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The Conceptual Design of a Hybrid UO_2-MOX Pellet

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Abstract

The conventional MOX fuel shows adverse controllability in view of its neutronic characteristics such as decreased soluble boron worth and effective delayed-neutron fraction compared to the UO_2 fuel. In order to mitigate these disadvantages, we devised a new concept of the hybrid UO_2-MOX fuel pellet with dual structure such that its outer annular section contains UO_2 fuel and its inner cylindrical bar contains MOX fuel. The lattice physics code HELIOS was used to evaluate the neutronic characteristics of three different types of fuel pellets ; UO_2 fuel pellet, MOX fuel pellet, and hybrid UO_2-MOX fuel pellet. Results show that the hybrid UO2-MOX fuel pellet generally has intermediate neutronic tendency between UO_2 fuel and MOX fuel which could diminish the problems arising from the use of the conventional MOX fuel.

1. Introduction

Many countries, which have chosen the nuclear power development, have been vigorously driving the policy to recover plutonium from the spent fuel and use it in the form of mixed-oxide(MOX) fuel. Today there is widespread acceptance of using MOX fuel in the thermal power reactors.

The neutronic properties of MOX fuel are much different from those of UO_2 fuel.

First, because the neutrons produced per fission in MOX fuel are more than those produced in UO_2 fuel and because Pu fertile isotopes (238, 240, 242) have very large capture cross section, MOX fuel shows higher conversion and slower depletion than the typical UO_2 fuel.

Second, the strong self-shielding effect of Pu-240 with a large resonance absorption cross section induce a neutron spectrum hardening, which reduces the neutron flux level in thermal energy range. The spectrum hardening associated with MOX fuel reduces the control rod worth and the boron worth which requires the more control rods and the higher capacity of boron injection system.

In addition, the effective delayed-neutron fraction of Pu-239 is about one third of that of U-235. It causes the LWRs loaded with MOX fuel to reduce the reactor safety margin and to limit a variety of reactor operations.

Although the use of MOX fuel has evolved gradually over the past several decades to the point where it can be freely used in the full core, the regulatory bodies of many countries allow the use of MOX fuel assemblies up to approximately one third of total assemblies in the core. The conventional reactor loaded with both UO_2 and MOX fuel assemblies has some problems as the following ; (i) the power peak in the periphery of the MOX fuel assembly adjacent to the UO_2 assembly, (ii) the simultaneous treatment of MOX fuel with different Pu contents for assembly zoning and (iii) the complicated fuel management.

To solve these problems, a hybrid UO_2-MOX fuel pellet - a cylindrical inner pellet containing MOX fuel and an annular outer pellet containing UO_2 fuel - was devised.

This paper presents the rationale for the use of newly devised UO_2-MOX fuel pellet and the calculated results for nuclear parameters improved by using it. Then, this paper presents also new strategy different from conventional one on partial MOX fuel loading.

2. Hybrid UO_2-MOX Fuel Pellet

As shown in Fig. 1, new fuel pellet has two sections, a cylindrical inner pellet containing MOX fuel and an annular outer pellet containg UO_2 fuel. A hollow UO_2 pellet having a hole along its central axis is spitted with a long bar of MOX, which is inserted and sealed in a cladding tube to form the fuel rod.

Substituting UO_2 fuel for MOX fuel within the outer region of the pellet can obtain a softer neutron spectrum and the larger delayed-neutron fraction than using the conventional MOX fuel pellet, maintaing the property of high conversion.

UO_2 fuel within outer pellet mitigates the strong self-shielding effect of Pu-240 with a large resonance absorption cross section. Therefore, the thermal flux distribiution in the new fuel pellet is flatter than that in MOX pellet.

A neutron spectrum softening is helpful for increasing the boron worth and the control rod worth which are closely related to the shutdown margin.

Because U-235 with relatively large delayed-neutron fraction is contributed in the fission reaction, this hybrid pellet produces the more delayed-neutrons that are vital for the effective control of the fission chain reaction.

3. Calculations

We have used HELIOS (developed by SCANDPOWER) as a lattice physics code to evaluate the nutronic characteristics of the hybrid UO_2-MOX fuel pellet

HELIOS is a two-dimensional transport theory program for fuel depletion and gamma-flux calculations based on the method of current coupling of the space elements, which are internally treated by the method of collision probabilities.

