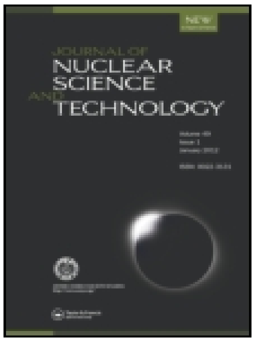

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ARTICLE

Neutronics studies on the feasibility of developing fast breeder reactor with flexible breeding ratio

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This paper investigates the feasibility of designing a flexible fast breeder reactor from the view of neutronics. It requires that the variable breeding ratio can be achieved in operating a fast reactor without significant changes of the core, including the minimum change of fuel assembly design, the minimum change of the core configuration and the same control system arrangement in the core. The sodium cooled fast reactor is investigated. Two difficulties are overcome: (1) the different excess reactivity is well controlled for different cores, especially for the one with small breeding ratio; (2) the maximum linear power density is well controlled while the breeding ratio changes. The optimizations are done to meet the requirements. The U–Pu–Zr alloy is applied to enhance the breeding. The enrichment-zoning technique with unfixed blanket assembly loading position is searched to get acceptable power distributions when the breeding ratio changes. And the control system is designed redundantly to fulfill the control needs. Then, the achieved breeding ratio can be adjusted from 1.1 to 1.4. The reactivity coefficients, temperature distributions and preliminary safety performances are evaluated to investigate the feasibility of the new concept. All the results show that it is feasible to develop the fast reactor with flexible breeding ratios, although it still highly relies on the advancement of the coolant flow control technology.

Keywords: fast reactor; fuel breeding; flexible breeding ratio

1. Introduction

The fast reactor can produce more fissile fuel than it consumes. It is very important to realize the sustainable developments of nuclear energy. Although the development of fast breeder reactors was slowed down after 1980s [1], some countries like India and Russia maintained the research, development and demonstration to face the potential uranium shortages. At present, China decides to speed up the nuclear power development. Aiming for the sustainable and quick increase of the nuclear energy supply, the basic strategy of PWR–FBR matched development with closed fuel cycle is adopted [2].

In order to address the problem of possible fuel shortage following the rapid nuclear energy development, the fast reactors mainly targeted at breeding are required and are under investigation. Nevertheless, the development of fast reactor is influenced by the fuel supply and nuclear waste disposal. Future demand for the breeder reactor is still inconclusive. Furthermore, the breeder reactor is more costly to construct and operate than the existing PWR. Therefore, the changes of nuclear energy developments should be taken into account when designing a breeder reactor.

Similar concept has been proposed in the United States, although it was indicated in designing a fast burner reactor. The concept of advanced burner reactor over a wide range of conversion ratio was put forward in 2006 [3]. This concept indicated that it is feasible to design a reactor with flexible conversion ratio just by employing diverse assembly design and arrangement, without changing the vessel and the core structure. Compared to the traditional fast reactors which are always designed at a fixed conversion ratio, it can effectively respond to the dynamically changing needs and priorities of the nuclear industry as well as those of the society [4].

Similarly, to make the fast breeder reactor design ahead of the industry requirements, this paper investigates the feasibility of designing the fast reactor with flexible breeding ratio (BR). It aims to show that the wide range of BR can also be achieved, with the least change in a designed reactor core. Therefore, a new fast reactor design is proposed with the name of flexible fast breeder reactors (FFBR). It is a 1500 MWt, pool-type, sodium-cooled fast breeder reactor.

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The metallic fuel (U–Pu–Zr alloy) is used to bring better breeding performance. The same fuel assemblies are used. This feature makes it possible to change BR merely by refueling scheme design whereby it greatly simplifies the fuel manufacture. Special treatments are carried out to flatten the power distribution to maintain the maximum linear power density in reasonable range while the BR changes. With the same number of control assemblies and the fixed location, the optimization of the control system is conducted to ensure enough reactivity worth to accommodate the different excess reactivity for each BR.

The reactivity feedback coefficients are calculated and evaluated in the equilibrium cycle. Preliminary safety analysis is taken based on the quasi-static safety analysis method to evaluate the inherent safety. The fuel and cladding temperatures and inherent safety-related parameters are given to show the change for the varying BR. Based on the results, it is shown that within a wide range of BR, the new concept of designing the FFBR is feasible.

2. Calculation method and parameter definitions

2.1. *Breeding performance parameters*

The BR and doubling time (DT) are the main parameters for evaluating the breeding performance of fast breeder reactors. However, several different methods have been used to calculate these two quantities [5]. Obviously, the consequences of diverse defined method will vary greatly.

The BR used in this paper is defined as

$$\mathrm{BR} = \frac{\mathrm{RR}_c}{\mathrm{RR}_a}, \tag{1}$$

where RR_c is the capture reaction rate of fertile isotopes, RR_a is the absorption reaction rate of fissile isotopes.

DT is given by

$$\mathrm{DT} = \frac{M_0}{M_g}, \tag{2}$$

where M_0 is the initial fissile inventory in the core, M_g is the net increase of fissile material per year.

