DOI: 10.15669/pnst.7.20

Progress in Nuclear Science and Technology
Volume 7 (2025) pp. 20-26

ARTICLE

Core concept of innovative small SFR with metal fuel for deployment in Japan

Sho Fuchita [a, *], Koji Fujimura [a], Daisuke Watanabe [a], Hirotaka Nakahara [a], Kazuhiko Matsumura [a], Hirokazu Ohta [b] and Masatoshi Iizuka [b]

[a] *Hitachi-GE Nuclear Energy, Ltd., 1-1 Saiwai-cho 3-chome, Hitachi-shi, Ibaraki 317-0073, Japan;* [b] *Central Research Institute of Electric Power Industry, 2-6-1 Nagasaka, Yokosuka-shi, Kanagawa 240-0196, Japan*

We are developing a metal-fueled sodium-cooled fast reactor (SFR) to solve social issues regarding nuclear power for the future. Fast reactors are effective for efficient use of uranium (U) resources and reducing the volume and radiotoxicity of high-level radioactive wastes (HLWs). In particular, the combination of fast reactors with metal fuel and pyrometallurgical reprocessing has the potential to achieve both at an early stage. In this development, we studied the fast reactor with metal fuel implementation scenario in Japan, in which the 1st reactor is introduced in 2040 and existing light water reactors (LWRs) are replaced by the fast reactor in stages. In this paper, based on this scenario, we evaluated the performance of cores with three types of fuel; LWR UO_2 spent fuel, LWR MOX spent fuel, and the fast reactor spent fuel. As a result, all cores satisfied design criteria, and the Pu breeding ratio was sufficient for the scenario to work. We have also shown that transmutations of minor actinides (MAs) such as neptunium (Np), americium (Am) and curium (Cm) in the fast reactor and the recycling of MA remaining in the spent fuel through pyrometallurgical reprocessing can remarkably reduce MA migration to HLWs with proven technology. These results confirmed the feasibility of the implementation scenario that contributes to the effective use of resources and the reduction of the hazardous level of HLWs.

Keywords: innovative small SFR; metal fuel core; radially heterogeneous core; inner blanket; Minor Actinide; transumutation

1. Introduction

1.1. Japanese approach to fast reactor development

The Japanese government has declared a policy to achieve carbon neutrality by 2050, and in accordance with this policy, the use of nuclear energy is specified in the Sixth Strategic Energy Plan and Energy White Paper 2022 [1]. Furthermore, from the viewpoint of satisfying social demands for nuclear power, the following are required: high safety, excellent economic efficiency, stable energy supply, reduction of the radiotoxicity and volume of high-level radioactive wastes (HLWs), non-proliferation of nuclear materials, and coexistence with renewable energy. Fast reactors that can effectively resolve these issues are attracting attention, and in the Japanese government's conference on fast reactor development in 2016, a policy regarding the continued development of fast reactors was presented in anticipation of the effective use of resources by fast reactors, the effect of reducing the potential hazards of HLWs, and the volume reduction of HLWs. In response to this policy, a strategic road map for fast reactor development was published in 2018. According to this roadmap, full-scale use of fast reactors is expected in the second half of the 21st century.

*Corresponding author. E-mail: Sho.fuchita.bj@hitachi.com

1.2. Issues in implementing fast reactors in Japan and Advantages of Metal-Fueled Fast Reactor Systems

From the viewpoint of effective use of uranium (U) resources, fast reactors and the nuclear fuel cycle must be introduced. In addition, plutonium (Pu) derived from light water reactors (LWRs) could be used in the start-up phase of fast reactors. Most current LWRs use enriched U as a fissile nuclide, but sustainability is an issue because this method requires continuous importation of natural U. For this reason, in Japan, nuclear fuel cycle facilities are under construction to recover Pu from spent fuel and manufacture mixed oxide (MOX) fuel. Japan already has a large amount of LWRs spent fuel for manufacturing MOX fuel, and we expect that MOX fuel fabricated from them will not only be loaded into LWRs but will also be used for the start-up of fast reactors. Eventually, achieving sustainability requires multi-recycling with fast breeder reactors. In addition, separation and recovery of minor actinides (MAs) contained in HLW and their transmutation in fast reactors can reduce the radiotoxicity of HLW [2].

