BIOMEDICAL APPLICATIONS OF TWO- AND THREE-DIMENSIONAL DETERMINISTIC RADIATION TRANSPORT METHODS

David W. Nigg
INEL BNCT Program
Idaho National Laboratory
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho, 83415-1575

ABSTRACT:

Multidimensional deterministic radiation transport methods are routinely used in support of the Boron Neutron Capture Therapy (BNCT) Program at the Idaho National Engineering Laboratory (INEL). Typical applications of two-dimensional discrete-ordinates methods include neutron filter design, as well as phantom dosimetry. The epithermal-neutron filter for BNCT that is currently available at the Brookhaven Medical Research Reactor (BMRR) was designed using such methods. Good agreement between calculated and measured neutron fluxes was observed for this filter. Three-dimensional discrete-ordinates calculations are used routinely for dose-distribution calculations in three-dimensional phantoms placed in the BMRR beam, as well as for treatment planning verification for live canine subjects. Again, good agreement between calculated and measured neutron fluxes and dose levels is obtained.

INTRODUCTION:

Multidimensional deterministic methods for radiation transport analysis have a wide variety of applications in the biomedical field, particularly in radiation oncology. Standard photon therapy, as well as some of the less-common modalities such as fast-neutron therapy and neutron capture therapy, are all dependent for their success upon the ability to calculate, *a priori*, accurate radiation dose distributions in the target volume. This seminar will focus on the use of various multidimensional, deterministic transport methods in support of BNCT research at the INEL and elsewhere. Particular emphasis will be placed on comparisons of calculation and measurement in a variety of applications.

There has been a worldwide resurgence of interest in BNCT for the treatment of certain types of currently-intractable malignancies (e.g. Glioblastoma Multiforme). The INEL is currently the focus for a comprehensive effort directed toward the development of this complex modality. The INEL BNCT Program includes: (1) the design, construction, and testing of neutron sources that are appropriate for therapy, and (2) coordination of various radiobiological and biochemical research activities, as well as the development of several ancillary technologies needed for eventual human clinical trials of BNCT. Although the program is centered at, and administered by, the INEL, the participants include experts in the appropriate research disciplines at a large number of institutions nationwide.

BNCT involves a two-step treatment procedure. In the first step, an appropriate drug tagged with Boron-10 (^{10}B) is administered by injection. This drug preferentially accumulates in the malignant tissue. A thermal-neutron field (from a beam generated by a nuclear reactor or other neutron source) is then applied, leading to selective destruction of the cancerous tissue by the high linear energy transfer (LET) secondary charged-particles that result from the boron capture reaction, specifically ^{4}He and ^{7}Li. The energy of this charged-particle pair is 2.35 MeV, on the average. This energy is deposited over a range comparable to cellular dimensions (a few micrometers), resulting in a high probability for cancer cell destruction with limited damage to surrounding healthy tissue. An epithermal-neutron beam (neutrons predominately in the energy range of 0.5 eV to 10 keV) is expected to have an advantage compared to a thermal-neutron beam because the epithermal neutrons will penetrate a few centimeters into tissue before forming a thermal peak, yet the epithermal neutrons are themselves relatively nondamaging. Therefore, with an epithermal beam, it may be possible to treat some deep-seated tumors without surgery and spare healthy surface tissue.

Multidimensional deterministic radiation transport analysis methods are used at the INEL in connection with the INEL BNCT Program to perform physics design calculations for various neutron sources, as well as to perform detailed radiation-flux and dose-distribution calculations for actual experimental irradiations. Both applications will be summarized in this seminar. References to additional, more detailed information available in previous open publications will be provided for those who wish to pursue the subject further.

NEUTRON SOURCE DESIGN:

Design of epithermal-neutron sources at the INEL has concentrated primarily on reactor-based systems.[1-3] Reactor-based, epithermal-neutron beam facilities designed at the INEL have included the previously-considered Power Burst Facility (PBF) beam (located at the INEL), which is capable of providing a high-intensity, high-purity epithermal source, and the currently-operating, medium-intensity epithermal-neutron beam at BMRR. This latter beam was designed and installed in collaboration with Brookhaven National Laboratory. In addition, there is a currently-ongoing collaborative effort between the INEL and the Georgia Institute of Technology to design a high-purity, medium-intensity epithermal-neutron beam for the Georgia Tech Research Reactor (GTRR).

