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A Determination of Bias between Calculational Methods for the Criticality Safety Analysis of Spent Fuel Storage Pool with Burnup Credit

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연소를 고려한 사용후핵연료저장조 핵임계 안전성분석에서 계산체제간의 편차결정

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Abstract

A test is made for a method to determine reliable bias in the criticality safety analysis of spent fuel storage pool with burnup credit between the reference and rack criticality calculation methods. The spent fuel pool of Kori Unit 1 is conceptually redesigned to the most compact rack with burnup credit, and its multiplication factors are calculated depending on fuel enrichment and burnup, by the Monte Carlo code KENO-IV as a reference and by a two-dimensional collision probability code FATAC as a practical method. Then, the computed values with the help of the above two computer codes are compared to evaluate the bias and its trend in terms of multiplication factor on fuel enrichment and burnup. The result indicates that the bias can be determined with reliability basis but without any disadvantage in criticality safety margin compared with the conventional method.

요 약

연소를 고려하는 사용후핵연료저장조의 핵임계 안전성 분석에서 검증용 계산 체제와 rack계산 체제 사이의 편차를 신뢰성 있게 결정하는 방법을 시험하였다. 이를 위하여 고리 1호기의 사용후핵연료저장조를 연소를 고려하는 가장 조밀한 rack으로 개념설계하고, 핵연료의 농축도 및 연소도에 따라 증배계수를 계산하였다. 표준값 생산용 Monte Carlo 코드로는 KENO-IV를 그리고 실제 rack 설계용으로는 2차원 충돌확률 코드인 FATAC을 사용하였다. 이 두 계산의 결과를 상호 비교하여 계산 체제 사이의 편차와 이의 경향성 및 신뢰도를 평가하였다. 이 방법을 사용하면 확실한 신뢰도 근거를 마련할 수 있을 뿐만 아니라 반응도 여유면에서 기존의 방법보다 불리하지 않음이 입증되었다.

I. Introduction

From the early 1970's all the nuclear power plants began to be faced with the shortage of the spent fuel storage capacity due to the discouraging prospects for the chemical reprocessing and the ultimate radwaste disposal. The most economic solution to this problem has been the expansion of the already existing pool capacity, and several methods have been employed for the more compact storage.

The efforts to enlarge storage capacity consequently reduced the safety margin of nuclear criticality, and it evoked the importance of stringent accuracy verification in criticality calculation. Critical experiments which were being faded away in USA were performed once and again for the experimental data to verify the accuracy of the calculation. In the conference on nuclear critical experiments held in Rocky Flat, Colorado, 1975, the importance of nuclear critical experiments simulating the spent fuel storage rack was emphasized[1]. In 1976, BPNL carried out a series of critical experiments simulating the fuel shipping cask and the storage pool sponsored by US NRC[2,3]. In 1977, the US DOE (US ERDA at that time) awarded a contract to BAW to carry out critical experiments simulating the spent fuel racks of new concept[4,5].

Meanwhile the US NRC changed its policy of admitting burnup credit in the criticality safety analysis of the spent fuel storage facility[6]. Under such circumstances, the gap size between fuel assemblies has been minimized down to less than 2cm in the currently introduced reracking concept by inserting neutron absorber in the rack and by analyzing criticality safety with burnup credit.

The criticality calculation with burnup credit ensued on some difficulty in using the conventional Monte Carlo procedures. There have been no systematic procedures to produce fission product cross-sections benchmarked for the input of Monte Carlo calculation because there have been no critical experiments ever conducted for burned fuel. Thus it has been obliged to use the reactor core design codes for the criticality calculation with burnup credit.

When a core design code is applied to the criticality safety analysis, its accuracy must be pre-evaluated for each specific problem because it is tuned to a specific core. This accuracy evaluation has been carried out by determining bias due to the use of a design code in stead of the general Monte Carlo method. This bias has been determined by such way that two multiplication factors are calculated for a pool containing new fuels of the maximum enrichment permissible for storage by the design code and the reference method, respectively, and then the difference between these two values is decided as bias. This method has been employed in every burnup credit rack design since the first license of spent fuel storage pool with burnup credit had been granted Callaway[7].

