flow calculation for helium and water.

The ACCORD consists of modules for nuclear calculation, heat transfer calculation in reactor, heat exchangers and piping, fluid flow calculation of helium and water, and control system and safety protection system of the HTTR. Figure 5 shows a calculation system of the ACCORD.

Nuclear characteristic is evaluated by conventional point kinetics model with six delayed neutron groups. Thermal power is calculated by a balance of feedback reactivity due to fuel and moderator temperatures of the reactor core with additional reactivity caused by inserting and withdrawing the control rods. Decay heat of fission products after the reactor scram is estimated according to Shure’s formula and decay heat of actinide.
one fuel rod taking into consideration thermal conduction through components and heat transfer of helium. Each heat exchanger is simulated by one channel model with one heat transfer tube. Helium and water flow is approximated by one-dimensional flow network model including flow line and pressure point for calculating the flow rate and pressure. Improvement of the reactor core model to multi-channel will be conducted utilizing results of the safety demonstration tests.

The control system module incorporates proportional and integral control applied to the HTTR control system for reactor power, inlet and outlet temperatures of the reactor, primary and secondary helium flow rate, etc. The safety protection system module incorporates the HTTR scram signal system including time until the occurrence of the scram signal and time until the startup of the auxiliary cooling system after the achievement of scram set value.

The validity of the ACCORD was confirmed through cross checks with other available codes of ASURA and THYDE-HTGR used in safety analysis of the HTTR. The ASURA and the THYDE-HTGR have some disadvantages to the ACCORD in analyzing actual behavior of the HTTR. The ASURA cannot compute transient thermal behavior of the reactor cooling system because static approximation is made for the energy conservation equation for the coolant. The THYDE-HTGR is not suitable for evaluating long term transient behavior because it does not model heat capacity of the core.

V. Reactivity insertion test – Control rod withdrawal test
1. Test objective
The reactivity insertion 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, and the transition of fuel temperatures is slow. Obtained test data is used for development and validation of codes for safety evaluation of HTGRs.

2. Test methods and instrumentation
In the reactivity insertion test, the central pair of control rods out of 16 pairs in the core is withdrawn with the reactor power control system being disabled. In this test, power supply for driving all the control rods excepting the central pair is cut off to prevent withdrawal of control rods by mistake. In addition, the normal value of 20 mm for the control rod pattern interlock is modified to 50 mm, which corresponds to the limitation of continuous withdrawal of control rods for the HTTR. The cut off of power supply driving the control rods and modification of the control rod pattern interlock is done automatically upon selection of the test mode switch. The measurement points and instrumentation are shown in Fig. 6.

3. Test conditions
Test conditions of the reactivity insertion test are shown in Table 3. For the rated operation mode, primary coolant flow rate is 12.4 kg/s, which is larger than that of 10.2 kg/s for the high temperature test operation mode. In the single loaded operation mode, only the PPWC in primary cooling system is in use. The maximum control rods withdrawal rate for the central pair is 5 mm/s to observe the limitation of the reactivity insertion rate for the HTTR, which is $2.4 \times 10^{-4}$ $\Delta k/k/s$, though the value for the other pairs of control rods is 10 mm/s. The maximum withdrawal distance of the control rods is set to 40 mm to observe the pattern interlock of 50 mm.

4. Pre-test analysis and test results
The reactivity insertion tests at the initial reactor thermal power of 15 MW (50%) was conducted in March 2003. Preliminary reactivity insertion tests at the initial power of 9 MW (30%) had been conducted in June 2002.

Figure 7 shows pre-test analysis and test results of the test case when the central pair of control rods is withdrawn 30
mm at the withdrawal rate of 1 mm/s.

The core dynamics parameters used in the analysis are as follows:

- Delayed neutron fraction ($\beta$): $6.5 \times 10^{-3}$
  
  (at 0 Effective Full Power Operation Day (EFPD))
- Prompt neutron lifetime ($\lambda$): $7.3 \times 10^4$ s (at 0 EFPD)
- Reactivity coefficient: see Figure 7

The values of total reactivity insertion and reactivity coefficients in Fig. 7 were calculated using Monte Carlo code, MVP.