Analytical methods employed for beam design at the INEL typically include one- and two-dimensional deterministic radiation transport calculations with limited confirmatory calculations performed using the Monte Carlo technique. The ANISN[4] and DORT[5] discrete-ordinates codes, developed at Oak Ridge National Laboratory (ORNL) are employed for the deterministic calculations. Cross sections are typically taken from the BUGLE-80[6] library distributed by ORNL. These methods have proven to be very effective and accurate.

The epithermal-neutron beam at BMRR provides a typical example of the application of multidimensional transport techniques. Figure 1 shows the BMRR reactor, the reflector, the thermal shield, and the three filter compartments where various materials may be placed, providing the means to tailor the neutron and gamma flux that is produced at the irradiation point. Large aluminum containers, filled with aluminum oxide tiles and aluminum spacers, were tailored to these pre-existing compartments. A layer of cadmium was installed at the downstream end of the filter to minimize the thermal-neutron component. Additional bismuth was placed at the beamport to minimize the gamma component of the beam.

The geometry of the BMRR beam represents a typical deep-penetration type transport problem. Final design studies were performed with the DORT[5] code and the BUGLE-80[6] coupled 47-neutron, 20-gamma group, P3 cross-section library. Cylindrical r-z geometry was used, with the Z-axis coinciding with the beam centerline. The reactor core was modeled as a homogeneous cylinder. This is a rough approximation to the actual geometry, but it was found to be adequate as long as certain key volumes and material thicknesses are preserved.

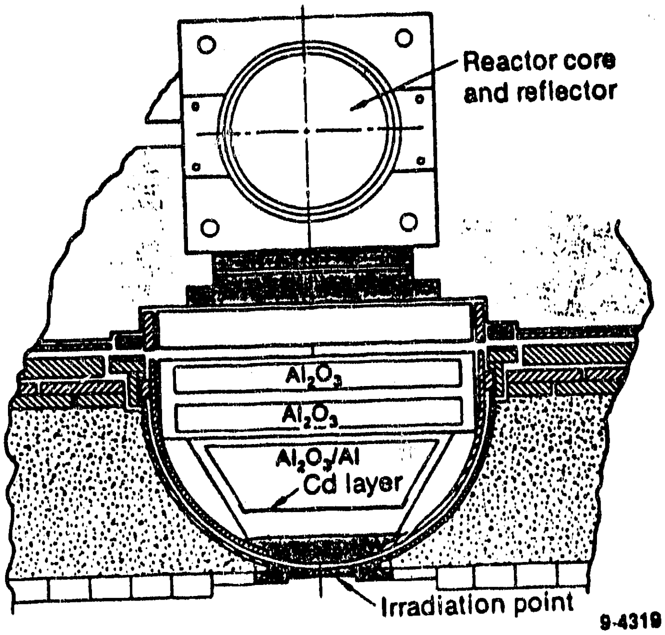

Figure 1. BMRR with epithermal-neutron filter for BNCT.

It is theoretically possible to solve the transport problem for the combined core and filter in a single enormous computer run. It was, however, much more efficient to obtain the solution for the neutron and gamma flux at the exit port using a two-step procedure. In the first step, the reactor core and a small portion of the filter were calculated as an eigenvalue problem. In the second step, the filter was modeled separately, with a fixed angular flux boundary condition at the entrance to the filter taken from the results of the first step. (The DORT code allows this to be done using a binary angular flux file). This results in the same solution as if the entire problem were solved in one run, as long as the reactor and filter models have sufficient overlap (to account for scattering back and forth between the two segments of the geometry).

The BMRR core and filter models have been successfully run using the standard CRAY version of DORT. Each problem segment requires about four hours of CPU time on the INEL CRAY-XMP/24. In addition, the models have been run using a version of DORT locally-adapted for use on the IBM RISC-6000 series of engineering workstations. Overnight execution times are easily achieved for the BMRR models on this type of hardware. The calculated solutions from the workstation version of

DORT are typically in agreement to about three decimal places with the corresponding CRAY results, even though the workstation uses a 32-bit word length, as opposed to 64 bits for the CRAY.

