JAERI-Tech
2005-015

SAFETY DEMONSTRATION TEST (SR-3/S1C-3/S2C-3/SF-2) PLAN USING THE HTTR (CONTRACT RESEARCH)

March 2005

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編集兼発行　　日本原子力研究所

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(Received January 28, 2005)

Safety demonstration tests using the HTTR are to be conducted from the FY2002 to verify the inherent safety features and to improve the safety design and evaluation technologies for HTGRs, as well as to contribute to not only the commercial HTGRs but also the research and development for the VHTR that is one of the Generation IV reactor candidates.

This paper describes the reactivity insertion test (SR-3), the coolant flow reduction test by tripping of gas circulators (S1C-3, S2C-3), and the partial flow loss of coolant test (SF-2) planned in March 2005 with their detailed test method, procedure and results of pre-test analysis. From the analytical results, it was verified that the negative reactivity feedback effect of the core brings the reactor power safely to a stable level without a reactor scram.

Keywords: Safety Demonstration, Inherent Safety, Gas Circulator Trip, Accident Simulation, Safety Design, Safety Evaluation, Generation IV, HTGR, VHTR, HTTR

Present Study is entrusted from Ministry of Education, Culture, Sports, Science and Technology of Japan.

※ Cooperative Staff

JAERI-Tech　2005-015

HTTR 安全性実証試験（SR-3/S1C-3/S2C-3/SF-2）の試験計画
（受託調査）

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（2005 年 1 月 28 日受理）

高温ガス炉固有の安全性を定量的に実証し、また実用高温ガス炉及び第 4 世代原子炉（Generation IV）の候補のひとつである VHTR の研究開発に資するため、HTTR（高温工学試験研究炉）を用いた安全性実証試験が 2002 年より実施されている。

本報は、2005 年 3 月に計画している制御棒引抜き試験（SR-3）、循環機停止試験（S1C-3,S2C-3）、流量部分喪失試験（SF-2）の試験内容、試験条件、事前解析結果等について述べたものである。事前解析の結果、炉心の負の反応度フィードバック特性により原子炉出力が低下し、原子炉が安定に所定の状態に落ち着くことが確認された。

本報告書は、文部科学省からの受託事業「高温ガス炉固有の安全性の定量的実証」の成果である。
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1. Introduction

Safety demonstration tests using the HTTR[1] (High Temperature Engineering Test Reactor) are in progress in order to verify the inherent safety features and to improve the safety design and evaluation technologies for HTGRs (High Temperature Gas-cooled Reactors). The HTTR, which is the first HTGR in Japan with thermal power of 30MW and 950°C maximum reactor outlet coolant temperature, was constructed at the Oarai Research Establishment of JAERI (Japan Atomic Energy Research Institute) for the purpose of establishing and improving technologies of HTGRs as well as nuclear heat utilization. The HTTR attained its first criticality in November 1998. The rise-to-power tests were started in September 1999 and at the fourth phase of the tests, the HTTR reached its full power of 30MW with reactor outlet coolant temperature of 850°C in December 2001[2][3]. In March 2002, JAERI received a certificate of the pre-operation test, which is an operation permit of the HTTR at the rated operation mode (operation at a reactor outlet coolant temperature of 850°C), from the government. In April 2004, during the fifth and final phase of the rise-to-power test, the HTTR reached its maximum reactor outlet coolant temperature of 950°C[4]-[6]. In June 2004, JAERI received an operation permit for the high temperature test operation mode.

In safety demonstration tests using the HTTR, anticipated operational occurrences (AOOs) and accidents will be simulated mostly without reactor scrams, though most of the postulated AOOs and accidents for the HTTR safety evaluation initiate scrams. 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.

This paper describes the plan of reactivity insertion test, the coolant flow reduction test by tripping one and two out of three gas circulators, and the partial flow loss of coolant test planned during the RS-6 reactor cycle in February and March 2005 with detailed test method, test conditions, and pre-test analysis. These tests will be performed with the reactor power at 80%.

2. Summary of HTTR and Schedule of Safety Demonstration Test

2.1 Outline of HTTR

The main specification of the HTTR is shown in Table 2.1 and the vertical cross section of the HTTR reactor is shown in Fig. 2.1.

The reactor consists of a reactor pressure vessel, fuel elements, replaceable and permanent reflector blocks, core restraint mechanism, control-rods, etc. Thirty columns of fuel blocks and seven columns of control-rod guide blocks form the reactor core, called the fuel region, which is surrounded by replaceable reflector blocks and large-scale permanent reflector blocks. The fuel element of the HTTR is a pin-in-block type. Enrichment of U-235 is 3 to 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 in the control-rod guide blocks by the force of gravity at a constant speed, shutting down the reactor safely.

As shown in Fig. 2.2, the cooling system of the HTTR consists of a main cooling system operating at normal operation; and an auxiliary cooling system and a vessel cooling system, the engineered safety features, operating after a reactor scram to remove residual heat from the core. The main cooling system, which consists of a primary cooling system, a secondary helium cooling system, and a pressurised water cooling system, removes heat generated in the core and dissipates it to the atmosphere by a pressurised water air cooler. The primary cooling system consists of an intermediate heat exchanger (IHX), a primary pressurised water cooler (PPWC), a primary concentric hot gas duct, etc. Primary coolant of helium gas from the reactor at 950°C maximum flows inside the inner pipe of the primary concentric hot gas duct to the IHX and PPWC. The primary helium is cooled to about 400°C by the IHX and PPWC and returns to the reactor flowing through the annulus between the inner and outer pipes of the primary concentric hot gas duct. The HTTR has two operation modes. One is the single loaded operation mode using the PPWC for the primary heat exchange, and the other is the parallel loaded operation mode using the PPWC and IHX. In single loaded operation mode the PPWC removes 30MW of heat and in parallel loaded operation mode the PPWC and IHX remove 20MW and 10MW, 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 stand-by 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 decay heat from the core by natural convection and radiation outside the reactor pressure vessel under 'accident without forced cooling' conditions such as a 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 is controlled by the reactor power control system and reactor-outlet coolant-temperature control system. These control systems are cascade-connected: the latter control system ranks higher to give demand to the reactor power control system. The signals from each channel of the power-range monitoring system are transferred to three controllers using microprocessors. In the event of a deviation between the process-value and set-value, a pair of control-rods is automatically inserted or withdrawn at the 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 a reactor scram, are controlled within 20 mm of one another by the control-rod pattern interlock to prevent any abnormal power distribution. The plant control system keeps plant parameters such as the coolant temperature of the reactor inlet, flow rate of the primary coolant, pressure of the primary coolant, and differential pressure between the primary cooling system, and pressurised water cooling system operating constantly according to the power level. The schematic diagram of the plant control system is shown in Fig. 2.3.

2.2 Schedule of Safety Demonstration Test

The safety demonstration tests are divided into two phases, the first phase (phase I) and second phase (phase II). In the phase I safety demonstration tests, AOO simulation tests including the tests of anticipated transient without scram are conducted. The phase I safety demonstration tests consist of:

[1] Reactivity insertion tests by means of a pair of control-rods withdrawal (SR),

[2] Coolant flow reduction tests by tripping of gas circulator(s) (S1C and S2C), and

[3] Partial flow loss of coolant test (SF).

The phase I tests have already been licenced and are to be conducted from the FY2002 to FY2005. Phase I test items and schedule are summarised in Table 2.2 and Table 2.3, respectively.

The initial reactivity insertion test (SR-0) had started in the FY2002 at the reactor power of 30% (9MW) as the safety demonstration pre-test. The reactivity insertion tests were conducted from the reactor power at 50% (15MW) as SR-1 in the FY2002 and from the reactor power at 60% (18MW) as SR-2 in the FY2003. The coolant flow reduction tests by tripping of one out of three gas circulators were performed from the reactor power at 30% (9MW) as S1C-1 in the FY2002 and at 60% (18MW) as S1C-2 in the FY2003. The coolant flow reduction tests by tripping of two out of three gas circulators were performed from the reactor power at 30% as S2C-1 and at 60% as S2C-2 in the FY2003. The partial flow loss of coolant test (SF-1) was performed from the reactor power at 60% (18MW) in the FY2003.

In this FY2004, the reactivity insertion tests (SR-3), the coolant flow reduction tests by tripping of gas circulators (S1C-3 and S1C-3) and the partial flow loss of coolant test (SF-2) will be conducted from the reactor power at 80% (24MW) and their details are shown in this report.