2.2. *Coupled neutronics and thermal–hydraulic calculation*

A home-developed code package SARAX1.0 (System for Advanced Reactor Analysis at XJTU) is used in the calculations in this paper. The calculation flow scheme is given in **Figure 1**. Based on the JENDL-3.3 data library and the collision probability method, the 16-group cross-section set is generated for the 3D whole core calculation from 107-group cross-section library by assembly calculation.

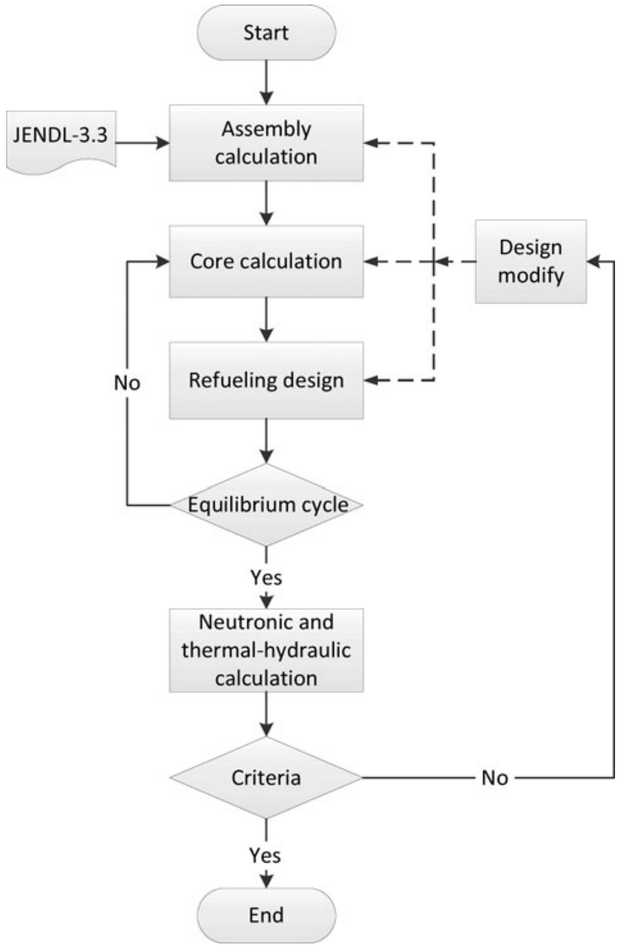

Figure 1. Computational flow chart in the calculations.

The single- and multi-assembly models are used to generate the homogenized cross section. For instance, the fuel assemblies are elaborated as what the actual assembly is. The heterogeneous model of the control assembly is illustrated in **Figure 2(a)**, for which the multi-assembly calculation is performed. The equivalent concentric ring geometry of reflector and shield model is illustrated in Figure 2(b) to consider the spatial dependence, in which only the homogeneous structure of reflector and shield assemblies is considered for simplification. The spatial homogenization for all types of assemblies including control rods has been done by the volume-flux weighting. The resonance calculations are conducted with the fuel pin to generate the effective shielding cross section first. The narrow resonance approximation is applied for the resonance calculation in slowing down process while the hyper-fine method is used for the resonance calculation in low energy. The collision possibility method is used to calculate the spectrum of the assemblies. Based on the spectrum, the fine energy group is collapsed into 16 groups for the homogenized assemblies.

A multi-dimensional diffusion code is involved in SARAX1.0 for whole core calculation in hexagonal-z geometry. The equilibrium cycle analyses are performed.