An attempt is employed in this paper to scrutinize the bias determined by the aforementioned method, and a test is made for developing a new method to determine the bias reliably. KENO-IV[8] with 19 group constants is adopted for the purpose of the accuracy verification whereas the two-dimensional collision probability code FATAC[9] is employed as the practical rack calculation tool. 19 group constants are generated by AMPX-II[14] for KENO-IV input.

Since burned fuel contains plutonium nuclides, a series of benchmark calculations of KENO-IV are carried out for MOX (PuO_2+UO_2) fuel critical experiments so as to establish its reliability basis for the burned fuel calculation. For the purpose of this study the spent fuel storage pool of Kori Unit 1 is chosen herein, and it is redesigned by

the most compact rack. Then the multiplication factors are calculated in due respect of various fuel enrichments and burnups by each method. Finally the results are evaluated to determine the bias and its trend with the confidence level.

II. Calculation

2-1. Calculational Methods

The 19 group cross-section library generation procedure developed at ORNL[10] is adopted for the broad group cross-section generation. The structure of the 19 group library contains 4 thermal energy groups which have been extended from the 16th of the 16 group Hansen-Roach library.[11] The modular code package AMPX-II[14] contains the modules for the broad group cross-section generation from the 218 group master library named CSRL-IV[15] which is processed from the ENDF/B-IV by one of the AMPX modules.

This library is collapsed to the intermediate library, 51 or 60 groups in this study, by space-independent spectrum weighting. A neutron energy spectrum as a weighting function for the collapsing should be changed for a different medium. But the recommended one consisting of fission, 1/E, and Maxwellian spectra is used for every case in this analysis because this constant weighting did not cause any significant errors in other works.[12,13]

Nordheim Integral Treatment[16] is selected among three options to evaluate the resonance integral, and the Dancoff correction factor for the shadowing effect is calculated by Sauer's approximation.[17]

The final 19 group constants are generated from the intermediate library by the space and energy weighted collapsing using the 1-dimensional cell calculation.

Reference multiplication factors of the storage facility are obtained from KENO-IV after its benchmarking for MOX fuel. Average neutron weights of 0.5 in the core or fuel assembly and 1.0 in the reflector are given independent of the group.

The geometry simulating the benchmark experiment is 1/4 of the full core with sufficiently thick reflectors at the bottom and two side boundaries. And the storage facility to be tested is represented by 1/4 of unit rack with the zero current conditions for 4 sides and with the sufficiently thick reflectors at the top and bottom. The number of neutron generations is roughly adjusted from 73 to 103 such that the resulting multiplication factor has 1 standard deviation of about 0.004. The first three generations are skipped in the statistical process computing the multiplication factor, standard deviations, etc., and 300 histories are given to each generation. The fuel and guide thimble cells are homogenized, respectively, due mainly to the limited computer capacity.

FATAC, a 2-dimensional collision probability code which is a modified version of WIMS,[18] is used as the practical rack criticality calculation. Because of the fixed neutron poisons in a form of Boraflex sheet in the rack, the transport code is selected rather than the diffusion code which would cause significant errors. Hoover, et al. showed that the diffusion code had underpredicted 7% on multiplication factors in the lumped poisoneous critical experiment.[4] The fine group library for the code has 69 group structure rooted on UKNDL.

The rack structure is modeled as a homogeneous medium of SS-304 and Boraflex because the code cannot describe these two discrete structures. 2×2 meshes are given to each fuel or guide thimble cell and 1 mesh each in the transverse direction to rack structure or water gap.

