Proceeding of the Korean Nuclear Society Spring Meeting
Kori, Korea, May 2000

The Concept of Proliferation-resistant, Environment-friendly, Accident-tolerant, Continual, and Economical Reactor (PEACER)

S.H. Jeong, I.S. Hwang, W.S. Yang*, B.G. Park, K.Y. Suh and C.H. Kim

Department of Nuclear Engineering, Seoul National University,
56-1 Shinlim-dong Gwanak-ku Seoul 151-742, Republic of Korea
**Department of Nuclear Engineering, Chosun University,*
375 Sursuk-dong Dong-gu Kwang-ju 501-759, Republic of Korea
Tel : +82-2-880-7215, Fax : +82-2-889-2688, E-mail : hisline@snu.ac.kr

ABSTRACT

As an effort to ameliorate generic concerns with current power reactors such as the risk of proliferation, radiological hazard of the spent fuel, and the vulnerability to core-melt accidents, the concept of a revolutionary reactor, named as PEACER, has been developed as a proliferation-resistant waste transmutation reactor with its technical footing on proven technologies of critical reactors and heavy liquid metal coolant. In this paper, results of PEACER conceptual design are summarized with the focus on the neutronic characteristics and the expected general system performance. The proliferation resistance of PEACER is built by installing both institutional and technical barriers. The latter includes denaturing of fissile materials, Pu in particular, as well as the intense radiation field associated with the pyrochemical partitioning method. When the fuel volume fraction and the core aspect ratio(L/D) are optimized, the transmutation capability of PEACER for long-living wastes from LWR spent fuels is found to exceed their production rate of two LWRs each at the same electric rating. In contrast with current power reactor design principles, the lower power density and the higher neutron leakage rate lead to the higher performance on the proliferation-resistance, transmutation capability and the accident-tolerance. The conceptual design result has shown promising characteristics in all the five target areas defined by its name PEACER, warranting more detailed studies.

1. Introduction

Due to the scarcity in domestic energy resources and strong energy dependence of her industry and society, Korea relies heavily on nuclear power for electricity generation. The nuclear power in the 21st century, however, is expected to confront with global controversy related with the risk of proliferation, the radiological hazard of the spent fuel, and the depletion of uranium resources if the disadvantage of water reactor technology is not overcome. The expectation calls for more fundamental steps to resolve the issues for the sake of the national energy security of resource-poor countries with high nuclear energy dependency. Recently, strong technical drive for nuclear wastes transmutation has been initiated and spreading world-wide to stimulate revolutionary approaches. Consensus is being developed on the fact that most significant long-living radioactive isotopes in the current power reactor spent fuels can be transmuted into short-living elements by using fast neutron spectrum with localized thermal traps[1].

Geological disposal of the spent fuel, while being actively developed as one of the most visible options today, may be confronted with societal obstacles especially in countries with dense population and with political questions raised by the expected accessibility of plutonium by the decay of intense radiation barriers over hundreds of years. For sustained growth, plutonium and other actinides are better to be utilized as valuable fuel resources. Therefore the development of transmutation technology with built-in proliferation-resistance and inherent safety features can receive an international support. We believe that economical transmutation systems can be developed through creative combination of available technologies and institutional approach among world nuclear R&D community. The PEACER concept is driven towards this goal by utilizing a low power density core consisting of metallic fuels cooled by chemically stable Pb-Bi eutectic liquid metal.

In the development of the conceptual design, the design targets and approach are established as follows.

1) Proliferation-resistance: Both technical and institutional barriers are employed with the defense-in-depth approach in each step of nuclear materials processing. The adequacy of the barrier system shall be judged as to international acceptance.

2) The transmutation capability: One PEACER shall transmute long-living wastes in the spent fuel from at least two LWRs with the same power rating, leaving behind only short-lived wastes suitable for land-based disposal.

3) Inherently safety: Using the inert Pb-Bi coolant with no fire hazard, the nuclear system shall be adequately cooled by natural convection even under the loss of ultimate heat sink accident scenarios.

4) Fuel continuity: All actinides shall be made useable as fuel in order to extend fuel resources by order of magnitude from that for LWRs.

5) Improved economy: The power generation cost of the PEACER shall be competitive with that of LWRs with a design life of 60 years.

