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Development of high temperature gas-cooled reactor (HTGR) fuel in Japan

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ABSTRACT

This paper describes experiences and present status of research and development works for the high temperature gas-cooled reactor (HTGR) fuel in Japan. Recently, Very High Temperature Reactor (VHTR) is evaluated highly worldwide, and is a principal candidate for the Generation IV reactor systems. In Japan, HTGR fuel fabrication technologies have been developed through the High Temperature Engineering Test Reactor (HTTR) project in Japan Atomic Energy Agency since 1960's. In total about 2 tons of uranium of the HTTR fuel has been fabricated successfully and its excellent quality has been confirmed through the long-term high temperature operation. Based on the HTTR fuel technologies, SiC TRISO fuel has been newly developed for burnup extension targeted VHTR. For ZrC-TRISO coated fuel as an advanced fuel designs, R&Ds for fabrication and inspection have been carried out in JAEA. The irradiation with the Japanese uniform stoichiometric ZrC coating has been completed in the cooperation with Oak Ridge National Laboratory of the United States.

1. Introduction

The High Temperature Gas-cooled Reactors (HTGRs), with its inherent safety and high temperature heat supply of about 1000 °C at the exterior of the reactor, can achieve effective utilization of nuclear energy in various fields by stages. For example, HTGRs make it possible to produce hydrogen with its high temperature heat supply. Hydrogen is expected as alternative energy source for oil near future. Therefore, HTGRs are expected to contribute to the reservation of the global environment and to provide a diverse energy supply (IAEA, 1986, JAERI, 1996).

In Japan, Japan Atomic Energy Agency (JAEA) has carried out the research and development of HTGRs since 1960's. The construction of High Temperature Engineering Test Reactor (HTTR) of 30 MW thermal power started in 1991 (Saito et al., 1994). The HTTR has attained full power with 850 °C of the outlet coolant temperature in December 2000 (Fujikawa et al., 2002), and with 950 °C in April 2004 (Fujikawa et al., 2004). Also, HTTR has been demonstrating the inherent safety features of HTGR by implementing the safety demonstration tests. As a recent status, in Japanese fiscal year 2009, the HTTR has performed the long-term high temperature operation during 50 days with 950 °C of reactor outlet coolant temperature, which was the first experience in the world HTGR history.

In the field of the HTGR fuel, two main fuel element concepts are presently in use, the spherical fuel element and the block-type fuel element as shown in Fig. 1. In both concepts, the high temperature heat supply and inherent safety features of the HTGRs are mainly achieved using refractory coated fuel particles. For current HTGRs, such as the HTTR at JAEA in Japan and the HTR-10 (Xu and Zuo, 2002) at the Institute of Nuclear and New Energy Technology (INET) of Tsinghua University in China, Tri-Isotropic (TRISO)-coated fuel particles with diameter of around 1 mm are employed. TRISO fuel consists of a micro spherical kernel of oxide or oxycarbide fuel and coating layers of porous pyrolytic carbon (buffer), inner dense pyrolytic carbon (IPyC), silicon carbide (SiC) and outer dense pyrolytic carbon (OPyC). The principal function of these coating layers is to retain fission products within the particle. Particularly, the SiC coating layer acts as a barrier against the diffusive release of metallic fission products and provides mechanical strength for the particle (Sawa et al., 2001a).

In Japan, HTGR fuel fabrication technologies have been developed by JAEA with the collaboration of Nuclear Fuel Industries, Ltd. (NFI) through the HTTR project since 1960's. NFI successfully fabricated many campaigns of irradiation test samples for the Japan Material Testing Reactor (JMTR), and the first and second loading fuels of the HTTR with in total about 2 tons of uranium. Its excellent

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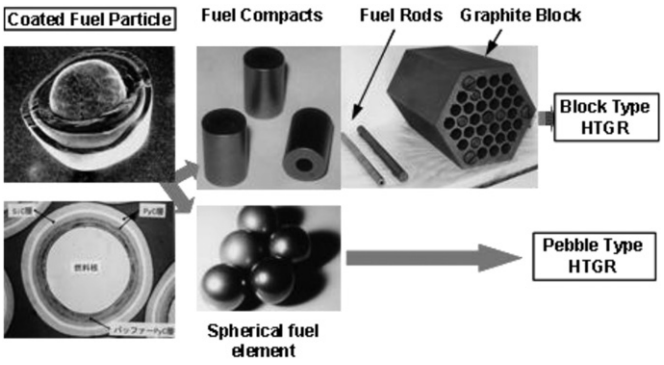

