Numerical Analysis of the Reactivity for the Dry Lattices above the Water Level of the Critical Fuel Cores

Yasushi NAUCHI* and Takanori KAMEYAMA
Central Research Institute of Electric Power Industry, 2-11-1, Iwado-Kita, Komae-shi, Tokyo 201-8511, JAPAN
* *nauchi@criepi.denken.or.jp*

Criticality analysis has been performed for dozens of tank type cores in which fuel lattices are loaded vertically and partially immersed in light water. The reactivity effect of dry part of lattices stuck above the critical water level has been calculated using the continuous energy Monte Carlo method. The reactivity effect exceeds 0.8% both for MOX and UOX fuel lattices of large buckling (B_z^2>0.0025cm^{-2}). It is evaluated that at least 20cm length of fuel rods above the critical water level has significant reactivity effect.

KEYWORDS*: *dry part of lattice, reactivity effect, critical water level, buckling

1. Introduction

Criticality data have been measured and reported for light water moderated lattices in tank type core facilities in which fuel rods are loaded vertically and reactivity is controlled by level of moderator[1]. Since those cores normally include no control rods or soluble boron, different size of critical lattices can be mocked up easily. The data measured for such series of lattices are useful for the verification of neutronics calculation methods and nuclear data libraries.

Criticality of the cores is achieved when parts of fuel rods are immersed in the moderator and the rest parts are left in dry air. The reactivity effect of the dry part is usually considered much smaller than that of the moderated part. In the calculations with deterministic methods, the geometry of the dry part of lattices is not modeled rigorously, but modeled by the assumed boundary conditions such as extrapolated length, buckling, albedo or something[2]. On the other hand, in the calculations with the continuous energy Monte Carlo method, the total length of fuel rods including the dry part is often modeled to predict the eigenvalue with high accuracy[3], because the three-dimensional transport calculation is available with the method. Accordingly, it is indispensable to clarify the reactivity effect of the dry part for the criticality calculations.

In the present paper, criticality calculations were performed for UOX and MOX fuel lattices with the continuous energy Monte Carlo method. The reactivity effect of the dry part of fuel lattices are identified in relation to the vertical length of moderated part and volumetric ratio of fuel to moderator.

2. Reactivity Effect of Dry Lattice

2.1 Critical Fuel Lattice

Criticality calculations were performed for UOX and MOX fuel lattices in light water where dry fuel rods stand above the critical water surface.

The criticality data of UOX fuel lattices had been reported from Tank-type Critical assembly (TCA) of Japan Atomic Energy Research Institute (JAERI)[3]. 2.6% ^{235}U enriched UO_2 fuel rods of 144.15cm length were used. The fuel pellets were enclosed in aluminum tube of 0.76mm thickness with aluminum end plug of 16.8cm length on bottom side. They were loaded in square shaped lattices of 18.49mm, 19.56mm, 21.50mm or 22.93mm pitches, which determine the volumetric ratio of moderator to fuel (V_m/V_f) to be 1.50, 1.83, 2.48 or 3.00, respectively. Neutron was reflected in the water region outside the lattices, since the width of the water region is 70cm horizontally and 16.8cm bellow the bottom. The lattice data used are summarized in Table 1.

The data of MOX fuel lattices of Critical Reactor Experiment facility (CRX) had been reported by the Westinghouse Reactor Evaluation Center[4]. The lattices were mocked up with MOX fuel rods of 92.96cm length containing 6.6% PuO_2 in natural UO_2

and 5.74% ^{235}U enriched UO_2 fuel rods. Those rods were loaded in square shaped lattices of several kinds of pitches. The (V_m/V_f) ratios of the lattices were varied from 1.5 to 11. The region outside the lattice is in water of more than 30cm horizontally, and is filled with water and aluminum below the bottom. The size, pitch and critical water level of the lattices are listed in Table 2.

2.2 Criticality Calculation

Criticality calculations were performed with the continuous energy Monte Carlo code, MCNP-4C[5]. The pointwise nuclear data set FSXJ3-R2 library based on JENDL-3.2 [6] was applied for the calculations. TMCCS library included in DLC-200 library set was also used for inelastic scattering on H_2O molecular[7].