Upon withdrawal of the central control rods, positive reactivity is inserted. The reactor power increases because of no reactivity compensation by movement of other control rods. Then the reactor power decreases due to negative reactivity feedback effect with core temperature rise. As shown in Fig. 7, in the test, the reactor power increased to about 56 % from the initial reactor power of 51 % and decreases afterward, whereas the pre-test analysis predicted that the reactor power increases to about 59 %. Cause of the difference between the test and pre-test analysis results has been investigated. The reactor power finally approached to about 52 % in the test as the positive reactivity by control rod withdrawal compensates with the negative one by core temperature rise.

VI. Partial loss of coolant flow test

1. Test objective

The partial loss of coolant flow test demonstrates that the reactor becomes stable by inherent safety characteristics and control systems of HTGRs even when partial loss of coolant flow occurs. Obtained test data is used for development and validation of codes for safety evaluation of HTGRs.

2. Test methods and instrumentation

In the partial loss of coolant flow test, primary coolant flow rate is slightly reduced by the primary coolant flow control system with the reactor outlet coolant temperature control system as well as the reactor power control system kept operated. In this test, primary coolant flow rate is reduced to the level between 100 and 93 % with all control systems operated normally. The HTTR scrams when the primary coolant flow rate reduces to 93 % of rated flow rate. The measurement points and instrumentation are shown in Fig. 6.

3. Test conditions

Test conditions of the partial loss of coolant flow test are shown in Table 4. The changes of flow rate are determined considering the scram level of 93 %.

<table>
<thead>
<tr>
<th>Subject gas circulators</th>
<th>Gas circulators A, B and C at the primary pressurized water cooler</th>
</tr>
</thead>
<tbody>
<tr>
<td>Change of flow rate</td>
<td>-2 %, -4 % (scram level: 93 % of total flow)</td>
</tr>
<tr>
<td>Rate of change for rotation of GCs</td>
<td>100 min$^{-1}$/s</td>
</tr>
<tr>
<td>Reactor power control system</td>
<td>Not disabled</td>
</tr>
</tbody>
</table>

4. Pre-test analysis results

Pre-test analysis results of the partial loss of coolant flow from 100 % to 96 % at the rated reactor power of 30MW are shown in Figs. 8 and 9. The rate of change for primary coolant flow rate is assumed 0.4 %/s in this analysis, that is, primary coolant flow rate is reduced from 100 % to 96 % in 10 seconds.

The core dynamics parameters used in the analysis are as follows:
- Delayed neutron fraction ($\beta$): $6.5 \times 10^{-3}$
  
  (at 0 Effective Full Power Operation Day (EFPD))
- Prompt neutron lifetime ($\lambda$): $7.3 \times 10^4$ s (at 0 EFPD)
- Reactivity coefficient: Values at 0 EFPD which were used in safety analysis of HTTR7)
Figure 8 shows the transitions of reactor power, and fuel temperature. Because the core temperature rises due to the reduction of primary coolant flow rate, the reactor power decreases to about 98% by negative reactivity feedback effect. Figure 9 shows the transition of the coolant temperature. The reactor outlet coolant temperature rises slowly up to about 859 °C due to large heat capacity of structural components at the core bottom. Because the reactor outlet coolant temperature control system is operating, the coolant temperature is controlled to about 850 °C. Therefore, the reactor power and reactor inlet coolant temperature decrease so as to keep a heat balance in the primary cooling system. The maximum fuel temperature rises slowly from the initial temperature of 1300 °C to about 1320 °C according to the reduction of primary coolant flow rate.

**VII. Gas circulators trip test**

**1. Test objective**

The gas circulator trip test demonstrates that rapid decrease of coolant flow rate brings reactor power to stable level by negative reactivity feedback of the core without a reactor shutdown, and the transition of fuel temperatures is slow. Obtained test data is used for development and validation of codes for safety evaluation of HTGRs.

