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Spread Sheet Macro Utilities for MCNP Program for Research Reactor Operation
C P Reddy, Indira Gandhi Centre for Atomic Research, Kaplakkam (T.N), India

Investigating the response function of HPGe detector with below 100 keV energy gamma rays by using MCNP code
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Dose distribution of electron beams in inhomogeneous phantoms by MCNP calculation and it's verification with experimental measurements.

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ABSTRACT

Monte Carlo simulation of radiation transport is considered to be one of the most accurate methods of radiation therapy dose calculation. MCNP is a general-purpose Monte Carlo code based on ETRAN/ITS codes. There are a few differences in electron transport between MCNP and EGS based codes. The Aims of this work was: (1) To evaluate the MCNP electron beam dose calculation capabilities in homogenous and heterogeneous phantom and determine dose distribution within or around small inhomogenities by electron beams from the new Elekta Synergy linear accelerator.

Detailed geometry of the Synergy treatment head for 8 and 15 MeV energies and 10×10 applicator as well as different heterogeneous phantoms which consisted of tissue equivalent materials were simulated. Experimental depth dose and lateral profiles at different depths under heterogeneities (e.g. 1, 3, 4cm) were measured using a P-type diode detector with a 2.5 mm diameter.

MCNP discrepancies in both energies were within 2% and within 2 mm in the homogenous phantom. The prediction of MCNP dose distributions in presence of heterogeneities were in the range of 1 to 3% discrepancy. The maximum dose enhancements in bone were approximately 7 and 5 % of the maximum dose in water for 8 and 15 MeV beam respectively. The dose values in air cavity are within 7 % less than adjacent tissues for 8 MeV beam and within 4 % for 15 MeV beam.

The results obtained with the MCNP4C agree well with measured electron radiation dose distributions. This combined with its user friendliness and ability to handle complex geometries, makes MCNP4C a very attractive and accurate code for electron transport calculations. In addition, the results give a basis for clinical judgment when electron beams are used to treat tissues behind bone and air or near bony structure.

Calculation of the gap corrections for the UK primary standard therapy-level electron beam calorimeter, using DOSRZnrc on the NPL Grid

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ABSTRACT

Hospital electron beam radiotherapy dosimetry standards are traceable using the 2003 Code of Practice[1], to the NPL primary standard therapy-level electron beam calorimeter, which has been in use for a number of years. This is a graphite calorimeter comprising a 2mm thick, 5cm diameter graphite disc surrounded by 1mm air gaps within a graphite phantom, and operates in a quasi-adiabatic mode by measuring the temperature rise for irradiations of a few seconds corresponding to absorbed doses in the region 2 – 3 Gy. The gap correction has been estimated by comparison with other calorimeters at about 0.3%, but is presently included as a contribution to the uncertainty budget.

The NPL Grid consisting of over 350 desktop computers has been used with the EGSnrc[2] user code DOSRZnrc[3], to simulate the absorbed dose distribution in the calorimeter when irradiated by electron beams from the NPL research LinAc, and thereby to calculate estimates of the gap corrections at nominal beam energies of 4, 10 and 19 MeV.

The poster illustrates the methods and results from the calculation, and describes how the United Devices Grid software[4] is used.

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Design of a Flattening Filter for a 50kVp Contact X-ray Therapy Unit using BEAMnrc

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ABSTRACT

A new contact X-ray therapy machine is being developed by Ariane Medical Systems for the treatment of early stage rectal cancer. Monte Carlo simulations have been used to design a filter to flatten the beam and to produce the required dose at depth. The 50kVp X-ray unit was modelled in BEAMnrc[1] with a realistic geometry and a parallel monoenergetic electron source incident on the transmission type tungsten target. The phase space file from this simulation was then used as the source for a series of simulations with varying filter geometries. DOSXYZnrc[2] was used to produce central axis depth dose and off axis profiles in a water tank for each filter geometry. Initial work concentrated on an early version of the X-ray unit fitted with a 0.6mm Aluminium filter. For this geometry there was good agreement between the Monte Carlo simulation and measurements using EBT radiochromic film. In addition the energy spectrum was measured using a Canberra GUL0035 Germanium detector. Although the general shape was replicated some differences were seen between this measured spectrum and the Monte Carlo generated energy spectrum. Monte Carlo simulations were run with flattening filters built up using 2 to 5 Aluminium disks 0.1 to 0.6 mm thick and 2 to 6 mm in diameter. The flattening filter which best met the flatness and depth dose requirements was selected for manufacture and testing.