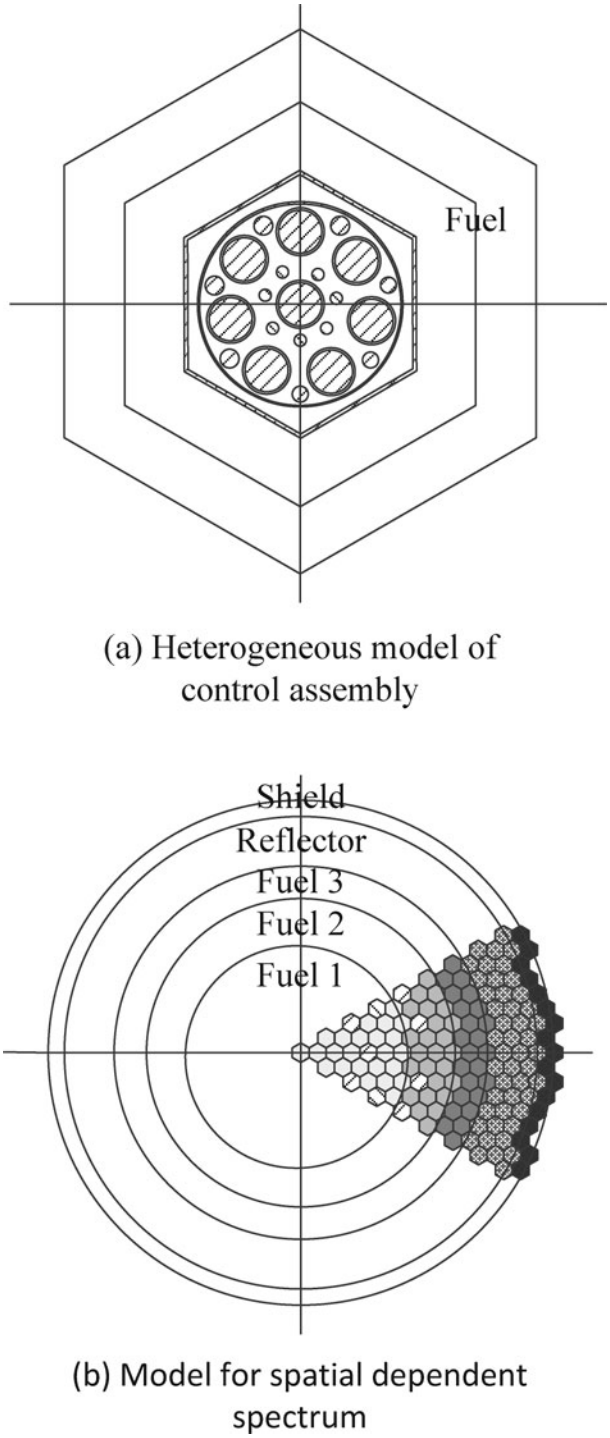

Figure 2. Calculation models for different assemblies.

The model used here assumes no fuel shuffling in the refueling process.

The temperature distribution is calculated by simplified thermal–hydraulic calculation based on the single channel heat transfer model. The coolant inlet temperature is 350 °C. The flow rate is adjusted such that the outlet temperature is 510 °C. The solidus temperature of U–Pu–Zr alloy is about 1100 °C and the fuel cladding eutectic temperature is limited to 650 °C [3,6]. The maximum linear power is limited to guarantee enough safety margins for the fuel centerline temperature and the cladding inner wall temperature.

For the U–Pu–Zr ternary metallic fuel, the thermal conductivity for fuel temperature calculation is given as follows [7]:

$$k_{(T)} = a + bT + cT^2 \tag{3}$$

where

$$a = 17.5\left(\frac{1 - 2.23\,W_{\mathrm{Zr}}}{1 + 1.61\,W_{\mathrm{Zr}}} - 2.62\,W_{\mathrm{Pu}}\right) \tag{4}$$

$$b = 1.154 \times 10^{-2}\left(\frac{1 + 0.061\,W_{\mathrm{Zr}}}{1 + 1.61\,W_{\mathrm{Zr}}} + 0.9\,W_{\mathrm{Pu}}\right) \tag{5}$$

$$c = 9.38 \times 10^{-6}\,(1 - 2.7\,W_{\mathrm{Pu}})\,, \tag{6}$$

where W_{Zr}, W_{Pu} are the weighting fractions of Zr and Pu, respectively.

With burnup, the increasing of pores results in the reduction of thermal conductivity till the porosity of 0.23 for the metallic fuel. Therefore, the porosity correction factor is applied to consider the irradiation effect of metallic fuel. The correction factor is given by Equation (7) considering the biggest value with burnup.

$$X = (1 - P)^{1.5\varepsilon} \tag{7}$$

where, P is the porosity.

Here, P is set to be 0.23 and ε is 1.75 for the gas-filled pores [7]. The most conservative value of thermal conductivity is used to calculate the maximum fuel temperature, for which the corrected thermal conductivity $Xk_{(T)}$ equals to $0.5k_{(T)}$.

2.3. Reactivity coefficients calculation

Using SARAX1.0, the Doppler coefficient, radial and axial expansion coefficients, coolant density coefficient and void worth are calculated to evaluate the reactivity effects.

It has been found that, the Doppler coefficient, dk/dT varies closely to $1/T^m$, where the exponent m is spectrum-dependent. Therefore, the Doppler coefficient is calculated by performing a set of core calculations at different fuel temperatures. Then, the coefficients are obtained from the derivative of fitted function of the fuel temperature.

The axial expansion coefficient is the main prompt negative feedback available in a metal-fueled fast reactor. To calculate the axial expansion effect, all dimensions and concentrations through the core are kept fixed except that the fuel pins are uniformly elongated by 1% in the axial dimension [8]. Correspondingly, the fuel density is lowered by approximately 1% to keep the fuel mass constant.

The core radial expansion coefficient strongly depends on the core design and is typically negative. It is concerned with the increase of core radius due to the grid expansion. Thus, the radial expansion coefficient is calculated for a 1% uniform radial expansion of the subassembly pitch [9].

The control rod driveline expansion reactivity coefficient is calculated by assuming the control rod driveline heated by the outlet coolant and the control assemblies are inserted into the core.

The coolant temperature coefficient is defined as the difference of reactivity in response to the coolant density changing with the coolant temperature increasing by 1K.

The coolant void worth is also evaluated to consider the reactivity effect after the boiling out of the coolant. It is calculated by voiding the coolant in the active core and its above regions.