Table I shows some calculated[1] and measured[7] free-field integral neutron-beam characteristics. Figure 2 shows the calculated and measured free-field neutron spectra.[1] The agreement between the design calculations and the measurements is excellent for the principle beam characteristics. The calculated epithermal and total neutron fluxes are within 2% of the measured values. The fast-neutron (above 10 keV) flux is underestimated in the calculation by about 30%. This discrepancy is believed to be caused by certain modeling approximations, and by the fact that the energy group structure of the BUGLE library does not allow an accurate representation of the neutron cross-section minimum in aluminum in the energy range near 75 keV. (The 25-keV window in aluminum is well-represented in the BUGLE library).

RADIATION DOSE-DISTRIBUTION ANALYSES:

Given a neutron beam of acceptable purity and intensity, it then becomes necessary to determine the expected radiation flux and dose levels in the irradiation target. This is a complex, three-dimensional problem. There are several dose components (boron neutron capture dose, hydrogen capture dose, incident fast-neutron dose, etc.), all having different spatial distributions and relative biological effectiveness (RBE). Either stochastic (Monte Carlo) or deterministic calculational methods may be used. At the INEL, three-dimensional, deterministic calculations are routinely performed for BNCT

Table I. Calculated and Measured Integral Neutron Fluxes for the Unperturbed BMRR Epithermal-Neutron Beam.

Flux ($n/cm^2/s/MW$)	Design Calculation	Measurement
Total neutron flux	6.75E+8	6.65E+8
Epithermal-neutron flux	6.39E+8	6.20E+8
Fast-neutron flux	0.25E+8	0.34E+8
NOTE: All data are normalized to 1 MW indicated BMRR core power.		

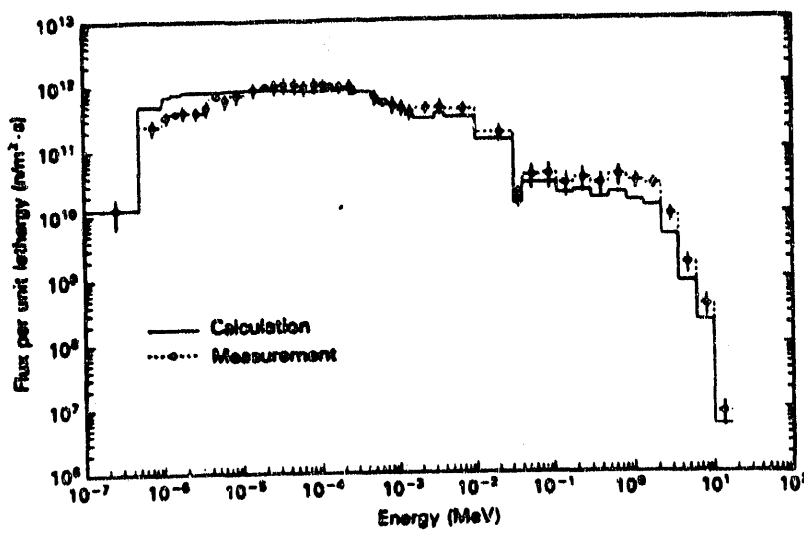

Figure 2. Comparison of calculated and measured neutron spectrum for the BMRR aluminum-oxide filter.

dosimetry using a local adaptation of the TORT[8] discrete-ordinates code. Typical applications include calculations of radiation flux and dose distributions in three-dimensional tissue-equivalent phantoms, as well as calculations for planning and interpretation of actual canine irradiation experiments, currently being conducted in connection with the INEL BNCT Program. An example involving neutron flux calculations for a three-dimensional canine head phantom will be briefly described below. Further details of this and various other three-dimensional applications are available in References 9-11.

A canine brain-irradiation program[12] for research into the biological effects of BNCT forms a major part of the overall set of BNCT-related activities being coordinated by the INEL. This experimental campaign uses the epithermal-neutron beam at the BMRR. Prior to the use of this beam with live subjects, it was necessary to experimentally characterize the neutron flux levels produced in realistic target volumes. To that end, a Lucite dog-head phantom, with dimensions typical of a Labrador Retriever, was constructed (Figure 3). The phantom was positioned in the BMRR beam as shown in Figure 4. (The extent of the region modeled by TORT is shown in bold outline.) The location and positioning of polyethylene bricks used to simulate the body of a typical dog are also shown in this figure.