Fig. 1. High temperature gas-cooled reactor fuel.

quality was confirmed by quite low release-to-birth rate ratio, (R/B), of fission gases from the first loading fuel through the long-term high temperature operation (Ueta et al., 2010b).

As worldwide interests in the HTGR development, Very High Temperature Reactor (VHTR) is noticed as one of the most promising candidates for the Generation IV Nuclear Energy System (U.S.DOE, 2004). The VHTR reference concept is a helium-cooled, graphite moderated, thermal neutron spectrum reactor with an outlet temperature of about 1000 °C. The VHTR will produce both electricity with a gas turbine and process heat for diverse applications, including hydrogen production, process heat for refineries, petrochemistry, and metallurgy. The VHTR fuel must exhibit excellent safety performance up to accident temperatures of 1600 °C, burnups of about 15–20% fissions per initial metal atom (FIMA), and fluences of about 6×10^{25} n/m^2 ($E > 0.1$ MeV). For the concept of the VHTR mentioned above, the emphasis for the VHTR fuel R&D will be placed internationally on the feasibility of the standard SiC TRISO fuel and on the advanced designs that perform well at high burnup, power densities, and temperatures, including manufacturability, irradiation testing, fuel performance, and fuel behavior.

As Japanese R&Ds on the advanced HTGR fuel for HTGR such as VHTR, new SiC TRISO fuel for extended burnup has been fabricated based on the HTTR fuel technologies (Ueta et al., 2010a). Furthermore, replacement of the SiC layer by a zirconium carbide (ZrC) layer is known as a very promising example. JAEA has performed ZrC coating tests on surrogate kernels which represent the oxide fuel kernel to investigate the influence of coating parameters and material properties such as stoichiometry and density of ZrC (Ueta et al., 2008a, 2008b). These ZrC coated surrogate particles have been irradiated at High Flux Isotope Reactor (HFIR) in Oak Ridge National Laboratory (ORNL) in the United States in order to clarify the fundamental irradiation response of ZrC (U.S.DOE, 2009).

This paper describes experiences and present status of research and development works for the HTGR fuel in Japan.

2. Fabrication of the HTTR fuel

In safety design of the HTGR fuels, it is important to retain fission products within particles so that their release to primary coolant does not exceed an acceptable level. From this point, the basic design criteria for the fuel are to minimize the failure fraction of as-fabricated fuel coating layers and to prevent significant additional fuel failures during operation (Sawa et al., 2001a). The safety design criteria for the HTTR fuel were settled at first, and then, research and develop works were carried out to confirm safety characteristics of the fuel corresponding to the criteria. The inspection items were also determined to confirm specifications, which certify nuclear and thermal-hydraulic design, irradiation performance and so on (Sawa et al., 2001a). Table 1 shows major specifications applied in the fabrication of the HTTR fuel (Sawa et al., 1999).

By the commercial scale plant with the licensed capacity of 400 kgU/yr launched at NFI, fabrications of the first and second loading fuels of the HTTR were carried out from June 1995 to December 1997 and from October 2002 to March 2005, respectively (Sawa et al., 1999; Kato et al., 2001; Ueta et al., 2006). Fig. 2 depicts a flow diagram of the HTTR fuel production process, which consists of the fuel kernel, coated fuel particle and fuel compact processes. The UO_2 kernels were fabricated in a gel-precipitation process (Kato et al., 2001). After formation of uranyl nitrate solution containing methanol and an additive, spherical droplets are produced by a vibration dropping technique. The diameter of uranyl nitrate droplet is determined by the combination of the flow rate of metal solution and the frequency of the nozzles as indicated in Eq. (1):

$$Q = \left(\pi D^3/6\right) f \tag{1}$$

where, Q is flow rate of metal solution, D is diameter of droplet and f is frequency of vibrating nozzle. Following the drying and calcinating, reduction of the calcinated kernels to UO_2 was carried out. Kernel fabrication was completed by a sintering process to produce dense UO_2 kernels.

