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MCNP–REN: a Monte Carlo tool for neutron detector design

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Abstract

The development of neutron detectors makes extensive use of the predictions of detector response through the use of Monte Carlo techniques in conjunction with the point reactor model. Unfortunately, the point reactor model fails to accurately predict detector response in common applications. For this reason, the general Monte Carlo code developed at Los Alamos National Laboratory, Monte Carlo N-Particle (MCNP), was modified to simulate the pulse streams that would be generated by a neutron detector and normally analyzed by a shift register. This modified code, MCNP-Random Exponentially Distributed Neutron Source (MCNP–REN), along with the Time Analysis Program, predicts neutron detector response without using the point reactor model, making it unnecessary for the user to decide whether or not the assumptions of the point model are met for their application. MCNP–REN is capable of simulating standard neutron coincidence counting as well as neutron multiplicity counting. Measurements of mixed oxide fresh fuel were taken with the Underwater Coincidence Counter, and measurements of highly enriched uranium reactor fuel were taken with the active neutron interrogation Research Reactor Fuel Counter and compared to calculation. Simulations completed for other detector design applications are described. The method used in MCNP–REN is demonstrated to be fundamentally sound and shown to eliminate the need to use the point model for detector performance predictions. © 2001 Elsevier Science B.V. All rights reserved.

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1. Introduction

The development of neutron detectors for nondestructive assay applications in nuclear safeguards, as well as nuclear waste characterization, makes extensive use of the predictions of detector response through the use of Monte Carlo computer modeling techniques in conjunction with the point reactor model. Unfortunately, the point reactor model fails to accurately predict detector response in commonly encountered applications for both Neutron Coincidence Counting (NCC) and neutron multiplicity assays. This forces the detector designer to make a careful evaluation of

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the point model assumptions and how their use affects the simulation results.

The point model fails to accurately predict detector response in commonly encountered applications if certain physical conditions are not met. The following assumptions are required to be valid.

- Multiplication, (α, n) source neutron production rate, spontaneous fission source neutron production rate, detection efficiency, and detector die-away time are constant across the sample volume.
- Multiplication and detection efficiency are energy independent.
- Induced fissions occur at the same time as the spontaneous fission or (α, n) source neutron production. This is commonly referred to as the superfission concept.
- There is no neutron return from the detector to the sample volume.

The instrument designer or researcher must perform a careful analysis of each of these assumptions before using the point model. It is frequently very difficult to do this without extensive knowledge of the system and the material to be assayed, both of which may not be available to the designer. These assumptions may be valid for one combination of assay system and material to be assayed while not being valid for different combinations of other materials with the same assay system. Therefore, use of the point model may limit the versatility and accuracy of the detector design. Its use also requires extensive modeling experience to avoid unanticipated biases in the answer.

Past efforts have been made to modify or develop new Monte Carlo codes or to use alternative analytical techniques to predict neutron detector response [1–5]. These efforts have either not been designed to model standard coincidence or multiplicity techniques, have relied on the assumptions inherent to the point model, or have not been demonstrated to meet the current simulation needs for nondestructive assay instrument design and calibration in the safeguards and nuclear waste assay communities.

For these reasons the general purpose Monte Carlo code, Monte Carlo N-Particle (MCNP, version 4a) developed at Los Alamos National Laboratory, was modified to simulate the pulse streams that would be generated by a neutron detector and typically analyzed by a shift register. This modified code, MCNP–Random Exponentially Distributed Neutron Source (MCNP–REN), along with the Time Analysis Program (TAP), which simulates the pulse processing typical of a shift register-based coincidence circuit, allows the prediction of neutron detector response without using the point reactor model, thus making it unnecessary for the user to decide whether or not the assumptions of the point model are met for their particular application. MCNP–REN and TAP are capable of simulating standard, shift register-based, NCC as well as neutron multiplicity counting. Minor modifications of TAP would be all that are required to simulate other neutron coincidence systems or detector signal analysis techniques.

Simulations and measurements of mixed oxide (MOX) fresh fuel made using the Underwater Coincidence Counter (UWCC) and measurements of highly enriched uranium (HEU) reactor fuel taken with the active-neutron Research Reactor Fuel Counter (RRFC) are compared with MCNP–REN calculations in the following sections. These comparisons demonstrate that the method used in MCNP–REN is fundamentally sound and that it eliminates the need to use the point model for detector performance predictions.