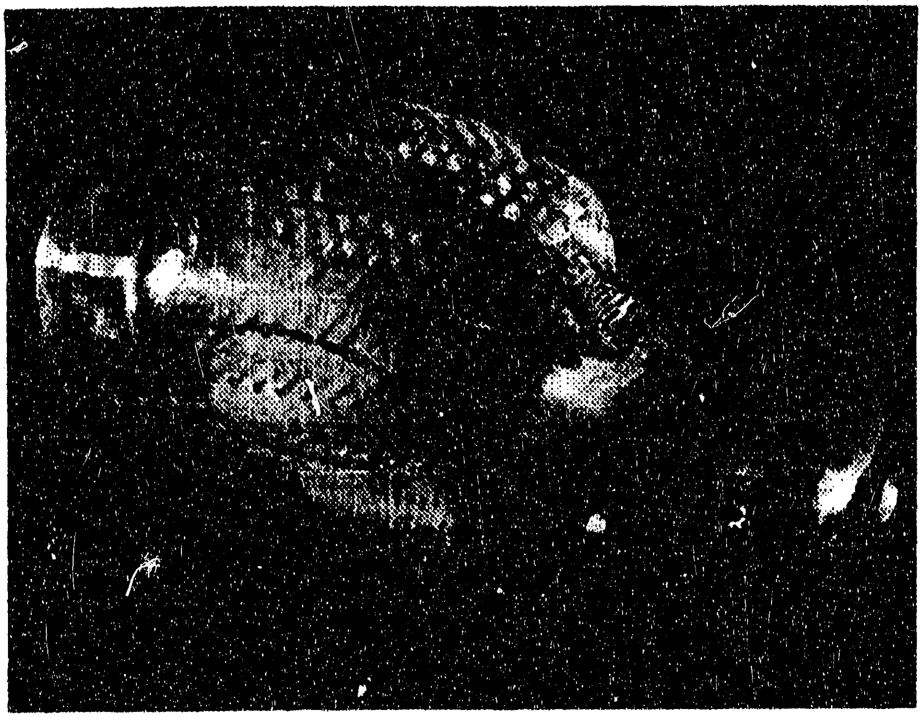

Figure 3. Lucite dog-head phantom.

The dog-head phantom has a large number of regularly-spaced holes drilled into it (as can be seen in Figure 3). Copper-gold alloy wires were inserted into these holes to measure neutron activation (and, therefore, neutron fluence) versus depth below the top of the head in several locations. The use of these two materials allows the thermal-neutron flux to be measured separately from the total flux. It is the thermal-neutron flux that is of particular interest in this work.

Figures 5 and 6 show how the dog-head phantom was modeled using the X-Y-Z representation available in the TORT code. Only the head and upper neck regions of the phantom were included in this particular model. The curved surface of the phantom was approximated as well as possible with a 1-cm mesh as shown. There were 32 mesh intervals in the X-direction, 16 intervals in the Y-direction. and 22 intervals in the Z-direction. A symmetric 96-direction quadrature set was used. All cross-section data were again taken form the BUGLE-80 library. A lithiated-polyethylene beam

