MODELING OF THE RACE-ISU SUBASSEMBLY TO ANALYZE NEUTRONICS EXPERIMENTAL DATA

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The objective of this work is to provide an understanding of the RACE (Reactor Accelerator Coupling Experiments) experimental results, obtained in the Idaho State University subassembly between May and October 2006, by means of neutron transport computations. MCNPX, the general-purpose Monte Carlo radiation transport code developed by Los Alamos National Laboratory was first employed in order to carry out a critical calculation for assessing the effective multiplication factor and total delayed neutron fraction. Second, we computed correction factors to the experimental reactivity estimates obtained from the area-ratio technique, which is the most appropriate with a pulsed neutron source. Finally, we derived reactivity estimates from simulated flux transients caused by the shutdown of the pulsed neutron source. This way, we were also capable of providing correction factors to the reactivity estimates obtained from experimental transients.

I. INTRODUCTION

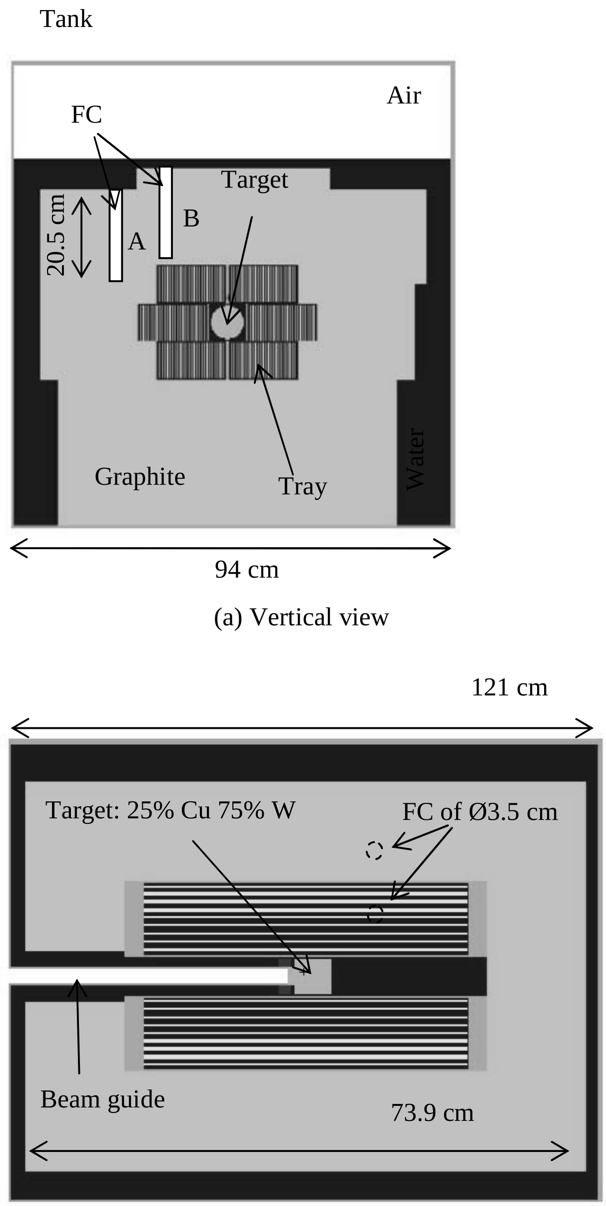
As it is well known, the concept of accelerator-driven systems (ADS) can provide a solution for the nuclear waste management issue by burning minor actinides. The subcritical nature of ADS makes them safer regarding prompt criticality accidents. However, the question of the reactivity control of ADS remains a main concern since their power is inversely proportional to their reactivity level.

The objective of the European Integrated Project EUROTRANS (Ref. 1) of the EURATOM 6th Framework Program is to bring answers to the high level nuclear waste transmutation in ADS. The EUROTRANS experimental activities have been joined into the ECATS domain, namely Experiment on the Coupling of an Accelerator, a spallation Target and a Sub-critical blanket. In the year 2003, Idaho Accelerator Center (IAC), part of Idaho State University (ISU), proposed to couple a subcritical assembly and a linear accelerator (LINAC) in order to contribute to the international research on ADS: It was the beginning of the RACE (Reactor-Accelerator Coupling Experiments) project (Ref. 2 and 3). Although the original goal, which was to carry out such a coupling with a TRIGA reactor, was abandoned, a series of experiments were successfully performed in the ISU thermal subassembly from May to October 2006 in tight collaboration with Commissariat L'Energie Atomique (CEA).

