A Neutronic Analysis of the Dry Processed Nitride Fuel in a Sodium-Cooled Fast Reactor

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1. Introduction

Nitride fuel is being considered as an alternative to oxide fuel for fast reactors owing to its high thermal conductivity. Because the nitride fuel is composed of one nitrogen atom per one heavy metal atom instead of two for the oxide fuel, it leads to a hardened neutron spectrum and consequently to a higher breeding ratio. In addition, the nitride fuel has other excellent characteristics such as a high material density (13.5 g/cm^3), high melting temperature (~2,800℃), no interaction with a sodium-bond, good compatibility with the cladding material, etc. Recently, researches on the nitride fuel such as the fuel fabrication process and irradiation test have been actively carried out in the U.S., Russia, Japan, etc.

In this study, the neutronic feasibility of the dry process nitride fuel cycle is assessed for a sodium-cooled fast reactor (SFR), which was recommended as one of the Generation-IV (Gen-IV) reactor systems by the Gen-IV international forum. The reactor analysis is performed for two core configurations: Hybrid BN-600 benchmark core with an enlarged lattice pitch (Case 1) and a modified BN-600 core (Case 2). Both cores are composed of two core regions and set as a breakeven core without blankets, which avoids the separation process of transuranic (TRU) elements from the spent fuel and the supply of additional fissile material.[1] In this study, the reactor characteristics such as the TRU enrichment, breeding ratio, peak linear power, burnup reactivity swing, etc. are obtained for the equilibrium core under a fixed fuel management scheme.

2. Sensitivity Calculation

The core calculation was performed by the REBUS-3 code [2] using the KAFAX-F22 library [3], which is an 80-group neutron and 24-group gamma cross-section library based on JEF-2.2. The TRANSX [Ref. 4] and TWODANT [Ref. 5] codes were used to generate 9-group region-wise effective macroscopic cross-sections. The fission products not included in the burnup chain were represented by lumped fission products (LFP), of which the cross-sections were generated from the KAFAX-E6FP cross-section library.

In this study, only neutronics calculations are performed based on the following external fuel cycle strategies: i) 95% of the rare-earth and all other fission products are removed, ii) all the uranium isotopes and 99.9% of the TRU are recovered, and iii) all surplus fuel materials after the dry process are sold. The external feed materials were composed of the TRU recovered from the typical light water reactor spent fuel and depleted uranium. For the nitride fuel with the natural nitrogen, the hazardous ^{14}C is formed by the ^{14}N(n,p)^{14}C reaction. Therefore, 100% enriched ^{15}N was used in this study. The equilibrium mode calculation was performed to establish a self-sustaining breakeven core without an external source of fissile material, aiming at a breeding ratio of 1.05.

2.1. TRU content

Because the breeding ratio strongly depends on the neutron spectrum of the core, an appropriate volume fraction of the fuel was searched. Then the sensitivity calculations for the TRU enrichments of the inner and outer core were performed to reduce the peak linear power density and flatten the power distribution of the core. The results showed that the breeding ratio decreases and the peak power position moves from the core center to the outer core region as the TRU enrichment of the outer core increases. For Case 1, when the fuel volume fraction was 39%, the TRU enrichment of the inner/outer core is 12.1%/15.3% to achieve the target breeding ratio of 1.05. The peak linear power density and burnup reactivity swing of this case are 272 W/m and 488 pcm, respectively. The calculation results for the optimum cases are summarized in Table 1.

2.2. Reactivity coefficient

The fuel temperature coefficients and coolant void reactivity were calculated using the finite difference method in the triangular-z node.

