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An innovative fast reactor core design for rapid reduction of separated Pu and its proliferation concerns

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ABSTRACT

This report examines a proposed innovative fast reactor (FR) core design that enables rapid reduction of separated plutonium (Pu), while presenting the equivalent or, in some cases, even better operation and safety performance compared to a typical FR. The impact of the innovative core design was evaluated based on mass balance and non-proliferation features.

The fundamental characteristics of FRs for the rapid reduction of separated Pu indicate that a core design with the blanket fuel regions removed and a higher-enriched mixed oxide (MOX) fuel loaded in the core regions should result in the highest Pu loading and Pu reduction ratio. Furthermore, three approaches of the age impact of MOX fuel, the placement of gas plenum in the axial blanket and Pu enrichment zoning gradually dense in the outer core were comprehensively examined for operation and safety performance. These led to be higher Pu loading and Pu reduction with the requirements of a typical FR.

In the mass balance and non-proliferation evaluation, it was clarified that the proposed FR had higher speed of separated Pu reduction than the light water reactor (LWR) or long storage. Consequently, the MOX fuels irradiated in the FR and the LWR were the downgraded categories in the nuclear security and safeguards regulations, which was the same as those of conventional spent nuclear fuel that exists abundantly worldwide.

1. Introduction

The total amount of separated plutonium (Pu) available worldwide from civilian nuclear power reactors in 10 countries was 275 t at the end of 2014. From 1996 to 2005, the world's stock of civil separated Pu grew at an average rate of about 10 t per year. From 2005 to the end of 2014, the average growth rate has slowed to about 2 t per year (Albright et al., 2014). Needless to say it is considered as energy resource in each country, the proper balance would be strongly desired to reduce the potential nuclear proliferation concerns by states or non-state actors.

To alleviate this concern, one option is separated Pu reduction, which means Pu mass reduction by nuclear transmutation physically, and Pu quality downgrading by its conversion to an irradiated form managerially in nuclear security and safeguards regulations (International Atomic Energy Agency, 2001; International Atomic Energy, 2011). Separated Pu mass reduction by nuclear transmutation has been studied the investigation of specific measures, such as Pu burning, to address nuclear proliferation concerns. Some core designs for Pu burning by fission in fast reactors (FRs) have been proposed (Kim et al., 2009; Yamaoka and Fujita, 1997). In those studies, it was determined that a FR core design with a high Pu enrichment, lower core height, and smaller fuel volume could present a higher Pu burning capability with reduced void reactivity. On the other hand, Pu quality downgrading by its conversion to an irradiated form in a nuclear reactor leads to a downgrade of the categories and potentially to an abatement of nuclear proliferation concerns in those regulations. From this point of view, a high loading reactor as a FR with a higher capacity than a light water reactor (LWR) can facilitate the rapid downgrade of the Pu categories.

In the present study, an innovative FR core design was proposed to enable the rapid reduction of separated Pu not only physically by nuclear transmutation but also managerially by conversion to a downgraded irradiated form in the nuclear security and safeguards regulations. In addition, the proposed design presented equivalent or, in some cases, even better operation and safety performance compared to a typical FR, as was verified by numerical neutronic analysis. The impact of the innovative core design for the rapid reduction of separated Pu was evaluated on mass balance

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and non-proliferation features, and compared to separated Pu storage on the ground or rapid reduction of separated Pu in FR or LWR.

2. FR core design for rapid reduction of separated Pu

Firstly, to identify the fundamental characteristics of FRs for rapid reduction of separated Pu, four core FR designs based on the reference reactor design (REF) were evaluated and compared in terms of Pu loading and Pu reduction. Based on the results of this analysis, the FR core design and specifications for rapid reduction of separated Pu is derived based on multi parametrical surveillance which satisfy nuclear safety and performance requirements.

Table 1
Reactor design parameters.

Item	Unit	Specification
a. Plant		
Reactor thermal power	MW_{th}	3570
Coolant temperature (inlet/outlet)	°C	395/550
Fuel/colant/structure	vt.%	43.9/30.3/25.8
Subassembly pitch	mm	206.0
b. Fuel		
Fuel material		$TRUO_2$-UO_2
Pu enrichment in HM (inner core/outer core)	wt%	18.3/20.9
^{235}U enrichment	wt%	0.2
Refueling patern		four-batch
Irradiation time per one batch	day	800
Core diameter/core height	m	5.38/1.0
c. Blanket		
Blanket fuel material		UO_2
^{235}U enrichment	wt%	0.2
Pattern of fuel exchange		four-batch
Thickness of axial blanket (upper/lower)	m	0.2/0.2

2.1. Reference core design and calculation method

The REF in Table 1 and the core layout are based on the demonstration type fast reactor i.e., Japan sodium-cooled fast reactor (JSFR), which is a high internal conversion type with mixed oxide (MOX) fuel in the core regions and depleted Uranium (DU) in the blanket regions (Naganuma et al., 2005). The calculation model was created from the core layout as a two-dimensional (RZ) model in Fig. 1, in each component of the fuel assembly composed of 182 calculation zones which were divided into 41 × 71 mesh intervals, and 9 material regions. In this study, SLAROM-UF (Hazama et al., 2009) and CITATION (Nakagawa et al., 1983) are used; and the cross-section library is UFLIB.J4.0 (Sugino et al., 2011) which is based on the Japanese Evaluated Nuclear Data Library JENDL-4.0. 70-group effective cross-sections for each material region are made by homogeneous calculation using SLAROM-UF. In calculations based on CITATION, reactivity and burnup analysis are performed based on RZ diffusion theory with a nuclide burnup chain in previous study (Hamase et al., 2011).