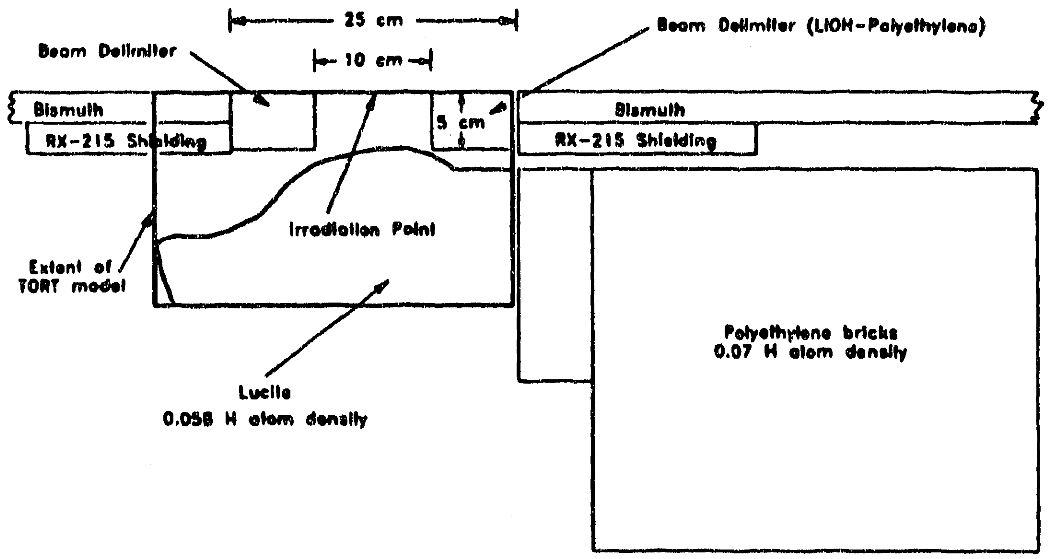

Figure 4. Positioning of canine phantom and beam delimiter at the BMRR beam output port.

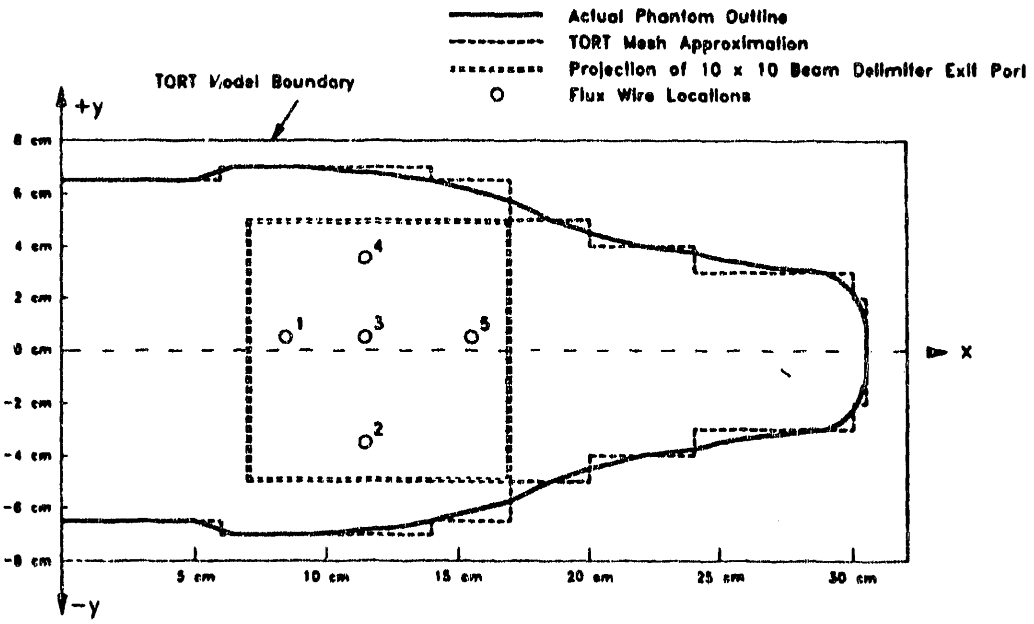

Figure 5. Dog-head phantom model (top view).

