Status report 101 - Gas Turbine High Temperature Reactor (GTHTR300C)

Overview

Full name	Gas Turbine High Temperature Reactor
Acronym	GTHTR300C
Reactor type	Block Type Reactor
Coolant	Helium
Moderator	Graphite
Neutron spectrum	Thermal Neutrons
Thermal capacity	600.00 MWth
Gross Electrical capacity	274.00 MWe
Design status	Conceptual Design
Designers	JAEA
Last update	21-07-2011

Description

Introduction

The GTHTR300C is a passively-safe Generation IV reactor plant design that enables production and co-production of electricity, hydrogen and industrial process heat. It consists of a modular VHTR (very high temperature reactor) rated up to 600 MWt each reactor, an efficient direct cycle gas turbine for electricity generation, and, for co-production, an intermediate heat transport loop for high temperature heat supply for hydrogen and process steam generation.

JAEA has been building the design basis for the GTHTR300C through several long term research and development programs. The program that has resulted in the proven construction and operation of the high temperature engineering test reactor (HTTR) with thermal power of 30MW and coolant outlet temperature of 950^{o}C has established a substantial design, operational and maintenance database for the VHTR. The HTTR is currently the largest operating test reactor for the VHTR technology in the world. It has demonstrated long-term stable operation at about 950^{o}C and the passive reactor safety performance for the loss of coolant circulation events.

The design and development program for a highly efficient electric power conversion system for the GTHTR300C has been conducted. This program included one-third scale tests of the helium gas turbine equipment and validated the high aerodynamic efficiency of the helium compressor technology. In parallel, a research program for hydrogen production by a thermochemical iodine-sulfur (IS) process had been conducted. The combination of the IS process with the VHTR heat source offers large-scale centralized hydrogen production without CO_2 emission.

On the basis of the reactor and application technologies obtained in the above mentioned programs and with the joint efforts of JAEA and domestic nuclear industries, the GTHTR300C has been designed to provide flexible production and a clean and economical source of hydrogen fuel and electricity while protecting the environment from global warming and so on. The GTHTR300C generates up to 300 MWe electricity at 45-50% thermal efficiency by a direct cycle gas turbine power conversion system and up to 1.4 million Nm^3 hydrogen / day at about 45% efficiency by the IS process, or cogenerates both electricity and hydrogen in these ranges. These production capabilities are evaluated to be economical and can meet the anticipated domestic demand for electricity and hydrogen after 2030 in Japan.

The baseline system is the reactor power plant shown in Figure 1, which includes the reactor system with the gas turbine power conversion system. The reactor is rated 600 MWt thermal power and 850~950^oC reactor outlet temperatures. By an intermediate heat transport loop, a share of the high temperature reactor heat is delivered in piping as high temperature process heat to the adjacent hydrogen plant. The electricity need for hydrogen production is met in house from the efficient gas turbine power cogeneration. Table 1 summarizes the GTHTR300C design and production specifications.

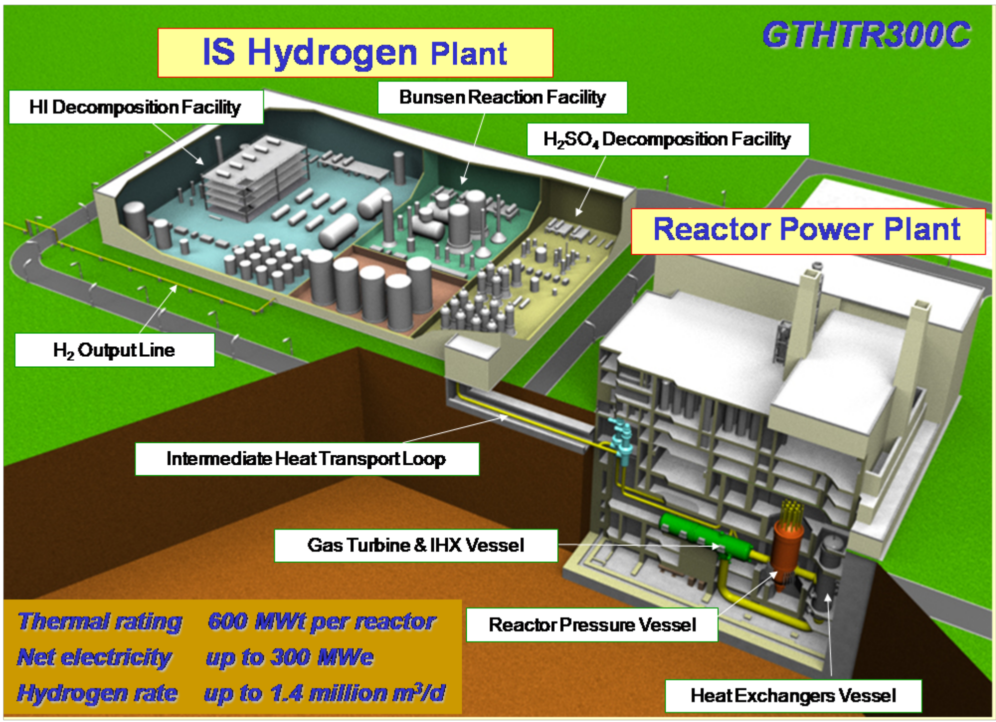

Figure 1: JAEA's GTHTR300C design to produce electricity, hydrogen or simultaneously both

The salient design features of the GTHTR300C include: 1) TRISO coated particle fuel; 2) The highest thermal power commensurate with passive safety; 3) Sandwich shuffling of fuel blocks with the proper installation of burnable poison to limit peak fuel temperature and to extend refueling interval; 4) Conventional steel material (SA533) for reactor pressure vessel construction; 5) Reactor confinement instead of containment; 6) 40-50% efficient production of electricity and hydrogen; 7) Non-intercooled, horizontal-oriented, single-shaft gas turbine generator for system simplicity; and 8)Air cooled spent fuel storage. These design features will be described in detail in the later sections.

Table 1 GTHTR300C design and major production specifications

	GTHTR300C power generation only	GTHTR300C power & H_2 cogeneration	GTHTR300C Mainly H_2 generation
Reactor thermal power	600 MWt/module	600 MWt/module	600 MWt/module
Reactor lifetime	60 years	60 years	60 years
Plant availability	90% +	90% +	90% +
Reactor fuel cycle	LEU, MOX, others	LEU, MOX, others	LEU, MOX, others
Reactor fuel design	TRISO coated particles	TRISO coated particles	TRISO coated particles
Reactor pressure vessel	SA508/SA533 steel	SA508/SA533 steel	SA508/SA533 steel
Reactor core coolant	Helium gas	Helium gas	Helium gas
Core coolant flow	439 / 403 kg/s	324 kg/s	324 kg/s
Core inlet temperature	587 / 663°C	594°C	594°C
Core outlet temperature	850 / 950°C	950°C	950°C
Core coolant pressure	6.9 MPa	5.1 MPa	5.1 MPa
Core power density	5.4 W/cc	5.4 W/cc	5.4 W/cc
Average fuel burnup	120 GWd/ton	120 GWd/ton	120 GWd/ton
Refueling interval	24 / 18 months	18 months	18 months
GT conversion cycle	non-intercooled direct Brayton cycle	non-intercooled direct Brayton cycle	non-intercooled direct Brayton cycle
GT cycle pressure ratio	2.0	2.0	1.5
Power generation efficiency	47% / 51%	47%	38%
Net electricity output	274 / 300 MWe	174 MWe	34 MWe
H_2 plant effective heat rate	n/a	220 MWt	505 MWt
H_2 conversion process	n/a	thermochemical (e.g. S/I) or hybrid (thermal-electro)	thermochemical (e.g. S/I) or hybrid (thermal-electro)
H_2 conversion efficiency	n/a	43%	41%
H_2 Production	n/a	58 tonnes /day (0.64 million m^3/day)	126 tonnes/day (1.41 million m^3/day)
Total plant efficiency (net)	45~50%	45%	40%

In sum, the GTHTR300C design is based the technologies already accumulated in JAEA such that any new technology development required to commercially deploy the system is limited and the investment risk is minimized. The system is considered by the designer to be economically attractive and deployable as a new energy source in as early as 2020-2030.

Description of the nuclear systems

The reactor design is based on the basic techniques accumulated in HTTR design with the design modifications intended to maximize the economics of the commercial design.

2.1 Fuel design

Based on the design and operational experiences of the HTTR fuel and experiments for high burnup fuel, the major specifications of the GTHTR300 fuel were determined. Two sets of fuel design specifications have been prepared by JAEA, one of which is given in Table 2. In this section, the outline of fuel design and fuel integrity evaluation is described.