2. Simulation Results

2.1. Calculation Method

2.1.1. Core parameters

Transmutation process requires partitioning that inherently raises concern with the possibility of diversion or theft of sensitive nuclear materials. As the PEACER is designed to have high transmutation rate, it is assumed that the facility is centralized as a waste burning energy park that can be operated under the international control. The utilization of pyroprocessing technique will enhance the proliferation resistance due to its inherent limit in separability and intense radiation field.[2] An additional approach of denaturing the sensitive fissile element is pursued especially for Pu. As a measure of the denaturing extent for proliferation-resistance, the odd ratio is defined as follows;

$$OR = \frac{Mass\ of\ Odd\ number\ isotope}{Mass\ of\ selected\ element} \tag{1}$$

The OR indicates the possibility of explosion of nuclear fuel material. Volpi suggested that the explosive power of a nuclear weapon is drastically decreased when the OR value is lowered.[3]

Environment-friendliness is measured by the transmutation rate that is defined by support ratio, as below;

$$SR\ (Support Ratio) = \frac{TRU\ reduction\ mass\ scaled\ to\ same\ electric\ power,\ 1 year\ cycle\ length}{TRU\ mass\ per\ year\ in\ 1GWe\ 150 days\ cooled\ LWR\ spent fuel} \tag{2}$$

For clarity in the performance comparison, the reactor power is measured in electric rating. The calculated TRU transmutation rate of PEACER is scaled linearly with the electrical power and cycle length of corresponding LWR. Considering its usefulness as a transmuter and associated economy, the target SR value for PEACER is set to exceed 2.0.

For the conceptual design study, a reference power is selected to be 1,575 MWt after the case of Integral Fast Reactor (IFR).[2] It is desired to maximize SR while minimizing OR by optimization of core design parameters including L/D and FVF, the ratio of core active height to diameter and the fuel volume fraction, respectively. The fuel volume fraction is defined as the ratio of metallic fuel (U+TRU+Zr) volume to total core volume. It is expected that the neutron leakage is increased with decreasing the L/D. The more neutron leakage represents less production of higher actinides due to neutron capture at a given transmutation rate that in turn will increase SR and decrease OR. Therefore the design approach is directed toward high leakage geometry. Decrease in volume fraction also increases the neutron leakage. In a typical Na-cooled fast reactor the fuel volume fraction is ranged from 35% to 40%.[2][4] If core power density is lowered, for example to the level of Russian Pb-Bi fast reactors, the transmutation rate is significantly increased from that of IFR burner design with corresponding increase in TRU loading.

2.1.2. Reference Core Design

Figure 1 and 2 shows the schematic of PEACER reference design with a pancake geometry to increase axial leakage. Radial leakage is enhanced by introducing a central reflector region. The reference PEACER design also has significantly high pitch to diameter ratio of fuel pin compared with that of Na-cooled fast reactors. The feature is utilized to allow sparse square lattice in a fuel assembly with adequate structural reinforcement against potential flow induced vibration with the heavy liquid metal coolant.[5][6][7]

U-TRU-Zr metallic fueled core having 365 days burn cycle time is compared with core design of 330 days case about Pu OR and SR. Accident-tolerance estimation was carried out in U-TRU-Zr fueled 330 days burn cycle time case.

2.2. Proliferation-Resistance

Parametric study of proliferation-resistance and environment-friendliness was performed about 365 days burn cycle time in equilibrium. The design of burn cycle time 330 days can almost achieve the objects of previous design and also is accord to general 3-year fuel recycling plan.

Pu OR of 330 days burn cycle time core on L/D and FVF is same as results in previous study, 365 days burn cycle time. As shown in figure 3(a) and figure 3(b), Pu OR decreases as L/D and FVF decreases. Most of fission reaction is occurred at Pu odd isotope especially Pu-239. Thus, at equilibrium cycle, EOC Pu OR is decreased. U-238 captures the neutron and converts to Pu-239 continuously, but Pu is destroyed more rapidly because of high leakage design of burner core. Therefore, it can be said that increased transmutation of Pu is same as decreased reproduction of Pu. A little shift up is caused by decrease of transmutation capability owing to reduced burnup. The discharge burnup in this study is described with unit of at%. The GWD or MWD which is the unit of energy will be mentioned particularly.

We can easily find out that the overall trends on design parameters are equivalent both of 365 days burn cycle time case and 330 days burn cycle time case. Thermal power and total core volume, and other design values are remained with 365 days burn cycle time case, thus the discharge burnup should be decreased. And reduction of TRU transmutation will be followed, causing increase of Pu OR.

2.3. Environment-Friendliness

The SR trends on L/D and FVF is same as results of U-TRU-Zr fueled 365 days burn cycle time case. Decrease of L/D and FVF enhance neutron leakage and the possibility of fission in fuel is reduced. Therefore, TRU wt% must increase to compensate the loss of neutron in fuel. Additionally the U-238 capture reaction is decreased, and TRU regeneration decreases. TRU regeneration from U-238 is most important factor in controlling the transmutation capability.

The shift down of SR as shown in figure 4 is caused by also decrease of discharge burnup. Discharge burnup is related with total thermal energy produced during cycle length, what is thermal power times total burn cycle time or destruction mass by fission. With same cycle length and power, more charged fuel amount means less discharge burnup. In other ways, with same power and fuel amount, less cycle length means less burnup. Therefore the TRU wt% will be increased by extension of burnup. while the reproduction of TRU would decrease. In this case, the burn cycle time is reduced to 330.0 days, and burnup is also reduced to about 10 at%.