In the FY2005, the coolant flow reduction tests by tripping of gas circulators (S1C-4 and S1C-4) and the partial flow loss of coolant test (SF-3) will be conducted from the reactor power at 100% (30MW). Also the reactivity insertion tests (SR-3) and the coolant flow reduction tests by tripping of one out of three gas circulators (S1C-5) will be conducted from the reactor power at 30% (9MW) to confirm the reactor behaviour in the condition after getting up the reactor burn-up.

The safety demonstration tests will be conducted step by step with increasing the reactor power. The safety demonstration tests performed previously are summarised in Table 2.4.

The phase II tests, which are more severe than the phase I tests, will be performed in the FY2006 after confirming safety features of the HTTR by the phase I tests and obtaining new licences. The phase II tests include:

[4] Loss of forced cooling test by trip of all the three gas circulators (S3C) and

[5] All-blackout test by stopping the vessel cooling system (SV).

Also a reactivity insertion test (larger than the phase I test), loss of heat removal test by stopping of secondary cooling system which simulate an abnormal event in the heat utilization system such as the hydrogen production system, etc(7)(8)(18) are the candidates for the phase II tests.

Table 2.1 Major specification of HTTR

Item	Specification
Thermal power	30 MW
Coolant	Helium gas
Reactor outlet coolant temperature	850 °C (Rated operation mode) 950 °C (High temperature test operation mode)
Reactor inlet coolant temperature	395 °C
Primary coolant pressure	4.0 MPa
Primary coolant flow rate	12.4 kg/s (Rated operation mode) 10.2 kg/s (High temperature test operation mode)
Core structures	Graphite
Core height	2.9 m
Core diameter	2.3 m
Power density	2.5 MW/m^3
Fuel	Low enriched UO_2
Enrichment	3～10 wt% (avg. 6 wt%)
Fuel element type	Prismatic block
Pressure vessel	Steel (2・1/4Cr – 1Mo)
Number of main cooling loop	1

Table 2.2 Items of phase I safety demonstration test

Test item	Test conditions	Data to be obtained
Reactivity insertion test - Control-rod withdrawal test	• Rated operation mode • Single loaded operation mode • Reactor power : 30%～80% • Central pair of control-rods are withdrawn • Control-rods withdrawal rate : 1 or 5 mm/s • Control-rods withdrawal distance : 50mm (max.) • Control-rods positions will be maintained except the operating one.	Transient data • Reactor power • Reactivity • Primary coolant temperatures • Temperatures of reactor internals, etc.
Coolant flow reduction test - Partial loss of coolant flow test	• Rated operation mode • Single loaded operation mode • Reactor power : 30%～100% • Parameters : change of primary coolant flow rate and rate of change • All of the control systems are operating.	Transient data • Reactor power • Reactivity • Primary coolant temperatures • Primary coolant flow rate, etc.
Coolant flow reduction test - Gas circulators trip test	• Rated operation mode • Single loaded operation mode • Reactor power : 30%～100% • Gas circulators to be stopped : one or two out of three at the primary pressurised water cooler • Each control-rod power supply is intercepted.	Transient data • Reactor power • Reactivity • Primary coolant temperatures • Primary coolant flow rate • Temperature of reactor internals, reactor pressure vessel, etc.

Rated operation mode	Operation at reactor outlet temperature of 850°C
Single loaded operation mode	Operation using primary pressurised water cooler for primary heat exchange

Table 2.3 Schedule of safety demonstration tests

Fiscal Year	2002	2003	2004	2005	2006
Duration of HTTR Operation for Safety Demonstration Tests	Pre-test		Safety Case		
Safety Demonstration Tests *Phase I Tests* *SR: Reactivity Insertion Test* *SF: Partial Flow Loss of Coolant Test* *S1C: 1 Gas Circulator Trip Test* *S2C: 2 Gas Circulators Trip Test* *Phase II Tests* *S3C: 3 Gas Circulators Trip Test* *SV: Vessel Cooling System Stop Test*	***SR-0*: 30%** ***SR-1*: 50%**	***SR-2*: 60%**	***SR-3*: 80%**	***SR-4*: 30%**	***S3C-1, SV-1*** *will start after obtaining new licences.*
		***SF-1*: 60%**	***SF-2*: 80%**	***SF-3*: 100%**	
	***S1C-1*:30%**	***S2C-1*: 30%** ***S1C-2*: 60%** ***S2C-2*: 60%**	***S1C-3*: 80%** ***S2C-3*: 80%**	***S1C-4*: 100%** ***S2C-4*: 100%** ***S1C-5*: 30%**	

Table 2.4 Safety demonstration tests performed previously using the HTTR

Code	Duration	Item	Reactor power (%)	Condition	References Plan	References Results
SR-0	24-30 June 2002	Reactivity insertion test by means of a pair of control-rods withdrawal	30	Low and high speed withdrawal	(8)	-
SR-1	10-11 March 2003		50		(9)	(10)-(12)
SR-2	22-23 February 2004		60		(13)	-
S1C-1	14 March 2003	Coolant flow reduction test by tripping of a gas circulator	30	One out of three gas circulator trip	(9)	(14)(15)
S1C-2	25 February 2004		60		(16)	(17)
S2C-1	11 August 2003	Coolant flow reduction test by tripping of gas circulators	30	Two out of three gas circulator trip	(16)	(16)(18)
S2C-2	5 March 2004		60		(13)	(17)
SF-1	24 February 2004	Coolant flow reduction test by partial flow loss of coolant	60	2% reduction of coolant flow rate	(13)	(17)

SR: Safety Demonstration Test, Reactivity Insertion Test

S1C: Safety Demonstration Test, 1 Gas Circulator Trip Test

S2C: Safety Demonstration Test, 2 Gas Circulators Trip Test

SF: Safety Demonstration Test, Coolant Flow Reduction Test by Partial Flow Loss of Coolant

