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Conceptual Design Study of a High Performance Commercial HTGR

for Early Introduction

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Abstract

Conceptual design study of a commercial High Temperature Gas-cooled Reactor (HTGR) for early introduction has been performed based on the cumulated experience in design, construction, and operation of the High Temperature engineering Test Reactor (HTTR) and design of the commercial Gas Turbine High Temperature Reactor 300 (GTHTR300). The power output is 165MWt and the inlet and outlet coolant temperatures are 325°C and 750°C, respectively, to provide steam for industrial utilization. However, given a requirement for the reactor pressure vessel to be smaller even that of the 30 MWt HTTR, several challenging technical problems have to be dealt with to arrive in a high performance core design that provides extended fuel burnup, prolonged refueling period, improved fuel refueling scheme, improved fuel element and so on from the HTTR.

KEYWORDS: commercial HTGR, early introduction, HTTR, GTHTR300

1. Introduction

Nuclear power is an attractive energy source of clean air and carbon-free electricity that produces no greenhouse gases or air pollutants unlike power generation with fossil fuel. To reduce CO_2 gas emission, the simplest way is to replace the fossil fuel utilization to nuclear power. In addition, High-Temperature Gas-cooled Reactor (HTGR) attracts attentions due to the inherent safety features (Ohashi et al., 2011). From this point of view, we investigated the conceptual design of commercial HTGR for early introduction to provide steam for industrial heat utilization based on the cumulated experience of a design, construction, and operation of High Temperature engineering Test Reactor (HTTR) (Saito et al., 1994), whose power output is 30 MWt, and a design of a commercial HTGR of Gas Turbine High Temperature Reactor 300 (GTHTR300) (Yan et al., 2003), whose power output is 600 MWt. Both core design employed prismatic and pin-in-block type fuel. To introduce early, we should inherit HTTR technologies as much as possible. However, we may have to improve the technologies to suit the request for commercial HTGR by employing the technologies developed for GTHTR300. By this treatment, not only achieving early introduction but also saving the project cost are expected.

The power output is assumed to be 165 MWt, and the inlet and outlet coolant temperature are assumed to be 325 °C and 750 °C, respectively. Moreover, we limited a diameter of the pressure vessel to 4.5 m, which is smaller than that of HTTR of 5.8 m, from the viewpoint of transport by load, rail, ship, and air plane by assuming a certain nuclear developing country. Major specifications are determined by considering the feasibility with limited R&D and performance as the first commercial HTGR in the following sections.

2. Major Specifications and Technology for High Performance

We are planning to employ the technologies to realize high performance under the condition described in the previous section.

Confinement

To realize higher level of safety, we selected a confinement instead of a containment, which is employed in HTTR. The confinement is employed GTHTR300 as well. With a containment, the pressurized and high temperature coolant gas would circulate into the area inside of the containment and inside of a reactor vessel with a raptured pipe in a depressurization accident. On the contrary, the confinement releases the pressure via a vent, which has a filter to prevent the unacceptable release of radioactive nuclides, if necessary, and a dumper working passively. The gas release can prevent the pressurized high temperature gas circulation. Moreover, due to the simple confinement structure, the construction cost is expected to be reduced up to 10 % in the evaluation of GTHTR300 (Kunitomi et al., 2007).

SiC matrix monolithic fuel

To satisfy the safety requirement after the Fukushima Daiichi Nuclear Accident (Atomic Energy Society of Japan, 2015), oxidation durability of fuel element should be enhanced. Therefore, we employed SiC matrix monolithic fuel because of the high durability for the oxidation in the depressurization accident condition. In the accident, the coolant helium is released from the raptured pipe and air flow into the core. The graphite material is thinning by the active oxidation due to the air ingress when the graphite temperature is high until the decay heat reduces. HTTR fuel employs fuel with graphite sleeve. With the confinement, which has not high air tightness than the containment, significant thinning of sleeve may occur and the fuel element may also be damaged with the existing sleeve fuel. However, oxidation of SiC matrix is passive even with the severe condition with air ingress (Minato et al. 1993). Thinning does not occur by protection of the passive film formed on the surface of SiC matrix.

Moreover, we have been developed a pressure less casting method, in which the Coated Fuel Particles (CFPs) packed into a mold first and liquid matrix material is poured into it. It is different with ordinally method, the overcoat method (Yoshimura et al., 1990). In the method, overcoat is evaporated to the CFPs, and the overcoated CFPs are pressed in a mold. At this stage, fuel failure occurs, and it is called initial failure. By this reason, the packing fraction is limited up to 33 % (Mizuta et al., 2017) with the method. On the other hand, JAEA started the development of new fabrication method named pressure less casting method. In the method, CFPs is charged into a mold in the first place, and liquid matrix is poured into the mold. The matrix is solidified by drying, and it is sintered. Unlike the overcoat method, the initial failure is not problematic by the virtue of the pressure less process. The high packing fraction of 60 % also expected with this method because the highest packing fraction by the random packing is 64 % (Berryman, 1983). General Atomic company (GA) attempted similar injection method in random close packing of particles by using thermosetting resins to make graphite matrix compact with packing fraction of ~60 % for Fort Saint Vrain Reactor (Bullock, 1977).