2.4. *Quasi-static approach to safety*

The quasi-static safety analysis method, proposed by Argonne National Laboratory, is applied to single out the cases that obviously do not satisfy the safety criteria [10]. Equations (8)–(10) are used to evaluate whether the reactor is inherent safety or not.

$$\frac{A}{B} \le 1 \tag{8}$$

$$1 \le \frac{C\Delta T_c}{B} \le 2 \tag{9}$$

$$\frac{\Delta\rho_{TOP}}{|B|} \le 1, \tag{10}$$

where A is the net power reactivity coefficient, B is the power/flow reactivity coefficient, C is the inlet temperature reactivity coefficient, ΔT_c is the coolant temperature rise at nominal full power, and $\Delta\rho_{TOP}$ is the reactivity introduced in the control rod withdrawal accident.

The coefficients A, B and C are defined as follows:

$$A = (\alpha_D + \alpha_e)\,\Delta T_f \tag{11}$$

$$B = \left\{\alpha_D + \alpha_e + \alpha_{Co} + 2\left(\frac{\alpha_{RD} + 2}{3\alpha_R}\right)\right\}\frac{\Delta T_c}{2} \tag{12}$$

$$C = \alpha_D + \alpha_e + \alpha_{Co} + \alpha_R, \tag{13}$$

where α_D is the Doppler coefficient, α_e is the axial expansion coefficient, α_{Co} is the coolant temperature coefficient, α_{RD} is the control rod driveline expansion coefficient, α_R is the radial expansion coefficient, and ΔT_f is the temperature rise of fuel relative to the coolant.

3. Core design

3.1. *Fuel selection*

It is known that U-238 could capture neutrons and thereby convert into Pu-239. In a fast reactor, the U–Pu fuel has better potential for superior breeding performance than other combinations. Another reason for giving preference to the U–Pu fuel in fast reactors is that it employs the same fuel cycle as the present commercial thermal reactors.

Among several types of U–Pu fuels available, the mixed oxide fuel is the most widely utilized one. However, the thermal conductivity and fissile atom density of the oxide fuel are relatively low, and the presence of oxygen atoms results in softened spectrum, which reduces the BR. The carbides and nitrides fuel have excellent properties, but failed to reach the practical stage because of lacking tests. Compared to the fuels mentioned above, the metallic fuel has higher fissile atomic density, more superior thermal conductivity and harder neutron spectrum. Therefore, the U–Pu–Zr alloy with 10% Zr is chosen as the candidate fuel to enlarge the range of flexible BR. The plutonium isotopic composition is 3.0%, 48.1%, 25.7%, 15.4% and 7.8% for Pu-238, Pu-239, Pu-240, Pu-241 and Pu-242, respectively, which comes from the reprocessed PWR spent fuel at 50 Gwd/tU [11]. Considering the fuel swelling, the smear density of 75% theoretical density is used [4].

3.2. *Assembly design*

The fuel assembly design is depicted in **Figure 3**. The pin diameter of 6.9 mm and P/D of 1.15 are adopted. Each assembly contains 127 pins loading U–Pu–Zr alloy fuel. The bond material is sodium. The axial blankets are located at both ends of the active core to further enhance the breeding, and the gas plenum is located on the top of upper blanket to accommodate the fission gas. The active length of fuel pin is 105 cm.

The blanket assembly has similar structure with the fuel assembly, but with the pin diameter of 8.9 mm and P/D of 1.09. The PWR spent fuel at 50 GWd/tU is used as the blanket fuel [11].

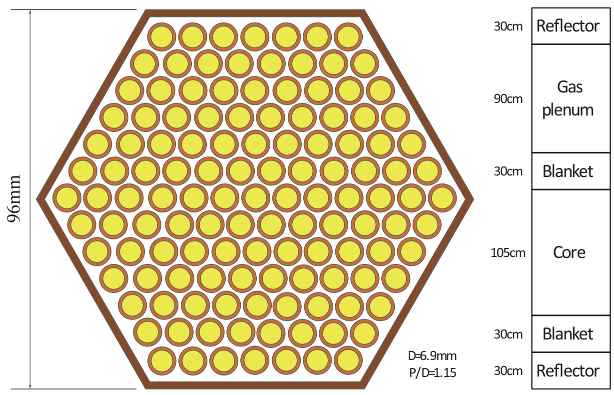

Figure 3. Structure of the fuel assembly.

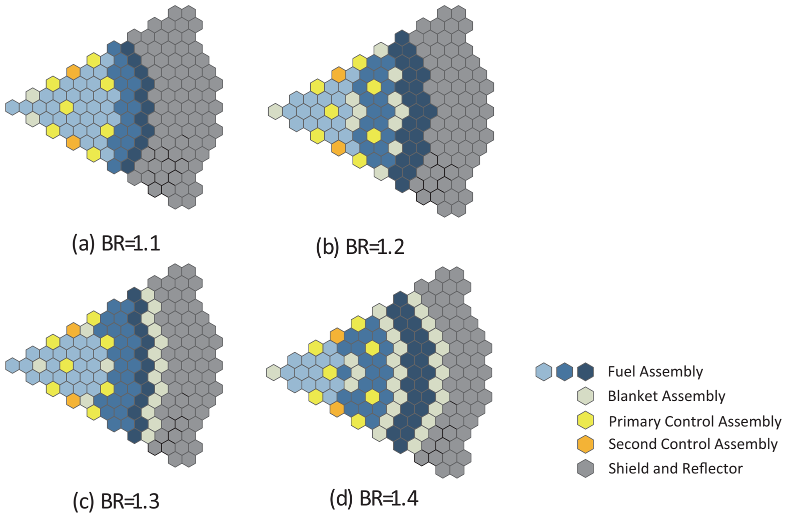

Figure 4. Layout of reactor cores with different BR.