Eigenvalue and vertical (axial) fission rate distributions were calculated for the 183U21 lattice and the CRX8 lattice, in which the long parts of fuels were dry out of water region. In the calculations, the total length of fuel rods is modeled, while the region above the top of fuel rods was assumed to be vacuum. The initial source point was selected at the middle of moderated part of the central fuel rod. In the calculation for the 183U21 lattice, 160000 histories were sampled for each generation and the eigenvalue estimators were accumulated for 150 generations after skipping 100 generations.

The eigenvalue was calculated to be 1.00162 (statistical error 0.00015) for the 183U21 and to be 1.00594 (0.00022) for the CRX8. The fission rate distribution F_z along the vertical axis (z-axis) of the CRX8 lattice is shown in Fig. 1. In addition to general cosine shaped distribution and reflector saving effect near the bottom part, the considerable amount of fission rate is observed in dry part. The ratios of the dry part for the total number of neutron generation were 2.0% for the 183U21 and 2.2% for the CRX8. Those ratios were not negligible for the criticality calculations.

2.3 Reactivity of Dry Lattice

The reactivity effect of the dry part was investigated in series of eigenvalue calculations for the lattices with varying the boundary position from the criticality water level H_c to the top of fuel. The calculated eigenvalue for the 183U21 and CRX8 are shown in Fig. 2. For both UOX and MOX fuel lattices, the eigenvalue increased with the height of the boundary up to 0.8% and were leveled off around the height of 20cm. The length is longer than the extrapolated length λ_z of 11.5cm for 183u21 and 10.8cm for the CRX8, when λ_z were deduced by fitting fission distribution F(z) as

$$F(z) = F\cos\left(\pi\frac{(z-z_0)}{H_c+\lambda_z}\right) = F\cos(B_z(z-z_0)) \quad (1).$$

Here, B_z is the square root of axial buckling and z_0 is the loop of the cosine shaped distribution.

Recently, It has been pointed out the influence of initial source distribution on calculated k_{eff} with the continuous energy Monte Carlo method[8]. The two lattices were checked using three kinds of source distribution,

1) the calculated vertical fission rate distribution,
2) identical to 1) without the dry part,
3) point source on the center of moderated part of the lattice. The calculated eigenvalues are described in Table 3. The dependence of eigenvalue upon the initial source is limited to 0.05%. The reactivity effect of the dry part of lattices was not so much influenced by the initial source distribution.

3. Reactivity of the Dry Lattice in Relation to Lattice Property

For critical lattices of TCA and CRX facility listed in Table 1 and 2, the reactivity effect of the dry part were identified by two kinds of modeling of the whole fuel rods and the partial fuel rods below the critical water level H_c. The effect was deduced by taking the difference of the two eigenvalues. In Fig. 3, the reactivity effects are plotted to the axial buckling B_z^2 which were involved in the equation (1). The effect shows roughly linear relation to the B_z^2 for UOX lattices as well as MOX fuel lattices.

In this section, the reactivity effect is investigated in detail to clarify which parameters mainly influence the reactivity effect.

3.1 Neutron Gain in Dry Lattices

Neutron gain in the dry part of lattices is defined by the sum of inward current and fission neutron yield in each lattice. In square shaped lattices, the inward current consists of neutrons from moderated part of lattice and those from outer moderator (see Fig. 4). Each component of the neutron gain for TCA 183Uxx lattices is listed in Table 4. The inward currents from the moderated part shared over 90% of the total one and they exceed the fission yield. In Fig. 5, the inward current of thermal neutron of E_n < 0.14eV is plotted to the axial buckling B_z^2 and fission yield is also done. The both thermal neutron currents from moderated lattices and from outer moderator is significantly proportional to the B_z^2, and as the results, the fission gain also shows proportional relation to the B_z^2.

3.2 Multiplicity of Neutrons emitted from Dry Lattice

The reactivity effect of dry part is limited to 0.8% for the TCA 183U21 lattice, although neutron yield in the lattice is almost 2%. Thus the multiplicity of neutron yielded in the dry part was surveyed.