2.4. Core Stability and Accident-Tolerance

Four reactivity coefficients and integral reactivity parameters will be calculated as function of L/D and FVF using the quasi-static reactivity balance. These reactivity parameters will represent the stability and the accident-tolerance of the reactor core. Then the anticipated transients analysis will be followed derived from quasi-static reactivity balance equation. In this calculation, perturbed outlet temperature will be compared with other fast reactor designs in case of LOFWS(Loss Of Flow Without Scram) and LOHSWS(Loss Of Heat Sink Without Scram).

Reactivity feedback parameters and neutron kinetics parameters were obtained by utilizing the DIF3D computer code.[8][9] The DIF3D code was utilized to determine neutron flux distribution and spectrum, using 9 group or 33 group cross section libraries, with 3-D cartesian geometry. The 9 group cross section library does not have temperature dependent database that is required to calculate Doppler coefficient. For this reason, only 33 group cross section library, having 980K and 1580K core temperature data generated for the ATW MA burner analysis, was employed in this study.

A safety analysis with complete transient calculations falls outside the scope of this study. Wade and et al. introduced a quasi-static approach to evaluate the consequences of anticipated transients without scram(ATWS).[2][10][11][12] Using some simplifications, The core response to external events can be described by the quasi-static reactivity balance,[12] as follows;

$$0 = \triangle\rho = (P - 1)A + (P/F - 1)B + \delta T_{in} C + \triangle\rho_{ext} \quad , \tag{3}$$

where,

P, F = normalized power and flow, respectively

δT_{in} = change from normal coolant inlet temperature.

$\Delta\rho_{ext}$ = externally imposed reactivity

A, B, C = integral reactivity parameters that are measurable at the operating plant via perturbations introduced through the communication paths.

C = inlet temperature coefficients of reactivity(cent/℃).

(A+B) = reactivity decrement experienced in going to full power and flow from zero power isothermal at coolant inlet temperature.(cents)

B = power/flow coefficient (cent/100% P/F).

A = net (power-flow) reactivity decrement (cents).

A, B and C are calculated from the reactivity coefficients by;

$$A = [\frac{\partial\,\delta\rho}{\partial P}]_{(P/F,\,T_{in} constant)} = (\alpha_D + \alpha_H)\,\Delta T_f \tag{4}$$

$$B = [\frac{\partial\,\delta\rho}{\partial P/F}]_{(P,\,T_{in} constant)} = (\alpha_D + \alpha_H + \alpha_{Pb} + 2\cdot\alpha_R)\,\Delta T_c / 2 \tag{5}$$

$$C = [\frac{\partial\,\delta\rho}{\partial P}]_{(P,F\ constant)} = \alpha_D + \alpha_H + \alpha_{Pb} + \alpha_R \tag{6}$$

where ΔT_f is the core-average temperature increase across the radius of the fuel pin, and ΔT_c is the core-average coolant temperature increase from inlet to outlet. T_{in}, T_{out}, ΔT_c, ΔT_f , and T_f are taken to be 600 K, 700 K, 100 K, 120 K, and 770 K, respectively, based on simple thermohydraulic calculation.

$$\alpha_D = \frac{1}{\beta}\left[T_f \frac{dk}{dT_f}\right]\frac{1}{T_f} \tag{7}$$

$$\alpha_H = \frac{1}{\beta}\left[H \frac{dk}{dH}\right]\gamma_f \tag{8}$$

$$\alpha_{Pb} = \frac{1}{\beta}\left[\rho_{Na}\frac{dk}{d\rho_{Na}}\right]\left[\frac{1}{\rho_{Pb}}\frac{d\rho_{Pb}}{dT}\right] \tag{9}$$

$$\alpha_{R} = \frac{1}{\beta}\left[R\frac{dk}{dR}\right]\gamma_g \tag{10}$$

where γ_f is the linear thermal expansion coefficient of the fuel, ρ_{Pb} is the Pb, Pb-Bi coolant density, $[(-1/\rho_{Na})(d\rho_{Na}/dT)]$ is the volumetric expansion coefficient of Pb(= K^{-1}). R is the equivalent core radius, and γ_g is the linear expansion coefficient of the grid plates made of HT-9. For the metallic fuel, it is assumed that the fuel is in contact with the cladding even at low burnup. Therefore, the fuel expansion of the metallic fueled reactor is controlled by the cladding expansion that is , in turn, determined by the coolant temperature. Hence, the coefficient A for the metallic fuel does not contain the axial expansion coefficient.[10][11][12]

Reactivity coefficients of equations from (7) to (10) are plotted in figure 5 vs. L/D and FVF. The integral reactivity parameters A, B, and C of equation (4), (5), (6) are plotted in figure 6, and these are used to calculate the perturbed outlet temperature in transients.

