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ARTICLE

Coolant void coefficient in a sodium-cooled rotational fuel-shuffling breed-and-burn fast reactor

Tsendsuren Amarjargal, Jun Nishiyama and Toru Obara *

Laboratory for Zero-Carbon Energy, Institute of Innovative Research, Tokyo Institute of Technology
N1-19 2-12-1 Ookayama, Meguro-ku, Tokyo 152-8550 Japan

In this study, the sodium void reactivity coefficient in a small rotational fuel-shuffling breed-and-burn fast reactors with nitride fuel and sodium coolant (RFBB-NS) was investigated. As it is possible that voids may be created if the coolant boils in hotter portions of the core, the sodium void coefficient is a crucial safety parameter of sodium-cooled fast reactors and must be monitored with the utmost attention. Sodium void coefficients were calculated in various coolant void situations at the beginning of the equilibrium cycle (BOEC) of the reference core and at the end (EOEC). Calculations were performed using Serpent continuous-energy Monte Carlo code 2.1.0. The results of this study indicate that the sodium void coefficient was close to that obtained with the current design of sodium-cooled fast reactors.

***Keywords:** RFBB; breed-and-burn reactor; rotational fuel shuffling; sodium-cooled fast reactor; sodium void coefficient; nitride fuel*

1. Introduction

Breed-and-Burn (B&B) reactor can be fueled with natural uranium or depleted uranium only, once initial criticality is established. It is based on breeding fissile material and fission of it in situ in a once-through fuel cycle. However, to obtain the B&B mode of operation, the neutron economy in the core should be enhanced. The reactor in this study could produce 180MWe which puts the reactor into a small modular reactor category. According to International Atomic Energy Agency (IAEA), small modular reactors are defined as advanced reactors that produce electricity of up to 300 MWe per module. However, as compared with conventional large reactors, neutron leakage is higher in small reactors which is a negative contributor to the neutron economy. Furthermore, the B&B core is dominated by neutron absorption materials, including fertile materials and fission products.

To achieve criticality and better neutron economy in the small B&B core, Rotational Fuel shuffling [1] was proposed in which fresh fuel is loaded from the edge of the core and then moved forward stepwise to the center of the core and discharged there. By doing so, high-reactivity fuels are kept continuously in high neutron-importance regions during reactor operation. Detailed analyses of small rotational fuel-shuffling breed-and-burn fast reactors (RFBB) have been studied [2,3].

The previous study investigated the feasibility of the RFBB-NS [4]. It was shown the concept is feasible. In addition to that excellent heat transfer characteristics of sodium make the fuel pin pitch small, which can make the neutron economy better by small neutron leakage. The better neutron economy with small neutron leakage makes the coolant void reactivity large in positive. Therefore, the sodium void coefficient needs to be confirmed that it is comparable with conventional SFRs. The purpose of the study was to clarify that the sodium coolant void coefficient is acceptable in a small RFBB-NS.

2. Reference core

Neutronic and heat removal analyses have been performed on the core concept of a Small RFBB-NS [4] with a power of 450MWt and 168 natural uranium fuel assemblies separated into 6 symmetry regions. The nitride fuel smear density was set at 83% of the theoretical density according to a previous study on fuel integrity during high burnup [5]. The fuel cladding material was oxide dispersion-strengthened (ODS) ferritic steel. The fuel pin pitch was set at 0.9434 cm which is smaller than conventional SFRs where fuel pitch is usually over 1.0cm for the nitride fuel with a helium bond. Small pin pitch results in a small active radius of core 106.5 cm and core height 220.0 cm. Detailed parameters are provided in **Table 1**.

After constructing the core, burnup analysis and fuel shuffling were performed to evaluate the fuel cycle length of the shuffling in small RFBB-NS. Shuffling was repeated in each symmetry region in the same shuffling pattern until burnup reached equilibrium. An equilibrium state was obtained after 56 shuffling steps. At this

*Corresponding author. E-mail: tobara@zc.iir.titech.ac.jp

Table 1. Core specification.