In the MCNP-4C code, the eigenvalue k_{eff} is defined by the ratio of fission neutron yield between successive generations. Defining M(**r**,E) as the expectation number of neutrons yielded by a fission neutron of energy E radiated at position **r**, here, the k_{eff} is expressed as follows.

$$k_{eff} = \frac{\int_{total-fuel} dV \int dE \left[M \int \chi \nu \Sigma_f \phi' dE' \right]}{\int \int \nu \Sigma_f \phi' dE' dV} \qquad (2)$$

By taking the integration of the numerator of equation (2) over any regions, we can evaluate the contribution of the regions to the eigenvalue.

The authors developed the new function calculating the contribution and attached it to the MCNP-4C code. Using them, the contributions of the dry part were calculated for 183Uxx lattices and are summarized in Table 5. On the right edge column of the Table, ratios of the contributions to the neutron yield in the dry part of lattices are shown, which means the multiplicity of neutrons emitted from the part. The multiplicity is limited to 0.38. Thus, the reactivity effect of 183U21 lattice did not exceed 1% in spite of fission yield of 2% to the total.

The multiplicity in dry lattice listed in Table 5 is slightly increased with B_z^2 but rather flatten for the 5 lattices. Attribute to this tendency together with the linear relation of the inward neutron current to the reactivity effect of the dry part, the reactivity effect roughly show linear relation to B_z^2 as shown in Fig. 3.

3.3 Reactivity Effect of Dry Lattice by (V_m/V_f)

Further investigation had been conducted on the reactivity effect of the dry part associated with the (V_m/V_f) in the moderated lattice.

Since the same axial buckling B_z^2 and the same kind of fuel rods were needed to clarify the influence of the (V_m/V_f), the reactivity effects of CRX MOX fuel lattices of (V_m/V_f)=1.68 and 2.16 were evaluated for B_z^2=0.0014cm^{-2} by the linear relation of the reactivity effect to the B_z^2.

For lattices of (V_m/V_f)=4.71, 5.67 and 10.75, simulated lattices were constructed by supplement of a row of fuel rods and decrement of critical water height to the experimental critical lattices with roughly conserving the sum of axial and horizontal buckling. The reactivity effect of dry part of those lattices were evaluated for the B_z^2=0.0014cm^{-2} using the data for experimental and simulated lattices.

In Table 6, sum of inward thermal neutron currents into the dry lattices are listed with neutron yield in them. In the lattices of small (V_m/V_f), the ratio of neutron yield to the sum of inward current is large owing to the small leakage probability. As the results, the evaluated reactivity of the dry lattices decreases as the (V_m/V_f) increase as shown in Fig. 6.

4. Reactivity Effect and Boundary Conditions

The dry part of lattice is commonly treated with some kinds of boundary conditions in deterministic methods. Among them, a half of extrapolated length, $\lambda_z/2$ is generally used to determine the calculation boundary outside a reactor. Here we discussed the relation of calculated eigenvalues with the use of the extrapolating length modeling the dry part of fuel.

For square shaped critical lattices, the neutron flux distribution in them can be expressed by a cosine function as the equation (1) except for the edge region. Hypothetical boundary between dry part and moderated part can be established by deduction of Z where F(Z)=0. The Z values were calculated for TCA and CRX. Then we performed criticality calculations by raising the moderator surface level to the Z and setting vacuum condition there. In Fig. 7, the eigenvalues on the improved conditions are compared with those by modeling the total length of fuel rods. The former were larger than the latter for every lattice, though the difference of eigenvalues depended on the axial buckling B_z^2.

5. Summary and conclusion

Criticality calculations had been performed for dozens of tank type cores in which criticality is achieved by the moderated part of fuel rods and the dry part of fuel rods above the critical water surface. This paper revealed that the reactivity effect of dry part was more than 0.8% both for MOX and UOX fuel lattices of larger buckling (B_z^2>0.0025cm^{-2}). It was also evaluated that 20cm length of fuel rods above the critical water level has significant reactivity effect. The reactivity effect was found linearly increased with the axial buckling and monotonously decreased with volumetric ratio of moderator to fuel of the lattices.

