Innovative TRU Burning Fast Reactor Cycle Using Uranium-free TRU Metal Fuel - Core Design Progress -

Kazuo Arie[1], Yasushi Tsuboi[1], Akihiro Hara[1], Yuji Arita[2], Hirokazu Ohta[3]

[1]Toshiba Corporation, Yokohama, Japan

[2]University of Fukui, Fukui, Japan

[3]Central Research Institute of Electric Power Industry, Yokosuka, Japan

E-mail contact of main author: kazuo.arie@toshiba.co.jp

Abstract. Fast reactors have a capability to effectively burn TRU (transuranic) compared to LWR due to its higher fission-to-capture ratio of TRU and to reduce the burden of radioactive waste disposal. The most effective way to burn TRU is to use uranium-free TRU fuel since it does not produce any new TRU. In order to clarify the feasibility of uranium-free TRU burning fast reactor cycle with metal fuel, we have been investigating the related key technical issues not only on fuel cycle area but also reactor core area since October, 2014 under the contract with Ministry of Education, Culture, Sports, Science and Technology (MEXT) in Japan. In this paper, among the various investigation items in this study, the progress of the core design study is presented, which shows the feasible core to simultaneously achieve enhanced Doppler feedback and low sodium void reactivity for the uranium-free TRU metal fuel fast reactor, considering the evaluated fundamental properties of the fuel and secured core safety.

Key Words: Fast reactor, Uranium-free TRU metal fuel, TRU burning, Core design

1. Introduction

Fast reactors burn plutonium and minor actinides (i.e., TRU: transuranic elements) effectively due to its higher fission-to-neutron-capture ratio compared to LWR. However, U-TRU-fuel fast reactors also produce TRU while burning TRU because its fuel contains uranium. Then, conventional fast reactors, even with low conversion ratio of 0.75, burn only 0.18 tTRU/GWe/y, while LWRs produce about 0.22 tTRU/GWe/y. It means almost the same capacity of fast reactors as that of LWRs is needed to continuously burn TRUs from LWRs.

Considering that LWR would be the dominant nuclear power plants in the next few decades at least, it seems to be difficult to deploy many fast reactors in the near term. Therefore, in order to burn TRU from LWRs using fast reactors under such circumstance, TRU burning capability of a fast reactor needs to be improved. The most effective way is to use uranium-free TRU fuel since it does not produce any additional TRU. Such system could reduce the capacity of the TRU burner units and the associated fuel cycle facilities to about 1/5 and 1/8 respectively.

There have been many studies on uranium-free or fertile-free fuel system in the past [1-3]. The French study showed potential solutions using a neutron moderator against the uranium-free core issues such as the reduced Doppler feedback and the increased sodium void reactivity [3]. However, such types of the inert matrix fuel seem to need new reprocessing technology. On the contrary, uranium-free TRU metal could be reprocessed and fabricated based on pyroprocess and injection casting technologies without large modifications.

As for uranium-free metal fuel concept, there were many studies in the US. One of them was the US Accelerator-driven Transmutation of Waste (ATW) program which investigated an accelerator-driven transmutation system coupled with a subcritical fast reactor using uranium-free metal fuel [4-7]. However, accelerator facility causes economic penalty due to its large electricity consumption for the beam production. The US had also studied sodium-cooled Advanced Burner Reactor cores with various conversion ratios from 1.0 to 0.0 to burn TRU based on a critical fast reactor. It had shown that the Doppler feedback was significantly reduced and the sodium void reactivity was increased to greater than $10 for the uranium-free core [8]. However, our recent study has shown that introduction of neutron moderator with keeping coolant volume fraction conventional level would enhance the Doppler feedback while keeping low sodium void reactivity for uranium-free metal fuel core [9].

Therefore, we have initiated the four years' research program on TRU burning sodium-cooled fast reactor cycle using uranium-free TRU metal fuel since October, 2014 under the contract with Ministry of Education, Culture, Sports, Science and Technology (MEXT) in Japan. It consists of technology development on fuel, pyroprocess and core. Some of the interim results of the program had been presented at the past international conferences [10-13].

This paper updates and integrates the core design study including the previously presented results.

2. Core Design Approach

The schematic diagram of our core design approach for a uranium-free TRU metal fuel core is shown in *FIG.1*.

At first, we reconfirmed the core design issues mainly caused by absence of uranium in the metal fuel alloy. Table I shows the nuclear physics comparison of a uranium-loaded and a uranium-free metal fuel core characteristics under the same condition of core and fuel geometry such as core height, number of fuel pins and fuel pin diameter to clarify just "uranium-free" influence. The core criticality was adjusted by the ratio of TRU to Zr alloy amount for the uranium-free core. It does not employ any special countermeasure against the issues. The result indicates the following issues.

- If the criticality is adjusted only by TRU/Zr ratio, Zr content of the metal fuel alloy becomes too high for injection casting in fuel fabrication since the melting point of the TRU-62wt.%Zr metal alloy would be around 1,600 C.
- Doppler coefficient is too small to ensure prompt reactivity feedback during transients. This is the most critical issue.
- Burnup reactivity swing is too large to have a reasonable operating cycle length as a commercial reactor.