In Japan, aqueous reprocessing will be used to reprocess LWR UO_2 spent fuel. The technology readiness levels (TRL) for aqueous reprocessing are in the practical stage if only U and Pu are recovered. However, additional processes are required to separate and recover MAs from HLW in aqueous reprocessing. Although various

approaches to separation and recovery are currently being considered for aqueous reprocessing, no methods are immediately available for demonstration on an engineering scale. On the other hand, pyrometallurgical reprocessing is applicable when metal fuels are used in fast reactors, and this technology is highly suitable for group separation of transuranic (TRU) elements. There is no need for an additional process for MA separation, and it is more reasonable when Pu and MAs need to be co-recovered [3]. In addition, although this technology has low decontamination factor, it has a high proliferation resistance because the recovered Pu is inevitably accompanied by MAs and a small amount of rare earth elements. Although the economies of scale may be limited for large-scale operations, pyrochemical reprocessing comprises small-scale batch processes, making it feasible to maintain economic viability even in small-scale facilities. However, establishment of the solid waste treatment technology remains an issue and ensuring operability in high temperature and inert atmosphere is a design consideration.

The Central Research Institute of Electric Power Industry is studying the practical application of pyrometallurgical reprocessing using LiCl-KCl molten salt. Fuel cycle tests using U and simulating elements have already been conducted using process equipment with a throughput of 1-ton heavy metal (HM)/yr. [4], and a series of electrorefining - cathode treatment - casting tests have achieved a mass balance of over 99% for Pu and americium (Am) [5]. In addition, electrochemical reduction tests have also been conducted on MOX fuel including MAs irradiated in a commercial reactor [6].

Based on the above, we have determined that a combination of pyrometallurgical reprocessing and metal fuel fast reactors is effective from the viewpoint of satisfying social demands for nuclear power. Examples of the combination of pyrometallurgical reprocessing and SFRs with metal fuel include the fast reactor cycle technology development project (FaCT) and the development of highly flexible technology for recovery and transmutation of MAs [7]. The former is based on the plan developed before Japan's nuclear power and fuel cycle policies were revised, so it is necessary to reconstruct a scenario that reflects the timing of SFR introduction and nuclear power generation capacity based on the current plan. The latter is based on a coexistence scenario of LWRs and SFRs. In this paper, we showed a fuel cycle scenario and material balance analysis based on the Japanese situation in 2022, considering the replacement of all Japanese LWRs by SFRs with metal fuel, as well as the corresponding core concept and its neutron analysis results.

2. Domestic deployment fuel cycle scenario of sodium-cooled metal fuel fast reactor

According to Japan's Energy White Paper, the share of nuclear power generation in 2030 is set at 20-22% [1]. This study examines the transition from LWR fuel cycle to a fast reactor fuel cycle, assuming that this nuclear power generation capacity will continue into the future.

Figure 1 shows the fuel cycle configuration considered in this study. LWR spent fuel is processed into MOX powder by aqueous reprocessing. We considered a scenario in which the MOX powder (mixed oxide of reprocessed uranium and plutonium) is electrochemically reduced to produce metal fuel. The metal fuel core loaded with this fuel is hereafter referred to as core A. The fresh core fuel isotopic composition of core A is shown in **Table 1**. The Pu isotopic composition was based on the ratio of PWR and BWR spent fuel masses and the Pu isotopic composition evaluated by Ando et al. [8] considering PWR and BWR spent fuels with discharged burnup of 45