Parameters	Values
Thermal power (MW)	450
Core Height (cm)	220
Core equivalent radius (cm)	106.5
Total number of assemblies	168 + 1(coolant channel)
Number of fuel pins in an assembly	271
Fuel assembly pitch (cm)	15.8
Fuel material	N-15 isotope 99% enriched UN (UN99)
Coolant material	Sodium
Cladding material	ODS
Fuel rod radius (mm)	0.405
Cladding thickness (mm)	0.55
Fuel pin pitch (cm)	0.9434
Pellet density	90% TD
Smeared density	83% TD
Bond material	Helium
Fuel volume fraction	66.85%
Cladding volume fraction	19.39%
Coolant volume fraction	13.76%
Reflector material	Sodium
Shuffling pattern	Rotational
Fuel cycle length and steps	860 days and 56 steps*
Average fuel temperature (K)	800
Average coolant temperature (K)	700
Coolant speed (m/s)	10.0

*Steps mean a number of shuffling. In this case, shuffling is repeated with an interval of 860 days burnup.

equilibrium state, the average effective multiplication factor was 1.0100, the average discharge burnup was 187 MWd/kg-HM, and the corresponding cladding material damage was 656 DPA. The change in power distribution between the beginning of the fuel cycle in the equilibrium condition (BOEC) and the end of the fuel cycle (EOEC) was small. A steady-state heat removal analysis was performed on the hottest channel of the core. It was found that the maximum fuel centerline temperature was lower than the nitride fuel operational limit temperature. However, it is important to understand the stability of the system for safe operation. Reactivity coefficients are the crucial safety parameter. As sodium is used as a coolant in the reference core analyses, there is the possibility of the coolant boiling. Thus, the coolant void coefficient of reactivity must be monitored.

3. Calculation conditions

The sodium void coefficients were calculated in various coolant void situations at the BOEC and EOEC. **Figure 1** shows the ID numbers of the fuel assemblies in a one-sixth segment of the core and the zone ID numbers for the fuel axial regions. The following voiding scenarios were analyzed:

- Scenario 1: A small portion of the core center was voided when assemblies 26, 27, and 28 in Zone 6 were voided;
- Scenario 2: The middle part of the core was voided when assemblies 26, 27, and 28 in Zones 5 to 7 were voided;
- Scenario 3: A small portion of the core center was voided when assemblies 23 to 28 in Zone 6 were voided;
- Scenario 4: The middle part of the core was voided when assemblies 23 to 28 in Zones 5 to 7 were voided;
- Scenario 5: The whole core was voided and all fuel assemblies in all zones were voided.

The calculation tool used in this study was Serpent 2.1.0 [6], a continuous-energy Monte Carlo Reactor Physics Burnup calculation code with the ENDF/B-VII.0 [7] cross-section library. Calculations were performed with 50,000 neutron histories in 200 batches, excluding the first 50 batches to obtain results within the statistical error.

4. Results and discussions

A summary of the present findings is shown in **Table 2**. First, the sodium void reactivity coefficient is estimated to be 0.88$ at BOEC and 1.04$ at EOEC in Scenario 1 where only Zone 6 was voided in assemblies 26 to 28 while the coefficient was increased by almost two times when the voiding areas were broadened in axial regions in Scenario 2. In the case of Scenario 3, the estimated reactivity coefficient

