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《**Original**》

An Effective Multiplication Factor Calculation of Uniform Lattices of UO_2-PuO_2 Fueled System

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UO_2-PuO_2 노심에서의 유효증배계수계산

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Abstract

A theoretical basis for analysis of plutonium-bearing fuel in a thermal nuclear power reactor has been established. The analysis of UO_2-PuO_2 fueled, light water moderated uniform lattice experiments has been performed. A unit cell program, KARATE, which is based on the theoretical models of GAM and THERMOS with some modifications, has been developed to generate a few-group cross-sections. These cross-sections are subsequently used in the diffusion theory code, KIDD, to compare the calculated values of the effective multiplication factor with the measured. The average value of the effective multiplication factor for 41 selected critical experiments is estimated to be 0.9997 with standard deviation of 0.43%. This illustrates the fact that KARATE/KIDD system can be effectively used for the analysis of uniform lattices of UO_2-PuO_2 fuels.

요 약

열중성자로에 플루토늄의 재순환이 실현될 것에 대비하여 이의 해석을 위한 이론적인 기초를 수립하였다. 특히 상업적인 목적으로 플루토늄 재순환이 이루어질 때를 위하여 UO_2-PuO_2를 핵연료로 하는 균일격자 임계실험로에 대한 분석을 수행하였다. 이 목적을 위하여 GAM과 THERMOS에 이론적인 기초를 두고 UO_2-PuO_2계통의 해석에 적합한 이론을 가미하여 KARATE라는 이름의 단일세포격자 코드를 개발하였다. KARATE에서 산출한 소수군 중성자 단면적과 확산이론 계산코드인 KIDD를 이용하여 임계실험로에 대한 유효증배계수를 계산하였다. 우리가 선정한 41개의 임계실험로에 대하여

유효증배계수의 평균치는 0.9997, 그 표준편차는 0.43%이었다. 이러한 결과는 KARATE/KIDD 전산체계가 UO_2-PuO_2를 핵연료로 하는 계통의 핵적 특성분석에 아주 유효하게 사용될 수 있다는 사실을 밝혀 준다.

I. Introduction

The economic benefit of large-scale plutonium recycling in thermal nuclear power reactors is well recognized. To begin with, however, an appropriate design method for plutonium-fueled reactors should be established in due course a priori so as to make this application into being. The existing and available design methods which have been normalized for uranium-fueled cores would not be valid for plutonium-fueled cores.

The neutronic analysis of light water reactors fueled with plutonium-enriched rods is much more complicated than that of light water reactors fueled with uranium-enriched rods. The phenomena of importance for plutonium systems analysis are as follows:

1. The plutonium isotopes have significant resonance absorption in thermal energy range.
2. In mixed oxide fuels the overlap of resonances of uranium isotopes and plutonium isotopes causes a reduction in the resonance integrals and thus, causes an increase in the effective multiplication factor.
3. The use of mechanical blending in the manufacture of mixed oxide fuels results in a structural heterogeneity effect due to self-shielding of the PuO_2 particles relative to the UO_2 matrix. This effect causes a reduction in the effective multiplication factor for practical lattices.

An analytical method for uniform lattices of mixed oxide fuel rods has been presented in this work. Since the mixed oxide fuels draw considerable commercial interest, our effort is concentrated on the lattices of mixed oxide fuel rods. We have analyzed forty-one critical and approach-to-critical experiments conducted in Westinghouse[1] and Pacific Northwest Laboratories.[2] These experiments, all being moderated by light water, include great variety in uranium-plutonium contents, plutonium isotope concentrations, lattice pitch, lattice type, and extent of moderator poisoning. An analysis is also made for four critical assemblies containing UO_2 as fuel. In this paper the results of a theoretical analysis of mixed oxide, light water-moderated critical experiments are presented.

II. Analytical Methods

The selection of experiments for this analysis has been limited to those critical assemblies which are normally considered uniform. The results of measurements of the critical size for uniform arrays of fuel rods in a light water moderated lattices form the base on which the calculational methods are tested. A newly-developed computer code, KARATE,[3] is utilized to generate a few group cross-sections, which are subsequently used in the diffusion theory code, KIDD,[4] to compare the calculated values of the effective multiplication factor with the measured values.