LOFWS and LOHSWS are postulated as follows: [2][10][11][12]

(a) Loss of Flow Without Scram (LOFWS)

This transient is defined by a pump failure event. The flow is reduced to the level of natural circulation. P/F becomes larger than 1, and the core average temperature will rise. Then a negative reactivity will be induced and the core power will reduce. Finally, the power will be much smaller than the normal value ($P \ll 1$) and Equation (3) reduced to;

$$\frac{P}{F} = 1 + \frac{A}{B} \tag{11}$$

$$\delta T_{out} = \Delta T_c \delta\left(\frac{P}{F}\right) = \frac{A}{B}\Delta T_c \tag{12}$$

(b) Loss of Heat Sink(LOHS)

The inlet temperature of the coolant rises due to the loss of cooling by the secondary circuit. The increase in inlet-temperature causes a decrease in power. Asymptotically, the power will become zero and the outlet and inlet temperature of the coolant will be the same. Then:

$$\delta T_{in} = \frac{A+B}{C} \tag{13}$$

$$\delta T_{out} = \delta T_{in} - \Delta T_c = \Delta T_c\left[\frac{A+B}{C\,\Delta T_c} - 1\right] \tag{14}$$

Both the Pb or Pb-Bi coolant density and the radial expansion coefficients can be made more negative by designing the core to enhance axial leakage. The positive coolant density coefficients is reduced by increasing the negative leakage component through the use of the appropriate height-to-diameter ratio(L/D in PEACER) and also by use of heterogeneous layouts as shown in figure 5. The negative radial expansion coefficient (due to increasing core inter-assembly gap upon core support thermal dilation) becomes more negative as the axial leakage fraction is increased. That is to say, it is also sensitive to L/D. Thus, it would be desirable to hold L/D ratios in a range favoring axial leakage and to employ heterogeneous core layouts to further reduce the heavy liquid metal coolant density coefficients.[12]

β-eff is substituted by β^{ph} defined by the ratio of ν_d (delayed neutron fraction of neutrons per fission) to ν (average total number of neutrons per fission).[13] With the ν_d data in ref[13] and isotope reaction information about each core of ν, β^{ph} is estimated about 0.0033.

We can find out that the radial expansion coefficient is the largest among 4 reactivity coefficients. The coolant density coefficient is much lower than that of Na, and the L/D is more dominant than FVF observing the trend of reactivity coefficients.

Now let's classify the core designs as follows;

1. Reference design, case 1: L/D= 0.102, FVF= 15.8%,
2. Case 2 : L/D= 0.117, FVF= 15.8%,
3. Case 3 : L/D= 0.135, FVF= 15.8%,
4. Case 4: L/D= 0.102, FVF= 18.0%,
5. Case 5: L/D= 0.102, FVF= 20.4%,

In this quasi-static analysis, the influence of afterheat and passive heat removal were not considered. These effects will play an important role for the events leading to passive shutdown of the reactor, i.e. the LOFWS and the LOHS events. Assumed is that the afterheat will be removed by the passive heat removal system (RVACS). In that case, the in-core temperatures will be similar, because the temperatures are set to compensate for the reactivity change.[10]

The reactivity coefficients and reactivity integral parameters are tabulated with IFR and ALMR burner core design.[10] Outlet temperature for LOFWS and LOHSWS is also tabulated, for reference core design, case 1. The coolant outlet temperatures for this high leakage transmutor core design is less than for the IFR breeder design, or ALMR burner design, which is due to the smaller coolant density coefficient, as described in Table 1. As shown in reactivity coefficients and outlet temperatures in main ATWS, the conceptual design of dilute high leakage transmutor core can be self-regulated and it is easily designed to have accident-tolerance. Outlet temperatures in LOHSWS and LOFWS is below 970 K which is strong fuel-clad interaction occurs resulting in high pin failure rates, and above 398 K which is the melting point of Pb-Bi coolant.

2.5. Others

Other criteria, continuity of fuel resource and Economy are not considered in this study. Use of Th cycle can be a good option to improve the fuel continuity. Economy analysis needs more detailed and further design. In simple preliminary calculation, it was showed that SR(Support Ratio) 2.0 is sufficient for reducing the disposal site cost greatly.

3. Conclusion

In this analysis improved from previous 365 days burn cycle time case, the new burn cycle time of 330 days was taken as a basis in order to adjust with the general 3-year fuel cycle interval. Because of this correction, the fuel discharge burnup was decreased to 10 at% or below, which is 1 2 at% lower than the case of 365 days burn cycle time. Hence the reactor performance parameters such as OR of Pu, and SR became somewhat inferior than the case of 365 days cycle. The results of U-TRU-Zr fueled 365 days total burn cycle time design are as following table. Peak power density shows more dilute core design than current fast breeder reactor design. For example, IFR fast reactor has about 600 kW/L peak power density.

In above sections, it is shown that dilute and high leakage core can increase the transmutation capability related to proliferation-resistance and environment-friendliness. Quasi-static reactivity characteristics are also improved by use of metallic fuel and Pb-Bi heavy liquid metal coolant, as shown in Table 1.

References

[1] W. S. Park, T. Y. Song, Y. Kho, H. S. Shin, "Development of Transmutation Technology.", KAERI/RR-1638/95, Korea Atomic Energy Research Institute, 1995.