This paper focuses on the analysis of the RACE-ISU experiments, which are presented and described in details in Ref. 4. The analysis was carried out with the use of the well-known and widely-used MCNPX code (Ref. 5). That Monte Carlo code was used to deal with neutron-transport problems only. First, we computed the delayed neutron parameters and more particularly the total delayed neutron fraction also known as the beta effective. The beta effective value was calculated by means of three different methods. Second, we propose correction factors to the area-ratio reactivity estimates in order to take into account spatial effects. Finally, we also simulated a flux transient caused by the shutdown of the pulsed neutron source. From that simulated transient, we could provide another estimate of the subcriticality level. This way, correction factors to that method are proposed as well.

I.A. Computational model of the ISU subassembly

The ISU subassembly, which was water-moderated and graphite-reflected, was coupled to a LINAC providing neutrons by shooting electrons onto a tungsten-copper target. Two uranium-235 fission chambers were



(a) Vertical view

(b) Horizontal view

Fig. 1. The IAC subassembly, the target and fission chambers (FC) emplacements

placed in the graphite reflector (See Fig. 1). More details are given in Ref. 4.

This work needed only the neutron transport capability of the MCNPX code (Ref. 5). Actually not only MCNPX was used but also a modified version of MCNP 4C capable of computing the beta effective (Ref. 6). In addition, one has to note that all simulations were made with the same geometry and material data. Vertical and horizontal cross sections of the assembly are shown in Fig. 1. Two types of calculations were accomplished: criticality calculations, which allowed us to compute both the effective multiplication factor and the beta effective, and source calculations to compute the area-ratio reactivity estimates and to simulate flux transients. We used the standard definition of the reactivity that assumes the existence of the fundamental mode that takes place in the neutronics system with no external source. That choice was motivated by the fact that there is no easy way to compute with the help of any present MCNP versions neutronics effective parameters, like either the effective multiplication factor or the beta effective. However, we simulated all the transients using inhomogeneous models, that is, source-type calculations. As a consequence, it is noteworthy that all the correction factors we calculated are by definition related to the reactivity concept we adopted. In addition, all those source calculations were based on a fission-spectrum neutron source uniformly distributed in the target volume and with an intensity consistent with the real neutron source. Such defined neutron source is thought to be satisfactory in order to account for the neutron-transport-related aspects investigated in this work (more arguments to do that are given in Sect. III.A and IV.A).

One can also note most of the nuclear data we used are those of the endf66c ACE library (Ref. 7). Using that MCNP model of the RACE-ISU subassembly, the effective multiplication factor was estimated to be

$$k = 0.91862 \pm 0.00022 \tag{1}$$

with 68% confidence level.

II. COMPUTATION OF DELAYED NEUTRON PARAMETERS

All reactivity estimates that can be assessed from experimental data are expressed in dollar unit. Given that one dollar exactly amounts to one beta effective, it makes sense that the knowledge of the beta effective is required in order to be able to compare or correct experimental reactivity values by means of computed ones.

II.A. Theoretical aspects

The kinetics equations are derived from the transport equations by integration using a weighting function. As a consequence, all the new so-defined kinetics parameters depend on that weighting function that is generally chosen to be the solution of the adjoint criticality problem. That adjoint flux, which is also named the importance function or importance, and the spectrum energy of the fission neutrons account for what one can call the effectiveness[1]

[1] The original definition of the effectiveness proposed by G. R. Keepin is somewhat different (Ref. 8).

of the delayed neutron fraction. The beta effective β is thus defined by the following ratio between two scalar products:

$$\beta = \frac{\langle \Psi, \mathbf{P}_d \Phi \rangle}{\langle \Psi, \mathbf{P} \Phi \rangle}. \tag{2}$$

The functions Φ and Ψ are the flux and importance, respectively. The operators $\mathbf{P}$ and $\mathbf{P}_d$ are the total and delayed neutron productions, respectively. The spectrum of all the fission neutrons and that of the delayed neutrons are included in the production operators.

