International Conference on Mathematics, Computational Methods & Reactor Physics (M&C 2009)
Saratoga Springs, New York, May 3-7, 2009, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2009)

MONTE CARLO MODELLING OF NEUTRON COINCIDENCE COUNTING SYSTEMS FOR NUCLEAR SAFEGUARDS

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ABSTRACT

This paper describes Monte Carlo methodologies applied to the simulation of neutron coincidence or multiplicity counting systems, which are generally installed at a variety of nuclear facilities around the world and rather heavily relied on by Euratom and IAEA inspectors for the safeguards and non-proliferation of nuclear material (e.g. Mox fuel, PuO_2 etc.) using Non Destructive Analysis (NDA) techniques. MCNP-PTA code has been developed at the Joint Research Centre of the European Commission in Ispra, in order to simulate the electronics pulse train analysis (PTA) systems that prevail in these counters in addition to the neutron radiation transport simulated by the MCNP part of the code on which it is primarily based. The methodology has been successfully applied for the design and optimization of new systems, for the numerical calibration and cross-calibration of various neutron counters and is to be deployed for the "verification" of nuclear materials by inspectors. This will considerably reduce both the measurement effort and the heavy reliance on reference materials required (but not always available) for the calibration of instruments. Agreement between Monte Carlo based calculations and measurements carried out either on site at nuclear installations or in laboratory conditions using reference materials are generally better than 3% as shown in this paper.

Key Words: MCNP-PTA, Monte Carlo, NDA, Nuclear Safeguards

1. INTRODUCTION

Under either their separate institutional duties or bilateral arrangements with member states, the Euratom and the IAEA nuclear inspectors routinely carry out a variety of Non-Destructive Analysis (NDA) measurements using various instruments and tools in order to verify that the material declared by a nuclear plant operator for example is correctly accounted for and satisfies its purpose. This is often done following the commonly used classical paradigm shown in Figure 1 whereby the mass declared is compared to the one deduced from measurements. Over many decades, neutron coincidence counters and their extension, namely neutron multiplicity counters, have been amongst the most important counting systems routinely applied for the mass determination of fissile material for safeguards purposes. This is due to the basic principle that the mass of the fissile material is proportional to the spontaneous fission rates, during which process a multitude of neutrons are known to be essentially emitted in "coincidence" (i.e. simultaneously) and are detected within a given short time gate. The number (two or more) of detected coincident neutrons, generally referred to as Reals, is a function of the spontaneous fission rate that in turn is proportional to the nuclear mass sought. The so-called neutron coincidence counters, with their electronics circuitry purposely set up to register only those neutrons which are simultaneous, are the most commonly used tools for neutron measurements, as they are able to differentiate between fission neutrons and those originating from competing events such as (α,n) reactions and background.

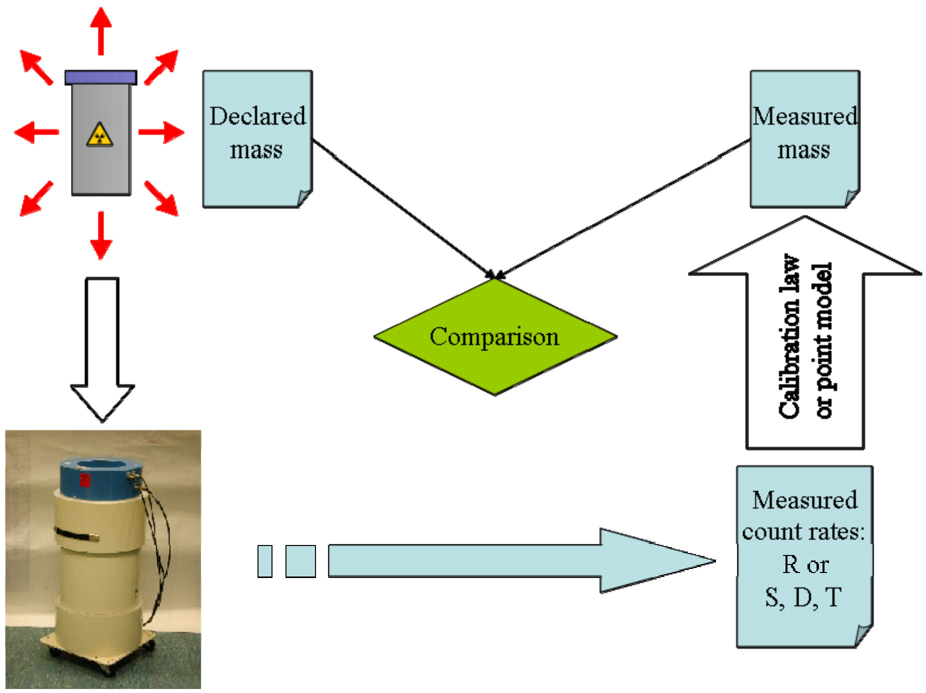

Figure 1. Classical verification paradigm.