By considering early introduction (approximately 10 years), the maximum burn-up is limited due to a lack of experimental data. We are planning to employ the Extended Burn-Up (EBU) CFP, but the discharged burn-up is limited up to 100 GWd/t, which was confirmed experiment in Kazakhstan (Ueta et al., 2016) as described latter. On the contrary, the maximum burn-up of GTHTR300 is 160 GWd/t. To realize longer cycle length and burn-up period, the high packing fraction is necessary to compensate the disadvantage of lower burn-up. In the future, the burn-up is expected to extend 200 GWd/t because the design of ISTC CFP is similar to AVR GLE-4 design in HFR-EU1 experiment (Petti et al., 2011) whose maximum burn-up is 200 GWd/t.

Moreover, we propose the fuel concept where the both side of annular fuel compact are directly cooled by coolant. The SiC matrix of thermal conductivity is inferior to that of graphite powder matrix. In HTTR condition, the maximum fuel temperature reduction by fuel cooling methods was investigated (Inaba et al. 2017). In the original fuel design of HTTR, the annular fuel compact is confined into graphite sleeve. In this evaluation, the coolant flow in the center hole of the fuel compact was hypothetically assumed. The maximum fuel temperature reduces by approximately 200 ℃. By the both side cooling, the lower thermal conductivity would be conquered.

In addition, reduction of auxiliary rate is expected because of the large flow area composed of inner cylindric flow path and outer annular flow path. Those reduces pressure drop of the coolant in the core because it is proportional to coolant flow velocity, which is slowed by the expanded flow area with the constant mass flow rate.

EBU CFP

For CFPs, there are two candidates, HTTR A-type and EBU type. The specifications illustrated in Fig.1 (Nuclear Fuel Industries Ltd, 2005, Aihara et al., 2014) and listed in Table 1. The diameter of CFP is the same for both type of CFPs. For the HTTR type-A CFP, kernel migration is problematic. In HTTR condition, the life time due to the kernel migration is 660 days (Hayashi et al., 1989). The life time is too short for commercial reactor with multi-batch core. On the contrary, EBU CFPs would be expected high durability for the kernel migration due to the thick buffer layer and I-PyC. Kernel migration should be limited not to fail the SiC layer. In other words, the migration length is arrowed up to sum of the thickness of buffer and I-PyC layer with the margin of 35 μm evaluated by manufacturing tolerances. As a result, the allowable migration lengths are 55 μm and 100 μm, respectively for the HTTR A-type and EBU CFPs. With considering the mechanism which depends on only temperature condition and time, basically, the burn-up period is proportional to the allowable lengths of the kernel migration when it is limited by the kernel migration. Then, we selected EBU CFP to realize the longer burn-up period of 1,200 days. We are planning to three batches core as described later. The cycle length should be 400 days, which is the shortest interval of periodic inspection in Japan. In the future, the cycle length may be extended to 2 years if the maximum burn-up would be extended to 200 GWd/t and oxygen absorber such as ZrC (Bullock and Kaae, 1983) would be employed in the EBU CFP design.

<Table 1><Fig.1>

Horizontal geometry of HTTR core and core barrel

To satisfy the limitation for reactor vessel diameter due to the transport, one of the solutions is to employ the horizontal geometry of HTTR core. By this treatment, it is expected to save the project cost as well. Even though, the permanent reflector thickness, which is important to protect a reactor vessel from irradiation damage, should be reduced. Therefore, we employ the core barrel, which is developed for GTHTR300. The core barrel can hold the permanent reflector only with the thickness of 5 cm and realize the 10 cm thickness of coolant flow path because of the simple plate structure unlike the core restraint mechanism, which is employed in HTTR design and occupies 50 cm thickness and mechanically assembled. The reduction of the thickness of the reflector region can be minimized and the reactor vessel would be feasible from the viewpoint of irradiation durability. <Fig.2>

Fuel loading and discharging scheme without reloading and shuffling

In general, reloading and/or shuffling are performed when the spent fuel is discharged and fresh fuel is loaded. However, fuel reloading scheme should be performed with short duration and feasible. We have been investigated axial shuffling such as sandwich shuffling (Yan et al., 2003). In this scheme, the fuel blocks are separated into two batches in the axial direction. The fuel blocks of the fresh batch and old batch are piled alternately. However, the all fuel blocks should be removed to out of the core to refuel the bottom layer of fuel blocks, and long duration is necessary.