3.3. *Core design for flexible BR*

Two core design choices exist in terms of the location of the blanket assemblies. In the homogeneous core, the relatively uniform or homogeneous mixture of the fertile and fissile fuel is spread throughout the core. While in the heterogenous core, the blanket assemblies containing pure fertile material are distributed throughout the core region. Since it is desirable to have a wide range of BR, the heterogenous core with higher breeding capability is the better choice. In addition, the in-core arrangement of blanket assemblies can be used to reduce the local peak power factors for the core with small BR.

The enrichment zoning strategy is employed to flatten the power distribution. The weight fraction of Pu is 12.5 w/o, 15.8 w/o and 21.4 w/o for the inner, middle and outer enrichment zone, respectively. Even for the core with significantly different BR, the strategy is kept to minimize the change of the core. The number of assemblies and their arrangement are adjusted to make the maximum linear power density be acceptable and ensure the excess reactivity being within the control range of the primary control system.

The blankets are arranged both on the axial and radial regions. In this way, the coolant void reactivity is reduced. In addition, the inner blankets play an important role in flattening the radial power distribution. Since the fuel enrichment zoning is optimized based on the BR of 1.1, the arrangement of blanket assemblies is an effective approach to flatten the power distribution, especially for the core with higher BR. It is different from the conventional fast reactor design, in which the arrangement of blanket assemblies is fixed.

Figure 4 shows the layout of the core with different BR. This series of design contain four schemes, with the corresponding BR of 1.1, 1.2, 1.3 and 1.4, respectively. Combining the periphery shield and reflector assemblies, all cores have the same geometry shape and size, which are the basic premises of the core design with flexible BR. The number and locations of control assemblies are also shown in Figure 4. The control system is optimized to provide enough control worth during the BR transition. In this concept, the coolant flow control is necessary when the BR is changing, since a part of fuel assemblies will be replaced by the blanket assemblies or the opposite. It needs to apply the combination of fixed and variable orifices [12]. The main contribution for flow control comes from the fixed orifices, which are designed to match the different assembly power. The variable orifices, which are not popularly applied in current reactors but have been proposed in some advanced concepts [12], are referred and used in this design to supplement the required small flow adjustment, especially in the process of BR transition.

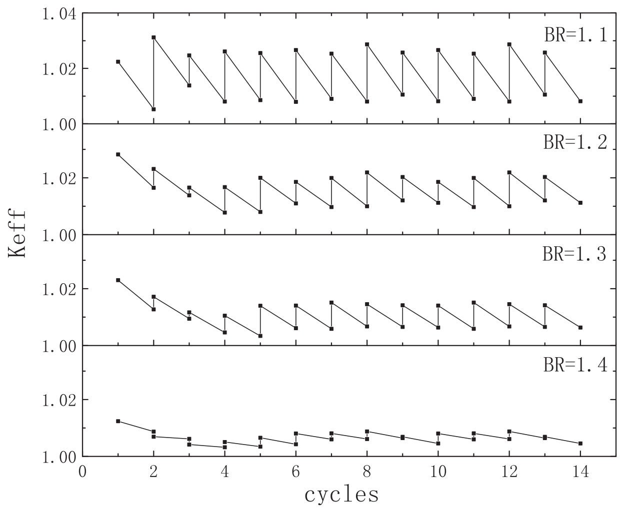

Figure 5. Multiplication factors in each cycle for different BR.

Figure 5 shows the multiplication factors in each cycle for different BR. The maximum multiplication factor decreases with the increasing BR. Though the number

Table 1. The assembly number and core performance for different BR.

BR	1.1	1.2	1.3	1.4
Core power (MWt)	1500	1500	1500	1500
Cycle length (EFPD)	260	280	260	240
Fuel exchange batch	4	4	4	4
Calculated BR	1.12	1.21	1.34	1.41
DT (y)	27.3	19.3	13.7	10.7
Fuel				
Low enrichment	181	102	127	72
Medium enrichment	90	108	138	126
High enrichment	66	138	72	138
Total	337	348	337	336
Blanket	6	55	96	157
Primary control assemblies	30	30	30	30
Second control assemblies	6	6	6	6
Active core height (cm)	105	105	105	105
Fuel pin diameter (cm)	0.69	0.69	0.69	0.69
Fuel pin number per assembly	127	127	127	127
Smear density (%TD)	75	75	75	75
Plutonium enrichment (%)	12.5/15.8/21.4	12.5/15.8/21.4	12.5/15.8/21.4	12.5/15.8/21.4
Zr content (%)	10	10	10	10
Axial blanket height (cm)	$30+30$	$30+30$	$30+30$	$30+30$
Blanket pin diameter (cm)	0.89	0.89	0.89	0.89
Blanket pin number per assembly	91	91	91	91
Maximum linear power (kW/m)	49.2	49.0	48.0	49.5
Average discharge burnup (MWd/kg)	67.4	66.5	62.9	55.4
Peak fast neutron fluence ($10^{23} n/\mathrm{cm}^2$)	3.79	3.96	3.53	3.66

of fuel assemblies is close, more blanket assemblies located in the core make the excess reactivity small for large BR.