According to our past study, Doppler coefficient can be enhanced by introduction of neutron moderator into core. However, sodium void reactivity increases as Doppler coefficient improves in general as shown in *FIG.2* [10]. Therefore, optimization of both safety parameters is one of the important issues for a uranium-free fuel core design.

Another issue is reduction of thermal conductivity. For example, Pu-40Zr metal fuel could be approximately 60% of conventional U-20Pu-10Zr metal fuel as shown in *FIG.3* [13].

Table II shows the countermeasures against the core design issues. Among them, the optimization of Doppler feedback and sodium void reactivity seems to be most complicated since they influence each other via nuclear physics parameters such as change of neutron

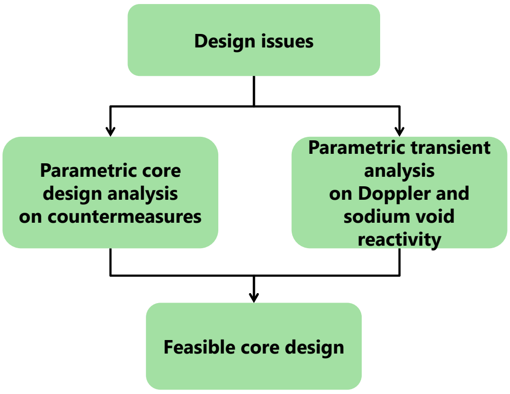


FIG. 1. *Core design approach.*

TABLE I: Comparison of U-TRU-Zr and TRU-Zr cores.

Characteristics	U-TRU-Zr Core	TRU-Zr Core w/o countermeasures
Reactor power (MWt)	714	same
Cycle length (days) x Refueling batch	148 x 5	same
Core height (cm)	93	same
No. of fuel assemblies	198	same
Fuel pin diameter (cm)	0.65	same
Volume fraction (Fuel/Na/SS) (%)	35.46 / 39.88 / 24.66	same
Fuel composition (U/TRU/Zr) (wt.%)	69.3 / 20.7 / 10.0	- / 38 / 62
Burnup reactivity swing (%dk/kk')	1.7	6.5
Ave. discharged burnup (at.%)	6.6	27.8
Doppler coefficient (Tdk/dT)	-0.0032	-0.00095
Na void reactivity (%dk/kk')	2.23	-0.02
Delayed neutron frac., βeff	0.0035	0.0025
Prompt neutron life time, lp (μs)	0.34	0.65

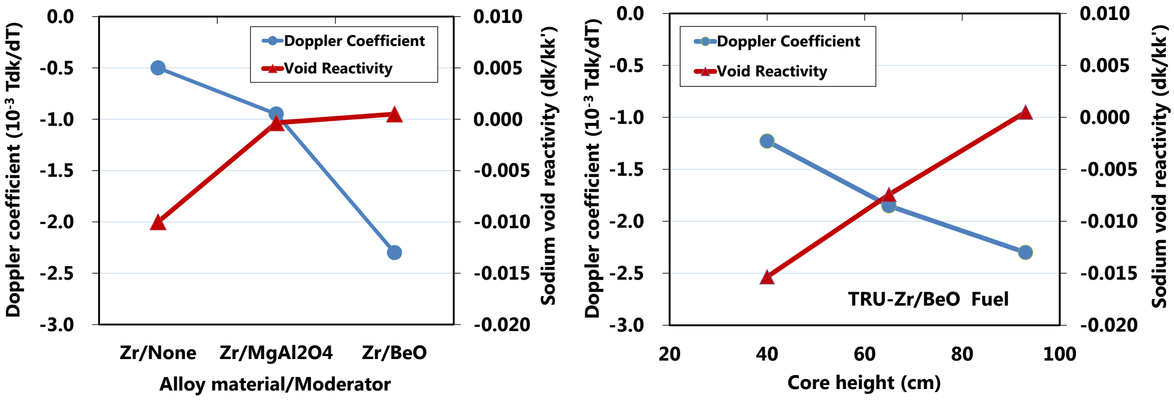


FIG. 2. *Tendency of Doppler coefficient and sodium void reactivity.*

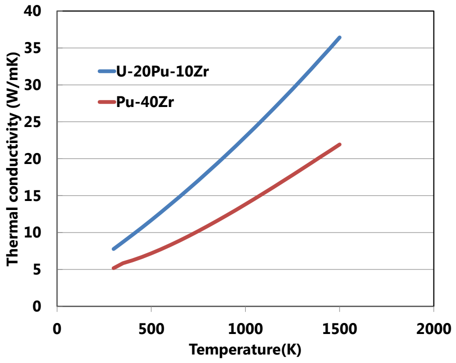


FIG. 3. *Thermal conductivity of Pu-40Zr.*

TABLE II: Countermeasures against uranium-free TRU metal fuel core.

Issues	Countermeasures	
Fuel meting point	- Zr content in fuel alloy, 30 – 40 wt.% to keep melting point around 1,100 C to 1,400 C. - Reduce number of fuel pins to avoid high Zr content	
Fuel thermal conductivity	Reduce linear heat rate	
Enhance Doppler coefficient	Spectrum softening by neutron moderator	- $MgAl_2O_4$ - BeO
	Alternative fuel alloy with large resonance absorption	- TRU-Mo - TRU-Nb (instead of TRU-Zr)
Sodium void reactivity upon Doppler enhancement	Increase of neutron leakage	- Reduced core height
Burnup reactivity swing	Increase of fuel inventory	- High neutron-leakage core
	Burnable poison	- B_4C with neutron moderator

spectrum and neutron leakage. In addition, burnup reactivity needs to be considered in couple with those safety parameters.

