CANDLE BURNUP FOR DIFFERENT CORES

Hiroshi Sekimoto and Kohtaro Tanaka
Research Laboratory for Nuclear Reactors
Tokyo Institute of Technology
O-okayama, Meguro-ku, Tokyo 152-8550, JAPAN
hsekimot@nr.titech.ac.jp

ABSTRACT

The new burnup strategy CANDLE is proposed, and calculation procedure for its equilibrium state is presented, where the power shape does not change along passing time in this strategy, and the excess reactivity and reactivity coefficient are constant during burnup. Any control mechanism for the burnup reactivity is not required, and power control is very easy. The life of reactor can be made longer by elongating the core height. This burnup strategy can be applied to several kinds of reactors, whose infinite neutron multiplication factor changes from less than or nearly unity to considerably more than unity, and then to less than or nearly unity. In the present paper it is applied to two types of fast reactors. One is a natural uranium fueled reactor. For this reactor some fissile material such as plutonium is required only for the nuclear ignition region of the initial core, but only natural uranium is required for the other region of the initial core and for succeeding cores. The average burnup of the spent fuel is about 40 % that is equivalent to 40 % utilization of the natural uranium without the reprocessing and enrichment. The other is a long-life small reactor. Since the speed of the burning region is only 1.8cm/y for the present design, the size of the reactor can be set small even for a very long-life reactor. This reactor also shows excellent simplicity and safety features.

1. INTRODUCTION

A new reactor burnup concept CANDLE (Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy producing reactor) is proposed [1, 2], where neutron flux, nuclide densities and power density do not change their shapes along burnup but move in the axial direction of a core with a constant velocity for a constant power operation during the whole reactor life as shown in Fig. 1. The moving direction can be either upward or downward, but in the present paper only the upward direction is considered for making description of the paper simple. If this concept is feasible, a long life reactor can be designed, whose life is easily set by adjusting its core axial length. The change of excess reactivity along burnup is theoretically zero for ideal equilibrium condition, and shim rods will not be required for this reactor. The reactor also becomes free from accidents induced by unexpected control rods withdrawal. The core characteristics, such as power feedback coefficients and power peaking factor, are not changed during life of operation. Therefore the operation of the reactor becomes much easier than the conventional reactors.

The upper part of the core is axially uniform, but the distribution of each nuclide density is complicated in the burning region. It may be difficult to construct the ignition zone for this burnup strategy. However, the configuration of the succeeding core becomes very easy. The burning zone at the end of reactor life can be used as the ignition zone of the succeeding core as shown in Fig. 2.

The CANDLE burnup may not start successfully without a good initial nuclide density distribution for the ignition region. Usually we do not have such a distribution. It is better to find the equilibrium state at first, and to construct the ignition region by modifying the burning region. The calculation method to get the equilibrium state directly is given in this paper.

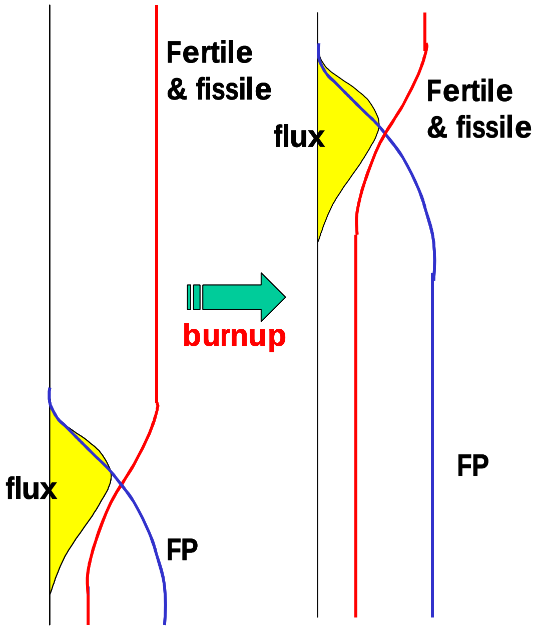


Figure 1. Concept of CANDLE burnup strategy.

