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Subject Monte Carlo Status Report

ASTRONUCLEAR LABORATORY

TO: F. D. Retallick
J. G. Gallagher

cc: W. H. Esselman
W. G. Roman
W. H. Arnold
D. W. Drawbaugh
D. McCutchan
H. C. Woodsum
R. N. Nassano

The purpose of this report is to summarize briefly the status of the work on shield Monte Carlo analysis, and to establish a tentative schedule for future work.

1.0 Description of Monte Carlo Program 18-0

Specialized Monte Carlo Program 18-0 is a digital computer code that applies Monte Carlo methods to simulate neutron and gamma ray histories in reactor-shield assemblies. This program is designed to determine nuclear heating rates in cylindrically symmetrical reactor-shield systems, and neutron and gamma ray leakage distributions in energy and angle for an equivalent point source. In addition, an "escape tape" option is provided in the program such that the parameters of all histories leaving the outer boundaries of the shield system are available on tape. This tape can be analyzed to determine flux distributions outside the reactor-shield assembly.

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Neutron events treated by the program are:

1. Elastic scattering
2. Inelastic scattering
3. Radiative capture
4. Neutron capture with alpha emission
5. $N, 2N$ reaction in beryllium
6. Absorption with no secondary emission.

Gamma rays events treated by the program are:

1. Compton scattering
2. Absorption
3. Gamma ray absorption with neutron emission (photoneutron).

The importance sampling techniques of splitting and Russian roulette in energy and region for neutrons, and in region for gamma rays, are optionally allowed in the program.

Program 18-0 was selected for analyzing the NERVA shield for several reasons:

1. It was readily available
2. It was used to great advantage by the author in analysis of ANP-type shields
3. It is a specialized code specifically designed for speed.

Although the code is not as versatile as the Flexible Monte Carlo Codes (FMCN and FMCG) in the use of importance sampling, in geometrical capabilities, and output, it does provide the use of the "escape tape" option. The usefulness

of this option cannot be over-emphasized. Essentially this option permits the running of any configuration in parts. For example, one could analyze in detail the NRX configuration without the shield in position. All histories escaping would be placed on tape permanently. The shield could then be analyzed using the histories on the tape. If any significant design changes should occur in the shield, the "escape tape" could be rerun with the "new" shield, thus eliminating the necessity of restarting neutrons and gamma rays from the reactor core. This option is a time saver. In the Flexible Monte Carlo Code, the entire reactor-shield assembly would have to be described in one run.

2.0 Status of Shield Monte Carlo Analysis

In order to describe adequately the status of this work, it is necessary to define the important types of input data required by the program. The present status of each type is indicated along with its description.

2.1 Neutron Data

2.1.1 Microscopic Cross Sections for Each Type of Event and Each Material

Basic neutron cross sections must be entered in tabular form as a function of energy. The energy group structure is limited to 25 groups. The basic cross sections are approximated by discontinuous straight line segments across the specified energy groups. STATUS: Cross sections have been compiled for the following elements: hydrogen, lithium, iron, chrome, nickle, carbon, beryllium, uranium 235 and 238, titanium, aluminum, manganese, and niobium. This compilation did not involve an extensive literature search because of limitations of time. Nor were any

nuclear or optical model codes used in the data preparation. It is believed that that the present compilation is sufficiently reliable to demonstrate the feasibility and reliability of the Monte Carlo method. Further cross sections compilation would be in the realm of basic research. The compiled cross sections were then reduced by Program 20-2 into the form required by the Monte Carlo code. Program 20-2 approximates the energy dependence of the cross section by discontinuous straight line segments across specified energy groups. The energy group structure was set as follows:

<u>Group</u>	<u>Lower E, Mev</u>	<u>Upper E, Mev</u>
1	1×10^{-8}	1.86×10^{-6}
2	1.86×10^{-6}	5×10^{-6}
3	5×10^{-6}	2×10^{-5}
4	2×10^{-5}	1×10^{-4}
5	1×10^{-4}	5.8×10^{-4}
6	5.8×10^{-4}	1×10^{-3}
7	1×10^{-3}	2×10^{-3}
8	2×10^{-3}	3×10^{-3}
9	3×10^{-3}	1×10^{-2}
10	1×10^{-2}	.025
11	.025	.045
12	.045	.07
13	.07	.1
14	.1	.2