2. Code description

The general purpose Monte Carlo code, MCNP, developed at Los Alamos National Laboratory, has been modified to simulate the timing of the pulse streams that would be generated by a neutron detector and typically analyzed by a shift register, such as is commonly used for NCC or neutron multiplicity counting.

The first modification required was to add accurate representations of the source spontaneous fission multiplicity distributions. The source distribution sampled by MCNP–REN can be an isotopic source or an (α, n) source. The distributions for ^{240}Pu, ^{252}Cf, and ^{244}Cm are all available for use as the isotopic source. MCNP–REN will determine an effective multiplicity distribution to account for the mixed source when this option is used. The energy-dependent, induced-fission multiplicity distributions were also modified for the common isotopes of interest. Specifically, the energy-dependent distributions as published by Zucker and Holden for ^{235}U, ^{238}U, and ^{239}Pu were added to the code [6]. These distributions are sampled directly and the average number of neutrons emitted per fission is therefore the same as that published by Zucker and Holden for these three isotopes. These isotopes comprise the vast majority of fissions in systems of interest for nuclear safeguards or waste assay applications.

An alternative use of the Zucker and Holden data has been proposed that would allow consistency between neutron multiplicity distributions and the evaluated nuclear data files. By fitting the Zucker and Holden data as a function of the average number of neutrons from fission, neutron multiplicity distributions can be generated from the average number of neutrons in the evaluated nuclear data files thereby eliminating the need to modify the evaluated nuclear data. Such a modification could be incorporated into MCNP–REN in the future. The goal of MCNP–REN development was to provide the safeguards community with a useful tool to predict doubles and triples, results that cannot already be accurately generated by the standard version of MCNP. The ultimate approach would be to produce an evaluated set of transport cross-sections coupled to measured multiplicity distributions that can predict unbiased singles, doubles, *and* triples for a large set of detector/sample benchmarks. Such an evaluation is clearly beyond the scope of the current work.

For a given source event, MCNP–REN will sample the effective multiplicity distribution for the number of neutrons "born," see Fig. 1, then tag those neutrons with a birth time that is based on the sampling of the elapsed time, Δt, since the previous source event.

$$\Delta t = \frac{\bar{v}}{S}\ln[1 - ran\#]$$

where $\bar{v}$ is the average multiplicity of the effective multiplicity distribution, S is the source neutron production rate, Δt is the time interval between source events, and *ran#* is a random number uniformly sampled between 0 and 1.

A source neutron is then tracked until it is absorbed while other neutrons born at the same time are banked for subsequent tracking. If the absorption is an (n,p) reaction in the active region of a ^{3}He detector, the event time is written to an output file, and if the absorption results in a fission, the neutrons resulting from the reaction are tagged with the event time and stored in the bank (Fig. 1). For a fission event in ^{235}U, ^{238}U, or ^{239}Pu, the energy-dependent multiplicity distribution added to the code is sampled while for other isotopes, such as ^{233}U or ^{241}Pu, the standard MCNP ACENU subroutine sampling is used. MCNP–REN is not intended to be used to model ^{233}U systems. This approach can create a bias in the predicted doubles and triples rates for some problems seen in safeguards where ^{241}Pu fission contributes. However, in practice this bias is small as demonstrated in the MOX fuel validation to be discussed below and problems where ^{241}Pu may contribute a significant bias are rare. After writing to an output file, MCNP–REN returns to the bank of source particles and tracks the next particle to its endpoint. This is repeated until all particles in the bank have been tracked.

The ^{3}He (n,p) reaction times are stored in one of two types of output files. One type, the "total" file, contains reaction times for all detectors with active regions specified in the MCNP input deck while the other type, "cell" files, record the reaction times for each individual detector tube as defined in the MCNP input deck. This allows the user to examine detector performance on a tube-by-tube basis or to add additional modeling details, such as pre-amplifier dead time, at a later date without requiring any further runs of the code.

A standard MCNP input file can be used by MCNP–REN with only three modifications. The modifications needed are: (1) addition of an "IDUM" card with entries to define the source (^{240}Pu, ^{252}Cf, ^{244}Cm, α-n), the cell numbers of the active ^{3}He regions, and other control parameters; (2) addition of a "RDUM" card with the