Therefore, we propose to employ the fuel loading and discharge scheme without reloading and shuffling. The scheme is illustrated in Fig. 3. In this scheme, the fuel columns are divided into several batches. The fuel columns in the batch spent life time are discharged from core, and fresh fuel columns are loaded there. As a result, there are several equilibrium cores as shown in Fig. 4. Three-batch core should be employed by considering symmetry with hexagonal fuel blocks. This scheme does not have difficulties of irradiation transformation, and the simplest and fastest compared with any other schemes.

<Figs.3 and 4>

Pressure vessel of Mn-Mo steel

To reduce construction cost, Mn-Mo steal is employed as vessel material instead of 2・1/4Cr-1Mo steel employed for HTTR due to high temperature durability. The Mn-Mo steel is employed by Light Water Reactor (LWR) as well, and cheaper than 2・1/4Cr-1Mo steel.

To be feasible the Mn-Mo steel, the inlet coolant temperature is reduced to 325 °C. With considering heating by radiant heat transfer, the temperature does not exceed 371 °C. In the code for pressure vessel (The American Society of Mechanical Engineers, 2017), which is determined by the American Society of Mechanical Engineers (ASME), high temperature structural design to consider creep rupture should be employed for SA508/533 (Mn-Mo steel) when the temperature exceeds 371 °C (700F). Therefore, there is no problem to use Mn-Mo steel under the temperature condition without the thickness margin for creep rupture.

Low purify nuclear graphite of IG-11

HTTR employs high purify nuclear graphite named IG-110 from the viewpoint of criticality. However, it is more expensive than the low purify nuclear graphite of IG-11. The low purity nuclear graphite is classified by the equivalent boron content from 2 ppm to 10 ppm. That is also classified by ash content from 300 ppm to 1,000 ppm (ASTEM, 2015). The equivalent boron content and ash content of IG-11 are 3 ppm and 500 ppm, respectively (Féron, 2012). The boron content depletes criticality of core, and the ash content activate oxidation by oxygen and/or moisture of air ingress accident or impurity of helium coolant in the normal operation condition. In this context, GTHTR300 employs IG-11 for fuel block and inner side of the inner reflector. The reflector facing fuel block employs IG-110 from the viewpoint of criticality (Fukaya et al., 2018). In the design of GTHTR300, the reactivity poison effect is treated as non-burnable poison material. Later, we researched the burnup dependency of the impurities and found the fact that it can be treated as burnable poison (Fukaya et al., 2018). In the general commercial design, it can be burned clearly, and the reactivity defect in End Of Cycle (EOC) is negligible (Fukaya et al., 2018). In other words, the 3 ppm of equivalent boron content of IG-11 can be burned clearly as well. Therefore, we propose to employ IG-11 for all blocks: fuel blocks, removable reflector blocks, and control rod guide blocks.

From the safety aspect, the oxidation of IG-11 is often regarded as problematic because of the higher oxidation rate (Kawakami, 1986). However, the oxidation of the blocks in depressurization accident is not problematized from the viewpoint of safety because the blocks never collapsed by the oxidation in the depressurization accident condition. With the accident condition described above, the thinning of graphite material occurs by the active oxidation. But the amount is small enough to sustain the core structure according to an analysis result. In addition, oxidation of graphite occurs even in the normal operation condition. Table 2 shows the upper limitation of impurity included in the helium coolant of HTTR (Saito et al. 1994). The oxidation type is passive oxidation which organize passive film on the surface of the graphite material. Therefore, thinning of the graphite material does not occur during the normal operation.

In addition, some impurity nuclides turn to radioactive nuclides, and it make decondition difficult. To solve this problem, counter measure may become a R&D subject.

With the technologies to realize high performance reactor, the major specifications are determined as listed in Table 3. In addition, the core height is determined by fuel inventory whose detail is described latter. And the power density is determined at the same time with the core volume and the thermal power. <Table 2 and 3>

3. Feasibility of Design and Detailed Specifications

To confirm the feasibility of proposed design and determine detailed specifications, evaluations are performed as described as follows.

3.1 Irradiation Damage of Pressure Vessel

As described above, the reduction of reflector thickness is minimized by employing the core barrel. Even though, the thickness of the reflector, including side reflector, is reduced by approximately 30 % as shown in Fig.2. That is important to moderate and shield fast neutron causing irradiation damage, and evaluated as difference between the radius of the permanent reflector and equivalent active core. Moreover, operation duration should be extended over 40 years with the load factor over 90 % from that of HTTR designed 20 years with the load factor of 60 %, and power density increases approximately 1.5 times. The irradiation damage may be problematic for the proposed reactor.

