LA-UR -87-4129

LA-UR--87-4129

DE88 004306

Los Alamos National Laboratory is operated by the University of California for the United States Department of Energy under contract W-7405 ENG 36

TITLE AN EFFICIENT SIMULATION MODEL FOR NUCLEAR GEOPHYSICAL MEASUREMENTS

AUTHOR(S) Robert D. Wilson, Highland and Scientific, Ltd.
Tavoni K. Cook, Highland Scientific, Ltd.
Sumner H. Dean, Los Alamos National Lab, X-7

SUBMITTED TO Proceedings for the "2nd Internat'l Symp. on Borehole Geophysics for Minerals, Geotechnical & Groundwater Applications"

October 6-8, 1987
Golden, CO

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

By acceptance of this article the publisher recognizes that the U.S. Government retains a nonexclusive, royalty-free license to publish or reproduce the published form of this contribution, or to allow others to do so, for U.S. Government purposes.

The Los Alamos National Laboratory requests that the publisher identify this article as work performed under the auspices of the U.S. Department of Energy.

MASTER

Los Alamos
Los Alamos National Laboratory
Los Alamos, New Mexico 87545

FORM NO 836 R4

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

AN EFFICIENT SIMULATION MODEL FOR NUCLEAR GEOPHYSICAL MEASUREMENTS

by Robert D. Wilson
Highland Scientific, Ltd.
and Southern Oregon State College
Tavoni K. Cook, Highland Scientific, Ltd.
Sumner H. Dean, Los Alamos National Laboratory

ABSTRACT

Nuclear measurements that provide data related to geophysical parameters have been used for many years in well-logging, surface surveys, and laboratory core studies. The measurement process can be considered in two parts: 1) the transport of nuclear radiation from source to detector, and 2) the interaction of the transported radiation with the detector. For certain measurement problems, the radiation transport of step 1) is not strongly affected by the presence of the detector in step 2). For such problems a simulation approach that decouples radiation transport from detection suggests itself. Many nuclear geophysical measurements can be modeled in this way. This method was first utilized to simulate aerial radiometric surveys and borehole logs for potassium, uranium, and thorium. A one-dimensional deterministic technique was used to model the gamma-ray transport from ground sources and the stoichastic Monte Carlo technique to model the detector response. The results of the two separate simulations are convolved to model a given source and detector combination. The method is an efficient way to model combinations of various radiation transport geometries with differing detector types and sizes.

This simulation approach has been extended in the present work to include two-dimensional geometries. The two-dimensional model applies exactly to centralized well-logging tools and in an approximate form for many eccentered measurements. The model is now being used to simulate neutron porosity and gamma-gamma density measurements.

INTRODUCTION

Many of the nuclear measurement problems encountered in borehole logging can be simulated by combining numerical solutions to the two-dimensional radiation transport equation with an appropriate detector response library. The resulting simulation model is capable of predicting the response of a variety of nuclear geophysical measurements whose geometries exhibit symmetry about one axis. In particular, the model is used in this work to determine the response of nuclear measurement systems during the process of drilling a well.

In order to efficiently model a wide variety of source, detector, borehole, drilling fluid, and formation conditions, the philosophy of the simulation model is to decouple the gamma-ray transport process from the detection process. It requires the assumption that the presence of the detector material has a negligible effect on gamma-ray transport to the detector surface. This modeling approach was originally developed for use with one-dimensional systems by M. Evans[1] at the Los Alamos National Laboratory.

The present work extends this modeling approach to two-dimensional systems. The two-dimensional discrete-ordinates neutral particle transport code DOT[2] is used to simulate the transport of neutron or gamma radiation and the analog Monte Carlo detector response code GAMRES[3] is used to transform the DOT computed radiation field to a detected pulse height spectrum. GAMRES can be used for scintillators, such as sodium iodide, cesium iodide, and bismuth germanate. The code can also be modified to simulate the response of GM tubes and solid state detectors such as germanium. An efficient model results that can couple many radiation transport calculations to a single detector response library. In this way, the more costly detector response calculations using GAMRES are performed just once.