Then, core design parametric analysis has been done to clarify each effect of the countermeasures and the relationship of the core design parameters. Although alternative fuel alloys were evaluated as a potential way, this paper mainly focuses on core design with conventional metal fuel alloy, Zr.

Parametric transient analysis has been also done to clarify the requirements to Doppler coefficient and sodium void reactivity which enable to prevent fuel melting and sodium coolant boiling even during Unprotected Transient Overpower (UTOP) event and Unprotected Loss of Flow (ULOF) event.

Finally, a feasible core design using uranium-free TRU metal fuel has been developed.

3. Parametric Core Design Analysis

In order to clarify each effect of the countermeasures against the issues (i.e., Doppler coefficient, sodium void reactivity and burnup reactivity swing), parametric core design analysis has been done.

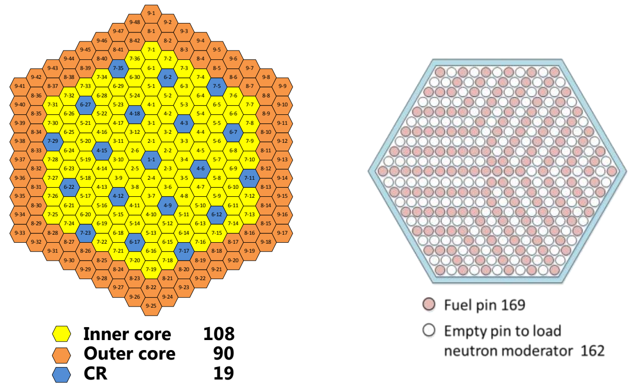
3.1. Analysis Conditions and Methods

FIG.4 and Table III shows the assumed core layout and specifications. The reference core has TRU-Zr fuel pins and empty pins to accommodate neutron moderators in fuel subassembly for parametric analysis. The fuel subassembly geometry and dimension of horizontal section are the same for all cases.

The effects of neutron moderators (i.e., $MgAl_2O_4$ and BeO) and alternative fuel alloy material (i.e., Mo and Nb) on Doppler coefficient were evaluated. The effect of core height reduction on sodium void reactivity was evaluated while keeping core volume constant based on 2-dimentional R-Z geometry. The effect of B_4C introduction on burnup reactivity swing was also evaluated as a function of the B_4C amount for homogeneous and heterogeneous loading. In homogeneous loading, it is assumed that natural boron B_4C is loaded in BeO moderator pins and its amount is varied by the ratio of B_4C to BeO. In heterogeneous loading, B_4C subassemblies with 80%-^{10}B enrichment are loaded as shown in *FIG.5*.

Regarding the calculation method, core burnup characteristics were analyzed by the 2-dimentional burnup calculation code, "STANBRE"[1]. Reactivity coefficients were analyzed using the diffusion calculation code, "DIF3D"[1]. The effective cross sections used in these calculations were obtained by the "SLAROM-UF"[1] code based upon 70-group cross sections from JENDL-4.0 [14] with self-shielding factor table. Homogeneous composition of fuel, structural material, neutron moderator, and coolant is assumed in core region.

The Doppler coefficient was evaluated by raising temperature of fuel and alloy material from 1000K to 1500K uniformly. Sodium void reactivity was evaluated assuming 100% coolant voiding in core and upper gas plenum region.



Core layouts Subassembly geometry

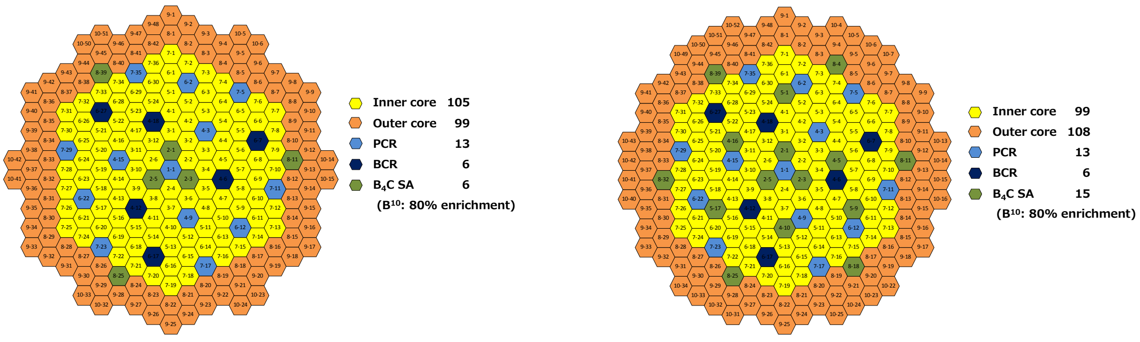
FIG. 4. Core and fuel layout for parametric analysis

TABLE III: Core specifications for parametric analysis

Items	Conditions
Reactor power	714 MWt
Cycle length	148 days
Metal fuel alloy	Zr (ref.), Mo or Nb
Core height	93, 65 or 40 cm
Core volume	2335 L (constant)
No. of fuel assemblies	198
Fuel pin diameter	0.475 cm
No. of pins per assembly - TRU fuel pin - Neutron moderator or burnable poison pin	Total 331 169 162
Neutron moderator	BeO, $MgAl_2O_4$ or none
Burnable poison	B_4C: Homo. or hetero. layout

---

[1] Product names may be trademarks of their respective companies.