A Monte Carlo code like MCNP can easily compute the so-called fundamental delayed neutron fraction β_0:

$$\beta_0 = \frac{\langle \nu_d \mathbf{F} \Phi \rangle}{\langle \nu \mathbf{F} \Phi \rangle}, \tag{3}$$

which is a ratio between two integral terms. $\mathbf{F}$ is the fission operator. The total and delayed neutron yields are denoted ν and ν_d, respectively. Although the prompt and delayed fission neutron spectra are implemented in recent versions of MCNP (Ref. 10), it is more difficult to take into account the importance. The k-ratio method (Ref. 9), also known as the prompt method (Ref. 6), is fairly good at estimating a beta effective value using the following approximation:

$$\beta \cong 1 - \frac{k_p}{k} \tag{4}$$

where k and k_p are the effective multiplication factor for the case when considering all the fission neutrons and the case when considering only the prompt neutrons, respectively. That method that consists in making two eigenvalue calculations seems to be more particularly applicable to the case of a bare and homogeneous core (Ref. 6).

The main point is thus how to directly compute the neutron importance with a continuous energy Monte Carlo code. One can do that by computing an estimate of the iterated fission probability that is shown to be proportional to the importance (Ref. 11). On its turn, that probability is right proportional to the count of fission events resulting from one neutron all along its history, which is defined as the sequence of events taking place from the neutron birth to its death by leakage or non-fission absorption. As shown in Ref. 6, that estimate of the importance can be calculated in the same time as calculating the effective multiplication factor. However, the history definition is somewhat different since a history is assumed to end at any sort of leakage or absorption event. This way, one actually estimates the next-fission probability that is shown to allow a satisfactory approximation of the beta effective (Ref. 6). In this work, we used a modified version of MCNP 4C capable of computing the beta effective using the above described calculation scheme (Ref. 6). In the following, that method will be referred as to the NRG method[2].

We also developed a new method to estimate the beta effective. That method lies on the probability p_p for a fission event to be caused by a prompt neutron and the probability p_d for a fission event to be caused by a delayed neutron. Those two probabilities can be regarded as the prompt and delayed components of the iterated fission probability. The beta effective can be thus rewritten as:

$$\beta = \frac{\beta_0 p_d}{(1-\beta_0) p_p + \beta_0 p_d} = \frac{\beta_0 R}{(1-\beta_0) + \beta_0 R} \tag{5}$$

That method will be referred as to the probability ratio or p-ratio method. The objective is now to be able to compute the probability ratio R. That can be carried out using any standard version of MCNP, that is, without modifying the code. It is however required to perform several criticality calculations and to generate a special source data. The procedure consists in computing two effective multiplication factors k_p and k_d, one for the case of prompt neutrons only and the other one for that of delayed neutrons only, respectively. The ratio of those two multiplication factors provides a value of the sought probability ratio R. The k_p calculation is easily performed using MCNP by turning off the delayed neutrons thanks to the use of the TOTNU NO card (Ref. 7). The main and more subtle point is to compute k_d. Such a calculation has to be made using a source data file, namely a RSSA file (Ref. 7), built from another RSSA file generated by MCNP when running the prompt case. The original RSSA file has to be modified by replacing the spectrum of prompt neutrons with that of delayed neutrons. This way, the two estimations of k_p and k_d, are highly correlated and the final statistical uncertainty is then less than in the case of the k-ratio method.

II.B. Simulation results

The beta effective was estimated using the three described methods, namely the k-ratio, NRG and p-ratio methods. Figure 2 shows the convergence of each estimator. Clearly, the k-ratio estimate exhibits the strongest fluctuations and thus the worst convergence. The beta effective values obtained with each method are reported in Table I. The uncertainty with the p-ratio

[2] NRG is the Dutch company where that method was developed and implemented in MCNP 4C.

method is smaller given that the same number of histories of both prompt and delayed neutrons is taken into account unlike with the NRG method that takes into account less many delayed neutron histories by a factor of beta effective.