This modeling approach was first used for measurement-while-drilling calculations in work reported by Wilson, Koizumi and Dean[4]. The model was one-dimensional and was applied to the problem of natural gamma-ray measurements in the drilling environment.

The coupling of detector response with the DOT computed radiation field is accomplished with the code ENFOLD[5]. Both GAMRES and ENFOLD had to be modified for use with the DOT computed radiation fluxes. Figure 1. is a block diagram of the simulation model.

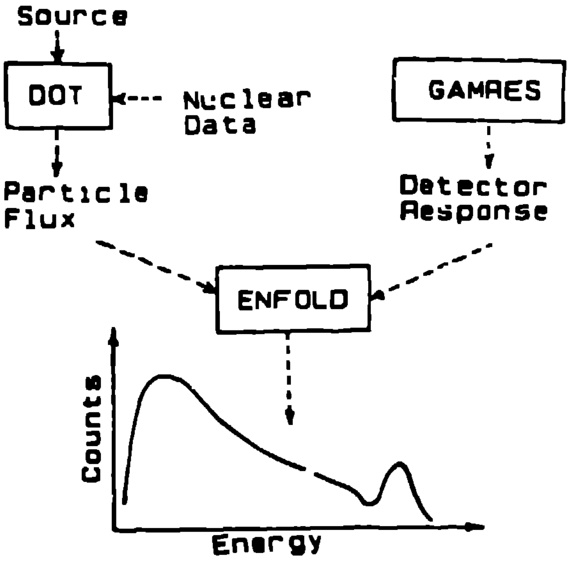

Figure 1. Block Diagram Of The Major Software Components Of The Simulation Model

The two-dimensional simulation model is very efficient, permitting extensive physics and design studies of nuclear logging tools at reasonable cost. The model has been extensively benchmarked with experiments and with Monte Carlo calculations. The model has been installed on an engineering work station, and promises in the future to be a low cost alternative and supplement to the use of experimentation with physical models.

Some uses of the simulation model and the results obtained are presented in later sections of the paper. The next sections describe the simulation model in more detail and present the results of benchmark comparisons that establish the model's validity.

SIMULATION MODEL

A two-dimensional simulation model that includes an accurate representation of scintillation detector response has been developed in order to solve a certain class of measurement problems encountered in the subsurface borehole environment. These problems are characterized by the presence of a symmetry axis, the axis of the borehole. Both the material properties and the distribution of radiation sources must exhibit this cylindrical symmetry.

The first applications of this model have been to simulate the response of neutron porosity and gamma-gamma density measurements. The model could also be applied to natural gamma-ray measurements through formation bed boundaries where the beds are separated by plane interfaces normal to the borehole axis. It applies to wireline logs when the tool is centralized within the well, and to natural gamma-ray logs in dry wells regardless of whether the tool is centralized.

The neutron porosity and gamma-gamma density geometry may lack the required cylindrical symmetry, depending on where the point source is located and whether the tool is centralized within the well. Although the detector is not actually present during the radiation transport portion of the simulation and hence does not destroy the cylindrical symmetry, a point source of radiation located off the borehole centerline, along the tool housing edge, clearly destroys the cylindrical symmetry of the measurement geometry. However, this difficulty has been avoided for such cases by representing the neutron or gamma-ray source as a ring-shaped distribution "wrapped around" the housing edge. This source representation is an approximation but has been shown to be accurate by comparison to physical measurements using actual point sources. Of course, if the point source resides along the borehole centerline, there is no difficulty in representing the point source as such in the simulation model.