Table 2: A set of GTHTR300C fuel specification

Table 3

Specifications of GTHTR300 fuel

Fuel rod length	1000	mm
Coolant channel diameter	39	mm
Fuel compact		
Length	80	mm
Inner diameter	9	mm
Outer diameter	24	mm
Sheath thickness	1	mm
Packing fraction of fuel particle	29	vol%
Coated fuel particle		
Coating type	TRISO	
Diameter	1010	μ m
Fuel kernel		
Material	UO_2	
Enritchment	14	wt%
Diameter	550	μ m
Density	10.80	g/cm^3
Buffer layer		
Thickness	140	μ m
Density	1.15	g/cm^3
IPyC layer		
Thickness	25	μ m
Density	1.85	g/cm^3
SiC layer		
Thickness	40	μ m
Density	3.20	g/cm^3
OPyC layer		
Thickness	25	μ m
Density	1.85	g/cm^3

2.1.1 Fuel design description

The fuel assembly is a so called pin-in-block type, which is composed of fuel rods and a hexagonal fuel block (Figure 2). A fuel rod consists of 12 hollow fuel compacts with 9 mm in inner diameter and 26mm in outer diameter. The fuel compacts are vertically piled up on the bottom plate of a central rod. The GTHTR300 fuel is not contained in a graphite sleeve, different from the HTTR fuel. That makes the heat transfer from fuel compacts to coolant more effectively, and makes it possible to keep the fuel temperature as low as possible. The elimination of the sleeve reduces the fuel temperature of as much as 100°C. Since the fuel rods are not supported by the sleeve, the clearance between fuel rods and inner walls of coolant path is adjusted by some graphite spacers.

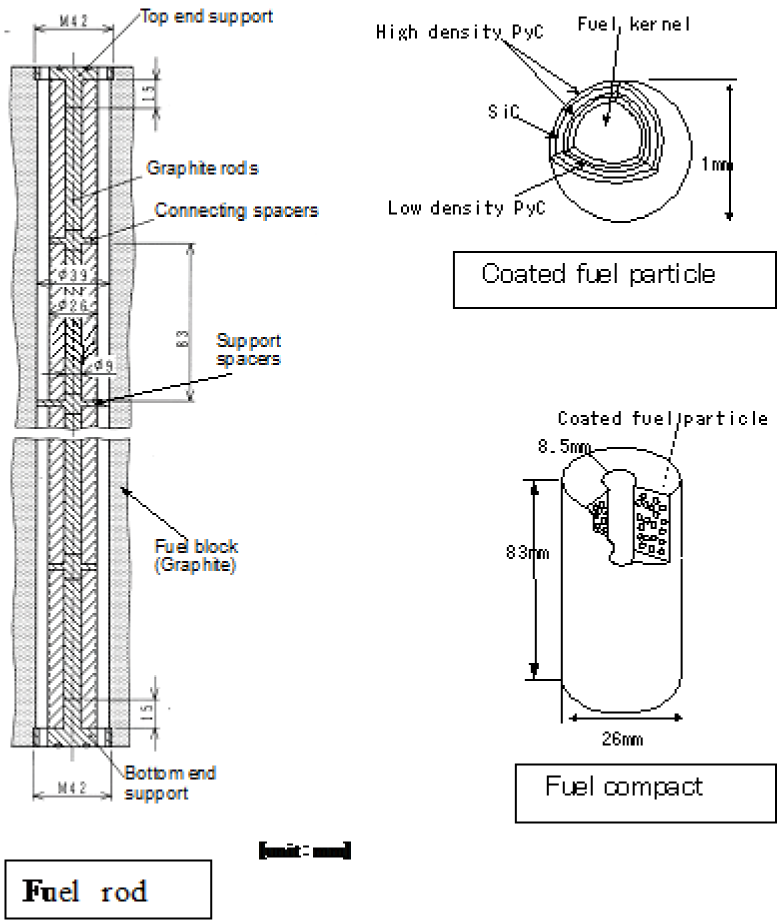

Figure 2: Pin-in-block fuel assembly design

To prevent the corrosion by the exposure to the primary helium containing impurity gases, the graphite layer of 1 mm thickness coats the surface of fuel compacts. Each fuel compact contains TRISO coated fuel particles (CFPs) with uranium enrichment of 14%, which are embedded in graphite matrix at the packing fraction of 29%. The TRISO coating consists of a low-density porous pyrolytic carbon (PyC) buffer layer adjacent to the spherical fuel kernel, an isotropic PyC (inner PyC : IPyC) layer, a silicon carbide (SiC) layer and an outermost PyC (outer PyC : OPyC) layer. The thickness of each layer is modified from that of the HTTR fuel particle so that the fuel particle can retain fission products during the operation with the average burnup of 120GWd/ton and the maximum 155GWd/ton; from 60μm to 140μm for PyC buffer layer, from 30μm to 25μm for IPyC layer, from 25μm to 40μm for SiC layer, from 45μm to 25μm for OPyC layer.

2.1.2. Evaluation of fuel integrity

The failure mechanism of the coated fuel particle had been investigated in JAERI for over 20 years. The study proved that following three causes were the most dominant for the fuel failure.

1. Failure by the internal pressure in the coated fuel particle.
2. Failure by the interaction of Pd, one of fission products, with the SiC layer.
3. Failure by the kernel migration inside of the coated fuel particle.

The integrity of the coated fuel particle was evaluated by a methodology developed for the HTTR fuel

As the burnup proceeds, the tensile stress acted on coating layers increases by the accumulation of fission gases and CO/CO_2 gases. When the tensile stress of the SiC layer caused by fission gases and CO/CO_2 gases exceeds the stress limit, the SiC layer would fail. The FIGHT code was used for calculating the stress of each layer and the failure probability based on a Weibull distribution. In this calculation, the initial fraction of through coating failure and the initial fraction of SiC coating failure were assumed to be 2.5×10^{-6} and 8×10^{-5}, respectively. They were determined based on the initial failure fraction measured for the HTTR fuel. The average fraction of through coating failure was evaluated to be 8.3×10^{-5} for the discharged fuel, which is as much as the total initial failure fraction of 8.25×10^{-5}. It proves that all coated fuel particles with SiC coating failure change to coated fuel particles with through coating failure and coated fuel particles without flaws at the initial condition still keep their integrities. This has to be confirmed by irradiation tests which simulate the high burnup condition.

Pd reacts with the SiC layer and damages the SiC layer. In the evaluation of the fuel failure caused by this chemical reaction, the penetration depth of the SiC layer was evaluated as a parameter of the amount of Pd released from the kernel. The amount of the released Pd was calculated by using the model of Pd diffusion in UO_2 kernels, depending on irradiation time, temperature and burnup. The maximum corrosion depth was evaluated to be 13μm which is far less than the thickness of the SiC layer of 40μm.

The kernel migration inside the coated fuel particle had been investigated in a capsule irradiation and post-irradiation examinations (Sawa, 1996). According to a proposed empirical equation, the maximum length of the kernel migration was 102μm which is less than the thickness of the buffer layer of 140μm.

The fuel integrity during the operation was confirmed in the same approach as the HTTR fuel design.

In a depressurization accident, the maximum fuel temperature was kept lower than 1600°C. The fraction of fuel failures additionally generated at this temperature was estimated to be negligibly low compared with initial failure fraction of fuels.

However, further irradiation tests in high burnup condition for the GTHTR300 fuel integrity are necessary to obtain the safety licensing.

2.2 Reactor core description

The active core of the GTHTR300 consists of 90 fuel columns in an annular ring, shown in Figure 3 and it is about 5.5m in outer diameter, about 3.6m in inner diameter, and about 8m in height. Inner and outer reflector regions consist of 73 columns and 48 columns (including control rod columns), respectively. Fixed reflector region surrounds these regions. A fuel column consists of 8 layers of hexagonal fuel blocks and 2 layers of graphite blocks placed as top and bottom reflectors. Each fuel block, 0.41m across flats and about 1m in height, has 57 fuel rods in its 57 coolant holes. Dowel pins and sockets for fixing fuel blocks are arranged respectively at the top and the bottom of three corners in the block. Burnable poisons (BPs) are inserted in the holes under three dowel pins. Coolant helium gas flows downward in annular spaces around fuel rods and removes the heat from fuel rods. Major specifications of the reactor core design are shown in Table 3.

