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Research Article

A Small-Sized HTGR System Design for Multiple Heat Applications for Developing Countries

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Japan Atomic Energy Agency has conducted a conceptual design of a 50 MWt small-sized high temperature gas cooled reactor (HTGR) for multiple heat applications, named HTR50S, with the reactor outlet coolant temperature of 750°C and 900°C. It is first-of-a-kind of the commercial plant or a demonstration plant of a small-sized HTGR system to be deployed in developing countries in the 2020s. The design concept of HTR50S is to satisfy the user requirements for multipurpose heat applications such as the district heating and process heat supply based on the steam turbine system and the demonstration of the power generation by helium gas turbine and the hydrogen production using the water splitting iodine-sulfur process, to upgrade its performance compared to that of HTTR without significant R&D utilizing the knowledge obtained by the HTTR design and operation, and to fulfill the high level of safety by utilizing the inherent features of HTGR and a passive decay heat removal system. The evaluation of technical feasibility shows that all design targets were satisfied by the design of each system and the preliminary safety analysis. This paper describes the conceptual design and the preliminary safety analysis of HTR50S.

1. Introduction

Nuclear energy is one of the most promising energy sources to satisfy energy security, environmental protection, and efficient supply. Many developing countries have expressed their interest in deploying nuclear power plants in their own country. After the accident at the Fukushima Daiichi Nuclear Power Station, some developed countries which already installed the nuclear power plant have changed their policy for the nuclear energy. However, many developing countries still show their interest in the nuclear power plants.

Since the small- and medium-sized reactors [1] can reduce capital cost and can provide electric power away from large grid systems, they are suitable for the developing countries. The high temperature gas cooled reactor (HTGR) [2] is one of the small modular reactors and has attractive inherent safety features. It is a helium cooled graphite moderated reactor employing a ceramic coated fuel particle with high temperature capability. It can operate at reactor outlet temperature of about 1,000°C, much higher than conventional light water reactor (LWR). Accordingly, HTGR can be applied to many kinds of heat applications such as hydrogen production, electricity generation by gas turbine and steam turbine, process heat supply, district heating, and sea water desalination [3–6]. HTGR has a superior safety potential as the residual heat of the core can be removed without any active devices or power supplies (i.e., station blackout) [7, 8]. It also has the capability of incinerating the plutonium from the spent fuel of LWR [9].

Since the 1960s, seven HTGRs have been built and operated in the United Kingdom, the United States, Germany, China, and Japan. At present, the HTGR technology

development has been performed worldwide. The Republic of Kazakhstan issued "Development Program of Nuclear industry in the Republic of Kazakhstan for 2011–2014 with the prospect of development until 2020" in June 2011, which includes the construction plan of a Kazakhstan high temperature gas cooled reactor (KHTR). It is planned to start the operation of KHTR, which is a 50 MWt HTGR, in 2020s to develop technological capabilities and related industries by the supply of electricity and process heat from KHTR in the small city and the introduction of the nuclear technology in Kazakhstan. Japan Atomic Energy Agency (JAEA) signed the memorandum of understanding between JAEA and the National Nuclear Center in Kazakhstan (NNC) for the future cooperation in nuclear energy research and development and has conducted the cooperation for the development of KHTR upon the request of the technical support from NNC since 2007.

JAEA constructed and has operated the Japanese first HTGR, named high temperature engineering test reactor (HTTR) [10, 11]. The inherent safety features of HTGR have been demonstrated by the safety tests using HTTR. A 50-day continuous operation with its reactor outlet temperature of about 950°C and thermal power of 30 MW performed in early 2010 showed that the HTTR is a reliable energy system to provide high temperature heat for nonelectric application [12]. In addition to the HTTR operation, JAEA has performed the design of an advanced commercial plant of 600 MWt-class very high temperature reactor (VHTR), the GTHTR300 series [3, 4], for the electricity generation using the gas turbine and hydrogen production using the water splitting iodine-sulfur process (IS process). The GTHTR300 series are the final target of the HTGR development in JAEA.

In order to deploy the HTGR in developing countries at an early date, JAEA has started a conceptual design of a 50 MWt small-sized HTGR for multiple heat applications, named HTR50S, using more conventional and proven technologies than those used in the GTHTR300 series with support of Japanese vendors. This paper describes the conceptual design results and preliminary safety analysis results for HTR50S.

2. Design Philosophy

The design philosophy of the HTR50S is a high advanced reactor, which is reducing the R&D risk based on the HTTR design, upgrading the performance for commercialization by utilizing the knowledge obtained by the HTTR operation and the GTHTR300 design. The balance of plant shall be designed for the heat application of district heating and process heat supply based on the steam turbine system and for the demonstration of the power generation by helium gas turbine and the hydrogen production by the thermochemical water splitting IS process to satisfy the user requirement for multiple heat applications. The HTR50S is to be upgraded in its performance as first-of-a-kind of the commercial plant or a demonstration plant and expanded in its application as the following steps:

(i) the 1st step of phase I: the power generation using a conventional steam turbine at the reactor outlet temperature of 750°C using the performance-proven HTTR fuel to demonstrate the technologies which are improved from HTTR,

(ii) the 2nd step of phase I: using a higher performance fuel to demonstrate the technology for the extension of the burnup by employing shuffling refueling,

(iii) phase II and III: increases of the reactor outlet temperature to 900°C and installing an intermediate heat exchanger (IHX) to demonstrate helium gas turbine and hydrogen production using the IS process.

The philosophy of safety design is the adoption of the defense-in-depth concept and the utilization of the inherent features of HTGR to protect people and the environment from the harmful effect of the radioactive materials, which is the same as the HTTR [13]. On the other hand, the following strategies were applied for the safety design of HR50S to fulfill the high level of safety.

(i) The vessel cooling system (VCS), which is engineered safety features to remove decay heat from the outer surface of the reactor pressure vessel (RPV), is designed as a passive means to utilize natural pressure difference and buoyancy forces.

(ii) The shutdown cooling system (SCS), of which role is to remove decay heat by forced cooling in the core, is designed as not safety class system but nonsafety class system so that the protection is optimized to provide the highest level of safety that can be reasonably achieved.

3. Design

3.1. Major Specification. The major design specification of HTR50S is listed in Table 1. The reactor outlet temperature in phase I was determined as 750°C taking into account that the conventional heat-resistant alloy, Alloy 800H, of which the restrictive temperature is 760°C (1400°F), can be applied to the heat exchanger tube of the steam generator (SG). The reactor outlet temperature will be increased to 900°C in phase II so that the helium gas turbine, of which inlet temperature is 850°C, can be demonstrated at the same temperature condition as that of the commercial plant. The higher reactor inlet temperature is better for the core thermal design because the coolant flow rate is increased due to the decrease in temperature difference between the core inlet and outlet. However, it affects the material selection for RPV. The reactor inlet temperature was determined as 325°C to apply the low-alloy steel for LWR, for example, SA533B and SA508 to the RPV, of which restrictive temperatures are 370°C (700°F) at normal operation and 540°C (1000°F) in accidents. Temperature margin is about 45°C considering the HTTR design. The primary coolant pressure affects the thickness of RPV, that is, the RPV weight. The RPV, the most heavy component, of HTR50S will be transported to the developing countries not only the coast country but also

Table 1: Major design specification of HTR50S.

Parameters	Single operation mode (phase I)			Serial operation mode (phases II and III)
Thermal power (MW)	50			50
Coolant	Helium			Helium
Reactor inlet temperature (°C)	325			325
Reactor outlet temperature (°C)	750			900
Coolant pressure (MPa)	4			4
Coolant flow rate (kg/s)	22.3			16.5
Core structure material	Graphite			Graphite
Core type	Prismatic/pin-in-block			Prismatic/pin-in-block
Effective core height (m)	3.48			3.48
Equivalent core diameter (m)	2.30			2.30
Numbers of fuel blocks	180			180
Fuel	Low enriched UO_2 TRISO coated fuel particle			Low enriched UO_2 TRISO coated fuel particle
	HTTR type fuel	High burn-up fuel		High burn-up fuel
Refueling	Whole	Half core		Half core
Reactor pressure vessel	Mn-Mo steel (SA533B/SA508)			Mn-Mo steel (SA533B/SA508)
Number of main cooling loop	1			1
Heat removal (SG/IHX (MWt))	Single mode (50/0)			Serial mode (30/20)
Steam turbine power generation (MWe)	17.2	13.5	8.6	10.3
Gas turbine power generation (MWe)	—	—	—	6.9
District heating (MWt) (water at 95°C and 0.1 MPa (t/h))	—	25 (857)	—	7.6 (259)
Process heat (MWt) (helium at 850°C and 4 MPa (t/h))	—	—	—	20 (47.3)
Process heat (MWt) (helium at 310°C and 10 MPa (t/h))	—	—	25 (45)	—
Hydrogen (Nm^3/h)	—	—	—	800 (phase III only)

the inland country. Whereas the primary coolant pressure of GTHTR300 is 7 MPa, the primary coolant pressure of HTR50S was determined as 4 MPa as the same pressure of HTTR to reduce the weight of RPV and utilize the experience obtained by the HTTR design and construction. The number of fuel blocks in axial direction was increased from 5 of HTTR to 6 in order to apply the two-batch shuffling refueling for axial direction to demonstrate the high burn-up fuel after the 2nd step of phase I. The size of the fuel block is the same as that of HTTR. These design specifications result in the effective core height of 3.48 m. The number of column is 30, which is the same as that of HTTR. Therefore, the equivalent core diameter is the same as that of HTTR of 2.30 m.

3.2. System Design. To satisfy the user requirement for the multiple heat applications, the system was designed so as to use heat from HTR50S for district heating and process heat based on the steam turbine system in phase I, for the demonstration of the helium gas turbine in phase II and for the demonstration of the hydrogen production technology in phase III. Figure 1 shows the overall plant configuration of HTR50S. The gas turbine system and IHX and the hydrogen production system will be installed in phase II and phase III, respectively. The products in each system are also listed in Table 1. The condition of steam at the inlet of steam turbine was determined as 533°C at 12.0 MPa taking into account of the proven technology of the Japanese vendor and the condition in the former HTGR steam turbine system in USA (i.e., 538°C at 16.6 MPa). The steam turbine system was designed so as to adjust the thermal duty for the district heating network and the process heat from 0 to 25 MW. It means that the thermal duty of steam turbine can be adjusted from 50 to 25 MWt as 50% partial load operation. The gross electricity generation varies from 17.2 MWe at full load operation of steam turbine to 13.5 MWe at 50% partial load operation for district heating. The overall nuclear reactor thermal power utilization is 77% at 50% partial load operation of steam turbine with the 25 MW heat supply to district heating network. The detail of the system design is described in the previous paper [5, 14].