Although the p-ratio method would have allowed us to calculate the effective delayed neutron fractions for the six delayed neutron groups defined in the employed nuclear data library (cf. Sect. I), those values were calculated with the NRG method only. The decay constant of each group was calculated by taking

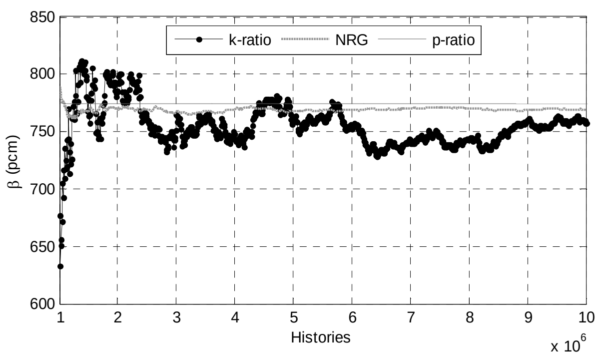


Fig. 2. Convergence of beta-effective estimators. The number of histories per cycle was 10,000 for the k-ratio and NRG calculations and 8,000 for the p-ratio calculation.

TABLE I. Beta effective estimates

Method	β (pcm)	
k-ratio	757.1 ± 33.4[a]	(4.41%)
NRG	768.4 ± 4.2	(0.55%)
p-ratio	771.0 ± 0.4[b]	(0.05%)

a. Standard uncertainty
b. The uncertainty of β_0 is not included.

TABLE II. Delayed-neutron group data. The delayed neutron fractions were calculated with the NRG method.

Group	Fraction (pcm)	Decay constant (s^{-1})
1	26.6 ± 0.8[a]	0.0136
2	141.3 ± 1.8	0.0316
3	131.1 ± 1.7	0.1228
4	297.4 ± 2.6	0.3195
5	121.1 ± 1.7	0.8947
6	50.9 ± 1.1	3.0095

a. Standard uncertainty

into account the isotopic composition of the fuel. All those group data are shown in Table II. For the sake of information, the fundamental delayed neutron fraction β_0 was estimated to be 689.6 ± 2.7[3] pcm with means of the same modified version of MCNP 4C (Ref. 6).

As a result, one can note that all the estimated beta effective values are consistent with each other within 68% confidence interval. The NRG and p-ratio estimates are however very close to each other since the discrepancy is less than 0.3%. The NRG beta-effective estimate will be used as the reference value in the following. This way, the reactivity proposed in Eq. (1) can be expressed in dollar unit:

$$\rho = -11.528 \pm 0.072 \text{ dollars} \tag{6}$$

with 68% confidence level.

III. CORRECTION FACTORS TO THE AREA-RATIO METHOD

The area-ratio method is based on the ratio of the area under the response peak, that is the decaying part of the PNS response, and the area under the background level due to the delayed neutron decay (Ref. 4, 12 and 13). The negative value of that ratio gives an estimation of the reactivity in dollar unit. Assuming the reactivity value ρ derived from the effective multiplication factor k given by a criticality calculation is considered as a reference subcritical level, one can calculate the theoretical ratio between that reference reactivity and a calculated reactivity estimate given by the area-ratio method. This ratio plays the role of a correction factor accounting for both reactivity bias and spatial effects.

III.A. Theoretical aspects

Theoretically, the response to a pulse train is equivalent to that of a unique pulse $S\,\delta(t)$, where $\delta(t)$ is the Dirac function. The flux $\phi(t)$ that is the response to such a pulse is then described by the following transport equation (Ref. 14, 15 and 16):

$$\frac{1}{v}\frac{\partial}{\partial t}\phi(t) = \mathbf{P_p}\phi(t) - \mathbf{L}\phi(t) + S_d(t) + S\delta(t) \tag{7}$$

$\mathbf{L}$ is the migration and loss operator. The operator $\mathbf{P}_p$ represents the prompt neutron production. The delayed neutron term is denoted by S_d and v is the neutron speed. For the sake of simplicity, the spatial and energy flux dependence is not explicitly indicated. Since the area-method deals with time-integrated terms only, it is more relevant to integrate Eq. (7) over time (Ref. 14, 15 and 16)

[3] The uncertainty is expressed in terms of standard uncertainty.

knowing that the flux is null in the beginning and goes back to zero at infinite time:

$$0 = \mathbf{P}\Phi - \mathbf{L}\Phi + S \tag{8}$$

where Φ is the time-independent flux or fluence. With the use of the adjoint flux defined after the following equation

$$\mathbf{L}^{*}\Psi - \frac{1}{k}\mathbf{P}^{*}\Psi = 0 \text{ or } \mathbf{L}^{*}\Psi - (1-\rho)\mathbf{P}^{*}\Psi = 0\,, \tag{9}$$

one obtains the important relationship:

$$\langle \Psi, \mathbf{P}\Phi \rangle = \frac{\langle \Psi S \rangle}{-\rho}\,. \tag{10}$$