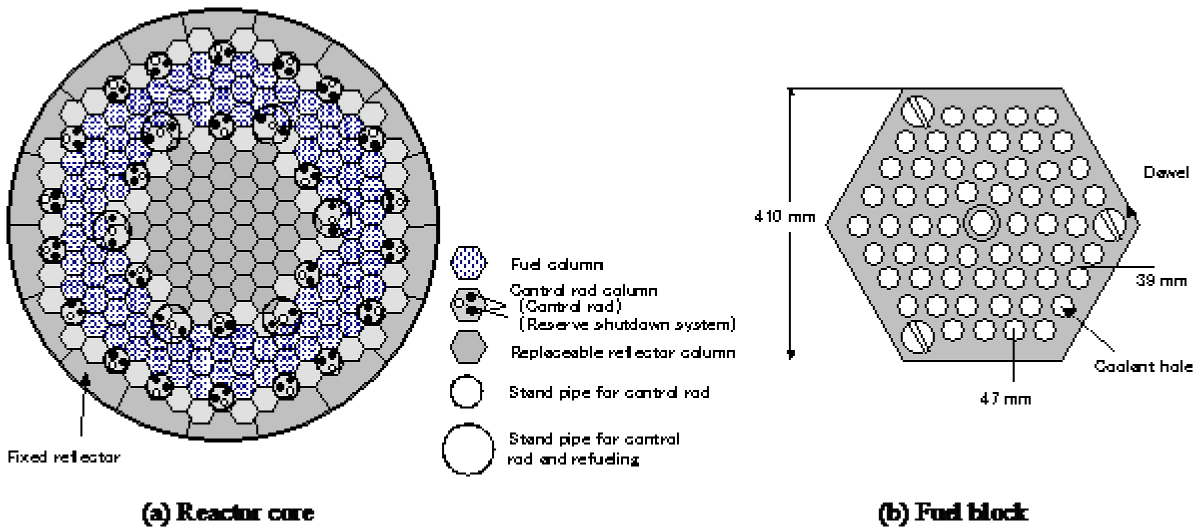

Figure 3 Horizontal view of GTHTR300C reactor core and fuel element

The GTHTR300C core is refueled by a "sandwich shuffling" refueling method shown in Figure 4. One out of every two axial fuel blocks is discharged from the core every two years, and the remaining fuel blocks are shifted to one-block downward. Newly charged fuel blocks are loaded on each remaining fuel block. "Sandwich shuffling" is named because the remaining fuel block is placed between newly charged fuel blocks like a sandwich.

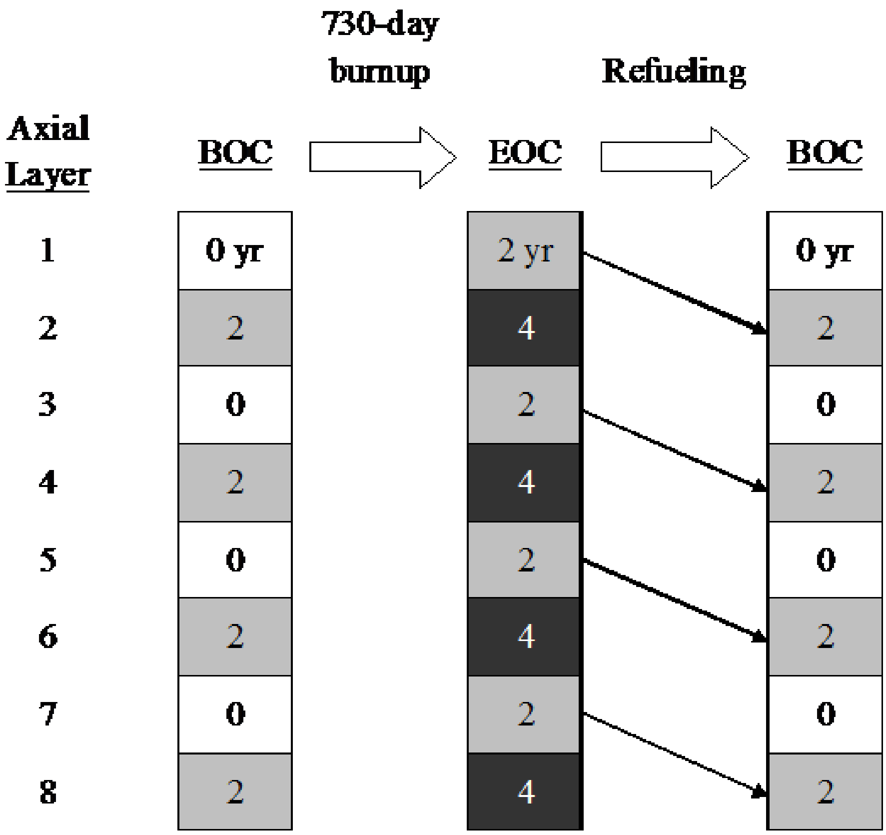

Figure 4: GTHTR300C refuelling by "sandwich shuffling" of fuel blocks in a fuel column

Table 3 GTHTR300C Reactor Design Optimization Results

	GTHTR300C - power plant - baseline design 850°C	GTHTR300C - power plant - growth design 950°C	GTHTR300C - H_2 cogen. plant - Pin-in-block 950°C
Reactor rating	600MWt	<— same	
Core design	prismatic annular core	<— same	
Fuel particle	TRISO UO_2	<— same	
Fuel block design	pin-in-block	<— same	
Reactor pressure vessel steel	SA533/508	<— same	
Reference fuel cycle	LEU	<— same	
Refueling method	2-batch axial shuffling	<— same	
Core physics design			
Fuel columns	90	<— same	
Inner reflector columns	73	<— same	
Outer reflector columns	48	<— same	
Core height (m)	8.4	<— same	
Average power density (W/cm^3)	5.4	<— same	
Average burnup (GWd/t)	120	<— same	
Fuel block height/across flat (mm)	1050/410	<— same	
Fuel rods per block	57	<— same	
Fuel rod diameter (cm)	2.6	<— same	
Core enrichment count	8	7	<— same
Average Enrichment (%)	14.3	14.5	<— same
Burnable poison (BP) count	6	5	<— same
BP pin diameter (mm)	4.8	3.6	<— same
Refueling interval (months)	24	18	<— same
Max. fuel power peak factor	1.16	1.12	<— same
Core thermal hydraulic design			
Coolant (helium) flow rate (kg/s)	439	403	322
Coolant inlet temperature (°C)	587	663	594
Coolant outlet temperature (°C)	850	950	950
Coolant inlet pressure (MPa)	7.0	6.4	5.1
Core coolant pressure drop (kPa)	58	50	35
Min. fuel coolant flow fraction	0.82	0.82	0.82
Max fuel temperature (nominal) (°C)	1108	1206	1244
Max fuel temperature (LOCA) (°C)	1546	1564	1535

2.2.1 Core neutronic design

The GTHTR300C neutronic design was conducted by the same procedure developed and proven for the HTTR core design shown in Figure 5. The DELIGHT code is a one-dimensional lattice burnup cell calculation code to provide group constants of fuel blocks,

reflector blocks, etc. The TWOTRAN-2 code is a transport code to provide detailed flux distributions of control rod regions where the neutron flux distribution changes largely. The CITATION code is a reactor core analysis code based on diffusion theory. This procedure and the analytical codes were already verified.

In the core analysis, effective averaged 6-group macroscopic cross sections were calculated in each burnup region from each macroscopic cross section for fuel, reflector and CR provided by the DELIGHT code, control rod shielding factors provided by the TWOTRAN-2 code and other correction factors. With these 6-group macroscopic cross sections, a spatial power distribution was calculated by the CITATION code with a three-dimensional 1/6-core model. The calculated spatial power distribution was used as an input for the next burnup step.

From the core analysis, it became clear that the excess reactivity had to be compensated only by the burnup of burnable poisons until the middle of the operation cycle so that an operation cycle of 2 years (730days) / batch was achieved. And despite of the initial high uranium enrichment of 14%, the residual uranium enrichment is 4.42% that is lower than the initial uranium enrichment of 5% of light water reactors.

Since the inner control rods with large reactivity are not almost inserted into fuel regions during the operation, the worth of each inner CR is as much as that of each outer CR. When one pair of CRs is withdrawn during the operation, the CR worth is 0.15%Δk/k at maximum. Even if the calculation uncertainties are considered, it is within 0.2%Δk/k which is sufficiently smaller than 0.5%Δk/k of the GTHTR300 safety requirement. The average burnup reaches the target value of 120GWd/ton. The maximum burnup is 155GWd/ton.

Figure 6 shows the axial power distribution change during the operation cycle. The maximum power density is limited to less than 13 W/cm^3 through the operation cycle except its last period. Due to the low peak power density, the maximum fuel temperature is lower than 1400°C as to be discussed in the following thermal hydraulic design.

Regarding the shut down margin, to evaluate it with less calculation uncertainty, supplemental analyses were carried out using the three-dimensional full-core model of the CITATION code and the detailed model of the continuous-energy Monte Carlo code MVP.