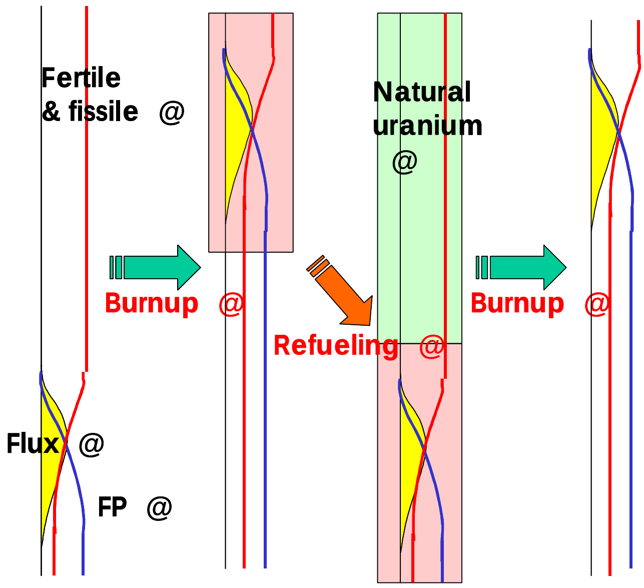


Figure 2. Concept of CANDLE burnup and refueling strategy.

The CANDLE burnup strategy can be applied to several reactors, when the infinite neutron multiplication factor of fuel element changes along burnup in a proper way such as it changes from less than or nearly unity to considerably more than unity, and then to less than or nearly unity. Only fast reactor cases are presented in the present paper, since for thermal reactors many design parameters should be considered compared to the fast reactor, though even thermal reactor can offer interesting examples by introducing highly enriched fuel and burnable poisons.

For fast reactors we consider two extreme cases. One is the most neutron economical case, where neither enrichment nor reprocessing is required once the burnup starts. Namely only depleted or natural uranium is required for succeeding cores. It is good not only for economy performance but also proliferation resistance. It is a difficult problem to make the system critical.

The other is a small reactor. It is also difficult problem, since neutron leakage is large and neutron economy is poor. Present power-plant constructors hesitate to build a larger reactor for the reason of its large economical risk, and prefer to build a smaller reactor if possible. The small reactor has several merits. It can be built on a less graded land such as a small land and less stable land. Therefore, it is much easier to find a proper site for them. The small reactor is suitable for several purposes other than electricity production, such as heat generation, desalination, etc, since their total demand is much smaller than electricity production by nuclear plant. The small reactor is also preferable for exportation to developing countries. For these cases the small reactor becomes better, if it is long-life, since it usually shows much better performance for maintenance and proliferation problems. Therefore the CANDLE burnup strategy is applied to the small long-life reactor.

2. CALCULATION METHOD FOR CANDLE BURNUP IN EQUILIBRIUM STATE

2.1 BASIC EQUATIONS FOR CANDLE

The present work is the first step of feasibility study of the CANDLE burnup strategy. The purpose of the present paper is to find the equilibrium state to satisfy the CANDLE requirements that the distributions of nuclide densities, neutron flux, and power density move axially with corresponding constant shapes and the same constant speed along burnup for constant power operation. Once these shapes are obtained as realistic values for a given reactor design, the CANDLE burnup strategy can be considered feasible for the design. Nuclide density distributions similar to the equilibrium ones may be used as proper initial distributions for the actual reactor as already mentioned. Many kinds of density distributions can be considered as such a distribution. However, the present study includes only finding a steady state of the CANDLE burnup, and the study on initial conditions and practical simulations from these conditions are the next subjects and not discussed in this paper.

For calculating the steady state CANDLE burnup, a coordinate transformation is employed in order to make the burning region at rest in the transformed coordinate system. By this way the calculation region can be limited to a proper size of region. Otherwise, the necessary calculation region is expanding, since the burning region moves steadily with repeating iteration of calculation. The convergence judgment becomes also easy for this coordinate system. The actual mathematical treatment is shown in the followings.