Table 1 summarizes the core design parameters for each BR, including the final BR and DT, of which, the capacity factor is not taken into account in the current investigation. Although the BR increases from 1.1 to 1.4, the number of fuel assemblies is kept similar. However, the number of blanket assemblies is increased significantly, from 6 for BR = 1.1 to 157 for BR = 1.4.

3.4. Control system evaluation

The control system consists of two independent subsystems. The primary control system is used to shut down the reactor from operating condition to the refueling temperature, and the secondary system is used to shut down the reactor from operating condition to the hot standby state [5]. In **Table 2**, the control requirements, available reactivity worth and shutdown margin for separated cores with different BR are listed. The contributions of the Doppler effect, coolant density change, axial and radial expansion are taken into account in cooling the core from full power to hot standby or refueling temperature. The maximum worth of single control assembly is used to quantify the reactivity fault introduced in some accident situation. The reactivity worth available is evaluated by assuming the most reactive control assembly stuck out. The excess reactivity is obtained from the maximum multiplication factor in the cycles. Besides, additional 3$ is added in evaluating the required control worth to consider the computational deviation

Table 2. The control requirements and available reactivity worth.

	1.1		1.2		1.3		1.4	
	Pri.	Sec.	Pri.	Sec.	Pri.	Sec.	Pri.	Sec.
Full power to hot standby ($)	0.71	0.71	0.70	0.70	0.67	0.67	0.63	0.63
Hot standby to refueling ($)	0.35	–	0.37	–	0.34	–	0.32	–
Reactivity fault ($)	0.78	0.78	0.65	0.65	0.75	0.75	0.99	0.99
Excess reactivity ($)	7.21	–	6.25	–	5.06	–	2.70	–
Uncertainties ($)	3		3		3		3	
Total ($)	12.05	1.49	10.98	1.35	9.81	1.42	7.64	1.62
Worth of one rod out ($)	24.12	9.20	21.77	7.90	22.44	7.32	19.30	6.72
Shutdown margin ($)	12.06	7.71	10.79	6.54	12.63	5.90	11.66	5.10

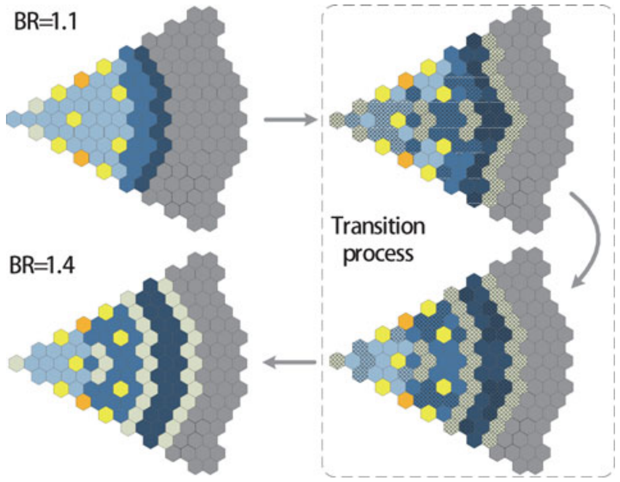

Figure 6. Refueling scheme for the BR transition.

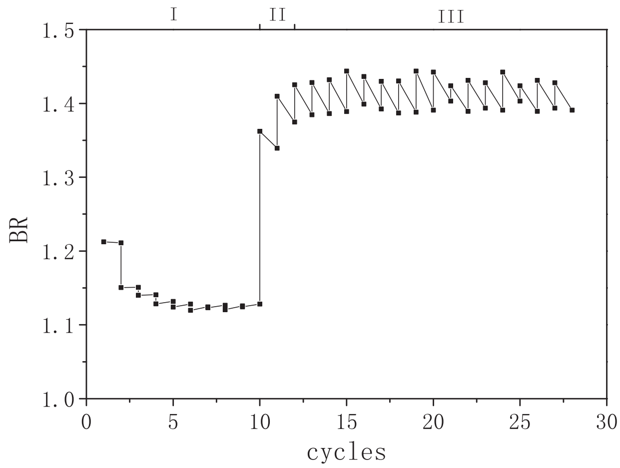

Figure 7. BR in each cycle in the transition process.

and uncertainties [5]. It can be seen that the worth of designed control system is sufficient for all the cores with different BR.