The neutron distribution in space and energy within a unit lattice cell is computed using the multigroup transport theory code KARATE, which computes the spectrum-averaged multigroup cross sections and then collapses them into two to five broad group values for subsequent use in the reactor calculation. Exact theoretical methods and analysis procedures are not generally used in reactor design because of an

excessive amount of computer storage capacity and running time. In practice, various assumptions and approximations are made to simplify the calculation of cell-averaged parameters.

KARATE is a combination of well-known computer codes, GAM-I[5] and THERMOS,[6] with some modifications. In thermal energy range below 1.855 eV, KARATE solves the integral transport equation in one dimension (THERMOS). In the fast energy range, KARATE solves the time-independent Boltzmann equation with isotropic sources using P_1 approximation (GAM).

The fission spectrum for Pu-239 has been used in all plutonium fueled lattices. For those lattices containing only uranium fuel, the U-235 fission spectrum has been used. The effect of heterogeneity in the resonance region, that is, the shadowing effect caused by neighboring rods in the lattice, is accounted for using the method presented by Carlvik.[7]

The calculation of resonance integral is adapted from WIMS[8] approach. This approach includes an intermediate resonance approximation for both the resonance absorber and an admixed moderator. This employs tables of groupwise resonance integrals as a function of temperature and potential scattering and relies on resonance equivalent theorems to interpolate within the table in the determination of the resonance integral appropriate for a given situation.

Some investigations[9,10] have shown that the reflecting boundary conditions in the one-dimensional cylindrical cells should lead to serious errors in the spatial flux distribtuion for tightly packed uranium/water lattices. Subsequent investigations of the effects of cell boundary approximations have also shown that the so-called white boundary conditions[11] lead to some significant improvements in the description of thermal neutron distributions in space and energy. Thus, a white boundary condition is incorporated in THERMOS part of KARATE.

KARATE computes the thermal neutron distribution in space and energy in a cylindrical geometry unit cell composed of up to ten material regions. Space point designation and material placement in each region is quite flexible.

The reactivity effect of the finite PuO_2 particles in the mixed oxide fuels is also incorporated in KARATE. Since KARATE thermal energy range extends to 1.855 eV, encompassing the 1.056 eV resonance of Pu-240 and the 0.3 eV resonance of Pu-239 which are two major contributors to substructure grain heterogeneity effects, the approximative method to treat this effect is incorporated with the THERMOS module only. Substructure grain heterogeneity effect was originally identified by Lane, Nordheim, and Sampson[12] in their analysis of granular HTGR fuels. Windsor and Goldstein[13] adopted this method in the analysis of mixed oxide critical experiments, combined with the isolated grain escape probability as a shielding factor for the thermal energy range. Their results were in good agreement with those of Cobb[14] using more sophisticated approach developed by Wälti.[15] Becase of the simplicity and accuracy of the method by Windsor and Goldstein, it has been chosen for its inclusion into KARATE.

In the slowing down calculation for fast and epithermal energy ranges, GAM assumes the lattice to be a homogeneous mixture of fuel, cladding and moderator. The fast flux is assumed to be spatially flat throughout the unit cell. In reality, however, the fast neutron flux is peaked in the fuel rod and depressed in the moderator which is the opposite of the thermal neutron flux distribution. Thus, a correction is made to account for heterogeneous fast effects. The method incorporated in KARATE is based on the prescription devised by R.L. Hellens which is described in Reference 16.

The GAM code does not take into account the effects of resonance overlap. The presence of

more than a single resonant absorber in a system can lead to an interaction in which neutrons are not available for absorption in a resonance of one isotope due to depletion by resonances of other isotopes at nearby energies. An approximate treatment of this effect as used in WIMS[17] has been incorporated into KARATE.

The solution of the reactor neutron balance equation is obtained using the diffusion theory code, KIDD. The KIDD program solves the homogeneous or non-homogeneous few group diffusion theory equation in one, two and three space dimensions for its lowest eigenvalue and the corresponding direct and/or adjoint vectors, in both multiplying and non-multiplying systems.