The coating layers were deposited on the kernels in a chemical vapor deposition (CVD) process using a fluidized coater (Kato et al., 2001). The buffer and high densed PyC coating layers were derived from C_2H_2 and C_3H_6, respectively, and the SiC layer from CH_3SiCl_3. The amount of charged particles was 3 kg uranium per one batch of coating. All UO_2 kernels and coated fuel particles are classified by

Table 1
Major specifications of the HTTR fuel.

Fuel kernel	
Diameter (μm)	600 ± 55
Density (g/cm^3)	10.63 ± 0.26
Impurity (ppm EBC[a])	≤3
Coating layers	
Buffer layer thickness (μm)	60 ± 12
IPyC layer thickness (μm)	30 ± 6
SiC layer thickness (μm)	25^{+12}_{-0}
OPyC layer thickness (μm)	45 ± 6
Buffer layer density (g/cm^3)	1.10 ± 0.10
IPyC layer density (g/cm^3)	$1.85^{+0.10}_{-0.05}$
SiC layer density (g/cm^3)	≥3.20
OPyC layer density (g/cm^3)	$1.85^{+0.10}_{-0.05}$
OPTAF[b] of IPyC and OPyC layers	≤1.03
Coated fuel particle	
Diameter (μm)	920^{+50}_{-30}
Sphericity	≤1.2
Fuel compact	
Coated fuel particles packing fraction (vol%)	30 ± 3
Impurity (ppm EBC[*1])	≤5
Exposed uranium fraction	$\leq 1.5 \times 10^{-4}$
SiC-failure fraction	$\leq 1.5 \times 10^{-3}$
Outer diameter (mm)	26.0 ± 0.1
Inner diameter (mm)	10.0 ± 0.1
Height (mm)	39.0 ± 0.5
Matrix density (g/cm^3)	1.70 ± 0.05
Compressive strength (N)	≥4900
Fuel rod	
Uranium content (g)	189 ± 6
Total length (mm)	577 ± 0.5
Fuel compact stack length (mm)	≥544

[a] EBC: Equivalent boron content.
[b] OPTAF: Optical anisotropy factor.

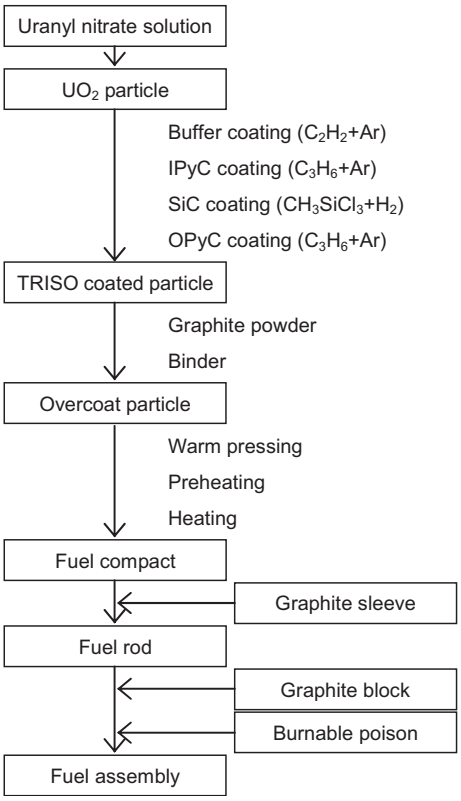

Fig. 2. Flow Diagram of HTTR Fuel Fabrication.

means of a vibrating table to exclude odd shape particles. The as-manufactured quality of the fuel has been improved by the modification of fabrication conditions and processes. The coating failure during coating process was mainly caused by the strong mechanical shocks to the particles given by violent particle fluidization that happened with a lot of force of spouting gas in the coater and by the unloading procedure of the particles. The coating process was improved by optimizing the mode of the particle fluidization and by developing the "continuous" coating process without unloading and loading of the particles at the intermediate coating process (Minato et al., 1997a).