HELIOS can analyze any type of fuel pellet and assembly. The geometry of the system to be calculated, typically a fuel assembly, can be built-up arbitrarily of heterogeneous space element. HELIOS can use any of 34, 89 and 190 group library which has been generated from ENDF/B-IV. In order to treat the complicated resonance structure of plutonium isotopes adequately, a 89-group library was seleted for this study.

In order to compare the neutronic characteristics of the hybrid UO_2-MOX fuel pellet against those of the UO_2 fuel pellet and MOX fuel pellet, three types of pin cells shown in Fig. 2 were evaluated as the calculation problem.

- Type1 consists of UO_2 (3.14% of U235) fuel only
- Type2 consists of MOX (6.0 % of Pu contents) fuel only
- Type3 consists of UO_2 fuel and MOX fuel with dual structure

The dimension of a pin-cell refered to WH V5H fuel rod is presented in Talble I. The pin cell was idealized in some respects to simplify problem and to keep the calculation consistent : (i) the cladding is taken to be natural zirconium, (ii) the cladding gap is homogenized with the cladding to form a single region, (iii) structural components, fission products, and actinides have been omitted, (iv) the reflecting boundary condition is used.

The following parameters were calculated as functions of burnup on three types of pin cells :

- k-infinity
- the ratio of the fast flux to the thermal flux
- moderator temperature coefficient (calculated at 540 °K and 580 °K)
- boron worth (calculated at 750ppm and 1000ppm)
- effective delayed-neutron fraction

4. Results and Discussion

The results shown in Fig. 3 through 5 and TABLE II through IV are obtained from the HELLIOS evaluations. Fig. 3 gives the k-inf for each of the three types at identical burnup conditions. Although two thirds of fuel pellet consists of UO_2 fuel, the depletion characteristics of the hybrid UO_2-MOX fuel pellet is rather similar to that of MOX pellet than that of UO_2 pellet. This effect is caused by the larger convertion of the U-238 to Pu-239 as shown in Fig. 4. The depletion characteristics of the hybrid UO_2-MOX fuel pellet is helpful for increasing the operation cycle and reducing the boron concentration to control excessive reactivity at BOC.

The burnup-dependent MTCs for each of the three types are given in TABLE III. The MTC of new fuel pellet is similar to that of MOX pellet.

As shown in TABLE IV, the boron worth (or conrol rod worth) of the hybrid UO_2-MOX fuel rod is larger than that of MOX fuel rod. It reduces the reguired capacity of the boron control system during reactor operations, cold to hot reactivity swing and accidents.

Fig. 5 illustrates the effective delayed-neutron fraction for each of the three types. The hybrid UO_2-MOX fuel rod has a larger delayed-neutron fraction than MOX fuel rod, because U-235 with relatively large delayed-neutron fraction is contributed in the fission reaction. The increased effective delayed-neutron fraction improves the rod ejection accident margin and the flexibility of reactor control.

5. Conclusions

The neutronic characteristics of the hybrid UO_2-MOX fuel pellet was evaluated. The results show some good features as described in the previous section.

If the core is fully loaded with the hybrid UO_2-MOX fuel assemblies without zoning, the problems caused from the partial MOX fuel assembly loading, - the power peak in the periphery of the MOX assembly aadjacent to the UO_2 assembly, the simultanious treatment of MOX fuel with different Pu contents for assembly zoning and the complicated fuel management - could be solved.

In the future, we are to evaluate the nutronic charateristics of fuel assembly consisting of these hybrid UO_2-MOX fuel pellets through the assembliwise color-set calculation.

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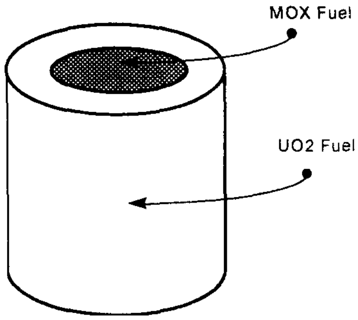

Fig. 1. The configuration of a hybrid UO2-MOX fuel pellet

TABLE I
Dimensions for calculations (cm)

Outer radius of pellet	0.4096
Outer radius of cladding	0.4750
Pin cell pitch	1.2598
Clad thickness	0.71206

