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Burnup Effects of MOX Fuel Pincells in PWR

- OECD/NEA Burnup Credit Benchmark Analysis -

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Abstract

The burnup effects were analyzed for various cases of MOX fuel pincells of fresh and irradiated fuels by using the HELIOS, MCNP-4/B, CRX and CDP computer codes. The investigated parameters were burnup, cooling time and combinations of nuclides in the fuel region. The fuel compositions for each case were provided by BNFL (British Nuclear Fuel Limited) as a part of the problem specification so that the results could be focused on the calculation of the neutron multiplication factor. The results of the analysis show that the largest saving effect of the neutron multiplication factor due to burnup credit is 30 %. This is mainly due to the consideration of actinides and fission products in the criticality analysis.

I. Introduction

The nature of the burnup credit problem requires the capacity to calculate both spent fuel composition and reactivity. Since 1991, the criticality working group of the NEANSC (formerly the NEACRP) has been investigating the methods and data associated with the calculation of burnup credit in criticality safety assessments. In the Phases I to III benchmark exercises, consideration has been given to the uranium oxide fuels in both PWRs and BWRs. The next challenge (denoted as Phase IV) for the burnup credit analysis lies in its application to the mixed oxide (MOX) fuels.

In this paper, the Phase-IV burnup credit criticality benchmarks established by the OECD/NEA Burnup Credit Criticality Benchmark Group[1] were analyzed by four computer codes (HELIOS, MCNP-4/B, CRX and CDP).

Experiences from the earlier benchmark exercises of OECD/NEA/NSC have shown that the first step in any benchmark programme must not be overly ambiguous to succeed in establishing some common ground among the OECD/NEA/NSC participants. In recognition of this, it was proposed that the initial MOX burnup credit benchmark should be centered upon a simplified MOX fuel configuration.

The fuel compositions have been derived by the benchmark co-ordinators of BNFL[2)] with the WIMS7 reactor lattice code. The nuclear data was provided by the 172-group WIMS′ 1996′ nuclear dataset, which is based upon JEF2.2 evaluations. These calculations used a simplified MOX only representation of the core, and irradiated the fuel in a single cycle at a power of 35,000 MWD/HMT, with a constant boron concentration of 500 ppm in the core coolant[2)].

II. Computational Methods

1. Case Numbers and Parameters

The infinite multiplication factor is required for a total of 63 cases that cover various combinations of initial MOX fuel composition, burnup, cooling time and spent fuel representation (i.e., "actinide" or "fission products"). The selected parameters and case numbers are given in Table 1. In the context of this benchmark exercise, the terms "major actinides", "all actinides", and "fission products" are considered as follows :

	Nuclides
Major Actinides (12)	· U-234, 235, 236, 238, Pu-238, 239, 240, 241, 242 · Np-237, Am-241, 243
All Actinides (16)	· Major Actinides + (Cm-242, 243, 244, 245)
Fission Products (15)	· Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133 · Nd-143, 145, Sm-147, 149, 150, 151, 152 · Eu-153, Gd-155

2. Compositions and Geometry Data

Three different MOX fuels were chosen to represent the range of potential interest within the group for MOX fuels. Therefore, the MOX fuels were categorized by 3 groups as follows ;

i) A reference MOX fuel case appropriate to a typical plutonium vector for material derived from the reprocessing of thermal reactor UO_2 fuels, often referred to as "first generation" MOX ("MOX-Case-A")

ii) A MOX fuel case appropriate to the disposition of weapons plutonium in MOX, ("MOX-Case-B").

iii) A MOX fuel case appropriate to future MOX fuels that might be produced using the plutonium recovered from the reprocessing of irradiated MOX, referred to as the "later generation" of MOX fuel from a plutonium recycling strategy ("MOX-Case-C").

The plutonium isotopic compositions for these MOX fuels are given in Table 2. In all cases, the uranium oxide component of the MOX is assumed to be depleted to a content of 0.25 w/o U-235/U, which is typical of current MOX fuel fabrication. The uranium isotopic composition is shown in Table 3. The initial MOX fuel enrichments for the three MOX cases (A~C) are shown in Table 4. The geometry for this paper is an infinite PWR fuel cell lattice as shown in Fig. 1.

