A NEW REACTOR BURNUP CONCEPT "CANDLE"

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ABSTRACT

A new reactor burnup concept CANDLE (Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy Energy production) is proposed, where relative shapes of neutron flux, nuclide densities and power density are constant but moves to an axial direction with a constant velocity during the whole life of reactor operation. It is applied to a lead-bismuth-eutectic cooled metallic fueled fast reactor, and a feasible result is obtained. The life of reactor can be extended as long as required by elongating its natural uranium region. The average burnup of spent fuel is 426GWd/t for the present design.

1. INTRODUCTION

Long life nuclear reactors are expected to be used for many places, where infrastructure is poor and/or refueling operation is difficult. Fast reactors have higher conversion capability from fertile to fissile materials and show less depression of excess reactivity along burnup. They are proper to be designed for long life reactors, and several designs have been proposed[1, 2]. Hattori, et al. try to move a fission region along core axis from its bottom to top by employing movable reflector in their reactor 4S[1]. This method enables to design a reactor whose life can be elongated as long as required by choosing its core height long enough. However, movable part (movable reflector) may cause reactivity insertion accident by malfunction. Sekimoto and Zaki[2] can make a reactor life long without any movable reactivity control instrument by choosing the proper initial fuel-loading-pattern to shift the fission region from outside (lower neutron importance region) to inside (higher neutron importance region) of the core along burnup. However, the reactor life is limited by its core size and power density.

In the present paper a new reactor burnup strategy CANDLE (Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy production) is proposed. In this burnup strategy, the fission region moves from bottom to top of the core like 4S reactor, but it does not require any movable reactivity control instrument. Relative shapes of neutron flux, nuclide densities and power density are constant but move to an axial direction with a constant velocity during the whole life of reactor operation as shown in Fig. 1. The burning direction can be reversed, if necessary. In the present design the upper region of the core is charged with natural uranium only. Fissile material is charged in the bottom region, whose amount does not depend of the life of reactor. The reactor life can be extended by elongating the natural uranium region.

It is apparent that the requirement of neutron economy should be severe to realize this burnup strategy. In the present paper metallic fuel is employed, since it shows excellent neutron economy[3]. Lead-bismuth-eutectic (LBE) is employed as coolant, though gas shows a much better performance[3], since its safety performance is superior compared to the gas.

The reactor design and burnup scenario of investigated reactor is presented in Section 2, calculation method is given in Section 3, and calculation results are given in Section 4 with some discussions.

2. REACTOR DESIGN AND FUEL CYCLE SCHEME

In the present study 3GWt LBE cooled metallic fuel fast reactor is investigated. As a cladding material HT-9 is employed. The diameter of fuel pin is 8 mm, and the cladding thickness is 0.35 mm. The core volume fractions of fuel, cladding and coolant are 0.60, 0.14 and 0.26, respectively. The reactor core is cylindrical, whose radius is 2 m. The core height is infinite for studying the characteristics of ideal CANDLE burnup scheme. However, in the actual calculation the height is finite but large enough, where at top and bottom core boundaries the neutron flux and leakage are negligibly small, and the changes of nuclide densities are also negligibly small. The reflector thickness is 0.5 m. The upper part of the core is filled with natural uranium. Below the natural uranium region U-238 is transformed to Pu-239. The lower part of this region works as fission region. The region below the fission region works as an inactive region, where a lot of fission products are located.

3. CALCULATION METHOD

Direct simulation of CANDLE burnup is not proper, since calculation is required for a very long reactor core. Convergence judgements on the neutron flux, nuclide densities and power density become also difficult because of the shift of the distributions along burnup iteration. In the present paper an artificial movement of fuel is employed to overcome this problem. By choosing its velocity proper the fission region becomes at rest. By this way the calculation region can be limited to a proper size of region, in which neutron flux shows meaningful value. The convergence judgements become also easy.

