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MCNP-REN - A MONTE CARLO TOOL FOR NEUTRON DETECTOR DESIGN WITHOUT USING THE POINT MODEL

Author(s):

M. E. Abhold and M. C. Baker .

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MCNP-REN – A Monte Carlo Tool for Neutron Detector Design Without Using the Point Model

Mark E. Abhold
Safeguards Science and Technology Group
Los Alamos National Laboratory
Los Alamos, NM 87545

Michael C. Baker
Environmental Science and Waste Technology Group
Los Alamos National Laboratory
Los Alamos, NM 87545

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 N-Particle code (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 (TAP) predict 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 MOX fresh fuel made using the Underwater Coincidence Counter (UWCC) as well as measurements of HEU reactor fuel using the active neutron Research Reactor Fuel Counter (RRFC) are compared with calculations. 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.

INTRODUCTION

The development of neutron detectors for nondestructive assay makes extensive use of the predictions of detector response, both coincidence and multiplicity counting, 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. This forces the detector designer to make a careful evaluation of the point model assumptions and how their use effects the simulation results.

The reason that the point model fails to accurately predict detector response in commonly encountered applications is that its use requires that certain physical conditions are met. The following assumptions are required to be valid:

- Multiplication, (alpha,n) source neutron production rate, spontaneous fission source neutron production rate, detection efficiency, and 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 (alpha,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 prior to using the point model. It is frequently very difficult to do this without extensive knowledge of the system and material to be assayed, and it may require 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[1-4]. 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 , neutron coincidence counting as well as neutron multiplicity counting. Minor modifications of TAP would be all that is required to simulate other neutron coincidence systems or detector signal analysis techniques.

Measurements of mixed oxide (MOX) fresh fuel made using the Underwater Coincidence Counter (UWCC) as well as measurements of highly enriched uranium (HEU) reactor fuel using the active neutron Research Reactor Fuel Counter (RRFC) are compared with MCNP-REN calculations below. 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.

CODE DESCRIPTION

The general purpose Monte Carlo N-Particle code, (MCNP, version 4a) 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 neutron coincidence counting 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, (alpha,n) source, or a mixture of the two in a ratio specified by the user. 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 for ^{235}U, ^{238}U, and ^{239}Pu were added to the code.

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.

$$\Delta t = \frac{\overline{\nu}}{S} \cdot \ln[1 - ran\#]$$

where ν 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.

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 the standard ACENU subroutine sampling, as in the unmodified code MCNP, is used. 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. This allows the user to examine detector performance on a tube-by-tube basis or to add additional modeling details such as pre-amplifier deadtime at a later date without requiring any further runs of the code.