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Validation of the PENELOPE Monte Carlo code to calculate the absorbed dose in heterogeneous phantoms

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ABSTRACT

This work compares some PENELOPE code (Salvat *et al*, 2001) simulation results to measurements in heterogeneous phantoms.

The 12 MV photon and 18 MeV electron beams from the LINAC Saturne 43 of the LNHB (the French standard laboratory) were simulated. Four different phantom configurations with inclusions of lung and / or bone heterogeneities have been specifically defined. A low uncertainty on the dosimeters measurements, better than 0.8% at one standard deviation, is necessary to validate the Monte Carlo simulations. Ionization Chambers (IC) were used to measure the depth-dose curves upstream and downstream the heterogeneities. Close to the interface, the Fricke solution essentially composed of water was selected instead of the IC because of its smaller dependency on the electronic spectrum. The dosimeters were not modeled in the PENELOPE geometry as only the capability of the PENELOPE code to mimic the physical phenomenon at the vicinity of the heterogeneities was tested. The simulations mimic the experimental system. First, the accelerator geometry and the materials were modeled according to the manufacturer's specifications. Then, several simulations have been carried out to characterize the initial electron source for both photon and electron beams by comparing simulated and measured depth-dose curves. The phase space files generated during the beam characterization have been used as input data to finally calculate the dose distributions in the four phantoms.

The results achieved in the different phantoms show a very good agreement between the measurements and the MC calculations. Some discrepancies exist near interfaces, between ionization chamber and PENELOPE due to the attenuation of the lower energy electrons in the wall of the ionization chamber.

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The influence of the detector dead layer thicknesses on the detector efficiency

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ABSTRACT

The simulation by Monte Carlo method for calculation of the detector efficiency is done by SIGARA software. Simulation are run for different couple of values of front and side dead layer thicknesses. The optimal values are found. The result is compared with the result taken by MCNP5 software.

Calculation of detector efficiency and coincidence summing corrections by Monte Carlo simulation.

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ABSTRACT

The steps of simulation by Monte Carlo method for calculation of the detector efficiency are described. Acquisition of a gamma ray spectrum with and not taking into consideration the coincidence sum effect by SIGARA software is described. The peak efficiency of different energy lines for these two cases and for some different geometries and matrixes is calculated. The pictures of simulated spectra and comparisons of efficiency values and fitting curves, together with the experimental efficiency values, are presented. The coincidence sum effect corrections for certain nuclides and different geometries and matrixes are presented. The dependence of these corrections from the detector dimensions and matrixes is given, too.

An investigation into the differences between electron and positron internal gas proportional counting using the PENELOPE code.

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ABSTRACT

The increasing use of Positron Emission Tomography (PET) has resulted in the commissioning of new cyclotron facilities for the production of positron-emitting radionuclides. Any radioactive effluents from these cyclotron sites are measured and monitored by dedicated instrumentation, but traceability to a primary standard is essential if such measurements are to be accurate.

A system already established at NPL for the standardisation of β^--emitters in gas is being adapted to provide a UK primary standard for positron emitters in gas. Differences between the effects of electrons and positrons (such as energy deposition, and counting losses in internal gas proportional counting) are being investigated using the PENELOPE Monte Carlo code.

Validation of the PENELOPE code for internal dosimetry

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ABSTRACT

Recent developments of internal radiotherapy lead to a growing interest for dosimetry of β or α emitters located in organs or tumours. One calculation method is to convolve the dose response from a point source (termed dose kernel) by the spatial distribution of radiation.