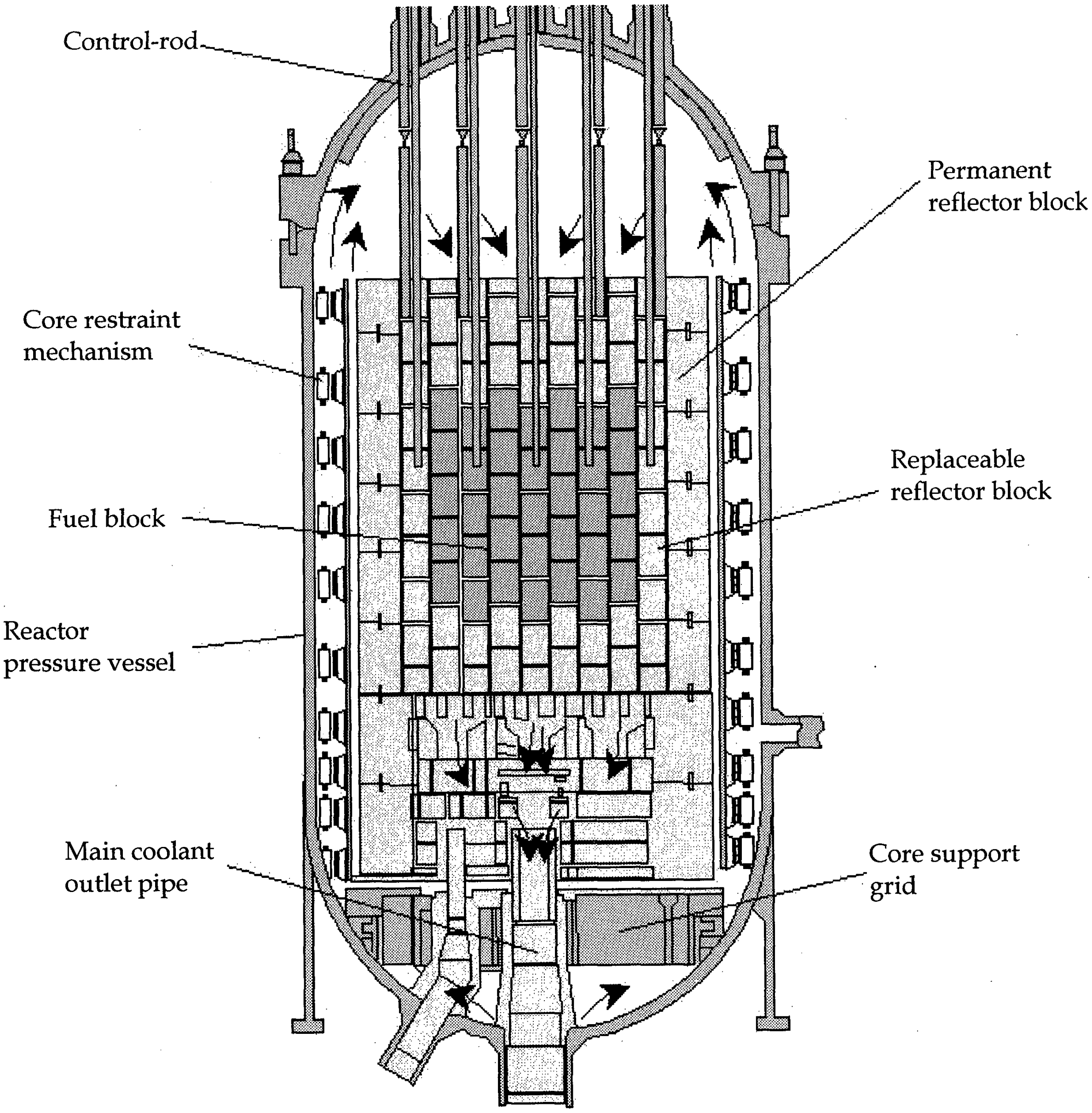

Fig. 2.1 Vertical cross section of HTTR reactor

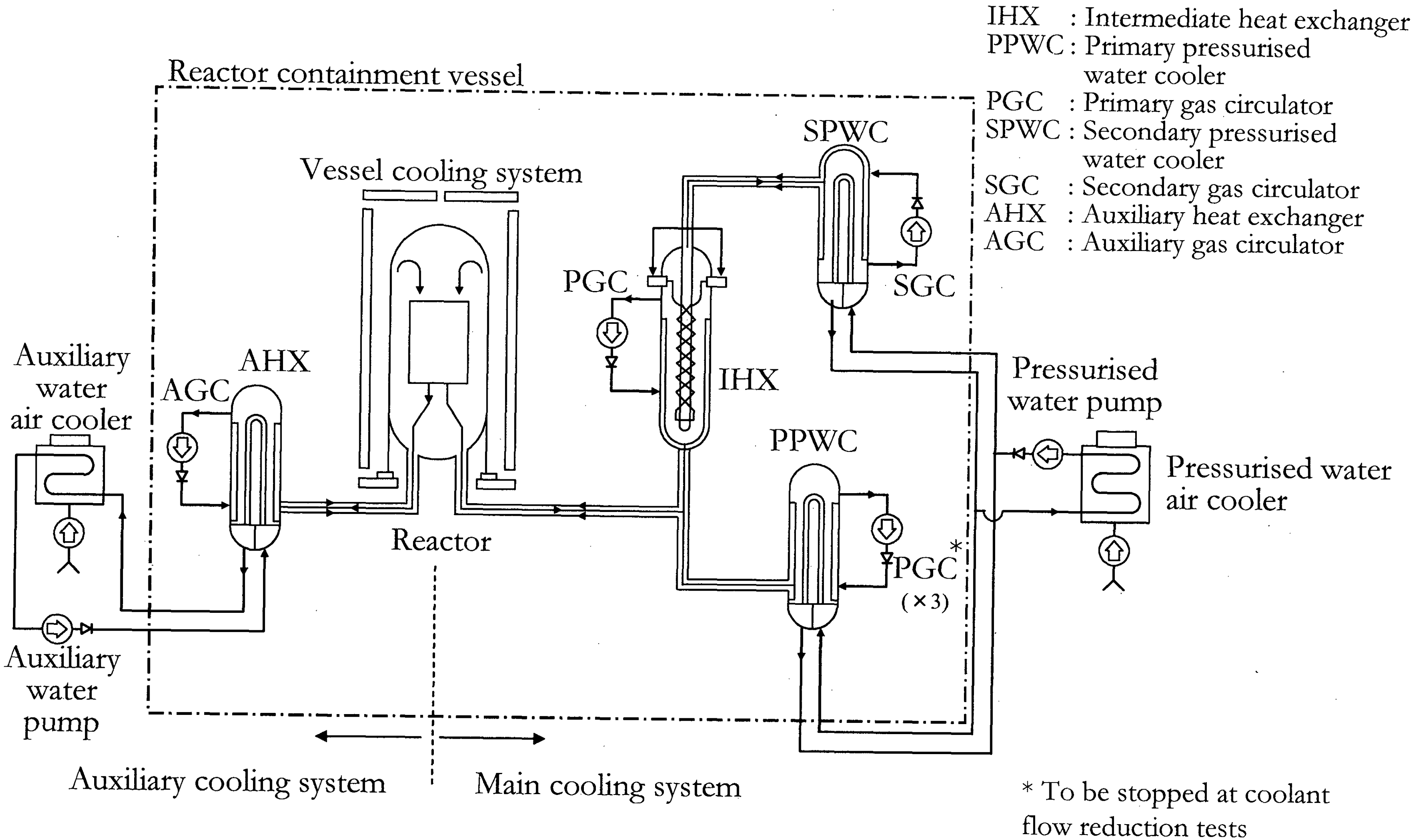

Fig.2.2 Schematic diagram of reactor cooling systems consisting main cooling system, auxiliary cooling system and vessel cooling system

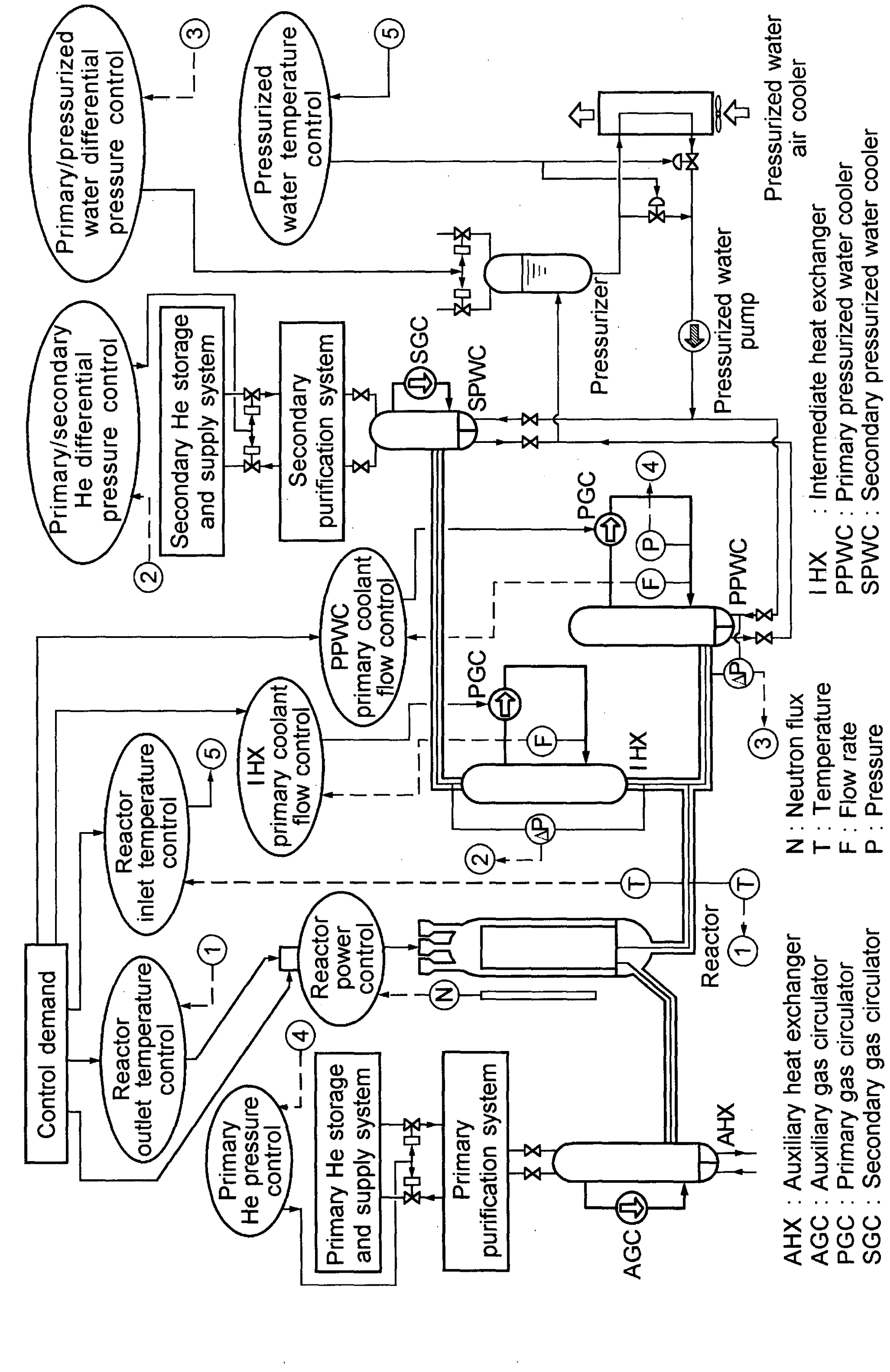

Fig.2.3 Plant control system of HTTR

3. Reactivity Insertion Test by Control-rod Withdrawal (SR-3)

3.1 Test Program

The reactivity insertion test (SR-3:Safety demonstration test, Reactivity insertion test -3) performing without operating the reactor power control system demonstrates that the rapid increase of reactor power caused in such a case of the rupture of the stand-pipe simulating by a centre pair of control-rods withdrawal does not initiate a severe accident due to the inherent safety features which are the negative reactivity feedback effect of the core and the slow temperature transient in HTGRs. The peak power value, decreasing power speed, etc. obtained during the reactivity insertion test will be utilised for the development and validation of analytical models, reactivity coefficients, temperature coefficients, etc. by using their best-estimated values.