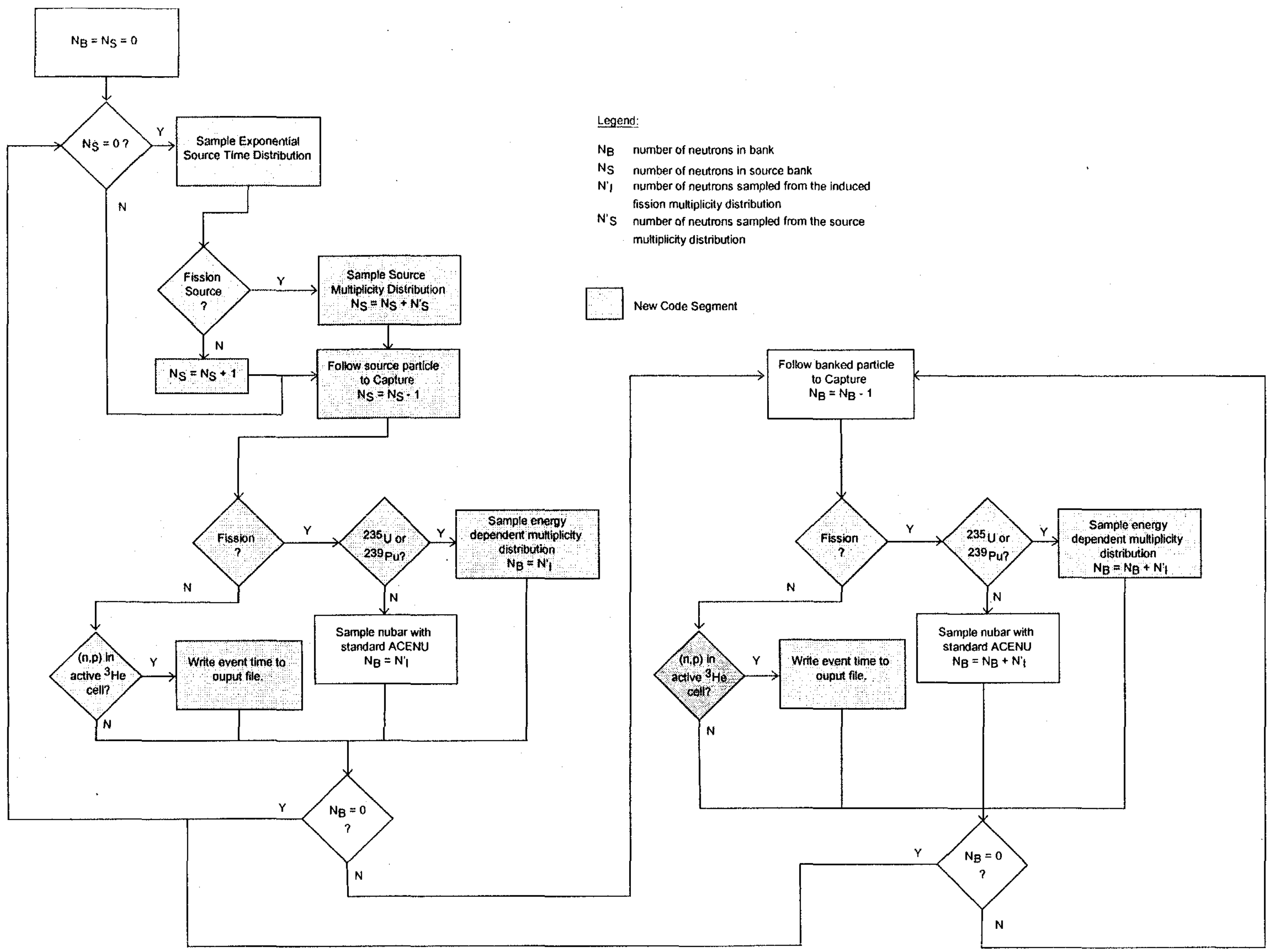

Fig. 1. Flowchart describing modifications to the Monte Carlo code MCNP4a.

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, alpha-n), the cell numbers of the active ^3He regions, and other control parameters; 2) addition of a "RDUM" card with the spontaneous fission neutron production rate and the alpha-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 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 deadtime as separate parameters.

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[5]. The UWCC consists of eight ^3He 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 [6]. Both series used 17×17 arrays of MOX fuel that was 97.30% UO_2 and 2.70% PuO_2 by weight. This corresponds to 7.00 g of plutonium and 5.00 g of uranium per rod. Uranium enrichment was 2.00% while the plutonium weight fraction was 5.55% [7]. 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 PWR MOX fuel assembly, the point model 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 I) for the first series was in good agreement with the experimentally determined detector response. The average relative error between the prediction and experimental measurement was less than 1.3%.

Number of events analyzed: 446436

DATA FOR GATE 1

Bin Number	R+A Gate	A Gate
0	34510	39574
1	81914	89682
2	102324	105831
3	90352	88831
4	63666	59182
5	37804	33942
6	19942	16778
7	9309	7607
8	4095	3126
9	1575	1253
10	610	416
11	213	127
12	76	46
13	26	26
14	11	8
15	7	3
16	1	3
17	0	0
18	0	0
19	0	0
20	0	0
21	0	0

Gate Length:	64.000	usec
Count Time:	10.870	seconds
Singles:	41071.547	counts/sec
Doubles:	6613.896	counts/sec
Triples:	1454.317	counts/sec

Corrected rates for gate 1

Singles:	41071.547	counts/sec
Doubles:	10754.140	counts/sec
Triples:	3844.994	counts/sec

Uncertainty Estimates (gate 1)