III. Calculational Procedure

The reactor model assumed for all of the critical assembly calculations is a simple two-region reactor consisting of a homogeneous core and a water reflector. This reactor is assumed to be a right circular cylinder whose radius is determined from the measured critical number of fuel rods by conserving volume. A one-dimensional description of this model in the radial direction is used in calculating the neutron leakage and associated value of effective multiplication factor for the assembly. The two region cylindrical reactor consisting of a homogeneous core and a water reflector is equivalent to the fully inundated heterogeneous assembly. The analysis proceeds with the calculation of lattice constants for both the homogeneous core and the reflector.

KARATE program has been utilized to obtain four broad energy group macroscopic cross-sections for homogenized core region. The energy boundaries for these four broad groups are given in Table 1. The mixed number density (MND) model is used to obtain thermal neutron cross sections. The thermal neutron distribution is computed in space and energy in a cylindrical geometry unit cell composed of four material regions; namely, fuel, cladding, moderator and white scattering ring. Twenty-one space points describe this unit cell with eight being allocated to the fuel, one to the clad, ten to the moderator, and two to the white scattering ring. Approximate treatments which are made of the resonance overlap effect, heterogeneous fast fission effect, and the grain size of PuO_2 particles have been applied. Measured critical bucklings or approximate material bucklings derived from the critical's equivalent cylindrical dimensions and estimated reflector savings are used to take into account the leakage from the unit cell in the spectrum calculations.

Table 1. Boundaries of Neutron Energy Groups

Group	Energy Range
1	10 MeV — 0.821 MeV
2	0.821 MeV — 5530 eV
3	5530 eV — 0.625 eV
4	0.625 eV — 0.0 eV

The macroscopic cross-sections of the reflector region are taken from the Reference 18. Eich and Williams[19] obtained three fast group water reflector cross-sections by application of the spatial multigroup transport code, P3MG.[20] They also applied AIM-5[21] and SLOP-1[22] codes to the generation of thermal MND cross-sections for water reflector. These four group reflector representations are used throughout our analysis.

Table 2. Reflector Constants*

Group	Σ_a	Σ_r	D
1	0.001190	0.10909	1.4700
2	0.000012	0.14555	1.1145
3	0.000887	0.13873	0.7650
4**	0.021830	—	0.1739

* Constants based on room temperature and atmospheric pressure.

** Thermal MND representation.

The four group constants used in the reflector representation of all room temperature critical analyses are listed in Table 2.

Another item of importance in reactor calculation is the transverse buckling, B_z^2. The radial neutron flux distribution in each broad group is determined exactly whereas the axial neutron distribution is approximated by using a mono-energetic transverse buckling in in both the core and reflector regions. The transverse buckling in critical experiments is determined from the physical height of the fuel or critical water height, H, and measured axial reflector savings, λ_z, by the relationship

$$B_z^2=\left(\frac{\pi}{H+\lambda_z}\right)^2 \quad \cdots\cdots(3.1)$$

In the case of approach-to-critical experiments conducted in Battelle Northwest Laboratories, where the reflector savings are assumed to be the same in the radial and axial directions, the transverse buckling is determined from the physical height of the fuel, H, and the measured reflector savings, λ, by the relationship

$$B_z^2=\left(\frac{\pi}{H+2\lambda}\right)^2. \quad \cdots\cdots(3.2)$$

The KIDD code is run to obtain the direct flux solution and corresponding eigenvalue for the four-broad-group system. The mesh points utilized are 40 in the core and 40 in the reflector region. Thus, the mesh spacing in the core and reflector regions is dependent on the radius of the core. The radius of core, R_c, is determined from the meaured critical number of rods, N_c, by conserving volume. The relationship is

$$R_z^2=N_cA, \quad \cdots\cdots(3.3)$$

where A is the area of a unit cell. Thus, for a square pitch lattice the critical radius is

$$R_c=P(N_c/\pi)^{1/2} \quad \cdots\cdots(3.4)$$

and for a triangular pitch lattice it is

$$R_c=P(\sqrt{3}N_c/2\pi)^{1/2} \quad \cdots\cdots(3.5)$$

where P is the lattice pitch.

IV. Results

The selection of experiments for this analysis is limited to those critical assemblies considered to be uniform. The experiments reported at various laboratories provide the base of data for the theory-experiment correlation. These include five enrichments of plutonium in UO_2-PuO_2 fuels in triangular array and two enrichments of plutonium in UO-PuO_2 fuels in square arrangement. Summaries of these critical experimental results are given in Tables 3 through 5.