3. Computer Codes

We used the HELIOS, MCNP-4/B, CRX and CDP computer codes for the calculations of infinite multiplication factor. The HELIOS computer code is a multi-group (ENDF/B-VI) two-dimensional transport theory program for fuel burnup and gamma-flux calculations[3].

The MCNP-4/B[4] computer code is a general-purpose, continuous-energy (ENDF/B-VI), generalized geometry, coupled neutron-photon-electron Monte Carlo transport calculation, including the capability to calculate eigenvalues for critical systems. Also, this code has ten statistical checks that provide a meaningful measure of false convergence. The MCNP-4/B allows one to calculate the effect of a small perturbation in a problem. This feature estimates the tally differences due to changes in nuclear cross-section data, material density, and material composition[5].

The CRX code[6,7] is a multi-group transport theory code based on the method of characteristics for heterogeneous cell and assembly calculations. Since the method of characteristics uses the characteristic form of the Boltzmann transport equation, CRX has no practical limitations on the geometry of the problem and on the heterogeneous calculations.

The CDP code[8,9] has been recently developed at KAIST to reduce the computational burdens of the method of characteristics. The code uses a new transport theory method called as “Method of Characteristic Direction Probabilities” and can perform the heterogeneous lattice calculation. In this paper, the multi-group (34 groups) cross sections of HELIOS are used both in CRX and CDP calculations.

III. Numerical Analysis

1. Comparison of MCNP, HELIOS, CRX and CDP

The standard deviations of MCNP-4/B and the relative discrepancies (%) of HELIOS(89) with respect to MCNP-4/B are given in Fig. 2. The results indicate that the relative discrepancy of HELIOS is very small. The infinite multiplication factors of HELIOS, CRX, and CDP are compared in Table 5. In these calculations, HELIOS used the transport corrected multi-group cross sections to approximate the linearly anisotropic scattering, and CRX and CDP performed calculations both with isotropic scattering only and with the exact treatment of linearly anisotropic scattering. The results show that CDP and CRX with linearly anisotropic scattering cross sections give quite close results to those of HELIOS (34 groups). The burnup credit saving effect (%) calculated by MCNP, HELIOS, CRX and CDP are given in Table 6. The maximum burnup credit saving effect is nearly 30% by these codes.

2. MOX Case-A

In Fig. 3, the maximum values of the infinite multiplication factor for MCNP-4/B and HELIOS are 1.29869 and 1.30368 at the 1st case number, respectively. On the other hand, the minimum values of the infinite multiplication factor for NCMP-4/B and HELIOS are 1.00257 and 1.00834 at the 13th case number, respectively. From this, we can get the burnup credit saving effect of 23 % by considering both major actinides and 15 major fission products for this case.

3. MOX Case-B

In Fig. 3, the maximum values of the infinite multiplication factor for MCNP-4/B and HELIOS are 1.41117 and 1.41841 at the 20th case number, respectively. On the other hand, the minimum values of the infinite multiplication factor for NCMP-4/B and HELIOS are 0.98535 and 0.99362 at the 32th case number, respectively. From this, we can get the burnup credit saving effect of 30 % by considering both major actinides and 15 major fission products for this case.

4. MOX Case-C

In Fig. 3, the maximum values of the infinite multiplication factor for MCNP-4/B and HELIOS are 1.19444 and 1.19862 at the 39th case number, respectively. On the other hand, the minimum values of the infinitene multiplication factor for NCMP-4/B and HELIOS are 0.97283 and 0.97638 at the 54th case number, respectively. From this, we can get the burnup credit saving effect of 30 % by considering both major actinides and 15 major fission products for this case.

IV. Conclusion

In this paper, the burnup effects of PWR fuel pincell reactivity for fresh and irradiated MOX fuels were analyzed by using the HELIOS, MCNP-4/B, CRX and CDP computer codes. The investigated parameters were burnup, cooling time and combinations of nuclides in the fuel region. Overall, the most interesting result in this analysis is that the largest saving effect of neutron multiplication factor due to burnup credit is 30%. This is mainly due to the consideration of actinides and fission products in the criticality analysis.