Then, the fast neutron fluence is evaluated in the pressure vessel using the one-dimensional neutron transportation calculation by collision probability method for radial direction in the module of Pij, which is named after first flight collision probability from the i-th to j-th region, of MOSRA-SRAC code (Okumura, 2015) with evaluated nuclear data of JENDL 4.0 (Shibata et al., 2011). By correcting with the result of detailed calculation for HTTR (evaluated as 8×10^{20} m^{-2} with the neutron energy over 1 MeV), the effect of power distribution can be considered even with the simple one-dimensional model as shown in Fig. 5. As same as HTTR, side shielding of B_4C-C composite is set out of core barrel. The limitation of fluence is determined to be 1×10^{23} m^{-2} in the guideline of material strength standards for HTGR (Terado et al., 1996).

The calculation results are listed in Table 4. The case with the life time of 60 years is also feasible with large safety margin according to the guideline of neutron fluence limitation of HTGR. It is also feasible if the lifetime is extended more in the future because of the large safety margin. The side shielding is not effective for the fast neutron. That is employed to absorb thermal neutron to protect the pressured vessel and equipment from the activation (Japan Atomic Energy Agency, 2006). However, there is no clear criteria to protect the vessel and equipment from the activation. Then, we evaluated the thermal neutron fluence (< 2eV) with varying B_4C composition of the shielding with the lifetime of 60 years. The original design of HTTR is 3 wt% of B_4C with natural boron. In addition, reactivity defect is evaluated as well. The result is listed in Table 5. To reduce the thermal neutron fluence to HTTR level, the B_4C composition should be increased to 20 wt%. Even though, the reactivity defect is merely 0.1 %Δk/kk'. It is acceptable compared with the design margin of 1.0 %Δk/kk' (Nakata et al., 2003).

<Tables 4 and 5> <Fig.5>

3.2 Safety for Depressurization Accident

The safety of a depressurization accident strongly depends on the core geometry, which determines a heat removal feature by thermal conductivity in core and radiant heat transfer to the silo wall, and power density, which determines decay heat. In other words, the design may not be feasible, the core geometry or power density should be changed. Then, we estimated the maximum fuel temperature in depressurization accident at this stage as shown in Fig. 6. The temperature was estimated by the one-dimensional thermal conductivity calculation and thermal properties set proposed to estimate depressurization accident (Sato et al., 2014). The calculation geometry is same as shown in Fig. 5. To evaluate decay heat, axial power peeking factor of 2.0 is assumed, and decay heat curve recommended by American Nuclear Society (American Nuclear Society, 2005) is used. For the initial fuel temperature condition, 1000 °C , which is lower than the nominal value of GTHTR300 of 1152 °C (1425 K) (Nakata et al., 2003) by considering the 100 °C lower outlet temperature and 30 % reduced power density, is assumed. To calculate the thermal conductivity equation, ANSYS code (ANSYS Inc., 2013), which solves the thermal equation by the finite element method with an implicit time integral technique, was used. The peak temperature 1567 °C appears at 10 hours after depressurization. It satisfies the limitation of 1600 °C. The design is feasible enough from the viewpoint of safety for a depressurized accident. <Fig.6>

3.3 Specifications of Fuel

The specifications of fuel blocks are listed in Table 6. The fuel block dimensions for horizontal direction is designed as same as HTTR's because the core diameter should be small due to the limitation of the transportation.

The core height is determined by fuel inventory as described above. To realize the target burn-up period of 1,200 days, the fuel inventory of 3.96 t is necessary by assuming the averaged burn-up of 50 GWd/t. To contain the fuel inventory with the packing fraction of 60 % and the fuel compact cross section area of HTTR, the core height of 10.5 m is necessary.

On the contrary, the core height is 3.6 times higher than HTTR, and coolant flow rate should increase 5 times larger along with the power increase. Pressure drop in core would increases to approximately 90 times (3.6×52) by simple estimation where it proportional to flow path length and to the square of coolant velocity. The core pressure drop should be reduced by revising the HTTR fuel design.

As described above, we employ SiC matrix fuel with the both sides directly cooling concept. The fuel compact dimensions are same as that of HTTR fuel compact. The fuel compact is surrounded by 1 mm thickness of support layer. The coolant flow area per fuel rod can be increased from 4.12 cm^2 (HTTR design) up to 7.55 cm^2 (over GTHTR300 design of 6.64 cm^2). With the flow area of 7.55 cm^2, the pressure drop can be decreased by approximately 70 %. The detailed dimension of the fuel element should be determined with considering maximum temperature and kernel migration in the stage of detailed design. <Table 6>

3.4 Graphite Block Durability for Irradiation

As described above, we employ the cycle length of 13 months according to Japanese regulation of the interval of periodic inspection. As a future option, the cycle length of 24 months become also a candidate if the long-life fuel technology would be developed. Those are corresponding to the burn-up period of 1200 and 2190 days, respectively with assuming the three-batch core.