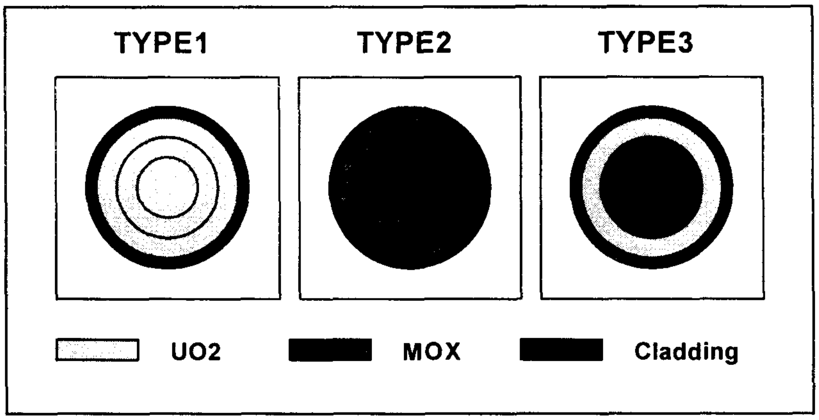

Fig. 2. Pin-cell geometries

TABLE II
The burnup-dependent ratio of fast flux to thermal flux for the three pin types

	TYPE1	TYPE2	TYPE3
BOC	7.331	33.657	14.753
MOC	9.254	31.567	15.069
EOC	9.775	28.619	14.416

TABLE III
The burnup-dependent MTC(pcm/℃) at 0ppm

	TYPE1	TYPE2	TYPE3
BOC	-25.501	-38.212	-37.181
MOC	-36.900	-42.317	-42.630
EOC	-44.953	-48.688	-49.165

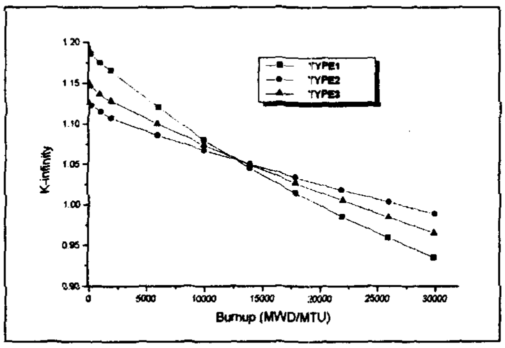

Fig. 3. The burnup-dependent K-infinity for the three pin types

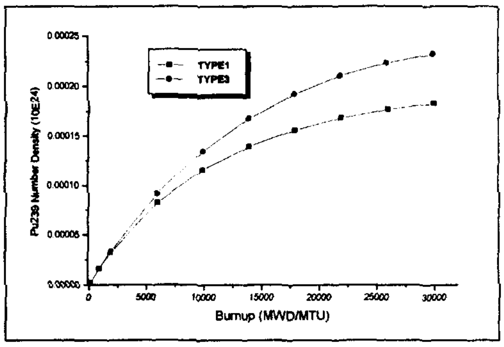

Fig. 4. The burnup-dependent number density of Pu-239

TABLE IV
The boron worth (pcm/ppm) vs burnup

	TYPE1	TYPE2	TYPE3
BOC	-6.756	-2.422	-4.157
MOC	-6.586	-2.697	-4.596
EOC	-7.108	-3.044	-5.167

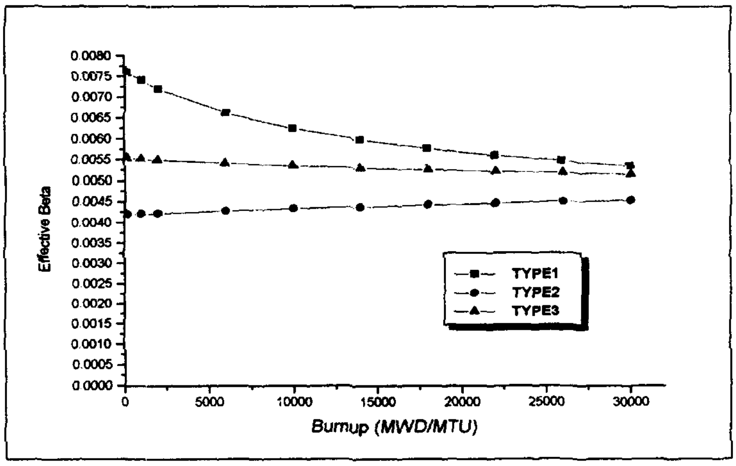

Fig. 5. The effective delayed-neutron fraction vs burnup