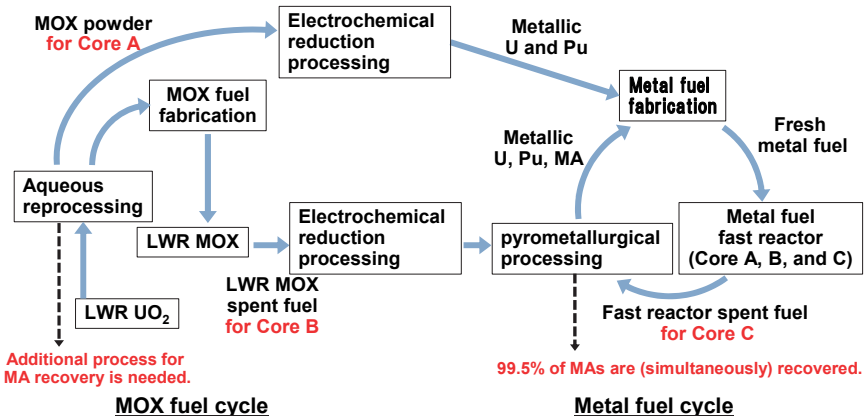

Figure 1. Fuel cycle scenario for this study.

Table 1. TRU composition of fresh fuel.

Core	TRU isotope weight ratio [wt.%] (Total TRU : 100)									
	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Am-241	Am-243	Np-237	Cm-244	Cm-245
Core A	2.5	56.4	27.0	5.9	8.2	—	—	—	—	—
Core B	2.1	35.7	31.6	6.2	9.5	11.2	2.3	0.6	0.5	0.1

GWd/t and cooling period of 20 years.

We considered that spent fuel from LWR MOX will be electrochemically reduced and pyrometallurgically reprocessed to produce metal fuel to be loaded into a fast reactor. The core loaded with this metal fuel is hereafter referred to as core B. The proportion of fissile Pu (Puf: Pu239+Pu241) in LWR MOX spent fuels is less than 50%, and even if reprocessed, it is difficult to reload them into LWRs from the viewpoint of criticality. Therefore, we intend to utilize core B as a recipient of LWR MOX spent fuel after 2080. The fresh fuel isotopic composition of Core B is shown in Table 1. The TRU composition was based on PWR and BWR MOX spent fuels considering 20 years cooling period with discharged burnup of 45 GWd/t and 33GWd/t respectively [8] and most of the TRU in LWR MOX spent fuel is transferred to the metal fuel cycle. As a result, the fresh fuel in the Core B contains certain amount of MA. Irradiation test results up to ~7 at.% burnup of fuel containing 5 wt.% MA per HM have reported no significant effects on physical properties [9].

The spent fuel from the metal fuel core is pyrometallurgically reprocessed and loaded back into the fast reactor itself. This metal fuel core is hereinafter referred to as the core C. When the breeding ratio exceeds 1.0, the Puf fraction of the fresh fuel does not decrease even after repeated reprocessing, thus enabling multiple recycling without replenishing fissile material from the outside. In addition, since more than 99% of MA is transferred to fresh fuel in pyrometallurgical reprocessing, the potential radiotoxicity of HLWs after the transition to the metal fuel cycle can be effectively reduced. In the study of core C, the spent fuel from core A is used as the starting point for multi-recycling, but it is expected that the spent fuel from core B can be repeatedly multi recycled to obtain an equilibrium core with the same fuel composition.

3. Development of Core Concept

3.1. Metal fuel core concept

Based on a small metal-fueled core developed in the United State [10,11], fuel assembly specifications and core configurations were determined. We conducted neutronic analysis for each of the cores (Core A, B and C) loaded with the three types of fuel described in Chapter 2. **Figure 2** shows the horizontal cross section and the vertical cross section approximated by the 2D R-Z model of the core A. **Table 2** shows the core specifications that are common to core A, B and C.