3.2 Test Condition

The test conditions of the SR-3 are as follows:

- Operation mode	Rated operation mode, Single loaded operation mode
- Initial reactor power	80% (24MW)
- Reactor outlet coolant temperature	Below 850°C
- Control-rod to be operated	Centre pair of control-rods
- Withdrawal speed	1.5mm/s, 4.5 mm/s
- Reactivity insertion rate	1.7×10^{-5} Δk/k/s, 5.1×10^{-5} Δk/k/s
- Withdrawal length	20mm, 30mm, 40mm
- Reactivity to be inserted	2.3×10^{-4} Δk/k, 3.4×10^{-4} Δk/k, 4.6×10^{-4} Δk/k

The test case of the SR-3 are shown in Table 3.1.

3.3 Measurement Items

Steady state and transient behaviours of the reactor and plant are measured by a plant computer, reactivity measurement instrumentation, TETRIS (High Temperature Engineering Test Reactor Emergency Response Information System), etc. The main items to be measured during the SR-3 are as follows:

- Reactor power (Power range monitoring)
- Reactivity
- Control-rod position
- Inlet temperature of the core
- Coolant temperature at the hot plenum
- Permanent reflector block temperature
- Plenum block temperature

3.4 Test Procedure

(1) Operation Restriction

During the safety demonstration test using the HTTR, there are some operational restrictions from its safety cases. The initial condition of the safety demonstration test, the reactor power and the reactor outlet coolant temperature are must below the value of 80% (24MW) and 850°C, respectively. Also, during the tests, the reactor power, inlet and outlet coolant temperatures, pressure, etc. will be kept within the normal operation conditions.

To set the plant parameters properly during the tests, the operation-mode selector is equipped to the HTTR system. Table 3.2 shows the function of the operation-mode selection equipment with not only the reactivity insertion test but the coolant flow reduction test by tripping of gas circulator and partial loss of coolant flow.

When the reactor power is between 30% and 80%, it is permitted to turn on the 'operation mode (iii)' shown in Table 3.2 to set the initial reactor power from 30% to 80% by the operation-mode selection equipment. After turning on the 'operation mode (iii)', the movement of the control-rods (15 pairs) except the centre pair of control-rods is prevented to avoid the miss-operation of control-rod withdrawal. Also, the pattern-interlock set-value of the centre pair of control-rods is changed from 20mm to 50mm to permit the withdrawal operation for the test. The pattern-interlock set-value at the abnormal conditions is not changed by the 'operation mode (iii)', so the maximum withdrawal distance of the centre pair of control-rods is limited to 50mm.

(2) Test Procedure

The test procedure of the reactivity insertion test is shown in Fig. 3.1. The 'operation mode (iii)' is selected after confirming the initial conditions. In the reactivity insertion test, the centre pair of control-rods is withdrawn and the neutron flux, reactor outlet temperature, primary circuit flow rate, pressure, etc. are measured. After the test, the reactor power is adjusted to the initial value and then the 'operation mode (iii)' is turned off.

3.5 Pre-test Analysis

(1) Analytical Conditions

Pre-test analysis of the reactivity insertion test was conducted using the core and plant dynamics analysis code 'ACCORD' developed by JAERI(19). The characteristics of the code are:

(a) Plant behaviour can be analysed for over several thousand seconds after an event occurrence by modelling the heat capacity of the core,

(b) In-core and plant dynamics of any plant system can be analysed by rearranging packages which simulate plant system components one by one, and

(c) Thermal hydraulics can be analysed for each component from fluid flow calculation for helium and pressurised water systems.

The inserted reactivity and the reactivity coefficients of fuel temperature and moderator temperature used for the analysis are the following values:

- Inserted reactivity (Withdrawal length) — 2.3×10^{-4} Δk/k (20mm)
  3.4×10^{-4} Δk/k (30mm)
  4.6×10^{-4} Δk/k (40mm)
- Reactivity coefficient of fuel temperature — -4.0×10^{-5} Δk/k/°C
- Reactivity coefficient of moderator temperature — -2.1×10^{-5} Δk/k/°C

(2) Analytical Results

The transients of the reactor power at the reactivity insertion test with the reactor power at 80% (24MW) are shown in Fig.3.2.

The positive reactivity is inserted by the centre pair of control-rods withdrawal. Since the reactor power control system is cut off and the movement of the control-rods except the centre pair is prevented during the test, the reactor power is increased to about 85%-91% within about 30 seconds depending on the withdrawal rate and distance. However, the reactor power is diminished to a stable level of about 81%-82% by the negative reactivity feedback effect caused by the temperature increase of the core. The maximum powers and final steady-state powers of the pre-test analysis are shows in Table 3.3.

Table 3.1 Test case of control-rod withdrawal test

Test case	Initial power	Control rod withdrawal rate	Control rod withdrawal distance	Expected inserted reactivity (Δk/k)	Expected reactivity insertion rate (Δk/k/s)
24MWL20	80% (24 MW)	1.5 mm/s	20 mm	2.3×10^{-4}	1.7×10^{-5}
24MWL30			30 mm	3.4×10^{-4}	
24MWL40			40 mm	4.6×10^{-4}	
24MWH20		4.5 mm/s	20 mm	2.3×10^{-4}	5.1×10^{-5}
24MWH30			30 mm	3.4×10^{-4}	
24MWH40			40 mm	4.6×10^{-4}	

Table 3.2 Function of operation-mode selection equipment

Test item	Turning on conditions	Turning off conditions	Functions after turning on	Mode number
Reactivity insertion test - Control-rod withdrawal test	(1) Rated operation mode (2) Single loaded operation mode (3) Reactor power 30% – 80%	The difference between the set-value of the reactor power control system and the measurement value of the reactor power is in the regulated value.	(1) Change of the pattern-interlock set-value of the centre control-rod (2) Withdrawal of the control-rods (15 pairs) except the centre control-rod is prevented.	(iii)
Coolant flow reduction test - Partial loss of coolant flow test	(1) Rated operation mode (2) Single loaded operation mode	None	At the 850°C operation, the set-value of primary coolant flow rate cannot change to the value below the set-value of reactor scram.	(ii)
Coolant flow reduction test - Gas circulators trip test	(1) Rated operation mode (2) Single operation mode	The difference between the set-value of the reactor power control system and the measurement value of the reactor power is in the regulated value.	(1) Withdrawal of the control-rod (16 pairs) is prevented. (2) The set-values of the scram of "primary coolant flow rate of PPWC low", "core differential pressure low" and "reactor outlet coolant temperature high" are changed.	(i)

Table 3.3 Calculation results

Case	Initial power	Control-rod withdrawal rate	Control-rod withdrawal distance	Maximum power	Final steady-state power
24MWL20	80% (24 MW)	1.5 mm/s	20mm	85%	81%
24MWL30			30mm	88%	82%
24MWL40			40mm	90%	82%
24MWH20		4.5 mm/s	20mm	85%	81%
24MWH30			30mm	88%	82%
24MWH40			40mm	91%	82%