2-2. Benchmark Calculation

The usefulness of 19-group KENO-IV to the

UO_2 fueled system was verified by the benchmarks performed by E.R. Roh[12] and C.K. Lee.[13] In the above two independent works, the criticality was analyzed on the critical experiments of BPNL[2,3] and BAW[4,5], respectively, which simulated spent fuel facilities. The results of those benchmarks indicated that the criticality estimations of this procedure were very accurate for UO_2 fuel without any error trend on the enrichment, rod pitch, boron concentration, lumped poison, etc. However, there had been no domestic works on the MOX fuel by this procedure. 6 benchmark calculations are carried out on the BPNL experiments using the MOX fuel in order to judge if this procedure is applicable to burned fuel calculation.

2-3. Selection of the Object and Its Analysis

The spent fuel storage facility of Kori Unit 1 can accommodate the spent fuels for 11 years after the reracking with a smaller assembly pitch. The conceptual redesign of this facility is made with the use of Boraflex sheet as the neutron poison and with the shortest assembly pitch which reflected only the mechanical tolerance. Thicknesses and contents of the SS-304 plates and Boraflex sheet are the same as those used in the work by C.K. Lee, et al. Fig. 2-1 illustrates the rack structure. Such a conceptual reracking should increase the capacity of the facility to about 3 times of the present capacity. Detailed safety analysis on a new design is not included in the scope of this study.

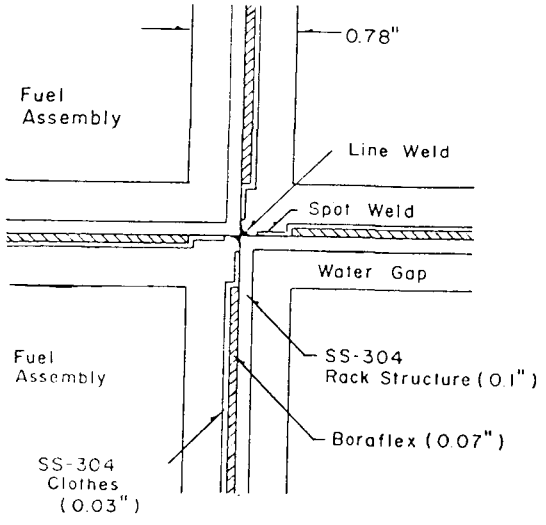

Fig. 2-1. Poisoned Rack Structure Design

Calculations by KENO-IV and FATAC are performed for 6 different enrichments from 1.5 to 4.0% and for 7 burnup steps from 5,000 to 35,000 MWD/MTU of 3.5% fuel. The burnup dependent nuclide inventory data are obtained from the burnup calculation by FATAC under the reactor conditions such as hot full power, 500ppm of the soluble boron concentration, and equilibrium xenon. Fission products are excluded from the rack calculations with burnup.

III. Results and Discussion

3-1. Benchmark Calculation

The results of 19 group KENO-IV benchmark calculation for the BPNL's MOX fuel critical experiments are summarized in Table 3-1. While the number of data is insufficient to demonstrate a statistical result, the estimated values show a distinct tendency depending on the rod pitch, i.e., underestimate about 1.3% when under-moderation (pitch 0.7") and overestimate about 2.4% when over-moderation (pitch 0.99"). The cases containing soluble boron show a little bit higher values than those without boron, but the tendency is not distinctive.

R.M. Westfall[10] having tested the 19 group cross-section generation procedure which was developed by himself, commented that this cross-section set might need to be expanded to adequately treat plutonium. Good agreement for UO_2 but not so good agreement for MOX fuel system connotes that the plutonium cross-sections were not properly treated.