From the viewpoint of graphite structure integrity, it is said that the irradiation should be lower than the irradiation at turnaround of the dimension change. The thermal conductivity reduces at a constant rate from the turnaround (Kunimoto et al., 2009). The irradiation, that is fast neutron fluence (> 0.1 MeV), of GTHTR300C, which is GTHTR300 for co-generation and is designed with the 75 % cycle length of GTHTR300, is reported 5×10^{25} m^{-2} (Shibata et al., 2010). The maximum fluences in graphite blocks are estimated $3.7\text{-}6.7\times10^{25}$ m^{-2} by correcting with the burn-up period and power density. The fluence at turnaround of IG-110 is reported 9.4×10^{25} m^{-2} at 900 ℃ (Kunimoto et al. 2009). The long fuel cycle option is feasible as well.

4. Summary

We designed conceptual commercial reactor with high performance. The detail is as follows. To improve safety performance, economy, and transportability, we employed:

- Confinement
  to enhance safety performance and to reduce construction cost.
- SiC matrix monolithic fuel
  to enhance oxidation tolerance, cooling function and to increase fuel inventory for longer cycle length.
- EBU CFP
  to enhance kernel migration durability for longer cycle length.
- Horizontal geometry of HTTR core
  to reduce the project cost and vessel diameter
- Core barrel
  to reduce irradiation damage of vessel by increasing reflector region.
- Fuel loading and discharging scheme without reloading and shuffling
  to avoid technical difficulties related to shuffling and to minimize the term of fuel loading and discharge.
- Pressure vessel of Mn-Mo steel

to reduce construction cost.

- Low purified graphite of IG-11

  to reduce fuel cost.

Moreover, we confirmed the feasibility of the design concerning to the following terms:

- Irradiation damage of pressure vessel
- Safety for depressurization accident
- Fuel design consistent to core specifications
- Graphite block durability for irradiation

As described above, we proposed the high performance commercial HTGR with the key technologies, and confirmed the feasibility.

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List of Tables and Figures

Table 1 Dimensions of CFPs

	HTTR A-type	EBU
Kernel diameter (μm)	600	500
Layer thickness		
Buffer (μm)	60	95
I-PyC (μm)	30	40
SiC (μm)	25	35
O-PyC (μm)	45	40
CFP diameter (μm)	920	920

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Table 2 Upper limitation of impurity included in the helium coolant of HTTR (vol ppm)

H_2O	CO_2	H_2	CO	CH_4	N_2	O_2
0.2	0.6	3.0	3.0	0.5	0.2	0.02

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Table 3 Major specifications of HTTR and proposal

	HTTR	Proposal
Thermal power (MWt)	30	165
Inlet temperature (°C)	395	325
Outlet temperature (°C)	850/950	750
Average power density (MW/m^3)	2.5	3.8
Core diameter / height (m)	2.3/2.9	2.3/10.5
Fuel block height (m)	0.58	1.05
Number of layer (-)	5	10
Fuel type	Sleeve	Monolithic
Material of fuel matrix	Graphite	SiC
Material of pressure vessel	2･1/4Cr-1Mo steel	59.30

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Table 4 Fast neutron fluence in pressure vessel

	Life time (year)	Load factor (%)	Fluence (m^{-2})
Case 1	40	90	1.3×10^{22}
Case 2	60	90	2.0×10^{22}

Table 5 Thermal neutron fluence in pressure vessel and reactivity defect

	HTTR	3wt%	10wt%	20wt%
Neutron Fluence (m^{-2})	2.1×10^{22}	4.3×10^{23}	5.4×10^{22}	1.8×10^{22}
Reactivity (%Δk/kk’)	-	0.0	-0.09	-0.13

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Table 6 Specifications of fuel blocks

	HTTR	GTHTR300	Proposal
Block length in across flat (cm)	36.0	41.0	36.0
Coolant hole diameter (cm)	4.10	3.90	< 4.10
Fuel rod diameter (cm)	3.40	2.60	2.8
Fuel rod inner diameter (cm)	-	-	0.8
Fuel rod pitch (cm)	5.16	4.70	5.16
Ligament (cm)	1.06	0.80	1.06
Fuel compact diameter (cm)	2.6	2.4	2.6
Fuel compact inner diameter (cm)	1.0	0.9	1.0
Coolant flow area per rod (cm^2)	4.12	6.64	$4.12 \leqq S \leqq 7.55$