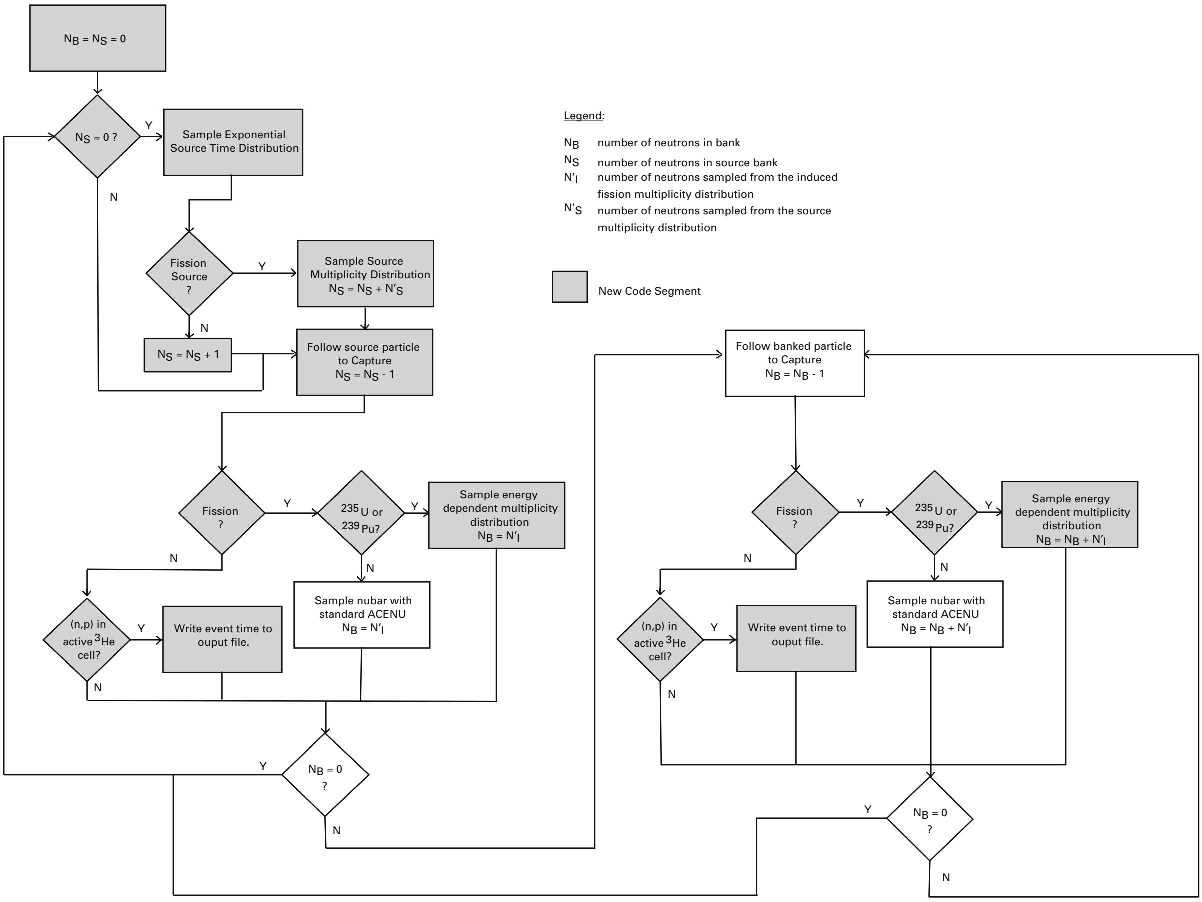

Fig. 1. Flowchart describing modifications to the Monte Carlo code MCNP4a.

spontaneous fission neutron production rate and the α-n production rate; and (3) a MCNP standard "PHYS" card to set the tracking to the analog mode.

The MCNP–REN output files are processed by TAP, a program that mimics a shift register and multiplicity analyzer. TAP produces a multiplicity distribution; singles, doubles, and triples count rates for the detector; estimates of the count rate uncertainties; as well as an estimate of the detector's die-away time (Fig. 2). TAP also has the capability to model detector, pre-amp, and shift register dead time as separate parameters.