The control rod shutdown margin in the cold condition (300K) with one pair of control rods fully withdrawn is 1.4%Δ k/k and it is confirmed that the sub-criticality in the cold condition can be maintained only with control rods. The control rod shutdown margin in the operation condition is 5.8%Δk/k that is large sufficiently. The controllable reactivity in the operation condition is smaller than that in the cold condition because the core temperature is high.

2.2.2 Thermal hydraulic design

The thermal hydraulic design was carried out by using the same code system developed for the HTTR design. As shown in Fig. 8, the power distribution of the core and the neutron fluence distribution were calculated by the neutronic code system and used as inputs for the thermal hydraulic code system. The coolant flow rate and the pressure in fuel blocks were evaluated by the flow network analysis code FLOWNET. The temperature of fuel rods and fuel blocks adjacent to a coolant path was calculated by the fuel temperature analysis code TEMDIM.

The FLOWNET model includes all flow paths affecting the fuel temperature, such as bypass flows between fuel blocks, horizontal cross flows between stacked fuel blocks, leakage flows between permanent reflector blocks which bypass the core region, and the other leak flows around core bottom and upper structures. The deformation of fuel blocks due to neutron fluence is also considered in this model. The effective coolant flowrate in fuel bocks is used as an input for the TEMDIM code. Hot channel factors were considered in the maximum fuel temperature calculated by the TEMDIM code. The hot channel factors consist of systematic subfactors such as uncertainties on reactor thermal power, coolant flow rate, core inlet coolant temperature, etc. and random subfactors such as manufacturing tolerance, uncertainties on physical properties, etc.

Figure 7 shows the maximum fuel temperature during the operation cycle calculated by the TEMDIM code. The maximum fuel temperature peak is 1398°C at the beginning of the full power operation. Form the point of dose evaluation, amount of fission products released from fuels into the coolant system at this temperature is negligibly low during the normal operation. Newly generated fuel failure at this temperature is also negligibly low in comparison with the initial fuel failure as described in Section 2.1.2.

2.3 Final reactor designs

The preliminary reactor core design for the 850^{o}C reactor outlet coolant temperature as described above uses uniform fuel enrichment of 14%. Final reactor core design is optimized by using multiple enrichment zones with approximately same average enrichment. The optimization is intended to minimize local power peaking factors and thus peak fuel operating temperature. The results of three optimized core designs are summarized in Table 3. The maximum fuel temperature for the 850^{o}C core design is reduced to 1108^{o}C nominal value and 1216^{o}C conservative value that considers all system random and uncertainty factors. The same optimization approach is used to finalize two core designs for 950^{o}C core outlet coolant as listed in Table 3. The two 950^{o}C designs select core inlet temperatures and coolant pressures that are optimum for power generation and hydrogen cogeneration, respectively. As intended, the 950^{o}C reactor core designs share a great deal of common design features with the baseline 850^{o}C design. The major differences are the corresponding higher fuel operating temperature and a shortened refueling interval from 24 to 18 months.