By introducing the fuel movement velocity, v, the density of i-th nuclide, n_i, satisfies the following equation:

$$
\begin{aligned}
v\frac{\partial n_i}{\partial s} = {} & \sum_j l_j a_{j\to i} n_j + \sum_g f_g \sum_j s_{a,j,g} b_{j\to i,g} n_j + \sum_g f_g \sum_j s_{f,j,g} g_{j\to i,g} n_j \\
& - l_i n_i - \sum_g f_g s_{a,i,g} n_i \quad ,
\end{aligned}
$$

where

f_g	: neutron flux of group g,
s	: axial distance in the transformed system,
l_i	: decay constant of nuclide i,
$s_{a,i,g}$	: absorption cross section of nuclide i for group g,
$a_{j\to i}$	: probability that decay of nuclide j produces nuclide i,
$b_{j\to i,g}$	: probability that neutron absorption in nuclide j produces nuclide i,
$g_{j\to i,g}$	: yield of nuclide i due to fission in nuclide j.

The core height is chosen to be large enough for suppressing the effect from the core axial edges of the model. The integration starts from the core top with natural uranium as an initial condition.

The neutron balance is denoted by 10-group diffusion equation. The vacuum boundary condition is employed at top and bottom core boundaries. The effects of this choice on the characteristics of CANDLE can be considered negligible, since the core height is chosen large enough. The group constants and their changes with respect to temperature and atomic density are prepared using SRAC code system[4] with JENDL-3.2 library[5].

The analysis consists with two stages. The first stage is a rough estimation of the proper fuel velocity, and the second is fine adjustment of the velocity to make the fission region at rest. The obtained fuel velocity is exactly opposite to the shape velocity in the actual reactor. In the first stage, a parametric survey is performed for the fuel velocity. When the fuel velocity is not proper, the shape shifts with iteration at the beginning, but the leakage from closer axial boundary suppresses this motion, so that this motion will stop at last. The fuel velocity is considered proper, when the neutron leakage required to stop the shape motion is lowest. Since the lowest neutron leakage brings the highest neutron multiplication factor, finding the fuel velocity that gives the highest multiplication factor is equivalent to the estimation of the proper fuel velocity.

In the second stage the fuel velocity is corrected iteratively from one obtained in the first stage by the following equation which is inserted between diffusion and burnup calculations:

$$v^{(k+1)} = w^{(k+1)} - \frac{2A}{H}\left(Z^{(k)} - \frac{H}{2}\right) \quad ,$$

where H and A are the core height and a constant, and $w^{(k)}$ and $Z^{(k)}$ are given by the following equations:

$$w^{(k)} = \begin{cases} v_0 & (k \le 2) \\ \dfrac{v^{(k)}\Delta Z^{(k-1)} - v^{(k-1)}\Delta Z^{(k)}}{\Delta Z^{(k-1)} - \Delta Z^{(k)}} & (k > 2) \end{cases}$$

$$Z^{(k)} = \frac{\sum_m z_m f_m{}^{(k)} V_m}{\sum_m f_m{}^{(k)} V_m} \quad ,$$

where

$$\Delta Z^{(k)} = Z^{(k)} - Z^{(k-1)}$$

and z_m, $f_m^{(k)}$ and V_m are axial distance, neutron flux at k-th iteration and volume, respectively, at axial mesh m. v_0 is the fuel velocity obtained in the first stage. By this iteration it is expected that $\Delta Z^{(k)}$ tends to zero and $Z^{(k)}$ converges to $H/2$.

4. CALCULATION RESULTS AND DISCUSSIONS

We choose $H = 5$ m . It is considered large enough, since additional calculations have shown that the change of boundary condition from vacuum to reflective does not change the following results. The calculated results for the first stage is shown in Fig. 2. We can see the change of effective neutron multiplication factor along fuel velocity. From this figure 1.2×10^{-7}cm/s is chosen as the initial guess for the second stage. Finally the proper fuel velocity, that is the inverse of shape velocity, is obtained to be 1.263×10^{-7}cm/s.

The effective neutron multiplication factor is 1.011 for this shape velocity. This value can be considered large enough to realize CANDLE burnup, since the change of reactivity along burnup can be ignored.