3. Example of bias improvement

One example of an application of MCNP and the point model that leads to significant bias in the simulation results is a modeling effort completed by Phillip Rinard and Howard Menlove at Los Alamos National Laboratory [7]. In their work, an Active Well Coincidence Counter (AWCC) was modeled in a configuration commonly used to measure the uranium linear density in long fuel elements. The application of the point model to the standard MCNP results to calculate the coincidence rate resulted in an answer that was biased by nearly 50%.

```
Number of events analyzed:          446436

                    DATA FOR GATE 1                                    DATA FOR GATE 2
                    ---------------                                    ---------------

  Bin Number       R+A Gate          A Gate          Bin Number       R+A Gate          A Gate
      0             34510            39574               0              3595             4391
      1             81914            89682               1             15334            17961
      2            102324           105831               2             34704            37807
      3             90352            88831               3             54398            58995
      4             63666            59182               4             66519            69554
      5             37804            33942               5             68848            68979
      6             19942            16778               6             61204            60288
      7              9309             7607               7             48598            46145
      8              4095             3126               8             34903            32373
      9              1575             1253               9             23677            21206
     10               610              416              10             15097            12843
     11               213              127              11              9038             7547
     12                76               46              12              5160             4179
     13                26               26              13              2715             2147
     14                11                8              14              1382             1080
     15                 7                3              15               650              524
     16                 1                3              16               313              226
     17                 0                0              17               149               97
     18                 0                0              18                66               53
     19                 0                0              19                37               23
     20                 0                0              20                26               11
     21                 0                0              21                14                5
                                                        22                 4                1
 Gate Length:          64.000           usec             23                 4                0
 Count Time:           10.870          seconds           24                 0                0
 Singles:           41071.547       counts/sec
 Doubles:            6613.896       counts/sec     Gate Length:         128.000            usec
 Triples:            1454.317       counts/sec     Count Time:           10.870          seconds
                                                   Singles:           41071.547       counts/sec
          Corrected rates for gate 1               Doubles:            8957.478       counts/sec
          --------------------------               Triples:            3509.088       counts/sec

     Singles:       41071.547       counts/sec                Die Away Time:  61.6872 usec
     Doubles:       10754.140       counts/sec        Doubles Gate Fraction:   0.6150
     Triples:        3844.994       counts/sec        Triples Gate Fraction:   0.3782

     Uncertainty Estimates (gate 1)
     ------------------------------
     Singles:           0.128          percent
     Doubles:           2.976          percent
     Triples:          28.049          percent
```

Fig. 2. Sample of the type of data generated by TAP from the data files created by MCNP–REN. These data were generated for a 17×17 array of MOX rods with a linear effective ^{240}Pu loading of 6.65 g/cm in unborated water. $R + A$ indicates the reals plus accidentals shift register gate and A indicates the accidentals gate. (See Ref. [1] for further information on coincidence counting.)

Rinard and Menlove then segmented the standard MCNP model to reduce the bias. In this approach the fuel assembly was modeled in six segments and the results combined to determine the response for the entire assembly. Despite the added complexity and cost, this is a common approach for detector designers since the segments are a better approximation to the point model assumptions. Their results still showed a bias of $\approx 30\%$.

They then used MCNP–REN, which allowed the use of the original MCNP model without segmentation. The simulation results still showed a positive bias, but it was reduced to $\approx 10\%$. A normalization factor would still be needed to apply the modeling results to field applications using the AWCC in the configuration modeled, but the use of MCNP–REN substantially reduces the size of the normalization factor and subsequently the uncertainty introduced by its use.

4. MOX experimental comparison

Measurements of MOX fresh fuel at the Venus critical facility in Mol, Belgium, were made using the UWCC developed at Los Alamos National Laboratory [8,9]. The UWCC consists of eight ^{3}He tubes embedded in two polyethylene blocks, which are wrapped in cadmium and placed in a watertight stainless steel casing. Two of the series of experimental measurements made in Belgium were modeled using MCNP–REN [10]. Both series used 17×17 arrays of MOX fuel that was 97.30% UO_2 and 2.70% PuO_2 by weight. Uranium enrichment was 2.00% [11]. The detector and fuel were submerged in a tank of water that was unborated for the first series (Fig. 3). In the second series, Borax soap was added to the water to raise the ^{10}B concentration to a nominal concentration of 2250 mg/l.

Due to the high multiplication in a pressurized water reactor MOX fuel assembly, the point model

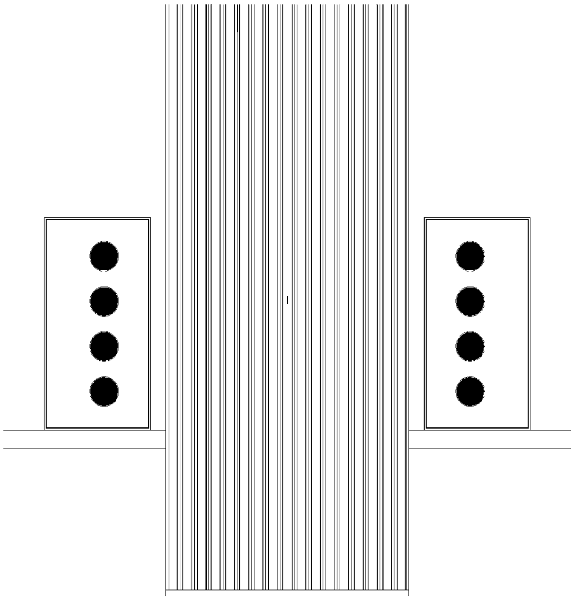

Fig. 3. MCNP model showing a slice through both heads of the detector, each with four ^{3}He detector tubes, with the 17×17 fuel bundle centered between the heads.