This would subsequently ensure a correct measure of the fissile material to be safeguarded, provided that the detector in use has been properly calibrated. The neutrons themselves provide distinct benefits, compared to gamma radiation for instance, due to the fact that they are highly penetrative thus can measure the entire volume to be assayed, and they are not easily shielded, except if hydrogenous materials or those containing neutron capturing poisons (e.g. Boron) are used. Unfortunately, there are many limitations and difficulties associated with on site measurements in general, when at all possible and affordable, added to the fact that an acceptable calibration of neutron counters with a range of representative reference standard materials in adequate conditions may not always be guaranteed as discussed in references [1-3]. This is especially true and crucial when for example new nuclear fuel materials in uncommon geometries are to be verified and safeguarded.

Consequently, this has in recent years led to an increased development and usage with success of computational methods in general and Monte Carlo simulation techniques in particular for not only the design and modeling of detection systems but for their calibration and cross-calibration and for the on-site verification of nuclear material as schematically represented in Figure 2. In this paradigm, a well validated Monte Carlo model of the counting system would be run for the nuclear material and the configuration declared by the operator which would result in some signature, namely the count rates which are then compared to the measured rates. Noteworthy that any Monte Carlo model is only as good as its validation and our measurements often carried out in our PERLA laboratories with an extensive set of reference standards agree well (within few %) with our Monte Carlo calculations for a variety of systems. It follows from above that a complete modelling of the counting system requires that in addition to simulating the generation and the transport of the neutrons in the sample-plus-detector system, one must also model the electronics and the pulse train analysis system.

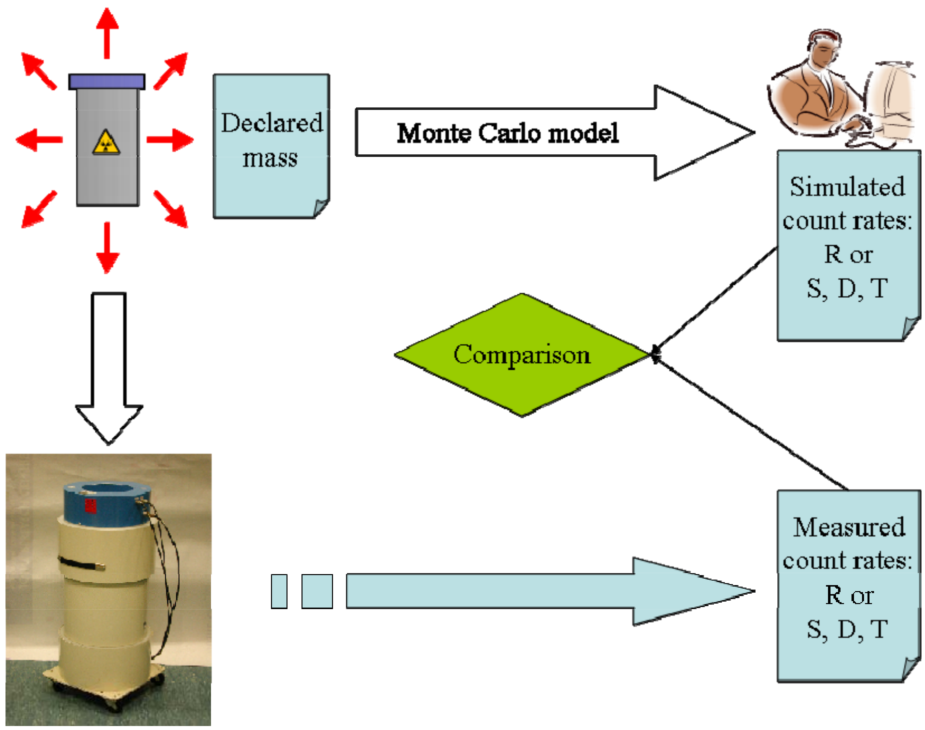

Figure 2. Monte Carlo based verification paradigm.

At the Joint Research Centre of the European Commission in Ispra (Italy), a Monte Carlo simulation code named MCNP-PTA, has been developed and successfully applied to carry out these tasks for many different neutron counting systems, including an appropriate treatment of neutron correlations and a simulation of the most commonly used pulse train analysers. A user interface has also recently been developed for automatisation in order to aid the nuclear inspectors whose task is to verify MOX (mixed Pu and U oxide) fuel magazine produced at a plant, in running the Monte Carlo simulations with ease and without knowledge of the MCNP-PTA codes, its input structure or even the counters' models.