Table 3. Summary of Experimental Results for UO_2-1.5 wt% PuO_2 Fueled Lattices

Laboratory	*Lattice Spacing (in.)	Moderator-to-Fuel Volume Ratio	Critical Number of Rods	Measured Buckling (m^{-2})
BNWL**	0.55	0.8382	1,487	48.0
BNWL	0.60	1.187	829	65.1
BNWL	0.71	2.063	484	78.5
BNWL	0.80	2.889	420	74.9
BNWL	0.90	3.922	452	60.9
BNWL	0.93	4.255	488	55.2

- Pellet outer diameter : 0.372 inch
- Clad thickness: 0.027 inch
- Plutonium isotopic composition:

Isotope	Atomic Percentage
Pu-239	91.41
Pu-240	7.83
Pu-241	0.73
Pu-242	0.03

- Cladding material : Zircaloy-2
- PuO_2 mixed in depleted (0.16 at% U-235) UO_2
- PuO_2 is 1.5 wt% mixture
- The PuO_2 particles are not spherical. They were passed through a 325-mesh screen. Their mean diameter is $\sim 25\ \mu m$.

* Triangular lattices.

** BNWL: Battelle Northwest (Pacific) Laboratories.

Table 4. Summary of Experimental Results for UO_2-2 w/o PuO_2 Fueled Lattices

Laboratory	Lattice* Spacings(in.)	Moderator-to-Fuel Volume Ratio	Critical Number of Rods	Buckling(m^{-2})	Boron Concentration (ppm)
		(8% Pu-240 Rods)			
BNWL	0.80	1.211	319.7	93.7	0
BNWL	0.93	1.987	192.4	103.3	0
BNWL	1.05	2.808	152.1	101.3	0
BNWL	1.143	3.513	147.5	97.0	0
BNWL	1.32	5.019	163.1	75.6	0
BNWL	1.386	5.635	179.5	68.9	0
WAPD	0.69	1.125	514	69.6	0
WAPD	0.75	1.557	321	90.0	0
WAPD	0.9758	3.502	160	104.72	0
WAPD	1.0607	4.365	152	98.4	0
WAPD	1.380	8.256	249	50.3	0
WAPD	0.69	1.557	631	62.6	261
WAPD	0.9758	3.502	243	83.7	261
WAPD	0.69	1.557	749	58.3	526
WAPD	0.9758	3.502	379	63.1	526
		(16% Pu-240 Rods)			
BNWL	0.93	1.987	245.6	86.3	0
BNWL	1.05	2.808	194.3	85.4	0
BNWL	1.143	3.513	187.5	81.5	0
BNWL	1.32	5.019	221.1	61.6	0
BNWL	1.386	5.635	254.6	55.6	0
		(24% Pu-240 Rods)			
BNWL	0.80	1.211	519.5	63.1	0
BNWL	0.93	1.987	286.1	79.4	0
BNWL	1.05	2.808	233.2	77.6	0
BNWL	1.143	3.513	232.1	72.2	0
BNWL	1.32	5.019	296.2	53.7	0
BNWL	1.386	5.635	365.3	44.3	0
WAPD	0.9758	3.502	247	79.5	0
WAPD	1.0607	4.365	243	73.3	0

- Pellet outer diameter : 0.505 inch
- Cladding material : Zircaloy-2
- Clad thickness : 0.03 inch
- Fuel density : 9.54 gm/cm^3
- Active fuel height : 36 inches
- PuO_2 mixed in natural UO_2
- PuO_2 is 2 wt% of total mixture
- Isotopic composition of plutonium (at%):

Isotopes	8% Pu-240	16% Pu-240	24% Pu-240
Pu-239	91.62	81.11	71.76
Pu-240	7.65	16.54	23.50
Pu-241	0.70	2.15	4.08
Pu-242	0.03	0.20	0.66

- The PuO_2 particles are not spherical. They were passed through a 325-mesh screens. Their mean diameter is $\sim 25\ \mu m$.

* All WAPD pitches are for square lattices, and all BNWL pitches are triangular lattices.