As described, the reactivity effect of the dry part of fuel rods is indispensable in analysis of critical experiments. Caution should be taken to the effect when we compare the accuracy of criticality calculation methods.

References

1) H. Tsuruta, I. Kobayashi, T. Suzaki, A. Ohno, K. Murakami and S. Matsuura, JAERI 1254, (1978).

2) B. Na and E. Sartori, Nucl. Sci. Eng., **142** 37 (2002).

3) T. Suzaki, K. Sakurai, K. Nakajima and O. Horiki, Proc. Int. Conf. on Nuclear Criticality Safety, ICNC'99, Versailles, France, Sept. 20-24, 1999, 386 (1999).

4) N. M. Abdurrahman, G. Radulescu and I. Carron, Nucl. Technol., **127** 315 (1999).

5) J. F. Briesmeister, "MCNPTM-A General Monte Carlo N-particle transport code," LA-13709-M, (2000).

6) K. Kosako, F. Maekawa, Y. Oyama, Y. Uno and H. Maekawa, "A CONTINUOUS ENERGY CROSS SECTION LIBRARY FOR MCNP BASED ON JENDL-3.2", JAERI-Data/Code 94-020, (1994).

7) J. S. Headricks, S. C. Frankle, J. D. Court, "ENDF-B VI Data for MCNP" LA-12891 (1994).

8) T. Kiyosumi and Y. Miyoshi, Proc. Int. Sympo. NUCEF 2001, Tokai, JAPAN, Oct. 31-Nov. 2, 2001, 415 (2002).

Table 1 Critical lattices of TCA

Lattice Name	Lattice Size	pitch (mm)	Vm/Vf	Critical Water Level(cm)
150U21	21x21	18.49	1.50	60.49
183U17	17x17	19.56	1.83	113.03
183U18	18x18	19.56	1.83	75.42
183U19	19x19	19.56	1.83	60.21
183U20	20x20	19.56	1.83	51.1
183U21	21x21	19.56	1.83	45.71
248U17	17x17	21.5	2.48	59.9
300U17	17x17	22.93	3.00	53.11

Table 2 Critical lattices of CRX facility

Lattice Name	Lattice size & configuration	pitch (mm)	Vm/Vf	Critical Water Level(cm)
CRX1	22x23MOX	13.21	1.68	82.90
CRX2	19x19MOX	14.22	2.16	80.75
CRX8	21x21MOX	14.22	2.16	50.38
CRX12	13x13MOX	18.68	4.71	68.41
CRX13	12x12MOX	20.12	5.67	76.76
CRX14	11x11MOX	26.42	10.75	79.50
CRX15	19x19UOX	14.22	2.16	81.94
CRX18	21x21UOX	14.22	2.16	51.00
CRX21	19x19UOX central 11x11MOX	14.22	2.16	89.43
CRX29	19x19MOX central 11x11UOX	14.22	2.16	74.43
CRX30	19x19UOX central 3x3MOX	14.22	2.16	82.17
CRX32	23x23MOX	13.21	1.68	74.33
CRX40	25x24MOX	13.21	1.68	57.22

Table 3 The initial source distribution and calculated eigenvalues with the continuous energy Monte Carlo Method. For both lattices, 100000 x 150 histories were sampled.

initial source distribution	183U21 eigenvalue error	CRX8 eigenvalue error
(1) Calculated axial fission rate distribution	1.00185 0.00019	1.00549 0.00022
(2) Same as (1) but no source in air	1.00162 0.00019	1.00592 0.00022
(3) Point source	1.00136 0.00018	1.00594 0.00022

Table 4 Neutron gain in dry lattices

Lattice	Current from Moderated Lattice	Current from Outer Moderator	Fission neutron yield
	(/total neutron yielded in the system)		
183U17	1.21E-02	1.08E-03	3.49E-03
183U18	2.56E-02	2.35E-03	7.85E-03
183U19	3.83E-02	3.37E-03	1.19E-02
183U20	5.11E-02	4.21E-03	1.61E-02
183U21	6.24E-02	4.86E-03	1.99E-02