2. BASICS OF NEUTRON COINCIDENCE COUNTING IN NDA

As in the counting systems of interest here, the neutrons detected, generally using ^{3}He(n,p) reaction based sensors, may originate either from spontaneous fission of the material to be measured (passive counting) as in section 6.3 or from reactions that are induced (active neutron counting) by an external neutron interrogation source such as an ^{241}Am-Li(α,n) as in section 6.2. In order to discriminate between the fission neutrons and others due to competing reactions such as background and (α,n) events, two categories of electronic systems that can process the time distribution of the detected neutrons are employed. The first category, used in coincidence counters is based on the traditional shift registers in which case one measures only the so called Single (or Totals) events, the Reals events (a Double events in coincidence, i.e. from the same fission) and the Accidental events (measured a certain time following the true coincidence gate). The second category uses multiplicity analysers in which case Singles, Doubles and Triples (higher order) events are counted as described below. These count rates are converted, as said above, to the measured mass through a model relating physical model calculations to experimental values, namely either a calibration law, Reals=*f*(mass), in coincidence counting or a solution of the point model equation system in multiplicity counting. Taking into account measurement uncertainties, the measured mass is finally compared to the declared value in order to either positively confirm or refute the sample verification.

Both the coincidence and multiplicity counters make use of ^{3}He gas proportional counters (usually 4 atm. pressure) embedded in low Z moderating material mainly Polyethylene (CH_2), because of the high thermal neutron capture probability in the ^{3}He gas. A neutron is thus detected as a result of a ^{3}He (n, p) T reaction generating a proton (p) and triton (T) pair and, due to the kinetic energy of these charged particles, a number of free electrons and positive ions (primary ionisations) are produced along their tracks. A high voltage is applied to the anode and the signal is amplified by a charge sensitive amplifier before a threshold is suitably adjusted on it using a pulse height discriminator (PSD). This makes it reasonable to assume that every (n,p) reaction will generate a short (50 ns) digital (TTL) output pulse on the PSD. Compared to coincidence neutron counters which use in general sixteen to fifty ^{3}He tubes, multiplicity counters, are built with more than a hundred of them in order to maximize efficiency which is especially necessary for the counting of higher order multiples. This means that tens of amplifiers and PSDs are utilized grouping a certain number of ^{3}He sensors together and in as many groups as possible (fewest tubes per bank possible) in order to reduce dead time effects. The digital pulses from a several discriminators are therefore electronically combined either by a digital mixer or a simple OR-chain in order to provide a single pulse train which can be fed into an instrument that measures its time and/or multiplicity characteristics.

As concerns modelling the operation modes of coincidence counting and counting the related events of interest, there are basically two much validated methods which are currently used. The first one is based on the so called theoretical “Point Model”, developed by Hage and Cifarelli at the JRC [4, 5, 6] in the 80’s, which yields theoretically exact equations for the probability distribution of neutron detection events. Therefore the probability of multiple (Singles, Doubles and Triples) detection of events within a time interval (gate) triggered from an initial detection event is calculated. All Single (also referred to as Totals), Doubles (or Reals) and Triples events are for instance shown to be proportional in first order to the spontaneous fission rate (F) in the sample, whereas they (S,D, T events) are respectively found to be proportional in first, second and third order, to both the neutron detection efficiency (ε) and the neutron multiplication (M). This explains for instance the need for much higher efficiency in order to be able count doubles compared to singles or triples (in multiplicity counters) compared to doubles. For the sake of completeness S, D and T events also depend on the (α,n) reaction events, to the spontaneous fission neutron ratio (usually labelled α), to the doubles and triples gate fractions (f_d, f_t), to the ν_{sj} = j-th moments of the spontaneous fission neutron distribution (ν_{sj}) and finally to the j-th moments of the induced fission neutron distribution (ν_{ij}).

Unfortunately the point model would fail if the following conditions are not met:
a) the detector's response functions must not vary with its geometrical profile,
b) the fission events which are induced by neutrons moderated in the counter and re-entering the sample is negligible.
c) the time for neutron diffusion is much larger than that of neutron cascade generation.
d) the decay in time of the neutron population within the detector obeys a single exponential law.

It is for example the case that the popular and much employed open geometry detectors, like neutron collars, do not fulfil the condition of constant efficiency profile (a) and thus an alternative solution was required.

A comprehensive guide to the usage of coincidence and multiplicity counting and an extensive description of both the theoretical and technical principles and considerations that lay behind them, including the electronics circuitry commonly used, has been given by Ensslin et al. [8].