In the present paper a cylindrical core is considered. For *r-z* coordinate system, the neutron balance equation and nuclide balance equation in the core can be written as the following equations, respectively:

$$\frac{1}{r}\frac{\partial}{\partial r} r D_g \frac{\partial}{\partial r}\phi_g + \frac{\partial}{\partial z} D_g \frac{\partial}{\partial z}\phi_g - \sum_n N_n \sigma_{R,n,g}\phi_g + \sum_n N_n \sum_{g'} f_{n,g'\to g}\sigma_{S,n,g'}\phi_{g'} + \frac{\chi_g}{k_{eff}}\sum_{g'}\sum_n N_n \nu\sigma_{F,n,g'}\phi_{g'} = 0 \quad (1)$$

$$\frac{\partial N_n}{\partial t} = -N_n\left(\lambda_n + \sum_g \sigma_{A,n,g}\phi_g\right) + \sum_{n'} N_{n'}\lambda_{n'\to n} + \sum_{n'} N_{n'}\sum_g \sigma_{n'\to n,g}\phi_g \quad (2)$$

where

$\phi_g = \phi_g(r,z,t)$ = neutron flux in *g*-th energy group,

$N_n = N_n(r,z,t)$ = nuclide number density of *n*-th nuclide,

$D_g = D_g(r,z,t)$ = diffusion coefficient for *g*-th energy group,

χ_g = probability that fission neutron will be born in g-th energy group

k_{eff} = effective neutron multiplication factor

$\sigma_{R,n,g}$ = removal cross-section of *n*-th nuclide for *g*-th energy group,
$\sigma_{A,n,g}$ = absorption cross-section of *n*-th nuclide for *g*-th energy group,
$\sigma_{F,n,g}$ = fission cross-section of *n*-th nuclide for *g*-th energy group,
$\sigma_{S,n,g}$ = slowing-down cross-section of *n*-th nuclide from *g*-th energy group,
$f_{n,g'\to g}$ = element of slowing-down matrix of *n*-th nuclide from *g*'-th to *g*-th energy group,
$\sigma_{n'\to n,g}$ = transmutation cross-section of *n*'-th to *n*-th nuclide for *g*-th energy group,
λ_n = decay constant of *n*-th nuclide,
$\lambda_{n'\to n}$ = decay constant of *n*'-th to *n*-th nuclide.

The neutron flux level is normalized by the total power of reactor.

The production of FP can be evaluated using $\sigma_{n'\to n,g}$ given by

$$\sigma_{n'\to n,g} = \sigma_{F,n',g}\gamma_{n'\to n}$$

where

$\gamma_{n'\to n}$ = yield of *n*-th nuclide (FP) from neutron-induced fission by *n*'-th nuclide (actinide).

The spontaneous fission is neglected in the present study. The removal cross-section is the sum of absorption and slowing-down cross-sections:

$$\sigma_{R,n,g} = \sigma_{A,n,g} + \sigma_{S,n,g}$$

and the absorption cross-section is the sum of fission and capture cross-sections.

The distributions of neutron flux and nuclide densities move with burnup. Their relative shapes are constant and their positions move with a constant speed V along z-axis for the CANDLE burnup. For this burnup scheme, when the following Galilean transformation [3] given by

$$r' = r$$

$$z' = z + vt$$

$$t' = t$$

is applied to Eqs. (1) and (2), then they will be changed to

$$\frac{1}{r'}\frac{\partial}{\partial r'} r' D'_g \frac{\partial}{\partial r'}\phi'_g + \frac{\partial}{\partial z'} D'_g \frac{\partial}{\partial z'}\phi'_g - \sum_n N'_n \sigma_{R,n,g}\phi'_g + \sum_n N'_n \sigma_{n,g-1\to g}\phi'_{g-1}$$

$$+ \frac{\chi_g}{k_{eff}} \sum_{g'} \sum_n N'_n \nu\sigma_{F,n,g'}\phi'_{g'} = 0 \qquad (3)$$