The fuel compacts of the HTTR are produced by warm-pressing of the coated fuel particles with graphite powder (Kato et al., 2001). In the first step, the coated fuel particles are overcoated by resin-ated graphite powder with alcohol. The aim of the overcoating is to avoid direct contact with neighboring particles in the fuel compact. The thickness of overcoating layer is about 200 μm, which is determined by the specification for the volume fraction of the coated fuel particles in the fuel compact (30 vol%). Then the over-coated particles are warm-pressed by metal dies to form annular green fuel compacts. The final step of the compaction process is the heat-treatment of the green fuel compacts at 800 °C in flowing N_2 to carbonize the binder and at 1800 °C in vacuum to degas the fuel compacts. To avoid the direct contact with neighboring particles in the fuel compact and to reduce the defective particle fraction during the compaction process, the combination of the pressing temperature and the pressing speed of the overcoated particles was optimized (Minato et al., 1997a; Sawa et al., 1999).

In each campaign of the HTTR fuel fabrication with optimized fabrication process mentioned above, 0.9 ton of uranium was used, and 66,780 fuel compacts corresponding to 4770 fuel rods for 150 fuel assemblies were successfully produced. As results, as-fabricated fuel compacts contained almost no through-coatings failed particles and few SiC-defective particles as shown in Fig. 3. Fig. 3 shows histograms of the presence of the fuel compact containing through-coatings failed or SiC-defective particles. Abscissa of Fig. 3 shows the number of through-coatings failed or SiC-defective particles per one fuel compact, which were measured by acid-leaching or burn-leaching methods, respectively (Sawa et al., 1999). Fuel compacts have been fabricated with 126 and 128 fuel compact batches of the first and second loading fuels, respectively. Through-coatings and SiC-defective fractions were measured with 2 and 3 fuel compacts sampled from 1 fuel compact batch (,i.e., 252/256 and 378/384 fuel compacts were used for through-coatings failure/SiC-defective fractions of the first and second loading fuel, respectively). Average through-coatings and SiC-defective fractions for the first and second loading fuels were 2×10^{-6} and 8×10^{-5}, and 8×10^{-5} and 1.7×10^{-4}, respectively (Sawa et al., 1999; Ueta et al., 2006). It was concluded that these values were quite lower than the criteria, 1.5×10^{-4} for through-coatings failure and 1.5×10^{-3} for SiC-defective fraction, and the HTTR fuel has been fabricated successfully with high quality.

3. Performance of the HTTR fuel during the long-term high temperature operation

In the safety design requirements for the HTTR fuel, it is specified that "the as-fabricated failure fraction shall be less than 0.2%" and "the additional failure fraction shall be little through the full service period" (Sawa et al., 2001a). For the safe operation of the HTTR, the continuous and reliable measurement of the coolant activity is required to allow the evaluation of the fuel performance and the radiological assessment of the plant during normal operating conditions. The fission gases are released from the through-coatings failed particle (i.e., there are no intact layers) and from the uranium contamination in the fuel compact matrix (Ueta et al., 2003). Since the released fission gases do not precipitate on the inner surface of the primary coolant system piping, their concentrations in the primary coolant reflect the core average through-

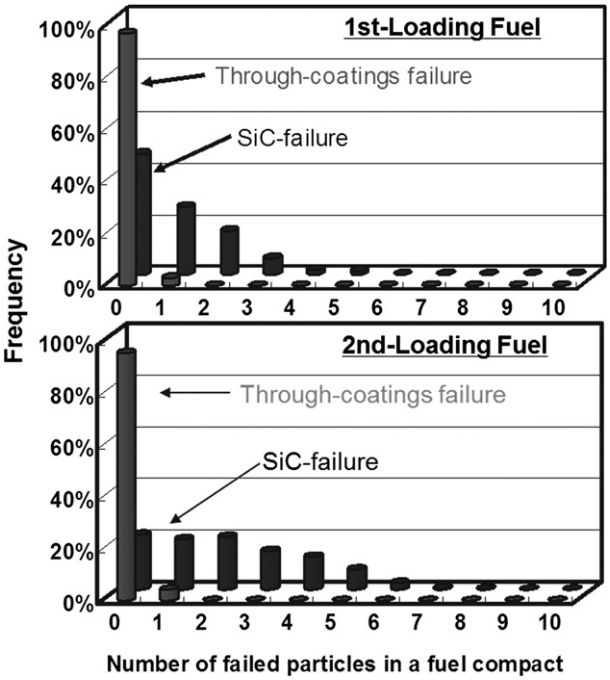

Fig. 3. Number of failed particles in a fuel compact in the first and second loading fuel fabrications.