3.3. Reactor Design

3.3.1. Core Design. The vertical and horizontal sectional views of the HTR50S core are shown in Figure 2. The reactor core is a prismatic cylindrical core and consists of 30 fuel columns, 18 replaceable reflector columns, and 13 control rod (CR) guide columns. The irradiation blocks and the outermost three columns of CR guide blocks in the HTTR were replaced with the replaceable reflector blocks in the HTR50S design. The number of the fuel block layers was increased

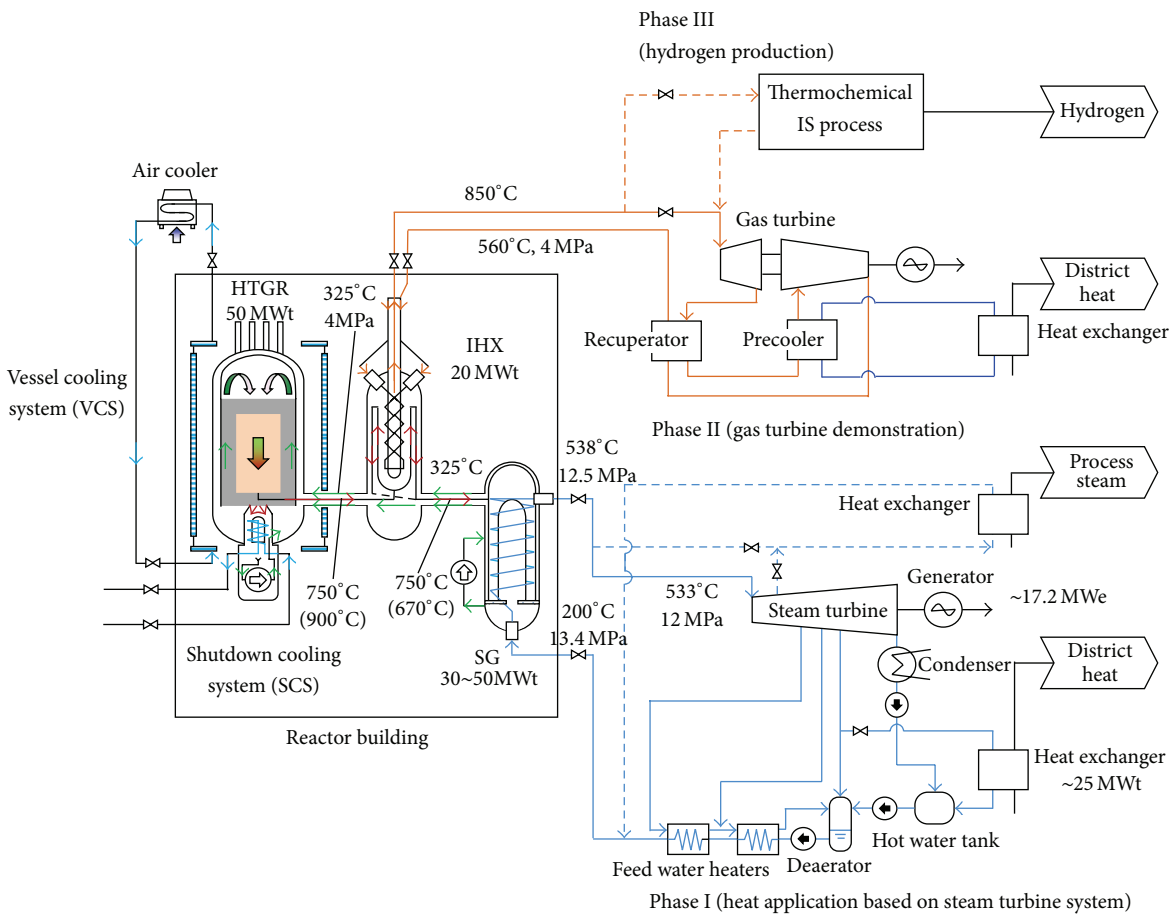


Figure 1: HTR50S plant configuration.

Table 2: Specifications of coated fuel particle.

Parameters	HTTR type fuel	High burn-up fuel
Diameter of UO_2 fuel kernel (μm)	600	500
Buffer layer thickness (μm)	60	95
Inner PyC layer thickness (μm)	30	40
SiC layer thickness (μm)	25	35
Outer PyC layer thickness (μm)	45	40
Diameter of coated fuel particle (μm)	920	920

Table 3: Specifications of fuel rod for phase I.

Parameters	HTTR type fuel	High burn-up fuel
Fuel compact		
Type	Hollow cylinder	Hollow cylinder
Packing fraction (vol%)	30	33
Outer diameter (mm)	26.0	28.0
Inner diameter (mm)	10.0	10.0
Height (mm)	39.0	39.0
Graphite sleeve		
Type	Cylinder	Cylinder
Outer diameter (mm)	34.0	34.0
Inner diameter (mm)	4.0	3.0
Height (mm)	580.0	580.0

from 5 of HTTR to 6. The configuration of the fuel element for phase I is shown in Figure 3. The specification of the graphite block for fuel element, which is made of IG-110 graphite, is the same as that of HTTR to utilize the knowledge obtained in the HTTR design and construction. It is the pin-in-block type fuel element which has 33 coolant channels in each block. The specifications of the coated fuel particles of the HTTR type fuel and the high burn-up fuel are listed in Table 2. The irradiation test for the high burn-up fuel is underway. The specification of fuel rod for phase I is also listed in Table 3.

This paper described the core design results for the 1st step of phase I, that is, the first core of HTR50S employing the HTTR type fuel. The target of core design is to enhance its performance more than that of HTTR. The number of uranium enrichment shall be reduced from 12 to less than 6 from the view point of the fuel fabrication process. The average core power density shall be increased from 2.5 MW/m^3 to 3.5 MW/m^3 and the burn-up days shall be increased from 660 days to 730 days (i.e., 2 years) with satisfying the maximum fuel temperature less than 1495°C to reduce the core size

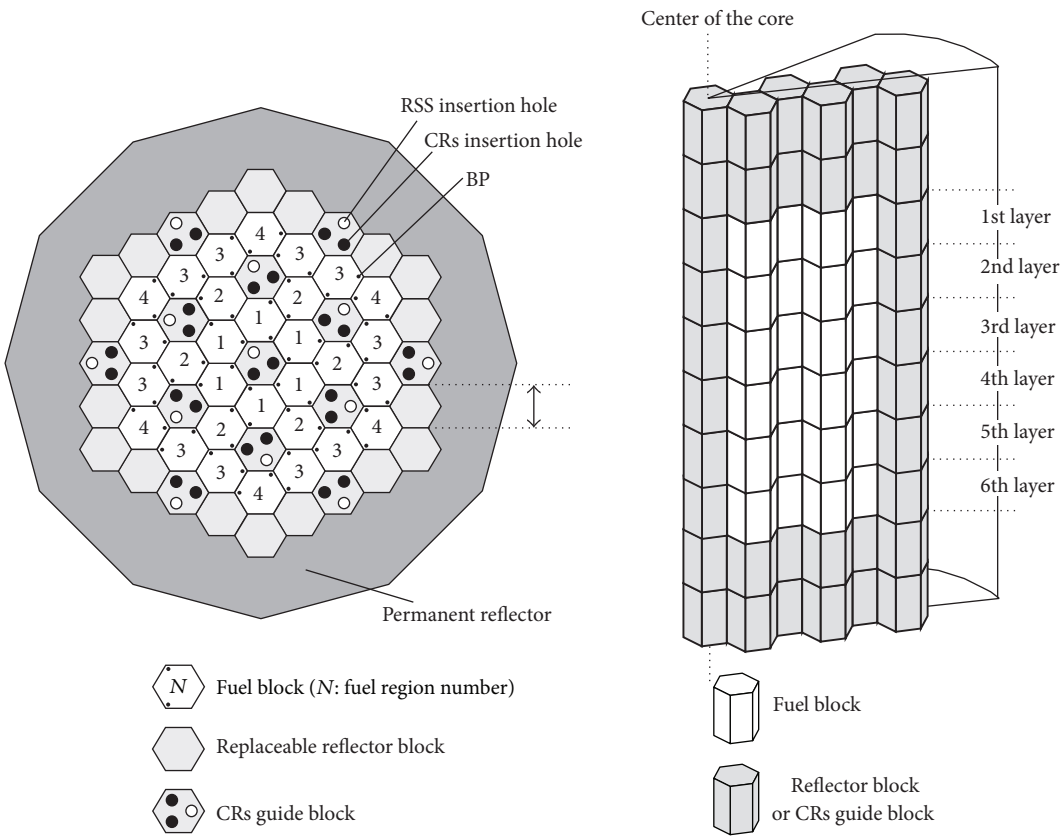


Figure 2: Vertical and horizontal sectional views of the HTR50S core.