In this study a test is made for the cross-sections at 0.3eV Pu-239 resonance which is

Table 3-1. The Result of the MOX Fuel Benchmark Calculation

No.	Pitch (in.)	Boron Con. (ppm)	Calculated k_{eff}*	
			51→19	60→19
1	0.700	1.7±0.1	0.9871±0.0040	0.9835±0.0043
2	0.700	680.9±2	0.9891±0.0039	0.9882±0.0033
3	0.870	0.9±0.2	0.9969±0.0041	0.9902±0.0042
4	0.870	1,090.4±2	1.0178±0.0034	1.0070±0.0031
5	0.990	1.6±0.1	1.0202±0.0041	1.0154±0.0047
6	0.990	762.2±2	1.0238±0.0032	1.0189±0.0028
Average			1.0058±0.0166	1.0005±0.0152

* Errors are by standard deviation.

the highest among its resonance peaks and whose reaction effect is far more significant because of its thermal energy range. As this energy range is not included in the resonance region of NITAWL, these cross-sections are treated as if they were of non-resonance. This energy range (0.15~0.6eV) is divided into 13 groups in the 218 group library. MALOCS collapses this by 4 groups to generate the 51 group library. It is suspected if these resonance cross-sections be treated properly in MALOCS because this does not consider the spatial flux distribution, while the group constants at resonance are sensitive to the spatial effect as well as energy spectrum. This energy region is carefully considered in CPM[19] and EPRI-CELL[20] which were developed mainly for the plutonium recycling. These codes perform a 1-dimensional lattice cell calculation from the fine group cross-section library in which the 0.3eV Pu-239 resonance region is finely divided. Similar cencept is tested by generating the 60 group intermediate library (no collapsing for the 0.3eV Pu-239 resonance region) instead of the 51 group to generate 19 group constants through the 1-dimensional lattice cell calculation by XSDRNPM.

The result of this test is compared with that from the 51 group library in Table 3-1. While overall agreement with the critical experiments is a little bit improved no special improvement is found from the pitch dependency viewpoint. If it is judged from the test result the main reason of pitch dependency might be accrued from the resonance region group structure of the 19 group which is not suitable to Pu-239. This group structure is just the same with that of Hansen-Roach[11] 16 groups. W.D. Bromley[21] reported that Hansen-Roach 16 group cross-sections showed very good agreement with UO_2 fuel critical experiments, but 6~8% overprediction for those of MOX fuel. When the fine group library is directly employed no such dependency is found[22]. In order to eliminate this pitch dependent errors by using 19 group constants, an extensive study should be performed for overall reconstruction of the group structure and its test.

Even though this procedure shows relatively wide distribution of prediction and pitch dependency for the MOX fuel benchmark, it is judged that it can be applicable to burned fuel criticality calculation of Kori Unit 1. The moderator-to-fuel ratio of the Kori Unit 1 fuel is 1.605, and this value is safely anchored between the cases (1, 2) and (3, 4) in the benchmark. Thus there is possibility to underestimate the multiplication factors for the burned fuel rack calculation. The effect of plutonium in the burned fuel calculation will, however, be less than a half of those shown in benchmark calculation, since the

Table 3-2. The Result of the Multiplication Factor Calculation Depending on the Fuel Enrichment of Kori Unit 1 (New Fuel)

Enrich. (w %)	KENO-IV	FATAC			
		Homogenized, Bias		Calibrated*, Bias	
1.5	0.8511+0.0037	0.8211	−0.0299	0.8465	−0.0045
2.0	0.9396+0.0039	0.9107	−0.0289	0.9361	−0.0035
2.5	1.0009+0.0039	0.9756	−0.0253	1.0010	0.0001
3.0	1.0583+0.0041	1.0286	−0.0297	1.0540	−0.0043
3.5	1.0894+0.0038	1.0688	−0.0207	1.0942	0.0049
4.0	1.1240+0.0037	1.1016	−0.0223	1.1270	0.0031

* Homogenization effect calibration by KENO-IV for 1.5% fuel (−2.54%)

relative amount of plutonium in the burned fuel is less than a half of that in the experimented one. Simple estimation of the error which would be caused by the introduction of plutonium in the burned fuel might be less than 0.2% if it is evaluated by the relative amount of PuO_2 and the pitch dependency. In the case of C.K. Lee's[13] study on the BAW critical experiments by the same procedure, the average estimation and its standard deviation were 1.00246 and 0.424% for 28 cases, respectively. If the result of the MOX fuel benchmark of this study is added to C.K. Lee's, the average and standard deviation are to be changed to 1.00352 and 0.751% for 34 cases. As a result bias and its uncertainty are increased to 0.106% and 0.652 %, respectively, due to the addition of the MOX fuel benchmark.