The simulation approach used in this work achieves its greatest efficiency when the radiation transport portion of the calculation is performed again and again for differing sources, logging geometries, and material compositions but in each case is convolved with the same detector response map. Such is often the case when performing tool design simulations. For example, the detector may have been fixed by tool housing and performance constraints so that its simulated response need be determined only once. Then environmental parameters, such as the fluid annulus thickness, can be characterized in terms of tool performance by repeating the radiation transport calculation for each thickness and convolving each time with the same, previously computed, detector response library. This then becomes a cost effective way to determine environmental effects, minimizing the need for experimental data. The detector response map requires several hours of computer time to produce but this is done just once for a given detector. The transport calculations require between 2 or 3 hours of computer time for each case. These computation times are for the SUN model 3/160 engineering work station.

The transport calculations are performed with the two-dimensional discrete-ordinates neutral particle radiation transport code DOT. This code was written by and has evolved over the years at the Oak Ridge National Laboratory[2]. Discrete values for the energy, space, and direction variables were established through a process of compromise between requirements for convergence of the iterative procedure for solving numerically the transport equation and the desire for a reasonable total running time on the computer. For gamma-ray calculations, comparisons were made to the analytical, infinite medium, point source calculations of Goldstein and Wilkins[6] to test the adequacy of the discrete mesh.

The discrete meshes employed in the DOT calculations for the several measurement types are:

1. Neutron-Neutron. 37 energy groups, 48 radial points, 42 axial points, 16 direction angles
2. Neutron-Gamma. 58 energy groups, 48 radial points, 42 axial points, 16 direction angles
3. Gamma-Gamma. 29 energy groups, 62 radial points, 60 axial points, 30 direction angles.

A finer mesh in space and direction was necessary for the gamma-gamma calculations because of the generally larger attenuation coefficients for gamma rays than for neutrons and because of the larger anisotropy exhibited by the Compton scattering process than for elastic neutron scattering. The larger number of energy groups for neutron-neutron calculations is necessary because the neutron cross sections exhibit a finer structure than do gamma-ray cross sections. The coupled neutron-gamma problem produces the secondary gamma rays as a transfer process from the 37 neutron groups to 21 added gamma-ray groups, hence the 58 energy groups for neutron—gamma problems.

The numerical solution to the two-dimensional radiation transport equation produces a large amount of highly differential flux data. The particle flux is provided as a discrete function of energy, radial and axial position, and direction. It is called the angular flux function and physically is the rate at which the neutral particles cross unit area oriented normally to one of the discrete directions at the point in space corresponding to a particular radial and axial position and to one of the several discrete energy groups. If the angular flux is totaled over all possible directions at this location, the result is called the scalar flux. Figure 2. shows a typical geometry with a source positioned for a gamma-gamma density measurement. Figure 3. compares the energy dependent scalar flux spectra at two locations, one on the housing edge, where the detector might be located, and another in the silica formation region. The source is Cs-137. Notice the dominance of multiply scattered gamma rays and the difference in photoelectric absorption at the low energies. Variations in the low energy absorption with differing formation types gives rise to the well-known lithology response of the litho-density tool in wireline logging.

The simulation is completed when the detector response function is convolved with the angular fluxes computed by DOT at specified detector locations. The analog Monte Carlo code GAMRES is

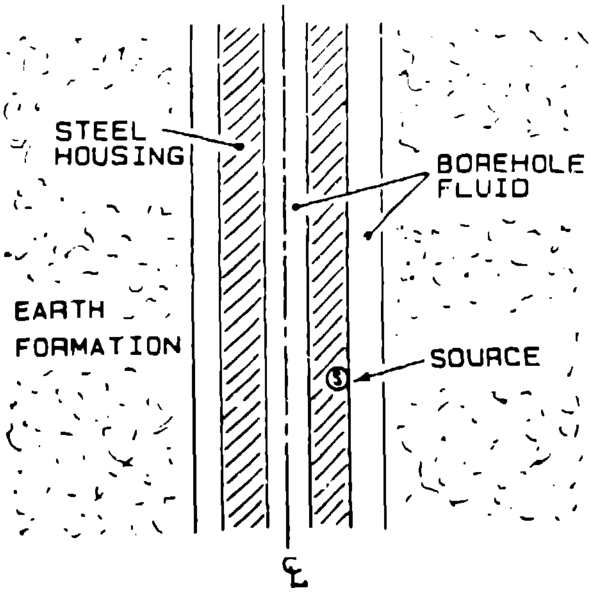

Figure 2.
Measurement Geometry.