Singles:	0.128	percent
Doubles:	2.976	percent
Triples:	28.049	percent

DATA FOR GATE 2

Bin Number	R+A Gate	A Gate
0	3595	4391
1	15334	17961
2	34704	37807
3	54398	58995
4	66519	69554
5	68848	68979
6	61204	60288
7	48598	46145
8	34903	32373
9	23677	21206
10	15097	12843
11	9038	7547
12	5160	4179
13	2715	2147
14	1382	1080
15	650	524
16	313	226
17	149	97
18	66	53
19	37	23
20	26	11
21	14	5
22	4	1
23	4	0
24	0	0

Gate Length:	128.000	usec
Count Time:	10.870	seconds
Singles:	41071.547	counts/sec
Doubles:	8957.478	counts/sec
Triples:	3509.088	counts/sec

Die Away Time:	61.6872	usec
Doubles Gate Fraction:	0.6150	
Triples Gate Fraction:	0.3782	

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 17x17 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.)

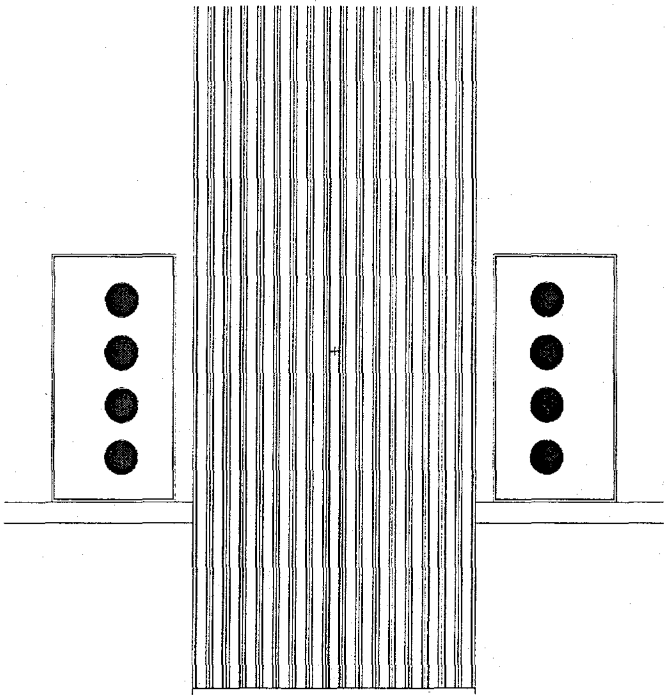

Fig. 3. MCNP model showing a slice through both heads of the detector, each with four ^3He detector tubes, with the 17 x 17 fuel bundle centered between the heads.

Fig. 4. Comparison of MCNP-REN simulation (▲) and the experimental measurement (■) of mixed oxide fresh fuel using the UWCC. Count rates greater than 4000 counts/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 less than 2250 mg/l used in the calculation owing to the solubility limit of Borax being locally exceeded. Borax precipitation was observed in the tank.

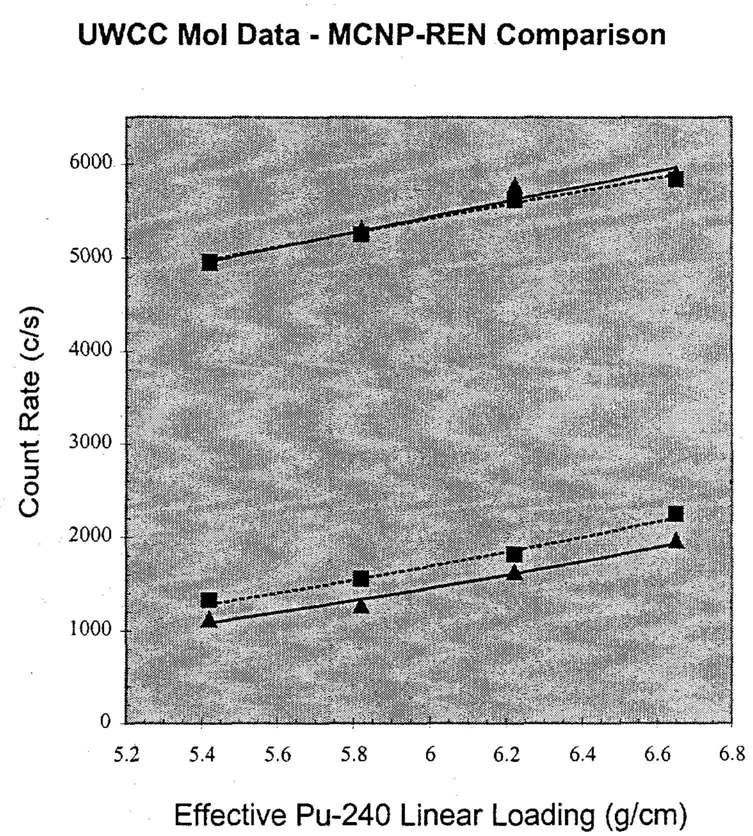

Table I. Data Comparison for UWCC Measurements and MCNP-REN Model.