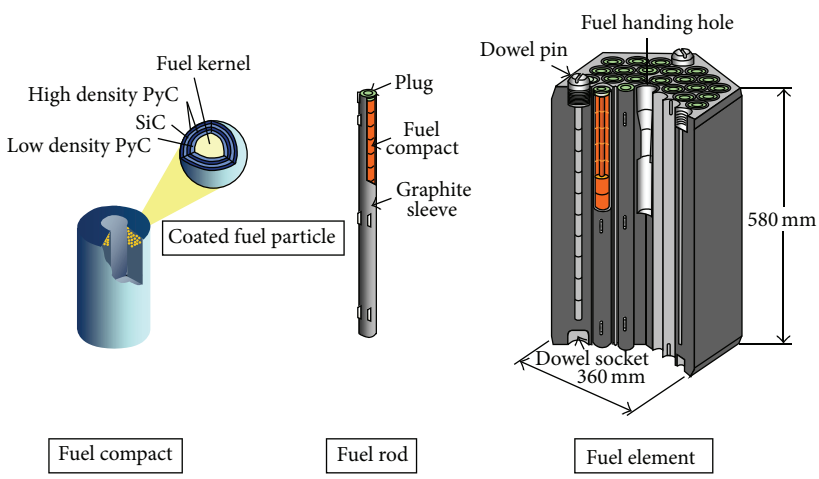


Figure 3: Configuration of the fuel element of HTR50S for phase I.

and to increase the plant availability. The most important subject in the core design is the optimization of the power distribution throughout the burn-up period satisfying the maximum fuel temperature criterion. The optimization of the power distribution was performed by changing the fuel enrichment, and the optimized power distribution shape was kept throughout the burn-up period utilizing the rod-type burnable poisons (BPs). The nuclear characteristics were evaluated by three-dimensional core burn-up calculations with the SRAC/COREBN code [15] based on a diffusion theory. The few group cross sections used for the core burn-up calculations were generated by the two-dimensional cell burn-up calculations with the SRAC/PIJ code [15] based on a collision probability method. The multigroup cross section sets used for the cell burn-up calculations are based on the JENDL-3.3 [16]. The core burn-up calculation method for a HTGR with the SRAC/COREBN was validated by using the HTTR burn-up data [17]. A three-dimensional triangular

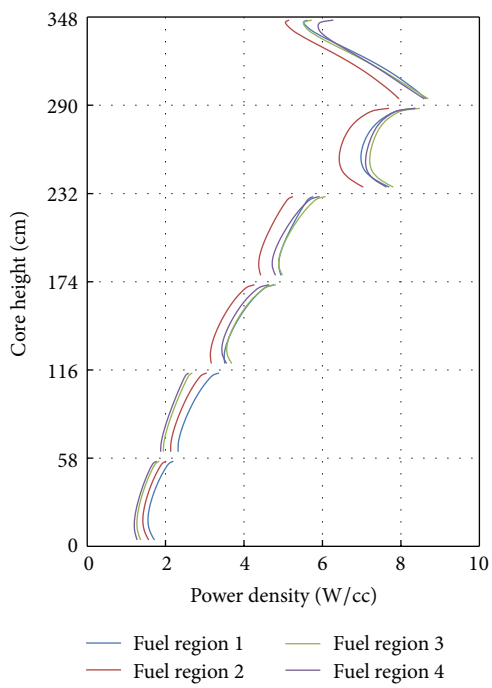


Figure 4: Power density at 30 effective full power days.

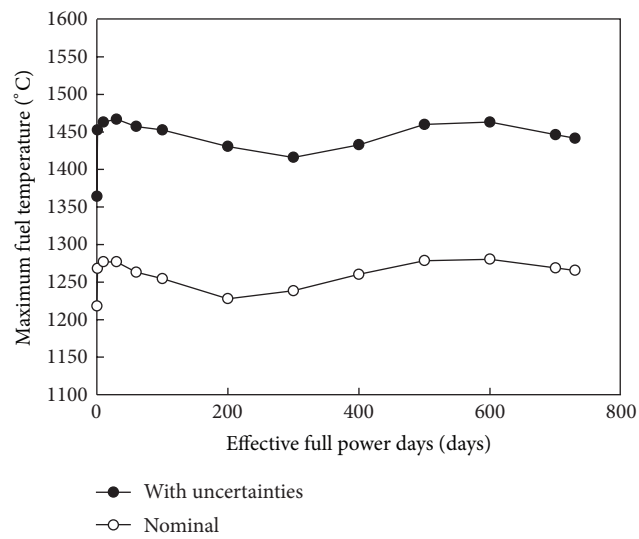


Figure 5: Maximum fuel temperature.

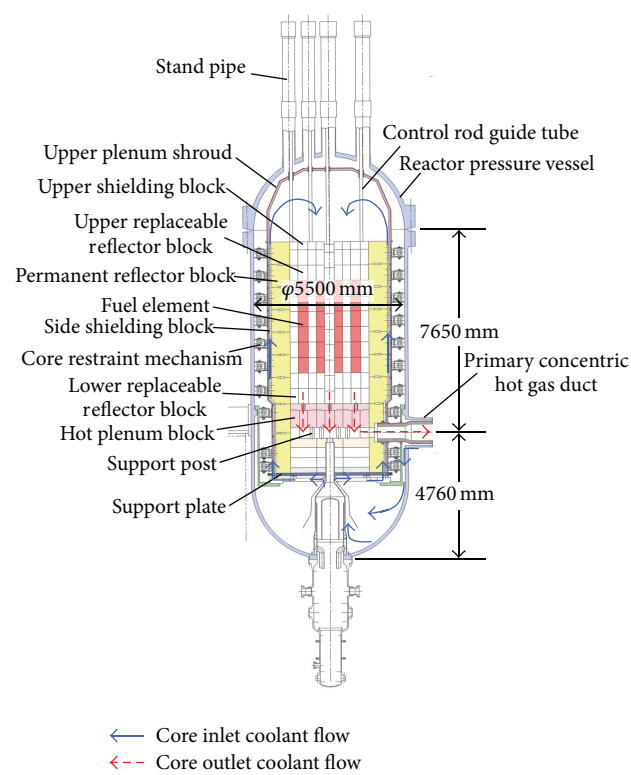


Figure 6: Structure of reactor core and reactor internals.

mesh was used in the core burn-up calculations for HTR50S. Each hexagonal block composing the core was divided into 24 triangular meshes horizontally and into 14 meshes vertically. The core burn-up calculations were performed with adjusting the CRs position at critical state at each burn-up step.

The design results of the fuel enrichment and BP alignment in the core are listed in Table 4. In the axial direction, high fuel enrichment was placed in the upper region of the core, where the high power density is required to optimize the power distribution. In the radial direction, high enriched fuel was placed in the outer region of the core, where neutron flux is low, to make the power density uniform. In the 5th and the 6th layers, the optimization for the radial direction was not conducted because the effect of the optimization on fuel temperature is relatively small due to the small power density in these regions. Thus, the same fuel enrichment was placed in the inner and the outer regions of the core in the 5th and the 6th layers. While the BPs were loaded into all of the fuel blocks in the HTTR, the BPs were not loaded into the 1st layer in the HTR50S design to optimize the power density prolife in an axial direction so as to reduce the maximum fuel temperature. Figure 4 shows the power density distribution in an axial direction at 30 effective full power days (EFPD) when the maximum fuel temperature appears throughout the burn-up period. The fuel temperature was calculated using FLOWNET [18] and TEMDIM [19] codes used in the HTTR design. Figure 5 shows the nominal peak fuel temperature and the peak fuel temperature including systematic and random uncertainties at each EFPD. The maximum fuel temperature including systematic and random uncertainties is 1467°C, which is below the design criterion of 1495°C.

3.3.2. Reactor Internal Design. Figure 6 shows the structure of reactor core and reactor internals. The reactor core is surrounded by the permanent reflector block made of PGX graphite for upper and middle side parts and IG-11 graphite for lower side part. The side shielding block and core restraint mechanism, which are installed around the permanent reflector block, restrain the displacement in the horizontal direction. The core is supported by RPV via the hot plenum blocks, support posts, and bottom structures. The primary coolant

Table 4: Design specification of HTR50S core for the 1st step of phase I.

Layer	^{235}U enrichment (wt%)				Burnable poison (BP) specifications Upper: diameter (mm) Lower: natural boron concentration (wt%)			
	Fuel region*1							
	1	2	3	4	1	2	3	4
1st	6.5	6.5	8.7	8.7	NA*2	NA*2	NA*2	NA*2
2nd	6.5	6.5	8.7	8.7	18.0 2.5	18.0 2.5	18.0 2.5	18.0 2.5
3rd	4.6	4.6	6.5	6.5	13.0 2.5	13.0 2.5	18.0 2.5	18.0 2.5
4th	4.6	4.6	6.5	6.5	13.0 2.5	13.0 2.5	18.0 2.5	18.0 2.5
5th	4.6	4.6	4.6	4.6	13.0 2.5	13.0 2.5	13.0 2.5	13.0 2.5
6th	4.6	4.6	4.6	4.6	13.0 2.5	13.0 2.5	13.0 2.5	13.0 2.5

*1Shown in Figure 2.
*2Not installed.

enters from the annular region of the primary concentric hot gas duct into RPV and flows upward in the side region inside RPV after cooling the underside of the support plate. It returns at the upper plenum of RPV and then flows downward through the fuel and the CR guide blocks. The primary coolant heated in the core is collected at the hot plenum blocks and goes to SG (or IHX for phase II) via inner pipe of the primary concentric hot gas duct. In the safety evaluation of HTR50S, all abnormal events of not only accidents but also anticipated operational occurrences (AOOs) result in the loss of forced cooling (LOFC) because SCS, which is the forced cooling decay heat removal system, is designed as the nonsafety class system. In this event, the overheating of RPV by the hot primary coolant which might flow up from the core by natural convection is a main concern. In HTTR, it was evaluated that the RPV temperature can be maintained lower than the temperature limit in the event of LOFC by the thermal shield equipped at the top head hemispherical closure of RPV [20]. The temperature limit of RPV of HTR50S in the event of AOO (e.g., loss of off-site power) decreases relatively largely, from 500°C of HTTR to 425°C, by the design change of RPV material from 2.25Cr-1Mo steel used for HTTR to Mn-Mo steel, whereas the reduction of temperature limit of RPV is from 550°C of HTTR to 540°C in the event of the accident. Therefore, the design issue of the reactor internals is to ensure the maximum RPV temperature in the event of LOFC less than 425°C so that the SCS can be designed as the nonsafety class system. The maximum RPV temperature during AOO is reduced by the following design approaches.

(i) An upper plenum shroud is installed in the upper plenum region in RPV to prevent the overheating of RPV by the hot primary coolant that flows up from the core by natural convection during AOOs.

(ii) The coolant flow path is changed from both of inside and outside of the side shielding block in the HTTR design to only inside of the side shielding block to reduce the RPV temperature at normal operation (i.e., initial temperature in the event of AOOs).

Since there is a gap between CR guide tube and top head dome shroud due to the error at the installation, the thermal and hydraulics analysis of the reactor upper region was performed using FLUENTv13.0 to evaluate the RPV temperature in the event of LOFC. The analytical model is shown in Figure 7. The upper plenum shroud was modeled as 40.0 mm thickness graphite, of which thermal conductivity is set as 10.0 W/m/K. Figure 8 shows the analysis results of the effect of the gap on the maximum RPV temperature for the reactor outlet temperature conditions of 750°C and 900°C. The RPV maximum temperature decreases with decreasing the gap. The RPV maximum temperature can be maintained below the criteria of 425°C by controlling the gap less than 50.0 mm. The gap can be controlled technically about from 30.0 mm to 40.0 mm based on the HTTR experience because the installation error of the CR guide tube in the HTTR was controlled for 41.7 mm against the design target of 45.1 mm.