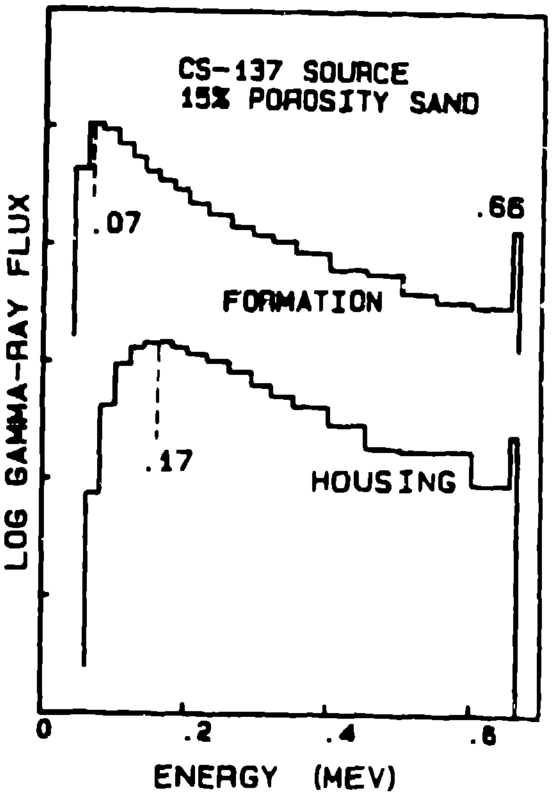

Figure 3. Gamma-Ray Flux Spectra For Locations Within The Geometry

used to compute the detector response. GAMRES was modified for use with the discrete directions chosen for the DOT calculations. GAMRES produces a simulated pulse-height spectrum for each specified incident gamma-ray energy and each discrete direction measured relative to the detector axis. For example, when used for a Cs-137 gamma-ray source, pulse-height spectra are computed at 10 keV intervals from 0 to 0.67 MeV and for six incident angles for a total of 67 x 6 or 402 spectra. This library of spectra, the detector response map, is then convolved with the DOT computed angular fluxes, at a particular location, to produce the total pulse-height response of the detector. Figure 4. is a pulse height spectrum for such a convolution calculation. It represents the count rate per .01 MeV wide channel for a scintillation detector placed along the housing edge at a particular spacing from the gamma-ray source. The source peak at .66 MeV is lost because of the relatively poor energy resolution of the detector and because the peak was a very small component of the total spectrum at this source-detector spacing.

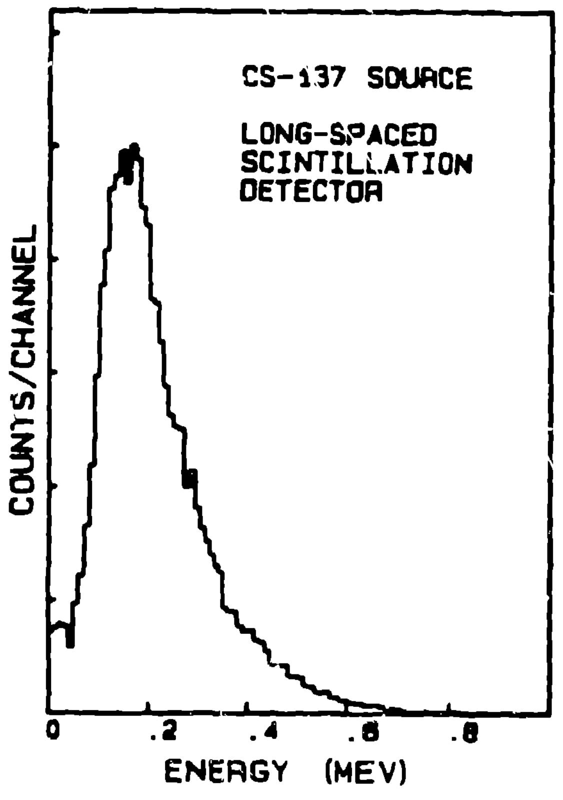

Figure 4. Pulse Height Response Of A Scintillation Detector Placed Within The Geometry