Table 5. Summary of Experimental Results for UO_2-4 wt% PuO_2 Fueled Lattice

Laboratory	*Lattice Spacing (in.)	Moderator-to-Fuel Volume Ratio	Critical Number of Rods	Measured Buckling (m^{-2})
BNWL	0.85	1.500	252.6	94.7
BNWL	0.93	1.993	178.9	107.9
BNWL	1.05	2.815	138.9	108.7
BNWL	1.143	3.521	122.4	107.9
BNWL	1.386	5.647	123.6	88.4
BNWL	1.60	7.859	180.9	59.5
BNWL	1.70	9.000	271.9	41.1

- Pellet outer diameter : 0.4975 inch
- Cladding material : Zircaloy-2
- Clad thickness : 0.0337 inch
- Active fuel height : 36 inches
- PuO_2 mixed in natural UO_2
- PuO_2 is 3.98 wt% of total mixture
- Fuel density : 9.46 gm/cm^3
- Isotopic composition of plutonium(at%):

Isotope	Composition
Pu-238	0.28
Pu-239	75.38
Pu-240	18.10
Pu-241	5.08
Pu-242	1.15

- The PuO_2 particles are not spherical. They were passed through a 325-mesh screen. Their mean diameter is~25 μm.

* Triangular lattices.

Table 6. Summary of UO_2 Fueled Critical Experiments

Laboratory	Fuel		Clad		Lattice* Spacing (in.)	Experimental Results	
	wt% U-235	Pellet Diameter (in.)	Material	Thickness (in.)		Critical Number of Rods	Measured Buckling (m^{-2})
WAPD	2.72	0.400	Zircaloy-4	0.03145	0.69	317	105.0
WAPD	2.72	0.400	Zircaloy-4	0.03145	0.9758	293	662.0
B&W	4.02	0.444	304-SS	0.0160	0.595	484	88.0
B&W	4.02	0.444	304-SS	0.0160	0.571	608	79.0

* Square lattices

Table 7. Calculated Values of Effective Multiplication Factor for UO_2 Lattices
(Measured K_{eff}=1.0000)

*Lattice Spacing (in.)	H/U Atom Ratio	Enrichment (wt% U-235)	Pellet Dia. (in.)	K_{eff}
0.69	5.48	2.72	0.400	0.9973
0.9758	17.83	2.72	0.400	1.0001
0.595	3.61	4.02	0.444	1.0056
0.571	3.03	4.02	0.444	1.0034
Mean K_{eff}=1.0016±0.0031				

* Square Lattices

Four cases of slightly enriched UO_2 lattice critical experiments are also included in the analysis[23),24)]. In Table 6 is presented a summary of experimental results for UO_2 criticals.

The adequacy of our calculational model in predicting the effective multiplication factor for various critical experiments is shown in Tables 7 through 10 where the calculated values of the effective multiplication factor for each experiment and the mean value of the effective multi-

plication factor for each experiment and the mean value of the effective multiplication factor for each series of experiments are presented.

Looking first at the uranium dioxide system, the number of selected experiments is so small that it is impossible to say as to whether there is any trend in the effective multiplication factors. However, the calculated values of the effective multiplication factor are in close agreement with experimental data. The results for the mixed oxide systems, shown in Tables 8 through 10, represent some of the systematic trends. A general trend is such that the effective multiplication factor decreases as the lattice spacing increases. It does not appear, however, that the effect of changes in plutonium composition exhibits any noticeable trend. The average value of the effective multiplication factor for 41 critical experiments is evaluated as 0.9997 with standard deviation of 0.43%.