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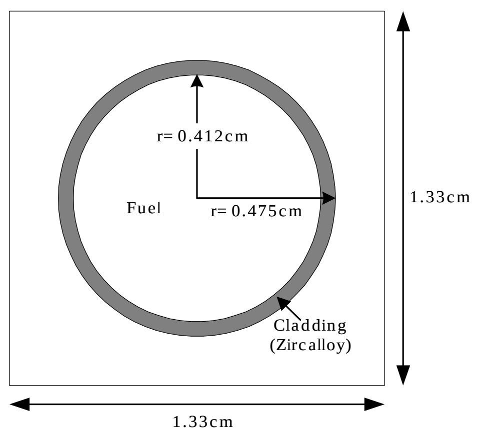

Fig. 1 Geometry of MOX Fuel Pincell for PWR

Table 1. Specified Calculations

MOX Case	Cooling Time (years)	Fission Products Modeled	Actinides Modeled	Burnup (GWd/teHM)			
				Fresh	20	40	60
MOX-A	1	Yes	Major	Case 1	Case 2	Case 3	Case 4
		No	Major		Case 5	Case 6	Case 7
			All		Case 8	Case 9	Case 10
	5	Yes	Major		Case 11	Case 12	Case 13
		No	Major		Case 14	Case 15	Case 16
			All		Case 17	Case 18	Case 19
MOX-B	1	Yes	Major	Case 20	Case 21	Case 22	Case 23
		No	Major		Case 24	Case 25	Case 26
			All		Case 27	Case 28	Case 29
	5	Yes	Major		Case 30	Case 31	Case 32
		No	Major		Case 33	Case 34	Case 35
			All		Case 36	Case 37	Case 38
MOX-C	1	Yes	Major	Case 39	Case 40	Case 41	Case 42
			All		Case 43	Case 44	Case 45
		No	Major		Case 46	Case 47	Case 48
			All		Case 49	Case 50	Case 51
	5	Yes	Major		Case 52	Case 53	Case 54
			All		Case 55	Case 56	Case 57
		No	Major		Case 58	Case 59	Case 60
			All		Case 61	Case 62	Case 63

Table 2. Plutonium Isotopic Compositions in Fresh MOX Fuel

Nuclides	Isotopic Compositions in Pu_{total} (w/o)		
	MOX Case A	MOX Case B	MOX Case C
Pu-238	1.8	0.05	4.0
Pu-239	59.0	93.6	36.0
Pu-240	23.0	6.0	28.0
Pu-241	12.2	0.3	12.0
Pu-242	4.0	0.05	20.0

Table 3. Uranium Isotopic Compositions in Fresh MOX Fuel

Nuclides	w/o in U_{total}
U-234	0.00119
U-235	0.25
U-238	99.74881

Table 4. Initial MOX Fuel Enrichments

MOX Fuel Type	MOX Fuel Plutonium Contents w/o Pu_{total}/[U+Pu]	MOX Fuel Enrichment w/o $Pu_{fissile}$/[U+Pu]
MOX-A	5.6	3.987
MOX-B	4.0	3.756
MOX-C	8.0	3.84

Table 5. K_{inf} for cases tested with CDP, CRX, and HELIOS

Fuel type (case number)	CDP		CRX		HELIOS	
	isotropic	linearly anisotropic	isotropic	linearly anisotropic	34 groups	89 groups
MOX-A(1)	1.30567(0.22[a])	1.30404(0.10)	1.30716(0.34)	1.30509(0.18)	1.30272	1.30368
MOX-B(20)	1.42293(0.08)	1.42194(0.01)	1.42449(0.19)	1.42308(0.09)	1.42174	1.41841
MOX-C(39)	1.19615(0.09)	1.19412(-0.07)	1.19750(0.21)	1.19508(0.004)	1.19503	1.19862
MOX-A(13)	1.00812(0.15)	1.00699(0.047)	1.00891(0.23)	1.00780(0.13)	1.00651	1.00834
MOX-B(32)	0.99460(0.16)	0.99386(0.09)	0.99538(0.24)	0.99466(0.17)	0.99296	0.99362
MOX-C(54)	0.97404(0.17)	0.97241(0.005)	0.97479(0.25)	0.97319(0.085)	0.97236	0.97638

[a]Relative discrepancy (%) from the result of HELIOS (34 groups).