This study is devoted to calculations of dose distribution using PENELOPE code for monoenergetic electrons between 50 keV and 3 MeV, in an infinite homogenous water medium and three radionuclides (^{32}P, ^{90}Y and ^{131}I) of used for internal radiotherapy investigations. Calculations involve geometries of concentric spheres of radius between 2 to 750 □m. The choice of simulation parameters has been determined by analysing their influence on the dose response and comparing it with event-by-event results for the lowest energies. Ten batches of 10^6 electrons and β emitters were run isotropically at the source point, involving an uncertainty less than 1%.

PENELOPE calculations were compared to published kernels data obtained by using various codes such as EGS4-PRESTA, ACCEPT, ETRAN, EGSnrc and DOSE3D. The PENELOPE results of distributions for monoenergetic electrons show a good agreement with all the other codes. Nevertheless, some differences appear near the maximum of dose curve for energies lower than 100 keV. For higher incident energies, all the Monte Carlo code calculations are very close except those obtained with DOSE3D. Moreover, ACCEPT calculations from 1 MeV are slightly far at very short distances. Dose kernels of spectrum sources are in excellent agreement whatever the code, except at radial distances near the source.

The ability of PENELOPE to score the energy deposited in several thin spherical shells including micrometric volumes was tested. Results allow the validation of the electron transport algorithm into this specific geometry. Thus, the PENELOPE reliability for dose kernel calculations is demonstrated.

Monte Carlo simulations for a dosimetry beam line at an 18 MeV proton beam

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ABSTRACT

NPL has performed dosimetric characterisation of various detectors in a 62 MeV beam in collaboration with the Clatterbridge Centre of Oncology. However, the dose-response of many dosimeters exhibits a significant energy dependence in the sub-10 MeV region, for which a good energy resolution can not be achieved by degrading a 62 MeV beam. In particular for NPL's radiotherapy level alanine dosimetry service we have an interest in characterising its response to protons in this low energy region. Biont's 18 MeV cyclotron (IBA Cyclone 18/9) covers this low-energy range and has an external beam line, which can produce a well-focussed beam by means of a steering magnet, two quadruplets and a collimation system. In order to make this beam line suitable for reference dosimetry, additional primary and tertiary collimators are installed as well as a primary scatter foil. This results, together with the vacuum exit window, in a double scattering system. The primary collimator can be either a sieve of 0.25 mm diameter pinhole collimators or a single 1 mm diameter collimator. Apart from its beam definition function it also serves in reducing the beam current to levels useful for dosimetry.

A preliminary experiment was performed using the pinhole-sieve as primary collimator. The response of a stack of seven 0.44 mm thick alanine pellets in a small PMMA phantom was compared with the response of a Markus chamber in a separate phantom. The current produced on the isolated exit window and a transmission ionization chamber between the tertiary collimator and the dosimetry phantom were used as monitors. Radiochromic film measurements of the depth dose curve were performed. Monte Carlo simulations using SRIM, MCNPX, McPTRAN.MEDIA and McPTRAN.RZ were performed to support the design of the beam line and interpretation of the measurements.

The results of Monte Carlo simulations using SRIM, MCNPX and two versions of McPTRAN through slabs representing scatter materials, exit window and air columns are in good agreement. Simulations using MCNPX and McPTRAN.RZ of the beam using pinhole collimator shows heavy low-energy proton contamination due to oblique incidence of the primary beam. The resulting depth dose curve is in good qualitative agreement with the alanine measurements and the radiochromic film measurements.

Monte Carlo simulations for absorbed dose standards and reference dosimetry in proton beams

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ABSTRACT

External proton beams are increasingly used in clinical radiotherapy. Protons have intrinsic advantages leading to better conformal target dose distributions with better sparing of surrounding tissue compared to x-rays. To exploit these properties to their full potential, dosimetry must be at the same level of accuracy which is at present not the case.

NPL uses the Monte Carlo method for various improvements to proton dosimetry:

- in the calculation of chamber dependent perturbation correction factors for proton dosimetry using ionisation chambers calibrated in ^{60}Co, high-energy electron beams or proton beams,
- in the calculation of perturbation factors and graphite to water dose conversion factors for graphite calorimeters that can potentially serve as primary standards for absorbed dose measurement in proton beams,
- in the calculation of perturbation factors for experiments that aim to determine interaction cross sections with improved accuracy,
- in the calculation of perturbation factors for a variety of other detectors that can be used for reference dosimetry as well as relative dosimetry, such as alanine and film.