coatings failure fraction and the fuel matrix contamination fraction. Therefore the failure fraction should be evaluated quantitatively during operation (Ueta et al., 2003).

In order to confirm the fuel integrity under severe irradiation conditions during the long-term high temperature operation of the HTTR, primary coolant sampling measurements were carried out (Ueta et al., 2010b). As results, concentrations of fission gas nuclides of ^{85m}Kr, ^{87}Kr, ^{88}Kr, ^{133}Xe, ^{135}Xe, ^{135m}Xe and ^{138}Xe were less than 0.1 MBq/m^3. Fig. 4 shows release-to-birth rate ratio, (R/B), of ^{88}Kr during this operation (Ueta et al., 2010b). Measured (R/B) values resulted less than 1.2×10^{-8}, how was by four orders of magnitude lower than the design limit for the normal operation, 1×10^{-4}, corresponding to 0.2% of fuel failure. According to the prediction by the fission gas release model (Ueta et al., 2003), it was suggested that the dominant mechanism of the fission gas release is the diffusion from the contaminated uranium in the fuel compact matrix, and the contribution of release from fuel kernel is relatively small. Finally, it was concluded that the HTTR fuel performed excellent quality during the long-term high temperature operation. The burnup in the end of this operation has reached around 11 GWd/t, although a little increase of the fission gas release was observed. It was suggested that the gap generating between fuel compact and graphite sleeve according to irradiation-induced shrinkage of fuel compact (Fukuda et al., 1989) would occur to make fuel temperature increasing.

By the way of the HTTR testing operation, the accelerated irradiation test with the HTTR first loading fuel by capsule irradiation in JMTR and its post-irradiation examination have been carried out to confirm the intactness under the irradiation (Sawa et al., 2001b; Ueta et al., 2007). Through these tests, it was concluded that no significant additional irradiation-induced failure occurred up to 6.1%FIMA (Fission per Initial Metal Atom) corresponding to about two times higher than the maximum burnup of the HTTR, which should also indicated the excellent properties of the HTTR fuel under the irradiation.

4. Advanced fuels

4.1. SiC-TRISO fuel for the burnup extension

In order to investigate fuel behaviors at extended burnup such as that of VHTR, irradiation tests have been performed by using so-called extended burnup fuel, whose target burnup and fast neutron fluence were higher than those of the HTTR fuel. In order to keep the fuel integrity against the internal pressure up to over 5%FIMA, thickness of buffer and SiC layers of the coated fuel particle were increased as shown in Table 2. As former researches of extended burnup fuels in JAEA, the fuel compact specimens were irradiated at the HFIR of ORNL up to 7%FIMA, and at the JMTR over 9% FIMA, respectively (Sawa and Tobita, 2003). Through these irradiation tests, it was suggested that the presumed failure mechanisms under high burnup condition were additional through-coatings failure of as-fabricated SiC-failed particles and/or an excessive increase of internal pressure by the accelerated irradiation, of which the results are well summarized by Sawa and Tobita (2003).