[2] Edited by T. D. Beynon and D. J. Dudziak, "Progress in Nuclear Energy, special issue: The Technology of the Integral Fast Reactor and its Associated Fuel Cycle", 1997.

[3] Alexander De Volpi, "Proliferation, Plutonium and Policy/Institutional and Technological Impediments to Nuclear Weapons Propagation", Pergamon Press, 1979.

[4] Walter, Alan E. and Albert B. Reynolds, "Fast Breeder Reactor," Pergamon Press, New York, NY, USA, 1981.

[5] O. M. Borisov, "Requirements to the core of Brest-type Reactors", Conference on Heavy liquid Metal Coolant for Nuclear Technology, 1998.

[6] B. F. Gromov et al., "The Problems of Technology of the Heavy Liquid Metal Coolants(Lead-Bismuth, Lead)", Proc. of Conf. on Heavy Liquid Metal Coolants in Nuclear Technology, IPPE, Obninsk, Russia, October 5-9, 1998.

[7] I. Gorynin et al., "Construction Materials for Nuclear Power Facilities Using Lead and Lead-Bismuth Coolant", Proc. of Conf. on Heavy Liquid Metal Coolants in Nuclear Technology, IPPE, Obninsk, Russia, October 5-9, 1998.

[8] B. J. Toppel, "A user's guide for the REBUS-3 fuel cycle analysis capability", 1990.

[9] W. S. Yang, etc, "Developmnet of A Three-Dimensional Depletion Analysis Code for Liquid Metal Reactors", KAERI/CM-135/96, Atomic Energy Research Institute attached to Chosun Univ., 1996.

[10] J. Bultman, "Actinide Transmutation in Nuclear Reactors", Thesis Delft University of Technology.-I11.

[11] D. C. Wade and E. K. Fujita, "Trends Versus Reactor Size of Passive Reactivity Shutdown and Control Performance", Nucl.Sci.Eng., 103, Oct. 1989, pp. 182-195

[12] D. C. Wade and Y. I. Chang, The Integral Fast Reactor(IFR) Concept: Physics of Operation and Safety, Proc.Int.Topl.Mtg. Advances in Reactor Physics Mathematics and Computation, Paris, France, April, 1987.

[13] Karl O. Ott, Robert J. Neuhold, "Introductory Nuclear Reactor Dynamics", American Nuclear Society, 1985.

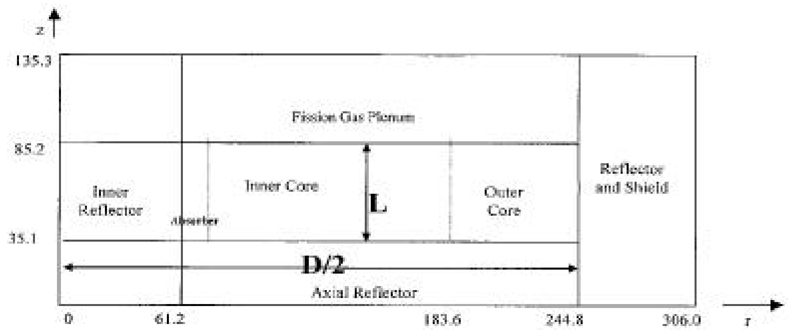


Figure 1. Core side configuration of reference design.
(radial and axial dimension from center(cm))

