Current Status of the Next Generation Fast Reactor Core & Fuel Design and Related R&Ds in Japan

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Abstract

The next generation sodium-cooled fast reactor is being investigated in Japan, with the aim of achieving several targets such as “safety”, “reduction of environmental burden” and “economic competitiveness”. As for the safety aspect, the fuel subassembly with inner duct structure (FAIDUS) concept is adopted to avoid re-criticality in core destructive accidents. The uranium-plutonium mixed oxide (MOX) fuel, in which minor actinide (MA) elements are included, will be applied to reduce the amount and potential hazards of radioactive wastes. The high burn-up fuel is being pursued to reduce fuel cycle costs. The candidate concept of the core & fuel design has been established, which can satisfy various design criteria by such approaches as a high internal conversion concept using large-diameter fuel pins. In addition, JAEA is developing oxide dispersion strengthened (ODS) ferritic steel cladding for the high burnup fuel, and JAEA is also investigating fuel material properties and irradiation behavior of MA-bearing MOX fuel. JAEA is developing a fuel design method, especially suited for fuel pins with annular pellets.

Key words: FR demonstration reactor, fuel design, MA-bearing MOX fuel, ODS

1. Introduction

The development of the sodium-cooled fast reactor (FR) and its related fuel cycle is being promoted in Japan to ensure energy security, to use uranium (U) resources efficiently and to reduce the amount and potential hazards of high-level radioactive wastes. In the FR cycle technology development (FaCT) project, design investigation and various R&Ds were conducted with the aim of realizing the FR and its fuel cycle [1]. The roadmap toward a FR demonstration reactor and a commercial reactor will be made in the near future using the experiences of Joyo operation and Monju construction and other activities.

This paper describes an actualized core & fuel design concept for a next generation FR demonstration reactor, which aims at contributing to construction of the safety design criteria, and the present status of its related R&Ds such as core materials development, fuel material properties research and fuel design techniques development.

2. Status of core & fuel design for the next generation fast reactor

2.1 Concept of core & fuel design and its targets

Design targets related to core & fuel design of the FR demonstration reactor [2] are summarized in Table 1. The electric output of the FR demonstration reactor is set to be 750 MWe, i.e., one half of that of a commercial reactor; this will be used in confirming its technical and economic performance. The breeding ratio is assumed to be 1.1 to demonstrate the capability as a sustainable energy source. Main characteristics of the core & fuel concept are described in the following four subsections.

2.1.1 Safety requirement

The countermeasures for severe accidents have become an essential issue, based on the lessons learned in the 2011 Fukushima-Dai-ichi nuclear power plant accident. The design target of a maximum sodium void reactivity in a fissile core region is provisionally set to be about 6

$ or less on the basis of the FR safety evaluation experience to avoid prompt criticality in the early phase of core disruptive accidents (CDAs).

The innovative fuel subassembly with inner duct structure (FAIDUS) concept is adopted to prevent the re-criticality in the transition phase of CDAs [3]. In the early phase of CDAs, molten fuel with pressurized fission product gases will burst through an inner duct and will be discharged upward to prevent the formation of a large-scale molten fuel pool which is one of factors leading to a severe power excursion.

2.1.2 Reduction of environmental burden

The fission cross sections of U, plutonium (Pu) and minor actinide (MA) nuclides are mostly higher than the capture cross sections of these nuclides in a FR neutron spectrum which has plenty of high-energy neutrons in comparison with a light water reactor neutron spectrum. Therefore, the isotopic composition of fuel nuclides will not be degraded even if fuel nuclides are multi-recycled in a homogeneous FR cycle system. Trans- uranium (TRU) nuclides including MAs can be circulated and managed in the FR fuel cycle system without discharging them outside the system. The heat generation and potential hazards of high-level radioactive wastes can be reduced by eliminating MAs from the wastes. An effective MA recovery system and remote fuel fabrication system in a hot cell are prerequisite techniques for utilization of MA-bearing fuel. The conventional U-Pu mixed oxide (MOX) fuel will be used in the early stage of the FR demonstration reactor operation until these techniques are established.