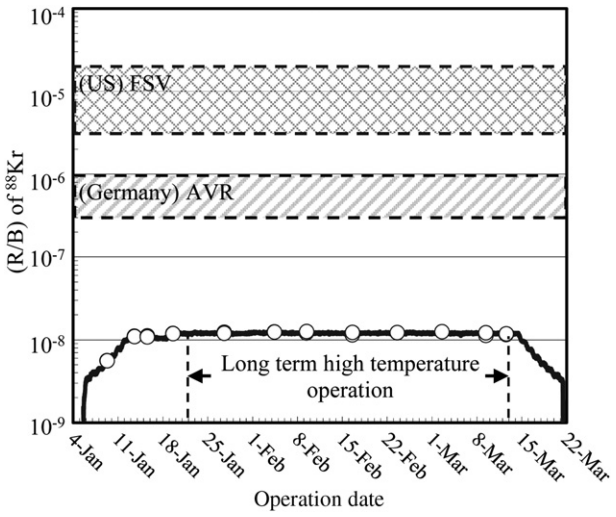

Fig. 4. Release-to-birth rate ratio (R/B) of ^{88}Kr during the long-term high temperature operation of the HTTR. Ranges of (R/B)s of German and US are referred from (IAEA, 1997).

To identify the fuel failure mechanism precisely, R&Ds on fabrication and irradiation with more highly qualified fuel specimen should be needed. In former researches, the coated fuel particles were fabricated by semi-commercial coater (2 kg uranium per coating batch) adopted the batch coating process, which caused to increase as-fabricate failure by handling (Minato et al., 1997a). And also, it is known that the mechanical strength of the coated layers during the irradiation depends on their thickness and sphericity. These characteristics strongly depend on the diameter and sphericity of the kernels. Therefore it is essential to establish the fabrication technology to obtain the kernels with more uniform diameter and excellent sphericity.

Based on these background, the new TRISO fuel for further extended burnup was designed as shown in Table 2, and the fabrication test was carried out with same process as the HTTR fuel using NFI's commercial scale coater (Ueta et al., 2010a). To manufacture the new HTGR fuel for extended burnup with high quality, some processes was optimized, those are, fuel kernel process and coating process. For the former, the vibrating nozzles emitted uranyl nitrate droplets were improved to make the fuel kernel with uniform diameter continuously. Then, Q and f in Eq. (1) as described in above section were optimized in order to make D smaller than that of the HTTR fuel. For the latter, the thickness and the homogeneity of the coating layer should be influenced by complicated parameters such as coating duration, concentrations of source gases, flow rate and so on. In order to obtain thicker and homogeneous buffer and SiC coating layers, the mode of the particle fluidization and the duration of each coating process were optimized appropriately, whereas the density of each coating layer was to be as same as that of the HTTR first loading fuel.

As results, the mean value of the diameter of the fuel kernel attained 504 μm, which agreed with the specification. For the coating process, neither as-fabricated through-coatings failure nor SiC-defective was observed in this test. Mean values of thicknesses of buffer/SiC resulted 97/34 μm, which also agreed with the specification. Especially, almost 100% of the coated fuel particle attained the sphericity less than 1.10, which was better than the former research ($\geq$1.11). Sphericity is defined as the ratio of maximum diameter to minimum diameter of the particle. Indicated by uniformity of the thickness of the coating layers, sphericity of the particle and so on, it was confirmed that the new fuel for extended burnup was manufactured with better quality than the former researches in JAEA.

Table 2
Major specifications of extended burnup fuel.

	New designed	Precursor researches	HTTR
Kernel diameter [μm]	500	500–550	600
Buffer layer thickness [μm]	95	90	60
SiC layer thickness [μm]	35	30–35	28
Target burnup [%FIMA]	$\geq$10	7–9	3.6

In the future, the irradiation test with this new designed TRISO fuel will be planned in order to examine its irradiation performance, which is scoped in the framework of the international cooperation, such as the Generation IV International Forum (GIF) and so on. As a recent status, the irradiation test with Japanese new designed fuel will be carried out under the research project organized by the International Science and Technology Center (ISTC) with National Nuclear Center (NNC) of the Republic of Kazakhstan from 2010 to 2013. In this project, several fuel compact specimens using the new TRISO fuel particle will be irradiated by an irradiation capsule at WWR-K reactor in NNC, targeted at high burnup level up to 100 GWd/t and at the range around 1100–1200 °C of irradiation temperature to scope VHTR irradiation conditions.