and

$$\frac{\partial N'_n}{\partial t'} = -v\frac{\partial N'_n}{\partial z'} - N'_n\left(\lambda_n + \sum_g \sigma_{A,n,g}\phi'_g\right) + \sum_{n'} N'_{n'}\lambda_{n'\to n} + \sum_{n'} N'_{n'} \sum_g \sigma_{n'\to n,g}\phi'_g \qquad (4)$$

where $-v$ is the relative speed of the transformed coordinate system to the original system. If $v = V$, the distributions of nuclide densities and neutron flux stand still and Eq. (4) becomes

$$-V\frac{\partial N'_n}{\partial z'} - N'_n\left(\lambda_n + \sum_g \sigma_{A,n,g}\phi'_g\right) + \sum_{n'} N'_{n'}\lambda_{n'\to n} + \sum_{n'} N'_{n'} \sum_g \sigma_{n'\to n,g}\phi'_g = 0 \qquad (5)$$

It can be expected from this equation that the speed of burning region, V, is proportional to the flux level, if the effects of radioactive decay of nuclides can be neglected.

2.2 ITERATION SCHEME

We have two equations, Eqs. (3) and (5), which should be solved simultaneously with the flux normalization condition for obtaining the equilibrium CANDLE burnup state. From this point the prime used for the Galilean transformed variables is omitted for simplicity, since we will not meet any confusion in the following discussion. We introduce an iteration scheme to solve these equations. From a given flux distribution nuclide density distributions are obtained using Eq. (5), and then from these nuclide density distributions a more accurate distribution of neutron flux is obtained using Eq. (3) with the normalization condition. This procedure is repeated until it converges. However, in usual cases, the exact value of V is unknown. Then an initial guess is introduced, and this value should be improved at each iteration stage. If the employed value of V is not correct, the distributions of neutron flux and nuclide densities are expected to move along z-axis. Therefore, the value of V can be modified from the value of distance by which those distributions move per each iteration stage.

In order to define the position of these distributions, the center of neutron flux distribution is introduced, which is defined as

$$\boldsymbol{r}_C = \frac{\int \phi(\boldsymbol{r})\boldsymbol{r}d\boldsymbol{r}}{\int \phi(\boldsymbol{r})d\boldsymbol{r}}$$

where $\boldsymbol{r}$ is the coordinate vector representing (r, z) and integration is performed over the whole core. In the present paper we consider our problem more theoretical than practical. Though the height of the core is finite for the practical case, the infinite length is considered as the core height for the present study since the height is an artificial parameter and can be changed.

Now the way of modification of V at each iteration step is discussed. When the velocity $V^{(i)} \neq V$ is employed, the distribution is considered to move with the velocity proportional to $V^{(i)} - V$ from the analogy of Eqs. (4) and (5). One cycle of iteration corresponds to passing the time proportional to $\Delta z / V^{(i)}$ considered from Eq. (5), where Δz is mesh width of the z-axis. Since the system in the present paper is cylindrically symmetric, only the z direction should be considered. Therefore, when z coordinate value of $\boldsymbol{r_C}$ is obtained as $z_C^{(i)}$ for the i-th iteration for a given velocity $V^{(i)}$, the following relation can be expected:

$$\Delta z_C^{(i)} = \alpha (V^{(i)} - V) / V^{(i)}$$

where $\Delta z_C^{(i)} = z_C^{(i)} - z_C^{(i-1)}$ and α is a constant. From this relation we can derive the following equation to get a proper estimate of V for the (i+1)-th iteration using the results for the i-th and (i-1)-th iteration:

$$V^{(i+1)} = V^{(i)} V^{(i-1)} \frac{\Delta z_C^{(i)} - \Delta z_C^{(i-1)}}{\Delta z_C^{(i)} V^{(i)} - \Delta z_C^{(i-1)} V^{(i-1)}} \tag{6}$$