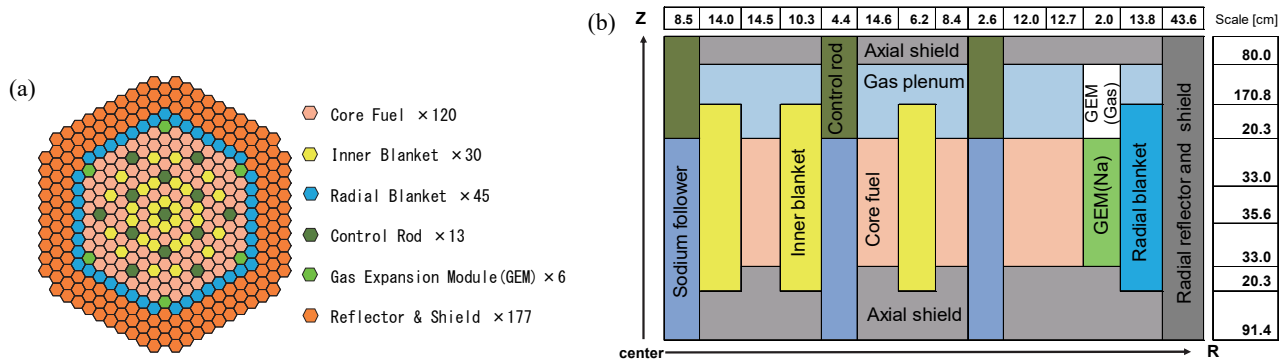

Figure 2. (a) Horizontal section of the core A, (b) Vertical section of the core A (1/2 core).

Table 2. Common core specifications.

Item	Unit	Value
Thermal power	MWt	840
Electric power	MWe	311
Primary sodium inlet/outlet temperature	℃	360/499
Core equivalent diameter(include shield)	cm	335.2
Assembly pitch	cm	16.142
Total number of C and IB assemblies	—	150
Height of fuel (C/IB/RB) *1	cm	101.6/142.2/142.2
Pin count (C/IB/RB) *1	—	271/127/127
Pin outer diameter (C/IB/RB) *1	mm	7.44/12.01/12.01
Fuel smeared density (C/IB/RB) *1	%TD	75/85/85
Fresh fuel materials (C/IB) *1	—	U-TRU-Zr/U-Zr*2
Fuel density	kg/cm^3	15.8（Constant value was used in this study）

*1 C: Core fuel, IB: Internal blanket fuel, RB: Radial blanket fuel.
*2 See Table 1 for TRU compositions. FPs are also included in fresh fuel of Core B and C. Zr isotopic composition ratio is natural.

These cores are radially heterogeneous in which the core region is also loaded with an inner blanket assembly composed of U-Zr fuel. In this core, the burnup of the inner blanket fuel affects the core criticality conditions. In contrast to the core fuel, which fissile material decreases with burnup, fissile material is produced with burnup in the inner blanket fuel. Therefore, the issue is to optimize the ratio of the number of assemblies of the core fuel to inner blanket fuel in order to achieve criticality conditions for the entire operation cycle. In a well-designed radially heterogeneous core, the inner blanket is loaded at a position with high neutron importance, which effectively suppresses the burnup reactivity swing during an operation cycle through Pu production. As a result, the reactivity required to the control rods in the beginning of cycle is reduced, which has the effect of reducing the control rod withdrawal reactivity during unprotected transient over power (UTOP). It also has the effect of suppressing the power peak in the center of the core and suppressing the reactivity increase when coolant sodium temperature rises or void in the core region.

In addition, gas expansion modules (GEMs) are installed in these cores. The GEM has an argon gas region and a sodium region inside, and its liquid level is set to be the same as the top of the core fuel during normal operation. The bottom end of the GEM is connected to the inlet plenum at the bottom of the core, and when the inlet plenum pressure drops due to the primary pump failure, the argon gas expands and the liquid level inside the GEM drops which promotes neutron leakage. The immediate negative reactivity effect by GEMs is effective against unprotected loss of flow (ULOF).

The inner blanket fuel assemblies loaded into core for 2 cycles are moved to radial blanket positions and further burned. This method can flatten the radial power profile and distribute the coolant flow more effectively. [11]

3.2. Neutronic analysis method

For nuclear property analysis, a seventy-group cross section library UFLIB.J40 for fast reactors based on JENDL-4.0 [12] was used and the effective cross section was calculated with SLAROM-UF [13]. Burnup characteristics and reactivity coefficients were calculated with CITATION-BURN [14] based on diffusion theory using a two-dimensional R-Z model. However, the neutron leakage effect dominates the GEM reactivity, and the evaluation by diffusion theory is not sufficient. Therefore, we evaluated the reactivity of GEMs using the SN neutron transport calculation code MINISTRI [15] using a three-dimensional TRI-Z model.