6 B_4C SAs   15 B_4C SAs

FIG. 5. Heterogeneous loading patterns for B_4C subassemblies.

3.2.Analysis Results

The analysis result on the countermeasures against Doppler coefficient and sodium void reactivity is shown as functions of neutron slowing-down power of the core region and core height in *FIG.6 [12]*. It is found that these reactivity parameters can be roughly correlated by the two metrics, i.e., the slowing-down power and the core height. The absolute value of the negative Doppler coefficient increases almost linearly and sodium void reactivity slightly increase as the slowing-down power increases. On the contrary, the Doppler coefficient degrades and the sodium void reactivity decreases as the core height reduces.

In order to easily extrapolate Doppler coefficient and sodium void reactivity as the functions of the slowing-down power and the core height, contour maps for them have been produced as shown in *FIG.7*. This enables to easily find feasible design points if the required Doppler coefficient and sodium void reactivity is clarified.

FIG.8 summarizes the influence on burnup reactivity swing by the countermeasures against Doppler coefficient and sodium void reactivity [10]. It has been found that burnup reactivity swing is almost linear relationship to fuel burnup per an operating cycle regardless of the various countermeasures. Core height reduction is effective to reduce burnup reactivity swing since it reduces fuel burnup due to the increase of TRU (i.e., heavy metal fuel) inventory.

Regarding B_4C introduction, burnup reactivity swing reduces as the amount of B_4C increases for both homogeneous loading and heterogeneous loading as shown in *FIG.9* since TRU inventory increases by about 16% for the 10%-B_4C homogeneous loading case and about 32 % for the heterogeneous loading case of 15-B_4C-SAs. However, Doppler coefficient is degraded for both loading schemes, and sodium void reactivity increases for homogeneous loading while it slightly decreases for heterogeneous loading as shown in *FIG.10*. Thus, heterogeneous loading seems to be preferable. However, heterogeneous loading needs to be further evaluated to apply the core design since it could cause distortion of power distribution and decrease of control rod worth.

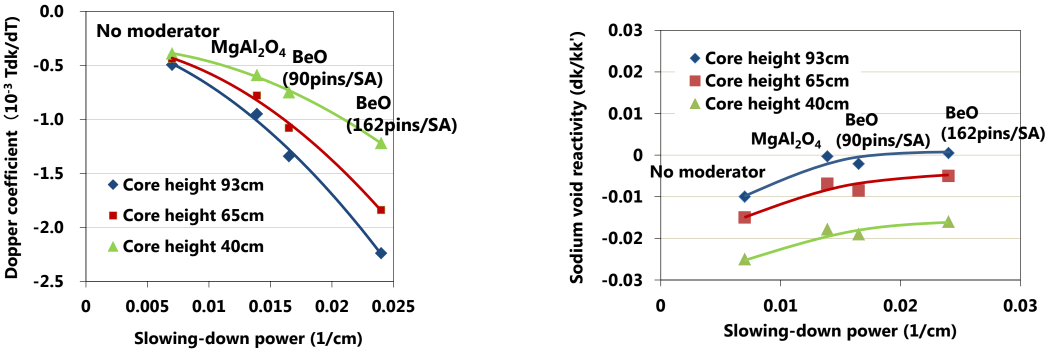


FIG. 6.Effect of countermeasures against Doppler coefficient and sodium void reactivity.

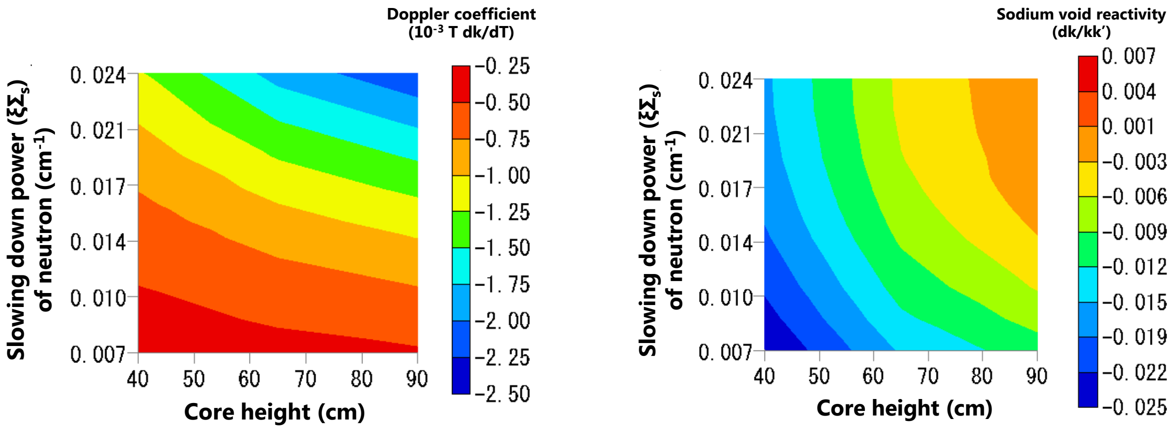


FIG. 7. Contour maps for Doppler coefficient and sodium void reactivity.