For simplification of the calculations, the four-batch method was changed to the one-batch of fuel exchange method using the following equation (Graves, 1979) to determine the fuel discharge burnup.

$$B_n^d = \frac{2n}{n+1} B_1 \quad (1)$$

where n is the batch number, B_n^d is the discharge fuel burnup at the end of the cycle (EOC) assuming n-batch, B_1 is the burnup at EOC assuming one-batch. From Eq. (1), B_n^d of the discharge fuel burnup for this core is 3200 days and B_1 is 2000 days. Therefore, the initial core fuel enrichment was adjusted to achieve over 2000 days of burnup when the effective multiplication factor (k_{eff}) was one. In

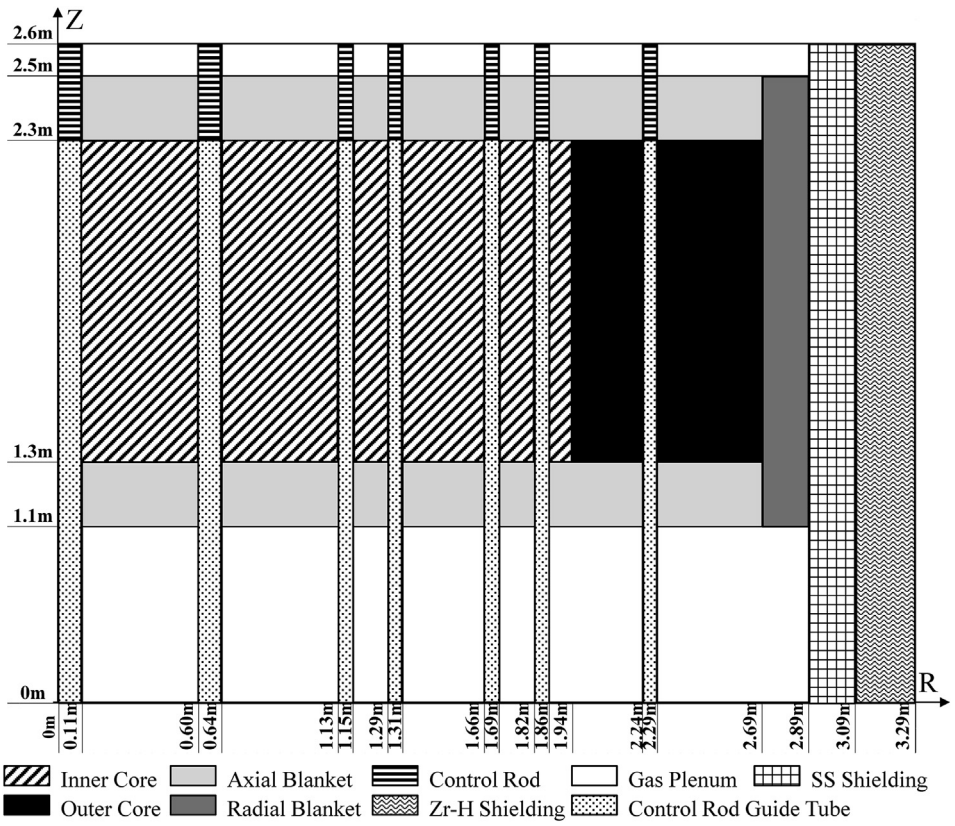

Fig. 1. CITATION RZ calculation model in the Reference core (REF).

this study, the effective multiplication factor at the beginning of cycle (BOC) with four-batch $k_{eff,BOC}$ is defined as the averaged k_{eff} at 0, 800, 1600 and 2400 days.

In the calculation of the peaking factor, the relative power density in the *i*-th radial region is defined as the average of the power density in the *i*-th radial region divided by that of the entire core. In addition, the radial peaking factor is defined as the maximum of the relative power density in the radial region. The relative power density in the axial region and the axial peaking factor is defined using the same method as the radial approach.

2.2. Evaluation of fundamental characteristics of FRs for Pu loading and Pu reduction

To identify the fundamental characteristics of FRs for rapid reduction of separated Pu, four core designs based on REF were characterized in terms of Pu loading and Pu reduction. The specifications of REF and the four core designs are shown in Table 2 and Fig. 2. In this calculation, REF and CASE 1 were used as base core design which refers to a previous study of JSFR. CASE 2 removed the blanket fuel regions, which implied the use of stainless-steel (SS) for shielding in these regions instead of DU, due to decreasing Pu production. CASE 3 loaded highly-enriched MOX fuel in the core regions, which meant that this case uses up to 50% of the high enrichment MOX fuel in the inner and outer cores by decreasing the density of U. Finally, CASE 4 was a combination of CASE 2 and CASE 3, which used up to 50% of the high enrichment MOX fuel in the inner and outer core, and used SS for shielding in the blanket. The composition of the transuranic (TRU) elements in each case is shown in Table 3. REF used TRU composition of the fast breeding reactor Multi-cycle (Multi-TRU) used in a JSFR study, which is recycled a few times with FR. On the other hand, the compositions of the CASE 1–4 were assumed as TRU through 10 years or 50 years of storage after reprocessing of the Uranium Oxide (UOX) fuel that was used in LWR. In CASE 1, CASE 2, CASE 3 and CASE 4, the TRU composition of Pu-10y was assumed to be separated Pu after 10 years of cooling time after reprocessing and extracting from irradiated fuels. These fuels are typical irradiated PWR fuels (44GWd/tHM) with 4.1% ^{235}U (Okumura and Okamoto, 2011; Katakura et al., 2004) followed by five years of cooling time. In these cases, the thermal power was fixed to 3333 MWth in the core region to avoid changing the operation performance and burnup day at $k_{eff} = 1$ was fixed to 2000 days by controlling the amount of Pu. The thermal hydronic and nuclear safety performance has been analyzed in a previous study (Naganuma et al., 2005).

Sensitivity analysis was performed for these cases. Fig. 3 shows the burnup reactivity in each case. From these results, it is seen that the blanket material had little impact on reactivity, however

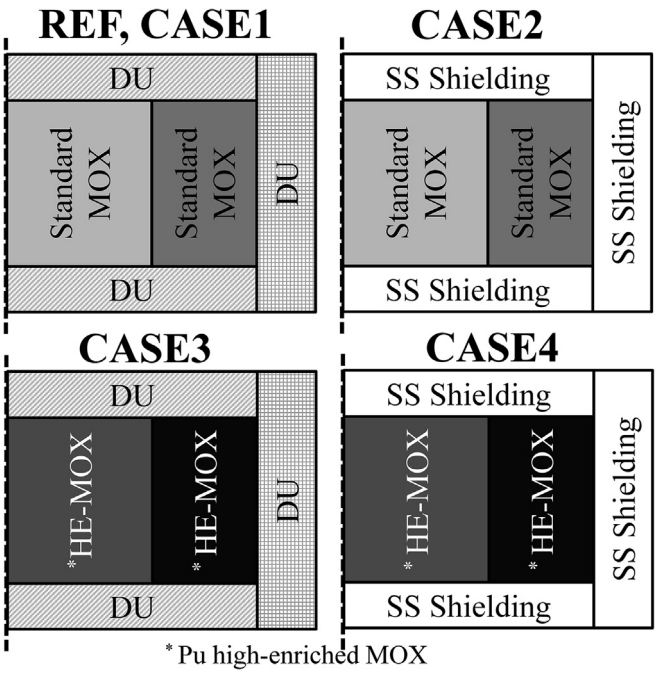

Fig. 2. Images of core designs for improvement.