One introduces the time-integral response A of a detector characterized by a cross section Σ_D and proportional to the weighted neutron production integral:

$$A = \varepsilon \langle \Psi, \mathbf{P}\Phi \rangle\,, \tag{11}$$

where the term ε is called the detector efficiency. Eq. (11) is then rewritten

$$A = \frac{\varepsilon \langle \Psi S \rangle}{-\rho}\,. \tag{12}$$

A corresponds to the area under an experimental histogram (Ref. 4). The flux contribution Φ_p due only to the prompt neutrons satisfies the following equation:

$$0 = \mathbf{P}_p \Phi_p - \mathbf{L}\Phi_p + S \tag{13}$$

that can be rewritten as

$$\mathbf{P}\Phi_p - \mathbf{L}\Phi_p - \mathbf{P}_d \Phi_p + S\,. \tag{14}$$

The latter equation can be finally turned into the following integral equation:

$$\langle \Psi, \mathbf{P}\Phi_p \rangle = \frac{\langle \Psi S \rangle}{\beta - \rho} \tag{15}$$

by assuming

$$\beta = \frac{\langle \Psi, \mathbf{P}_d \Phi \rangle}{\langle \Psi, \mathbf{P}\Phi \rangle} \cong \frac{\langle \Psi, \mathbf{P}_d \Phi_p \rangle}{\langle \Psi, \mathbf{P}\Phi_p \rangle}\,. \tag{16}$$

The prompt contribution A_p to the time-integral detector response is thus given by

$$A_p = \frac{\varepsilon \langle \Psi S \rangle}{\beta - \rho}\,. \tag{17}$$

It is noteworthy the detector efficiency ε has to be assumed the same as in Eq. (10). Combining Eq. (12) and (17), one obtains the so-called area-ratio relationship

$$\frac{\rho}{\beta} = -\frac{A_p}{A_d} \tag{18}$$

given that

$$A = A_p + A_d\,, \tag{19}$$

where A_d is the delayed neutron contribution. Eq. (19) comes directly from

$$\Phi = \Phi_p + \Phi_d \tag{20}$$

and from the assumption, as previously mentioned, that the detector efficiency ε is independent of the flux type.
The MCNP code allows us to estimate the reactivity in dollar unit as expressed in Eq. (18). More precisely, the following relationship has to be employed:

$$\frac{\rho}{\beta} = \left(1 - \frac{A}{A_p}\right)^{-1}. \tag{21}$$

Two fixed-source calculations have to be carried out with means of the TOTNU card (Ref. 7): a first calculation taking into account the prompt and delayed neutrons (TOTNU) and a second one taking into account only the prompt neutrons (TOTNU NO). It is important to note the area-ratio relationship (Eq. 18) is independent of the source term S. This is the reason why we did not perform the sophisticated and more time-consuming simulation of the real neutron source due to accelerated electrons impinging upon the tungsten-copper target.

III.B. Simulation results

The MCNP results are displayed in Tables III and IV. One can note that an uncertainty of 1% in the time-integrated detector responses A and A_p yields an uncertainty of 17% in the area-ratio reactivity value. As a consequence, a Monte Carlo calculation of the correction factor to the area-ratio method requires a significant number of histories (without any biasing technique). The correction factors at the detector locations A and B are approximately equal to 6.0% and 7.7%, respectively. However, those values are given with a standard uncertainty of almost 2.5%. Although more accurate results could be obtained, the reactivity value provided by the area-ratio method at the two studied detector locations is clearly greater than the reference reactivity (Eq. 6) derived from a criticality calculation.

TABLE III. Time-integrated detector responses simulated with MCNP. The number of histories was of 8 millions.