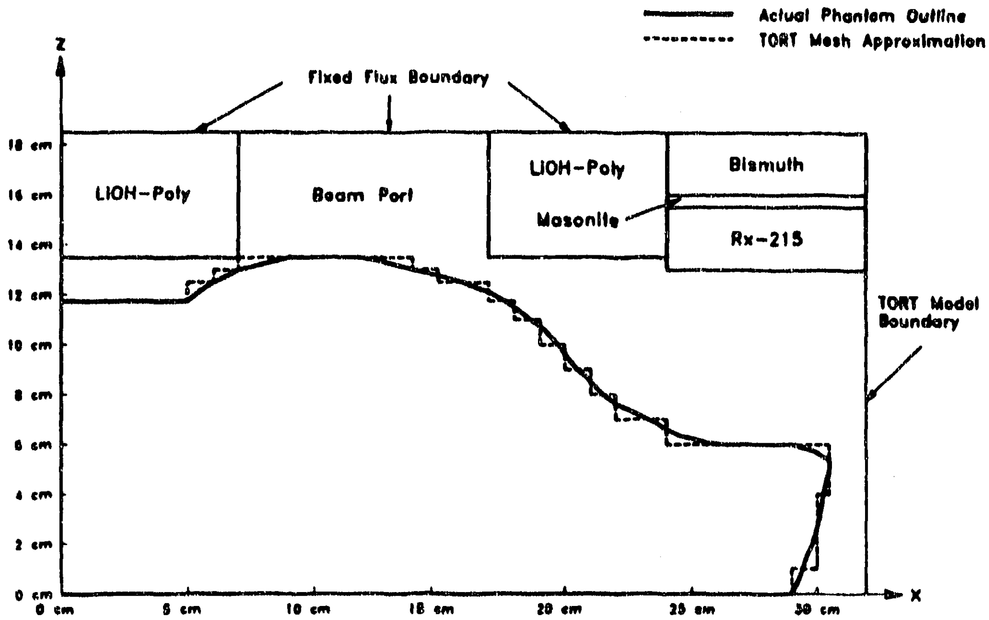

Figure 6. Dog-head phantom model (side view).

delimiter, with a 10-cm square aperture, was placed between the bismuth beam face and the top of the phantom as shown. This delimiter was explicitly included in the TORT model. Finally, Figure 6 shows details of the vertical flux-wire positions of interest in this work.

Figure 6 also shows the neutron flux at the entrance plane of the delimiter port was represented in the TORT model by a fixed angular flux boundary condition in the X-Y plane at the top of the calculated geometry. The magnitude, spectrum, and angular distribution for this source were calculated with the previously-described DORT model of the BMRR filter. The small incident gamma source present in the BMRR beam. was also included in the source representation for the TORT phantom model.

Calculations for the lucite phantom were carried out using a version of the TORT code adapted at the INEL for operation on an Apollo DN10000 engineering workstation. The entire 67-group calculation ran in approximately 670 minutes. Performance of the IBM RISC-6000 series of workstations for this application is similar to within a factor of two in execution time, depending on the exact hardware configuration employed.

The calculated axial thermal (0.0 to 0.414 eV) neutron-flux distribution for the flux wire nearest the beam axis (Wire 3) is shown in Figure 7. The measured distribution is shown for comparison and agreement is in the 10-20% range. This is quite acceptable for this type of calculation, given the measurement uncertainty (about 10%) and the modeling approximations that were used. Of particular note is the fact that all calculations for both the reactor-filter combination (DORT) and the phantom (TORT) are normalized directly to the indicated BMRR core power with no empirical correction factors of any type.

CONCLUSIONS:

Multidimensional, deterministic transport calculations have become essentially indispensable in the INEL BNCT Program. These types of calculations complement the extensive Monte Carlo calculations that are done for patient treatment planning. This provides a very useful validation tool. In some beam design applications, deterministic methods offer the only practical approach to the problem of calculating radiation transport through large, complex filter and moderator assemblies. Monte Carlo