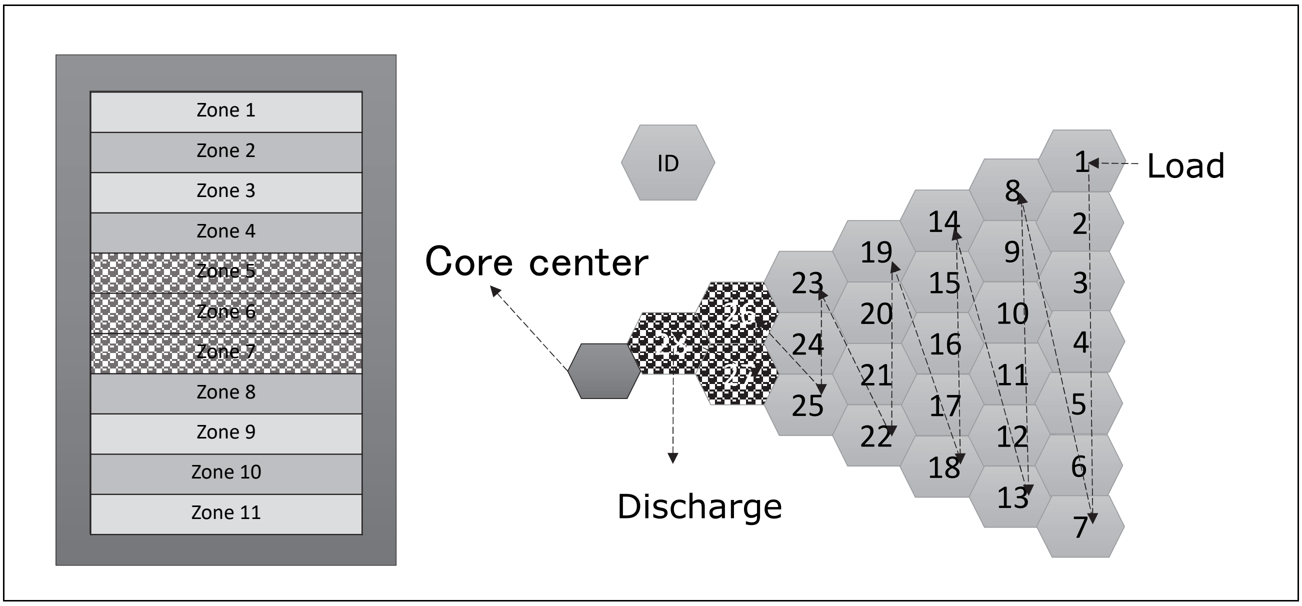

Figure 1. (a) core axial view, (b) one-sixth of core.

Table 2. Sodium void reactivity coefficients in various situations for reference core.

Void scenario	Sodium void reactivity coefficient ($) at BOEC	Sodium void reactivity coefficient ($) at EOEC
Scenario 1	0.88	1.04
Scenario 2	1.60	1.66
Scenario 3	1.15	0.92
Scenario 4	2.12	2.16
Scenario 5	3.96	4.08

was 1.15$ at BOEC and 0.92$ at EOEC, whereas those values were doubled in Scenario 4, in which a number of the voided assemblies were identical to the voided zones in Scenario 3. Finally, when all fuel assemblies in all axial zones were voided, the estimated reactivity was 3.96$ at BOEC and 4.08$ at EOEC. Delayed neutron fractions for all scenarios were ~0.0036.

In conventional SFR designs, the sodium void coefficient is usually around 4-6$ [8,9]. In the case of RFBB-NS, the estimated reactivity was under 2$ when the middle portion of the core center was voided, while when the whole core was voided in case of a severe accident, the estimated reactivity was about 4$ at both BOEC and EOEC. RFBBs with sodium coolant were designed to achieve maximum neutron economy by reducing neutron leakage. The present results suggest that, even though small RFBBs are designed to maximize neutron economy, the positive sodium void coefficient cannot be significantly large in a small RFBB with sodium coolant compared to conventional SFRs.

5. Conclusions

The sodium void reactivity coefficient was investigated in small RFBB-NSs and five different voiding scenarios were investigated at BOEC and EOEC. When a small portion of the core center was voided, the reactivity coefficient was around 1$ at both BOEC and EOEC, while when the middle portion of the core center was voided, it was around 2$. Finally, when the whole core was voided, the estimated reactivity was around 4$ at both BOEC and EOEC. It was thus confirmed that the sodium coolant void coefficient in RFBBs with nitride fuel and sodium coolant was the same as that in conventional SFR designs in the equilibrium condition. This suggests that the sodium void coefficient cannot be a serious issue from the point of view of safety in small RFBBs with sodium coolant and nitride fuel.

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