Neutronic Analysis of the Dry Process Oxide Fuel in Pb-Bi-Cooled Fast Reactors

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1. Introduction

Instead of the sodium, the Pb-Bi is considered as another option for the coolant of the fast reactors because the Pb-Bi has some merits in the neutronics aspects. Since the atomic weight of the Pb and Bi is about 9 times heavier than that of Na, the Pb-Bi-cooled fast reactor (LFR) has a harder neutron spectrum than that of Na-cooled reactor (SFR). Consequently, the breeding ratio is expected to be enhanced.

In this study, the reactor core analysis for the equilibrium fuel cycle of the dry reprocessed oxide fuel (DRF) in a Pb-Bi-cooled fast reactor was performed for two selected reference cores: the Hybrid BN-600 benchmark core with a enlarged lattice pitch (Case 1) and the modified BN-600 core (Case 2). The core calculation was performed by the REBUS-3 [Ref. 1] code and the reactor characteristics such as the transuranic (TRU) enrichment, breeding ratio, peak linear power, burnup reactivity swing, etc. were calculated for the equilibrium core under a fixed management scheme.

2. Physics Calculation

2.1. Physics analysis model

In order to assess the applicability of the DRF to the LFR, two core models are considered as the reference core. The first one is based on the BN-600 Hybrid core which is used as the benchmark core of the Co-ordinate research Project (CRP) [Ref. 2] and the second one (Case 2) is based on the original BN-600 core to consider the real operation conditions. For both cases, the reactor core is composed of two regions without the blanket region [3].

The core calculation is performed by the REBUS-3 code using the KAFAX-F22 library [4], which is a neutron 80-group and gamma 24-group cross-section library based on JEF-2.2. The TRANSX [Ref. 5] and TWODANT [Ref. 6] codes are used to generate the 9-group region-wise effective macroscopic cross-sections. The fission products not included in the burnup chain are represented by lumped fission products (LFP) and the cross-sections for the LFP are generated from the KAFAX-E6FP cross-section library [7].

In this study, only the neutronics calculation was performed. The external fuel cycle strategies for the equilibrium core are as follows: i) 95% of the rare-earth and all other fission products are removed, ii) all uranium isotopes and 99.9% of TRU are recovered, and iii) all surplus fuel material after reprocessing process are sold. The isotropic compositions of the external feed to achieve the equilibrium cycle are composed of the TRU recovered from the typical light water reactor spent fuel and the depleted uranium. The recycling equilibrium mode calculation was performed to establish a self-sustaining breakeven core without an external source of fissile material, aiming at the breeding ratio of 1.05.

2.2. Sensitivity calculation results

2.2.1. Fuel volume fraction

The breeding ratio is a measure of self-sustaining of the reactor without the fissile material supply. Because the breeding ratio strongly depends on the neutron spectrum of the core, appropriate volume fractions of the fuel, cladding and coolant of the fuel assembly were searched for the selected cores. The breeding ratio is maintained over 1.05 when the fuel volume fractions are over 42% and 44% for Case 1 and Case 2, respectively. Compared to the sodium-cooled reactor, the fuel volume fractions are reduced by 5% and 4% for Case 1 and Case 2, respectively.

2.2.2. TRU content

In order to reduce the peak linear power density and flatten the power distribution of the core, the equilibrium core calculations were performed for various TRU enrichment distributions in the inner- and outer-core at the beginning of the equilibrium cycle (BOEC). Because the breeding ratio has a decreasing tendency as the difference of the TRU enrichment of the inner- and outer-core increases, the reactor characteristics calculations for the TRU enrichment were performed for 47% and 51% of the fuel volume fraction for Case 1 and Case 2, respectively.