3.3.3. Reactor Pressure Vessel Design. The primary concentric hot gas duct is connected not at the bottom of RPV as HTTR but at the side of RPV to lay out the primary system vessels (i.e., RPV, IHX, and SG) inline side-by-side arrangement as shown in Figure 6. Whereas the diameter is the same as that of HTTR, the length is increased due to the increase in the number of the fuel blocks in axial direction. The design target is to reduce the weight of RPV in mode of packing less than 300 tons from the view pint of shipping to the inland countries. The reactor inlet temperature was reduced from 395°C of HTTR to 325°C. It enables employing Mn-Mo steel instead of 2.25Cr-1Mo steel used in HTTR. The wall thickness of the vessel body could be reduced from 122.0 mm of HTTR to 77.0 mm under the same design pressure condition of HTTR, 4.7 MPa. In this result, the approximate RPV weight from the bottom to the flange sheet was evaluated as 180 tons.

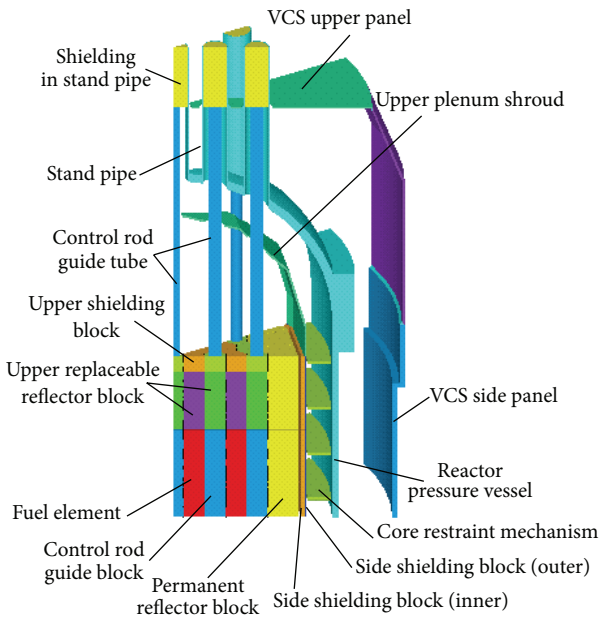


Figure 7: Analytical model for thermal and hydraulics analysis of reactor upper region.

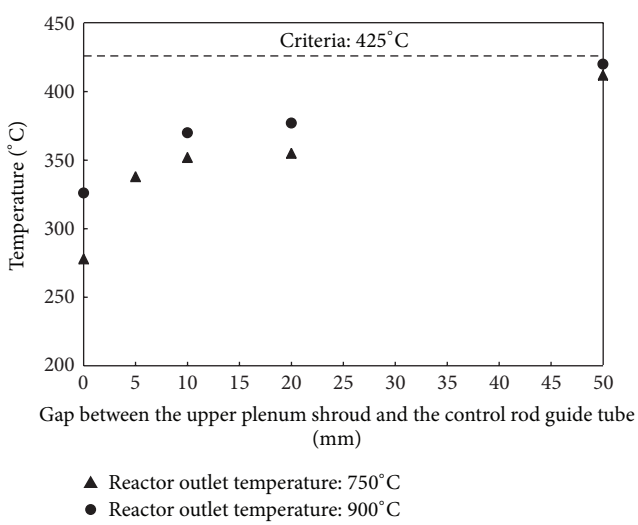


Figure 8: Effect of the gap between control rod guide tube and top head dome shroud on the maximum temperature of reactor pressure vessel.

It means that total weight of RPV including the transport cradle satisfies the design target of less than 300 tons.

3.4. Reactor Coolant System Design. The reactor coolant system of HTR50S consists of SG, the helium circulators, the primary hot gas duct, SCS, and VCS as shown in Figure 1. The IHX will be installed at the position between RPV and SG prior to phase II.

3.4.1. Vessel Cooling System Design. The VCS removes the residual heat during AOOs and the accident when neither the main cooling system nor the nonsafety-related SCS is available. The VCS is classified as the engineered safety features. The VCS shall assure the integrity of core and reactor coolant pressure boundary under the abnormal operating conditions. It removes heat from the outside surface of RPV to the VCS panel located on the surface of the biological shielding concrete wall by the radiation and the natural convection of air in the reactor cavity. It is operated even in normal operation to protect the biological shielding concrete wall from overheating. The VCS was designed as a passive water-cooled type with an air cooler by natural convection as shown in Figure 9. It consists of the cooling panels surrounding RPV, water tubes, heat removal adjustment panels, and two air coolers. The VCS was designed as a redundant means (i.e., two independent systems) because the active isolation valve is installed near the containment structure penetration in the water tube. The cooling water circulates in the water tubes by natural convection, and the air flows upward by natural convection to remove heat from the cooling water in the air cooler. The cooling panel consists of upper, side, and lower panels. The water flows inside the water tube of each panel in parallel.

The design issue is to ensure the heat removal performance by passive means. The heat and mass balance of the system was evaluated taking into account the natural circulation of the cooling water in the water tube and the natural convection of air in the air cooler to determine the specifications of water tube and heat transfer tube in the air cooler. The determined design specification of VCS is listed in Table 5. The inner diameter and the number of water tubes were increased from those of the HTTR VCS, which is a forced circulation system, to decrease the pressure drop for the natural circulation. For instance, the inner diameter of the water tube for the side and lower panels was increased from 19.0 mm of the HTTR to 26.0 mm by increasing the outer diameter from 25.4 mm to 31.8 mm and decreasing the tube thickness from 3.2 mm to 2.9 mm. The inner diameter of the tube for the upper cooling panel was increased from 9.5 mm of the HTTR to 10.1 mm by decreasing the tube thickness from 3.2 mm to 2.9 mm. The number of the tube in one system for the side panel was increased from 108 to 180. The air cooler is located at the rooftop of the reactor building to ensure the water head for the natural circulation. The differential heights between the center of the air cooler and that of upper, side, and lower panels are 23 m, 30 m, and 37 m, respectively. The number of water tubes and layers in the air cooler was determined as 120 tubes and 8 layers, respectively. The heat and mass balance under the conditions of not only two-system operation but also one-system operation was evaluated as listed in Table 6. The maximum temperature of the cooling panel is 64.8°C at one-system operation, which means that the temperature of the biological shielding concrete wall that is located outside the panel must be below the temperature limit of 65.0°C. Note that the reactor will be shut down if one out of two systems is unavailable during normal operation because VCS is the safety engineered feature. The integrity of core and reactor coolant pressure boundary in the event of accidents using the designed VCS is described in the safety analysis later.

The schematic of the cooling water panel is shown in Figure 10. The side panel consists of 12 panel units, which are located around the RPV in regular dodecagon. Each panel

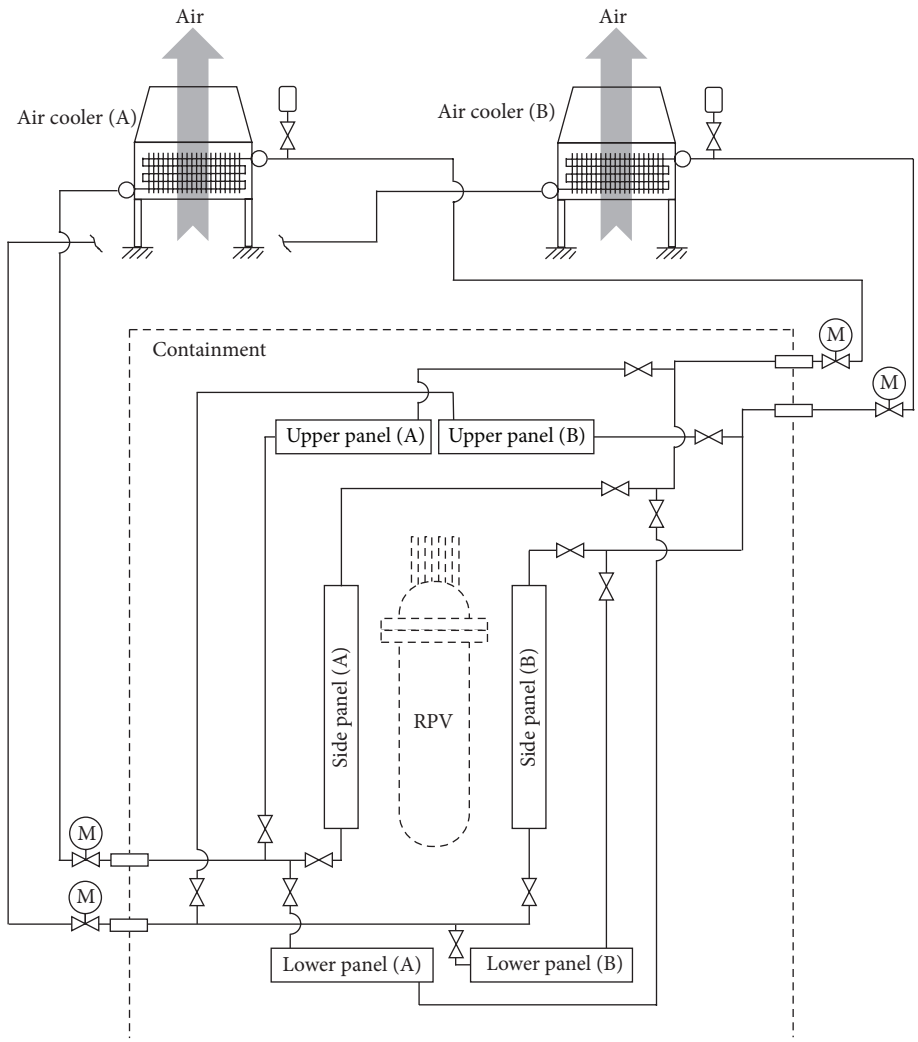


Figure 9: Schematic flow diagram of the vessel cooling system.

unit has 30 water tubes for two systems (i.e., 15 tubes for one system). The water tubes for each system are located alternately, and welded each other using plate as shown in Figure 10. The lower panel consists of 12 panel units as the side panel. Each panel unit has 16 water tubes for two systems.