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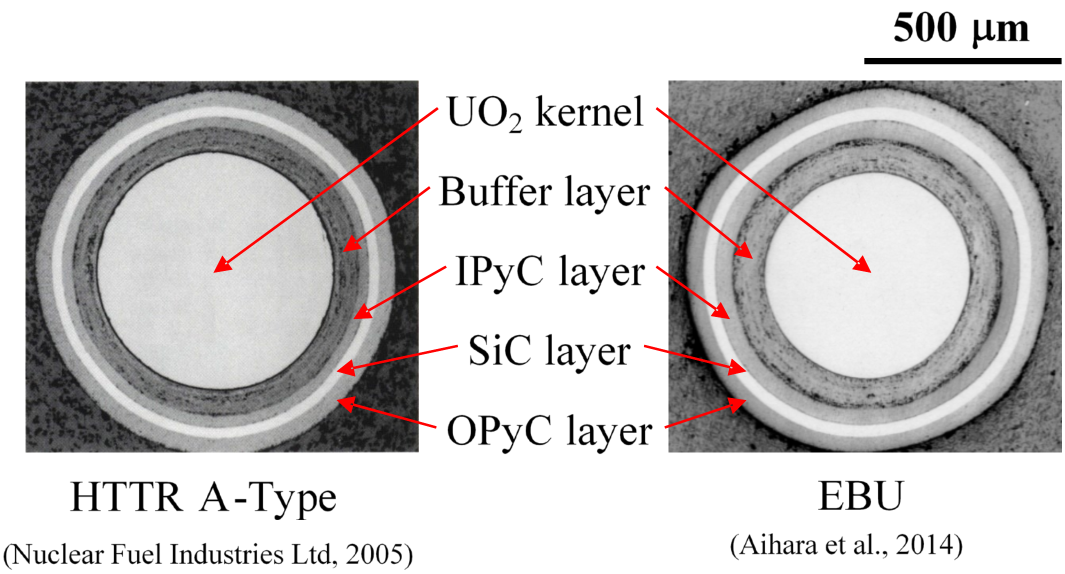

Fig.1. Geometry of CFPs

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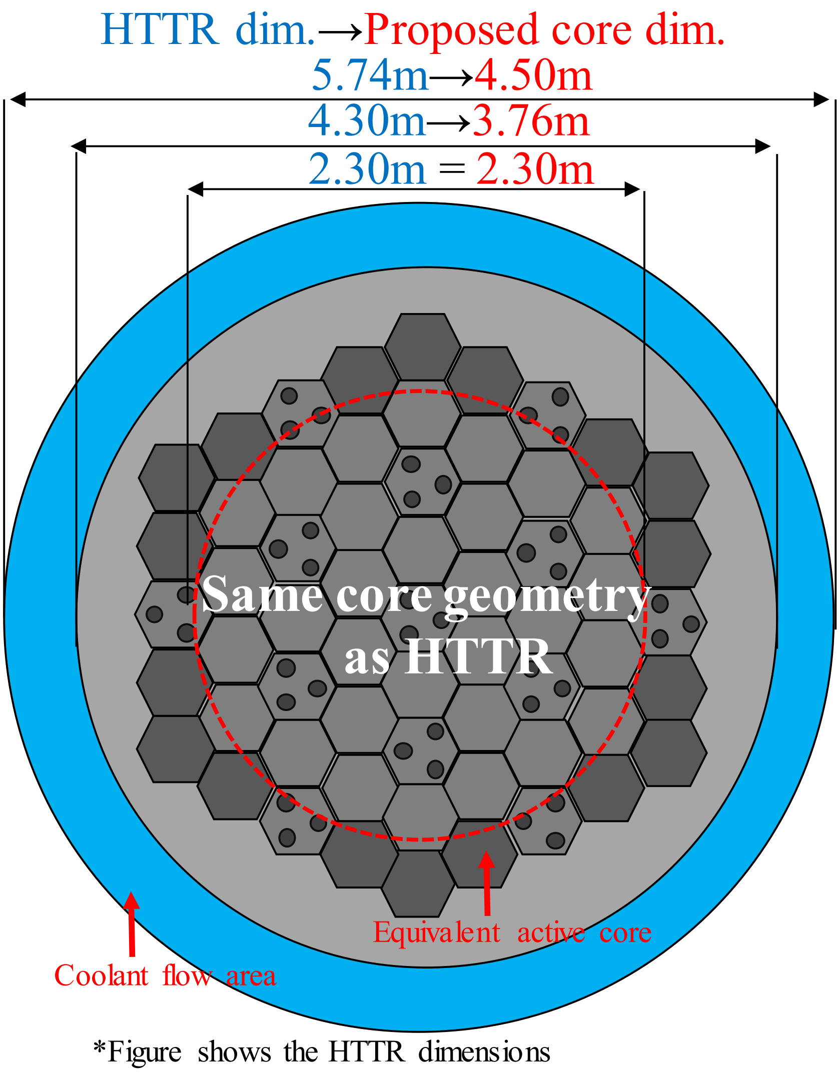