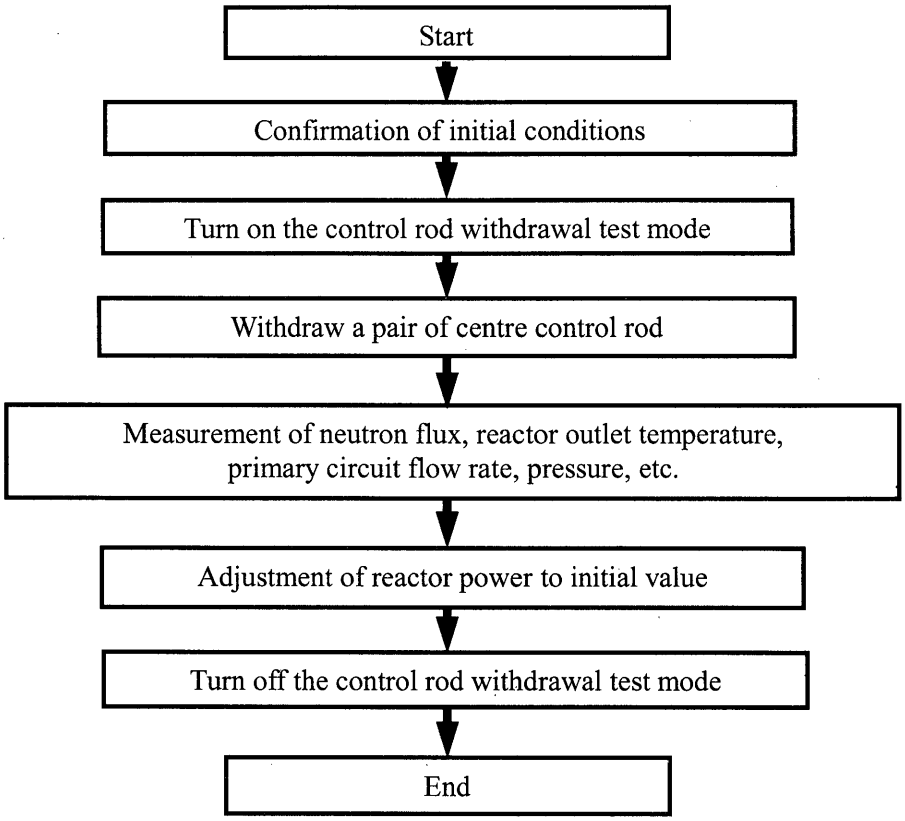

Fig. 3.1 Test procedure of reactivity insertion test by control-rod withdrawal

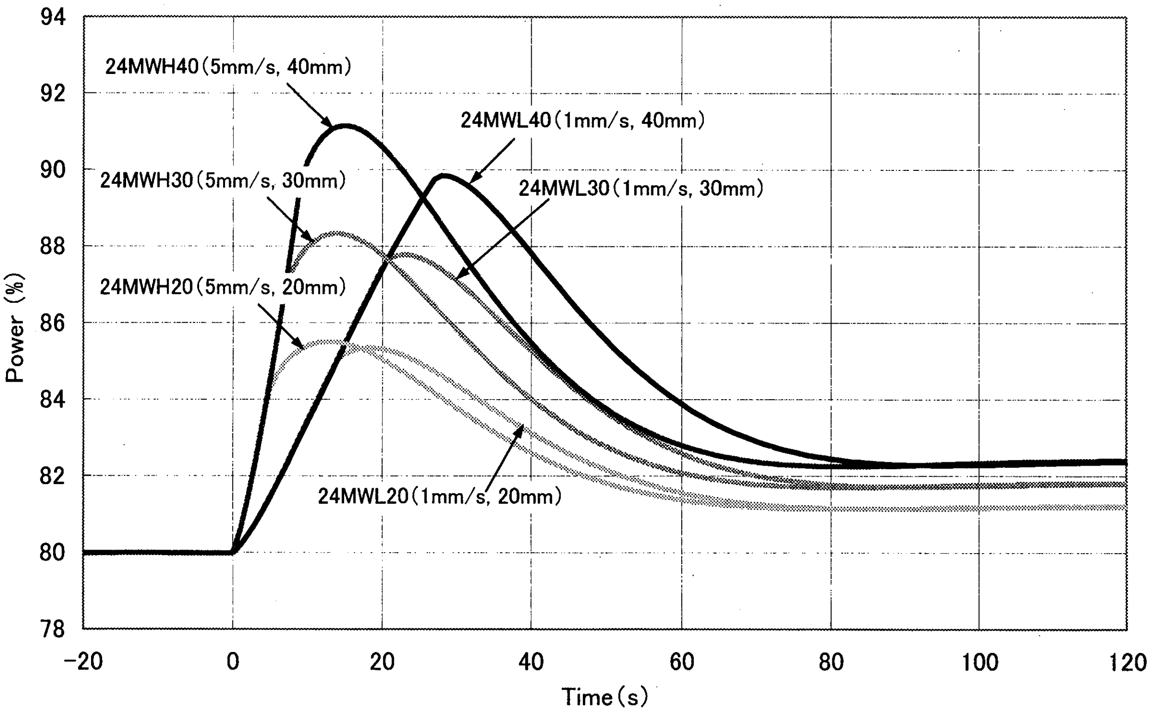

Fig. 3.2 Calculation result of transient of reactor power at reactivity insertion test by control-rod withdrawal

4. Coolant Flow Reduction Test by Trip of Gas Circulator (S1C-3, S2C-3)

4.1 Test Program

The coolant flow reduction test by tripping gas circulators (S1C-3; Safety demonstration test of One out of three gas circulators trip test -3, S2C-3; Safety demonstration test of Two out of three gas circulators trip test -3) demonstrates that the negative reactivity feedback effect of the core brings the reactor power safely to a stable level without a reactor scram, and that the temperature transient of the reactor core is slow in a rapid decrease of the coolant flow rate.

4.2 Test Condition

The test conditions of the S1C-3 and S2C-3 are shown in Table 4.1. In the tests, the primary coolant flow rate is reduced by running down one or two out of three gas circulators without operating the reactor power control system. For the test, some scram levels are changed automatically by the operation mode selector and the plant condition will be an anticipated transient without scram condition (cf. following 4.4). At the end of this test, the reactor is shut down by a manual scram.

4.3 Measurement Items

Steady state and transient behaviours of the reactor and plant are measured by a plant computer, reactivity measurement instrumentation, TETRIS, etc. The main items to be measured in the gas circulator trip tests are as follows:

- Reactor Power (Power range monitoring)
- Reactivity
- Reactor core differential pressure
- Coolant flow rate
- Inlet temperature of the core
- Coolant temperature at hot plenum
- Permanent reflector block temperature
- Plenum block temperature

4.4 Test Procedure

(1) Initial Test Condition

When the primary coolant flow-rate decreases by tripping of gas circulators, reactor scram could be caused by the shutdown signal: 'primary coolant flow rate of PPWC is low' or 'core differential pressure is low'. Also reactor scram could be caused by the shutdown signal: 'reactor outlet coolant temperature is high' by the increasing of reactor outlet coolant temperature due to the stop of gas circulators. To avoid the possibility of a reactor scram by the above three shutdown signals, the scram signal values will be changed automatically by synchronising of the operation-mode selection equipment as well as preventing wrong operation by an operator.

To maintain the control-rods positions during the tests, the selection equipment of the operation-mode is regulated such that the control system of the reactor power is a manual mode. The function of the selection equipment of the operation-mode is shown in Table 3.2. After turning on the 'operation-mode (i)' shown in Table 3.2, the each control-rod power supply is intercepted. Then, the gas circulator trip test can be conducted as an anticipated transient without scram.

(2) Test Procedure

The test procedure of the gas circulator trip test is shown in Fig. 4.1. The test mode is selected after the confirmation of initial conditions. In the one gas circulator trip test, the gas circulator No. A at the PPWC are stopped. In the two gas circulators trip test, the gas circulators No. A and No. C are stopped. After measuring the neutron flux, reactor outlet coolant temperature, primary circuit flow rate, pressure, etc. for about four hours, the reactor is shut down by a manual scram.

4.5 Pre-test Analysis

(1) Analytical Conditions

Pre-test analysis of the gas circulator trip test was conducted using the ACCORD code. The reactivity coefficients of the fuel temperature and moderator temperature used for the analysis are the following values:

One and two out of three gas circulators trip test with the reactor power at 80%

- Reactivity coefficient of fuel temperature $-4.0\times10^{-5}\,\Delta k/k/\,^{\circ}C$
- Reactivity coefficient of moderator temperature $-2.1\times10^{-5}\,\Delta k/k/\,^{\circ}C$

The reactor kinetics parameters of the effective delayed neutron fraction and prompt neutron lifetime used for the analysis are the following values:

- Effective delayed neutron fraction 6.5×10^{-3}
- Prompt neutron lifetime 9.4×10^{-4} s

The decrease flow-rate characteristics of the primary helium gas circulators are the measured data of the previous one or two gas circulators trip test with the reactor power at 30%.