2.1.3 Economic competitiveness

The high burnup fuel reduces fuel cycle costs by lessening the amount of nuclear fuel materials circulating in fuel fabrication and in reprocessing. From this point of view, it becomes important to enhance the burnup not only of a fissile part, but also of parts merged with a fertile part. The target burnup for fuel comprised of the fissile part and the fertile one is 60 GWd/t or more. In this case, the core discharged average burnup of the fissile part becomes approximately 150 GWd/t. The target coolant temperature at the reactor outlet is 550 ºC to achieve high generating efficiency.

2.1.4 Fuel concept

The enhanced burnup may pose thermal-hydraulic challenges because the extended power dispersion between fresh fuel and burned fuel causes an increase of the necessary coolant flow. The fuel concept using large-diameter pins with annular pellets is adopted to attain compatibility of the high burnup and the high coolant outlet temperature. The high internal conversion ratio of the concept with high fuel volumetric composition enables control of power fluctuation caused by the high burnup extension. Furthermore, the high internal conversion also enables reduction of the amount of surrounding fertile parts which is necessary to attain the target breeding ratio. This reduction of fertile parts enhances the discharged average burnup including fertile parts.

Ferritic steel which experiences small dimensional changes by neutron irradiation is applied for the cladding and wrapper tube, taking into account an increase of fast neutron fluence. The oxide dispersion strengthened (ODS) ferritic steel with high mechanical strength at an elevated temperature is adopted for the cladding materials to prevent the creep rupture by internal pressure of the fuel pin. The maximum fast neutron fluence (E>0.1 MeV) is approximately 5×10^{23} n/cm^2 (250 dpa) or less and the maximum cladding temperature is 700 ºC or less, taking into account development goals.

Fuel smear density, which is a bulk density averaged within a cladding inner surface, is set at about 82%T.D. to preclude excessive fuel-cladding mechanical interaction (FCMI) that may occur by increased fuel swelling owing to the high burnup. Annular pellets, which have excellent thermal performance, are adopted to prevent fuel melting in a power transition condition.

Table 1 Design targets and design results related to core & fuel design

Items	Design targets	Design results
Plant & core specifications		
Power output	750 MWe/1765 MWt	←
Primary coolant temperature (outlet / inlet)	550 ℃ / 395 ℃	←
Primary coolant flow rate	9000 kg/s	←
Operation cycle length	> 13 EFPM	18 EFPM
Number of fuel loading cycle	-	6 cycles
Discharged average burnup (fissile + fertile)	> 60 GWd/t	83 GWd/t
Breeding ratio (standard core condition)	about 1.1	1.1
Core height	100 cm	←
Upper / lower blanket height	-	20 / 25 cm
Max. sodium void reactivity	< about 6 $	5.2 - 6.0 $
Fuel specifications		
Subassembly pitch	-	206 mm
Number of pins /subassembly	-	255
Fuel pin diameter	-	10.4 mm
Fuel smear density	82 %T.D.	←
Fuel pellet density	-	95 %T.D.
Pu content (average)	-	22.1 - 24.7 wt%
MA content (average)	< 3.0 wt% < 5.0 wt% (local)	0.2 - 3.0 wt%
Cladding thickness	-	0.71 mm
Wrapper tube thickness	-	5.0 mm
Materials of cladding tube	-	ODS
Materials of wrapper tube and inner duct	-	PNC-FMS
Max. linear heat rate	< 430 W/cm	376 - 392 W/cm
Max. fast neutron fluence	< about 5×10^{23} n/cm^2	5×10^{23} n/cm^2
Max. cladding temperature	700 ℃	←
Max. bundle pressure drop	< about 0.2 MPa	0.2 MPa

2.2 Results of core & fuel design study

2.2.1 Core specifications and core configuration

Core & fuel specifications were determined in the design study [4] and they are also summarized in Table 1. The core configuration is shown in Figure 1. The core is comprised of 278 core fuel subassemblies and 66 blanket fuel subassemblies. The standard neutronic and thermal-hydraulic analysis methods are applied in the core characteristic evaluation. The core characteristics were evaluated for 6 types of fuel isotopic compositions, taking into account the composition trend of fuels supplied to the FR demonstration reactor during its lifetime [2]. The fuel isotopic compositions include U & Pu composition (fuel type: (U, Pu) O_{2-x}), U & TRU composition (fuel type: (U, Pu, Np, Am, Cm) O_{2-x}), both of which are recovered from LWR spent fuels, and U & TRU composition, which is recovered from LWR and FR spent fuels in the multi-recycle system.