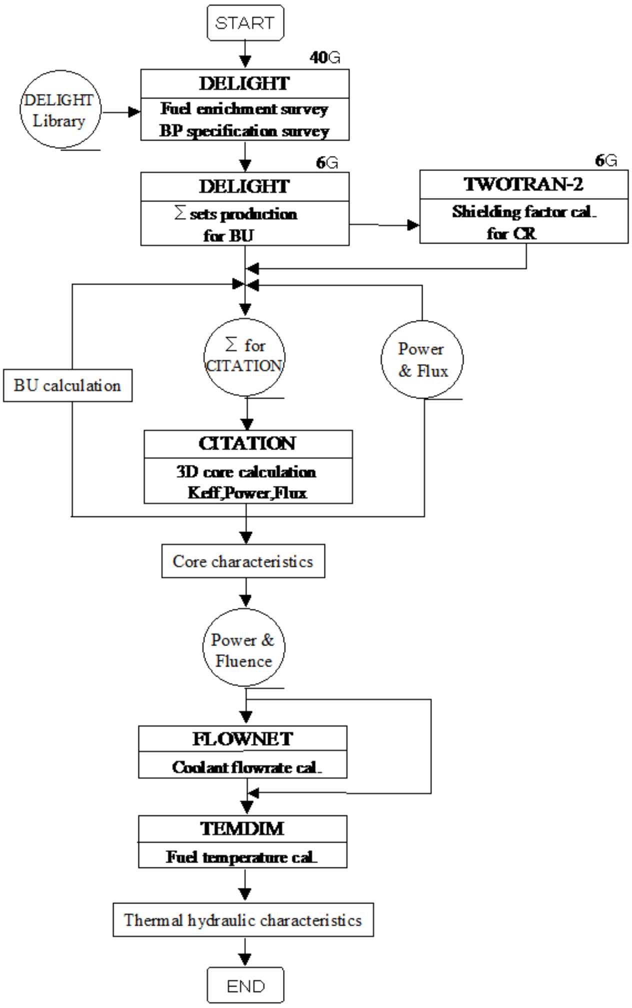

Figure 5: HTTR-proven core design process used in the GTHTR300C

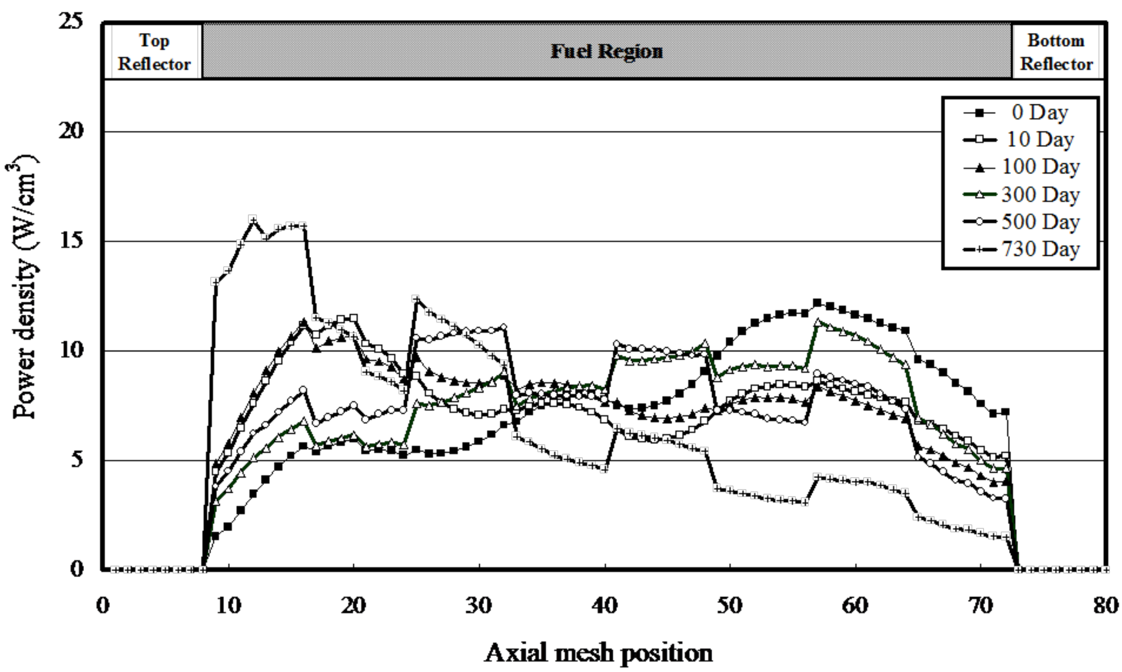

Figure 6: Change of axial power distribution with burnup

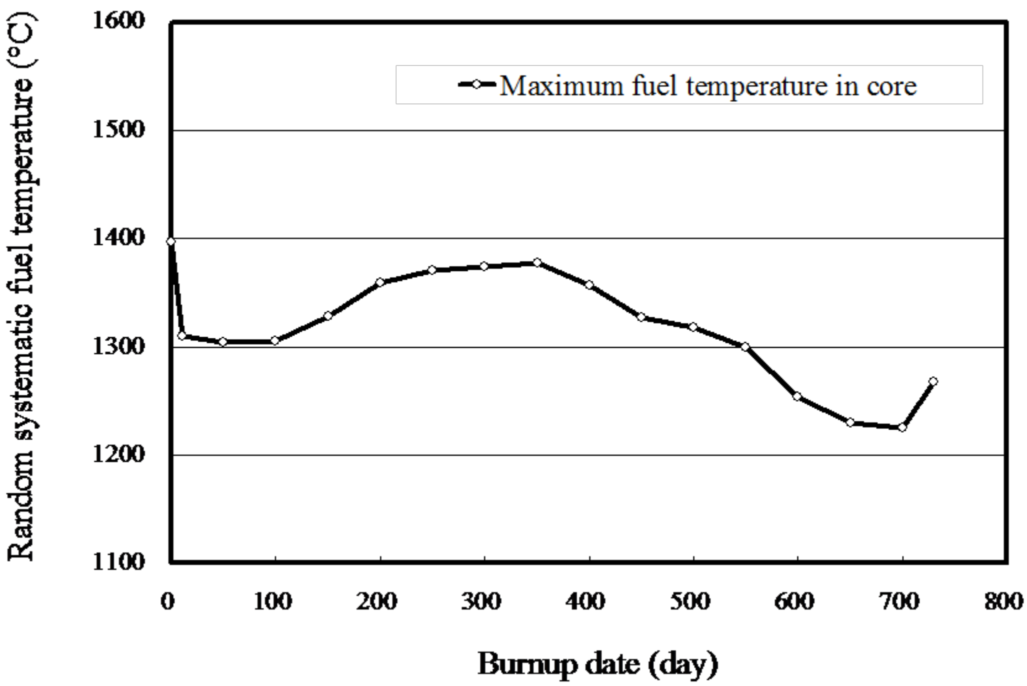

Figure 7: Maximum fuel temperature with burnup

Description of safety concept

Unique and inherent safety characteristics of HTGRs, such as durability of the reactor core in high temperature and slow transient during abnormal conditions, allow simplification of safety systems. For example, the residual heat of reactor core can be removed safely by passive cooling systems only. This point leads a large cost benefit for construction and operation of the plant without any deterioration of the safety. The major safety design features are described in this section.

3.1 Reactor building

A containment vessel is not necessary in the GTHTR300 because it is designed as severe-accident-free, that is, no large amount of fission product (FP) release from fuel in any postulated accident. However, the reactor building is designed to release the mixture of helium gas and air through pressure release stacks into the atmosphere in the depressurized accident, subsequently to avoid large amount of air flowing in through the stacks by closing closures. The reactor building is designed to be a confinement with limited leak rate in order to reduce the amount of air ingress into the reactor core. This is to prevent excessive oxidation of fuels after the depressurization accident which is induced by a large break of the helium coolant boundary.

The reactor building which is a reinforced concrete building with compartment structure has been designed to be intact against any pressure attack in the depressurization accident, in essence , kinetic blow-out pressure and static pressure increase in the inside of the building by helium gas release from primary coolant boundary. Since no apparent additional fuel particle failure occurs during the accident and failure fraction of the fuel particle during normal operation is very small, the release of the helium gas into the environment during the initial phase of the depressurization accident can be allowed. Therefore the confinement function of the reactor building is intact during the depressurization accident.

3.2. Vessel cooling system

A vessel cooling system is a residual heat removal system for the complete loss of forced cooling in the depressurization accident. The vessel cooling system was designed as a passive heat removal system only by the natural circulation of air in the cooling panels installed at the outside of the Reactor Pressure Vessel (RPV), as shown in Figure 8. Residual heat in the reactor core is transferred to the cooling panels by radiation from outer surface of the RPV and by natural convection of air in the cavity between the RPV and cooling panels. The vessel cooling system has been designed to keep the temperatures of the fuel and the RPV lower than temperature limitation by passively removing the heat radiated from the RPV during the accidents. The vessel cooling system is a fundamental safety function and the characteristic of heat removal by the vessel cooling system will be confirmed by the safety demonstration tests in the HTTR, under simulating conditions of loss of forced flow.

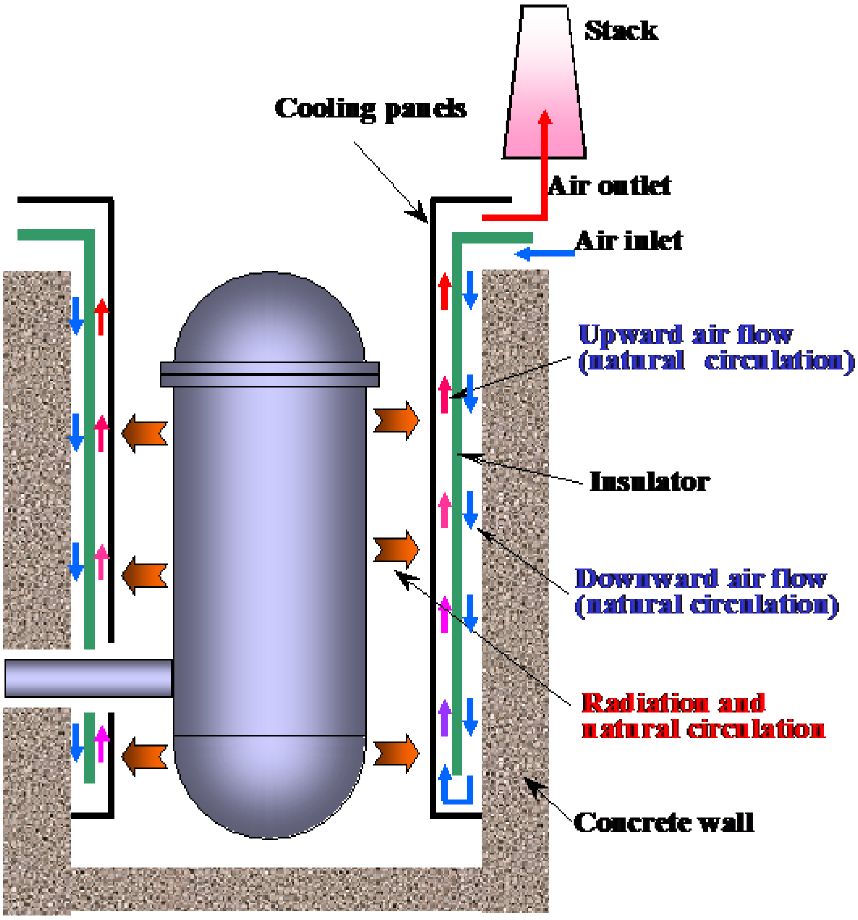

Figure 8: GTHTR300C reactor decay heat passive removal approach

3.3. Reactor shutdown system

A reactor shutdown system consists of two diverse and independent systems; a control rod system and a reserve shutdown system. The control rod system has enough reactivity worth to shutdown the reactor from operation condition to cold subcriticality condition with “one rod stuck margin”. The RSS has the reactivity worth to make the subcriticality condition in high temperature from the operation condition.

The control rods are driven and inserted by control rod drive mechanism into the channels of control rod blocks which are placed at the graphite reflector region in the inside and outside of the annular fuel region. Control rods can be dropped into the channels by gravity in case of the emergency shutdown. In the reserve shutdown system, cylindrical B_4C/C pellets are dropped into holes at the same reflector column as the control rod channel.

In the point of view for reactivity initiated events, the sum of reactivity worth of control rods which are ejected at the same time by one stand pipe break should be lower than 0.5%$\Delta k/k$ in order to prevent fuel failure by temperature increase (Kunitomi, 1998). This value has been attained by arrangement of control rods and by optimizing the operation method of control rods.

3.4 Coupling hydrogen plant to reactor

A hydrogen production plant connected to the HTGR is designed for construction, operation and maintenance as a general (non-nuclear) chemical plant, which has two design objectives of minimizing the production cost of the hydrogen plant and making the safety protection of the of the reactor plant independent of the hydrogen plant. As a result, the safety design of the test reactor HTTR and the commercial plant GTHTR300C assumes no functional role, such as a heat load, on the part of the hydrogen plant.

In addition, the hydrogen plant should be cited close to the reactor building to shorten the high temperature heat transport piping whose specific cost is high. The safe separation distance between the hydrogen production plant and the reactor building is decided based on the safety assessment of external events originated in the hydrogen production plant.

Hydrogen gas released from the hydrogen plant disperses and would then explode if an ignition source exists. It is important that the ignition timing be considered during the diffusion of the leaked hydrogen gas cloud because the explosive hydrogen gas cloud could move and close in the nuclear plant prior to ignition. A separation distance of over 150 m is estimated in case of the hydrogen release of 96.8 kg or 1000 Nm^3. The economics of system layout dictates that the separation distance be kept less than 100 m. There are several means to reduce the separation distance as follows.

- To reduce potential mass of hydrogen release
- To reduce moving distance by diffusion
- To reduce overpressure of hydrogen explosion

To reduce the potential mass of hydrogen release, it is required to detect an abnormal leakage speedily and to shut a hydrogen pipeline down. However the mass flow rate is 19.4 kg/s in case of pipe diameter of 100 mm and hydrogen pressure of 4 MPa. About 100 kg of hydrogen gas can be released in 5 seconds. It seems difficult to reduce the mass of hydrogen release under 100 kg.