(2) Analytical Results

The transients of the reactor power and primary coolant flow rate at the one and two gas circulator trip test with the reactor power at 80% (24MW) are shown in Fig. 4.2 (one gas circulator trip test) and Fig. 4.3 (two gas circulator trip test) respectively.

The flow rate of the primary circuit decreases within 10 seconds after tripping gas circulators. The reactor power is diminished by the negative reactivity feedback effect due to the increase of reactor core temperature caused by decreasing of rapid and large change of the coolant flow. After several hours have elapsed, the reactor power reaches to a stable condition. Since the remaining gas circulator operates continuously with its flow rate control devices, the flow rates become 66%(one gas circulator trip) and 33%(two gas circulator trip) of the initial values and the reactor powers are finally reached to about 58%(one gas circulator trip) and 31%(two gas circulator trip).

Table 4.1 Test condition of coolant flow reduction test by trip of gas circulators

	One gas circulator trip test	Two gas circulator trip test
Operation mode	Rated operation mode, single loaded operation mode	
Initial Reactor power	80% (24MW)	
Reactor outlet temperature	Below 850°C (Initial)	
	Below 950°C (During the test)	
HGC to be stopped	One	Two
	HGC for PPWC No. A	HGC for PPWC No. A and C

HGC	Helium Gas Circulator
PPWC	Primary Pressurised Water Cooler
Rated operation mode	Operation at reactor outlet coolant temperature of 850°C
Single loaded operation mode	Operation using primary pressurised water cooler for primary heat exchange

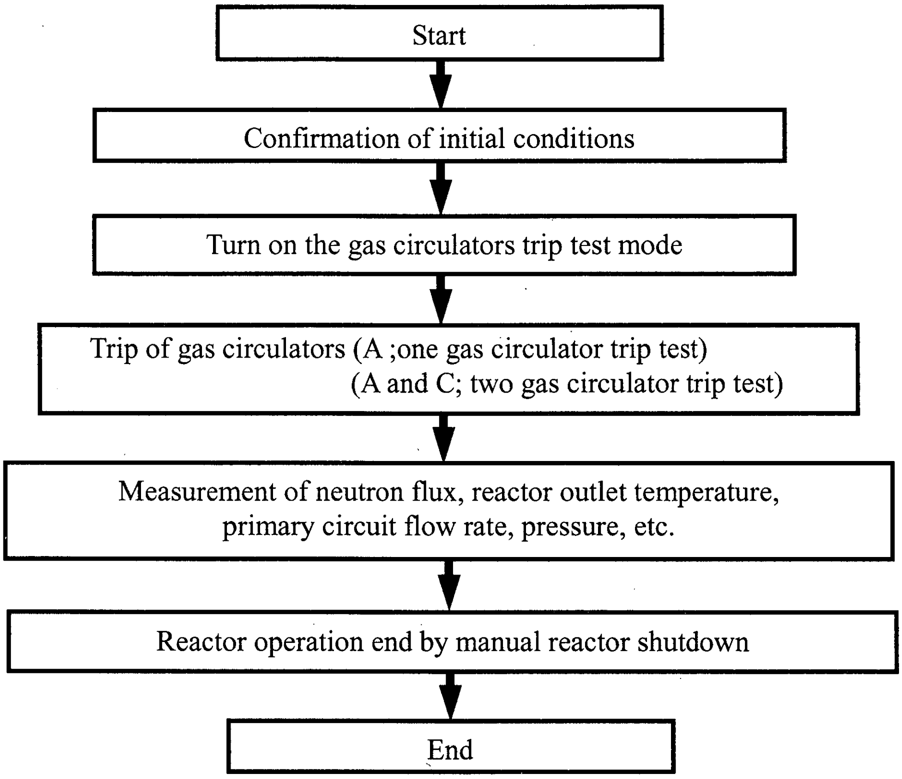

Fig.4.1 Test procedure of coolant flow reduction test by trip of gas circulators

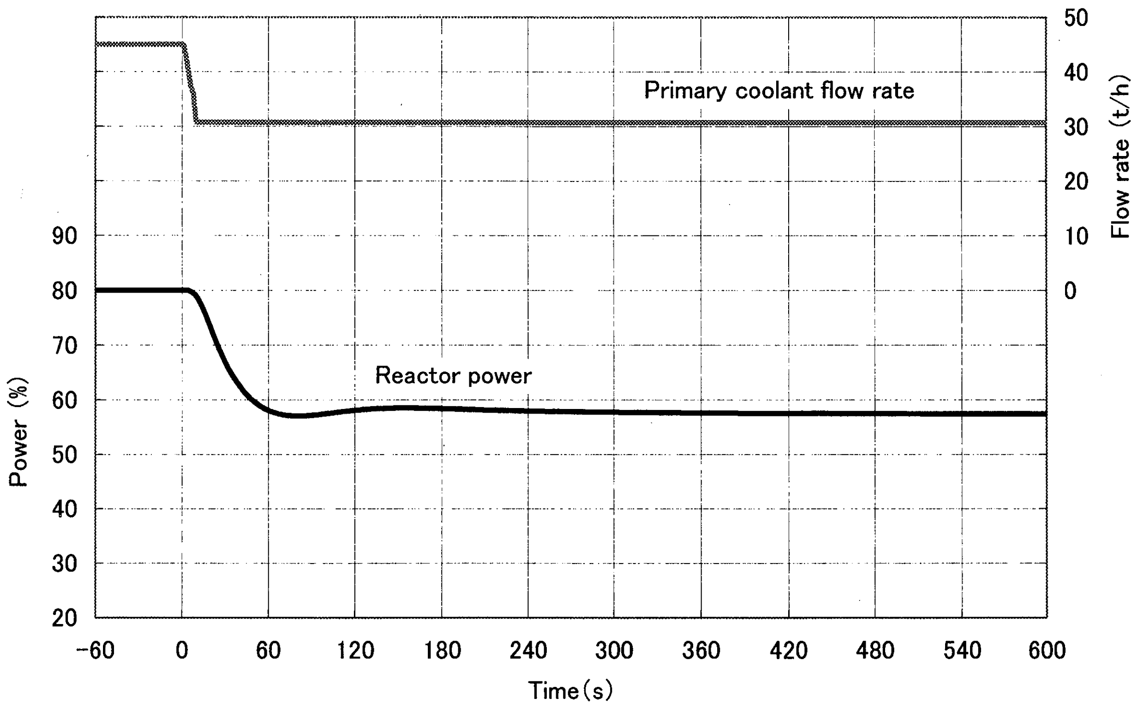

Fig. 4.2 Calculated result of transients of reactor power and primary coolant flow rate during coolant flow reduction test by tripping of one out of three gas circulators

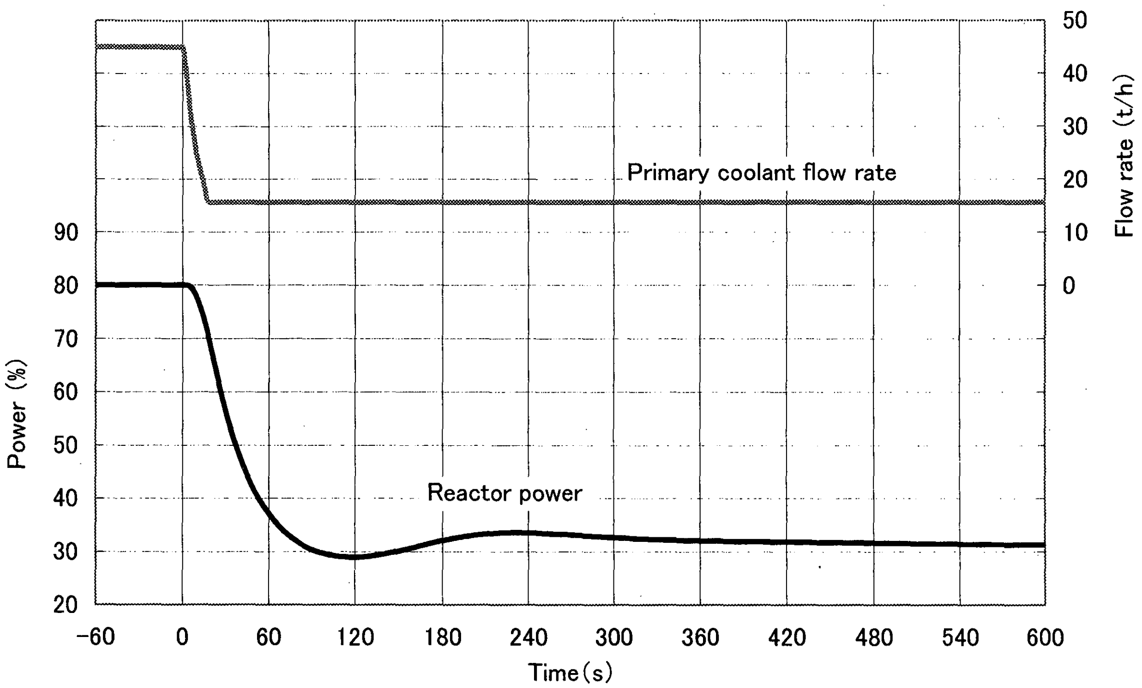

Fig. 4.3 Calculated result of transients of reactor power and primary coolant flow rate during coolant flow reduction test by tripping of two out of three gas circulators