Response	Detector A[a]
A	0.90512 ± 0.00129[b] (0.14%)
A_p	0.82891 ± 0.00120 (0.14%)
Response	Detector B
A	1.07865 ± 0.00146 (0.14%)
A_p	0.98651 ± 0.00135 (0.14%)

a. See Fig. 1.a
b. Standard uncertainty

TABLE IV. Area-ratio reactivity estimates and correction factors

Detector	Reactivity estimate
A	-10.877 ± 0.262 (2.41%)
B	-10.706 ± 0.241 (2.25%)
Detector	Correction factor
A	1.0598 ± 0.0264 (2.49%)
B	1.0767 ± 0.0251 (2.34%)

b. Standard uncertainty

IV. CORRECTION FACTORS TO THE TRANSIENT METHODS

As explained in Ref. 4, the well-known inverse kinetics method can be used to estimate the reactivity from a transient, also called beam trip, caused by the instantaneous shutdown of a non-steady neutron source. The resulting transient is then fully driven by the delayed neutrons and its shape, which is of high importance to estimate the reactivity value (Ref. 17), does not depend on the source type at all. In the same manner as in the case of the area-ratio technique, one can compute a correction factor accounting for both reactivity bias and spatial effects.

IV.A. Theoretical aspects

The objective is here to simulate such a beam trip. That sort of simulation can be carried out using the code MCNP. The reactivity is thereafter estimated with means of the inverse kinetics (IK) method. It is important to note that in the case of a simulated transient, the non-linear fitting (NLF) method (Ref. 4 and 17) can be also appropriate given that the simulated flux amplitude before the transient can be assumed to be steady.

IV.B. Simulation results

Since the investigated subcriticality level is very far from critical, the transient due to the delayed neutrons is very short and the counting statistics is low. As a consequence, it is likely that the reactivity estimation with the use of transient methods is sensitive to the analysis range and especially to the upper bound of that range. That effect is shown in Fig. 3 and 4. The IK method appears to be more sensitive to that upper bound and the analysis domain is smaller than that of the NLF method.

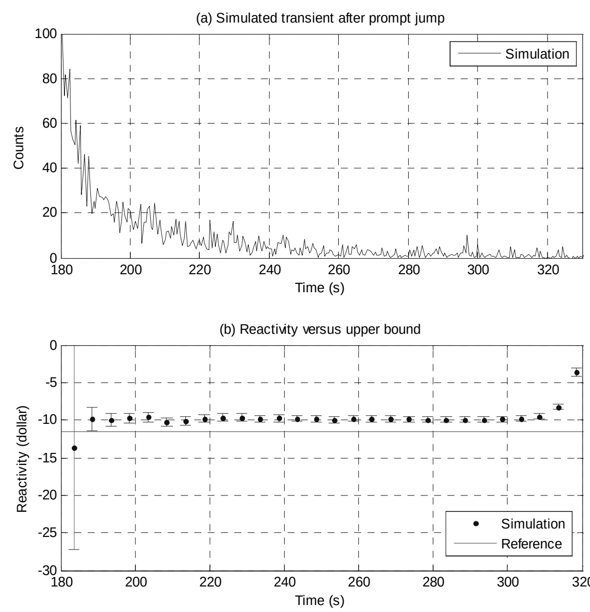


Fig. 3. Reactivity estimates using the inverse kinetics method. The transient was obtained at the detector location B with fixed source MCNP calculation of 4.5 millions of histories. The error bars represent one standard uncertainty. The analysis range spans from the perturbation time t_p fixed at 180 s to the variable upper bound t_1 (Ref. 17).

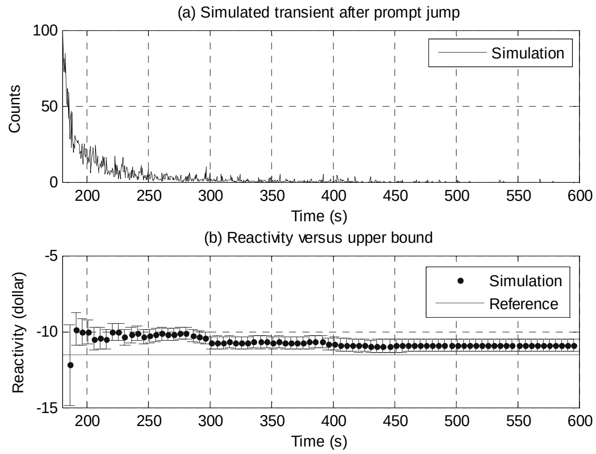


Fig. 4. Reactivity estimates using the non-linear fitting method. Same comments as in Fig. 3.