BENCHMARK COMPARISONS

The two-dimensional simulation model can be applied to both neutron and gamma-ray measurement systems as well as to coupled neutron-gamma systems where gamma rays are produced by neutron interactions. The model has been verified for such measurement problems by comparison to both experimental data and to calculations that utilize the Monte Carlo technique.

The benchmark experiments utilize a test assembly that may represent either wireline or measurement-while-drilling geometries and yet provides the close control of material composition and dimensions required for an accurate simulation. An experimental arrangement is shown in Figure 5. The fluid annulus is represented by a variable thickness of lucite pipe with an inner diameter of 6.5-inches. The tank is filled with either water-saturated Ottowa sand or with pure water.

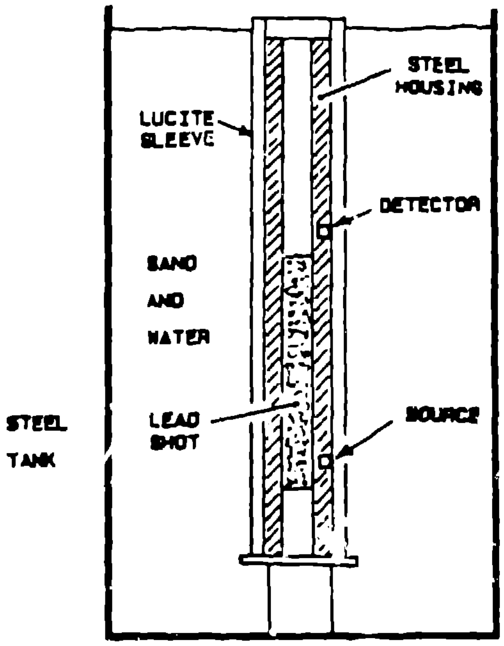

Figure 5. Test Assembly for the Benchmark Experiments.

A section of steel pipe is lowered into the lucite annulus. The pipe is instrumented with various sensors, depending on the measurement, and the radioactive source is mounted inside the pipe on the model centerline or near the edge. For neutron-neutron measurements a helium-3 detector is used and for neutron-gamma or gamma-gamma measurements a scintillator is used. Both are mounted along the pipe edge and can be moved along a channel to vary the source-detector spacing. For gamma-gamma backscatter measurements, the scintillation detector position is fixed and the source is moved among three positions to provide differing source-detector spacings. The pipe has a 3-inch inside diameter and 6.5-inch outside diameter. The inner core, which could contain fluid and various components, was filled to varying levels with lead shot or polyethylene to provide shielding.

The simulation model was used to compute detector response versus source-detector spacing for the neutron-neutron, neutron-gamma, and gamma-gamma measurements. Figure 6. is representative of the agreement obtained between the simulation model and experiment. It shows the total or gross count gamma-ray response of the gamma-gamma measurement as a function of spacing for a pure water formation, and for a water-saturated silica formation. The gamma-gamma simulations were also benchmarked by comparison to a completely independent simulation utilizing the Monte Carlo technique. The code MCNP[7] was used for the Monte Carlo simulations.