3. SIMULATION OF NEUTRON COUNTERS WITH MCNP-PTA

A good solution to the above limitations of the point model is based on the analogue Monte Carlo method which can simulate very accurately all the functions performed by the various analysers as they are all completely based on digital electronics. For this purpose, the nuclear safeguards group of the Joint Research Centre (Ispra, Italy) where "the point model" originated many decades ago, set up to develop MCNP-PTA. It is a modified version of MCNP [9], the well known and proven radiation (neutron, gamma, electron) transport code in complex 3-D geometries, to which it couples a second Monte Carlo code PTA (Pulse Train Analysers), able to simulate the operation of a number of electronic measurement systems used in coincidence counting.

In essence, the basis MCNP coding has been designed to calculate the average values and statistical errors of many physical quantities: in each simulation history the contributions of single starting neutrons are added to the so called tallies. A number of physical phenomena are modelled in a simplified way, which on average gives for a large number of events a correct result for functions related to the first order moments such as fluxes and reactions rates etc. The number of neutrons generated for instance in a fission event is simply calculated by sampling only two values of the average number of neutrons) namely $Int(\langle\nu\rangle)$ whose probability is equal to $1-(\langle\nu\rangle - Int(\langle\nu\rangle))$ and $Int(\langle\nu\rangle)+1$ with a probability which equals $\langle\nu\rangle - Int(\langle\nu\rangle$. This simplified distribution would however fail to reproduce correctly those fission events related to the higher moments like the coincidence rates. In MCNP-PTA the sampling of the number of fission neutrons is done using the correct multiplicity distributions in order to also reproduce correctly the higher moments. Furthermore, whereas for the simulation of spontaneous fission sources, MCNP can only simulate situations that generate a single starting particle per source event, MCNP-PTA generates *n* neutrons per spontaneous fission, where *n* is sampled from a multiplicity distribution. Moreover, facilities (cards) have been developed such that the input for fuel assemblies and passive measurements can be easily implemented and the source definition (sdef) is executed automatically. This is achieved by combining the sample characteristics (provided by the user on a special input card), the information available inside MCNP at initialization and a special nuclear data library, containing data for the most common isotopes such as decay constants, spontaneous fission yields, neutron multiplicity distributions and secondary neutron spectra, (α,n) yields and neutron spectra. It is assumed that every (n,p) event inside a ^{3}He gas always generates an electronic pulse (100% tube efficiency), and that time delays due to tube dynamics can be neglected (instantaneous pulse). Following an (n,p) capture reaction, the location where the pulse was generated, the source event identifier, the time elapsed between neutron generation and detection are stored in a special interface file for post processing by the PTA code. An MCNP-PTA run is thus executed in two distinct phases: the first one regards the "normal" MCNP neutron radiation transport simulation within the nuclear material (or sample) plus detector system, followed in phase two by the Pulse Train Analysis simulating the coincidence (multiplicity) counting electronics and pulse processing involved. It is an advantage that once the first and most time consuming phase is run, the PTA phase (which make take up to minute or two on a standard PC) and various related analysis that a user may want to make (e.g. varying various electronics parameters) can be done without repeating the full Monte Carlo simulations. The PTA code hence generates the pulse train, calculates the dead-time losses due to operation of the pulse-shaping amplifier and takes into account system parameters such as the pre-delay and the gate width. The usage of modern digital mixers, also modeled by MCNP-PTA, will minimize dead time effects especially in high efficiency Pu counters which operate at high count rates as dead-time losses due to the summing of the digital pulses (logical OR) become much more important. For a detector system using shift register electronics, the PTA code creates an output file that gives 1) the singles count rates "Totals", 2) "Reals+Accidentals" and 3) the "Accidentals", the same parameters the hardware instrument gives. In the case of Multiplicity Shift Registers which calculates the multiplicity distributions of detected signals within the gate (section 2.2), the PTA code will in contrast yield the first three moments of events namely Singles, Doubles and Triples, i.e. in exactly the same manner as the electronics units used.

4. MCNP-PTA CARDS

Special cards are developed in MCNP-PTA either to carry out new tasks that are beyond the normal MCNP applications or to facilitate others, for instance, the description of magazines of MOX fuel elements. This section describes some of the cards which are added to the normal MCNP input file prior to execution. For the purpose of post-MCNP analysis e.g. of dead time effects many cards can also be run separately in (PTA) phase two of the simulation.

4.1. The Sample card:

It is defined as: **sample** *cell_label x_cen y_cen z_bottom height radius*
and is used for the MCNP-PTA cylindrical source definition, sampling the coordinates of the source events using power law sampling for the radius, and uniform sampling for the z-coordinate. The volume described by the sampling must coincide with a cylindrical MCNP

cell. In the example: sample 2003 0.00 1.00 -0.5 1.0 0.5 MCNP-PTA will sample source events in MCNP cell 2003, which is a cylinder with centre (0.00,1.00,0.00), height 1.0 and radius 0.5.

