A SENSITIVITY STUDY ON NEUTRONIC PROPERTIES OF DUPIC FUEL

Hangbok Choi and Gyu-Hong Roh

Korea Atomic Energy Research Institute
P.O. Box 105, Yuseong
Taejon, 305-600, Korea
Tel:42-868-2793 Fax:42-868-8365 E-mail:choih@nanum.kaeri.re.kr

ABSTRACT

A sensitivity study has been done to determine the composition of DUPIC fuel from the viewpoint of neutronics fuel design. The spent PWR fuel compositions were generated and fissile contents adjusted by blending fresh uranium after mixing two spent PWR fuel assemblies. The ^{239}Pu and ^{235}U enrichments of DUPIC fuel were adjusted by controlling the amount of fresh uranium feed and the ratio of slightly enriched and depleted uranium in the feed uranium. Based on the material balance calculation, it is recommended that DUPIC fuel composition be such that spent PWR fuel utilization is more than 90%. A sensitivity study on the temperature reactivity coefficient of DUPIC fuel has shown that it is desirable to increase the ^{239}Pu and ^{235}U contents to reduce both the fuel and coolant temperature coefficients. On the other hand, refueling simulations of the DUPIC core have shown that the channel power peaking factor, which is a measure of the reactor trip margin, increases with the total fissile content. Considering these neutronic characteristics of the DUPIC fuel, it is recommended to have enrichments of 0.45 and 1.00 wt% for ^{239}Pu and ^{235}U, respectively.

I. INTRODUCTION

For direct use of spent PWR fuel in CANDU (DUPIC)[1], the DUPIC fuel should be compatible with the reactor system which was originally designed for the natural uranium fuel. Therefore, compatibility studies are being conducted on reactor physics, reactor safety, radiation physics, fuel handling and so on. As a basis for these compatibility studies, it is necessary to generate a reference DUPIC fuel composition which is most feasible from various aspects. However, because the isotopic composition of spent PWR fuel changes depending on the initial and discharge conditions of PWR fuels, preliminary studies have been done to reduce uncertainties

due to fuel composition variation by adjusting either the reactivity or fissile contents of DUPIC fuel.[2] These studies have shown that fuel composition heterogeneity can be reduced appreciably when the compositions of major fissile isotopes are fixed. Nonetheless, the effects of fuel composition on the nuclear characteristics of DUPIC fuel bundle design have not been fully studied. In this study, we have surveyed the sensitivity of the neutronics characteristics of DUPIC fuel for a wide range of fissile content variations. This study suggests a feasible value of DUPIC fuel composition and provides a data base for further studies.

II. DUPIC FUEL COMPOSITION

In this study, it was assumed that the spent fuel assemblies will be refabricated as DUPIC fuel in 2010. A total of 3600 spent PWR fuel assemblies, which include 14×14, 16×16 and 17×17 fuel assembly types, were used in this study. As previously mentioned, all actinides and fission products are reused as fuel material except the gaseous fission products which escape during the OREOX and sintering processes. Once the composition of spent PWR fuel powder is calculated, the DUPIC fuel composition is determined in two steps: mixing spent PWR fuel and blending fresh uranium.

A mixing process is done first in order to reduce the variation of isotopic composition in spent PWR fuel. Currently, the physics study recommends mixing the spent PWR fuels of high and low ^{239}Pu content together. The variation of ^{239}Pu is reduced appreciably once the mixing operation has taken place. The variation of ^{239}Pu content with a 95% confidence level decreases from 20.5% to 7.1% and 6.6% as the number of mixing assemblies increases to two and three, respectively. In addition to reducing the variation in the ^{239}Pu content, the distribution of ^{235}U is also reduced from 71.4% to 49.9% and 47.6%. Considering that ^{235}U and ^{239}Pu are the most reactive isotopes, the variation in fuel composition is reduced most effectively when two spent PWR fuel assemblies are mixed.

Though the mixing operation reduces the variation in the fuel composition, it does not provide a unique composition of any isotope in the fuel. It is therefore proposed to blend spent PWR fuel with fresh uranium to control the enrichment of ^{239}Pu and ^{235}U, which contribute most to the neutronics property of a fuel bundle. The blending of fresh uranium dilutes the ^{239}Pu content in the spent fuel and a target enrichment of ^{239}Pu can be achieved if the amount of fresh uranium is adjusted. At the same time, if the amount of ^{235}U in the fresh uranium is adjusted by mixing slightly enriched uranium (SEU) and depleted uranium (DU), the target enrichment of ^{235}U can also be satisfied. However, because of the limitation in enrichment of SEU and DU, some spent PWR fuels may not satisfy the target enrichment of ^{235}U. Therefore, a spent PWR fuel utilization factor (FAC) has been defined as the ratio of the amount of spent PWR fuel accepted for the enrichment adjustment over the total amount of candidate spent PWR fuel.