Table 3
TRU composition in MOX fuel.

Nuclide	Composition (wt%)		
	Multi-TRU	Pu-10y[a]	Pu-50y[b]
^{238}Pu	1.1	2.2	1.6
^{239}Pu	54.1	55.8	56.4
^{240}Pu	32.1	23.8	24.0
^{241}Pu	4.3	7.1	1.0
^{242}Pu	3.9	6.7	6.8
^{237}Np	0.5	0.0	0.0
^{241}Am	2.0	4.4	10.1
^{243}Am	1.0	0.0	0.0
^{244}Cm	1.0	0.0	0.0

Those after reprocessing from spent fuels which are irradiated in PWR (44GWd/tHM).
[a] Separated Pu through 10 years of cooling time.
[b] Separated Pu through 50 years of cooling time.

Pu enrichment in the core had a significant effect because of the decreasing the volume of ^{238}U with high a resonance absorption cross-section. Fig. 4 indicates the Pu reduction ratio ($\Delta M_{Pu}/M_{Pu,BOC}$), and the weight of the total Pu loading in the entire core for each

Table 2
Detail core designs for improvement.

Item	Unit	Specification				
		REF	CASE 1	CASE 2	CASE 3	CASE 4
a. Plant						
Reactor thermal power	MW_{th}	3570	3570	3333	3570	3333
b. Core Fuel						
Average core power density	W/cm^3	0.315	0.315	0.315	0.315	0.315
TRU composition typle		Multi-TRU	Pu-10y	Pu-10y	Pu-10y	Pu-10y
Pu enrichment in HM (inner core/outer core)	wt%	18.3/20.9	18.6/21.2	19.1/21.8	45.9/50.0	45.8/50.0
The weight of Pu in core (Inner core/Outer core)	t	6.75/7.35	6.85/7.45	7.06/7.68	7.81/8.49	8.25/8.98
c. Blanket						
Blanket fuel material		UO_2	UO_2	SS	UO_2	SS
^{235}U enrichment	wt%	0.2	0.2	–	0.2	–

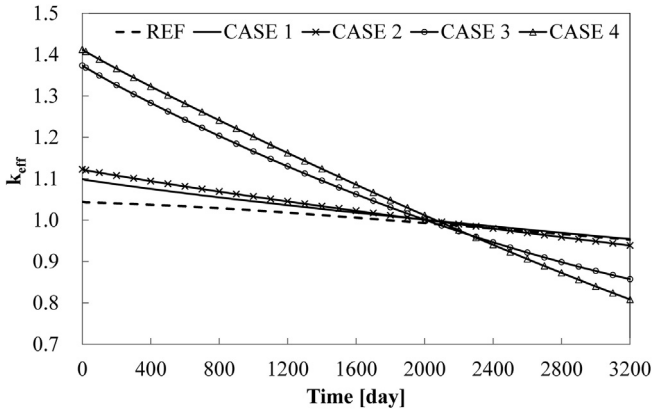

Fig. 3. k_{eff} with Burnup for REF and four representative designs.

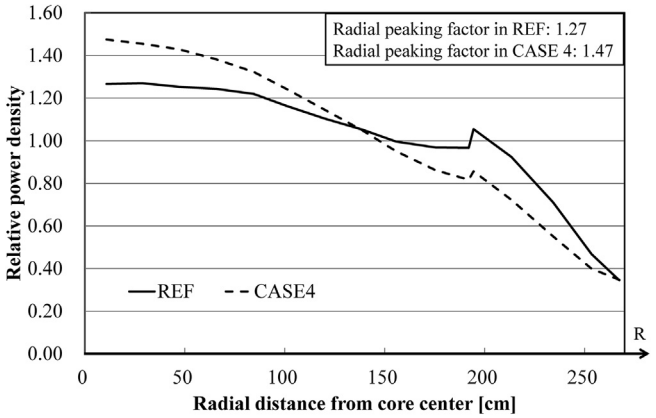

Fig. 5. Radial power distribution at BOC for CASE 4.

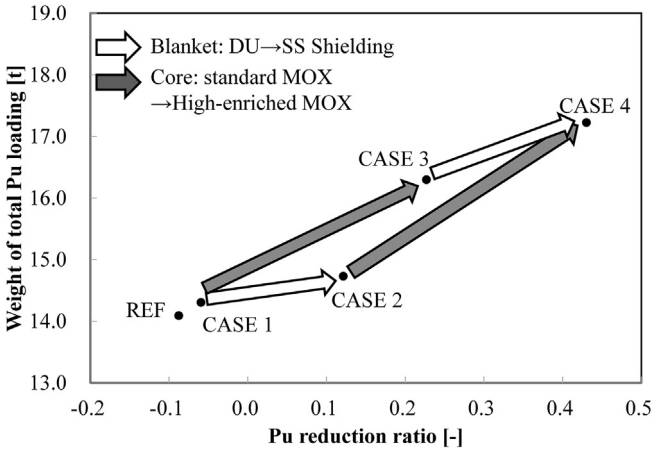

Fig. 4. Pu reduction ratio for REF and four representative designs.

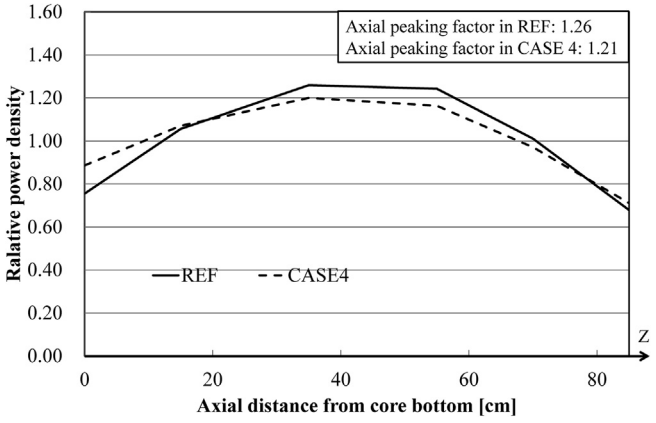

Fig. 6. Axial power distribution at BOC for CASE 4.

case. The Pu reduction ratio increased in CASE 2 because Pu was never produced in the blanket region without fuel. The weight of Pu loading and the Pu reduction ratio increased in CASE 3 using higher-enriched MOX fuel in the core because the neutron capture reaction rate of ^{238}U and the generation of Pu decreased in the fuel. Based on the aforementioned, it was revealed that a core design with the blanket fuel regions removed and the loading of higher–enriched MOX fuel in the core regions caused the highest Pu loading and Pu reduction ratio with referable characteristics for rapid reduction of separated Pu, especially a core design of CASE 4 with SS shielding in the blanket regions and 45.8 wt% and 50.0 wt% of Pu enrichment in the inner and outer core respectively, would present 17.2 t of Pu loading and 0.43 of Pu reduction ratio.