As the TRU enrichment of the outer core increases at the beginning of the equilibrium cycle (BOEC), the coolant void reactivity decreases and the fuel temperature coefficient increases in contrast to the void reactivity. However, the void reactivity and the fuel temperature coefficient at the end of the equilibrium cycle (EOEC) are greater than those of the BOEC, because the fissile plutonium isotopes are significantly accumulated and ^{238}U transmutes as the fuel is irradiated. For Case 1, if the fuel volume fraction is 39%, the void reactivity and fuel temperature coefficient are 3819 pcm and -1.05 pcm/K, respectively, at the BOEC, while they are 3891 pcm and -1.02 pcm/K at the EOEC.

2.3. Fuel mass flow and inventory

The fuel mass flow of the external fuel cycle was calculated to estimate the amount of fuel required and/or removed for each external cycle step when achieving the equilibrium core. The fissile plutonium gain during the equilibrium cycle is ~35 kg for all the selected cases, which satisfies the self-sustaining breakeven core without an excess fissile material. The amount of minor actinides is slightly reduced during the equilibrium cycle in the core but is accumulated a little during the external cycle by the decay. The total amount of minor actinides to be sold is ~1.8 kg for all the cases.

2.4. Fission product removal rate

For the dry process, it is important to remove some of the fission products which have negative offsets on both the fuel performance and mass balance of the recycled fuel. Parametric calculations were performed to estimate the fission products removal rate required to establish an equilibrium recycling fuel cycle. As shown in Fig. 1 for Case 1 with the fuel volume fraction of 39%, the rare-earth fission products should be removed by 35% if other fission products are removed by 70%. Because the removal rate of the fission products strongly depends on the breeding ratio, the fuel volume fraction should be high enough to achieve the equilibrium core, if the fission products removal rate is low.

3. Conclusion

In this study, two kinds of SFR with the nitride fuel were analyzed for the fuel volume fraction and TRU enrichment without considering the detailed design of the fuel assembly and the fuel channel. The SFR with the nitride fuel has shown better reactor characteristics than those of the oxide fuel SFR. However more studies are required for the fuel fabrication and irradiation test for the nitride fuel to be practically used in the reactor. If the design criteria used in this study is proved to be acceptable through a detailed physics design and thermal hydraulic analysis in the future, it is possible to construct an equilibrium fuel cycle of the SFR based on the nitride fuel by utilizing the dry process technology.

ACKKNOWLEDGEMENT

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Table 1. Summary of calculation results

	Case 1		Case 2	
V_f	39%	40%	40%	41%
TRU	12.1/15.3	11.7/14.8	11.9/15.2	11.3/15.7
BR	1.05333	1.06494	1.05445	1.05729
$\Delta\rho_{BU}$	488.0	597.4	943.6	1141.0
$P'_{\max}$	272.3	274.1	534.4	500.1
α_V^{BOEC}	3819	3749	3819	3760
α_V^{EOEC}	3892	3818	3804	3751
α_T^{BOEC}	-1.051	-1.043	-1.010	-1.004
α_T^{EOEC}	-1.021	-1.015	-0.949	-0.949

Note: V_f (fuel volume fraction), TRU (TRU enrichment inner/outer, %), BR (breeding ratio), $\Delta\rho_{BU}$ (burnup reactivity swing, pcm), $P'_{\max}$ (peak linear power density, W/m), α_V (void reactivity, pcm), α_T (fuel temperature coefficient, pcm/K)

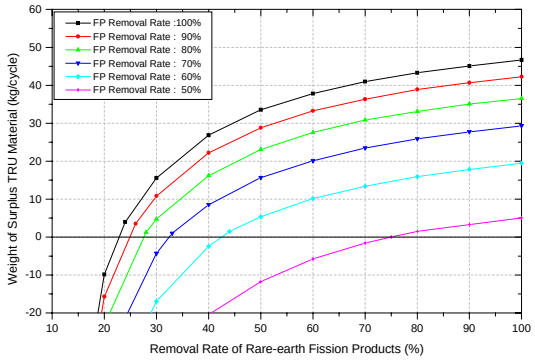

Fig. 1. Weight of the surplus TRU material depending on the removal rate of fission products