3.4.2. Shutdown Cooling System Design. The SCS provides the means of removing residual heat by forced cooling whenever the main cooling system, IHX, and SG are unavailable. Its role corresponds to that of the auxiliary cooling system (ACS) of HTTR, which is the safety engineered features. The SCS of HTR50S was designed as the nonsafety class system; that is, the purpose of its installation is not for the safety but for the increase in the plant availability by reduction of the operation time for the residual heat removal because it takes a long time only by VCS. The HTTR ACS is installed outside the RPV and connected with RPV by the auxiliary concentric hot gas duct. The auxiliary concentric hot gas duct forms the reactor coolant pressure boundary as well as ACS and RPV. The depressurization accident caused by the pipe rupture of the auxiliary concentric hot gas duct is postulated in the safety analysis of HTTR. In order to exclude the postulation of the pipe rupture of the hot gas duct between RPV and SCS and in order to reduce the amount of material of the primary cooling system, the SCS of HTR50S was designed so as to eliminate the primary piping connecting RPV and SCS. The main components of SCS are the heat exchanger and helium gas circulator. They are unitized and installed inside a housing, which is integrated with RPV at the bottom centerline of RPV as shown in Figure 11. The housing is separable into two parts: upper and lower housings. The helium gas circulator which requires its maintenance is installed inside the lower housing and is removable from RPV by lowering it in the space below RPV.

The thermal duty of SCS was determined as 4.0 MW so as to prevent overcooling of reactor internal graphite from the view point of structural integrity based on the HTTR design experience and to shorten the cooling time from the view point of the plant availability. The major specification is listed in Table 7. Figure 12 shows the effect of thermal duty of SCS on the reactor inlet and outlet helium gas temperatures. The reactor outlet helium gas temperature becomes below 100°C

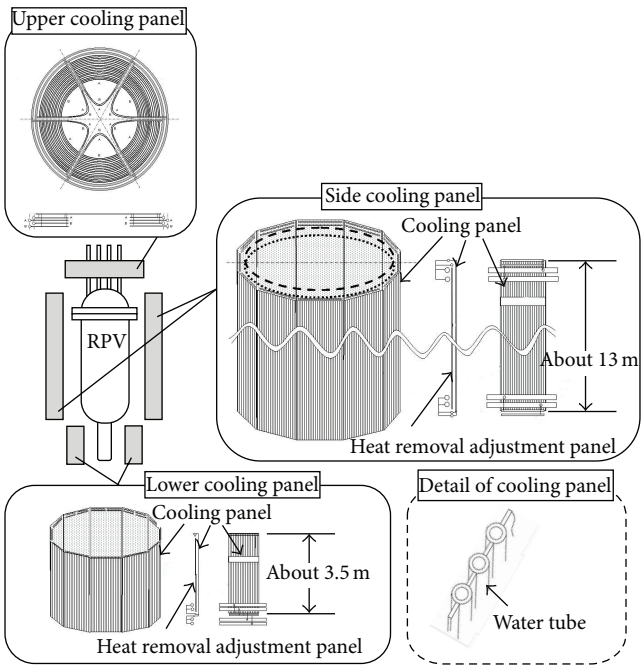


Figure 10: Schematic of cooling panels of vessel cooling system.

Table 5: Major specifications of vessel cooling system.

Parameters	Values		
Cooling water panel			
Number of systems		2	
Parts	Upper panel	Side panel	Lower panel
Panel units			
Number (upper panel: sectors, side and lower panels: units)	6	12	12
Wide (m)	—	2.0	1.15
Height (m)	—	13.0	3.5
Differential height between air cooler and panel center (m)	23	30	37
Cooling water tube			
Number (upper panel: per sector, side and lower panels: per unit)	18	30	16
Diameter (mm)	15.9	31.8	31.8
Thickness (mm)	2.9	2.9	2.9
Pitch (mm)	70.0	66.0	61.0
Air cooler			
Number of air coolers		2	
Stack height (m)		4.0	
Heat transfer tube			
Number of tubes		120	
Diameter (mm)		48.6	
Thickness (mm)		3.2	
Pitch (mm)		120.0	
Number of layers		8	
Fin			
Thickness (mm)		0.9	
Height (mm)		14.7	
Pitch (mm)		2.8	

after about 100 hours at the design point of 4 MW thermal duty; that is, the maintenance and refueling can be started at about 100 hours after the reactor shutdown.

3.4.3. Steam Generator Design. The steam is produced in SG using the heat from the primary coolant. The amount of heat-resistance alloy shall be reduced in the SG design from the view point of economy. The helical-coiled counter flow type heat exchanger that is used for the HTTR IHX was applied to the SG design as shown in Figure 13 because it can reduce the size. Both evaporation and super heater parts are installed inside the inner shell. The primary helium gas flows on the shell side and water/steam in the tube side. The major specification is listed in Table 8. To reduce the amount of heat-resistance alloy, the Alloy 800H steel is used only for the super heater part. On the other hand, the 2.25Cr-1Mo steel is employed for the evaporator part and welded using heterogeneous material joint to connect the heat exchanger tubes made of Alloy 800H steel.

Figure 14 shows the temperature profiles of the primary helium and water/steam inside SG at normal operation of phase I. The feed water of 200°C is evaporated and heated up to superheated steam of 538°C at the exit of SG, whereas the primary helium gas temperature decreases from 750°C to 325°C. The heat transfer area was determined as 260 m^2. The amount of heat-resistance alloy (i.e., Alloy 800H) was evaluated as about 1.5 tons. It is less than the amount of heat-resistance alloy in the HTTR IHX of 5.4 tons which is tentatively determined as the design target for the SG design.

3.4.4. Intermediate Heat Exchanger Design. The IHX of HTR50S is the helical-coiled type heat exchanger as same as the HTTR IHX as shown in Figure 15. One of the design issues is to ensure the strength of the heat transfer tube for its own weight because the thermal duty increases to twice (i.e., 20 MW) compared with that of HTTR whereas logarithmic mean temperature difference decreases as listed in Table 9. The specification of the heat transfer tube was determined to satisfy the stress limit for a 40-year design lifetime. The total weight of the heat transfer tube in IHX was decreased by means of the decrease in stress by increasing the tube diameter compared with HTTR IHX (from 32.0 mm to 45.0 mm) and by means of the increase in the heat transfer performance by increasing the flow velocity as listed in Table 9. The total primary stress satisfies the stress limit of 2.6 MPa for the creep damage factor of less than 0.1 for the 40-year design lifetime.

Another issue is the maximum temperature of the IHX pressure vessel in the configuration of the leveled pair of the concentric duct nozzles for the inline side-by-side layout of the vessels in the primary coolant system. The configuration of three helium flow paths, which is formed using an inner shell and a tube bundle barrel, was proposed to provide IHX pressure vessel cooling by using the bypass flow of

Table 6: Heat and mass balance of vessel cooling system.

Parameters	2-system operation			1-system operation		
	Upper	Side	Lower	Upper	Side	Lower
Boundary conditions						
RPV temperature (°C)		295			295	
Air temperature at air cooler inlet (°C)		29.4			29.4	
Results						
Cooling water panel temperature (°C)						
Maximum	52.2	51.7	51.6	64.8	64.0	63.3
Average	48.8	48.1	47.9	59.7	58.8	58.1
Cooling water temperature (°C)						
Panel inlet		42.6			50.5	
Panel outlet	49.4	49.7	50.0	60.7	61.1	61.1
Water flow rate (t/h)	4.9	68.7	7.6	3.3	44.7	4.9
Heat exchange (kW)	39.4	572.4	66.0	38.1	550.3	59.8
		677.8			648.2	
Air flow rate (m^3/s)		18.9			24.0	
Air temperature at air cooler outlet (°C)		45.1			53.6	

Table 7: Major specifications of shutdown cooling system.

Parameters	Values
Thermal duty (MW)	4.0
Primary helium	
Inlet temperature (rated condition) (°C)	900
Outlet temperature (rated condition) (°C)	325
Flow rate (kg/s)	1.3
Cooling water	
Inlet temperature (°C)	47
Outlet temperature (rated condition) (°C)	100
Flow rate (kg/s)	18
Heat exchanger tube	
Material	SUS321TB
Outer diameter (mm)	31.8
Thickness (mm)	3.5
Number of tubes	19
Helical tube bundle	
Number of layers	3
Effective heat transfer area (m^2)	16

Table 8: Major specifications of steam generator.

Parameters	Values
Thermal duty (MW)	50
Primary helium	
Temperature (inlet/outlet) (°C)	750/325
Flow rate (kg/s)	22.4
Water/steam	
Temperature (inlet/outlet) (°C)	200/538
Pressure (inlet/outlet) (MPa)	13.3/12.5
Flow rate (kg/s)	19.3
Heat exchanger tube	
Material	Evaporator: 2.25Cr-1Mo steel Super heater: Alloy 800H
Outer diameter (mm)	31.8
Thickness (mm)	3.5
Number of tubes	36
Helical tube bundle	
Number of layers	8
Outer diameter (m)	1.4
Inner diameter (m)	0.7
Effective heat transfer area (m^2)	260

low temperature helium at 325°C from SG as shown in Figure 15. The maximum IHX pressure vessel temperature was evaluated as 305°C at the design point of 1% bypass cooling flow rate, which satisfies the design criterion. The detailed design result is described in the previous paper [14].

3.5. Plant Layout. The design of the system and component layout in the reactor and turbine buildings and the site layout was conducted as well as the design of the containment structures. Figure 16 shows the plant layout after phase II, especially the reactor building and the steam turbine building. The steam turbine building is located at the side of the reactor building. The system and component layout in the reactor building was designed based on that of HTTR. The space for the installation of IHX, gas turbine, and hydrogen production systems after phase II was taken into account for the overall layout design. The RPV, IHX, and SG are located inside the containment structure, the reinforced concrete containment vessel (RCCV). The RCCV of HTR50S is located below grade as well as the containment vessel (CV) of HTTR which is a steel containment. The layout of the primary components, RPV, IHX, and SG, inside RCCV was determined as the

Table 9: Major specifications of intermediate heat exchanger.