5. Coolant Flow Reduction Test by Partial Flow Loss of Coolant (SF-2)

5.1 Test Program

The coolant flow reduction test by partial flow loss of coolant (SF-2: Safety demonstration test, coolant flow reduction test by partial flow loss of coolant -2) demonstrates that the both of negative reactivity feedback effect of the core and the reactor power control system brings the reactor power safely to a stable level without a reactor scram, and that the temperature transient of the reactor core is slow in a decrease of the coolant flow rate.

5.2 Test Condition

The test conditions of the SF-1 are as follows:

- Operation mode	Rated operation mode, Single loaded operation mode
- Initial reactor power	80% (24MW)
- Set-value of primary coolant flow rate	15.07t/h (Initial), 14.77t/h (During the test)

5.3 Measurement Items

Steady state and transient behaviours of the reactor and plant are measured by a plant computer, reactivity measurement instrumentation, TETRIS, etc. The main items to be measured during the SF-1 are as follows:

- Reactor power (Power range monitoring)
- Reactivity
- Reactor core differential pressure
- Coolant flow rate
- Inlet temperature of the core
- Coolant temperature at hot plenum
- Permanent reflector block temperature
- Plenum block temperature

5.4 Test Procedure

The test procedure of the coolant flow reduction test by partial flow loss of coolant is shown in Fig. 5.1. The test mode is selected after confirmation of initial conditions. The set-value of primary coolant flow rate is changed from 15.07t/h to 14.77t/h (diminution of 2%). After measuring the neutron flux, reactor outlet coolant temperature, primary circuit flow rate, pressure, etc. for about four hours, the primary coolant flow rate is increased by changing the set-value from 14.77t/h to 15.07t/h. The test mode selection equipment is turned off and returned to an initial condition.

5.5 Pre-test Analysis

The transients of the reactor power and primary coolant flow rate at the coolant flow reduction test by partial flow loss of coolant with the reactor power at 80% (24MW) calculated by the ACCORD code are shown in Fig. 5.2. The temperature of the core increased due to the diminution of the flow rate of the primary circuit after changing the flow-rate set-value from 15.07t/h to 14.77t/h. The reactor power reaches to the stable level of 80% by a reactor power control system, though the reactor power decreased about 0.5% due to the negative reactivity feedback effect of the core.

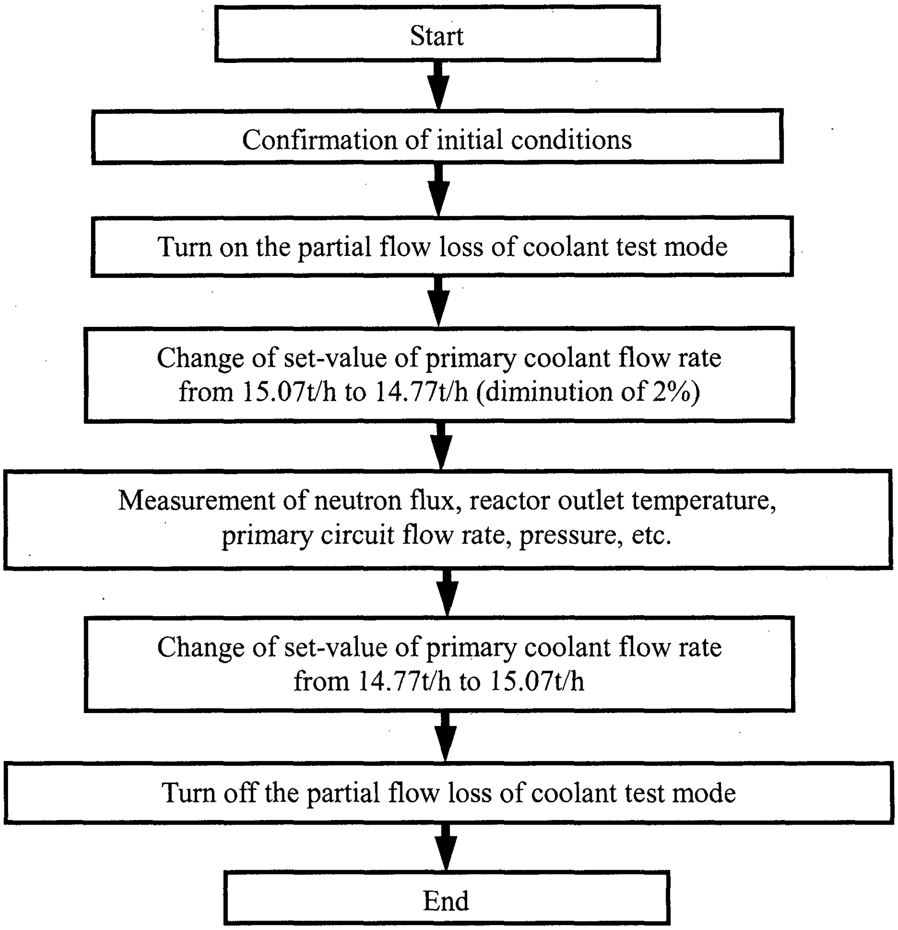

Fig 5.1 Test procedure of coolant flow reduction test by partial flow loss of coolant

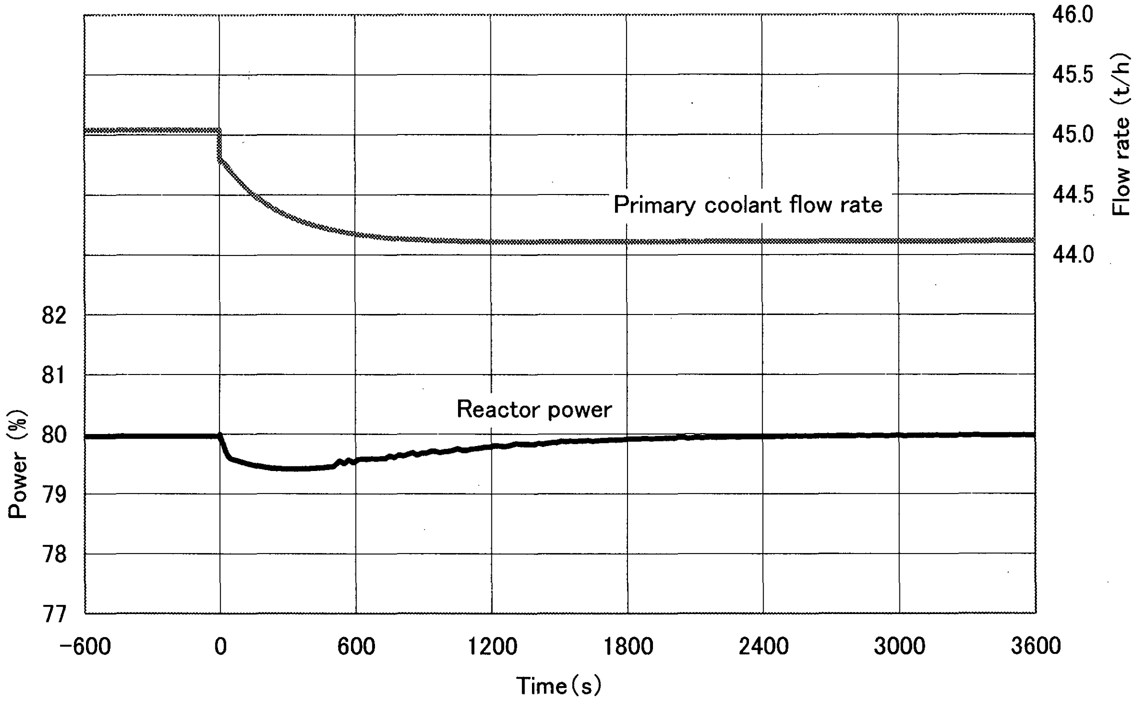

Fig 5.2 Calculated result of transients of reactor power and primary coolant flow rate during coolant flow reduction test by partial flow loss of coolant

6. Concluding Remarks

The reactivity insertion test by means of a pair of control-rod withdrawal, coolant flow reduction tests by tripping one and two out of three gas circulators, and partial flow loss of coolant test with the reactor power at 80% are planned in February and March 2005. Their test method, procedure and results of pre-test analysis are described in this report. From the analytical results of the steady state and transient behaviours of the reactor and plant of the HTTR, it was found that the negative reactivity feedback effect of the core brings the reactor power safely to a stable level without a reactor scram, and that the temperature transient of the reactor core is slow.