3-2. Evaluation of Bias between Calculational Methods

The spent fuel storage pool of Kori Unit 1 is conceptually redesigned to a most compact rack considering burnup credit, and the KENO-IV and FATAC are employed for the multiplication factor calculations so as to evaluate bias.

The calculated result depending on the initial enrichment is shown in Table 3-2. FATAC underestimates 2∼3% less than KENO-IV. It is confirmed that this phenomenon is primarily due to the rack homogenization of FATAC. The rack homogenization effect is −2.54% when it is tested by KENO-IV. If this effect is compensated for the results of FATAC, there is a fairly good agreement with KENO-IV as shown in Table 3-2.

In order to clarify whether FATAC shows an enrichment dependent bias or not, the data are fitted by the linear regression method. Fig. 3-1 shows the result of this fitting. The fitted slope is 0.33% delta-k/%-enrichment, and the correlation coefficient is 0.78. When the independency of bias on enrichment is tested with the correlation coefficient and the number of data, the probability to be independent is less than 5%. Thus it can be easily said that FATAC shows bigger bias as the enrichment reduces. This tendency as well as the relatively big bias may come from the rack homogenization or other

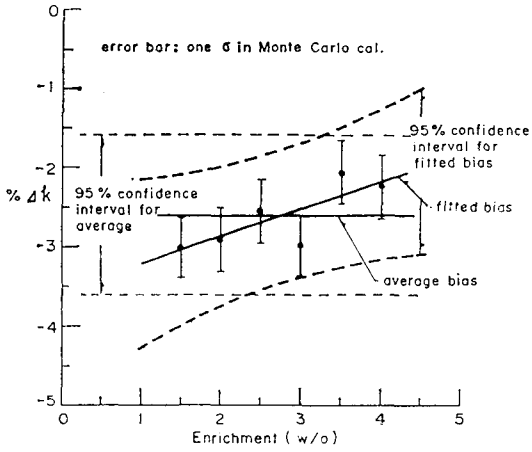

Fig. 3-1. The Variation of Bias Depending on Enrichment and its Confidence Interval

Table 3-3. The Result of the Multiplication Factor Calculation Depending on Burnup of Kori Unit 1 Fuel (3.5% Enriched Fuel)

Burnup (MWD/MTU)	KENO-IV	FATAC			
		Homogenized, Bias		Calibrated*, Bias	
0	1.0894±0.0038	1.0686	−0.0207	1.0942	−0.0049
5000	1.0732±0.0043	1.0493	−0.0240	1.0747	0.0015
10000	1.0434±0.0045	1.0227	−0.0207	1.0481	0.0047
15000	1.0229±0.0039	0.9965	−0.0263	1.0219	−0.0009
20000	0.9974±0.0037	0.9711	−0.0263	0.9965	−0.0009
25000	0.9724±0.0037	0.9456	−0.0267	0.9710	−0.0013
30000	0.9393±0.0037	0.9204	−0.0188	0.9458	0.0066
35000	0.9195±0.0035	0.8956	−0.0238	0.9210	0.0016

* Homogenization effect calibration by KENO-IV for 1.5% fule (−2.54%)

approximations in FATAC. In order to verify these problems, FATAC must be modified so as to treat the poisoned rack explicitly.