No. of rods	Effective ^{240}Pu (g/cm)	MCNP Prediction (counts/s)	MCNP Uncert (counts/s)	Exp Meas (counts/s)	Exp Uncert (counts/s)
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

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 [5] have led us to conclude that the solubility limits for Borax had been exceeded, and the actual boron concentration in the tank was less than 2250 mg/l. This conclusion was partially 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 very similar to that observed in the experimental measurements.

HEU EXPERIMENTAL COMPARISON

The RRFC is an underwater active neutron coincidence counter installed at the Receipts Basin for Offsite Fuel (RBOF) 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 (PSR) counting electronics module and a computer running a modified version of the Los Alamos National Laboratory Neutron Coincidence Counting (NCC) code called RRFC. Also located above the surface is a pico-ammeter connected to an ion chamber that is located inside the instrument. The ion chamber can be used to verify declared fuel burnup.

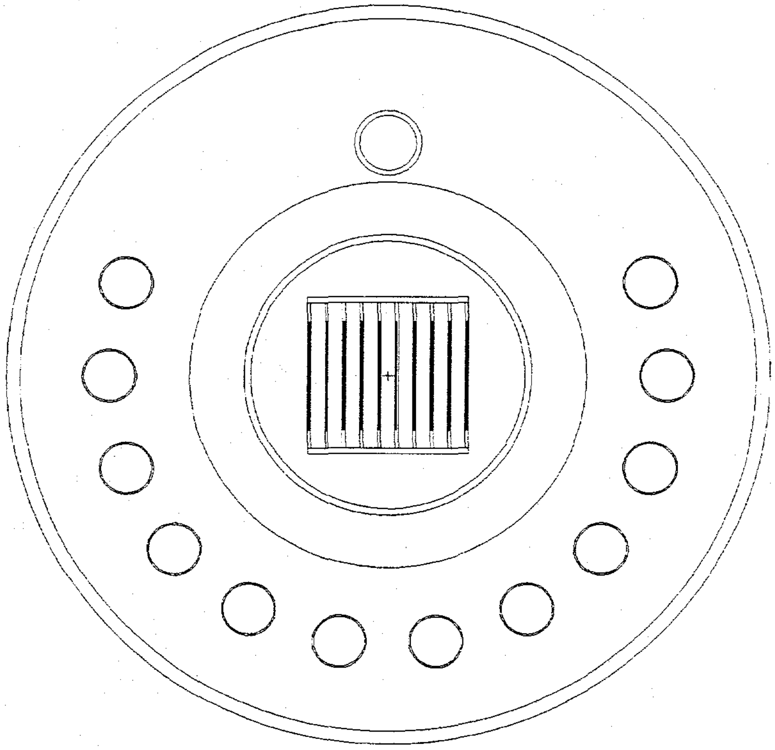

Fig. 5. Horizontal slice through the RRFC with an IAN-R1 fuel assembly centered in the detector. The slice shows the 12 detector tubes as well as the position of one of the AmLi sources located at the top of the image.

Typical assemblies are HEU, 93% enriched, aluminum clad, and contain 80 to 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 [8].

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 [9]. 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 II, 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 less than 1.3% for the mass range of 95-185 g.

Table II. MCNP-REN vs 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 (counts/s)	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 (counts/s)	164.3 ± 4.4	200.3 ± 5.2	243.9 ± 4.3	273.6 ± 4.4	284.6 ± 5.7

CONCLUSIONS

MCNP-REN was developed by modifying the general Monte Carlo code MCNP. This modification simulates the timing of neutron events in ^{3}He-filled detectors. The processing of those simulated times by a second program, TAP, then 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 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.

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