Fig.2. Change on dimensions from HTTR

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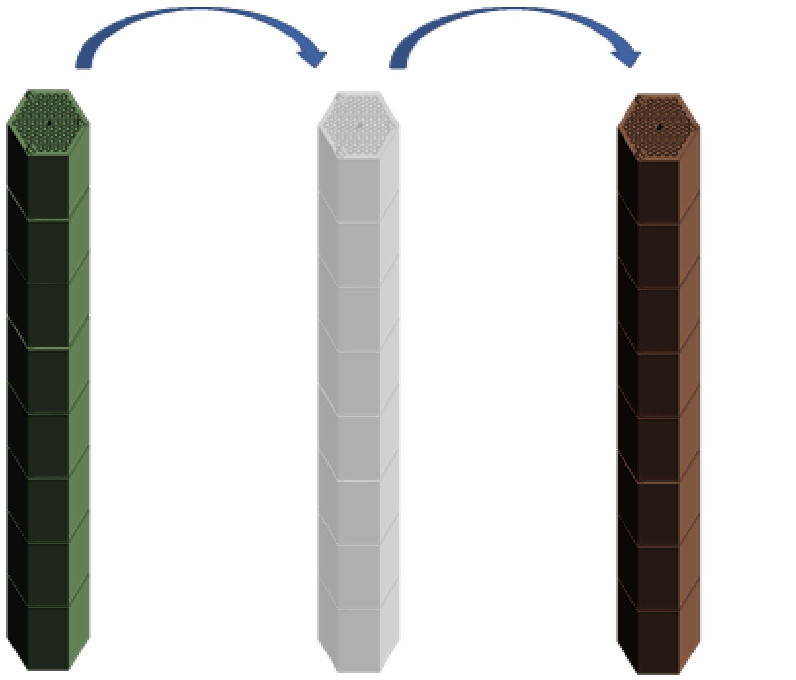

Fig.3 Fuel loading and discharge method

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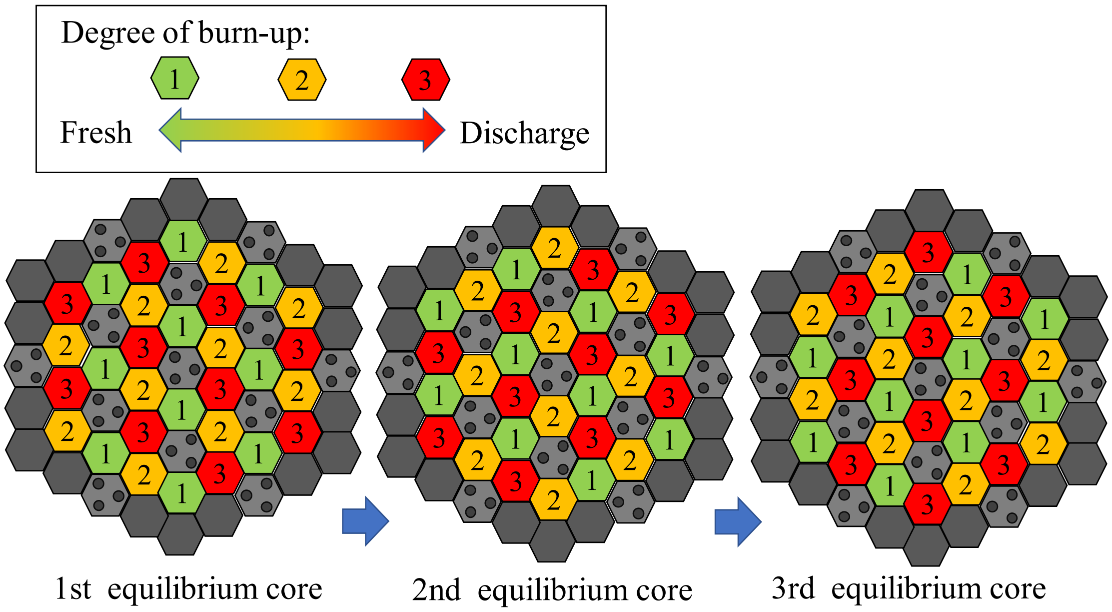