When the dependency is not encountered, the average bias is −2.61%, and the standard deviation is 0.4%. As the number of data is 5, it can be treated as t-distributin with 5 degrees of freedom. Then the uncertainty of the average bias with 95% confidence level is 2.571 times the standard deviation, which is 1.03%. In case of linear fitting the degree of freedom is 4, and the standard deviation is determined depending on enrichment by the following equation:

$$s_Y = s_{y\cdot x}\sqrt{1+\frac{1}{N}+\frac{(X-\bar{X})^2}{(N-1)s_x^2}},$$

where N=number of data points,

X=enrichment,

Y=bias,

$\bar{X}=\dfrac{\sum_i X_i}{N}$=the average value of enrichment,

$s_x^2=\dfrac{\sum_i (X_i-\bar{X})^2}{N-1}$=variance of enrichment data,

$s_{yx}^2=\dfrac{N-1}{N-2}(s_y^2-b^2\cdot s_x^2)$=variance of residuals (variance of biases from the fitted line),

$s_y^2=\dfrac{\sum_i (Y_i-\bar{Y})^2}{N-1}$=variance of bias data,

$\bar{Y}$=the averaged bias,

b=fitted slope.

The t-score corresponding to 95% confidence is 2.776 when the degree of freedom is 4.

The result for burned fuel calculation is summarized in Table 3-3 and depicted in Fig. 3-2. As shown in Fig. 3-2 there is no dependency of bias for burnup. The averaged bias and its standard deviation are −2.34% and 0.3%, respectively. The uncertainty with 95% confidence level is 0.71% which is about 0.3%

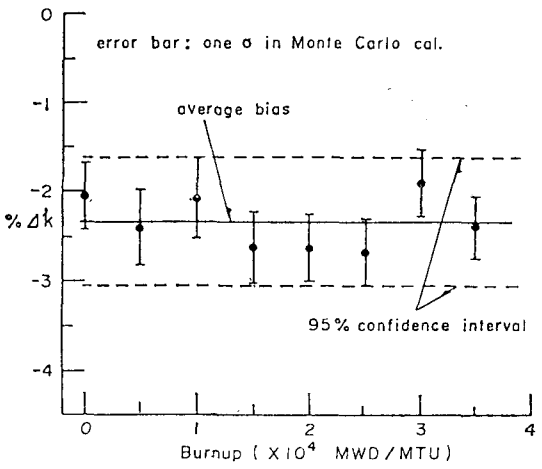

Fig. 3-2. The Variation of Bias Depending on Burnup and its Confidence Interval

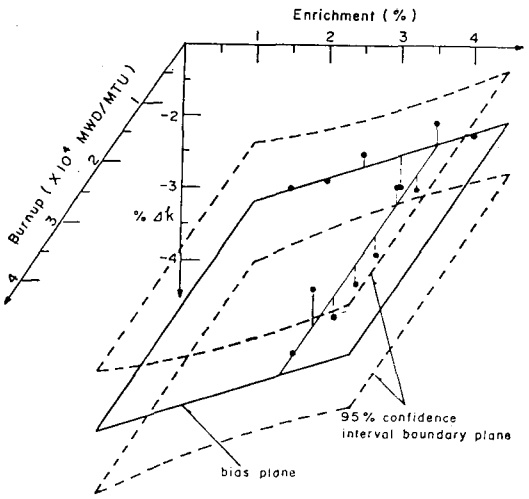

Fig. 3-3. The Variation of Bias Depending on Enrichment and Burnup and its Confidence Interval

smaller than that of enrichment.

As it is verified that the bias is dependent on the fuel enrichment but independent of the burnup, the final fitting is carried out by the linear regression on enrichment for all the data generated. Fig. 3-3 shows the result of this fitting. As the number of data employed in fitting is increased the uncertainty is reduced by about 0.2% compared with the fitting only in terms of enrichment (Fig. 3-1). The fitted slope of bias on enrichment is, however, almost the same for the two fittings. While the value of uncertainty varies with enrichment, its variation is very small. Thus 0.76% may be used conservatively, being independent of enrichment.