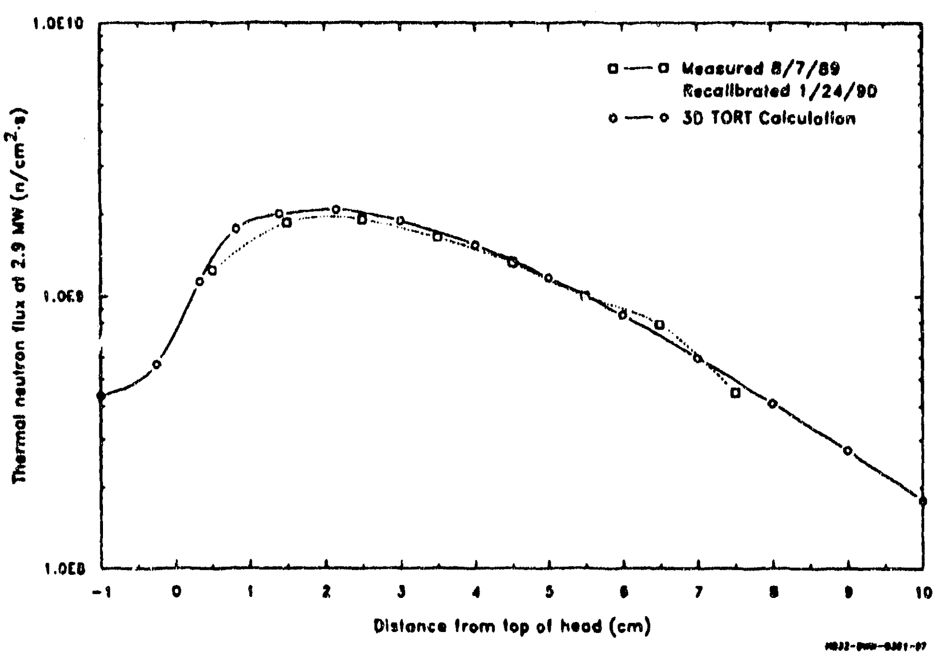

Figure 7. Depth profile of BMRR (2.9 MW) thermal flux. Dog-head phantom at x = 11.5 cm, y = 0.5 cm (Flux Wire 3).

techniques, even those that are biased in some manner, tend to have large statistical variances in such applications. The ongoing trend in the direction of increasing speed and decreasing cost of modern engineering workstations will allow even more sophisticated uses of deterministic techniques in the future.

REFERENCES:

1. Wheeler, F. J., et.al.,"Epithermal Neutron Beam Design for Neutron Capture Therapy at the PBF and BMRR Reactor Facilities," Nuclear Technology 92: 106-117 (1990).

2. Neuman, W. A. and J. L. Jones, "Conceptual Design of a Medical Reactor for Neutron Capture Therapy," Nuclear Technology 92: 77-92 (1990).

3. Nigg, D. W. and F. J. Wheeler, "Conceptual Physics Design of an Epithermal-Neutron Facility for Neutron Capture Therapy at the Georgia Tech Research Reactor," Transactions of the 1992 American Nuclear Society Annual Meeting, Boston, MA, June 1992.

4. Parsons, D. K., "ANISN/PC Manual," EGG-2500 (December 1988).

5. Rhoades, W. A., R. L. Childs, "Updated Version of the DOT4 One- and Two-Dimensional Neutron/Photon Transport Code," ORNL-5651, Oak Ridge National Laboratory (1982).

6. Roussin, R. W., "BUGLE-80, Coupled 47-Neutron, 20-Gamma Ray, P3 Cross Section Library for LWR Shielding Calculations," DLC-75, Radiation Shielding Information Center, Oak Ridge National Laboratory (1980).

7. Becker, G. K., et. al.,"Neutron Spectrum Measurements in the Aluminum-Oxide Filtered Beam Facility at the BMRR," *Neutron Beam Design, Development, and Performance for Neutron Capture Therapy*, O.K. Harling, et. al. (eds), Plenum Press, New York, NY, 235-248 (1990).

8. Rhoades, W. A. and R. L. Childs, "The TORT Three-Dimensional Discrete-Ordinates Neutron/Photon Transport Code,", ORNL-6268, Oak Ridge National Laboratory (1987).

9. Nigg, D. W., et. al., "Demonstration of Three-Dimensional Deterministic Radiation Transport Theory Dose Distribution Analysis for Boron Neutron Capture Therapy," Medical Physics 18: (1) 43-53.

10. Wheeler, F. J. and D. W. Nigg, "Three-Dimensional Radiation Dose Distribution Analysis for Boron Neutron Capture Therapy," Nuclear Science and Engineering (January 1992) (in publication).

11. Moran, J. M., et. al., "Macroscopic Heterogeneity Effects in Radiation Dose Distribution Analysis for Boron Neutron Capture Therapy," Medical Physics (1992) (in publication).

12. Gavin, P. R., et. al., "Canine Spontaneous Brain Tumors: A Large Animal Model for BNCT," Strahlentherapie und Onkologie 165: 225-228 (1989).

ACKNOWLEDGEMENT:

This study was performed under the auspices of the U.S. Department of Energy, Office of Energy Research, DOE Field Office, Idaho, Contract Number DE-AC07-76ID01570.