Figure 3 shows the neutron flux distribution, which is dangling in the radially outer region. It is attributed to the fact that the burnup progresses slower in this region, since the flux level is lower than the central region.

The average burnup of spent fuel is 426GWd/t, which is equivalent to more than 40% utilization of natural uranium. It shows a very high efficiency of natural uranium utilization and should be compared with the present situation. It is difficult for conventional light water reactor to utilize more than 1% of natural uranium, even reprocessing is employed.

4. CONCLUSIONS

The new burnup strategy CANDLE is proposed and applied to a LBE cooled metallic fuel fast reactor. The obtained effective neutron multiplication factor is 1.011, and shows the possibility to realize this burnup strategy. Though some amount of fissile material is required for bottom of the initial core, the life of reactor can be elongated to any value by elongating the core height. The burnup of spent fuel shows a very large value 426GWd/t. It means very efficient uranium utilization.

REFERENCES

1. S. Hattori, A. Minato, N. Handa, “Present Design Features of the Super Safe, Small and Simple Reactor,” *Potential of Small Nuclear Reactors for Future Clean and Safe Energy Sources*, pp. 225-234, Elsevier, Amsterdam, The Netherland (1992).

2. H. Sekimoto, Zaki S., “Design Study of Lead- and Lead-Bismuth-Cooled Small Long-Life Nuclear Power Reactors Using Metallic and Nitride Fuel,” *Nucl. Technol*., **109**, pp. 307-313 (1995).

3. A. Mizutani, H. Sekimoto, “Core Performance of Equilibrium Fast Reactors for Different Coolant Materials and Fuel Types,” *Ann. Nucl. Energy*, **25**, pp. 1011-1020 (1998).

4. K. Okumura, et al., “SRAC95; General Purpose Neutronics Code System,” JAERI-Data/Code 96-015, Japan Atomic Energy Research Institute (1996).

5. T. Nakagawa, et al., “Japanese Evaluated Nuclear Data Library Version 3 Revision 2: JENDL-3.2,” *J. Nucl. Sci. Technol*., **32**, pp. 1259-1271 (1995).