The discharged average burnup (fissile + fertile) reaches about 83 GWd/t under the conditions of an operation period of 18 EFPM and 6 fuel exchange batches. The core discharged average burnup becomes about 148 GWd/t. Pu content averaged in the core region is in the range from 22 wt% to 25 wt% and averaged MA content is in the range from 0.2 wt% to 3.0 wt%.

The maximum sodium void reactivity in the core region stays below 6 $ even if the high burnup and MA inclusion, which have a tendency to increase the sodium void reactivity, are taken into account. The reactivity change with burnup in the high fissile fuel composition becomes larger than that in the low fissile fuel composition. To compensate for the increased reactivity change, primary control rods will be inserted deeply at the beginning of operation cycles in the case of the high fissile fuel composition. The fluctuation of control rod insertion depth may cause an increased power distribution distortion in the core. The control rods are arranged suitably to minimize the change of core power distribution. An upper internal structure (UIS) with a slit, through which an arm of a fuel handling machine is driven, is adopted in order to downsize the reactor vessel and to shorten the fuel exchange period. The UIS and fuel handling system require zones in which control rods cannot be arranged. Four flux adjusters, i.e., a core insertion structure without fuel, are newly set at the zones to mitigate their power peaking.

A self-actuated shutdown system (SASS) is installed in some backup control rods (BCRs) to enhance passive safety [5]. An absorber rod is suspended using a temperature-sensing electromagnet in the SASS. When coolant temperature rises in the case of an anticipated transient without scram (ATWS), the electromagnet force is lost and the absorber rod drops into the core automatically.

The necessary coolant flow of the fuel subassembly is determined as the maximum flow to keep the maximum cladding temperature below 700 ºC or to keep the cumulative damage fraction (CDF) of cladding below 0.5 in normal operation. The limit of cladding CDF is set so that it does not reach creep rupture even if CDF progression is taken into account during anticipated operational occurrences and spent fuel handling. The necessary coolant flow is minimized by adoption of the high internal conversion concept with large-diameter fuel pins, the optimized arrangement of control rods, the adoption of flux adjusters, an adequate allocation of coolant flow zones, and a rationalization of the hot spot factor in the cladding temperature evaluation. The total coolant flow is concluded to be 9000 kg/s or less, which includes additional flow for control rods, neutron shielding, and inspection holes. Therefore, the target coolant outlet temperature, i.e., 550 ºC will be achieved by the well-organized core design.

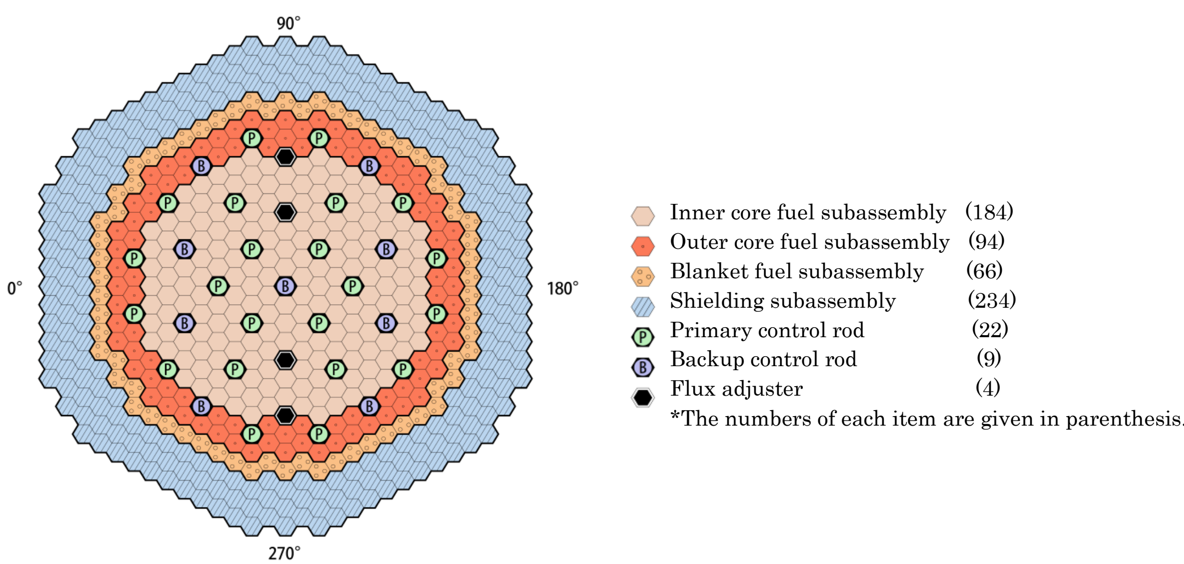

Figure 1 Core configuration of the next generation SFR in Japan.