<u>Group</u>	<u>Lower E, Mev</u>	<u>Upper E, Mev</u>
15	.2	.3
16	.3	.4
17	.4	.6
18	.6	.8
19	.8	1
20	1	2
21	2	3
22	3	4
23	4	6
24	6	8
25	8	10

All cross sections are now ready for use except those in the thermal group. The thermal cross sections are being obtained from the Reactor Analysis group for each particular material, and will be region averaged and flux weighted.

2.1.2 Anisotropic Elastic Scattering Probabilities as a Function Angle in the Center of Mass System

Anisotropic scattering data are required in tabular form as probabilities for each of the neutron energy groups described in Section 2.1.1. However, it is well known that at very low energies (for most materials) the elastic scattering can be considered isotropic. Therefore, an energy is specified in the program (in this case, 0.1 mev) below which all scattering will be treated as isotropic. STATUS: All known readily

available sources of elastic differential scattering cross section data were investigated. This investigation required an extensive literature survey of the data. No data are available for manganese; this element will be treated as iron or nickel pending the availability of new data. BNL 400 is to be revised this year and may contain such data. Program 20-4 was used to reduce the data to a form for the Monte Carlo code. This program averages the specified differential scattering cross sections over specified energy groups to obtain averaged angular distribution data in the form of cumulative probability tables. All elastic scattering data is now ready for use.

2.1.3 Low Energy Inelastic Scattering Data

Due to the fast core storage limitation of the 7090, Program 18-0 distinguishes between low and high energy inelastic scattering. Low energy is considered as the four lowest possible excited levels of the nucleus (each material). STATUS: The four lowest levels for each material have been compiled and are ready for use. The required probabilities of transition from level to level, and into each level, are among the most difficult data to obtain with any certainty. The availability data have been compiled and are ready for use. "Missing" data have been estimated.

2.1.4 High Energy Inelastic Scattering Data

2.1.4.1 Program 18-0 requires a tabulated distribution from which the energy of an inelastically scattered neutron particle is selected. These tables are in the form of probability distributions and were generated using Program 20-6. This program applies the evaporation model to obtain the neutron energy distribution. STATUS: Complete and ready for use.

2.1.4.2 The number of gamma rays emitted as a result of an inelastic scattering is necessary because the program only treats the first four levels exactly. STATUS: Basic data have been compiled. A Fortran Code, "Trap 1", was

written by Pat Heiser to facilitate the reduction of the basic data. No data reduction has been made. Therefore, these data are not yet ready for use.

2.1.5 The Number of Capture Gamma Rays Per n,γ Event For Each Material

Program "Trap 1" was used to reduce the basic available data.

STATUS: Complete and ready for use.

2.1.6 Total Macroscopic Cross Sections for Each Composition

Each geometrical region used to describe the reactor-shield assembly has associated with it a composition number. The composition number is a function of the volume fraction of each material in the region, and hence, the number density. Program 20-3 calculates the macroscopic cross sections given the microscopic cross sections and the volume fractions and/or number densities for each composition.

STATUS: Difficulty has been encountered in using the 20-3 binary deck. A new deck is being sent to WANL from General Electric. In the meantime, Pat Heiser coded a very condensed version of 20-3 in the event that the new General Electric 20-3 deck will not run. STATUS: Incomplete.

SUMMARY OF NEUTRON DATA

The incomplete data and approximate time required for completion are as follows:

1. Thermal cross sections - 3 weeks
2. Number of gammas from high energy inelastic - 1 week
3. Macroscopic total cross sections - 1 week

2.2 Gamma Ray Data

It is planned to generate all necessary gamma ray data during the time that the neutron calculations are being performed. There are two types of gamma ray input data closely allied to the neutron data, and must be generated in a consistent manner; a tabulated distribution for each material from which the energy of a gamma ray emitted in high energy inelastic scattering is chosen, and a similar table for selecting the energy of a radiative capture gamma. A program was written by Mr. Carl Saalbach of the Mathematics section to facilitate reduction of the basic data into tabular form.