Now we have a whole iteration scheme, but it is required to use two good initial guesses of V, $V^{(1)}$ and $V^{(2)}$. They can be estimated from several trial calculations with different values of V', where the value of V' is fixed during the iteration. Here Eqs. (3) and (5) are solved repeatedly. If the height of core is infinity, the burning region moves forever for $V' \neq V$. Though the infinite cylindrical core is considered in this paper from the theoretical purity, in this calculation stage for finding two good initial guesses, large but finite height is treated, and zero-flux boundary condition is set for both upper and lower core boundaries. Then, the motion of burning region stops finally after several iterative calculations even for $V' \neq V$, since the boundary condition does not permit the burning region to pass the boundary. The iteration is converged. If the value of tried *V'* is more different from *V*, then the burning region of core moves more until it converges and arrives nearer to the boundary. The case in which the burning region stays nearer to the boundary gives a smaller value of k_{eff}, since the neutron leakage becomes larger. Therefore the *V'* value, which gives the largest value of k_{eff}, should be usually the best candidate of the initial guess of *V* among all trial values. By using the best two values of *V* the iterative calculation mentioned above is started.

3. CALCULATION CONDITIONS AND RESULTS

3.1 CALCULATION CONDITIONS

The CANDLE burnup strategy can be applied to many kinds of reactors, when the infinite neutron multiplication factor of fuel element of the reactor changes along burnup as the followings. It starts from the value less than or nearly unity, and increases with burnup, and becomes considerably more than unity after certain amount of burnup, and then takes its maximum value at a certain value of burnup, and then decreases with burnup, and becomes less than or nearly unity, and continues to decrease with burnup. The spatial regions before and behind the burning region, where the infinite neutron multiplication factor is less than or nearly unity, are inevitable to fix the shape of burning region to a local area by shifting both front and back ends of the shape to the same direction with the same speed. The condition that infinite neutron multiplication factor for some interval should be more than unity is also inevitable for keeping the system critical. The amount of the surplus of infinite neutron multiplication factor above unity should be large enough to supply excess neutrons to the front region for fissile nuclide production.

To satisfy this condition an excellent neutron economy should be satisfied in the equilibrium state. Even for fast reactors it is not easy to realize this scenario for our cases: natural uranium fueled reactor and long-life small reactor. Only fast reactor with excellent neutron economy can realize them. A lead-bismuth-eutectic (LBE) cooled metallic fuel fast reactor, whose fuel volume fraction is 50%, is investigated. The infinite neutron multiplication factor of the natural uranium fuel is, ob course, less than unity, and the problem is to raise the effective neutron multiplication factor of the total system. On the other, making the effective multiplication factor more than unity is rather easy for the long-life small reactor, since plutonium can be added for this design. However, the addition of plutonium makes difficult to reduce the initial infinite multiplication factor.

Design parameters for these reactors are shown in Table I. Since in the present paper we try to find a theoretically general solution, the core height is set infinite. However, numerical calculation can be performed only for finite dimensions. We set the height of core 8m, and try to confirm the no effects on the results caused by changing the boundary conditions as mentioned as mentioned in the previous section.

The group constants and their changes with respect to temperature and atomic density are calculated using a part of SRAC code system [4] with JENDL-3.2 nuclear data library [5]. In the present calculation 20 actinides and 66 fission products are employed. The capture cross section of the nuclides produced by neutrons capture of the nuclide at the end of the nuclide chain is assumed to be the same as the cross section of the nuclide at the end of the chain. It is equivalent to that the nuclide at the end of the chain remains the same nuclide even after capturing neutrons.

Table I. Design parameters for natural uranium fueled reactor and long-life small reactor

		Natural Uranium Fueled Reactor	Long-life Small Reactor
Total thermal output		3000MW	150MW
Core & reflector dimensions	Core radius	2.0m	0.8m
	Radial reflector thickness	0.5m	0.5m
Fuel-pin structure	Diameter	0.8cm	1.0cm
	Cladding thickness	0.035cm	0.08cm
	Pellet density	75%TD	75%TD
Materials	Fuel (weight %)	90U-10Zr	71U-19Pu-10Zr
	Cladding	HT-9	HT-9
	Coolant	LBE	LBE
Fuel volume fraction		50%	50%