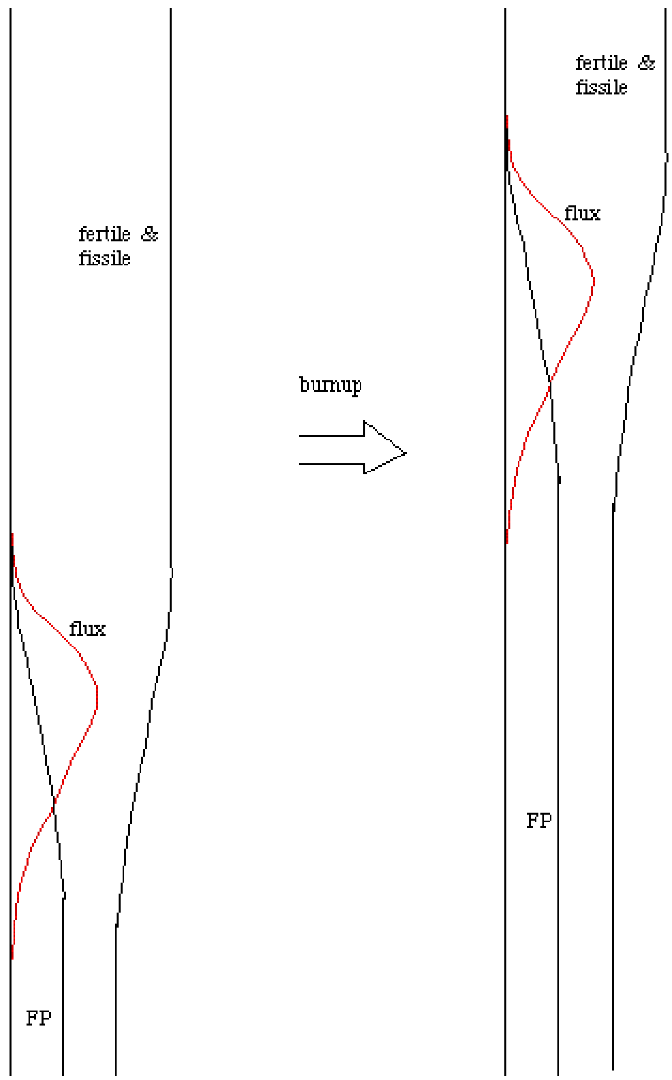

Fig. 1　Concept of CANDLE burnup concept

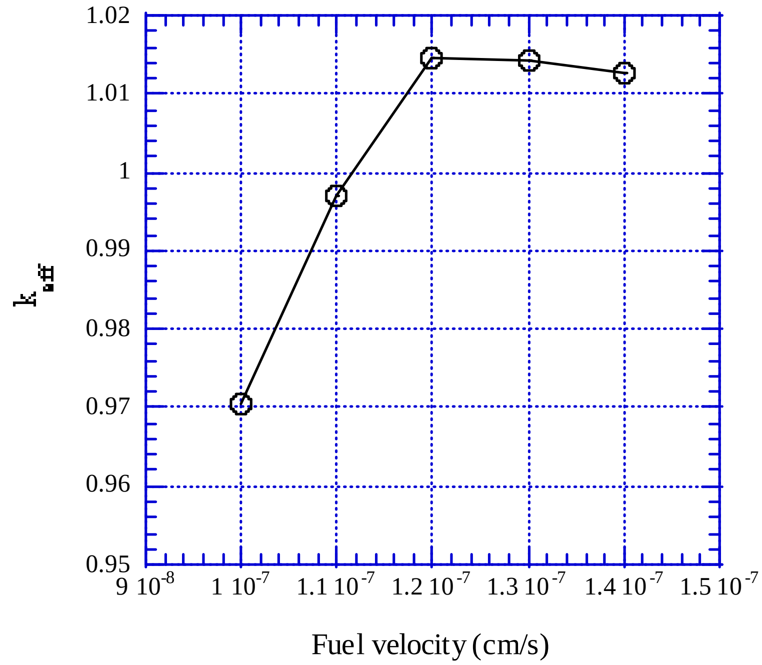

Fig. 2 Change of effective multiplication factor for different fuel velocities

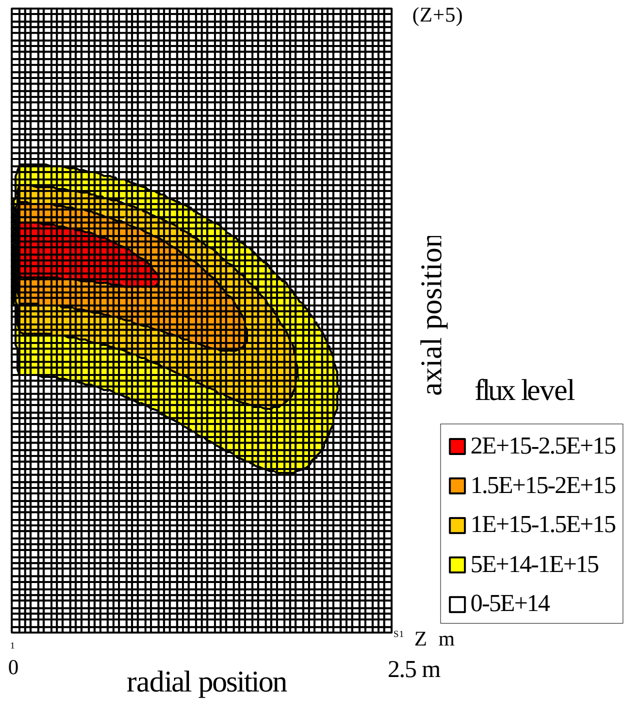

Fig. 3 CANDLE neutron flux distribution for a lead-bismuth cooled fast reactor with metallic fuel