2.2.2 Fuel specifications

Figure 2 offers a schematic view of fuel subassembly structure. The large-diameter fuel pin (10.4 mm diameter) is adopted to achieve the high internal conversion. Annular pellets made from MOX or MA-bearing MOX fuel are adopted to enhance the thermal performance in view of prevention of fuel melting. The maximum linear heat rate is 400 W/cm or less at the rated thermal power output and it satisfies the provisional limit 430 W/cm. The limit is configured considering a power excursion in the transient condition and the influence of MA inclusion on fuel material properties. The acceleration of helium (He) gas generation due to the decay of Cm-242 transmuted from Am-241 with a neutron capture reaction should be considered in the case of the TRU fuel composition. The length of a gas plenum is set so that fuel pin internal pressure will not become higher excessively due to the fission product gases and He gas. The gas plenum is located under the fissile fuel column so that the fuel pin length is shortened. The hydraulic pressure drop of the fuel pin bundle becomes less than 0.2 MPa owing to the shortened fuel pin and the moderated coolant flow rate. The low pressure drop of the fuel pin bundle is effective for removing the decay heat by natural circulation. The ODS cladding is connected to an upper end plug and a lower end plug, which are also made of ODS, by a pressurized resistance welding (PRW) technique developed at JAEA [6]. The fuel pin bundle is comprised of 255 fuel pins, which is the number left when 16 corner pins are subtracted from the 271-pin structure. The rhombic inner duct is installed in the space of the 16 corner pins of the fuel subassembly. The wrapper tube and the inner duct are made of ferritic / martensitic steel (PNC-FMS). In the early phase of CDAs, molten fuel breaks through the inner duct wall and is discharged upward through the remaining duct space by the elevated pressure of the molten fuel mixture with accompanying fission product gases and other vapors. This fuel dispersion prevents re-criticality in the transition phase of the CDA scenarios. The wall thickness of the inner duct is set to be thinner than that of the wrapper tube to ensure the molten fuel ejection before the formation of the molten fuel pool. An upper neutron shield is installed above the fuel pin bundle to reduce the neutron fluence of the UIS. The ejection path is grooved in the upper shield so as not to hamper the molten fuel flow. Orifice holes are bored in the entrance nozzle to control the coolant flow rate adequately in combination with the slits of the core support plate. A self-orientation structure is carved in the handling head and the entrance nozzle, which is also adopted in the Monju fuel subassembly. When the fuel subassembly is exchanged, it is inserted into the hexagonal space formed by the surrounding fuel subassemblies. The fuel subassembly automatically rotates when the spikes of the entrance nozzle slip down along slopes in the handing heads of surrounding fuel subassemblies. The mechanical connecting structure is configured to absorb the dimensional change caused by the thermal expansion difference between the austenitic steel handling head and the ferritic steel wrapper tube. Pads are arranged in the handling head, the wrapper tube and the entrance nozzle to prevent the direct contact with adjacent fuel subassemblies except for the pads. The pad shape is designed so that it does not prevent coolant flow in the case of natural circulation.