However, this CASE 4 design had some technical challenges regarding operation and safety performance, initial excess reactivity, and peaking factor. In terms of initial excess reactivity, k_{eff} in CASE 4 was 1.35 times higher than that of REF as shown in Fig. 3, and $k_{eff,BOC}$ in CASE 4 was 1.17. In terms of peaking factor, the relative power density of the radial power distribution for CASE 4 in Fig. 5 was higher than that for REF in the inner region. Moreover, the radial peaking factor in CASE 4 was 1.16 times higher than that of REF. The relative power density for CASE 4 in Fig. 6 of the axial power distribution was flatter than that for REF, and this peaking factor was also lower than the value for REF. Therefore, it was clarified the key issues of such a core design were the initial excess reactivity, positive void reactivity and the peaking factor.

2.3. Improvement of operation and safety performance for rapid reduction of separated Pu

2.3.1. Approach of improvement for technical challenges

For the two technical challenges of higher initial excess reactivity and peaking factor, two approaches are developed, and we attempt to incorporate them in the design for rapid reduction of separated Pu with the equivalent or, in some cases, even better operation and safety performance compared to a typical FR. They include the utilization of long storage MOX fuel with high ^{241}Am composition for the control of the initial excess reactivity, and Pu enrichment zoning is used to gradually increase the density in the outer core to facilitate a lower peaking factor and void reactivity. In general, fission to capture ratio of ^{241}Am is less than 1 in the core regions of the FR in normal operation and ^{241}Am plays a role of a burnable poison. However, this ratio would be increased when sodium void is generated and fast neutron fraction increases in the FR. Therefore it is expected the void reactivity would be increased when MOX fuels are stored for a long period of time and the inventory of ^{241}Am increases. To address this problem, another approach is added i.e., the placement of gas plenum in the axial blanket regions for improved void reactivity.

In the previous discussion, CASE 4 was shown to be suitable for the rapid reduction of separated Pu. However, the Pu enrichment zoning is reconsidered in the proposed approaches for multi-parametrical surveillance. Therefore, those three approaches were associated with CASE 2 core design for this process. Note that those improvements of FR core design gave priority to separated Pu

quality downgrading than to Pu mass reduction in rapid reduction of separated Pu in this study.

Considering operation and nuclear safety performance, three constraints were decided based on JSFR study data which has been verified. In that study, the initial excess reactivity was less than 1.07, which is derived from 6.8%Δk/kk′ of the main control rod worth. The power peaking factor, void coefficient, and Doppler coefficient of the nuclear safety reactivity must be less than the corresponding values in that study. The amount of Pu loading was maximized in the core under these constraints.

2.3.2. Utilization of long storage MOX fuel for the control of initial excess reactivity

If the amount of Pu loading increases too much as shown in CASE 3 or 4 in Fig. 2, the initial excess reactivity may exceed $k_{eff,\ BOC}$ = 1.07 of controllable reactivity. To control the initial excess reactivity, the storage year for the separated Pu was examined. Worldwide, separated Pu has been produced from irradiated fuels by reprocessing since about the 1960s. Therefore, the separated Pu is assumed to be stored on long-term for 50 years after reprocessing and extracted from the irradiated UOX fuels. This composition named Pu-50y in Table 3 increases the ^{241}Am ratio by beta decay of ^{241}Pu due to the short half-life (14.3 years). The initial excess reactivity of CASE 2 using the composition of Pu-50y was 0.95 times lower than that of Pu-10y in one-batch. Moreover, the effective burnup period at k_{eff} = 1.0 was about 100 days longer for Pu-50y than for Pu-10y. The $k_{eff,\ BOC}$ was 1.05 for Pu-10y while a value of 1.03 was obtained for Pu-50y in four-batch. This is because ^{241}Am produced by ^{241}Pu has a lower fission to capture ratio for fast neutrons under 1 in the core regions of the FR in normal operation and plays a role of a burnable poison which makes decrease reactivity. Transformations of ^{241}Am were promoted as presented (Hamase et al., 2011) in the following burnup chain:

$$^{241}\mathrm{Am(n,\gamma)85\%} \rightarrow {}^{242}\mathrm{Am}(\beta, T_{1/2} = 16.0\mathrm{h})82.7\% \rightarrow \mathrm{Cm}(\alpha, T_{1/2} = 162.8\mathrm{d}) \rightarrow {}^{238}\mathrm{Pu}$$

and ^{238}Pu was produced. Therefore, ^{238}Pu acted as fissile material in the fast reactors and produced higher reactivity.

However, the larger composition of ^{241}Am causes the void reactivity to be worse in the FR in general. Thus, the void coefficient has to be checked for this composition. In this study, the void coefficient is defined as follows:

$$\text{Void coefficient} = (k' - k)/\%\text{void} \tag{2}$$

where k is the effective multiplication factor assuming no void and k' is that assuming void existence, and %void is the void fraction. In this study, it is given as 72.7% assuming that the sodium completely transforms to a void inside the fuel assembly region except for the gap and inner duct areas. Consequently, the void coefficient was 2.07×10^{-4} [(k′ − k)/%void] at BOC in CASE 2 using Pu-10y and 2.41×10^{-4} [(k′ − k)/%void] at BOC in CASE 2 using Pu-50y composition. It was determined that the MOX fuel stored for 50 years made void coefficient was more positive up to 1.16 times compared with that for 10 years. This is mainly because the capture reaction rate decreases and the fission reaction rate increases for ^{241}Am in the harder neutron spectrum, resulting from the void generation. Thus, we need to improve the void reactivity in the ^{241}Am-riched fuel composition by redesigning the core without changing the core geometry.