4.2. The Fission Treatment Card

The card is written as: **pta_fission** *zaid_1 mode_1* [*zaid_2 mode_2*] ... [*zaid_N mode_N*] and is used to describe the type of fission treatment to be applied within MCNP-PTA in one of the following modes and for each material referenced by the MCNP zaid parameter (default =0) involved:

- **mode**=0 , MCNP: Standard MCNP.
- **mode**=1, FORT: FORT's method for fission neutron multiplicity distribution generation.
- **mode**=2 , POLY: Polynomial Approximation of P(nu) as a function of Energy.

Thus in the following example: pta_fission 0 1 92235 0 92238 2 94239 1
the default is set to FORT's method and for zaid equal 92235 (^{235}U) the standard MCNP mode is used while for 92238 (^{238}U) POLY mode is used.

4.3. The PTA_Record Card

It is written as: **pta_record card** *id [src src_id] [eps src_eps] [tag src_tag] [epd format]* and is used to select the track to be recorded, i.e. the pulse information obtained during the run will be stored on the selected PTA pulse file. In order to record a track, a neutron source must be selected and the src sub-command can be used to select one of the following sources:

- src 0 : MCNP, SDEF
- src 1 : MCNP-PTA, AN+SF
- src 2 : MCNP-PTA, SF only
- src 3 : MCNP-PTA, AN only

When no source intensity is specified, i.e. the number of **e**vents **p**er **s**econd (eps) sub-command is not given, MCNP-PTA calculates the source intensity based on the material definitions, densities and volumes of the sample. When the eps sub-command supplied, the specified value src_eps, is used. With the tag sub-command a label can be given to each track, making identification of the recorded tracks easier. With the epd sub-command the information on the pulse file can be selected and in addition to the event_id, time_of_flight and detector_cell, the energy, the position and direction of the neutron that caused the (n-p) event are stored on the pulse file. A variety of combinations consisting of energy, position and direction combinations are selectable with the epd parameter.

4.4. The PTA_Track Card

Written as: **pta_track** id [play status] [eps track_intensity] [tag track_tag]
one can use the card to select one or more tracks that will be played back during PTA execution and the pulse files will be used for the creation of the pulse train. The values stored in the *.PTA file (created amongst others following an MCNP-PTA run) can be changed by using the eps and tag sub-commands. With the play sub-command the track can be either switched on (status=1)

or off (status=0) during the PTA execution. As and example, pta_track 2 eps 2.5E6, sets the number of events per second (eps) of track 2 to 2.5E6 while the other parameters are kept unchanged.

4.5 The PTA_Detectors Card

The card is written as: **pta_detectors** *deadtime dt pulsetime pt*
and is used to define the detector and the OR chain deadtimes as in the following example:
pta_detectors deadtime 1.0e-6 pulsetime 50.e-9

4.6. The Counterbank Card

This card is written as : **counterbank** *id [deadtime dt] [pulsetime pt] cells lb1 [lb2 … lbN]*
and is used to represent a each group of detectors which are grouped together to one amplifier. For each counter bank the labels of counter cells must be given. Counter cells are standard MCNP cells containing e.g. ^{3}He (ZAID=2003) and a specific deadtime and pulsetime may also be given for each counter bank on this card as in the following two examples:
counterbank 1 deadtime 2.0e-6 pulsetime 51.e-9 cells 1 3 6
counterbank 2 cells 2 7 9