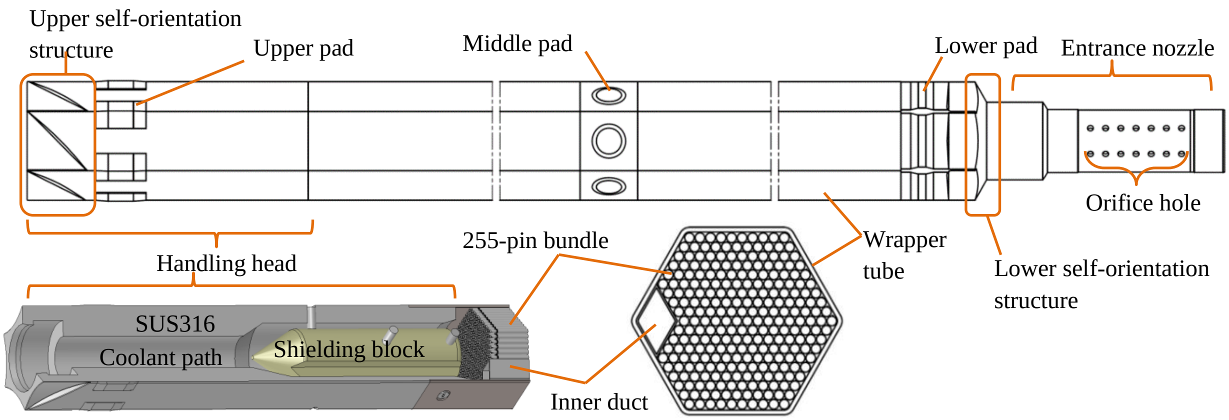

Figure 2 Structure of fuel subassembly with FAIDUS concept

3. Current status of FR fuel development

Japan Atomic Energy Agency (JAEA) developed MOX fuels for the FR step by step through experiences with the experimental reactor Joyo and the prototype reactor Monju. Various irradiation experiments have been conducted in Joyo and the many data obtained in these experiments have contributed to development of fuel design methods. Thus, the technical development of fuel for the next generation FR is in progress. The current status of main fields of the technical development is explained hereafter.

3.1 Development of core materials

JAEA has developed ODS ferritic steel for a long-life cladding tube, which is resistant to both fast neutron dose of 250 dpa and temperature of 700 ºC [7]. PNC-FMS (11Cr ferritic-martensitic steel) has been developed for wrapper tubes used at a lower temperature than cladding, i.e. < 600 ºC in normal operation [7]. Figure 3 shows a schematic view of the technological level of ODS ferritic steel cladding tube development within JAEA. The material development has been completed, and work is now in the transition stage to technology development for practical application. Irradiation tests of several fuel pins with 9Cr-ODS steel cladding tubes have been conducted in BOR-60 to demonstrate in-reactor performance of 9Cr-ODS steel and to accumulate in-pile data for preparation of a material strength standard [8],[9], where peak burnup and peak neutron dose were 112 GWd/t and of 51 dpa, respectively. Superior properties of the ODS claddings were confirmed such as fuel compatibility, dimensional stability, and integrity of a PRW part. However, in one of the irradiation tests, an irradiated fuel pin ruptured due to heterogeneity (metallic Cr inclusions) of the 9Cr-ODS steel cladding tube and exposure to higher temperature than designed value (> 700 ºC) [9], [10]. In response to these results, JAEA modified the fabrication process to suppress formation of metallic Cr inclusions in the tube. The modified process is designated as “the full pre-alloy process”, which uses only pre-alloyed powder and Y_2O_3 powder as the raw material powder (i.e., no elemental powder is used in the process) and it is carried out under the strict management of the powder metallurgy process. Two lots of 9Cr-ODS steel cladding tubes were successfully fabricated by the full pre-alloy process without any cracking during cold-rolling. It was confirmed that there were no metallic Cr inclusions and no detrimental heterogeneity in the manufactured tubes. This modification also achieved the reduction of non-metallic inclusions, thereby increasing creep strength of the 9Cr-

ODS steel cladding tube. Currently, out-of-pile mechanical tests including long-term creep rupture tests are in progress on the full pre-alloy 9Cr-ODS steel cladding tubes. Neutron irradiation tests of these cladding tubes are planned in Joyo. Towards the development of a mass production process, JAEA restarted the R&Ds to scale up the mechanical alloying process and mother tube size. Technological knowledge on the full pre-alloy process will be incorporated into this development.