Table 5 Contribution of dry lattice to the k_{eff}

Lattice	B_z^2 cm^{-2}	fission neutron yield	Contribution to k_{eff}	Contribution yield
183U17	0.00063	3.47E-03	1.21E-03	0.34823
183U18	0.00129	7.83E-03	2.76E-03	0.35255
183U19	0.00189	1.19E-02	4.34E-03	0.36587
183U20	0.00258	1.61E-02	6.16E-03	0.38328
183U21	0.00301	1.98E-02	7.89E-03	0.39809

Table 6 Sum of inward current to the dry part of lattices and fission neutron yield in the part

Lattice	(V_m/V_f)	B_z^2 cm^{-2}	(a) Sum of thermal current	(b) fission neutron yield	(b)/(a)
CRX32	1.68	0.00121	8.29E-04	2.35E-02	28.37
CRX40	1.68	0.00188	1.32E-03	5.24E-02	39.71
CRX12	4.71	0.00158	1.86E-03	7.34E-03	3.95
Simulation	4.71	0.00230	2.63E-03	1.26E-02	4.79
CRX13	5.67	0.00129	1.74E-03	4.57E-03	2.62
Simulation	5.67	0.00217	2.80E-03	1.02E-02	3.64
CRX14	10.75	0.00124	2.55E-03	3.57E-03	1.40
Simulation	10.75	0.00186	4.02E-03	7.52E-03	1.87

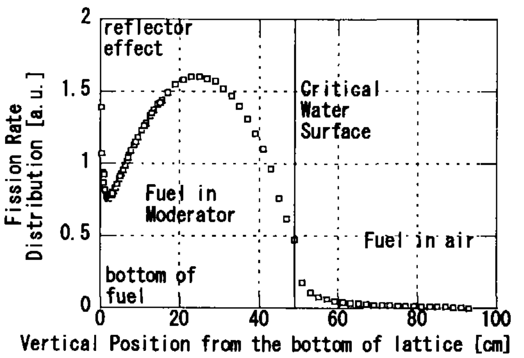

Fig. 1 Calculated fission rate distribution of the CRX8 lattice.

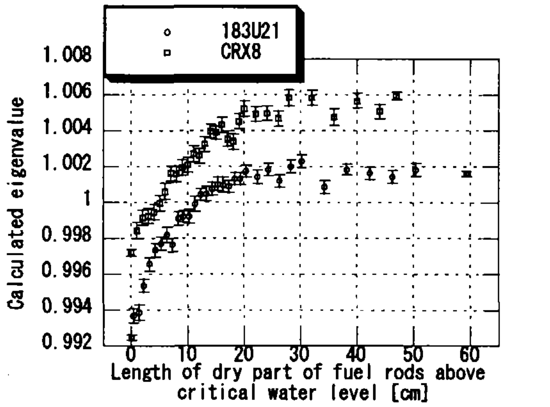

Fig 2 Evaluated eigenvalue of 183U21 and CRX8 lattices in relation to the length of the dry part above the water level in the calculation models.

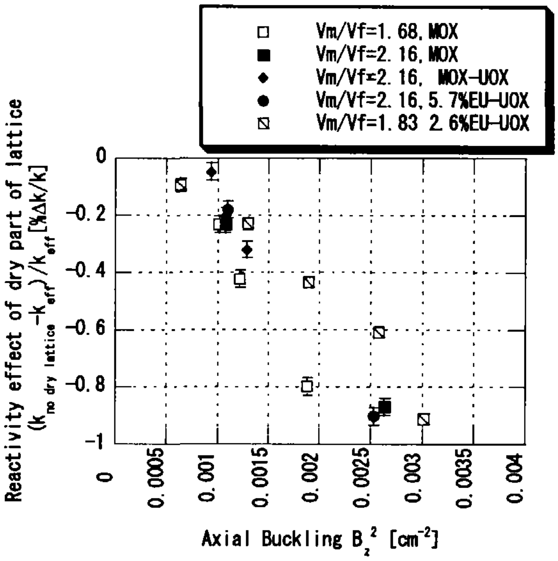

Fig. 3 Relation of axial buckling B_z^2 to reactivity effect of dry part of lattices.