To reduce the moving distance of the hydrogen cloud by diffusion, it is found that a protection wall is very performance and cost effective and that the moving distance can be reduced to less than 50 m by means of the adequate blast proof protection wall. For example, it is found that the wall of 2 m higher than the level of a potential leaking hydrogen pipe is enough to reduce to a half of the moving distance without a wall.

To reduce the overpressure of the hydrogen explosion, the presence of packed obstacles, confinement structure and strong ignition, which induces strong blast overpressure, must be avoided.

In sum, the best means to meet the design requirement of a separation distance within 100 m in the GTHTR300C is as follows.

- To erect a protection wall near hydrogen plant to prevent hydrogen dispersion by diffusion.
- To clear any packed obstacles in the separating space between hydrogen and reactor plants.

Description of turbine-generator systems

As shown in Fig. 9, the baseline design of helium gas turbine is a single-shaft, axial-flow design having six turbine stages and twenty non-intercooled compressor stages. The gas turbine rated at 300MWe and 3,600 rpm drives a synchronous generator from shaft cold end by a diaphragm coupling. The machine is placed horizontally to minimize bearing loads. These design features have been chosen in consistence with the established industrial practice in combustion gas turbines. The new gas turbine elements incorporated in the baseline unit are the narrow compressor flowpath, which is the result of working in helium, and the use of rotor magnetic bearings (MB) to avoid large pressure boundary penetration or potential lubricant contamination to reactor system. The development and test programs have been carried out to validate the new technology components uniquely present in this application.

Shown in Fig. 13 is a model test compressor consisting of four axial stages in one third dimensional scale of the full size compressor stages. The test compressor was modeled after the aerodynamic features, including alternative sets of airfoils, under design consideration for the GTHTR300 baseline gas turbine compressor. It was put in a dedicated helium loop for aerodynamic development testing. The data obtained in test are concerned with aerodynamic losses particularly near end walls and growth through multiple rotating blade rows, surge predictability, clearance loss and inlet and outlet performance effects, all to be correlated closely to the full-scale design conditions.

The multi-year compressor development and test program has been concluded including test of an one-third scale compressor shown in Figure 10. The program has achieved the intended goals of exploring basic helium compressor aerodynamics, relative to those of air compressors, and establishing the analytical tools qualified to design and evaluate the full scale compressor. With the qualified tools, the full scale compressor is predicated to over-achieve the design target of 90.5% flange-to-flange polytropic efficiency at design flow and surge margin. The level of performance matches those found in modern air gas turbine compressors. The helium compressor aerodynamics has been advanced ready for prototype demonstration.

A magnetic bearing development and test program is focused on evaluating optimal rotor-bearing clearance control method and developing magnetic bearing control algorithm to operate rotor above the 2nd bending critical speed. A test rig has been constructed and is presently undergoing commissioning. As shown in Fig. 11, the test rig is a one-third scale mockup for the generator rotor of the GTHTR300 and has further built-in capability to test the multi-span and multi-bearing rotor configuration modeled after the GTHTR300 turbine-generator drive train. Existing and new analytical techniques of rotordynamics and control will be test calibrated.

The baseline helium gas turbine design with its component development described so far is the unit in use in the power plant design variant, the GTHTR300. For the units used in other plant variants, geometric scaling from the baseline design has been applied to achieve design and technology simplification in accordance with the SECO design philosophy.

The scaling method is based on the principle that one can increase or decrease system pressure and alternatively or simultaneously increase or decrease the rotor diameter while holding the speed constant to produce aerodynamically and mechanically similar gas turbines of larger or smaller unit capacity. The complex blade airfoils, such as those obtained in the helium compressor development, become simply scalable from one machine to the other and the resulting aerodynamic working conditions and efficiencies are unchanged. The centrifugal stresses remain also unchanged in discs and blades. This makes the technologies developed for the baseline unit also applicable in other units.

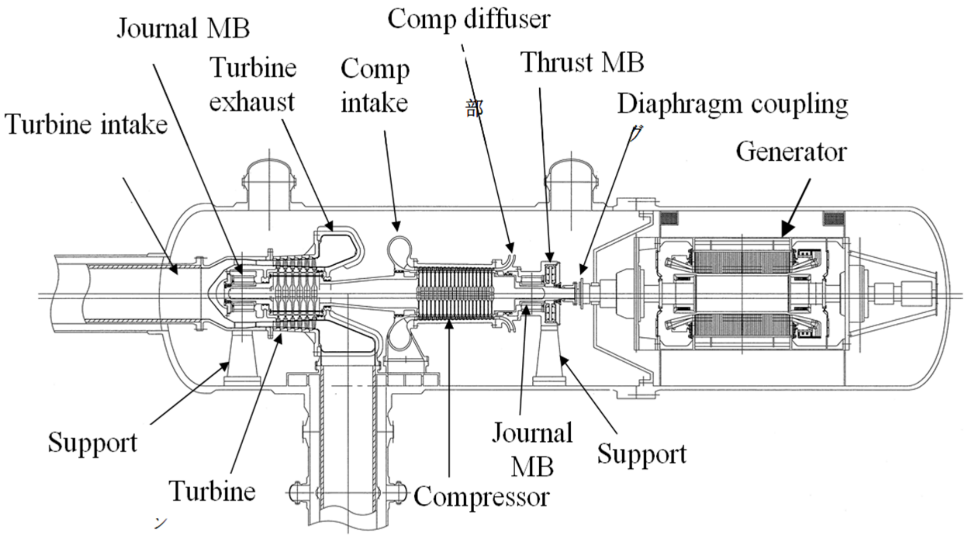

Fig 9. Baseline design of GTHTR300 horizontal helium gas turbine in pressure vessel

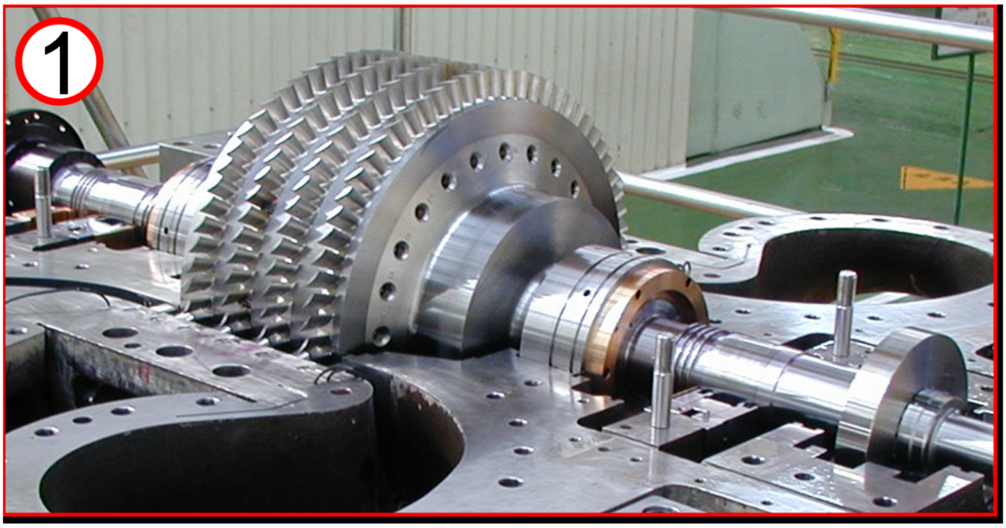

Fig 10. Test compressor of 1/3 of the GTHTR300C full scale

Fig 11. MB rotor test rig of 1/3 of the GTHTR300C full scale

Spent fuel and waste management

Storage options of spent fuels include pool storage, cask storage, vault storage and silo storage. The cask storage used for LWR spent fuels is selected for the estimate of the storage unit cost of GTHTR300C spent fuels. Starting the reference cost for the LWR spent fuel, the storage unit cost is assumed to be proportional to the heat rate and inversely proportional to specific uranium contents of spent fuel. The storage cost of the GTHTR300 spent fuels is conservatively determined to be about 1,467 US$/kg-HM, based on the trial cost estimation by the Federation of Electric Power Companies (FEPC) and on the comparison of costs between methods for water pool storage by Central Research Institute of Electric Power Industry of Japan.

The radioactive waste generated in the GTHTR300 fuel cycle includes high-level waste (HLW), radioactive waste containing transuranic elements (TRU) and irradiated graphite waste. The amount of HLW is 2.5 times that of the LWR, because the burn-up of the GTHTR300 is about 120GWd/t which is more than twice as much as that of the

LWR. The unit cost of about 5,183 US$/kg-HM for storage, transport and disposal of the HLW is assumed, by multiplying 2.5 times that of the LWR's. The 1,792 US$/kg-HM unit cost for storage and disposal of TRU waste is similarly estimated, proportional to fuel burnup, from the LWR cost.