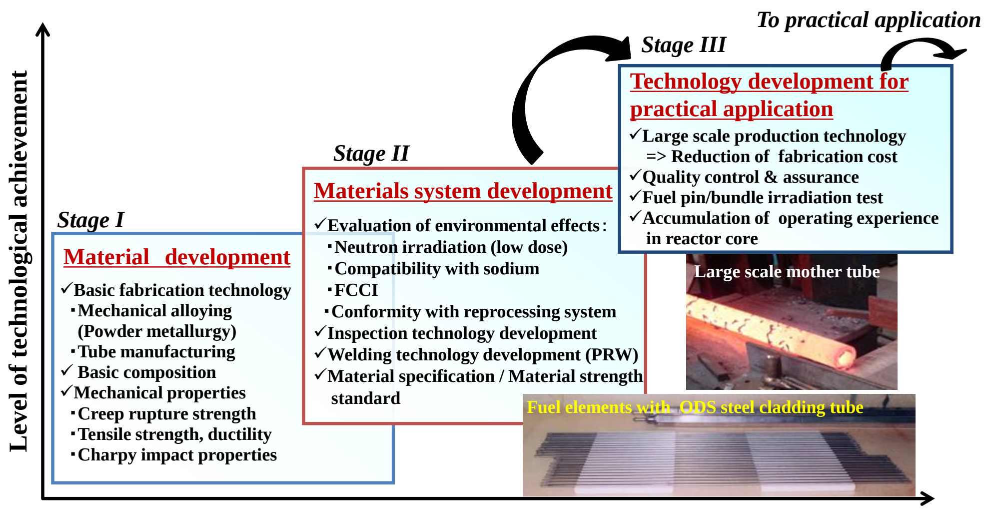

Figure 3 Schematic view of technological level of ODS ferritic steel cladding tube development for FR fuel

3.2 Research on fuel material properties

Material properties of MOX fuel and MA-bearing MOX fuel are key knowledge for the fuel design and for the fuel performance evaluation. Therefore, the thermal and mechanical properties of MA-bearing MOX, such as melting point and thermal conductivity and so on, are being systematically studied in JAEA [11].

The measurement of the melting point of MOX by the improved thermal-arrest method which uses an inner rhenium container has been carried out in JAEA. The dependence of melting point on Am content was obtained. The measurement revealed that the melting point of MA-bearing MOX changes continuously with Am contents and its value can be estimated with the ideal-solution model calculation [12], [13].

The thermal conductivity of MA-bearing MOX is affected slightly for the Am content of 3wt% except in the low-temperature region [14], but it is hardly affected at all even for the Np content of 12wt% [15]. Thermal conductivity data for the high Pu-content MOX in the high-temperature region have been limited. Recently, the thermal diffusivities of MOX with 30% Pu-content were measured at temperatures from 990 to 2190 K by preventing change of the oxygen to metal (O/M) ratio [16]. Thermal diffusivity is one of the main factors affecting thermal conductivity. It was found that when the O/M ratio of specimens was less than 1.95, the measurements could be done at temperatures up to 2190 K in vacuum because the change of O/M ratio was slight. The temperature dependency of thermal conductivity evaluated from the obtained thermal diffusivity data is shown in Figure 4.

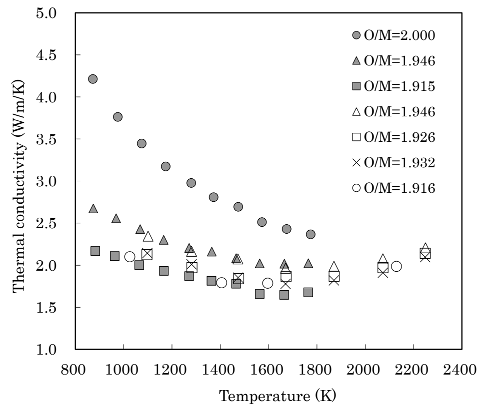

Figure 4 Temperature dependence of thermal conductivity of (U, Pu)

3.3 Fuel design technology development

It is necessary to develop a technique to evaluate the irradiation behavior of the fuel pins with annular pellets made from MA-bearing MOX fuel. The main material properties of MA-bearing MOX fuel such as the thermal conductivity and melting point are introduced into the fuel design method. Significant fuel restructuring occurs during irradiation of the FR fuel. Voids dispersed in an MOX pellet at its fabrication move towards the fuel pellet center by an evaporation-condensation process under a steep radial temperature gradient during irradiation. This phenomenon enlarges the central void of the annular pellets. On the other hand, the central void size is reduced owing to the compressive stress loaded around the central void by FCMI and differential thermal expansion. The fuel performance code CEPTAR is under development to simulate the irradiation behavior of fuel pins with annular pellets. The fuel temperature evaluation model of the CEPTAR code has been verified by fuel centerline temperatures obtained from INTA experiments in Joyo and "power-to-melt" data in Joyo and EBR-II [17].