5. COMPUTER PROCESSING AND NEW PROSPECTS

The success of Monte Carlo methods applied in many fields is owed to considerable developments in computer power and fast processing in addition to the increased availability of good cross section data which constitutes the basis of the physics involved in this method. For many on-site applications and if one is to ever reach real time or at least quasi real time simulations, high performance machines, usually one built as a cluster of processors, with better than say 100 GHz specification are being sought. A small prototype of compact Beowulf cluster, built with 4 nodes (a similar one was built for the IAEA with 16 nodes) has recently been obtained by the NDA group at the JRC (Ispra) and described in reference [10]. It is optimised for the particular case of Monte Carlo calculations in safeguard applications, which because of their nature are quite easy to execute in parallel using either the open source Parallel Virtual Machine (PVM) software library (although one can also use Message Passing Interface libraries), with which MCNP-PTA and PTA codes have been adapted to run. Running MCNP-PTA under PVM, the Monte Carlo simulation is split one hand into a single master task (on a server) and one the other hand to many micro-tasks (on the cluster nodes). As soon as a cluster node has finished the micro-tasks, a message will be sent providing the server with the calculation results. On reception of all messages, the master task calculates all intermediate results and re-organises the work load for the next rendez-vous. By optimising the number of rendez-vous, the bandwidth requirements of the interconnecting network can be reduced without losses in the cluster's computing performance using a Gigabit Ethernet standard (GbLAN) which provides high bandwidth (1000 Mbps) communication via twisted wire cabling (UTP) used in most supercomputers. The JRC is acquiring a high performance computer based on 256 core processors and 50 TBytes of data storage, which together with parallelization (PVM or MPI) will help a great deal for that purpose and in other time consuming simulations. It is also expected that new concepts for digital signal acquisition and processing would in the near future progressively replace the classical shift register based systems. The LIST mode concept whereby a high-speed PC is equipped with pulse acquisition cards, providing a time stamp (LIST mode acquisition) for every digital pulse has been developed and tested. The pulse train can be analyzed off line using software that simulates the shift register with all its parameters (e.g. dead time, pre-delay, gate-width, all of which can be modified) without repeating the acquisition. The other competing concept is that of a "virtual instrument" based on the incorporation within the neutron counting system of a digital card which: simulates the traditional shift register, integrates counts within given gates and directly provides a data acquisition PC with the singles, doubles and triples events. Both concepts have been investigated in an inter-comparison campaign within an ESARDA-NDA working group [11] at the JRC Perla Laboratory (Italy) and great progress was made.

6. MCNP-PTA APPLICATION AND RESULTS

The MCNP-PTA codes and the associated methodologies described above have been extensively and successfully applied to many different systems over recent years as published in references [1-4] to which the reader is directed for detailed descriptions. Most calculations agree within 3% or better with measurements as shown in the following example cases.

6.1. MCNP-PTA for the Verification of MOX Fuel Magazines

The code has been used to model neutron coincidence counters and their associated coincidence electronics, which for the purpose of Euratom Safeguards, have been installed at a nuclear plant where MOX (mixed Pu/U oxides) fuel elements are produced and arranged e.g. in a 14 by 14 or 17 by 17 matrix within magazines. Here as in many other situations, difficulties arise from the fact that no reference materials exist in order to experimentally calibrate the counter and even the access to the site is very restricted thus prohibiting regular safeguards measurements. Furthermore, the magazines to be verified may have a varying profile and configuration of filling with fuel pins. Thus the Monte Carlo solution following the paradigm described in Figure 2 has been adopted and used. The model of the detecting system has been validated comparing calculations with measurements carried out on site with ^{252}Cf radionuclide sources and real MOX fuel materials as more fully described in detail elsewhere [2]. The agreement as shown in Table 1 is very good.

Table 1. Computed and measured performance of the D4 detector with 3 fuel magazines. The last column gives the measured (ε_m) and calculated (ε_c) detector efficiency for ^{252}Cf.

Measurements	A	B	C	ε (^{252}Cf) cm^{-2} %
Measured Reals (s^{-1})	11328±996	11455±940	11197± 715	ε_m= 11.13 ± 0.06
Calculated Reals (s^{-1})	11396±593	11565±505	11159 ±422	ε_c= 11.14 ± 0.04
Discrepancy (%)	0.6	0.1	-0.3	< 0.3

A new user interface has been developed to automate the full process, starting from the operator's declaration of the material to be verified, to simulation and subsequent verification and comparison of the computational results against the declared masses (as in Figure 2 above) . Thus the nuclear inspectors can with the touch of a button create the MCNP-PTA input file for a specific pre-loaded and validated detector model and for a declared loading pattern and enrichments of the MOX fuel elements into a magazine, which can be changed in a simple and convenient way within the interface (Figure 3). All changes are then automatically written to the MCNP-PTA input file prior to running the code, thus allowing the inspector to verify reasonably quickly and almost in real time (depending on computing power) whether the measured and the calculated count rates, which correspond to a certain MOX mass, are consistent.