Parameters	HTR50S	HTTR
Thermal duty (MW)	20	10
Design life time (years)	40	20
Logarithmic mean temperature difference (°C)	76	101/113
Primary flow		
Inlet temperature (°C)	900	850/950
Outlet temperature (°C)	670	387/389
Velocity (m/s)	12.6	9.3/8.1
Secondary flow		
Inlet temperature (°C)	560	244/237
Outlet temperature (°C)	850	782/869
Velocity (m/s)	43.0	30.6/27.1
Heat transfer tubing		
Material	Hastelloy XR	Hastelloy XR
Outer diameter (mm)	45.0	31.8
Wall thickness (mm)	5.0	3.5
Number of tubes	159	96
Helical tube bundle		
Number of layers	9	6
Outer diameter (m)	2.24	1.31
Inner diameter (m)	1.20	0.84
Effective heat transfer area (m^2)	500	215

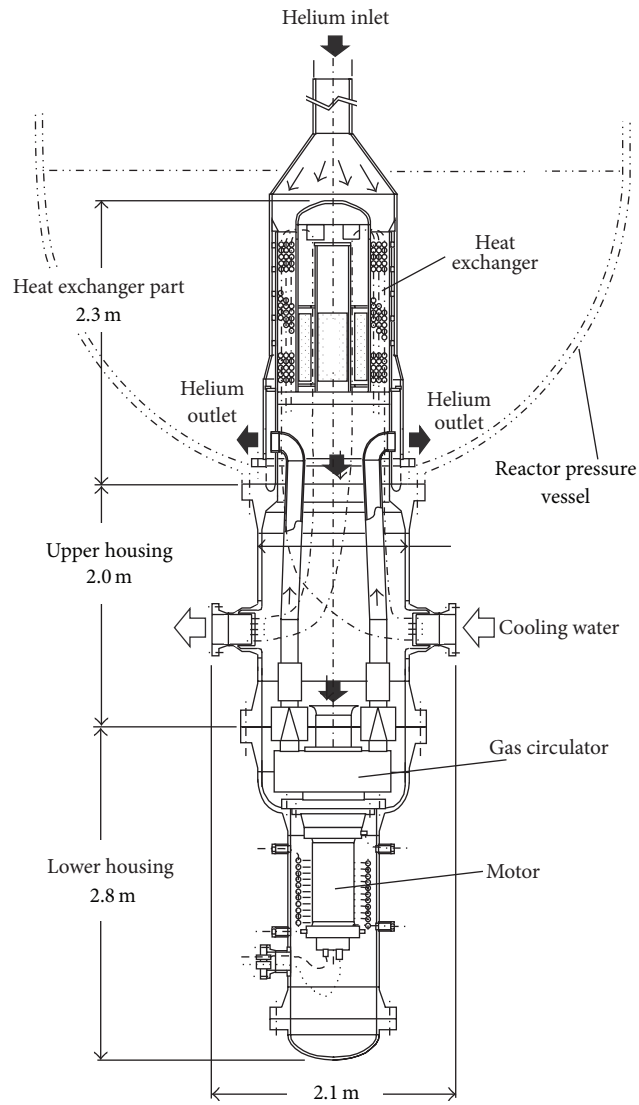


Figure 11: Configuration of the shutdown cooling system (SCS).

side-by-side arrangement using the leveled pair of concentric hot gas ducts taking into account the restriction of the free volume inside RCCV. The free volume inside RCCV shall be determined so as to minimize the amount of air which may react with graphite components in the event of the rupture of the reactor coolant pressure boundary and to withstand the pressure transient such as the rupture of the reactor coolant pressure boundary and rupture of the steam system piping inside RCCV. Both the amount of graphite oxidation and the flammable gas concentration inside RCCV in the event of the rupture of the reactor coolant pressure boundary can satisfy the design criteria conservatively when the free volume in RCCV is less than 3890 m^3 based on the HTTR safety analysis, whereas the detailed evaluation result of HTR50S is described in the safety analysis later. The peak pressure in the event of the rupture of the reactor coolant pressure boundary can be maintained less than the design pressure when the free volume in RCCV is larger than 3290 m^3 based on the HTTR safety analysis. Hence, the designed free volumes inside RCCV are about 3860 m^3 and about 3700 m^3 without and with IHX, respectively. The pressure in RCCV of which free volume is 3700 m^3 was evaluated as 0.12 MPa, which is less than the design pressure of 0.41 MPa, in the event of the rupture of the main steam and feed water piping. The amount of steel used in RCCV was estimated as about 420 tons, which is less than about 700 tons of the HTTR CV.

4. Safety Analysis

4.1. Analysis. The preliminary safety analysis for HTR50S of the 1st step of phase I was conducted to confirm the safety of HTR50S. The rupture of concentric hot gas duct in the primary cooling system (i.e., air ingress accident) and the tube rupture of SG (i.e., water ingress accident) were deterministically selected as the events for the preliminary safety analysis because these events cause the graphite core structure oxidation and are severe events for HTGR in general.

The acceptance criteria for the accident are listed in Table 10, which are basically based on those of HTTR except the criteria for the materials that are not used in HTTR. The conservative boundary conditions, the single failure criterion, and a loss of off-site power were applied in the same

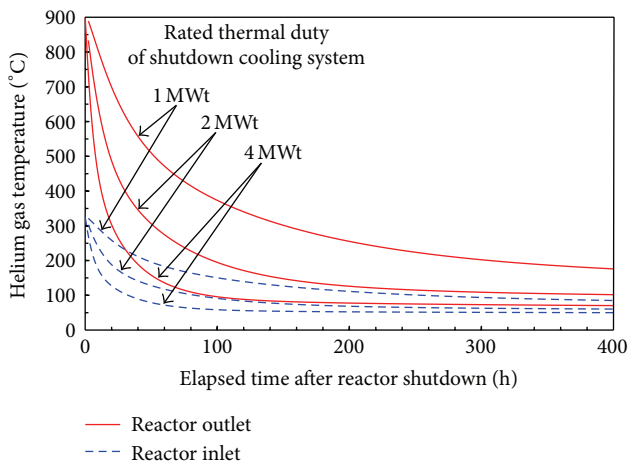


Figure 12: Effect of the thermal duty of shutdown cooling system on helium gas temperature at reactor inlet and outlet.

Table 10: Acceptance criteria in accident.

Acceptance criteria
The reactor core shall not be seriously damaged and can be cooled sufficiently
Pressure on reactor coolant pressure boundary is less than 1.2 times of maximum pressure in service
Maximum temperature of reactor coolant pressure boundary
Mn-Mo steel such as the reactor pressure vessel: <540°C
2.25Cr-1Mo steel such as heat transfer tubes in the steam generator: <550°C
Alloy 800H such as heat transfer tubes in the steam generator: <760°C
Maximum pressure on containment boundary is less than maximum pressure in service
No significant risk of radiation exposure to public

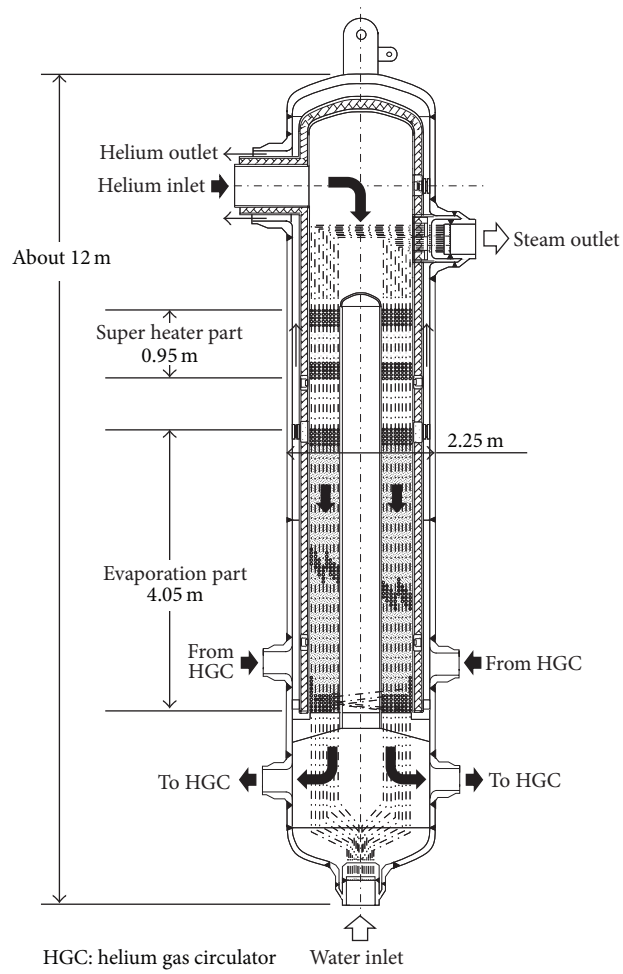


Figure 13: Configuration of the steam generator (SG).

manner as the HTTR safety analysis [20, 21]. The TAC-NC code [22] was used to calculate long-term transient behavior for the rupture of concentric hot gas duct in the primary cooling system. The RELAP5 code [23] was used to calculate short-term transient behavior for the rupture of concentric hot gas duct in the primary cooling system and transient behavior for the SG tube rupture. The graphite oxidation behavior of core internal structure was calculated using the THYTAN code [24], which was originally developed for the calculation of mass balance of tritium and hydrogen in the HTGR hydrogen production system using the flow network. The THYTAN code has been modified to calculate the mass balance of the graphite oxidation reaction and the gas concentration in the primary circuit and containment structure.

The source term at the accidents was conservatively evaluated by the same manner of the HTTR [20, 21]. The effective dose, which includes external gamma-ray exposure from the radioactive cloud containing noble gases and iodine, internal exposure by inhalation from the radioactive cloud, direct external gamma-ray exposure, and external skyshine gamma-ray exposure from fission products such as Cesium contained in the containment structure, was tentatively evaluated with the assumption of the same siting condition as the HTTR (e.g., the weather condition and the distance to the nearest site boundary (i.e., 280 m)). And the evaluation results were compared to the dose limit stipulated by the Nuclear Regulation Authority in Japan.