The B11 irradiation experiments of Np- and Am-bearing MOX fuels and Am-bearing MOX fuels were conducted in 2006 [18], and the B14 irradiation experiment was done 2007 in Joyo to investigate the thermal behavior of MA-bearing MOX fuels at the beginning of irradiation [19]. In the B11 experiments, $(Am_{0.02}, Np_{0.02}, Pu_{0.29}, U_{0.67})O_{2-x}$ and $(Am_{0.05},Pu_{0.29},U_{0.66})O_{2-x}$ were irradiated for 10 minutes and then 24 hours at the maximum linear heat rate of 45 kW/cm. In the B14 experiment, $(Am_{0.024}, Pu_{0.31}, U_{0.666})O_{2-x}$ was irradiated for about 48 hours at the maximum linear heat rate of 47 kW/cm with consideration of the O/M ratio as a parameter. As the result of post-irradiation examinations (PIEs), it was observed that the central void formation would be accelerated in stoichiometric Am-MOX fuels. The void migration model was improved to simulate the irradiation outcome adequately [20]. Vapor pressures of vapor species from the fuel matrix were calculated from correlation between vapor pressure and Gibbs free energy to evaluate the effect of O/M ratio and MA content [21]. The vapor pressure of UO_3, which is the dominant species, increases significantly as the O/M approaches stoichiometry, i.e., 2.00. This vapor pressure dependence explains the accelerated void formation in the stoichiometric fuel. Figure 5 compares the observed and calculated central void diameters. The central void formation can be estimated adequately taking into account each vapor pressure of the vapor species in the void migration model and thermal conductivity of the MA-bearing MOX fuel.

Restoration work in Joyo was completed in 2014, including the replacement of the upper core structure damaged by the material test rig which failed to be detached in 2007. Safety of Joyo is being reinforced to meet new regulation standards established after the Fukushima power plant accident. Various irradiation experiments are being planned after the restart of Joyo to investigate the irradiation behavior of MA-bearing MOX fuels not only in the steady-state but also at the high linear heat rate up to fuel melting. The outcomes of these experiments will contribute substantially to improve the fuel design method of MA-bearing MOX fuels.

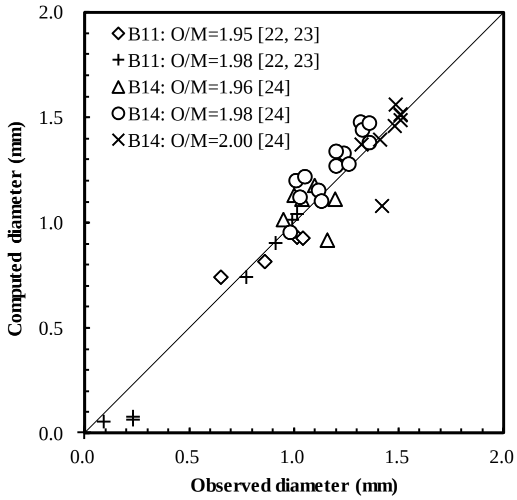

Figure 5 Comparison of observed and calculated central void diameters

4. Conclusion

In Japan, the core & fuel concept of the next generation FR is being established to contribute to the nation's energy security as a sustainable energy source and to contribute to the reduction of the amount and potential hazards of high-level radioactive wastes. Various fuel compositions including MA nuclides can be applied to the concept utilizing characteristics of the high-energy neutron spectrum of the FR. The fuel concept using large-diameter pins with annular pellets facilitates control of the fluctuation in the core power distribution and substantial high burnup owing to the improved internal conversion. The target of reactor outlet coolant temperature, i.e., 550 ºC, is achieved by the apposite arrangement of control rods, adoption of the flux adjuster and other measures. The discharged burnup averaged in the fissile and fertile parts reaches 83 GWd/t which enables the reduction of fuel cycle costs. To realize the concept, the development of core materials for high burnup fuels, the study of material properties of MA-bearing MOX fuels, and improvement of the fuel design method are now in progress.

Acknowledgments

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