assumptions are invalid. The application of the point model to this simulation problem was not attempted beyond some initial modeling for detector design purposes that were completed prior to the development of MCNP–REN. Subsequent to the fabrication of the detector, MCNP–REN initial development was complete and it was used for the more detailed modeling required for detector calibration.

The predicted detector response for the doubles rate (Fig. 4 and Table 1) for the first series was in excellent agreement with the experimentally determined detector response. The average relative error between the prediction and experimental measurement was $<1.3\%$.

The predicted response for the borated series of measurements was slightly lower than that observed in the experimental series (Fig. 4). In this series, the average relative error was 14.1%. A careful examination of these experimental series and others performed in Belgium [10] have led us to conclude that the solubility limits for Borax had

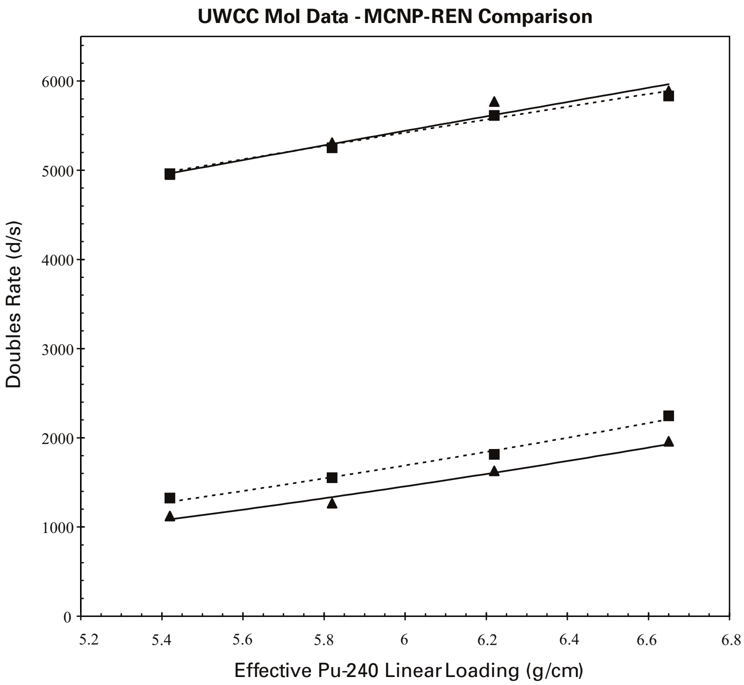

Fig. 4. Comparison of MCNP–REN simulation (▲) and the experimental measurement (■) of mixed oxide fresh fuel using the UWCC. Doubles rates >4000 doubles/s were obtained in unborated water while those below this were obtained in borated water (2250 mg/l of boron). The experimental boron concentrations actually achieved were most probably <2250 mg/l used in the calculation owing to the solubility limit of Borax being locally exceeded. Borax precipitation was observed in the tank.

Table 1
Data comparison for UWCC measurements and MCNP–REN Model

# of rods	Effective ^{240}Pu (g/cm)	MCNP–REN prediction (dps)	MCNP–REN uncert (dps)	Exp meas (dps)	Exp uncert (dps)
0 mg/l Boron					
264	6.65	5891	190	5834	21
247	6.22	5771	162	5614	10
231	5.82	5310	188	5253	24
215	5.42	4952	144	4961	23
2250 mg/l Boron					
264	6.65	1962	53	2247	13
247	6.22	1630	65	1814	13
231	5.82	1268	59	1552	5
215	5.42	1123	46	1326	11

been exceeded, and the actual boron concentration in the tank was <2250 mg/l. This conclusion was corroborated by the observation of Borax precipitates in the tank. Because the MCNP–REN model used the nominal boron concentration, the results show the model predicting count rates lower than that observed experimentally.

In both the borated and unborated cases, the observed trend in the predicted detector response as a function of effective ^{240}Pu loading was in excellent agreement to that observed in the experimental measurements.