3.2 CALCULATION RESULTS

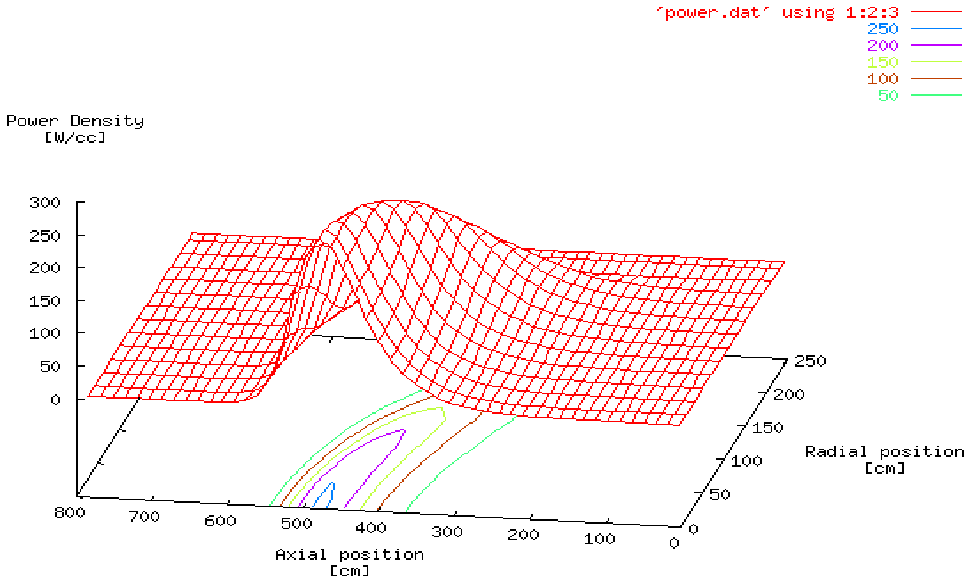
For both cases feasible solutions are obtained. The obtained results are shown in Table II. The effective neutron multiplication factors are more than unity. Since the calculation has been performed for the hot equilibrium core, the excess reactivity is not necessary. For the case of effective multiplication factor of unity, the reactor can be designed more feasible and attractive from engineering points.

Table II. Calculated results for natural uranium fueled reactor and long-life small reactor

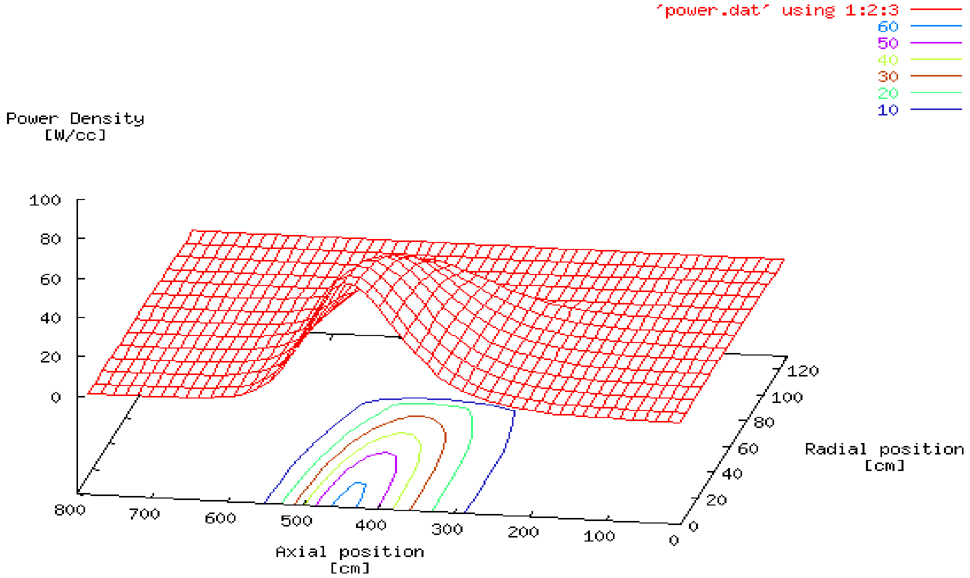
	Natural Uranium Fueled Reactor	Long-life Small Reactor
Effective neutron multiplication factor	1.012	1.001
Speed of burning region (cm/y)	4.40	1.80
Half width of axial power shape (cm)	150	145
Average burnup of spent fuel (GWd/t; %)	358 ; 38.2	267 ; 28.5