4.2. Analysis Results

4.2.1. The Rupture of Concentric Hot Gas Duct in the Primary Cooling System. Figure 17 shows the short-term reactor transient behavior of the reactor power and the peak fuel temperature. After the rupture of concentric hot gas duct, the primary pressure decreases rapidly and the reactor is scrammed after 4 seconds by detecting the decrease of the primary coolant flow rate. Despite that the reactor power increases at first, the peak fuel temperature does not increase due to the large heat capacity of the reactor core. Note that the initial reactor power is set as 102.5% of full power taking into account power calibration error, and so forth to obtain conservative results in the same manner of the HTTR safety analysis. Figure 18 shows the long-term transient of the peak temperatures of fuel and RPV. The residual heat is

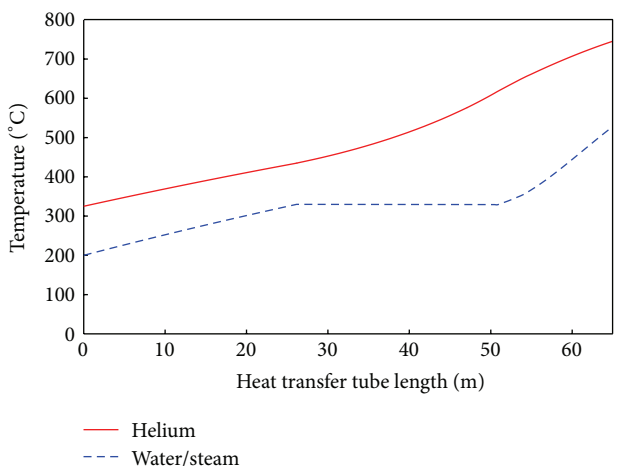


Figure 14: Coolant temperature profile in the helical-coiled type steam generator at nominal operation.

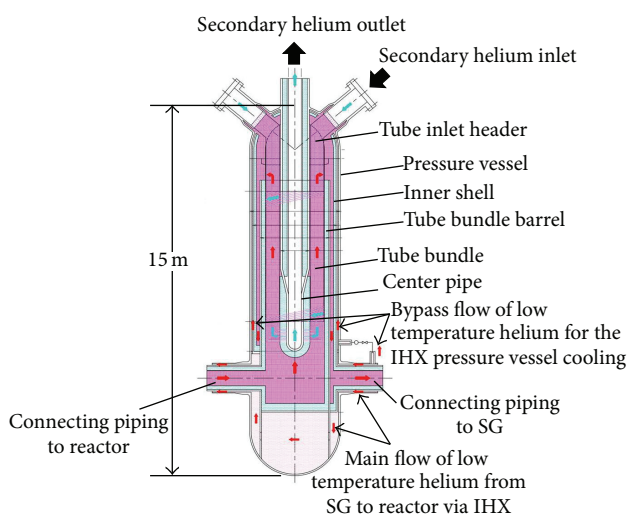


Figure 15: Configuration of the intermediate heat exchanger (IHX).

removed only by VCS. The peak fuel temperature initially decreases after the reactor scram, increases again, and shows the maximum temperature of 1386°C at about 28 hours after the initiation of the accident. However, it does not exceed the initial fuel temperature and decreases below 500°C after 2000 hours. The maximum RPV temperature appears at the side wall and is 364°C at about 20 hours after the initiation of the accident. The maximum RPV temperature remains below the temperature limit of 540°C, assuring the integrity of RPV.

In the graphite oxidation analysis, air induced into the core by natural circulation oxidizes the core bottom structure first then fuels. Figure 19 shows the oxidization profile of the support post located at the core bottom structure. The equivalent depleted thickness (i.e., equivalent thickness of the reacted graphite) is about 5.0 mm. The residual diameter of the support post is about 140.0 mm, which is thick enough to maintain the integrity of the support post. Oxidation of the bottom plate for the graphite sleeve of fuel rod was also evaluated. The oxidized thickness is only less than 1.0 mm from the bottom surface. The thickness of the bottom plate is more than 9.0 mm, and the graphite sleeve can sufficiently support the fuel compacts. The concentration of carbon monoxide (CO), which is produced by the graphite oxidation, in the containment structure is out of the explosive range. It can be concluded that the integrity of the reactor core can be maintained from these results.

The amounts of fission products released to the environment are 1.6×10^{12} MeV Bq for noble gases and 8.9×10^{11} Bq for ^{131}I equivalent. The effective dose equivalent is 4.9 mSv, which is lower than the dose limit of 5.0 mSv. The margin of the evaluated effective dose equivalent to the criterion is only 0.1 mSv. However, this preliminary safety analysis was performed under much conservative conditions so as to confirm the technical feasibility of the conceptual design. The safety analysis in the basic design phase must be conducted using the improved assumptions and conditions and should have large margin to the criterion of public dose.

4.2.2. The Tube Rupture of Steam Generator. The water ingresses into the helium side of SG through the rupture point of the heat transfer tube in SG. Figure 20 shows the reactor transient behavior of the reactor power, the primary cooling system flow rate, the peak fuel temperature, and the flow rate at rupture point. The reactor is scrammed after 1 second after the initiation of the accident by detecting the increase in the reactor power. Whereas the reactor power increases to about 129%, the peak fuel temperature increases by only 2°C then decreases gradually. The primary cooling flow rate decreases drastically due to the stop of the primary gas circulators at the same time of the reactor scram. The isolation valves are also started to be closed at the same time of the reactor scram. It takes about 73 seconds to close completely the isolation valves installed in the feed water and main steam lines. The amount of the water entering the primary cooling system was evaluated to be about 568 kg. The reactivity addition is approximately 1.3×10^{-3} $\Delta k/k$ which is less than the shutdown margin. The PRV peak temperature does not exceed the initial temperature. The residual heat can be removed by only VCS.

The decrease in the support post by the oxidation is about 3.0 mm in diameter, which is small enough compared to the support post diameter of 150.0 mm. The oxidized thickness of the bottom plate for the graphite sleeve is only less than 1.0 mm from the bottom surface, which means that the graphite sleeve can sufficiently support the fuel compacts as well as the rupture of concentric hot gas duct in the primary cooling system. Figure 21 shows the transient behavior of the pressure in the primary cooling system. The pressure increases due to the water ingress into the primary cooing system, then decreases drastically due to the opening of the safety valve, and stops decreasing due to the close of the safety valve when the pressure becomes lower than the preset value of the closing of the safety valve. In this result, approximately 15% of gas mixture in the primary cooing system is released into the containment structure. However, the concentration of the flammable gases (i.e., hydrogen and CO produced by graphite oxidation) is about 0.003%, which is very small compared to the explosive range.

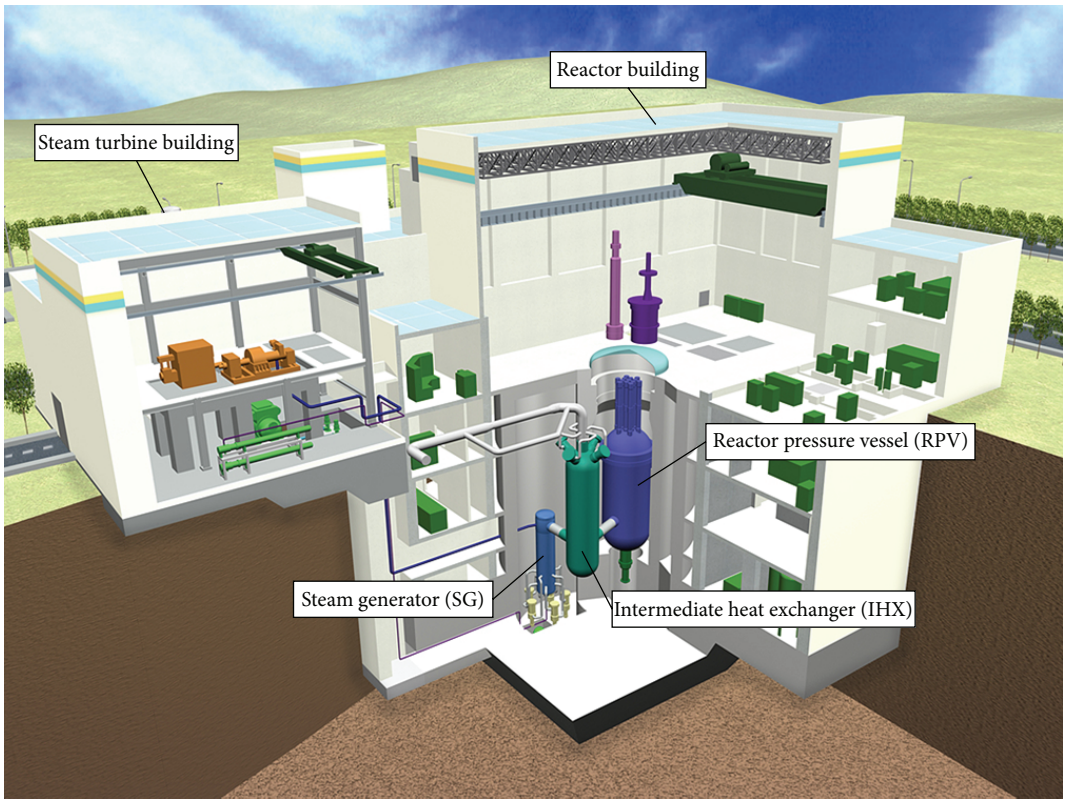


Figure 16: Plant layout of HTR50S.

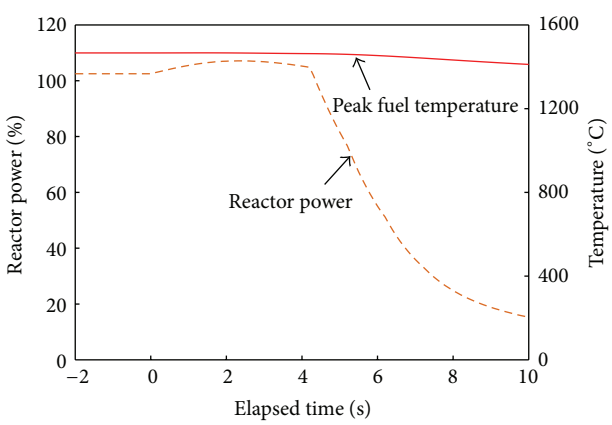


Figure 17: Short-term transient response during rupture of concentric hot gas duct in the primary cooling system in HTR50S.