2.3.3. Placement of gas plenum in the axial blanket for the improvement of void reactivity

Secondly, it was considered to use the gas plenum instead of SS shielding in the axial blanket regions in CASE 2, to improve the void reactivity. This is because gas plenums in the blanket regions increase neutron leakage due to a lower atomic density. Consequently, the void coefficient could be improved. The void coefficient was 1.62×10^{-4} [(k′ − k)/%void] at BOC in the CASE 2 after replacing the gas plenums with SS shielding in the axial blanket regions. Therefore, the result revealed that the utilization of gas plenum in the axial blanket could decrease and thereby improve the void coefficient by 21% compared with the void coefficient in CASE 2.

2.3.4. Pu enrichment zoning with gradually improved density in the outer core for low peaking factor

Thirdly, it was considered that Pu enrichment zoning with a gradual increase in density in the outer core for improving peaking factor. In this calculation, the inner and outer core region is segmented into 6 regions as shown in Fig. 7 from CASE 2, and Pu enrichment is defined for each region to maximize loading of Pu and satisfy the constraint for peaking factor. The inner core is composed of 2 regions; IC1 ($1.45 \le z \le 2.15$ [m]) and IC2 ($1.3 \le z \le 1.45$ or $2.15 \le z \le 2.3$ [m]). The outer core is composed of 4 regions; OC1 ($1.45 \le z \le 2.15$ and $1.94 \le r \le 2.53$ [m]), OC2 ($1.45 \le z \le 2.15$ and $2.53 \le r \le 2.69$ [m]), OC3 ({$1.3 \le z \le 1.45$ or $2.15 \le z \le 2.3$} and $1.94 \le r \le 2.53$ [m]) and OC4 ({$1.3 \le z \le 1.45$ or $2.15 \le z \le 2.3$} and $2.53 \le r \le 2.69$ [m]). Therefore, Pu enrichment is increased in each core region more than IC = 18.7 wt% and OC = 21.4 wt% for the core design with ^{241}Am-riched fuel and gas plenum in the axial blanket. Firstly, for fixing the radial peaking factor, Pu enrichment in the radial outer regions of OC2 and OC4 increased from 21.8 wt% until radial peaking factor = 1.27 as REF. Secondly, for fixing the axial peaking factor, Pu enrichment in the axial outer regions of IC2, OC3, and OC4 is increased by the same fraction of the enrichment distribution which resulted from the first step, until the axial peaking factor achieves a value of 1.26, the same as REF.

Therefore, Figs. 8 and 9 show the ratio of the radial and the axial power density and those of the peaking factor, and Table 4 shows the Pu enrichment distribution in REF and the proposed core of CASE 5. These results indicate that the ratio of the radial and axial power density in CASE 5 is higher in the outer region than in the inner region, but the same peaking factors as REF were achieved. For this result, CASE 5 could load Pu of 17.5 t from 14.1 t in REF.

2.4. The result of proposed core CASE 5

The proposed core of CASE 5 as shown in Fig. 10 for rapid reduction of separated Pu was decided as follows. Firstly, the separated Pu after long-term storage for 50 years after reprocessing presented as Pu-50y composition was used as MOX fuels. Secondly, gas plenum was used in the axial blanket region, and SS shielding was used in the radial blanket region. Finally, the core was divided into 2 axial and 4 radial regions to distribute and set the Pu enrichment as shown in Table 4.

Finally, CASE 5 of the high Pu loading core was evaluated in terms of initial excess reactivity, void coefficient, and Doppler coefficient additionally as Table 5. In the initial excess reactivity, CASE 5 presented $k_{eff,BOC}$ = 1.07, and this value was equivalent to the controllable reactivity of control rods. In the void coefficient, CASE 5 presented 1.89×10^{-4} [(k′ − k)/%void] at BOC less than 2.24×10^{-4} [(k′ − k)/%void] at BOC in REF. Finally, the Doppler coefficient is defined as follows:

$$\text{Doppler coefficient} = (k' - k)/\Delta T \tag{3}$$

where k is the effective multiplication factor in the normal operation and k' is for an abnormal condition assuming a rise in fuel temperature, and ΔT is the difference in the fuel temperature. In this study, the temperature increase was assumed to be 500 K in both the core and the blanket regions. Using this, it was determined that

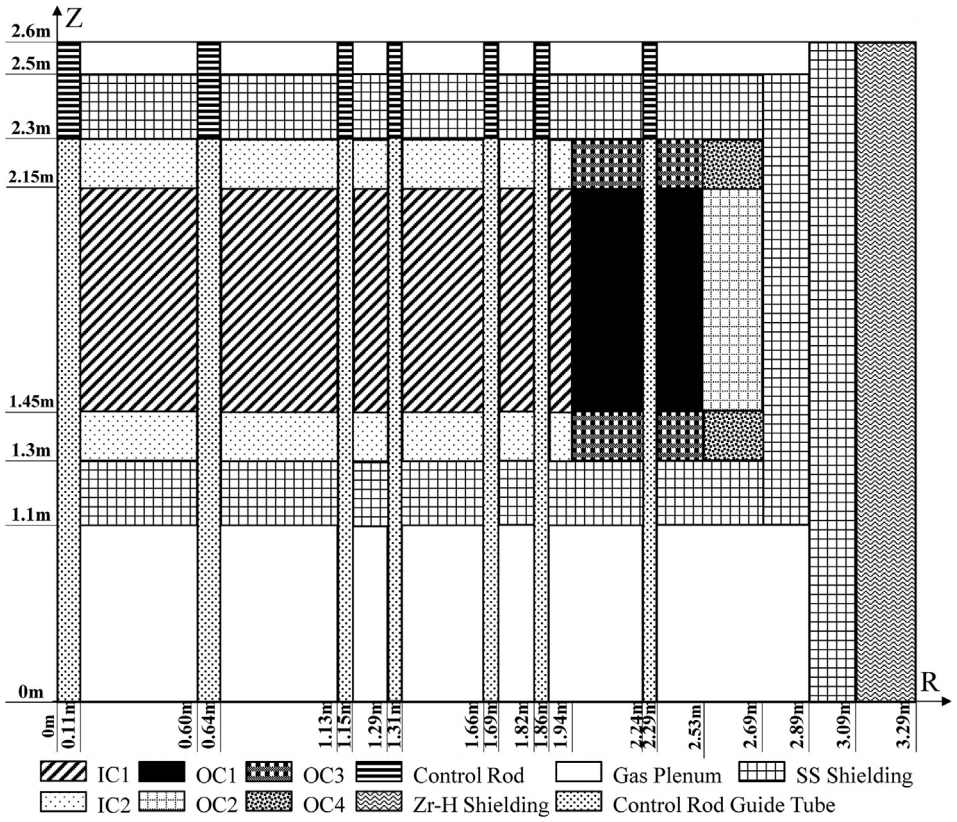

Fig. 7. RZ calculation model with inner and outer core regions segmented into 6 regions from CASE 2.