The power density distributions are shown in Fig. 3. For the natural uranium fueled reactor it is dangling in the radially outer region, but its effect is very small for the long-life small reactor. The dangling is attributed to the fact that the burnup progresses slower in peripheral region, since the flux level is lower than the central region. This effect is larger for the larger reactor. The dangling is not good from neutron economy and core design.

In the following, the obtained features of each reactor are presented and discussed.



(a) natural uranium fueled reactor



(b) long-life small reactor

Fig. 3. Power density distributions in natural uranium fueled reactor and long-life small reactor.

3.2.1 Natural Uranium Fueled Reactor

For the natural uranium fueled reactor, the fuel burnup is 38.2%. It means that 38% of natural uranium is completely utilized without reprocessing and enrichment. It should be compared with the present LWR system, where only less than 1 % of natural uranium can be utilized. Even when once-through cycle is applied, the total amount of disposed fuel per unit power production is much lower than the present LWR once-through system. Even the proposed future fast reactor system can utilize 60 to 70%, with which our system may be competitive from fuel utilization. Our system does not require reprocessing. The waste treatment for our system is much simpler then the system with reprocessing. The technologies and facilities of uranium enrichment and spent fuel reprocessing are the most important items for nuclear proliferation. Our system does not require these technologies and facilities. Since the reactor is long life and the refueling is not required, the reactor vessel can be sealed during reactor life. It helps physical protection. All these characteristics enhance the sustainability of nuclear energy.

Our system does not require any control system for compensating burnup excess reactivity. Since the coolant flows along the core axis, total heat input to each coolant channel does not change along burnup. Then time dependent control of flow distribution over coolant channels is not required. Once the distribution is optimized such as to equalize coolant output temperature for every coolant channel, the output temperature is always equalized during the reactor life without any control. Therefore, any control along burnup is not required. The reactor operation life can be extended by designing the core height long enough. The half width of axial power shape is 150cm. The speed of burning region is 4.40cm/y. Then even a 20 years operation adds only 88 cm to the core height. Therefore it is possible to design a reactor, which does not require refueling. The maintenance of this reactor becomes simple, safe and easy. The reactor is barely critical and redistribution of fuel usually reduces its criticality performance. Therefore, it is almost free from CDA accident. All these features make our reactor very safe and reliable, and its operation and maintenance become very simple and do not require any highly trained specialists.

The evaluation of economics of this kind of really innovative reactor is difficult, but it has a big potential to realize low price of produced energy, since safe and reliable system can eliminate a lot of safety oriented equipments, systems and personnel. The eliminations of burnup control system (control rods and orifices) and refueling mechanism are considerable for reducing the capital cost. The key issue for realizing our concept is to maximize neutron economy with lowest penalty of core cooling performance and to solve the problems associating to high burnup.

3.2.2 Long-life Small Reactor

The fuel burnup of the long-life small reactor is 28.5%. It is less than the natural uranium fueled reactor mentioned above, but still very attractive. It may be even better from its feasibility, since the maximum permissible burnup of the present advanced fuel is only around 10%. The speed of the burning region is only 1.80cm/y. The reactor size can be made small even for a very long-life reactor. Even the 20 years operation lengthens the core height by only 36 cm. Such a small reactor can be transported to a site as a completed reactor and will be installed there. After a certain period of operation it can be replaced with a new one. It is suitable to developing countries or local area. Since the reactor is long life and the refueling is not required, the reactor vessel can be sealed during reactor life. It enhances qualitatively physical protection performance.

Since the safety features of LBE cooled small long-life reactor have already been discussed intensively by one of the authors[6,7], general safety features are not mentioned here except only notifying its negative void coefficient and some good characteristics of LBE such as chemical inertness and high boiling temperature. The safety features mentioned for the natural uranium fueled reactor are also attained for the long-life small reactor.