4.2. ZrC-TRISO coated fuel particle

ZrC is one of the transition metal carbides, which are characterized by (1) the good-compatibility with structural metals, (2) the high melting point and the thermodynamic stability, (3) the wear resistance etc (Ogawa et al., 2003). Studies of ZrC-TRISO coated fuel particles have been developed at JAEA since early 1970s. ZrC has the melting point at 3420 °C, but it eutecticaily melts with carbon at 2850 °C when it is used for the coated fuel particle. As a result of post-irradiation heating test, ZrC coated fuel particles did not fail until ~6000 s at 2400 °C, while almost 100% of SiC-coated fuel particles failed instantaneously at 2400 °C (Ogawa et al., 1992). Also, high potentials of the ZrC layer with regards to the retention and the chemical stability of the metallic fission product were demonstrated by post-irradiation heating tests (Minato et al., 1997b, Minato and Ogawa, 2003; Ogawa et al., 1992).

For the fabrication of the ZrC coated fuel particle, the bromide process was developed to produce a stoichiometric composition of ZrC, that is, C/Zr ratio of 1.0 (Ogawa et al., 1981). In the bromide process, the ZrC coating layer is deposited with pyrolytic reaction of $ZrBr_4$, CH_4 and H_2 at about 1500 °C in a fluidized bed. Main reactions of the bromide process can be described as follows (Ogawa et al., 1981):

$$[CH_4] = (C) + 2[H_2] \quad (2)$$

$$[ZrBr_4] + [H_2] \rightarrow [ZrBr_3],\ [ZrBr_2] + [HBr] \quad (3)$$

$$[ZrBr_x] + (C) = (ZrC) + x[Br],\ x = 2, 3, \text{or } 4 \quad (4)$$

$$[Br] + 0.5[H_2] = [HBr] \quad (5)$$

$$0.25(C) + [HBr] = 0.25[CH_4] + [Br] \quad (6)$$

(In above equations, solid and gas phases are expressed in parentheses and square brackets, respectively.) Then, coating conditions to obtain ZrC with uniform structure and with good stoichiometry were acquired by a small-scale coater with 10 g-batch size. On the other hand, it was also revealed that the stoichiomety of ZrC might be sensitive to the coating condition because of the complex reaction. When the scale of the coater is enlarged, i.e., the volume of a fluidizing bed becomes large, the temperature distribution of coating region may change more easily. It would affect the stoichiometry of the ZrC layer.

Based on these backgrounds, the following investigations have been newly carried out from 2004 to 2009 to establish the technology of fabrication and inspection methods for the ZrC coated fuel particle at JAEA: (1) development of ZrC coating process with a large-scale coater with 0.2 kg batch size, (2) development of inspection method for ZrC coated particles, especially with regard to C/Zr ratio (stoichiometry), density of ZrC coating layer and ZrC failure fraction, and (3) irradiation test and post-irradiation examination of ZrC coated particles to characterize irradiation performance of the ZrC coating layer under the VHTR condition. These results have made the ZrC coating layer an interesting candidate to replace SiC for the advanced fuel particle (Ueta et al., 2008a).

For developing ZrC coating process with a large-scale coater, JAEA has newly constructed a ZrC coater with 0.2 kg batch scale. ZrC was deposited on dummy zirconia particles with weight close to UO_2 fuel particles. At first of the study, non-uniform stripes were observed in the deposited ZrC. It was found by transmission electron microscopy observation that segregated excess free carbon made the non-uniformity. Uniform ZrC was successfully deposited by controlling ZrC depositing temperature to be 1300–1400 °C in order to prevent excessive pyrolysis of methane (Ueta et al., 2008a; Yasuda et al., 2008). Also, advanced techniques to coat both ZrC and outer pyrolytic carbon continuously on the particle have been developed (Yasuda et al., 2009). In parallel, for the inspection of ZrC coated particles, especially a measuring technique for ZrC stoichiometry was newly developed by conducting both inductively coupled plasma – atomic emission spectrometry (ICP-AES) and infrared absorption spectrometry, enabling measurement of C/Zr atom ratio with high accuracy on the order of 0.01 (Ueta et al.,