When the half core of the GTHTR300C fuel is exchanged every two years, the half of replaceable reflector blocks and control rod blocks close to fuel regions are also exchanged. The graphite waste is generated from fuel blocks, fuel compact matrixes and coating layers of particle fuels, reflector blocks and control rod blocks. The disposal method of graphite waste is studied newly because no graphite waste is generated in the LWRs. In this estimate, the graphite waste is categorized into low-level waste and shallow land disposal with concrete pits is selected. The unit cost of the shallow land disposal with concrete pits is estimated to be 25,000 US$/m3 (transport unit cost of 8,333 US$/m3 + disposal unit cost of 16,666 US$/m3) based on the decommissioning cost estimate for reprocessing facility by the FEPC. The disposal unit cost of graphite waste converted from the unit cost of the shallow land disposal by using the amount of graphite waste and spent fuel is about 1,233 US$/kg-HM.

The decommissioning cost consists of dismantlement cost of facilities and disposal cost of waste materials. Allowance for dismantling a nuclear power plant is saved every year over plant operation life. The decommissioning starts after seven years of cooling after the end of operation. It is necessary to determine the radioactivity of equipment to be dismantled, including high-level waste, low-level waste, and extreme low-level waste. The dismantlement cost is estimated based on the amount of the waste generated the each group of radioactivity. The estimate method is based on the estimated decommissioning cost of a reprocessing facility reported by the FEPC. The estimate items include personnel cost, equipment cost, and management cost in the process of decontamination of systems, of dismantlement of facilities and buildings, and of sealing the waste into drum cans or vessels. The dismantlement cost is estimated by piling up each cost, considering dismantlement method, equipment, man-days and pay rate for personnel according to the level of contamination and dose rate of equipment. The disposal cost is the sum of transport cost and disposal cost of the waste. The total decommission cost is estimated to be US0.15¢/kWh

Plant performance

The GTHTR300C plant process as shown in Fig. 12 consists of a 600MWt nuclear reactor, a direct cycle gas turbine for power conversion, and a thermochemical iodine sulfur (IS) process for hydrogen production. The gas turbine circulates reactor coolant while driving electric generator. The cogeneration cycle includes an intermediate heat exchanger (IHX) in serial between the reactor and the gas turbine. The particular serial arrangement creates a logarithmic mean temperature difference as large as 157^{o}C between the primary and secondary sides of the IHX, making a compact-size IHX possible to minimize its construction cost. The 950^{o}C operating helical tube IHX built into the HTTR is the reference design for the GTHTR300C. The effective tube bundle of the 170 MWt IHX is sized 4.5 meter in diameter and about 3 meter in length. A 390 MWt IHX has also been designed. The closed intermediate loop circulates hot helium from IHX to the distant hydrogen plant and completes necessary environmental and material separation between the nuclear and chemical plants.

The helium gas turbine is a single-shaft, axial-flow design having six turbine stages and twenty compressor stages. The gas turbine is rated at 280 MWe and 3,600 rpm and drives a synchronous generator from shaft cold end by a diaphragm coupling. The machine is placed horizontally to minimize bearing loads. These design features are chosen in consistence with the established industrial practice in combustion gas turbines.

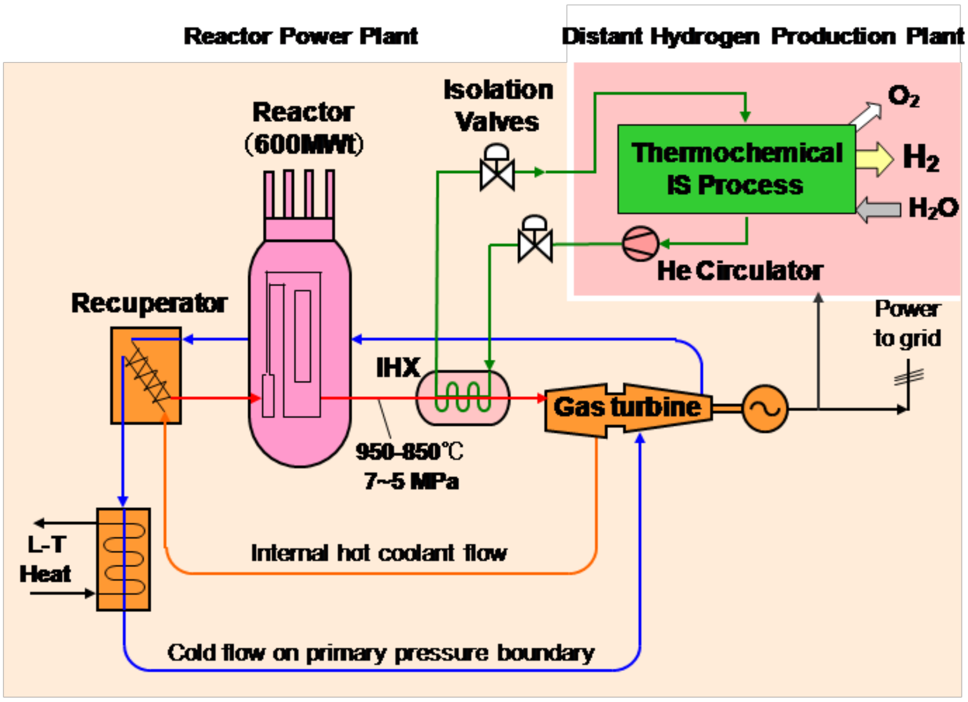

Figure 12: Schematic of GTHTR300C plant cogeneration process

The IS process involves three inter-cyclic thermo-chemical reactions for the decomposition of water molecules into hydrogen and oxygen gas products. The process consumes water as the only material feedstock and all other process materials used are chemical reagents. The process requires energy input of major heat and minor electricity with both supplied in house from the reactor plant. The heat supplied in form of hot helium gas is used to support the endothermic decomposition reactions and the electric energy is used to power process electrolyzers, gas circulators, pumps and other utilities.

Of the several nuclear hydrogen production options, JAEA selects the IS process as the basis of commercial nuclear hydrogen production in the consideration of the following:

1. The IS process consists simply of three essential chemical reactions, and the high temperature of the process is the basis for high thermal efficiency, which is estimated in the range of 40-50%.
2. The IS process is essentially a thermal fluid process that permits continuous operation and which scales with volume rather than areas, offering the incentive of economy of scale for large-scale hydrogen production from nuclear energy

6.1 Rated electricity and hydrogen cogeneration

In this rated mode of electricity and hydrogen cogeneration, the reactor outlet coolant of 950^{o}C enters the primary side of the IHX and heats the secondary helium to 900^{o}C. About 170 MWt of heat is transferred in the IHX and carried by the intermediate loop to the hydrogen process, which produces 0.64 million Nm3 (58 tonnes/day) hydrogen per day. The balance of the reactor thermal power is converted to power in the direct cycle gas turbine plant, in which the helium gas of 850^{o}C exiting the primary side of the IHX enters the gas turbine to convert the heat to electric power. Gross electricity of 202MWe is generated, about 12% of which is used in-house to supply the reactor plant consumption and the hydrogen plant uses. The net power output to the grid is 178 MWe.

6.2 Standalone electricity or hydrogen generation

When power generation, rather than cogeneration, is desired, the heat rate of the hydrogen plant is reduced to zero by stopping the intermediate loop flow circulation and taking the hydrogen plant off-line. Simultaneously the reactor power is reduced to 430 MWt and reactor outlet coolant temperature to 850^{o}C by using reactor control rods. The rate of reactor temperature change is limited, saying to 15°C/hr, to avoid thermal stress in reactor structure. Completing the above transient yields the standalone mode of electricity generation of 196 MWe net, which may be increased to 276 MWe by increasing helium inventory and the coolant pressure from 5 to 7 MPa while keeping reactor outlet coolant temperature at 850°C and returning the reactor power to 600 MWt, with the use of reactor control rods.

On the other hand, if only hydrogen production is desired, the power generation is reduced only to supply house loads of 25 MWe by using flow bypass valves. Additionally this power generation reduction can be assisted by the helium inventory reduction in the primary system circuit. The reactor outlet coolant temperature is maintained at 950°C by using reactor control rods, while turbine inlet temperature of 850°C is unchanged. The end of the transient yields the steady standalone production of 0.64 million Nm3/day hydrogen.

6.3 Plant economics

Figure 13 shows the power generation cost of the GTHTR300C baseline design, as predicted by the design organization, that employs the 600MWt reactor with 850°C reactor outlet coolant temperature for power generation only. The power generation costs are estimated to be US 3.5 cents/kWh under 80% load factor and US 3.2 cents/kWh at 90% load factor. The GTHTR300C baseline is designed to achieve 90% load factor. The power generation cost is almost 1 cent/kWh lower than that of the LWR of 1,300 MWe estimated by the FEPC. The results demonstrate the economical potential of the GTHTR300C. The GTHTGR300C with 950°C reactor outlet temperature is expected to improve the cost further because the higher temperature results in a significantly higher efficiency as given in Table 1.