Progress in each of these areas will be reported.

Linear Power Calculation for the Chipped Fuel Pellets using MCNP program

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ABSTRACT

The fuel pins used in nuclear reactor are made by stacking the sintered fuel pellets. During the process of compacting and sintering of the pellets, some physical defects are likely to be developed. Some of these defects are cracks, body chips and end chips. The material in the chipped portion is devoid of fuel and is filled with bonding gas. This has two effects, reduction in the power produced in the pellet and reduction in the thermal conductivity from the pellet to the clad. These two effects have contradictory effects on the hot spot of the fuel. The reduction of the thermal conductivity is more dominating and leads to the increase of hotspot temperature of the fuel. This will limit the operating linear power of the defective pellets. In order to have maximum power in the reactor it is advisable that these defective pellets are not used in the reactor or used such that these pellets are in the core outer region where the power density is lower. In the first option the throughput of the pellet production gets reduced affecting the economics of fuel pellet production and the second option complicates the fuel pin assembling process and also complicates the process of automation. A good balance is struck by using the pellets with minor defects. In quantifying the acceptable size of the defect requires the calculation of the temperature distribution in the defective pellet. Many of the thermal conductivity codes are not amenable for the calculation of the hotspot temperature in these defective pellets as the geometry is very complicated. Monte Carlo technique is one of the suitable techniques to solve such problems.

Thermal conductivity process is basically diffusion process. Monte Carlo based diffusion techniques can be used to solve the problems discussed above. Neutron transport codes like MCNP can also be used to solve these problems by appropriately modified cross sections. The neutron transport theory reduces to diffusion theory with the following changes in the cross section:

- No absorption of neutrons and
- Isotropic neutron scattering

It is very easy to create such a cross section file in ENDF format and generate cross sections in ACE format such that it can be used with the MCNP code. Only one cross section set is sufficient and various materials with different thermal conductivity can be dealt by taking appropriate number density in the cell cards of MCNP code. Neutron flux is analogues to the temperature and power density is analogues to the neutron source.

The paper presents calculated results for some simple problems and details some possible limitations of the code in real problems.

Spread Sheet Macro Utilities for MCNP Program for Research Reactor Operation

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ABSTRACT

India's first sodium cooled fast reactor is fuelled mixed carbide fuel with 70 percent PuC and 30 percent UC. This reactor achieved first criticality on 18th October 1985. The reactor presently utilizes three types of fuel, referred as Mark I, Mark II and MOX, with different compositions. The carbide fuel is a mixture of PuC and UC in the ratio 70:30 in Mark I subassemblies and 55:45 in the Mark II subassemblies. The MOX fuel is mixture of PuO_2 and UO_2 in the ratio of 44:56. Mark I fuel has three different Plutonium isotopic composition. There are six symmetrically located control rods (B_4C with 90% B_{10}) in the 4th ring. The active core is surrounded by Nickel reflector (neutron) up to 8th ring. The ninth ring is loaded with Thoria blanket subassemblies. Beyond the ninth ring, stainless steel reflectors are provided. The cross sectional view of the core mid-plane, with 49 fuel subassemblies, is given in Fig. 1.

Investigating the response function of HPGE detector with below 100 KEV energy gamma rays by using MCNP code

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ABSTRACT

In this paper, tally F8 with mode P,E including secondary electron processes in detector was used in MCNP version 5 to simulate the response function of HPGe detector when using Am^{241} (59 keV) and Cd^{109} (88 keV) sources with Doppler broadening effect. The results showed that there was a large difference between scattering backgrounds of spectra when using mode P and mode P,E with low energy gamma rays of Am^{241} and Cd^{109}. When supplementing the effect of secondary electron (mode P,E) , the scattering background of spectra would be higher and then nearer to the experimental spectra. However, the P/T ratios of two models with Am^{241} and Cd^{109} sources did not have considerable difference (about 2%). It means that the mode P can be used instead of mode P,E to determine the total efficiency curve and the P/T ratio even with low energy range. The Doppler effect wasn't clear in this case and would be investigated in the gamma scattering problem.

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