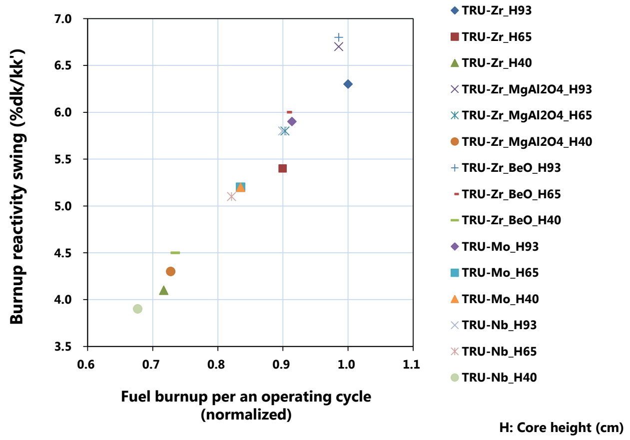
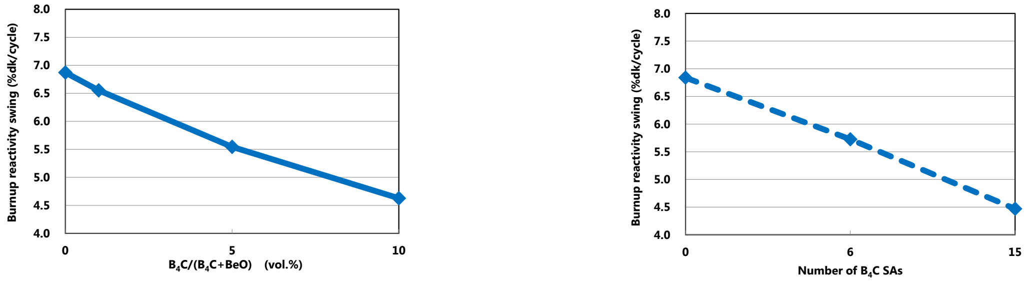
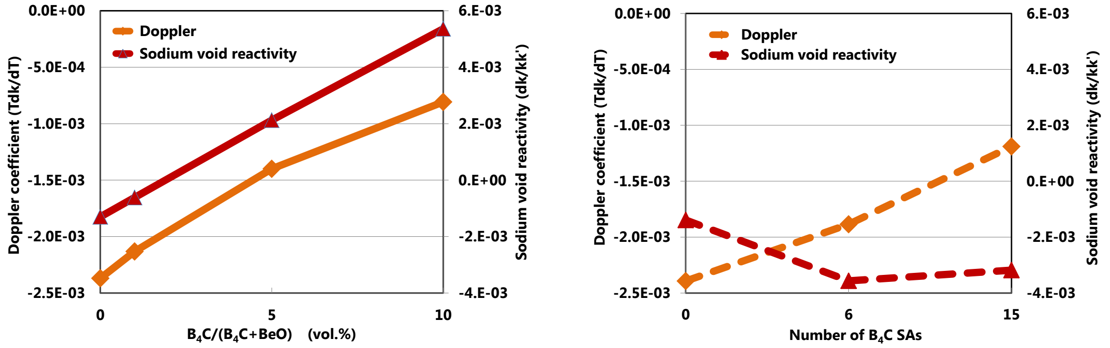


FIG. 8. Burnup reactivity swing for the countermeasures as a function of fuel burnup per cycle.



Homogeneous loading Heterogeneous loading

FIG. 9. Burnup reactivity swing as a function of B_4C amount.



Homogeneous loading Heterogeneous loading

FIG. 10. Doppler coefficient and sodium void reactivity as a function of B_4C amount.

4. Parametric Transient Analysis

Parametric transient analysis has been done to clarify the requirements to Doppler coefficient and sodium void reactivity which enable to prevent fuel melting and sodium coolant boiling even during Unprotected Transient Overpower (UTOP) event and Unprotected Loss of Flow (ULOF) event for a uranium-free TRU metal fuel core.

Table IV shows the transient analysis conditions. The fuel material is TRU-40Zr, which is considered as a representative fuel composition for a uranium-free TRU metal fuel. The reduction of thermal conductivity due the absence of uranium has been taken into account as shown in *FIG.3*, then the maximum linear heat rate at normal operating condition is assumed to be 210w/cm. The core height is assumed to be 65 cm considering the parametric core design analysis in Section 2. The insertion reactivity for the UTOP, 60 cents, is assumed considering relatively large burnup reactivity swing of TRU-Zr metal fuel core.

FIG.11 shows the obtained safety requirement map on Doppler coefficient and sodium void reactivity by this parametric transient analysis. It indicates no fuel melting area for UTOP event. As for ULOF event, the whole area within the figure can prevent coolant boiling on account of inherent safe feature of metal fuel core.