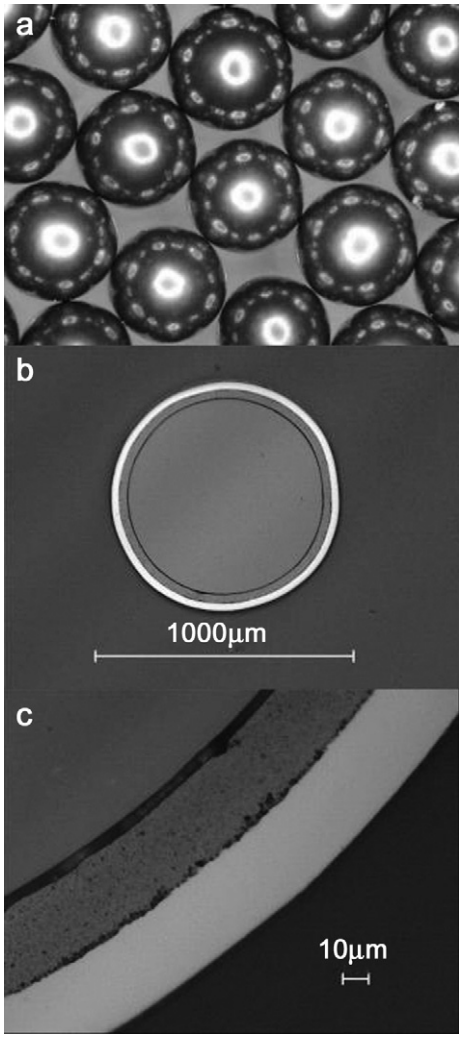

Fig. 5. Photos of ZrC coated particles, (a) appearance and (b)(c) cross sections.

2008b). Finally, the fabrication conditions for stoichiometric ZrC were successfully determined, and highly qualified stoichiometric ZrC coating has been obtained as shown in Fig. 5 (Ueta et al., 2008b).

Also, the irradiation test and the post-irradiation examination with Japanese CVD ZrC coated particle was carried out in the cooperation with ORNL under the framework of International Nuclear Energy Initiative (I-NERI) up to March 2010. In this irradiation test, ZrC coated particles (ZrC with C/Zr = 1.03) was used. ZrC coated particles treated post fabrication heating at 1800 °C for 1 h were used in the test to simulate particle thermal history during compact formation. The ZrC coated particles were irradiated in HFIR of ORNL at 800 °C and 1250 °C to 2 dpa ($\sim 2.12 \times 10^{25}$ neutrons/m^2), and at 1250 °C to 6 dpa ($\sim 6.72 \times 10^{25}$ neutrons/m^2). As results, a comparison of the irradiated specimens by SEM images indicated that the samples irradiated at 6 dpa had suffered less microstructural damage than the 2 dpa samples, although this could be attributed in part to the induced microstructural damage during handling of the two-dpa samples (U.S.DOE, 2009). Also, almost no change was observed on the strength of the ZrC coated particle by fracture test before and after the irradiation. Although Young modulus of the ZrC tended to slightly decrease as the neutron fluence increased, of which the level would not be significant for the irradiation-induced stress generated in the coating layer. Finally it was confirmed that no significant irradiation damage to the microstructure and mechanical properties such as fracture strength and Young modulus of CVD ZrC under the VHTR irradiation condition has been obtained, which will contribute to construct the ZrC fuel performance model.

5. Summary

This paper describes research and developments for the HTGR fuel in Japan. R&Ds for SiC TRISO fuel fabrication technologies through the HTTR project are described. Through the long-termed high temperature operation the excellent quality of the HTTR fuel as Japanese commercial products is confirmed. As an advanced HTGR fuel such as VHTR for burnup extension, fabrication of the new designed SiC TRISO fuel based on the HTTR fuel technologies is described. This new TRISO fuel will be irradiated under the ISTC framework to confirm the integrity under the irradiation condition targeted VHTR. In addition, R&Ds of ZrC coated particle are described to obtain uniform CVD ZrC coating and to investigate the influence of coating parameters and material properties such as stoichiometry and density of ZrC. For clarifying the fundamental irradiation response of ZrC, properties after irradiation of Japanese CVD ZrC particles investigated through the irradiation program under the I-NERI framework is described. These results will contribute to the development of the VHTR.

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