In cores B and C, which use pyrometallurgical reprocessed metal fuel, fission products (FPs) and MAs are included. As a result, there are effects of changes in neutron absorption and a decrease in the volume fraction of HM. On the other hand, it is not reasonable to calculate the burnup of all FPs in the neutronic analysis. In such cases, a method is known to select representative nuclides among FPs [16]. In this study, Nd-143 was selected as a representative nuclide of FPs in the core fuel, and the following equation was used to obtain an equivalence factor α such that the neutron absorption reaction rate is equivalent to that when all FPs are considered.

$$\alpha = \frac{\sum_{i\in FP}\sigma_c^i N_i}{\sigma_c^{Nd-143} N_{Nd-143}^{(reference)}} = \frac{\sum_{i\in FP}\sigma_c^i n_i/\sigma_c^{Nd-143}}{\sum_{j\in FP}\frac{A_{FP\,j}}{\rho_{FP\,j}}\Big/\frac{A_{Nd-143}}{\rho_{Nd-143}}} \quad (1)$$

where σ_c^i is the one-group neutron capture cross section of nuclide i (unit: barn), N_i is the number density of nuclide i, n_i ($\sum(i\in FP)n_i = 1$) is the FP isotopic composition ratio, A_{FP} is the chemical formula weight, and ρ_{FP} is density. Note that Nd-143 is not included for the blanket fuel. Therefore, Mo-95, which is the heaviest element among FPs, is stable, has a high isotopic fraction, and has a neutron capture cross section similar to that of Nd-143, was used as the representative FP nuclide in the blanket fuel. Next, the effect of decreasing the volume of other components by the volume fraction of FP is expressed by the following Eq. [16].

$$VF(FP) = \frac{V_{FP}}{V_{U-Pu-MA-Zr}+V_{FP}} \quad (2)$$

where $V_{U\text{-}Pu\text{-}MA\text{-}Zr}$ is the volume of metal fuel and V_{FP} is the volume of FP.

TRU and noble metals in FPs, which are transferred from spent fuel to fresh fuel by pyrometallurgical reprocessing, were considered to be 99.5 wt.% and 27 wt.%, respectively. Rare earth in FPs were to be transferred to make up 0.3wt.% of the fresh core fuel weight. However, no rare earth FPs were transferred to the blanket fuel. The ratio of Zr was adjusted so that the weight ratio of noble metal FPs and Zr is 10wt.%.

Inner blanket fuel assemblies are reloaded as a radial blanket fuel assembly. Therefore, an iterative calculation was performed in which the post-burnup composition of the inner blanket fuel was used as the initial loading composition of the radial blanket fuel until there was no change in the effective multiplication factor (keff) at the end of the equilibrium cycle (EOEC).

The FP isotopic composition of the spent fuel of cores A and C was evaluated by ORIGEN2.2 [17] applying ORLIBJ40 [18] based on JENDL4.0. The residence time and specific power of the fuel required for ORIGEN2.2 calculation were determined by CITATION-BURN. For the C core, the calculation of the fresh fuel composition by ORIGEN2.2 and the neutronic analysis by CITATION-BURN were repeated until there was no change in the Pu enrichment of the fresh fuel.

3.3. Neutronic design criteria

The goals and constraints of this study are described below. The maximum linear power should be less than 500 W/cm in accordance with the viewpoint of preventing fuel melt [19]. The Pu and neptunium (Np) enrichment should be less than 25 wt.% from the viewpoint of preventing liquid phase formation due to fuel-cladding chemical interaction (FCCI) [20]. Considering the close chemical properties of Pu and Np, the Pu enrichment limit is

conservatively considered as the ratio of Pu and Np to HM. The sodium void reactivity should be less than 8$ according to previous studies in order to prevent prompt criticality in ULOF initiating phase [19]. The breeding ratio is targeted to be breakeven (1.03 or higher), considering the large supply of Pu derived from LWRs spent fuel. The target average discharge burnup of the core fuel is more than 100 GWd/t. The GEM reactivity worth should be about -40¢ or smaller to avoid coolant boiling under ULOF conditions, referring to the ULOF evaluation results for small metal fuel fast reactor [21]

To achieve the above conditions, parameters such as the ratio of core fuel assemblies to blanket fuel assemblies, the number of refueling batches, Pu enrichment, and burnup period are determined by neutronic design criteria.