In case of the conventional method, the uncertainty of bias cannot be considered since the bias is determined only by one set of calculations on a certain new fuel. Therefore there is no statistical reliability basis for this value. The method proposed in this study, however, gives the best estimated bias with 95% confidence level. If there is no dependency of bias on both enrichment and burnup it then becomes relatively simple to determine the bias; only the arithmetic mean and its uncertainty can be obtained by simple statistics. When there is any dependency on parameter or parameters, as shown in this study, the trend of bias should call for its determination by the fitting of the data for the parameter(s) with dependency. Then the best estimated biases and their uncertainty can eventually be determined.

The quantitative comparison of this method to the conventional one from the viewpoint of the safety margin which should be added due to the use of design code, is as follows:

For 2% enriched fuel that is hypothetically assumed to have the minimum burnup of zero, difference between two values is −2.89%. For the case of conventional method this value will be applied to all the results of FATAC calculation without its uncertainty. When the bias is determined by the procedure studied, however, the absolute value decreases as enrichment increases. The fitted slope is 0.33% delta-k/%-enrichment, and its uncertainty with 95% confidence level falls within 0.64∼0.76% range. In case of 2% enriched fuel the best estimated bias is −2.87% and its uncertainty is 0.7%. While the bias itself is nearly the same with that of the conventional method, considerable uncertainty has to be reflected. The effect of this uncertainty to the total safety margin in the criticality safety analysis is not so significant since the total uncertainty is determined by the combination of every source. If the uncertainty combination of other sources——Monte Carlo method itself, estimated multiplication factor distribution in the benchmark calculation, sensitivity analysis, etc.——is assumed to be 1%, the combination of 0.7% to 1% is 1.22% $[(1^2+0.7^2)^{1/2}]$, and the increment is only 0.22%. As the usual value is greater than 1% in the actual analysis, the increment is smaller than the above one. 0.22% of reactivity worth is comparable to 0.01% of fuel enrichment.

Thus the deficiency in the determination of minimum burnup zero enrichment is only 0.01%.

In the case of minimum burnup determination for the more enriched fuel, the allowable minimum burnup can be reduced as the bias decreases. For the case of 3.5% fuel the best estimated bias is −2.38% and its uncertainty is 0.65%. Thus the bias reduces 0.51%, while the combined uncertainty increases by 0.19%. The total effect is a reduction of 0.32% in total error which is comparable with the reduction of allowable minimum burnup by 400 MWD/MTU.

The above comparison indicates that the new method does not accompany any disadvantage in the minimum burnup which is practically important than the enrichment with zero minimum burnup, even though the reliability basis is established by adding uncertainty.

IV. Conclusion

A computer code for reactor core design shall deliver a biased result when it is applied to the fuel storage facility. Even though the result of the code is unbiased, there is no way of verification in practical problems whether it is correct, except the comparison with a reference value predicted by the least approximative method. In turn the Monte Carlo code as a reference gives a value laid within a certain range because of its inherent stochastic characteristics. In other words, if a value far from being true is produced by chance and it is referred to as the unique standard, an improper bias will result in a highly overconservative or a dangerous, but not perceivable, conclusion. This is the fundamental reason why the conventional bias calculation is not so reliable and at the same time, why the method proposed in this study is necessary.

The procedure proposed herein provides a sound foundation on how to determine the bias between results by two different calculational tools such as Monte Carlo and other practical core design codes. Furthermore it has been shown that there is no or almost negligible disadvantage in the safety margin of reactivity.

There remains one question as to whether the fission products have a strong effect on the bias or not. Even though there is no way to verify this, its effect on the bias can be regarded as negligible since the total effect of fission products on reactivity ($<10\%$) is far less than that of plutonium in the discharged fuel, and the design codes show the very accurate treatment of fission products for core conditions.

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