CONCLUSIONS

The CANDLE burnup has the following general merits:

1) Burnup reactivity control mechanism is not required: The reactor control becomes simpler and easier. The excess burnup reactivity becomes zero, and the reactor becomes free from reactivity-induced accidents. The burnup of control rod becomes negligible. Neutrons are efficiently utilized.

2) Reactor characteristics do not change with burnup: The reactor characteristics such as power peaking, reactivity coefficients do not change with burnup. The expectation of core condition becomes very reliable. The reactor operation strategy remains same for different burnup stage.

3) Control of coolant flow rate is not required for each channel along burnup: Since the radial power profile does not change with burnup, the required flow rate for each coolant channel does not change. Therefore, the orifice control along burnup is not required. The operational mistakes are avoided.

4) Radial power distribution can be optimized more thoroughly: Since the radial power distribution does not change with time, it can be optimized more thoroughly. The optimization method is much simpler.

5) By simply increasing the core height, the reactor life can be elongated. Design of long-life reactor becomes easier. Even a very long-life reactor does not require refueling during its life. Such a reactor is well suited to the case that high infrastructures cannot be expected. ?

The CANDLE burnup is applied successfully to the natural uranium fueled and long-life small reactors.

The natural uranium fueled reactor shows the following excellent performances in addition to the general merits mentioned above:

1) Any enriched fuels are not required after the second core. Only natural or depleted uranium is enough to be charged to the core after the second core. Namely, if the fuel for the first core is available, neither enrichment nor reprocessing plant is required.

2) The burnup of the spent fuel is about 40%: This value is competitive to the value of the present system consisting with fast reactor plus reprocessing plant. The 40% of natural uranium is utilized for energy production without enrichment or reprocessing.

3) CDA accident is avoided: Since the reactor is just critical without any absorbers and contains no surplus fissile materials in its core, CDA accident is hard to be happened.

The long-life small reactor shows the following excellent performances in addition to the general merits:

1) The reactor size can be made small even for a very long-life reactor: Since the speed of the

burning region is only 1.80cm/y for the present design, the size of the reactor can be set small even for a very long-life reactor. Such a small reactor can be transported to a site as a completed reactor and will be installed there. After a certain period of operation it can be replaced with a new one. It is suitable to developing countries or local area. Since the reactor is long life and the refueling is not required, the reactor vessel can be sealed during reactor life. It enhances qualitatively physical protection performance.

2) LBE cooled small fast reactor can be designed very simple and safe: LBE cooled small fast reactor has a lot of safety features such as negative void coefficient, chemical inertness and high boiling temperature. The refueling is not required. The reactor characteristics do not change with burnup. The control of coolant flow rate is not required for each channel along burnup. Burnup reactivity control mechanism is not required either.

ACKNOWLEDGEMENTS

The present study is supported partly by the JAERI's Nuclear Research Promotion Program.

REFERENCES

1. H. Sekimoto and K. Ryu, "A New Reactor Burnup Concept "CANDLE"," *Proceeding of PHYSOR 2000*, Pittsburgh, May 7-11, 2000, CD (2000).
2. H. Sekimoto, K. Ryu and Y. Yoshimura, "A New Burnup Strategy CANDLE," *Nucl. Sci. Engin.*, **139**, 306 (2001).
3. H. Goldstein, *Classical Mechanics*, Addison-Wesley, Reading, Massachusetts (1950).
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: JENDLE-3.2," *J. Nucl. Sci. Technol.*, **32**, 1259 (1995).
6. H. Sekimoto and Zaki S., "Design Study of Lead- and Lead-Bismuth-Cooled Small Long-Life Nuclear Power Reactors Using Metallic and Nitride Fuel," *Nucl. Technol.*, **109**, 307 (1995).
7. Zaki S. and H. Sekimoto, "Accident Analysis of Lead or Lead-Bismuth Cooled Small Safe Long-Life Fast Reactor Using Metallic and Nitride Fuel," *Nucl. Engin. and Design*, **162**, 205-222(1996).