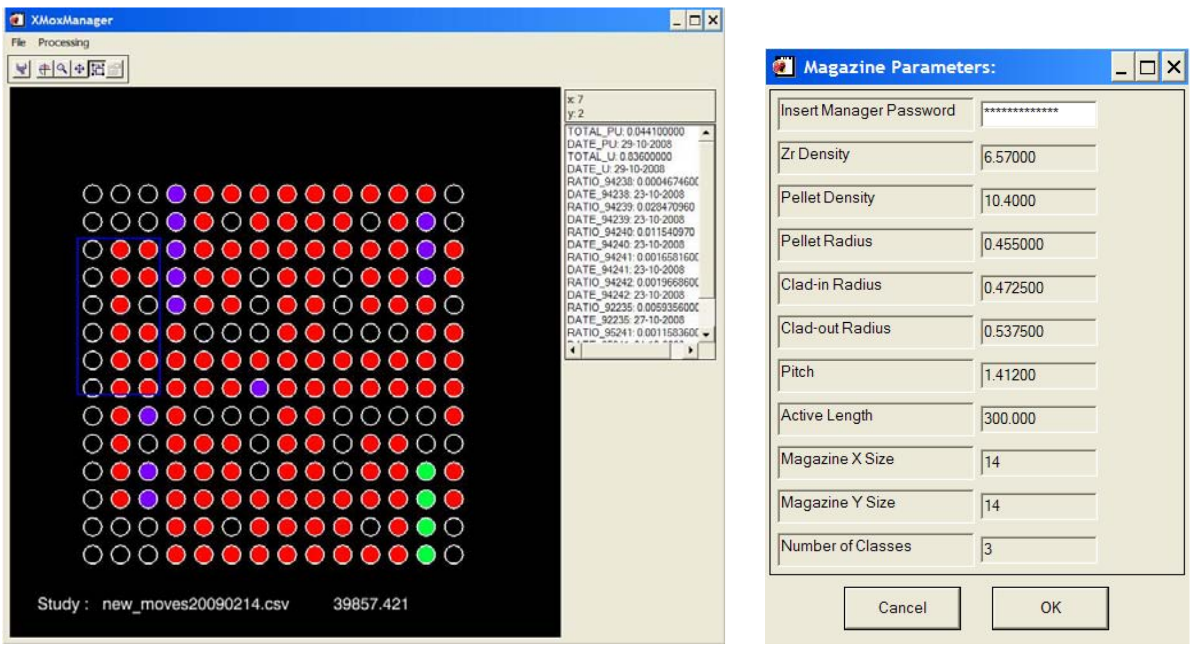

Figure 3. User interface pages for fuel type and pattern selection in a MOX magazine.

6.2. Modelling of the Guinevere AWCC Counter and its Fuel elements

This European based project, named Guinevere (Generator of Uninterrupted Intense Neutrons at the lead Venus REactor), conducted within the EC FP5 "IP-Eurotrans", addresses issues concerned with the development of Accelerator Driven Systems for the purpose of partitioning and transmutation for nuclear waste volume and radio toxicity reduction. It thus aims to obtain a validated methodology for on-line reactivity monitoring. At the SCK/CEN nuclear site in MOL (Belgium) a new GENEPI-C 14 MeV neutron generator (to be built by CNRS France) will be coupled to a modified VENUS-F zero-power fast lead reactor which will host a newly built (by CEA France) one meter in diameter fast lead core consisting of 88 fuel assemblies of about 60 cm active length. Each assembly contains nine uranium rods separated by lead rods as described in reference [12]. Here again there are no representative reference materials that can be used for calibrating any detection system to be employed in the traditional way for the determination of the mass of fissile material by the nuclear safeguards inspectors of Euratom. Thus the Monte Carlo based solution as described above will be used. The detector (an Active Well Coincidence Counter with two ^{241}Amli(α,n) sources) and the fuel magazines have therefore been extensively modeled, as shown in Figure 4, on the basis of technical details kindly supplied to us by SCK/CEN and the Reals and Totals events expected from the fuel have been calculated.

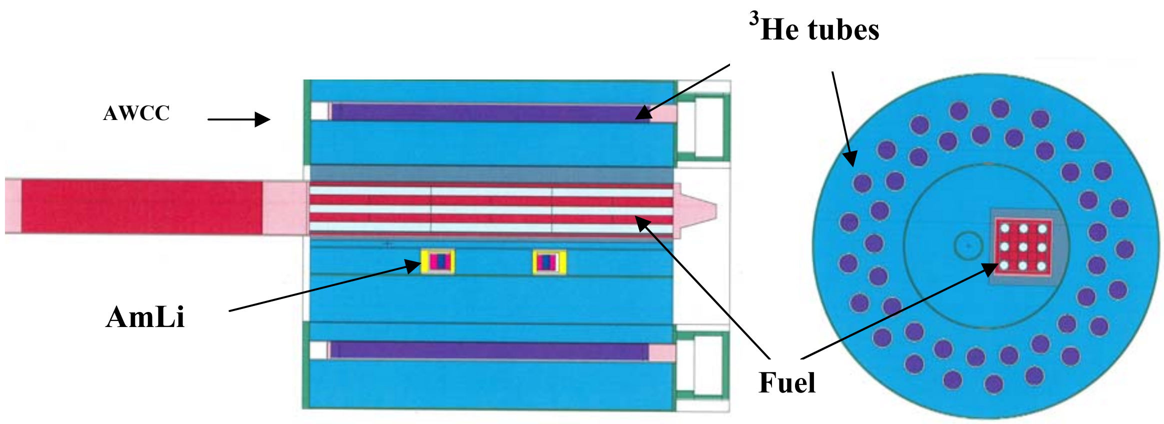

Figure 4. Monte Carlo Model of the Guinevere Active Well Coincidence Counter with two AmLi sources and showing also the uranium and lead fuel assembly.