Table I. Spent PWR Fuel Utilization for Two-Assembly Mixing

^{239}Pu	^{235}U (wt%)								
(wt%)	.80	.85	.90	.95	1.00	1.05	1.10	1.15	1.20
.20	98.1	99.0	100.0	100.0	100.0	100.0	100.0	100.0	100.0
.25	95.1	96.7	97.5	98.2	99.0	99.8	100.0	100.0	100.0
.30	91.2	92.6	95.1	95.3	97.1	97.6	98.6	99.0	99.8
.35	87.9	90.8	91.4	92.6	94.8	95.2	97.0	97.3	97.8
.40	78.5	84.8	89.9	90.9	91.8	92.6	94.8	95.1	96.4
.45	66.5	75.4	82.0	87.2	*90.3	90.4	88.1	83.1	74.3
.50	51.0	57.4	62.1	58.5	55.1	45.2	38.2	26.6	14.8

* Based on 100 kg of spent PWR fuel, 90.3 kg of spent PWR fuel is used for DUPIC fuel.

III. LATTICE PROPERTIES

For the various combinations of ^{239}Pu and ^{235}U enrichments of DUPIC fuel, the lattice properties were calculated by WIMS-AECL using the 89-group ENDF/B-V-based cross-section library. In order to see the sensitivity of the fissile content, a calculation has been done for the selected values of ^{239}Pu and ^{235}U enrichment. The temperature reactivity coefficients were calculated by perturbing the lattice conditions and compared to those of the natural uranium fuel.

The fuel temperature coefficient (FTC) of natural uranium is about $-10.9\,\mu k/^{\circ}K$ under fresh conditions and gradually increases as the uranium burns and plutonium is produced. At equilibrium burnup, it becomes $0.5\,\mu k/^{\circ}K$, which is nearly zero considering the number of significant figures. For DUPIC fuel, the FTC is also very small for the fissile contents considered in this study. There is a small dependence of FTC on fissile contents such that a more negative value is obtained when the contents of ^{239}Pu and ^{235}U are increased.

The coolant temperature coefficient (CTC) is an effect of coolant density change which results in a spectrum change in the fuel region. For the natural uranium fuel, the CTC is $53.9\,\mu k/^{\circ}K$ at equilibrium burnup. For DUPIC fuel, the effect of the coolant density change on fuel bundle reactivity is smaller compared to the natural uranium fuel because of the poison material in the fuel. It is also noticed that the CTC decreases as the enrichments of ^{239}Pu and ^{235}U increase.

The moderator temperature change results in a density change and spectral shift in a fuel lattice accordingly. Because the CANDU fuel lattice is over-moderated and the moderator behaves as a thermal neutron source, the moderator temperature coefficient (MTC) changes sensitively as the fuel composition changes when the fuel burnup increases. In the case of natural uranium fuel, the MTC increases from -57.9 to $35.0\,\mu k/^{\circ}K$ as the plutonium builds up when the fuel burnup increases from fresh to equilibrium value. For DUPIC fuel, the MTC and its variation as a function of burnup are much smaller than those of natural uranium fuel because DUPIC

fuel contains plutonium from the beginning and, therefore, the effect of the plutonium build-up is relatively small.

The coolant void reactivity is positive for the standard 37-element CANDU fuel bundle. The DUPIC fuel bundle also has a positive coolant void reactivity which is even higher than that of natural uranium fuel. For DUPIC fuel, the positive coolant void reactivity is suppressed by mixing a poison material with the center fuel rod of the 43-element DUPIC fuel bundle. The amount of poison (natural dysprosium) is determined such that the prompt inverse period of DUPIC fuel at equilibrium burnup is the same as that of natural uranium fuel.[3] Because of the poisoning, the coolant void reactivity of DUPIC fuel is smaller than that of natural uranium fuel (14.0 mk) for all combinations of fissile enrichments.

In general, the temperature reactivity coefficients are smaller in DUPIC fuel, which is a positive result in the nuclear design of DUPIC fuel. The effect of fissile enrichment on the temperature reactivity coefficient is relatively small but there is a small advantage for the temperature reactivity coefficients if enrichments of both ^{239}Pu and ^{235}U are increased.

IV. CORE PERFORMANCE

Refueling simulations have been performed for 600 Full Power Days (FPD) with a 2-bundle shift refueling scheme in which two fuel bundles are loaded into a channel at a time. The maximum channel power (MCP), maximum bundle power (MBP) and channel power peaking factor (CPPF) are summarized in Table II. The refueling simulation was performed such that the MCP was maintained as low as possible by adjusting the discharge burnups of the inner and outer core in the radial zone. The MCP of the DUPIC core was close to that of the natural uranium core and even lower when the ^{239}Pu content increased.

On the other hand, the MBP of the DUPIC core is much smaller than that of the natural uranium core because the axial power distribution is flattened when highly reactive fuel bundles are located at the channel inlet under the 2-bundle shift refueling scheme. However, when the ^{235}U content is more than 1.2 wt%, the MBP of the DUPIC core is more than that of the natural uranium core because the bundle power is peaked too much near the channel inlet.