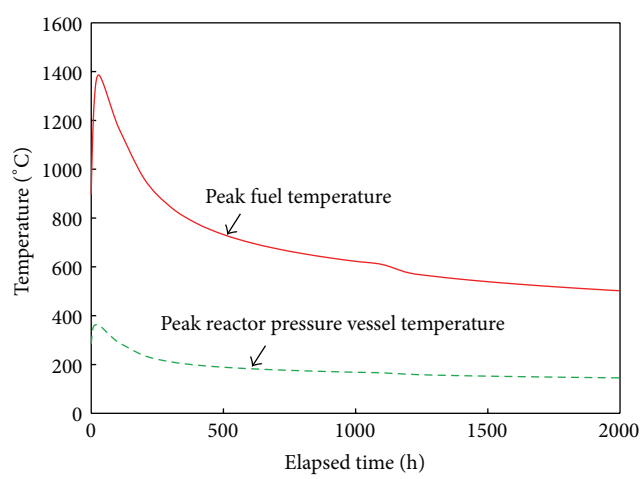


Figure 18: Long-term transient response during rupture of concentric hot gas duct in the primary cooling system in HTR50S.

The ratio for the removal of sorbed FPs from the primary metallic component (i.e., piping) by water and steam induced from the secondary circuit was conservatively assumed as 100% in the evaluation of source term. A part of FPs in the primary circuit is released into the containment structure by the opening of the safety valve. The amounts of fission products released to the environment are 8.6×10^9 MeV Bq for noble gases and 6.1×10^{10} Bq for ^{131}I equivalent. The effective dose equivalent is 3.5 mSv, which is lower than the dose limit of 5.0 mSv.

5. Evaluation of Technical Feasibility

The technical feasibility assessment for HTR50S is summarized in Table 11. The HTR50S can be utilized for the multiple heat applications such as the district heating based on the steam turbine system and the demonstration of the gas turbine and hydrogen production by using secondary side of IHX. The heat supply for district heating network can be adjusted from 0 to 25 MW. As for the core design, the

Table 11: Technical feasibility assessment for HTR50S.

Components	Design target	Criteria		Results
System design	Multiple heat applications	Heat supply for district heating network (MW)	0~25	0~25
		Demonstration of gas turbine and hydrogen production		Gas turbine and hydrogen production can be demonstrated by using the secondary side of IHX
Core	Reduce the number of uranium enrichments	Number of uranium enrichments	<6	3
	Reduce the core size	Average core power density (MW/m^3)	3.5	3.5
	Increase plant availability	Refueling interval (years)	2	2
		Maximum fuel temperature (°C)	<1495	1467
Reactor internal	Shutdown cooling system is designed as nonsafety class system	RPV temperature at AOO (°C)	<425	420
Reactor pressure vessel (RPV)	Reduce the weight	Weight in mode of packing (tons)	<300	about 180
Vessel cooling system (VCS)	Passive means	Biological shielding concrete wall temperature at normal operation (°C)	<65.0	64.8 (panel temp.)
		Fuel temperature at accident (°C)	<1600	The acceptance criteria were satisfied in the events of the rupture of concentric hot gas duct in the primary cooling system and the SG tube rupture
		RPV temperature at accident (°C)	<540	
Shutdown cooling system (SCS)	Reduce the reactor coolant pressure boundary	Eliminate the primary piping related to SCS		No primary piping related to SCS
Steam generator (SG)	Reduce the amount of heat-resistance alloy	Weight of heat-resistance alloy (tons)	<5.4	1.5
Intermediate heat exchanger (IHX)	Thermal duty: 20 MW	Primary stress limit (MPa) (creep damage factor <0.1, 40 years of lifetime)	≤2.6	2.6
	Inline layout of the concentric duct nozzles	Vessel temperature (°C)	<370	305
Containment structure	Reduce the weight	Weight of steel (tons)	<700	About 420
	Minimize the amount of air which may react with graphite components	Free volume (m^3)	>3290	3860 (without IHX), 3700 (with IHX)
		Free volume (m^3)	<3890	
	Withstand the pressure transient	Pressure at rupture of main steam piping (MPa)	<0.41	0.12
Safety analysis	Satisfy the acceptance criteria	Shown in Table 10		The acceptance criteria were satisfied in the events of the rupture of concentric hot gas duct in the primary cooling system and the SG tube rupture

AOO: anticipated operational occurrence.

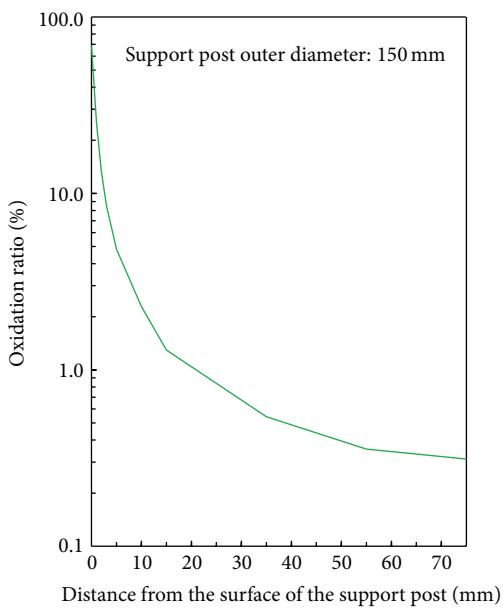
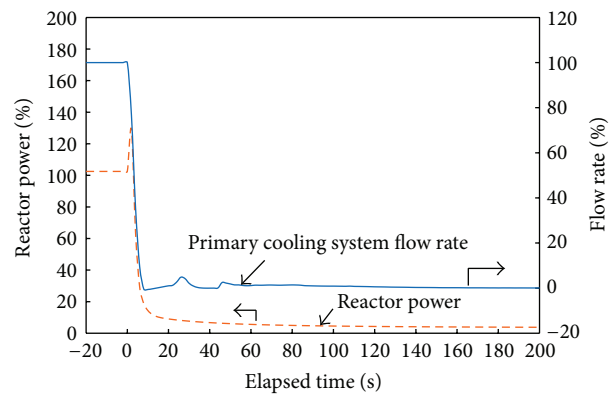
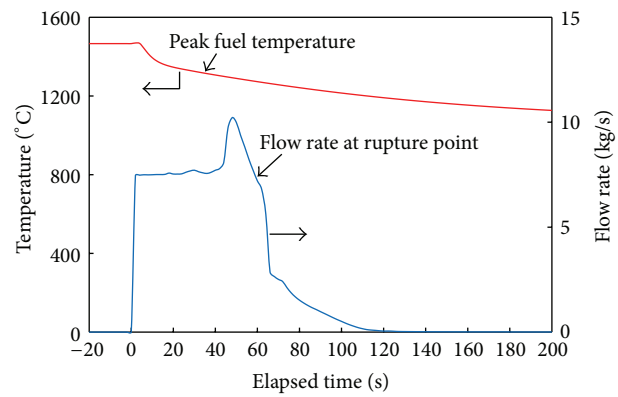


Figure 19: Oxidation profile of the support post during the rupture of concentric hot gas duct in the primary cooling system in HTR50S.



(a) Reactor power and primary cooling system flow rate



(b) Peak fuel temperature and flow rate at rupture point

Figure 20: Transient response during tube rupture of steam generator in HTR50S.

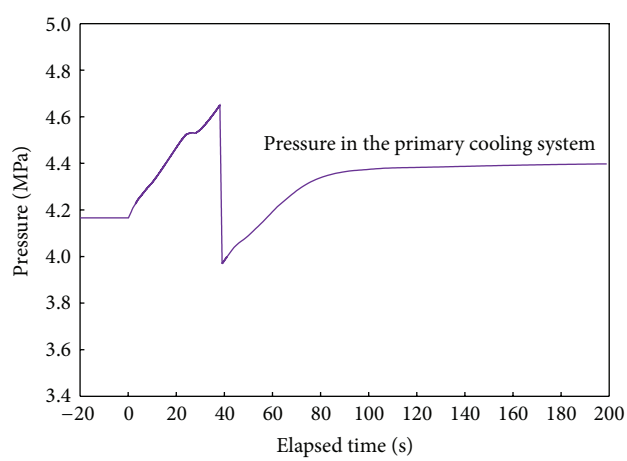


Figure 21: Transient behavior of the pressure in the primary system during tube rupture of steam generator in HTR50S.

number of uranium enrichments can be reduced from 12 to 3. The average core power density and the burn-up days can be increased from 2.5 MW/m^3 to 3.5 MW/m^3 and from 660 days to 730 days (i.e., 2 years) satisfying the maximum fuel temperature less than 1495°C, respectively. The reactor internal design employing the upper plenum shroud and alternative coolant flow path in the side shielding block satisfies the RPV maximum temperature at AOO less than 425°C, which means SCS can be designed as the nonsafety class system. The weight of RPV in mode of packing is less than the target weight of 300 tons by employing Mn-Mo steel instead of 2.25Cr-1Mo steel used in HTTR. The VCS can maintain the concrete wall temperature below the acceptable temperature during normal operation and fuel and RPV temperatures below each acceptable temperature in the event of accidents by passive means. The SCS is located inside the housing, which is integrated with RPV, so as to eliminate the primary piping related to SCS. In the SG design, the amount of heat-resistance alloy is reduced by employing heterogeneous material joint to connect 2.25Cr-1Mo steel and heat-resistance alloy of Alloy 800H steel. The IHX is designed for the thermal duty of 20 MW and inline layout of the concentric duct nozzles. The preliminary safety analysis for the rupture of concentric hot gas duct in the primary cooling system and the SG tube rupture shows the adequacy of the HTR50S safety design. It can be concluded that all design targets are satisfied by the design of each system and the preliminary safety analysis.

6. Conclusions

The HTGR, which is one of small-sized reactors, has attractive features, for example, inherent and passive safety features and multiple heat applications such as the electricity generation for dispersed power system, the district heating, the process heat for industrial growth, and the hydrogen production for the creation of industries in the future. It is a suitable nuclear reactor to be deployed in the developing countries. Conceptual design of a 50 MWt small-sized HTGR, HTR50S, for multiple heat applications with high level of safety has been conducted by JAEA with the support of Japanese

vendors. The design results satisfy the design targets such as the core performance, system design criteria, and the safety in the event of the representative accidents such as air and water ingress accidents. It can be concluded that HTR50S can satisfy the user requirements for multiple heat applications and its performance is upgraded compared to that of HTTR without significant R&D utilizing the knowledge obtained by the HTTR design and operation. It is expected that these conceptual design results are to be applied to the basic design of a small-sized HTGR in the developing countries.

Conflict of Interests

The authors declare that there is no conflict of interests regarding the publication of this paper.

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