4. Results and discussion

Table 3 shows the results of neutronic analysis for core A, B, and C. For all cores, all design criteria set in the previous chapter were achieved by adjusting the effective full power days (EFPD), core fuel/blanket assembly number ratio, and Pu enrichment. The maximum linear heat generation rate (MLHGR) of core fuel is less than 400 W/cm for all cores. The results are based on a two-dimensional R-Z model and dose not considered three-dimensional effects such as fuel exchange patterns, but even if they are considered, the MLHGR is not expected to exceed the limit of 500 W/cm based on our previous studies.

Core B had the largest number of core fuel assemblies. This change was made because the Puf fraction of the fresh fuel is small, and the Pu and Np enrichment limit is exceeded with the same configuration as Core A. On the other hand, the breeding ratio is almost the same as that of core A despite the decrease in the number of blanket fuel assemblies. This is due to the reduced consumption of Puf by MA fissions and the large fraction of the parent nuclide, Pu240, in the fresh fuel. The operation cycle period of core B is longer than that of core A. This is because the core thermal power and the average discharged burnup of the core fuel remain the same despite the increase in the number of core fuel assemblies. The sodium void reactivity of core B is 7.9$. The reason for the increase compared to core A is that the higher neutron energy due to sodium voiding, increases the fission reaction rate of MAs.

Figure 4 (a) shows the changes in the Pu enrichment and Puf content of the fresh fuel of core C because of repeated pyrometallurgical reprocessing. The Pu enrichment decreases with each cycle, corresponding to an increase in the Puf content of the core fuel with each reprocessing cycle because of the high Puf content extracted from the radial blanket fuel. We determined that Pu enrichment reached equilibrium when the fresh fuel was reprocessed twelve times and defined the core at this point as the

Table 3. Nuclear performance of the core A, B and C.

Item	Unit	Core A	Core B	Core C (Equilibrium)	Design criteria
Pu and Np enrichment of the fresh core fuel ((Pu+Np)/HM)	wt.%	23.7	24.3	23.4	≦ 25 [20]
MA content in the fresh core fuel (MA/HM)	wt.%	0	4.2	0.7	≦ 5 [9]
FP content in the fresh core fuel (FP/(HM+Zr+FP))	wt.%	0	0.9	0.9	No target
FP equivalence factor α of fresh fuel(C/IB)	—	—/—	1.58/1.26	2.26/1.15	No target
Number of assemblies (C/IB/RB)*	—	120/30/45	132/18/45	114/36/54	Total C and IB = 150
Refueling batches (C/IB/RB)*	—	3/2/3	3/2/5	3/2/3	No target
Burnup period	EFPD	598	611	567.5	No target
MLHGR of core fuel	W/cm	357	364	300	≦ 500 [19]
Average discharge burnup (Core fuel)	GWd/t	106.3	106.3	106.3	≧ 100
Pu-fissile Breeding ratio	—	1.11	1.09	1.09	≧ 1.03
Effective delayed neutron fraction	%	0.355	0.332	0.349	No target
Burnup reactivity	$	5.1	4.8	4.0	Controllable with CRs
Sodium void reactivity at EOEC	$	6.0	7.9	6.2	≦8$ [19]
GEM reactivity	¢	-42	-37	-43	≒ -40¢ or smaller [21]

* C: Core fuel, IB: Internal blanket fuel, RB: Radial blanket fuel, CR: Control rod.