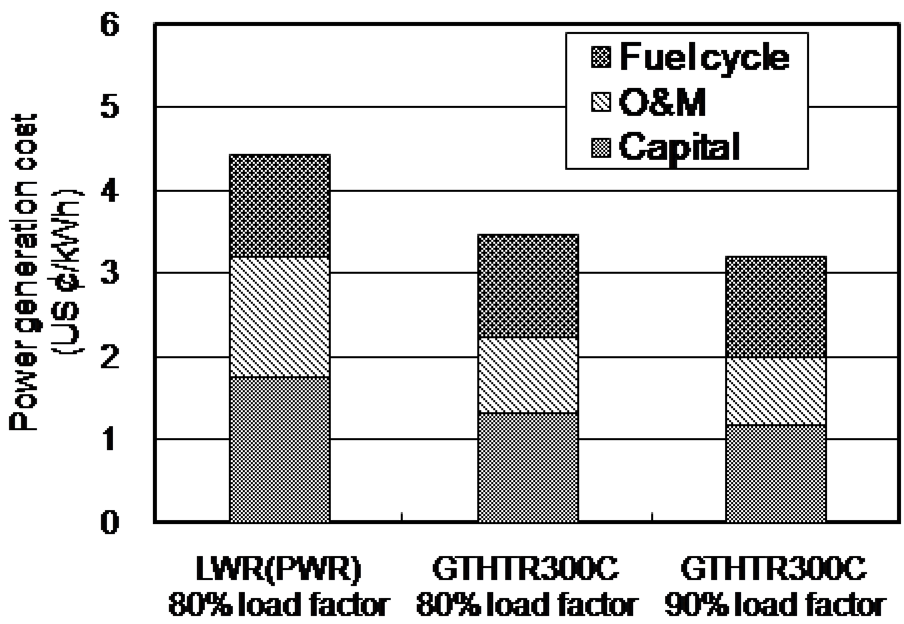

Figure 13: Comparative generation costs based on utility cost evaluation method in Japan

Development status of technologies relevant to the NPP

By sharing technologies of the HTTR and GTHTR300C, additional development needs for the GTHTR300C are reduced and the limited needs are being met in the ongoing development activities in JAEA.

The reactor technology for this system is based on the technology successfully developed and demonstrated in the HTTR. Additional irradiation performance data for the commercial fuel with 120MWd/t burnup, to extended from the current fuel of 90 GWd/t, is necessary to achieve the target plant economics. Research on graphite and carbon composite materials and non-destructive inspection techniques is be performed in order to extend the lifetime of the core components.

Advanced technologies of helium gas compressor and magnetic bearing are the key to this system. These technologies have been developed and validated through one-third scale (30 MWe level) component tests. The integrated gas turbine system demonstration including hot function tests will be needed, which can be achieved through the prototype plant development.

The same design philosophy of the IHX in the HTTR is applied for that in the GTHTR300C. Also, the same material, welding method, structure are used for the IHX. Design conditions of this system such as operating temperature and pressure difference between the primary and secondary helium gas are almost the same as those of the HTTR. For example, the pressure difference between the primary and secondary helium gas is controlled as low as 0.015MPa to keep the creep damage of the heat transfer tube as low as possible. Due to this design philosophy, no significant development is necessary for the GHTR300C-IHX in this system. Existing technologies are available.

The main primary helium gas circulator is not needed in the GTHTR300C because the direct cycle gas turbine circulates the reactor coolant. In the secondary circuit, a helium gas circulator with oil bearings can be used because potential lubricant ingress into the secondary circuit does not damage the reactor system and any problem with the secondary circuit circulator can be easily accessed and serviced.

The hydrogen production system by the IS process method is designed to be a non-nuclear plant for the installation in the third loop. The malfunction of the hydrogen production system does not impair the continuous operation of the reactor to generate power.

The GTHTR300C, which is meant for commercial unit, shall demonstrate its ability to operate in normal cogeneration mode or with electric or hydrogen system operating alone in case of a scheduled or forced shutdown of either system. As described in Section 6, the results of the system performance analysis showed that the reactor could be continuously operated with the above variable load conditions. However, actual demonstration test is necessary for performance confirmation, which is to be carried out in the prototype demonstration plant.

Deployment status and planned schedule

Figure 14 presents the technical roadmap of nuclear hydrogen production development in Japan. JAEA has formulated its "Nuclear Energy Vision 2100" towards a low-carbon society, which presents the technical feasibility and nuclear fuel sustainability for the installation of 120 commercial HTGRs (a total of 72GWt) in Japan, beginning in 2030, for use in the production of hydrogen to meet the projected demand for the nation's transportation, residential, and industry sectors and to contribute to achieve a national CO_2 reduction target of 50% and 90% by the year 2050 and 2100, respectively, from the year 2000 level.

By combining power generation and substantial production of hydrogen in an efficient commercial cogeneration plant, the GTHTR300C will provide cost-competitive, CO_2 emission-free electricity for traditional energy consumption, while meeting a significant new demand for hydrogen as fuel for fuel cell vehicles and as a manufacturing feed material for fertilizer, petrochemical, steelmaking and other industries.

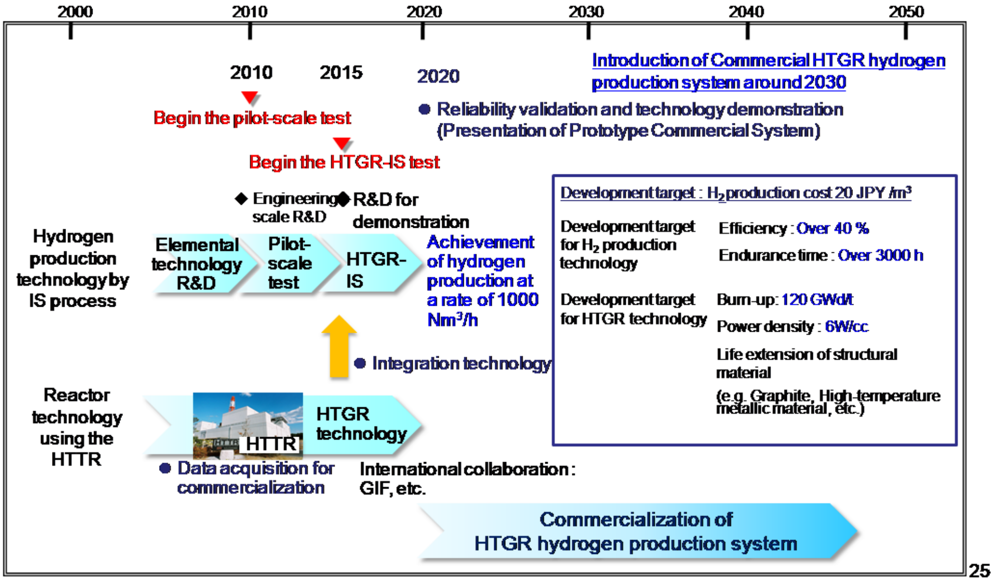

Figure 14. Technical Development Roadmap of nuclear hydrogen production by the Atomic Energy Commission of Japan in July 2008

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Technical data

General plant data

Reactor thermal output	600 MWth
Power plant output, net	274 MWe
Power plant efficiency, net	47 %
Plant design life	60 Years
Plant availability target >	90 %
Primary coolant material	Helium
Moderator material	Graphite
Thermodynamic cycle	Brayton
Type of cycle	Direct
Non-electric applications	H2 production

Reactor core

Active core height	8 m
Fuel column height	1.050 m
Equivalent core diameter	0.410 m
Average core power density	5.4 MW/m^3
Fuel material	UO2 and MOX
Fuel element type	Spherical
Outer diameter of fuel rods	26 mm
Outer diameter of elements	24 mm
Number of fuel assemblies	90

Number of fuel Elements in fuel assemblies	57
Enrichment of reload fuel at equilibrium core	14.3 Weight %
Fuel cycle length	24 Months

Primary coolant system

Primary coolant flow rate	439 Kg/s
Reactor operating pressure	7.0 MPa
Core coolant inlet temperature	587 °C
Core coolant outlet temperature	850 °C

Reactor pressure vessel

Base material	SA508

Fuel channel

Pressure Tube material	Zr 2.5wt% Nb alloy

Residual heat removal systems

Active/passive systems	Passive

Safety injection systems

Active/passive systems	Active and Passive

Turbine

Type of turbines	Single-shaft, axial-flow gas turbine
Turbine speed	3600 rpm