When the TRU enrichments of the inner- and outer-core are the same, the power peaking occurs at the core center and moves to the outer core region as the out-core enrichment increases. In order to achieve the target breeding ratio of 1.05, the TRU enrichments of the inner-/outer-core should be 10.3%/15.8% and 9.2%/17.9% for Case 1 and Case 2, respectively. The average linear power densities for Case 1 and Case 2 are 114.4 W/cm and 142.6 W/cm, respectively, and the peak linear power densities for the selected fuel volume fractions are 194.8 W/cm and 241.7 W/cm for Case 1 and Case 2, respectively.

2.2.3. Reactivity coefficient

The DIF3D [Ref. 8] code is used for the core reactivity calculations for various TRU enrichments of the inner-core and outer-core. The isotopic compositions of the BOEC and the end of equilibrium cycle (EOEC) states are obtained from the equilibrium cycle calculation of the REBUS-3 code for each TRU enrichment. Then the fuel temperature coefficients and the coolant void reactivity at the BOEC and EOEC are calculated by the DIF3D using the finite difference method in the triangular-z node.

The coolant void reactivity generally shows a decreasing tendency as the difference of the TRU enrichment between the inner- and outer-core increases. At the same time, the coolant void reactivity decreases as the fuel volume fraction increases because the fraction of the coolant relatively decreases in the core. In addition, the void reactivity of the EOEC core is larger than that of the BOEC core due to the plutonium buildup through the fuel irradiation. The void reactivity of Case 1 and Case 2 are 1239 pcm and 1272 pcm for the BOEC and 1487 pcm and 1514 pcm for the EOEC, respectively. Compared to the sodium-cooled reactor, the Pb-Bi coolant void reactivities are reduced to about a half for both cases.

In contrast with the void reactivity, the fuel temperature coefficient in general increases as the difference of the TRU enrichment between the inner- and outer-core. At the same time, the fuel temperature coefficient of the EOEC core is lower than that of the BOEC core, because the fissile plutonium isotopes are significantly built up and U-238 transmutes as the fuel is irradiated. For the TRU enrichment of the inner-/outer-core achieving the breeding ratio of 1.05, the fuel temperature coefficient ranges from -0.70 pcm/K to -0.74 pcm/K for both the Case 1 and Case 2. Generally the fuel temperature coefficients of Pb-Bi cooled reactor are increased than those of the sodium-cooled reactor.

2.2.3. Fuel mass flow and inventory

The fuel mass flow for each step of the external fuel cycle is calculated to estimate the amount of fuels required and/or removed in each external reprocessing step when achieving the equilibrium core. The fissile plutonium gains during one cycle of the equilibrium core are 34.0 kg and 31.7 kg for the Case 1 and Case 2, respectively, which satisfy the self-sustaining breakeven core without an excess fissile material. The amount of minor actinides is slightly reduced during one cycle in the core, and built up a little during the external reprocessing process by the decay. The total amount of minor actinides to be sold is 1.6 kg and 1.4 kg per cycle for the fuel volume fraction of 49% and 51%, respectively. The increase in the amount of fission products is 700 kg per cycle; 5% of the rare-earth fission products are recovered and all other fission products are removed during the reprocess. The amount of depleted uranium to be supplied through the external feed is about 750 kg per cycle.

3. Conclusion

In this study, two kinds of Pb-Bi-cooled reactors were analyzed for the fuel volume fraction and TRU enrichment, without considering the detailed design of the fuel assembly and fuel channel. Generally, the reactor characteristics of the Pb-Bi-cooled core were better that those of the sodium-cooled core. However, because the specific gravity of the Pb-Bi coolant is 10 times higher than that of the sodium, it is required the higher capacity of the coolant pump for the Pb-Bi-cooled reactor. In addition, the specific heat and the thermal conductivity of Pb-Bi are lower than those of sodium. Nevertheless, if the design criteria used in this study are proved to be acceptable through the detailed physics design and thermal hydraulic analysis in the future, it is practically possible to construct an equilibrium fuel cycle of the LFR based on the oxide fuel utilizing the dry reprocessing technology.

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