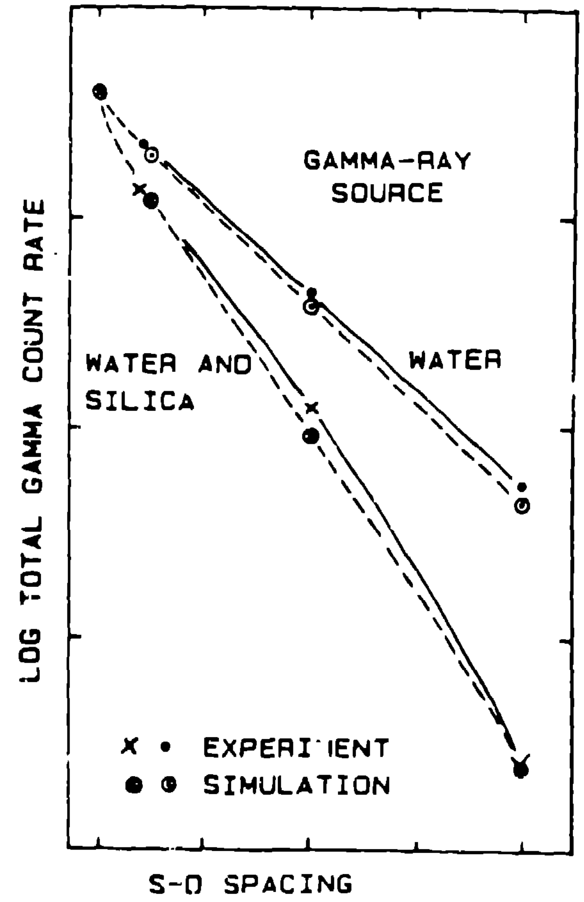

Figure 6. Comparison Of Benchmark Experiments To The Simulation Model.

The comparisons are absolute and generally quite good. They indicate the model is accurate to within 15 percent for the range of spacings, measurements, and formations studied.

APPLICATION OF THE MODEL

The simulation model has been used extensively in a continuing effort to evaluate various nuclear measurement concepts for determining formation porosity and density in a wellbore. Figure 1. showed earlier a typical arrangement in which such measurement systems might be located. The major features are the thick steel pipe, the inner and outer fluid annuli, and the formation region surrounding the well. Various source energies have been considered for both the neutron porosity and gamma-gamma density systems. The porosity measurement was evaluated for source locations inside the pipe on the pipe axis, and near the outer edge of the pipe. The simulations provide detector response for a range of formation porosities and densities for many fluid annulus conditions and as a function of source-detector spacing. Scintillation detectors used for gamma-ray detection are modeled with the GAMRES code. Neutron detectors, such as the helium-3 proportional counter, are modeled assuming each thermal neutron or epithermal neutron within the detection volume produces a count according to the energy dependent neutron-proton reaction cross section. All radiation transport calculations were performed with the DOT code, whether neutrons or gamma rays.

A neutron-gamma measurement has been chosen to illustrate the application of the simulation model. This measurement is potentially a technique for determining formation porosity through detection of hydrogen capture gamma rays. A scintillation detector is placed at a particular spacing from a neutron source. The scintillator responds to thermal neutron capture gamma radiation, principally from the elements hydrogen, silicon, and iron. If the counts are totaled across the entire spectrum, the so-called total gamma or gross-count gamma response obtains. Figure 7. is a plot of the total count response of the detector as a function of source-detector spacing. The figure shows curves for two different formation porosities and for two different fluid annulus thicknesses. Such curves can be developed for other fluid annulus thicknesses and compositions to assess the effect of the drilling fluid on the measurement.

Calculations like these have been used to select detector spacings, to establish source strength, and to develop data reduction and calibration procedures. Measurement systems that do not display an axis of symmetry must be simulated using a three-dimensional model. The Monte Carlo code MCNP has been installed on an engineering work station for this purpose. It has not yet been used for nuclear tool design but has successfully provided benchmark data against which the DOT simulation model has been compared. The Monte Carlo model is the model of "last resort" in the sense that it is a very time consuming calculation and is subject to statistical fluctuations because of the stoichastic nature of this simulation approach.