The CPPF is defined as the ratio of instantaneous channel power over the time-average channel power. The CPPF is directly used to determine the trip margin of the regional overpower protection system of a CANDU reactor. For natural uranium fuel core, the CPPF is around 1.064 based the 600 FPD refueling simulation, which is consistent with the typical value between 1.06 and 1.10 for the CANDU physics design. In the case of DUPIC core, the CPPF increases as the total fissile content increases because of the large perturbation in channel power when the refueling operation is performed. As seen in Table II, the CPPF can be kept below 1.065 when the ^{239}Pu

and ^{235}U contents are less than 0.5 and 1.0 wt%, respectively. However, it is expected that there will be a little room to have a slightly higher CPPF of the DUPIC core, considering that the MCP and MBP of the DUPIC core are less than those of the natural uranium core.

Table II. DUPIC Core Performance Parameters by 600 FPD Refueling Simulation

^{239}Pu (wt%)	^{235}U (wt%)				
	0.8	0.9	1.0	1.1	1.2
0.2	6832 840 1.051	6845 829 1.059	6867 810 1.065	6893 813 1.069	6899 847 1.077
0.3	6818 823 1.055	6833 807 1.062	6864 800 1.062	6871 824 1.070	6876 853 1.078
0.4	6815 816 1.059	6821 797 1.062	6849 801 1.065	6849 826 1.069	6845 854 1.077
0.5	6809 819 1.061	6811 797 1.063	6824 801 1.065	6825 828 1.071	6810 866 1.081
Natural Uranium	6859 Maximum channel power (kW) 854 Maximum bundle power (kW) 1.064 Channel power peaking factor				

V. CONCLUSION

The DUPIC fuel composition has been generated for various combinations of ^{239}Pu and ^{235}U enrichments. Depending on the target enrichments of ^{239}Pu and ^{235}U, not all the spent PWR fuel can be reused as DUPIC fuel. This is believed to be mostly due to the PWR fuel used for initial core loading, which has a quite different enrichment. For the spent PWR fuel used in this study, about 10% of it has an enrichment of less than 2.0 wt%. If we take into account these spent PWR fuels, it is recommended that the DUPIC fuel cycle should consume at least 90% of the spent PWR fuels. The spent PWR fuel utilization factors greater than 90% are shaded in Table I.

Lattice properties, such as the temperature coefficient, purity coefficient, shutdown reactivity, coolant void reactivity, etc. were calculated and compared to those of natural uranium fuel. Based on the parametric study, it can be concluded that:

- The effect of enrichment on the lattice property of DUPIC fuel is relatively small.
- For the major reactivity coefficients like fuel temperature, coolant temperature and moderator temperature coefficients, DUPIC fuel displays a slightly better performance over natural uranium fuel.
- The lattice property of DUPIC fuel could be slightly improved if the enrichment of ^{239}Pu and ^{235}U is increased.

The performance parameters of the DUPIC core, such as maximum channel power, maximum bundle power and channel power peaking factor, were obtained from the refueling simulation. The results have shown that:

- The maximum channel and bundle powers of the DUPIC core can be kept smaller than those of the natural uranium core.
- The total fissile content of DUPIC fuel should be limited in order to reserve a trip margin which is determined by the channel power peaking factor and other parameters.

This study has shown the sensitivities of fuel composition on the neutronics property of DUPIC fuel. It is difficult to choose the optimum value of DUPIC fuel composition based only on the physics calculation because the thermalhydraulics property could worsen while the fuel cycle economics improves when fissile content is increased. Nonetheless, it is possible now to suggest the feasible range of DUPIC fuel composition as far as neutronics design is concerned. Considering the fuel utilization, temperature reactivity coefficient and channel power peaking factor, it is recommended that the ^{239}Pu and ^{235}U enrichments be 0.45 and 1.00 wt%, respectively. It will be necessary to do further studies on the thermalhydraulic and economic characteristics of such a fuel.

REFERENCES

1. J.S. Lee et al., "Research and Development Program of KAERI for DUPIC (Direct Use of Spent PWR Fuel in CANDU Reactors)", Int. Conf. and Tech. Exhibition on Future Nuclear System: Emerging Fuel Cycles and Waste Disposal Options, GLOBAL'93, Seattle, USA, 1993.

2. H.B. Choi, B.W. Rhee, H.S. Park, and P.S.W. Chan, "Fuel Composition Heterogeneity Effect for Recycling of Spent PWR Fuel in CANDU", Int. Conf. on Evaluation of Emerging Nuclear Fuel Cycle Systems, GLOBAL'95, Versailles, France, Sept.11-14, 1995.

3. Hangbok Choi, Bo W. Rhee, Hyunsoo Park, "Burnable Dysprosium for Low Void Reactivity Fuel", Proc. of Am. Nucl. Soc. Topical Mtg - Advances in Nuclear Fuel Management II, Myrtle Beach, March 23-26, 1997.