TABLE IV: Transient analysis conditions

Items	UTOP	ULOF
Reactor power	714 MWt	
Reactor inlet/outlet temp	350 / 482 C	
Pri. coolant circulation time	approx. 20 s	
Pri. pump halving time	N/A	10 s
Core height	65 cm	
Fuel pin diameter	0.475 cm	
Fuel material	TRU-40Zr	
Rated max heat rate	210 w/cm	
Insertion reactivity	60 cents	N/A
Reactivity insertion rate	0.3 cents/s	N/A
Duration time	1,000 s	
Core radial expansion	considered	
Doppler coefficient	parameter	
Sodium void reactivity	parameter	
Criteria	No fuel meting (< 1,150 C)	
	No coolant boiling (< 920 C)	

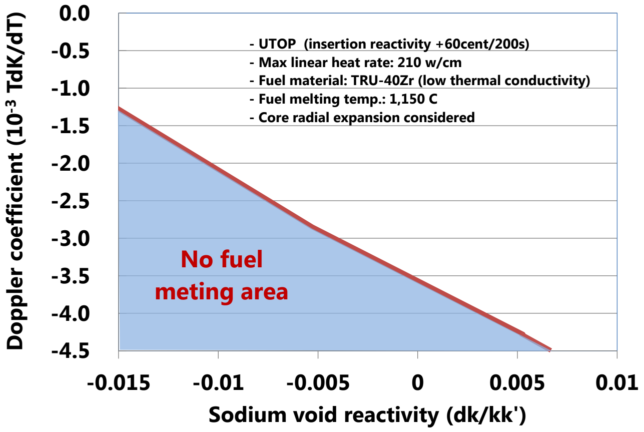


FIG. 11. Safety requirement map on Doppler coefficient and sodium void reactivity for UTOP.

5. Feasible Core Design

A feasible core has been developed using the obtained results described above.

At first, the key design parameters, i.e., core height and slowing-down power have been selected to satisfy the safety requirement as shown in *FIG.12*. Although the safety requirement map indicates there is a broad area to meet the requirement, the area for Doppler coefficient less than approximately -2.3 x 10^{-3} Tdk/dT could not be reasonably achieved according to the parametric analysis described in Section 3. Then, Doppler coefficient between -1.5 x 10^{-3} and -2.3 x 10^{-3} Tdk/dT is considered to be a practically achievable range. The corresponding sodium void reactivity are roughly -0.014 and -0.009 dk/kk' respectively. Those bounding lines are plotted on the contour maps for Doppler coefficient and sodium void reactivity. In order to satisfy the safety requirement, core height and slowing-down power should be selected between the two bounding lines on both Doppler coefficient map and sodium void reactivity map. Then, we have selected the core height, 55 cm and the slowing-down power, 0.024 /cm.

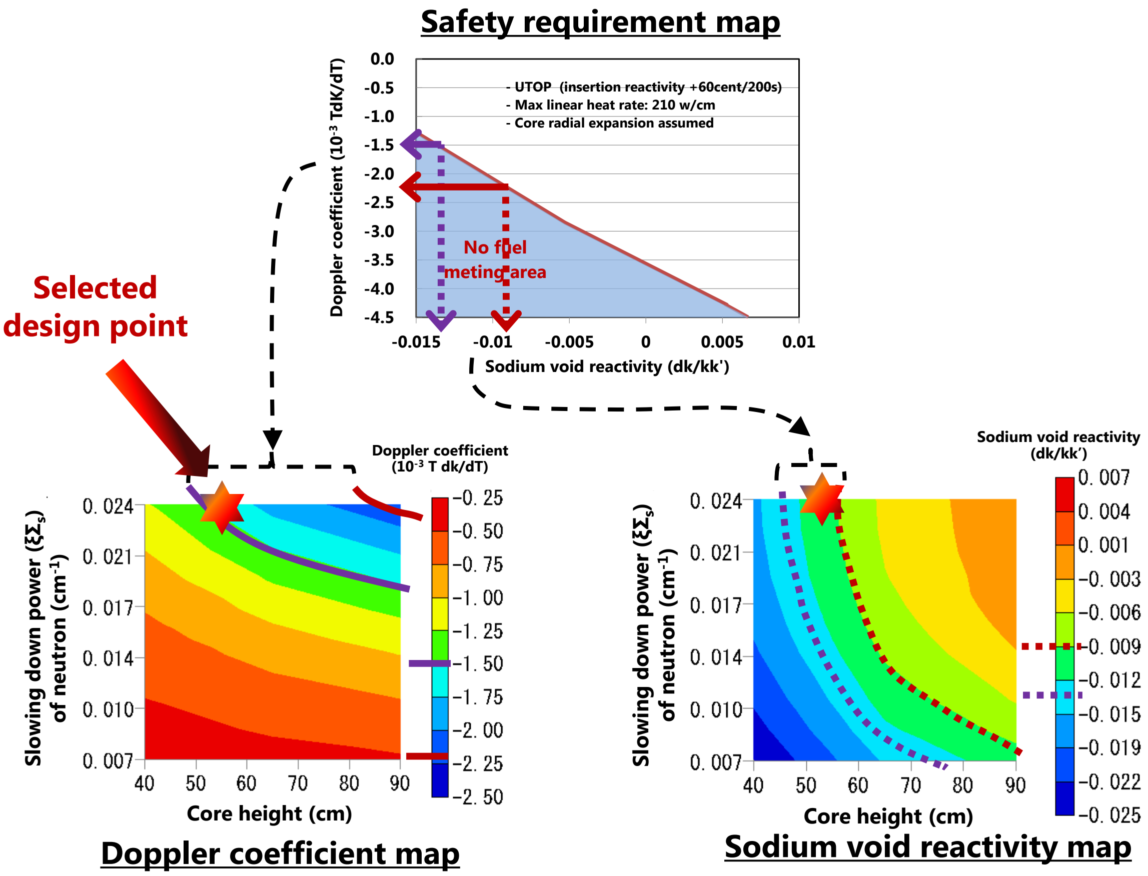


FIG. 12. Schematic diagram on how the core height and the slowing-down power was selected.