With the versatile fuel, more assemblies are needed to reduce the maximum linear power density, which results in the large excess reactivity. As shown in Table 2, the core with the smallest BR requires the most control worth to compensate the excess reactivity introduced by the increased fraction of fuel assemblies. Some methods are applied to reduce the multiplication factor at the beginning. For example, replacing the high-enriched fuels with the low-enriched ones in the initial layout, and replacing the fuel assemblies with the blanket assemblies in the core center.

3.5. *BR transition scheme design*

The design of FFBR aims to match the requirement of industry development. Therefore, the BR transition is necessary when the need changes. The feasibility research is performed by adjusting the core from the state of low BR ($BR = 1.1$) to the state of high BR ($BR = 1.4$).

Figure 6 shows the detail transition process, the hexagonal meshes with grids represent the new added assemblies. The BR transition is achieved by the refueling scheme design. A complete transition process involves three stages: (1) the initial stage is from starting up to the equilibrium cycle with BR equaling to 1.1; (2) The second stage is the transition stage; (3) the final stage is from the end of transition state to the equilibrium cycle with BR equaling to 1.4. The transition stage is completed through two cycles and each cycle is 240 effective full power days.

The BR of each cycle on the transition process is shown in **Figure 7**. The BR increases dramatically with the blanket assemblies added into the core. I, II, III stands for the three stages, respectively. **Figure 8** shows the multiplication factors in the transition process. **Figure 9** shows the maximum linear power density

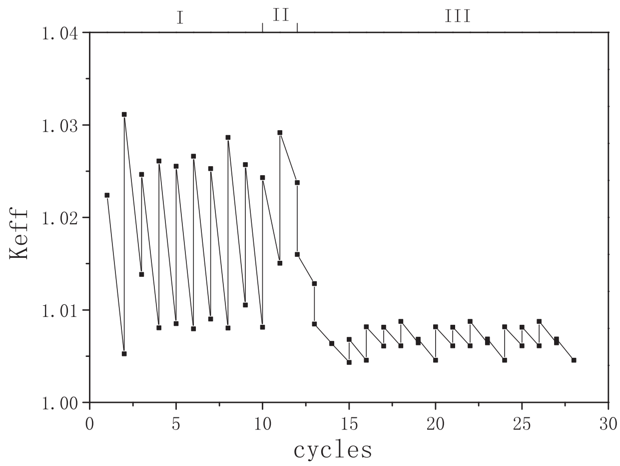

Figure 8. Multiplication factor in each cycle in the transition process.

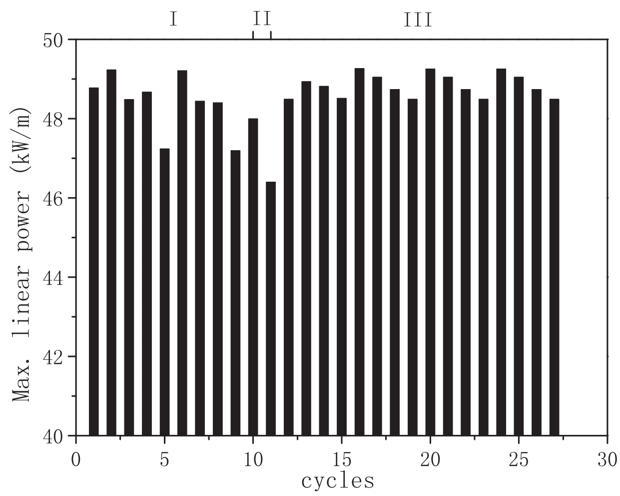

Figure 9. Maximum linear power density in each cycle in the transition process.

Table 3. The reactivity coefficients for different BR.

BR	1.1	1.2	1.3	1.4
Effective delayed neutron fraction (pcm)	417	426	429	436
Doppler coefficient (¢/K)	−0.098	−0.094	−0.091	−0.091
Sodium temperature coefficient (¢/K)	0.153	0.136	0.136	0.126
Axial expansion coefficient (¢/K)	−0.118	−0.110	−0.121	−0.107
Radial expansion coefficient (¢/K)	−0.231	−0.216	−0.225	−0.217
Driveline expansion coefficient (¢/K)	−0.043	−0.039	−0.040	−0.035
Sodium void worth ($)	6.25	5.50	5.50	5.05

Table 4. The decomposited Doppler coefficients for different regions.

Region	1.1	1.2	1.3	1.4
Active core (¢/K)	−0.0860	−0.0676	−0.0633	−0.0579
Axial blanket (¢/K)	−0.0096	−0.0068	−0.0060	−0.0060
Radial blanket (¢/K)	−0.0033	−0.0167	−0.0205	−0.0268

in each cycle. It can be seen that the maximum excess reactivity is under the control margin. Despite the power distributions and maximum linear power density vary with BR, the maximum fuel centerline temperature and maximum cladding inner wall temperature are all acceptable considering the thermal margin. The detailed data is given in Table 6 in Section 4.2.

4. Feasibility evaluation

The calculations in Section 3 have shown that the flexible BR is achievable for proper design of the reactor core. In this section, several key parameters are calculated to evaluate the change in core performance brought from the varying BR. The preliminary safety analysis is also performed to discuss the inherent safety of the new reactor.