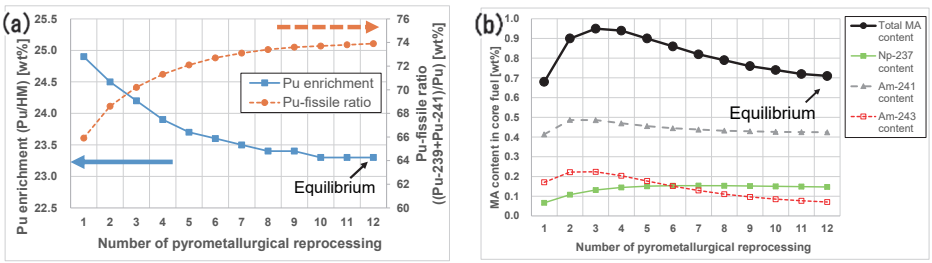

Figure 4. (a) Change in Pu enrichment and Puf fraction in multi-recycling of core C, (b) Change in MA fraction in multi-recycling of core C.

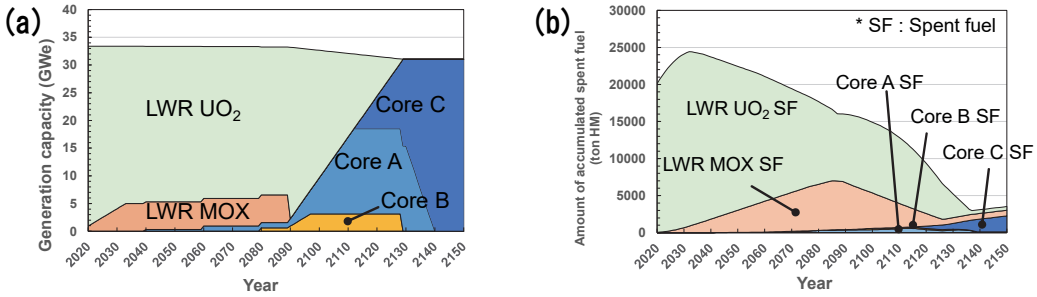

Figure 5. (a) Transition of nuclear power generation capacity, (b) Transition of amount of accumulated spent fuel.

equilibrium core.

The MA fraction peaked at about 0.95 wt.% and then gradually decreased, reaching equilibrium at about 0.7 wt.%. The reason for the decrease in the MA fraction is that the fraction of Pu-242 decreased due to multi recycling, resulting in a decrease in the fraction of Am-243 produced in the core fuel, as shown in **Figure 4 (b)**.

Figure 5 shows the power generation capacities and total amount of spent fuel accumulation of LWRs, core A, B, and C evaluated based on the mass balance of these spent fuel. The results showed that within the nuclear power generation capacity envisioned for Japan, the introduction of the metal fuel fast reactors and pyrometallurgical reprocessing would allow the transition to a metal fuel cycle after processing almost all spent UO_2 and MOX fuels from LWRs.

5. Conclusion

We have confirmed that cores of innovative small SFR with metal fuel of three fuel conditions based on the fuel cycle scenario satisfies all design criteria. The study of cores A and B using metal fuel fabricated from LWR UO_2 and LWR MOX spent fuel showed that, from the viewpoint of core design and mass balance of LWRs spent fuel , it is possible to transition from the LWR oxide fuel cycle using aqueous reprocessing to the metal fuel cycle using electrochemical reduction and pyrometallurgical reprocessing. The neutronic analysis of core C using metal fuel multi-recycled from metal fuel fast reactor spent fuel showed that the MA production and transmutation are balanced when the MA content reaches about 0.7 wt.%, and that most of the MA in the spent fuel is transferred to the metal fuel cycle through the pyrometallurgical reprocessing. On the premise of using already proven technology, the metal fuel cycle can be effective in reducing the radiotoxicity of HLWs.

In the future study, we plan to examine the possibility of recovering MA contained in spent fuel from domestic LWR UO_2 and burning it in a metal fuel core, as well as to conduct a safety evaluation for ULOF and UTOP accidents. In addition, it will be necessary to optimize the core design at the basic design stage although this study showed example cores suitable for the scenario based on material balance analysis from the implementation stage to the equilibrium period.

Acknowledgements

This research used resources of the Program to Support the Development of Innovative Nuclear Power Technologies in Response to Society's Needs of the Ministry of Economy, Trade and Industry.