Based on the above, the developed feasible core is indicated in *FIG.13* and Table V which shows the core configuration and the major design parameters respectively. BeO pins are introduced in fuel subassembly corresponding to the slowing-down power, 0.024 /cm. The reduced core height is also preferable for reduction of burnup reactivity swing as well as sodium void reactivity. To accommodate the low thermal conductivity of uranium-free metal fuel, average linear heat rate is reduced to 110 w/cm. Zr content in metal fuel is kept constant, 40 wt.% to keep appropriate melting point of fuel. Then, TRU inventory has been controlled by adjusting the numbers of fuel pins and BeO pins per fuel subassembly to keep core criticality, and the ratio of the number of the two kinds of pins has been selected differently for inner core and outer core to flatten radial power distribution.

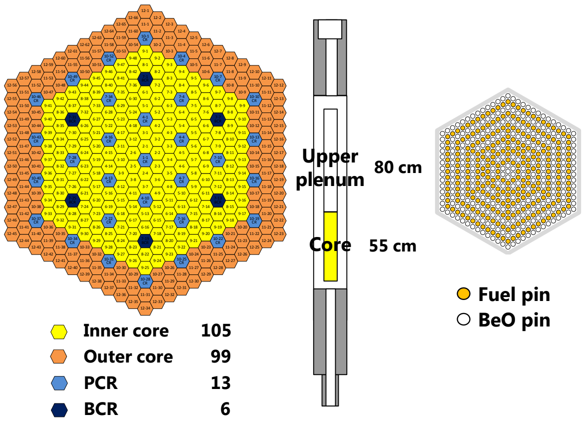


FIG. 13. Developed core configuration.

TABLE V: Developed core specifications

Items	Values
Reactor power	714 MWt
Reactor inlet/outlet temp.	350 / 482
Cycle length	148 days
Refueling batch	8
Core height / diameter	55 / 307 cm
Core volume	3695 L
Power density	193 w/cm³
Ave. linear heat rate	110 w/cm
SA pitch	14.68 cm
Fuel composition	TRU-40wt.%Zr
TRU inventory	3.2 tHM
TRU composition	LWR discharged, 10 yrs cooled
Fuel and BeO pin diameter	0.475 cm
Fuel smear density	75 %TD
No. of pins per SA - Fuel pin (IC/OC) - BeO pin (IC/OC)	547 per SA 279 / 389 268 / 158

TABLE VI: Developed core performances

Items	Values
Burnup reactivity swing	3.5 %dk/kk'
Ave. discharge burnup	24 at.%
TRU consumption	0.11 tHM/cycle
Max. linear heat rate IC (BOL/EOL) OC (BOL/EOL)	 215 / 127 w/cm 185 / 125 w/cm
Max. fast neutron fluence	1.6 x 10^{15} n/cm^2
Pri. control rod worth	14.4 %dk/kk'
Backup control rod worth	3.7 %dk/kk'
Doppler coefficient	-1.9 x 10^{-3} Tdk/dT
Sodium void reactivity	-1.1 $
Delayed neutron fraction, βeff	0.0025
UTOP behavior Insertion reactivity Peak fuel temperature	 60 cents 1,139 C
ULOF behavior Pri. pump halving time Peak cladding temperature	 10 s 853 C

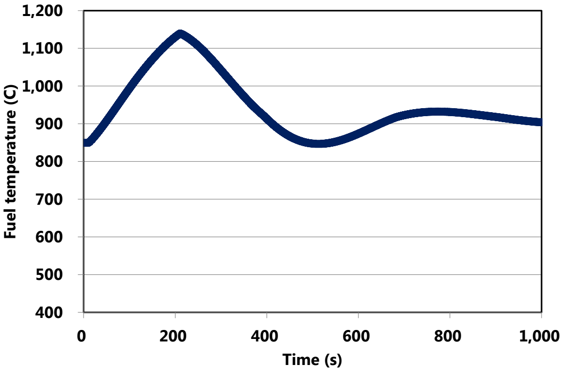


FIG. 14. Peak fuel temperature during UTOP.

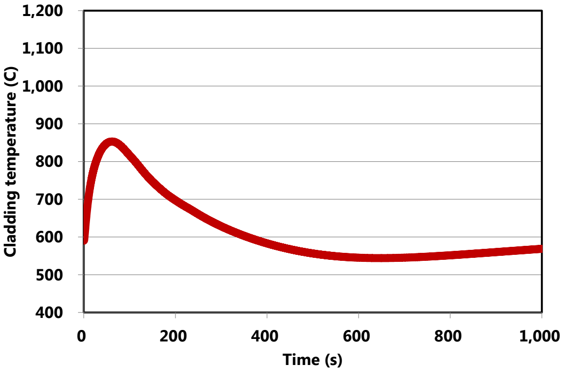


FIG. 15. Peak cladding temperature during ULOF.