4.1. *Reactivity coefficients*

The reactivity coefficients in the equilibrium cycle are summarized in **Table 3**. The effective delayed neutron fraction increases with the BR increasing. The increased blanket inventory is the primary reason, since U-238 is the key fission isotope determining the effective delayed neutron fraction.

The Doppler coefficients become less negative with the BR increasing. This is the result of the reduced concentration of U-238 in active core for the large BR. In **Table 4**, according to the decomposed Doppler coefficients, it can be found that the active core makes the primary contribution to the Doppler effect. The sodium temperature coefficient and the coolant void worth have the same trend. Both of them decrease with the BR increasing, since the volume fraction of coolant in active core becomes smaller when more blanket assemblies are loaded into the core. The core layout is the main factor affecting the axial expansion coefficient and radial expansion coefficient. The expansion coefficients for BR of 1.2 and 1.4 are less negative than others, since larger radius and more blanket assemblies interspersed among the fuel assemblies cause less leakage change when the geometry changes. The driveline expansion coefficients are directly related to the control assembly worth. Therefore, the coefficient gets the maximum value for the smallest BR.

However, by comparing all the coefficients, it can be seen that they are not affected much by the BR changing. Therefore, the varying of BR will not bring significant change of feedback and transient performance of the core.

4.2. *Temperature parameters*

The single channel heat transfer model is used to calculate the temperature distribution. The coolant inlet temperature is 350 °C and the flow rate is adjusted such that the outlet temperature is 510 °C. The hot channel factors [5] are used to consider the uncertainty in the temperature calculation. The values are presented in **Table 5**. **Table 6** shows the results, both the maximum cladding inner wall temperature and the maximum fuel

Table 5. The hot channel factors in different regions.

	Value
Coolant	1.26
Film	1.23 (2.10)*
Cladding	1.19
Gap	1.48
Fuel	1.10
Heat flux	1.13

*This factor affects the maximum cladding temperature only; it does not affect the maximum fuel temperature.

Table 6. The temperatures of reactor cores with different BR.

BR	1.1	1.2	1.3	1.4
Coolant inlet temperature (°C)	350	350	350	350
Coolant outlet temperature (°C)	510	510	510	510
Coolant temperature rise (°C)	160	160	160	160
Max. cladding inner wall temperature (°C)	556	556	555	556
Max. fuel centerline temperature (°C)	877	873	868	877
Temperature rise of fuel to coolant (°C)	132	122	124	120

Table 7. The reactivity parameters for quasi-static safety analysis.

BR	A (¢)	B (¢)	C (¢/K)	A/B	$C\Delta T_c/B$	$\Delta\rho_{TOP}/\lvert B\rvert$
1.1	−28.48	−36.63	−0.29	0.78	1.29	0.59
1.2	−24.72	−34.66	−0.28	0.71	1.31	0.54
1.3	−26.22	−36.48	−0.30	0.72	1.32	0.42
1.4	−23.70	−34.44	−0.29	0.69	1.34	0.23

centerline temperature are similar for all the BR. All the temperature is smaller than the temperature limit.

4.3. *Quasi-static safety analysis*

All the reactivity coefficients and temperature parameters needed for the quasi-static safety analysis have been presented in Tables 3 and 6, respectively.

The values of parameters A, B and C are given in **Table 7**. It can be observed that all the results are within the safety constraints defined in Equations (8)–(10) for different BR. It means that the reactivity feedback could bring the core to the shutdown condition at accidents. The cases considered here including the loss of flow without scram, the loss of heat sink without scram (LOHS) and the rod runout transient overpower event (TOP).

5. Conclusion

This paper investigates the feasibility of designing a new fast breeder reactor FFBR, which is a sodium-cooled fast reactor with flexible BR. Preliminary studies indicate that it can be realized by adjusting the number and fraction of fuel and blanket assemblies, along with changing the fuel management schemes, but without changing the core structure, fuel design and control system arrangements. The optimizations are performed on the number of assemblies, fuel enrichment and loading pattern to get the appropriate reactor core, overcoming two key difficulties: (1) the different excess reactivity swing can be accommodated by the existing control system; (2) the maximum linear power is fulfilled to satisfy the thermal limits.

The key parameters of the core are calculated to evaluate the change brought by the varying BR. The reactivity coefficients are not affected much by the BR changing. It benefits the transition between different BR. The quasi-static analysis shows that the cores are inherently safe over the entire BR range. All the results indicate that the flexible BR does not bring essential difficulties in designing the fast breeder reactor than the traditional ones. By restraining the initial excess reactivity, flattening the power distribution and redundantly arranging the control system, the reactor core can satisfy all the key requirements from the view of neutronics. However, loading blanket assemblies at unfixed positions, which is very important for the BR transition, brings a new difficulty. The coolant flow control technology must be advanced to fulfill the cooling requirements.

In this paper, the proposed BR range is from 1.1 to 1.4. However, it can be flexible in a wider range theoretically. The main constraint comes from using the versatile fuel design for all the BR. Although it simplifies the fuel manufacture and transition scheme greatly, the BR's range is limited.

More detailed studies will be done in the future on FFBR to consider the dynamically changing situation.

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