Table 8. Calculated Values of Effective Multiplication Factor for UO_2-1.5 wt% PuO_2 Lattices (Measured K_{eff}=1.0000)

* Lattice Spacing (in.)	H/U Atom Ratio	Core Radius (cm)	Transverse Buckling (m^{-2})	K_{eff}
0.55	230	30.054	5.700	0.9969
0.60	326	23.038	5.096	0.9985
0.71	567	20.831	5.203	0.9989
0.80	793	21.864	5.255	0.9953
0.90	1,077	25.517	5.282	0.9954
0.93	1,169	27.454	5.291	0.9968
Mean K_{eff}=0.9970±0.0014				

* Triangular lattice

Table 9. Calculated Values of Effective Multiplication Factor for UO_2-2 wt% PuO_2 Lattices (Measured K_{eff}=1.0000)

Laboratory	Lattice* Spacing(in.)	H/U Atom Ratio	a/o Pu-240	Core Radius (cm)	Transverse Buckling (m^{-2})	Boron Concentration (ppm)	K_{eff}
BNWL	0.80	238	8	19.076	8.864	0	0.9983
BNWL	0.93	391	8	17.203	8.705	0	1.0030
BNWL	1.05	552	8	17.270	8.646	0	0.9977
BNWL	1.143	692	8	18.512	8.841	0	1.0007
BNWL	1.32	988	8	22.482	8.891	0	0.9978
BNWL	1.386	1,110	8	27.764	9.100	0	0.9940
Mean K_{eff}=0.9986±0.0028							
WAPD	0.69	177	8	22.418	8.560	0	1.0051
WAPD	0.75	245	8	19.256	8.967	0	1.0093
WAPD	0.9758	552	8	17.688	9.466	0	1.0031
WAPD	1.0607	688	8	18.740	9.471	0	1.0025
WAPD	1.38	1,301	8	31.206	9.520	0	0.9930
WAPD	0.69	177	8	24.838	8.731	261	1.0028
WAPD	0.9758	552	8	21.798	9.526	261	0.9947
WAPD	0.69	177	8	27.061	8.954	526	0.9989
WAPD	0.9758	552	8	27.223	9.635	526	0.9929
Mean K_{eff}=1.0003±0.0054							

BNWL	0.93	391	16	19.437	8.598	0	1.0054
BNWL	1.05	552	16	19.519	8.579	0	1.0007
BNWL	1.143	692	16	20.872	8.771	0	0.9999
BNWL	1.32	988	16	26.175	8.894	0	0.9968
BNWL	1.386	1,110	16	29.493	9.278	0	0.9925
Mean $K_{eff}=0.9991\pm0.0043$							
BNWL	0.80	238	24	24.317	8.484	0	0.9989
BNWL	0.93	391	24	20.978	8.672	0	1.0001
BNWL	1.05	552	24	21.384	8.711	0	0.9991
BNWL	1.143	692	24	23.222	8.877	0	0.9999
BNWL	1.32	988	24	30.296	9.299	0	0.9953
BNWL	1.386	1,110	24	35.328	9.434	0	0.9932
Mean $K_{eff}=0.9978\pm0.0026$							
WAPD	0.9758	552	24	21.977	9.436	0	1.0009
WAPD	1.0607	688	24	23.694	9.639	0	1.0005
Mean $K_{eff}=1.0007\pm0.0023$							

* All BNWL lattices are triangular and all WAPD lattices are square

Table 10. Calculated Values of Effective Multiplication Factor for UO_2-4 wt% PuO_2 Lattices (Measured $K_{eff}=1.0000$)

Lattice Spacing* (in.)	H/Pu Atom Ratio	Core Radius (cm)	Transverse Buckling (m^{-2})	K_{eff}
0.85	153	18.016	8.585	1.0039
0.93	203	16.589	8.695	1.0022
1.05	289	16.503	8.698	1.0096
1.143	414	16.864	8.781	1.0070
1.60	805	28.699	9.479	1.0020
1.70	922	37.383	9.457	0.9993
Mean $K_{eff}=1.0043\pm0.0032$				

*Triangular lattices

V. Conclusion and Recommendation

The analyses onthe separate effects of various approximations incorporated in KARATE are not attempted in this paper. Furthermore, there are many other parameters, such as power distribution, reactivity coefficients, control rod worth, and burnup, that need to be investigated in connection with plutonium recycle. Since the reactivity calculation is a very basic problem, however, the results of this analysis illustrate the fact that KARATE/KIDD system can be effectively used for the analysis of uniform lattices of UO_2-PuO_2 fuels.

The extension of KARATE/KIDD system for the analysis of lattices whose spatial uniformity is interrupted at regular intervals by discontinuities such as control elements, water thimble, mechanical structure, etc., as in reactor cores is yet to be done and prerequisite for the future work.

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