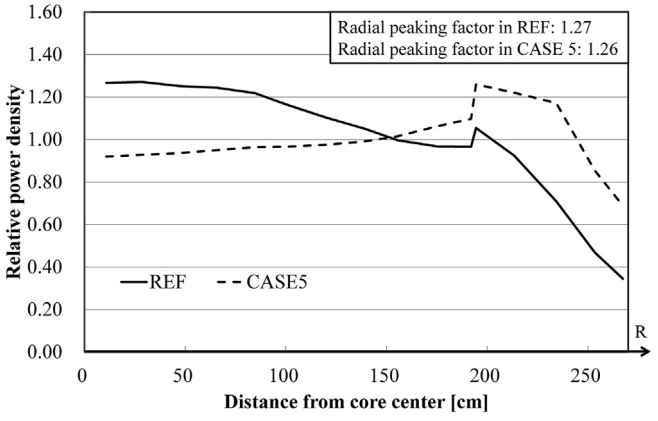

Fig. 8. Radial power distribution at BOC for CASE 5.

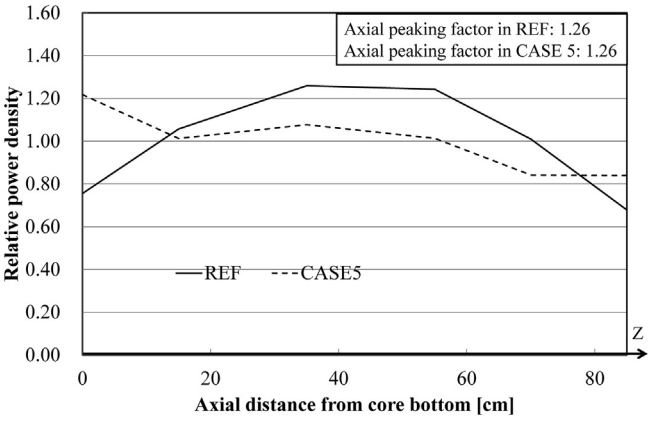

Fig. 9. Axial power distribution at BOC for CASE 5.

CASE 5 had a value of -1.92×10^{-6} [$(k' - k)/\Delta T$], which is less than -1.39×10^{-6} [$(k' - k)/\Delta T$] for REF.

Therefore, the FR core design and specification for rapid reduction of separated Pu was derived from multi parametrical surveillance, and provided equivalent or, in some cases, even better operation and safety performance compared to the typical FR. In order to overcome the aforementioned issues, three approaches were comprehensively examined; the age impact of the MOX fuel was very significant to the initial excess reactivity compensation to accept a high initial 17.5 t Pu loading from 14.1 t and 0.17 Pu reduction ratio from −0.09, placement of the gas plenum in the axial blanket led to improvement of the void reactivity and the Pu enrichment zoning gradually became denser in the outer core which led to an improvement of the peaking factor, comparable

Table 4
Pu enrichment of proposed core (CASE 5).

	Volume [m^3]	Pu enrichment in HM [wt%]	
		REF	CASE 5
IC1	7.41	18.3	18.7
IC2	3.18	18.3	28.1
OC1	3.19	20.9	21.4
OC2	3.86	20.9	24.0
OC3	1.37	20.9	32.1
OC4	1.65	20.9	36.0
Total	20.65	–	–
Average	–	19.6	23.8

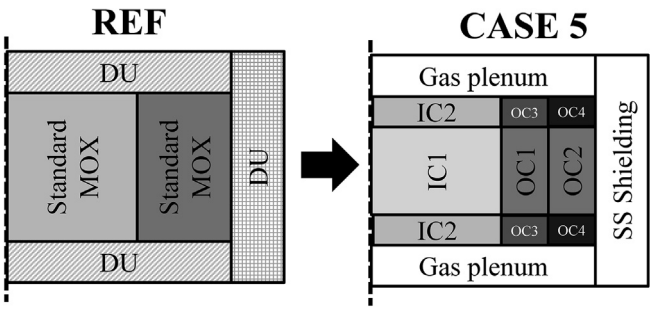

Fig. 10. Proposed core (CASE 5) design.

Table 5
Proposed core (CASE 5) features.

Item	Unit	REF	CASE 5
Weight of total Pu loading in core	t	14.1	17.5
Pu reduction ratio		−0.09	0.17
Axial peaking factor (BOC)		1.26	1.26
Radial peaking factor (BOC)		1.27	1.26
keff, BOC (CR can accept 1.07)		1.04	1.07
Void coefficient (BOC)	Δk/%void	2.24E−04	1.89E−04
Doppler coefficient (BOC)	Δk/ΔT	−1.39E−06	−1.918E−06

to the requirements of a typical FR such as REF. Finally, it was confirmed that CASE 5 had a rapid reduction of separated Pu with the equivalent initial excess reactivity and peaking factor to the typical FR such as REF and less positive void coefficient and more negative Doppler coefficient than that.

3. Evaluation of mass balance and non-proliferation

3.1. Pu management scenarios

Finally, mass balance and non-proliferation features of the proposed FR core for rapid reduction of separated Pu were evaluated and compared with three Pu management scenarios, as follows.