5. HEU experimental comparison

The RRFC is an underwater active neutron coincidence counter installed at the Reciepts Basin for Offsite Fuel facility at the Savannah River Site. This detector was developed at Los Alamos National Laboratory to assay the remaining ^{235}U content in Material Test Reactor (MTR) spent-fuel assemblies. The RRFC contains two AmLi neutron sources and 12 ^{3}He tubes (4 atm fill pressure), each with its own preamplifier (Fig. 5). Above the surface of the spent-fuel pool is a Portable Shift Register counting electronics module and a computer running a modified version of the Los Alamos National Laboratory NCC code called RRFC.

Typical assemblies are HEU, 93% enriched, aluminum clad, and contain 80–250 g of uranium. These assemblies have low multiplication and, as the active sources interrogate only a small region of the fuel, the point model assumptions are quite good. The point model was used for the detector design and calibration with excellent result [12]. The authors will provide a copy of the MCNP input deck upon request for these simulations to readers interested in repeating the simulations with MCNP or MCNP–REN. Further design information is available in the hardware manual [13].

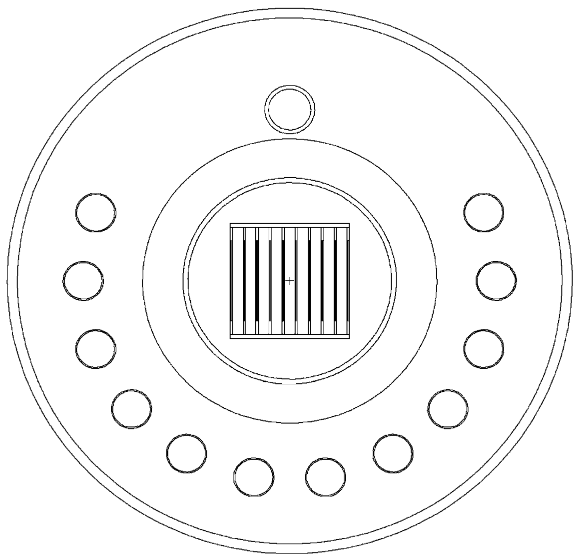

Fig. 5. Horizontal slice through the RRFC with an IAN-R1 fuel assembly centered in the detector. The slice shows the 12 detector tubes as well as the position of one of the AmLi sources located at the top of the image.

Table 2
MCNP–REN versus experimental measurements for the five HEU MTR fresh fuel calibration assemblies measured with the RRFC

Mass ^{235}U (g)	98.0	123.0	147.2	171.7	184.0
Experimental results doubles rate (dps)	159.2 ± 1.1	201.6 ± 0.6	242.9 ± 1.0	274.5 ± 1.2	290.3 ± 1.6
MCNP–REN results doubles rate (dps)	164.3 ± 4.4	200.3 ± 5.2	243.9 ± 4.3	273.6 ± 4.4	284.6 ± 5.7

Recently, it became necessary to develop a calibration curve for the RRFC that was specific to the HEU fuel used at the IAN-R1 reactor in Columbia [14]. We decided to use MCNP–REN to calculate the new calibration, but first a comparison was made between previous RRFC experimental measurements and MCNP–REN predictions for HEU MTR fresh fuel calibration standards at Los Alamos. As shown in Table 2, there is excellent agreement between MCNP–REN model predictions and the experimental data. The average unsigned difference between the measurements and MCNP–REN results is $<1.3\%$ for the mass range of 95–185 g.

6. Conclusions

MCNP–REN was developed by modifying the general Monte Carlo code MCNP, version 4a. This modification simulates the timing of neutron events in ^{3}He-filled detectors. The processing of those simulated times by a second program, TAP, allows one to determine the detector response as would be experimentally observed using shift registers.

This code has been used to model several neutron detection systems at Los Alamos National Laboratory. Two of these modeling efforts that included experimental data comparisons were described. Calculated responses for the MOX measurement exercise were in excellent agreement with measurements for the unborated series of experiments. Calculated trends as a function of effective ^{240}Pu loading were also in excellent agreement for both borated and unborated experiments. These results, as well as the excellent agreement observed between MCNP–REN predictions and calibration measurements made with the RRFC, leads us to conclude that the method used in MCNP–REN is fundamentally sound. MCNP–REN eliminates the need to rely on the point model and its assumptions for detector performance predictions and can be used with the standard MCNP input deck with only minor modifications.

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