Acknowledgment

The authors would like to express their appreciation to Messrs. Satoshi Shimakawa and Naoki Nojiri of JAERI for their efforts to evaluate nuclear parameters such as control rod worth, temperature coefficient of reactivity, etc. The authors also would like to express their appreciation to Mr. Shoji Katanishi, Mr. Yoshihiro Komori, Dr. Yoshinari Anoda, Mr. Hideyuki Hayashi, Mr. Toshio Nakazawa, Mr. Toshihiko Mizushima, Mr. Kozo Kawasaki, Mr. Yukio Tachibana, and Mr. Tatsuo Iyoku of JAERI for their useful comments and advice.

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国際単位系（SI）と換算表

表1　SI基本単位および補助単位

量	名　称	記　号
長　　さ	メートル	m
質　　量	キログラム	kg
時　　間	秒	s
電　　流	アンペア	A
熱力学温度	ケルビン	K
物 質 量	モ　ル	mol
光　　度	カンデラ	cd
平 面 角	ラジアン	rad
立 体 角	ステラジアン	sr

表2　SIと併用される単位

名　称	記　号
分，時，日	min，h，d
度，分，秒	°，′，″
リットル	l，L
トン	t
電子ボルト	eV
原子質量単位	u

$1\ \mathrm{eV}=1.60218\times10^{-19}\ \mathrm{J}$

$1\ \mathrm{u}=1.66054\times10^{-27}\ \mathrm{kg}$

表5　SI接頭語

倍数	接頭語	記　号
10^{18}	エクサ	E
10^{15}	ペタ	P
10^{12}	テラ	T
10^{9}	ギガ	G
10^{6}	メガ	M
10^{3}	キロ	k
10^{2}	ヘクト	h
10^{1}	デカ	da
10^{-1}	デシ	d
10^{-2}	センチ	c
10^{-3}	ミリ	m
10^{-6}	マイクロ	μ
10^{-9}	ナノ	n
10^{-12}	ピコ	p
10^{-15}	フェムト	f
10^{-18}	アト	a

表3　固有の名称をもつSI組立単位

量	名　称	記号	他のSI単位による表現
周　波　数	ヘルツ	Hz	s^{-1}
力	ニュートン	N	$m\cdot kg/s^2$
圧力，応力	パスカル	Pa	N/m^2
エネルギー，仕事，熱量	ジュール	J	$N\cdot m$
工率，放射束	ワット	W	J/s
電気量，電荷	クーロン	C	$A\cdot s$
電位，電圧，起電力	ボルト	V	W/A
静 電 容 量	ファラド	F	C/V
電 気 抵 抗	オーム	Ω	V/A
コンダクタンス	ジーメンス	S	A/V
磁　　束	ウェーバ	Wb	$V\cdot s$
磁 束 密 度	テスラ	T	Wb/m^2
インダクタンス	ヘンリー	H	Wb/A
セルシウス温度	セルシウス度	℃	
光　　束	ルーメン	lm	$cd\cdot sr$
照　　度	ルクス	lx	lm/m^2
放　射　能	ベクレル	Bq	s^{-1}
吸 収 線 量	グレイ	Gy	J/kg
線 量 等 量	シーベルト	Sv	J/kg

表4　SIと共に暫定的に維持される単位

名　称	記　号
オングストローム	Å
バーン	b
バール	bar
ガル	Gal
キュリー	Ci
レントゲン	R
ラド	rad
レム	rem

$1\ \mathrm{Å}=0.1\mathrm{nm}=10^{-10}\mathrm{m}$

$1\ \mathrm{b}=100\mathrm{fm}^2=10^{-28}\mathrm{m}^2$

$1\ \mathrm{bar}=0.1\mathrm{MPa}=10^5\mathrm{Pa}$

$1\ \mathrm{Gal}=1\mathrm{cm/s}^2=10^{-2}\mathrm{m/s}^2$

$1\ \mathrm{Ci}=3.7\times10^{10}\mathrm{Bq}$

$1\ \mathrm{R}=2.58\times10^{-4}\mathrm{C/kg}$

$1\ \mathrm{rad}=1\mathrm{cGy}=10^{-2}\mathrm{Gy}$

$1\ \mathrm{rem}=1\mathrm{cSv}=10^{-2}\mathrm{Sv}$

（注）

1．表1－5は「国際単位系」第5版，国際度量衡局1985年刊行による。ただし，1 eVおよび1 uの値はCODATAの1986年推奨値によった。

2．表4には海里，ノット，アール，ヘクタールも含まれているが日常の単位なのでここでは省略した。

3．barは，JISでは流体の圧力を表わす場合に限り表2のカテゴリーに分類されている。

4．EC閣僚理事会指令ではbar，barnおよび「血圧の単位」mmHgを表2のカテゴリーに入れている。

換　算　表

力	$N(=10^5dyn)$	kgf	lbf
	1	0.101972	0.224809
	9.80665	1	2.20462
	4.44822	0.453592	1

粘　度　$1\ Pa\cdot s(N\cdot s/m^2)=10\ P$（ポアズ）$(g/(cm\cdot s))$

動粘度　$1m^2/s=10^4St$（ストークス）(cm^2/s)

圧力	MPa(=10bar)	kgf/cm^2	atm	mmHg(Torr)	$lbf/in^2(psi)$
	1	10.1972	9.86923	7.50062×10^{3}	145.038
	0.0980665	1	0.967841	735.559	14.2233
	0.101325	1.03323	1	760	14.6959
	1.33322×10^{-4}	1.35951×10^{-3}	1.31579×10^{-3}	1	1.93368×10^{-2}
	6.89476×10^{-3}	7.03070×10^{-2}	6.80460×10^{-2}	51.7149	1

エネルギー・仕事・熱量	$J(=10^7\ erg)$	$kgf\cdot m$	$kW\cdot h$	cal(計量法)	Btu	$ft\cdot lbf$	eV
	1	0.101972	2.77778×10^{-7}	0.238889	9.47813×10^{-4}	0.737562	6.24150×10^{18}
	9.80665	1	2.72407×10^{-6}	2.34270	9.29487×10^{-3}	7.23301	6.12082×10^{19}
	3.6×10^{6}	3.67098×10^{5}	1	8.59999×10^{5}	3412.13	2.65522×10^{6}	2.24694×10^{25}
	4.18605	0.426858	1.16279×10^{-6}	1	3.96759×10^{-3}	3.08747	2.61272×10^{19}
	1055.06	107.586	2.93072×10^{-4}	252.042	1	778.172	6.58515×10^{21}
	1.35582	0.138255	3.76616×10^{-7}	0.323890	1.28506×10^{-3}	1	8.46233×10^{18}
	1.60218×10^{-19}	1.63377×10^{-20}	4.45050×10^{-26}	3.82743×10^{-20}	1.51857×10^{-22}	1.18171×10^{-19}	1

1 cal＝4.18605J（計量法）

＝4.184J（熱化学）

＝4.1855J（15℃）

＝4.1868J（国際蒸気表）

仕事率　1 PS（仏馬力）

＝75 $kgf\cdot m/s$

＝735.499W

放射能	Bq	Ci
	1	2.70270×10^{-11}
	3.7×10^{10}	1

吸収線量	Gy	rad
	1	100
	0.01	1

照射線量	C/kg	R
	1	3876
	2.58×10^{-4}	1

線量当量	Sv	rem
	1	100
	0.01	1

（86年12月26日現在）

Safety Demonstration Test (SR-3/S1C-3/S2C-3/SF-2) Plan Using the HTTR (Contract Research)