**2. Test methods and instrumentation**

In the gas circulators trip test, primary coolant flow rate is reduced by running down one and two out of three gas circulators with the reactor power control system being disabled. In this test, scram set values of primary coolant flow rate of PPWC (Low), core differential pressure (Low), and reactor outlet coolant temperature (High) are modified to prevent a reactor scram in the course of the tests. At the same time, power supply for driving all the control rods is cut off to prevent withdrawal of control rods by mistake. The modification of scram set values and cut off of power supply driving the control rods is done automatically upon selection of the test mode switch. The measurement points and instrumentation are shown in Fig. 6.

**3. Test conditions**

Test conditions of the gas circulators trip test are shown in Table 5. All gas circulators trip test simulating loss of forced cooling will be conducted in the second phase tests.

<table>
<thead>
<tr>
<th>Operation mode</th>
<th>Rated operation mode</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor thermal power</td>
<td>9 MW (30%) ~ 30 MW (100%) (during the test)</td>
</tr>
<tr>
<td>Reactor outlet coolant temperature</td>
<td>Lower than 850°C (initial)</td>
</tr>
<tr>
<td>Subject gas circulators</td>
<td>Gas circulators (GCs) A, B and C</td>
</tr>
<tr>
<td></td>
<td>at the primary pressurized water cooler</td>
</tr>
</tbody>
</table>

**4. Pre-test analysis and test results**

One out of three gas circulators trip test at the initial reactor thermal power of 9 MW (30%) was conducted in March 2003. Figure 10 shows pre-test analysis and test results of the test.

The core dynamics parameters used in the analysis are as follows:

- Delayed neutron fraction \( \left( \beta \right) : 6.5 \times 10^{-3} \)  
  (at 0 Effective Full Power Operation Day (EFPD))
- Prompt neutron lifetime \( \left( \lambda \right) : 7.3 \times 10^{-4} \) s (at 0 EFPD)
- Reactivity coefficient: see Figure 10

The values of reactivity coefficients in Fig. 10 were calculated using Monte Carlo code, MVP.

![Fig.10 Pre-test analysis and test results of the gas circulators trip test](image)

Figure 10 shows the transition of the primary coolant flow rate and reactor power. Because one gas circulator out of three is run down, the primary coolant flow rate decreases in accordance with the free coast down characteristics of the
gas circulator. The primary coolant flow rate is finally controlled to 67% of initial flow rate because the rest gas circulators keep running with primary coolant flow control system operated.

In this test, the core temperature increased due to reduction of primary flow, and the reactor power decreased to about 24% from the initial reactor power of 30% due to negative reactivity feedback effect. The saturated value of about 24% was in fairly good accordance with the pre-test analysis result of 22%.

VIII. Conclusion
In this paper, a planned test series for demonstration of the safety features (safety demonstration tests), and pre-test analysis and test results of the reactivity insertion test and the coolant flow reduction test were shown and discussed. The test results obtained up to now demonstrated the following phenomena (behaviors of HTTR and HTGRs):

i) 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, and the transition of fuel temperatures is slow.

ii) Rapid decrease of coolant flow rate brings reactor power to stable level by negative reactivity feedback of the core without a reactor shutdown and the transition of fuel temperatures is slow.

The safety demonstration test results are applicable to demonstration of inherent safety features as well as upgrading of safety evaluation technologies of HTGRs, and provided for consideration of cost reduction of future HTGRs by eliminating engineered safety features such as the auxiliary cooling system and reactor containment vessel. Thus, it is expected that the obtained results contribute to commercialization of HTGRs such as the Very High Temperature Reactor System selected as one of the most promising Generation IV systems.

Acknowledgment
The authors would like to express their appreciation to Dr. Yoshinari Anoda (JAERI) for his useful comments and advices.

This research has been conducted as the contract research from the Ministry of Education, Culture, Sports, Science and Technology (MEXT) of Japan since FY2002.

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