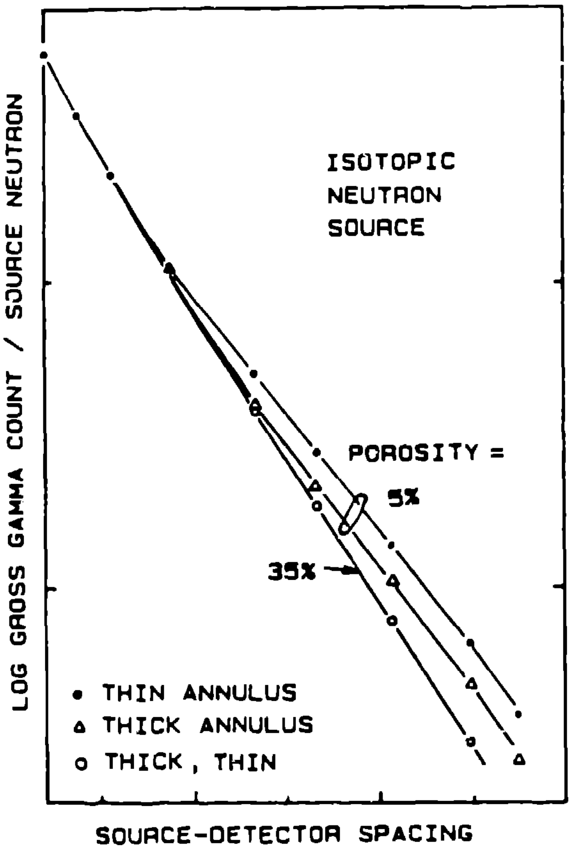

Figure 7. An Application of the Simulation Model Showing the Neutron-Gamma Porosity Response

CONCLUSIONS

The two-dimensional logging tool simulation model developed in this work has proved to be accurate, efficient and quite useful in tool design problems. The implementation of the simulation model on the newest class of very powerful engineering work stations has tremendously increased the usefulness of the simulation model because of the modest cost of the computations.

Present efforts to install Monte Carlo simulation models on the engineering workstation have been very successful and the future of three-dimensional and time-dependent modeling with Monte Carlo techniques appears bright.

ACKNOWLEDGEMENTS

This work has been supported by Exploration Logging, Inc. Critical guidance has been provided by Exploration Logging personnel, particularly by Jim Meisner, Roger Tresler, Hilton Evans, and Andrew Brooks. Roger Tresler and Ted Mumby of Exploration Logging, Inc., designed and fabricated certain critical hardware components for the benchmark experiments. The authors thank the School of Science-Mathematics of Southern Oregon State College for making facilities and equipment available for use in the benchmark measurements.

REFERENCES

1. M. L. Evans, ed., "NDA Technology for Uranium Resource Evaluation," Los Alamos National Laboratory report LA-8490-PR (August 1980).

2. W. A. Rhoades and F. R. Mynatt, "The DOT III Two-Dimensional Discrete Ordinates Transport Code," ORNL-TM-4280 (September 1973).

3. M. L. Evans and R. D. Wilson, "Calculation of Borehole Correction Factors for Gamma-Ray Spectral Logging, Bendix Field Engineering Corp. report GJBX-27(83) (August 1983).

4. R. D. Wilson, C. J. Koizumi, S. D. Dean, "Spectral Gamma-Ray Calculations with an Efficient and Accurate Radiation Transport Model," Paper B., Trans. SPWLA, Twenty-Sixth Annual Logging Symposium, Dallas, Texas (June 1985).

5. M. L. Evans, "NDA Technology for Uranium Resource Evaluation," Los Alamos National Laboratory Report LA-8967PR (August 1981).

6. H. Goldstein and J. E. Wilkins, Jr., "Calculations of the Penetration of Gamma Rays," NYO-3075, Nuclear Development Association, Inc. (1954).

7. J. F. Briesmeister, editor, "MCNP - A General Monte Carlo Code for Neutron and Photon Transport," Los Alamos National Laboratory Report LA-7396-M, Rev.2 (September 1986).