Fig.4 Equilibrium core of proposed scheme

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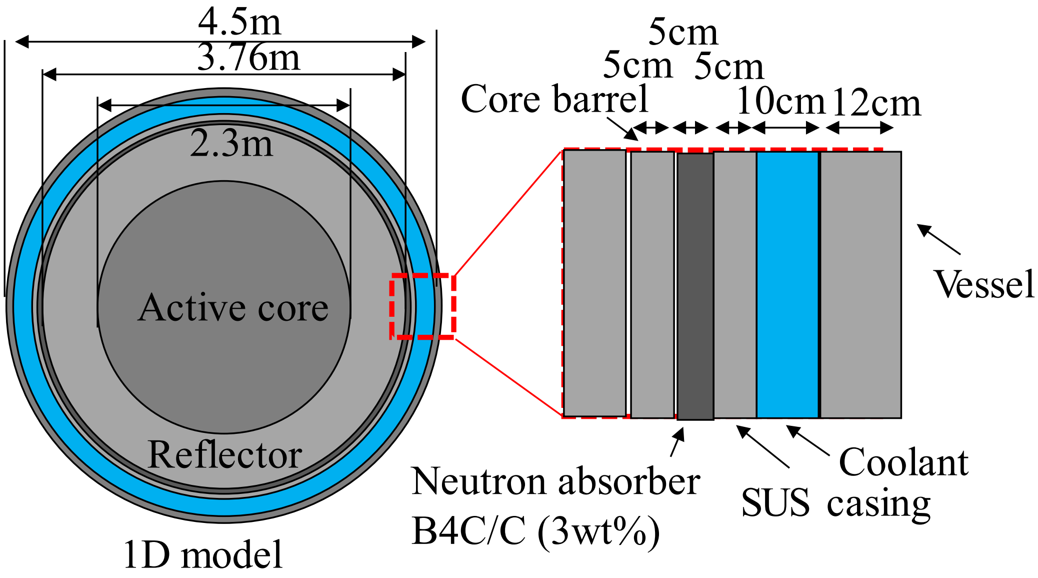

Fig.5 Irradiation damage evaluation model

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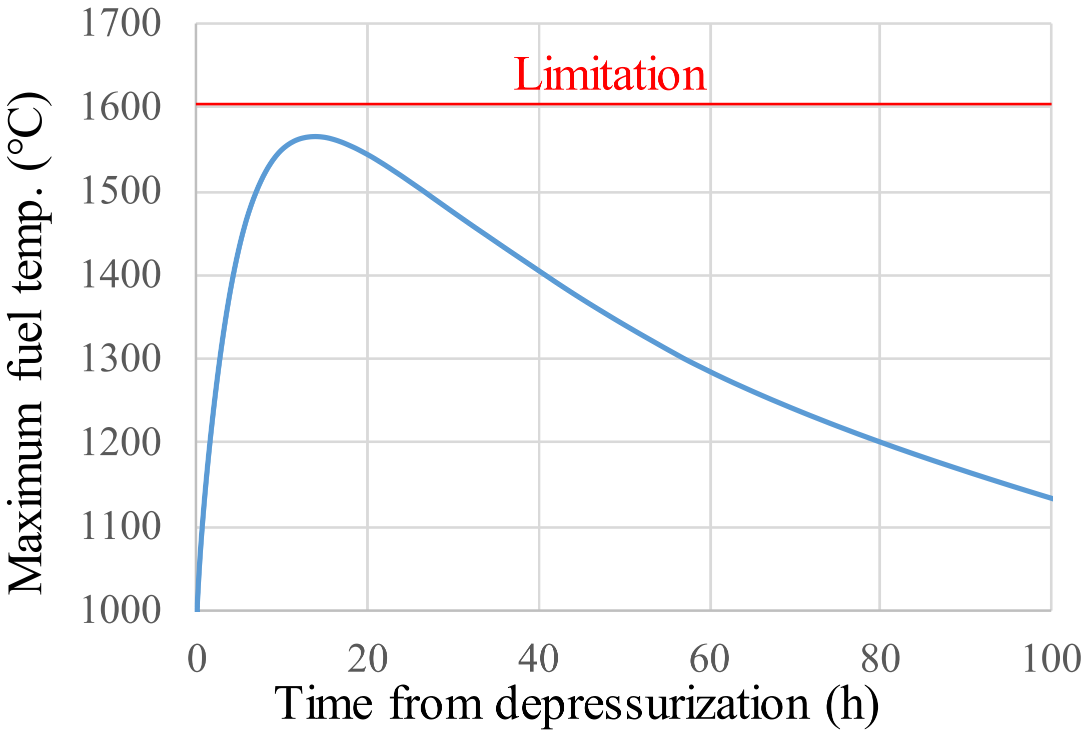

Fig.6 Maximum fuel temperature in depressurization accident

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