For worldwide management of separated Pu, the storage of separated Pu on the ground under strict surveillance is the general approach. On the other hand, for the disposal of separated Pu, it is common to convert to the irradiated form using nuclear reactors to decrease the risk of proliferation. Therefore, three Pu management scenarios are presented in Fig. 11 for separated Pu storage on the ground, separated Pu reduction in FR or in LWR, with 47.9 t separated Pu equivalent to that of Japan in 2016 (Japan Atomic Energy Commission, 2016); were evaluated in this study. The scenarios started at 0 year when the one FR and the one LWR start the operation and the whole separated Pu was moved to the storage on the ground. In scenario 1 of separated Pu storage on the ground, it was assumed that the separated Pu 47.9 t was formed as MOX powder with a composition of 1:1 for the Pu-to-U ratio, and it was stored semi-permanently in containers on the ground under strict surveillance immediately. The specifications of the storage containers are shown in Table 6. In scenario 2, FR started its operation for separated Pu reduction as MOX fuel in the whole core. In scenario 3, LWR was used for the separated Pu reduction as MOX fuels. Considering practical transition of UOX fueled LWR for MOX fuel utilization, it was assumed that the UOX fuels were gradually replaced to MOX fuel in 1/3 of whole core of LWR over 3 cycles. The LWR continued its operation in three batch incineration for each UOX and MOX fuel after the first 3 cycles. In scenario 2 of separated Pu reduction in FR and scenario 3 of separated Pu reduction in LWR, it was assumed that the separated Pu was burned in a FR or LWR as fresh MOX fuels and converted to irradiated fuels in equilibrium cycle. The FR in Scenario 2 is the proposed FR core for rapid reduction of separated Pu as shown in Table 7, whereby the weight of the structure refers to the JOYO data (Ohkawachi et al., 2003). The LWR in scenario 3 is the typical PWR using MOX fuel (45GWd/tHW) (Okumura and Okamoto, 2011; Yamamoto et al., 2013) as shown in Table 7 and ORIGEN-ARP (Bowman and Gauld, 2010) was used for the PWR burnup calculation.

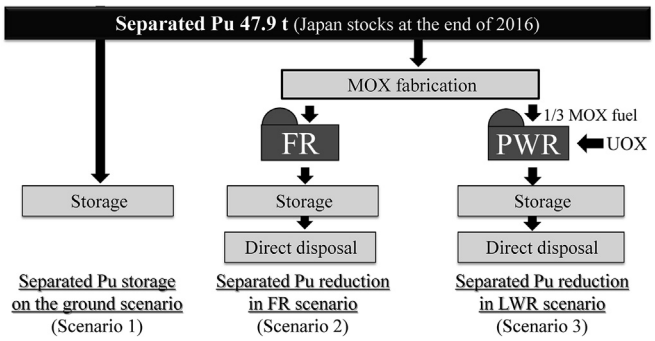

Fig. 11. Pu management scenarios for separated Pu 47.9 t.

3.2. Result of evaluation

The results for the mass balance in the three scenarios are shown in Figs. 12 and 13.

Fig. 12 shows the transition of the amount of separated Pu and irradiated Pu since the nuclear reactor started operation for each scenario. In this figure, in fuel reloading, a certain amount of separated Pu is introduced as fresh MOX fuels and its corresponding amount of irradiated Pu is discharged. In scenario 1, the separated Pu was stored for a long time and was not irradiated. Thus, the amount of all separated Pu stayed constant. In scenario 2, large amount of separated Pu was decreased at first cycle because of fresh MOX fuel loading into the whole FR core. Scenario 2 presented that the separated Pu 47.9 t was converted to the irradiated form in 19.5 years by one unit of FR and the amount of irradiated Pu increased and eventually achieved 41.6 t. In contrast, scenario 3 presented the separated Pu 47.9 t was converted to an irradiated form in 86.5 years by one unit of LWR and the amount of irradiated Pu increased and eventually achieved 35.3 t. Considering these three scenarios, the amount of separated Pu storage on the ground in scenario 1 would never change via separated Pu reduction in LWR or the FR in scenarios 2 and 3 would be completed within a finite period of time. Pu reduction speed in a FR is 4.4 times faster than that in a LWR compared with MOX fuel irradiation. The Pu inventory per MOX fuel assembly of the FR in scenario 2 was about 24% lower than that of the LWR in scenario 3. However, the number of exchanged fuel assembly per year in scenario 2 is about 290% larger than that in scenario 3. Larger number of exchanged

Table 6
Specifications of the MOX powder storage container for scenario 1.

Item	Unit	Specification
Shape		Cylinder type
Matarial		aluminium alloy
Outer diameter	mm	191
Height	mm	400
Pu inventory per container	kg	6.6
MOX inventory per container	kg	13.3
Container matarial weight	kg	5.7
Gross weight per container	kg	20.7

Table 7
Specifications of FR and LWR for scenario 2 and scenario 3.

	Item	Unit	Specification	
			Scenario 2	Scenario 3
Reactor design	Reactor type		FR	PWR
	Reactor thermal power	MWth	3333	3212
	Operation time per a batch	day	800	381
	Refueling pattern		four-batch	three-batch
	Refueling period	day	90	90
	MOX fuel loading area		All CORE	1/3 CORE
	The number of fuel assembly		562	52
MOX fuel	Fuel material		PuO_2-UO_2	PuO_2-UO_2
	Average Pu enrichment in HM	wt%	23.8	9.0
	^{235}U enrichment	wt%	0.2	0.2
	Subassemby type		–	17 × 17
Fuel assembly	Pu inventory	kg	31	41
	MOX inventory	kg	145	520
	Structural material weight	kg	304	154
	Gross weight per assembly	kg	449	674

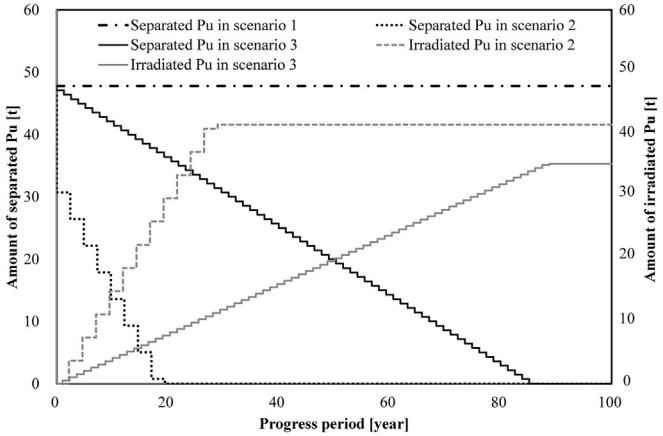

Fig. 12. Weight of separated Pu and irradiated Pu for 100 years.

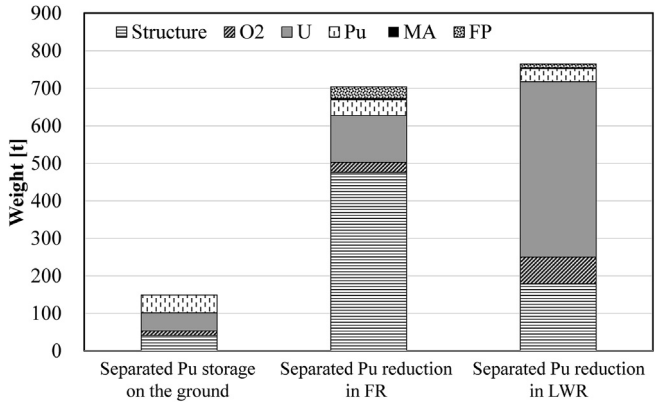

Fig. 13. Total weight of U, Pu, MA, O_2, FP and the structure in the canisters in scenario 1 and in irradiated MOX fuel assemblies in scenario 2 and 3.

fuel assembly per year in FR was one of dominant factor in the rapid Pu reduction speed.