Table VI, *FIG.14 and 15* show the performances of the developed core including transient behaviors for UTOP and ULOF. It is found that there is a feasible uranium-free TRU metal fuel core with controllable burnup reactivity swing, i.e., 3.5 %dk/kk' while achieving passive safety even for UTOP and ULOF events. Thus, the fundamental feasibility of the core design has been confirmed.

6. Conclusions

The core design progress in the four years' research program on TRU burning fast reactor cycle using uranium-free TRU metal fuel is presented. As the result, we have found that there is a feasible uranium-free TRU metal fuel core.

Further study is being done to improve the core characteristics from various viewpoints, such as higher core power density, longer operating cycle, TRU multi-recycling, severe accident behavior, alternative fuel alloy material and others.

The other planned researches, such as pyroprocess of uranium-free metal fuel and fuel irradiation performances, are also in progress. This program will continue through March, 2018.

7. Acknowledgements

This research is being conducted as the nuclear system research and development program under the contract with Ministry of Education, Culture, Sports, Science and Technology (MEXT) in Japan.

8. References

[1] YAMASHITA, T., et al., "Rock-Like Oxide Fuels and Their Burning in LWRs," Journal of Nuclear Science and Technology, Vol. 39, No. 8, p. 865-871 (2002).

[2] MIWA, S., et al., "MgO-Based Inert Matrix Fuel for A Minor Actinide Recycling in A Fast Reactor Cycle," Global 2011, Makuhari, Japan, December 11-16 (2011).

[3] MESSAOUDI, N. and TOMMASI, J., "Fast Burner Reactor Devoted to Minor Actinide Incineration," Nucl. Technol., Vol. 137, p.84-96 (2002).

[4] HILL, R. H. and KHALIL, H. S., "Physics Studies for Sodium Cooled ATW Blanket," Proceedings of the IAEA technical committee meeting on core physics and engineering aspects of emerging nuclear energy systems for energy generation and transmutation, Argonne National Laboratory (2000).

[5] BELLER, D. E., et al., "The U.S. Accelerator Transmutation of Waste Program," Nuclear Instruments and Methods in Physics Research, A 463, p.468-486 (2001).

[6] HEIDET, F., KIM, T. K. and TAIWO, T. A., "Multiple-Stage Fuel Cycle Options Based on Subcritical Systems," 2013 ANS Winter meeting, Washington DC, November 10-14 (2013).

[7] MEYER, M. K., et al., "Fuel Design for The U.S. Accelerator Driven Transmutation System," Nuclear Application in the New Millennium (accApp-ADITTA'01), Reno, Nevada, November (2001).

[8] HOFFMAN, E. A., TANG, W. S. and HILL, R. N., “Preliminary Core Design Studies for the Advanced Burner Reactor over a Wide Range of Conversion Ratios”, Argonne National Laboratory, ANL-AFCI-177, September 29 (2006).

[9] ARIE, K., YAMAOKA, M., MOIKI, Y., KAWASHIMA, M., OMORI, T., ISHII, K. and TSUBOI, Y., “TRU Burning Fast Reactor Cycle Using Uranium-free Metallic Fuel,” Paper ID 14136, ICAPP 2014, Charlotte, USA, April 6-9 (2014).

[10] ARIE, K., TSUBOI, Y., YAMAOKA, M., MORIKI, Y., ASANO, K., KIMURA, R., ARITA, Y., UNESAKI, H. and IIZUKA, M., “Innovative TRU Burning Fast Reactor Cycle Using Uranium-free TRU Metal Fuel (1) Overview and Progress of Core Design Study,” Paper ID 5096, Global2015, Paris, September 20-24, 2015.

[11] ARITA, Y., TSUBOI, Y. and OHTA, H., "Innovative TRU Burning Fast Reactor Cycle Using Uranium-free TRU Metal Fuel (2) Fundamental Properties of Uranium-free TRU-Zr Metal Fuel," Paper ID 5093, Global2015, Paris, September 20-24, 2015.

[12] ARIE, K., TSUBOI, Y., OMORI, T., ARITA, Y., UNESAKI, H., IIZUKA, M., and OHTA, H., “Program Overview on Development of Innovative TRU Burning Fast Reactor Cycle with Uranium-free Metal Fuel,” 14th Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation, San Diego, USA, October 17 -20, 2016.

[13] TSUBOI, Y., ARIE, K., PAPESCH, C., MILLER, B., MARIANI, R., ARITA, Y. and OHTA, H., “Fundamental Properties for Evaluation of Uranium-free TRU-Zr Metal Fuel Performance,” ANS 2016 Winter Meeting, p.463-466, Las Vegas, Nov. 6 -10, 2016.

[14] SHIBATA, K., et al., "JENDL-4.0: A New Library for Nuclear Science and Engineering," J. Nucl. Sci. Technol., 48(1), pp1-30 (2011).