Table 6. Burnup credit (%) for cases tested with several codes

Fuel type	CDP		CRX		HELIOS		MCNP
	isotropic	linearly anisotropic	isotropic	linearly anisotropic	34 groups	89 groups	
MOX-A	22.79	22.78	22.81	22.78	22.74	22.65	22.80
MOX-B	30.10	30.11	30.12	30.11	30.16	29.95	30.17
MOX-C	18.56	18.56	18.59	18.56	18.63	18.54	18.55

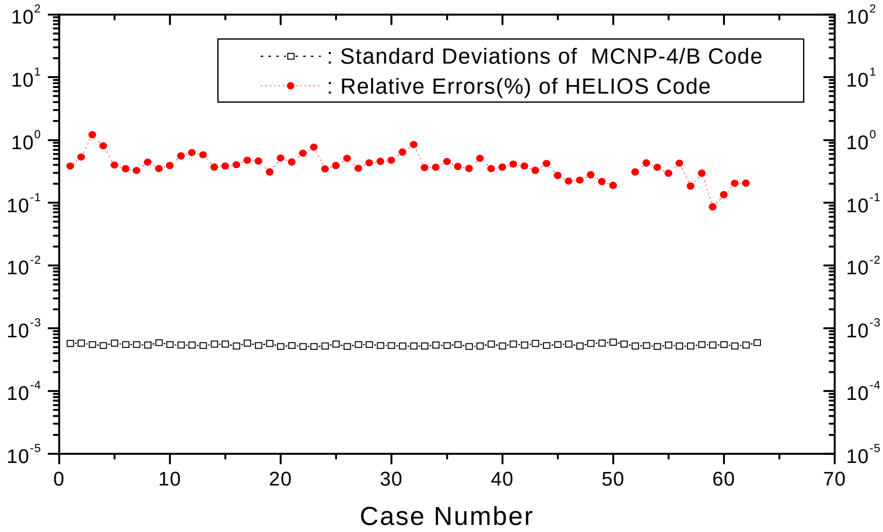

Fig. 2 Standard Deviation and Relative Errors(%) of MCNP-4/B and HELIOS Codes, Respectively.

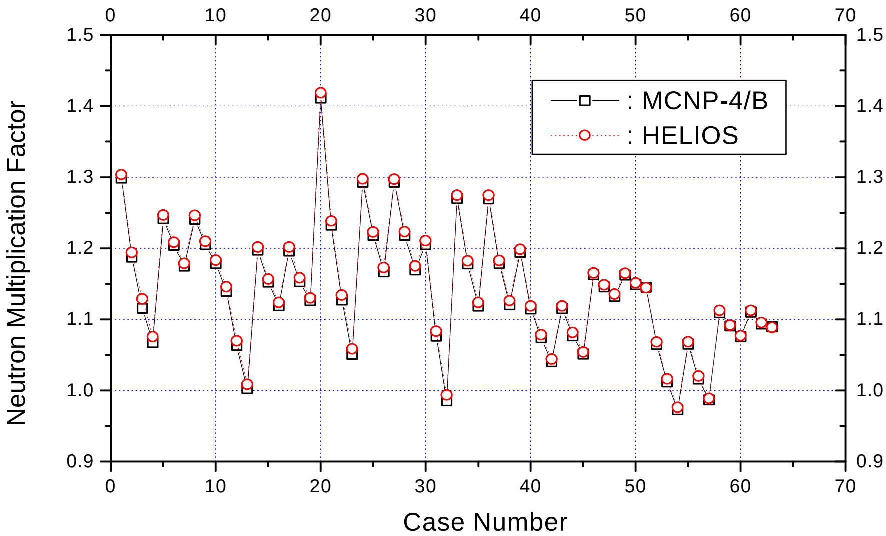

Fig. 3 Neutron Multiplication Factor of MOX Fuel Pincell for PWR By Using MCNP-4/B and HELIOS Codes.