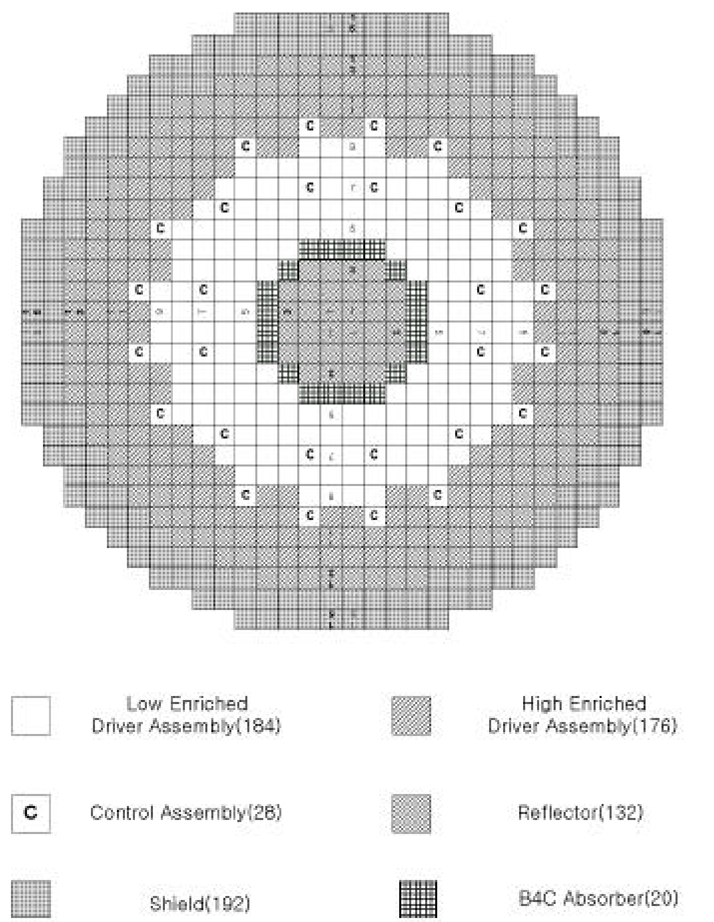
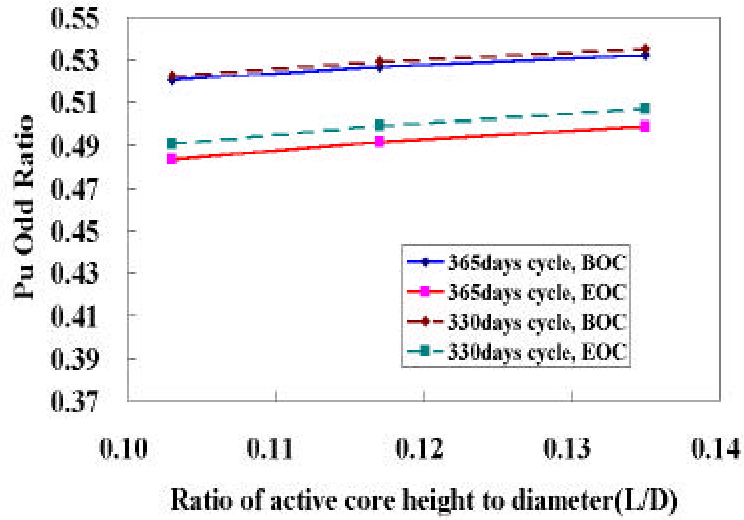
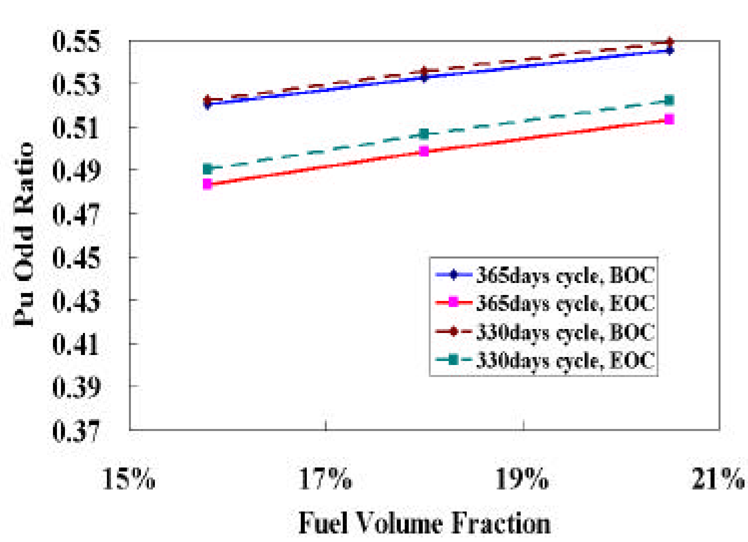


Figure 2. Cross sectional view of reference core



(a)

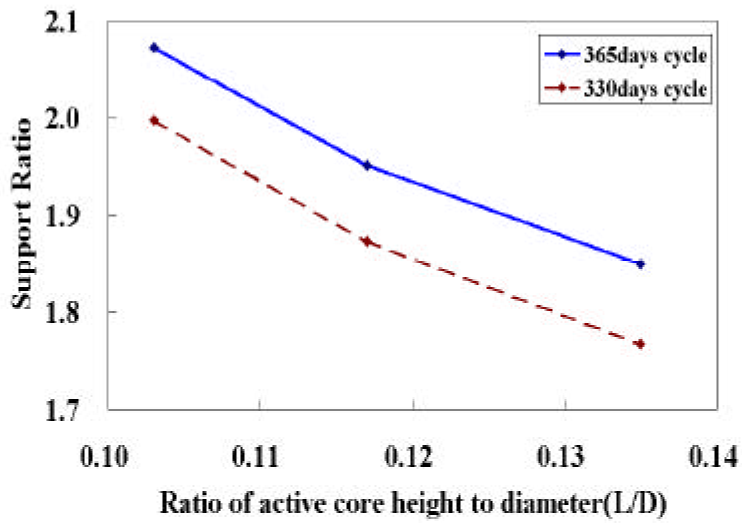


(b)