6.3. Monte Carlo Modelling of a High Efficiency Passive Counter

In order to be able to measure the uranium mass in large low enriched uranium (LEU) samples by detecting the rather week signals from the spontaneous fission of ^{238}U, a prototype high efficiency passive counter (HEPC), which has a 4π-geometry and tens of ^{3}He tubes was designed using MCNP-PTA (Figure 5) and subsequently built and tested at the JRC (Ispra) [13]. Once the feasibility of the technique was demonstrated with the prototype counter, two large units with 148 ^{3}He tubes each were commissioned, built by a commercial company and have been successfully operating in fuel fabrication plants. Measurements of the counters' characteristics carried out agree well with the Monte Carlo calculations as seen in Table 2.

Table 2. Measured and Monte Carlo Simulated characteristics of the HEPC

Physical characteristic	Measured	Calculated
Efficiency for ^{252}Cf source	56.3%	52.7%
Efficiency for AmLi source	57.1%	57.4%
Maximum efficiency variation within cavity	3%	3%
Die away time (μs)	47.1	46.3

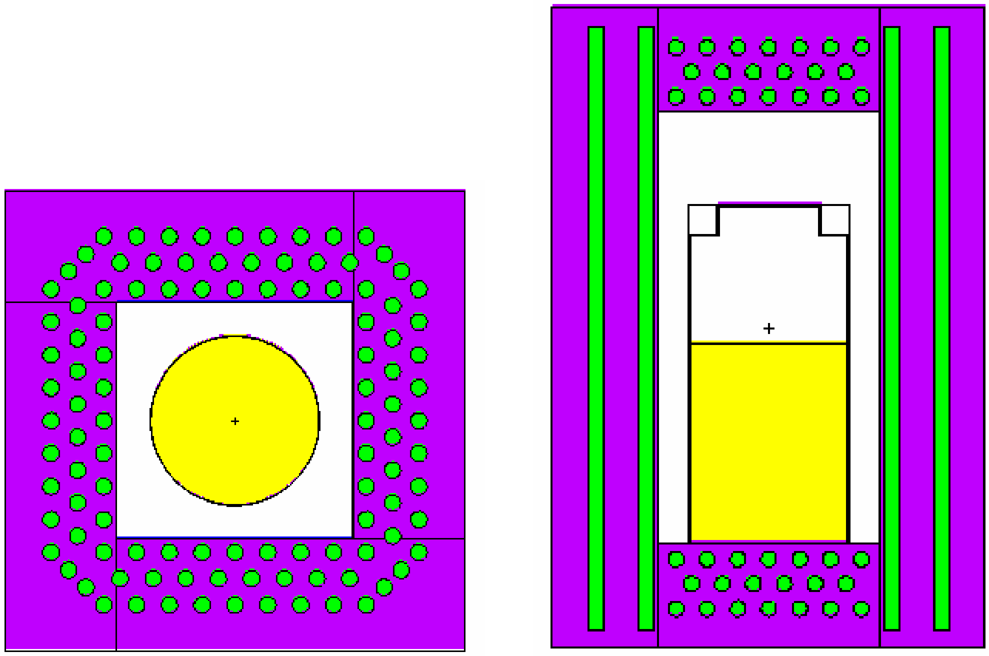

Figure 5. Transversal and longitudinal cross sections of the HEPC Monte Carlo Model.

7. CONCLUSIONS

The rationale behind the usage of Monte Carlo methodologies for the safeguards of nuclear materials, based on MCMP-PTA, has been fully established and validated. In many limiting situations whereby for instance either representatives reference materials are non-existent or regular measurements not always possible, this method may be the best if not the only solution possible. Agreement between measurements and Monte Carlo calculations is generally good (2 to 3% or better) for a variety of neutron coincidence counting systems studied [1-4]. With the quasi-real time if not real time simulation [3] now a distinct possibility due to considerable progress in computer power and parallelization techniques (PVM or MPI) the application of Monte Carlo techniques as "verification" tools by nuclear inspectors can soon be seriously envisaged. The JRC is acquiring a high performance computer based on 256 core processors and 50 TBytes of data storage, which together with parallelization (PVM or MPI) will help considerably for that purpose and in other time-processor-hungry simulations.

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