References

[1] Ministry of Economy, Trade and Industry. FY2021 Annual Report on Energy (Japan's Energy White Paper 2022).

[2] M. Salvatores and P. Giuseppe, Radioactive waste partitioning and transmutation within advanced fuel cycles: Achievements and challenges, *Progress in Particle and Nuclear Physics* 66 (2011), pp. 144-166.

[3] P. Baron, et al., A review of separation processes proposed for advanced fuel cycles based on technology readiness level assessments, *Progress in Nuclear Energy* 117 (2019), 103091.

[4] M. Iizuka, et al., Performance of Pyroprocess Equipment of Semi-Industrial Design and Material Balance in Repeated Engineering-Scale Fuel Cycle Tests Using Simulated Oxide/Metal Fuels, *Nuclear Technology* 184 (2013), pp. 107-120.

[5] T. Koyama and M. Iizuka, Pyrochemical fuel cycle technologies for processing of spent nuclear fuels: Developments in Japan, *Reprocessing and Recycling of Spent Nuclear Fuel*, Woodhead Publishing, (2015), pp. 457-519.

[6] T. Kato et al., Actinides recovery from irradiated MOX fuel by pyrochemical reprocessing, *Proc. GLOBAL 2011*, Makuhari, Japan.

[7] K. Ohgama, et al., A design study on a metal fuel fast reactor core for high efficiency minor actinide transmutation by loading silicon carbide composite material, *Journal of Nuclear Science and Technology* 59 (2022), pp. 735-747.

[8] Y. Ando and H Takano, Estimation of LWR spent fuel composition, *JAERI-Research 99-004* (1999). [in Japanese]

[9] H. Ohta, et al., Development of fast reactor metal fuels containing minor actinides, *Journal of Nuclear Science and Technology* 48 (2011), pp. 654-661.

[10] B.S. Triplett, P.L. Eric and J.D. Brett, PRISM: a competitive small modular sodium-cooled reactor, *Nuclear Technology* 178 (2012), pp. 186-200.

[11] A.E. Dubberley, et al., SuperPRISM oxide and metal

fuel core designs, *Proceedings of ICONE 8* (2000), pp. 2-6.
[12] K. Sugino, et al., Fast reactor nuclear data set UFLIB.J40 and JFS-3-J4.0 based on JENDL-4.0. *JAEA-Data/Code 2011-017*, 2011.
[13] T. Hazama, et al., SLAROM-UF: Ultra fine group cell calculation code for fast reactor. Version 20090113 (Translated document), *Japan Atomic Energy Agency* (2009).
[14] K. Yokoyama et al., Development of general-purpose core analysis system MARBLE2, *JAEA-Data/Code 2015-009*, 2015.
[15] K. Sugino, et al., Development of neutron transport calculation codes for 3-D hexagonal geometry NSHEX, MINIHEX and MINISTRI, *Japan Atomic Energy Agency* (2012).
[16] S. Ohki, Influence of Remaining Fission Products in Low-Decontaminated Fuel on Reactor Core Characteristics, JNC TN9400 2002-066, *Japan Atomic Energy Agency* (2002). [in Japanese]
[17] A.G. Croff, *A User's Manual for the ORIGEN2 Computer Code*, ORNL/TM-7175, Oak Ridge National Laboratory (1980).
[18] K. Okumura, et al., A set of ORIGEN2 cross section libraries based on JENDL-4.0. ORLIBJ40, (2013).
[19] M.M. Uematsu, et al., Fuel and core design studies on metal fuel sodium-cooled fast reactor (4), (5) and (6). *Joint Research Report for JFY2009 – 2012* (2012), Japan Atomic Energy Agency report, JAEA-Research 2012-041. [in Japanese]
[20] K. Nakamura, et al., Reactions of uranium-plutonium alloys with iron, *Journal of Nuclear Science and Technology* 38 (2001), pp. 112-119.
[21] GE Nuclear,GEFR-00793, PRISM Preliminary Safety Information Document, Nov. (1986).