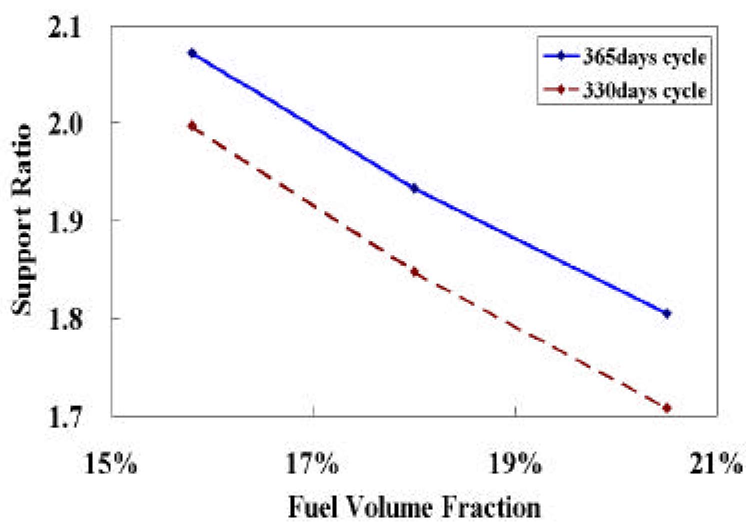
Figure 3. Pu Odd Raio comparison between burn cycle time.

(a) Pu Odd Ratio on L/D, FVF=15.8%.

(b) Pu Odd Ratio on FVF, L/D=0.102.



(a)

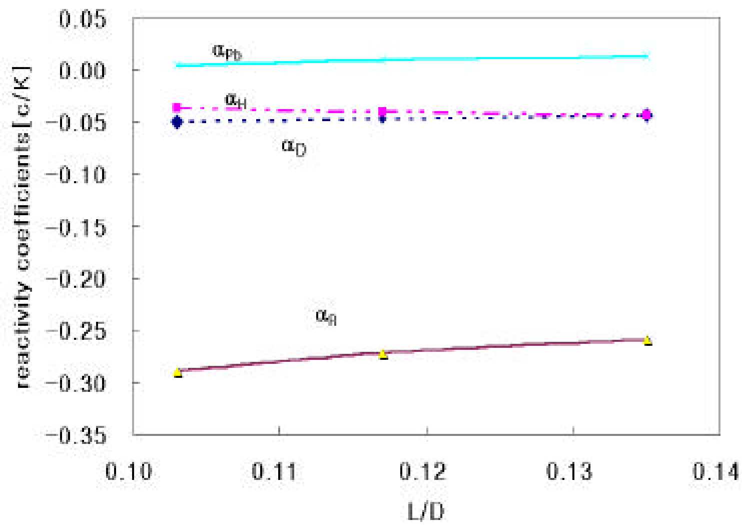


(b)

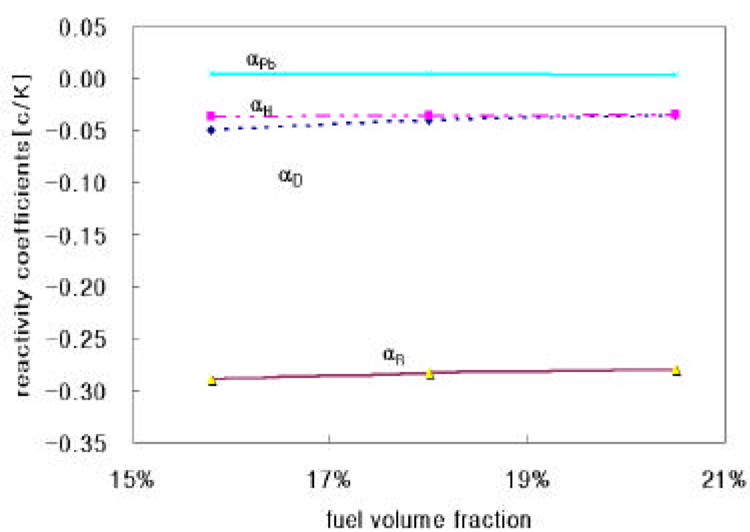
Figure 4. Support Ratio comparison between burn cycle time.

(a) Support Ratio on L/D, FVF=15.8%.

(b) Support Ratio on FVF, L/D=0.102.



(a)

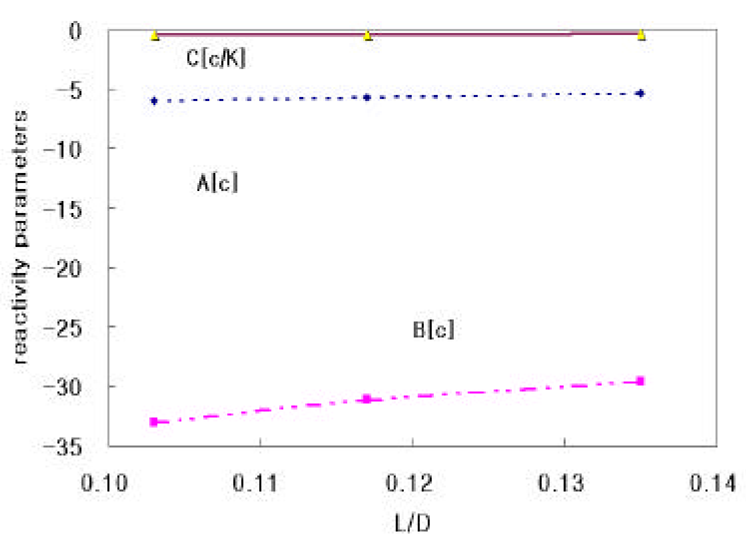


(b)