Fig. 13 shows the weight of U, Pu, MA, O_2, FP and structure material in MOX storage containers or irradiated MOX fuel assemblies on the storage or after irradiation of separated Pu 47.9 t in each scenario. In scenario 1, the total accumulated weight was the least of 123 t of the three scenarios because Pu enrichment of MOX powder was 29% higher than that of the MOX fuel assembly and the weight of the container's structure per unit is 40% lighter than that of the fuel assembly. The total accumulated weight in scenario 2 was equivalent to that of scenario 3. On the other hand, the accumulated weight of the Actinides and the FP in irradiated MOX fuel assemblies in the FR was 40% smaller than those of the LWR and the accumulated total weight of the fuel assembly structure in the FR was 2.66 times larger than those of the LWR. This is because the number of MOX fuel assemblies and the weight of the structure per fuel assembly of FR were 32% and 97% larger, respectively, than those of LWR. The fuel assembly of FR is heavier than MOX fuel assembly of LWR because of its massive structure of upper and lower assembly and duct structures. In terms of non-proliferation features, the nuclear material categories in the safeguards (International Atomic Energy Agency, 2001) and physical protection of nuclear security (International Atomic Energy Agency, 2011) regulations were evaluated for three scenarios as shown in Fig. 14. The container including the MOX powder in scenario 1 and the fresh MOX fuel for the FR and the LWR in scenario 2 and scenario 3 are categorized into un-irradiated direct-use material in the safeguards, and Category I in physical protection. However, irradiated fuel from the FR and the LWR is categorized into irradiated direct-use material in safeguards, and Category II in physical protection, because their radiation dose increased. Therefore, it causes a decreased frequency of inspection and the easing of restrictions on a monitoring system for burnup of fresh MOX fuel. As a consequence, MOX powder form in containers was the highest grade, and the strictest regulations would be maintained in both in

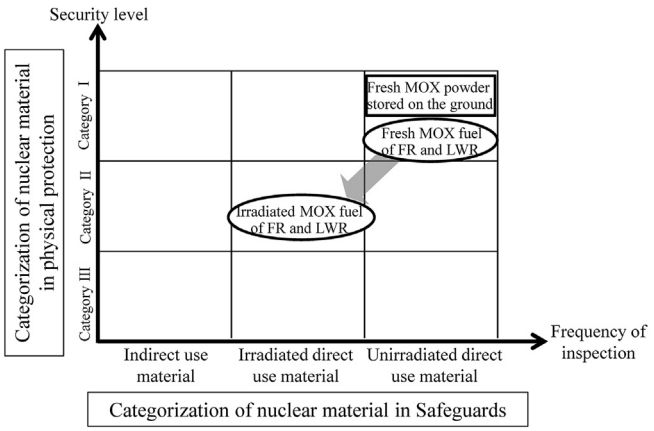

Fig. 14. Categorization of non-proliferation features in three scenarios.

nuclear security and safeguards. On the other hand, the Categories of MOX fuel irradiated in the FR or the LWR were downgraded in 1-step and the same as those of conventional spent nuclear fuel that exists abundantly worldwide.

4. Conclusions

An innovative FR core design was proposed to enable the rapid reduction of separated Pu physically by nuclear transmutation and managerially by conversion to an irradiated form, thereby downgrading its category in the nuclear security and safeguards regulations. The design also facilitated the equivalent or, in some cases, even better operation and safety performance compared to a typical FR, as determined by numerical neutronic analysis. The impact of the innovative core design for the rapid reduction of separated Pu was evaluated on mass balance and non-proliferation features, and compared to separated Pu storage on the ground or rapid reduction of separated Pu in the FR or the LWR.

Firstly, to find the fundamental characteristics of the FRs for rapid reduction of separated Pu, four core designs of FRs based on the demonstration type fast reactor, JSFR, were evaluated and compared in terms of Pu loading and Pu reduction. It was revealed that a core design with the blanket fuel regions removed and a higher–enriched loading of MOX fuel in core regions caused the highest Pu loading and Pu reduction ratio. These are preferable characteristics for rapid reduction of separated Pu, however the key issues of such core design were initial excess reactivity, positive void reactivity, and peaking factor.

Secondly, based on the results of fundamental characteristics analysis, the FR core design and specification for rapid reduction of separated Pu was derived for multi-parametrical surveillance, and provided the equivalent or better operation and safety performance than the typical FR. In order to overcome the key issues, three approaches were comprehensively examined; the age impact of MOX fuel was very significant to the initial excess reactivity compensation to accept a high initial 17.5 t Pu loading and 0.17 Pu reduction ratio. Placement of the gas plenum in the axial blanket led to the improvement of the void reactivity, and the Pu enrichment zoning gradually became denser in the outer core which led to an improvement of the peaking factor, comparable to the requirements of a typical FR.

Finally, mass balance and non-proliferation features were evaluated and compared between separated Pu storage as the MOX powder form in containers, separated Pu reduction by MOX fuel irradiation in that proposal core of the FR and separated Pu reduction by MOX fuel irradiation in the typical PWR type of LWR, with 47.9 t separated Pu equivalent in Japan. In terms of mass balance, the amount of separated Pu storage on the ground would never change via separated Pu reduction in LWR, or FR would be completed within a finite period of time. Compared with MOX fuel irradiation, all Pu reduction time in one unit of FR was 19.5 years, which implies that the reduced speed in FR was 4.4 times faster than that in LWR due to the higher initial Pu loading, and total accumulated weight of Actinides and FP in the irradiated MOX fuel assemblies in FR was 40% smaller than those in the LWR. In terms of non-proliferation features, the nuclear material categories in the nuclear security and safeguards regulations were evaluated; MOX powder form in containers was the highest grade and the strictest regulations would be maintained in both the nuclear security and safeguards. On the other hand, the Categories of MOX fuel irradiated in the FR or the LWR would be downgraded in 1-step and the same as those of conventional spent nuclear fuel that exists abundantly worldwide.

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