The photoneutron event in beryllium will be included in this study.

2.3 Source Particle Generator

Program 20-0 generates and writes on tape the necessary parameters describing the histories to be run in the Monte Carlo code. These parameters include the energy and the location of each history.

Part of the input required for Program 20-0 consists of appropriate reactor axial and radial power distributions. A final source tape will not be made until adequate power distributions are made available. However, a tentative particle history for neutrons was made for three reasons:

1. To check out the code
2. To test the adequacy of 10,000 histories to mockup the core
3. To use as a test run for a time estimate in the main Monte Carlo program.

The most recent power distributions were used and a tape was successfully generated and analyzed. Program 20-8 is used to analyze a source tape. This program uses a statistical estimation technique to determine if the number of histories generated in the reactor is adequate. A total of 10,000 histories were generated.

In addition to the power distribution, the energy distribution must be supplied in Program 20-0. In the case of neutrons, the commonly available neutron fission spectrum is used. From past experience, (APEX 707), it was shown that the entire fission spectrum from 0 to 10 mev should not be attempted in one run, but should be separated into group runs. The energy grouping will be as follows:

1. 0 - 1 mev
2. 1 - 2 mev
3. 2 - 4 mev
4. 4 - 10 mev

The present tape is for a neutron source of from 1 mev to 2 mev. The final Monte Carlo results will then be weighted appropriately to preserve the total energy generated from a neutron fission spectrum. STATUS: A source tape is now ready for neutron histories between 1 and 2 mev. This tape will be used in a test run in the main Monte Carlo code. A final tape will not be made until satisfactory reactor power distributions are available.

2.4 Geometry Generator

The parameters required to describe the geometry in Program 18-0 are quite numerous and are tedious to obtain. The reason for this is because the

geometry tracing routine was coded for speed on the computer and to do so required an elaborate input description scheme. Program 20-7 is used to facilitate making up the geometry input data. This program punches a set of binary cards which are inserted into Program 18-0. STATUS: Program 20-7 was used to describe the B-4 core, reflectors, and aluminum support plate. Fifty-four regions were used in this description. The binary deck is ready to be used in the main program. Some difficulty may be encountered in locating the geometry data at the correct loading address in the main program. This geometry will be used as a time test run.

3.0 Plans and Schedule

The present plans for Monte Carlo shield analysis are:

1. To determine heating rates in the NRX-A-1 configuration in the following order:
 - a. Neutron heating
 - b. Gamma heating from the reactor core
 - c. Gamma heating from secondary gammas (inelastic and n,γ) originating both inside and outside the reactor core.
2. To determine flux distributions outside the reactor-shield assembly for use in improved propellant heating calculations.
3. To make any possible comparisons with experimental test results.

Any analysis using the Monte Carlo technique in a computer code has inherent difficulties and assumptions present such as energy group structures and validity of basic data, etc. However, the technique itself is superior to any now used

in shield analysis. Most of the difficulties involve the capacity of present day computers and lack of basic data. Computers are improving. The use of Monte Carlo also has instigated a "new look" at basic data experiments and analytical approaches to fill in the gaps of basic data.

The present NERVA shield design is not a particularly thick shield, but is a complex structure. Both the shield thickness and the geometrical complexity of the design affect the running times for Monte Carlo calculations. Therefore, no time estimates can be established until initial runs are made. As mentioned in Section 1.0 of this report, the "escape tape" option will be used, and the first runs will not have the shield described. The shield geometry will be generated while the core, reflectors, and aluminum support plate data are being acquired. It is estimated that initial test runs will be made during the first week of December. As soon as feasible, after test runs are made, a time schedule (and cost schedule) will be determined. No final runs will be made until the NRX-A-1 drawings are released, and until the reactor loadings and power distributions are known. It may happen that one or two additional materials will be necessary for the NRX-A-1 configuration. In that event, a delay of about ten days will be necessary for tracking down basic data.

It should also be pointed out that the output from Program 18-0 will be compared to present heating rate data from diffusion, transport, and point kernel codes, all of which appear to differ in their results.

SIGNED

M. A. Capo