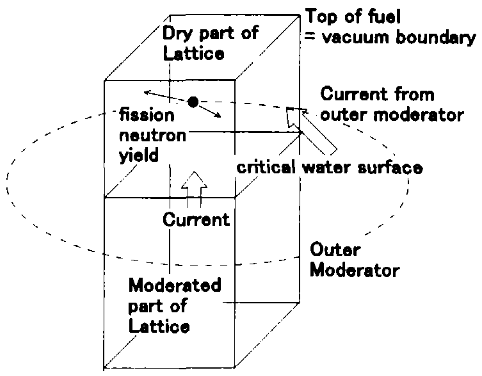

Fig. 4 Schematic view of neutron gain in dry lattice.

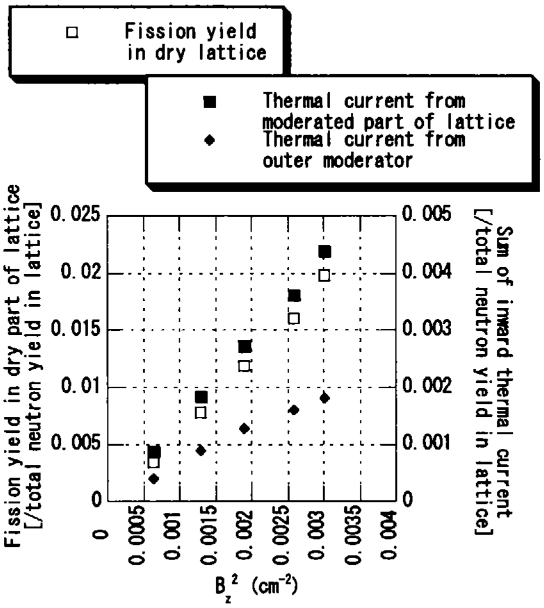

Fig. 5 Axial buckling and neutron current.

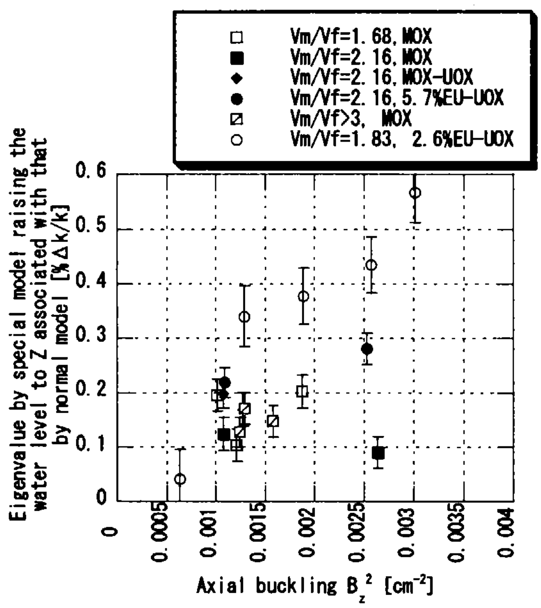

Fig. 7 Eigenvalue of lattices raising critical water height to Z on F(Z)=0 comparing with that in normal modeling.

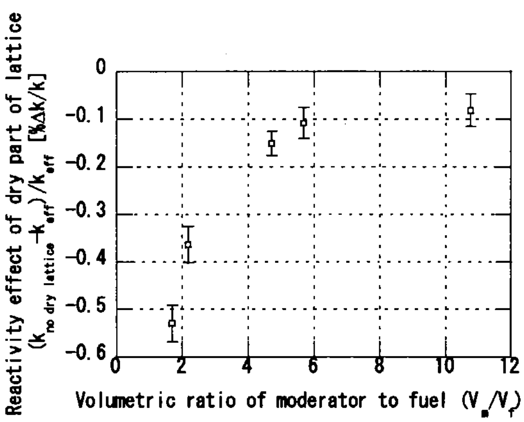

Fig. 6 Reactivity effect of dry part of lattices and volumetric ratio of moderator to fuel (B_z^2 are adjusted to be 0.0014cm^{-2} for all lattices).