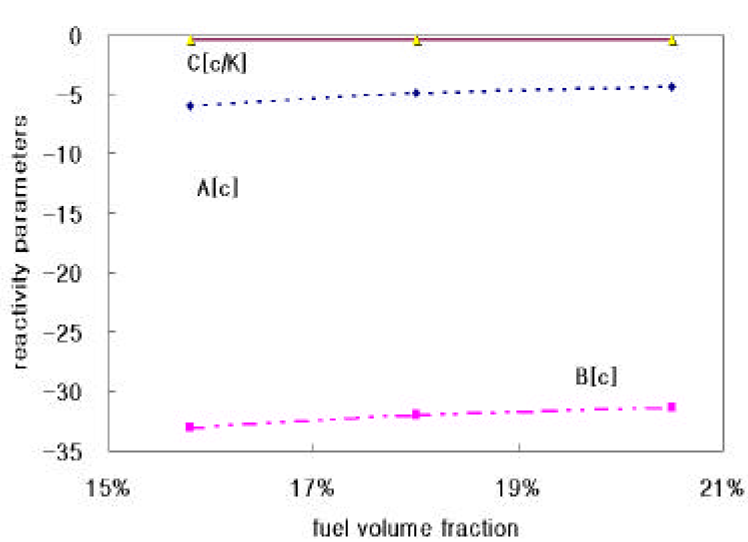
Figure 5. Reactivity coefficients of U-TRU-Zr fuel in 330 days burn cycle.

(a) Reactivity coefficients on L/D, FVF=15.8%.

(b) Reactivity coefficients on FVF, L/D=0.102.



(a)



(b)

Figure 6. Integral reactivity parameters of U-TRU-Zr fuel in 330 days burn cycle.
(a) Integral reactivity parameters(A, B, C) on L/D, FVF=15.8%.
(b) Integral reactivity parameters(A, B, C) on FVF, L/D=0.102.

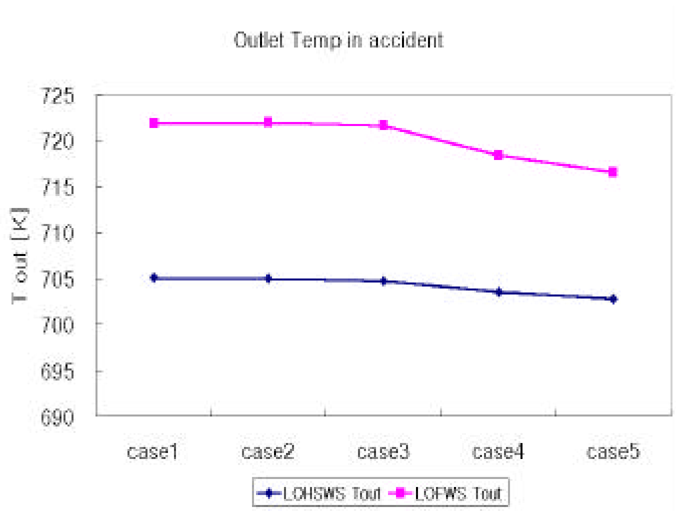


Figure 7. Outlet Temperature in LOHSWS and LOFWS.

In all case, outlet temperatures are below 970 K which is strong fuel clad interaction occurs resulting in high pin failure rates.

Parameters	Reference core (case 1 design)	ALMR metal fuel burner	IFR metal fuel fast reactor	Reference core**
T_f[K]	770	774	848	774
ΔT_f[K]	120	80	150	80
ΔT_c[K]	100	129	150	129
T_{in}[K]	600	629	623	629
T_{out}[K]	700	758	773	758
α_D[c/K]	-0.05009	-0.114	-0.10	-0.05009
α_H[c/K]	-0.03740	-0.117	-0.12	-0.03740
α_R[c/K]	-0.28857	-0.291	-0.25	-0.28857
* $\alpha_{coolant}$[c/K]	0.004704	0.0363	0.18	0.004704
A[c]	-6.011	-9.1	-15	-4.007
B[c]	-33.00	-50.1	-40.5	-42.57
C[c/K]	-0.3714	-0.486	-0.29	-0.3714
T_{out} in LOFWS[K]	722	781	829	770.1
T_{out} in LOHSWS[K]	705	751	814	770.4

Table 1. Comparison of reactivity coefficients, parameters, and outlet temperature in transients with other fast reactor design.[10]

*. Pb density coefficient for reference core, and Na for others.

**. To show the influence of reactivity coefficients only, the A, B, C for the PEACER have been calculated for the temperatures of the ALMR burner.