TABLE V. Inverse-kinetics reactivity estimates and correction factors.

Detector	Reactivity estimate
A	-9.779 ± 0.566 (5.79%)
B	-10.122 ± 0.628 (6.20%)
Detector	Correction factor
A	1.1788 ± 0.0686 (5.82%)
B	1.1388 ± 0.0710 (6.24%)

a. Standard uncertainty
b. The analysis range is [t_p t_1]=[180 s 213.5 s]

TABLE VI. Non-linear-fitting reactivity estimates and correction factors.

Detector	Reactivity estimate
A	-9.625 ± 0.216 (2.25%)
B	-10.765 ± 0.415 (3.85%)
Detector	Correction factor
A	1.1976 ± 0.0279 (2.33%)
B	1.0709 ± 0.0418 (3.90%)

a. Standard uncertainty
b. The analysis range is [t_p t_1]=[180 s 300 s]

The reactivity estimates and the correction factors to the IK and NLF methods are reported in Tables V and VI, respectively. The overall reactivity uncertainty takes into account the uncertainty of the counting statistics and the uncertainties of the delayed neutron parameters. The latter uncertainties are those proposed in Ref 18. One can note that the overall uncertainty is more significant in the case of the IK method. For that method, the main uncertainty contribution is due to the counting statistics uncertainty that is around 5%. That comes from the fact that the analysis range is smaller for the IK method. As a result, the number of reactivity points available to estimate the final reactivity value is less important (Ref. 17). It is noteworthy to mention the counting statistics uncertainty of the IK method is calculated using a Monte Carlo method (Ref. 17).

The correction factors depend on both the detector location and applied methods. Its value goes from 7.7% to 19.7%. However, since the experimental reactivity estimates could be obtained only with the IK method (Ref. 4), only the correction factors to that method have to be finally considered.

V. CONCLUSION

Using a MCNP model of the RACE-ISU subassembly, we were capable of calculating the correction factors to the reactivity estimation techniques employed during the RACE-ISU experimental program, namely the area-ratio and beam-trip techniques. One recalls that those correction factors account for both spatial effects and reactivity bias.

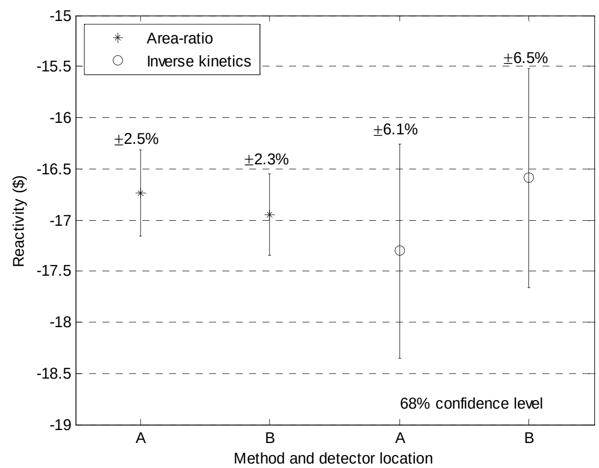


Fig. 5. Experimental reactivity estimates after calculated correction

Prior to that work, we had to compute the effective delayed neutron fractions with MCNP. In order to accomplish that task, we used three different methods. Among them, the NRG and p-ratio method provided consistent estimates of the total delayed neutron fraction, that is, the so-called beta effective. The beta-effective value given by the NRG method was equal to 768.4 ± 4.2 pcm.

The corrected experimental reactivity estimates are shown in Fig. 5. After correction, all of those reactivity estimates, which are consistent with each other within 68% confidence interval, are smaller. In other words, the subassembly appears to be more subcritical after correction. It is also important to note that the area-ratio method allowed us to obtain smaller overall uncertainties of 2.5% at most after correction. The beam trip or transient techniques furnished less satisfactory results in terms of accuracy. That comes from the fact that the subassembly was so much subcritical that the transient caused by the shutdown of the pulsed neutron source was too short to successfully perform a transient analysis with the inverse kinetics method, the only one applicable with a significantly non-steady source.

This work confirms the area-ratio technique seems to be the best candidate for reactivity calibration even after calculated corrections.

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