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USE OF THE MCNP CODE IN THE DESIGN OF

A ^{252}Cf IRRADIATOR

A Thesis

Presented to

The Faculty of Nuclear Science

San Jose State University

In Partial Fulfillment

of the Requirements for the Degree

Master of Science

by

Yadollah Yadi Aliakbar

December, 1996

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A handwritten signature in cursive script, reading "Kiumars Parvin".

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ABSTRACT

USE OF THE MCNP CODE IN THE DESIGN OF A ^{252}Cf IRRADIATOR

by Yadollah Yadi Aliakbar

The Nuclear Science Facility at San Jose State University is designing a ^{252}Cf -based neutron irradiator to permit neutron activation analysis of small samples. The Monte Carlo Neutron and Photon (MCNP) code was selected to help determine the design that would maximize the fully thermalized flux over a 30 ml cylindrical sample volume. The neutron flux, spectrum, and gradient over the sample volume were calculated. Use of the program has simplified the design process and provides the basis for estimation of minimum detectable concentrations for different analytes in different matrices. Although this was tanker developmental work, not final architecture, but it is a fine design for starting.

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To My Wife, Sadaf Najmabadi

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I. INTRODUCTION

1.1. Purpose

The subject of this thesis is the application of the Monte Carlo Neutron and Photon (MCNP) code (see App. I for more details on MCNP code) in the design of a ^{252}Cf irradiator (see Fig. 1). The Nuclear Science Facility at San Jose State University uses this irradiator to perform Neutron Activation techniques on some analytes of interest (Wang et al. 1975). The principal purpose of this project is to calculate the maximum thermalized neutron flux and gradient over the sample volume.

The emerging neutrons (with the energy ranges from below thermal energies to 20 MeV) from the ^{252}Cf sources are moderated by the surrounding polyethylene materials. Hereby the entire irradiator tank becomes a source of thermal neutrons. Some of these neutrons are absorbed by the sample (^{23}Na) and produce radioactive (^{24}Na). The radionuclide ^{24}Na beta decays with half-life 14.5 hours to ^{24}Mg . This decay is accompanied by emission of characteristic high-energy photons as the ^{24}Mg nucleus de excites to its ground state. By knowing the neutron flux in the sample and measuring the gamma radiation emitted by the sample, one can use Neutron Activation Analysis (NAA) to calculate the ^{23}Na atoms within sample (see App. II for more details on NAA). To maximize the fully thermalized flux over the sample (30 ml cylinder) it is necessary to compare the sample neutron flux for various positions (source-sample distance and vertical displacement). The charts of these tables helps us find the position of the optimum flux. Similarly, we make a table of neutron flux vs lengths and widths of the sample (see App. III). The three-dimensional chart of this table shows how the neutron flux is distributed within the sample (Stoddard et al. 1971).

To find the maximum neutron flux over the sample (Butler 1956), the relative position of the co-planar sample and sources was varied from 3.4 cm (center-to-center), the shortest distance, to 9 cm, a reasonable cut-off for calculation within the irradiator cask (see Table 6). The chart of flux vs sample-source distance shows the neutron flux changes and the 3.4 cm optimum point (see Chart 4).

After selecting the 3.4 cm - point as the preferred radial separation (see Table 7), we gradually move the sample away from the source plane both upward and downward (due to asymmetry of the source capsule). The chart of flux vs vertical position for the [-5, 5] centimeter range (see Chart 5) shows the symmetric property of the flux for the vertical displacements and suggests the achievement of the maximum flux at zero vertical displacement (i.e., 3.4 cm). It also shows how the efficiency of the flux decreases as the sample position is displaced from the plane of the sources. To study the flux gradient within the sample the panoramic neutron flux was plotted by a three-dimensional chart (see Chart 8).

1.2. Outline

This thesis is divided into three parts. Part I contains the description of the irradiator, geometry and materials, ²⁵²Cf source specifications, absorbed neutrons in cylindrical sample volume, and application of Neutron Activation Analysis and flux mapping. Part II focuses on the applications of the MCNP code as a design tool for simulation of Neutron Activation and its results and analyses. Finally, part III provides the summary and conclusion.

2. Radiation Properties of ^{252}Cf

^{252}Cf has two modes of decay (Stoddard et al. 1971; Mannhart 1987). For 97% of the ^{252}Cf atoms, decay is by the alpha particle emission mode, and for 3% by the spontaneous fission (SF) process with the emission of 2.4×10^{12} neutron / sec (see Tables 1, 2, and 3) from one gram of this material (see also Charts 1A, and 1B). The spontaneous fission decay follows the Watt formula (USDOC 1960) for the neutron-energy spectrum (see Charts 2, and 3).

$$N(E) = 0.373 \exp(-0.88E) \sinh(2.0E)^{1/2} \quad (1)$$

where $N(E)$ is the fraction of neutrons per unit energy range and E is the neutron energy in MeV.

3. Description of Irradiator

The irradiator tank (see Fig. 1) is filled with polyethylene moderator to thermalize the neutrons and to reduce the dose rate at the periphery (Wang 1992; Chilton 1984). There are four 50-mg ^{252}Cf sources, symmetrically distributed around the sample. To apply the MCNP program requires a sample suitable for neutron activation. Sodium is a good example due to its importance in food products. The suspended sample is at the center of the spherical cask. All sources and sample guide tubes within the irradiator cask are aluminum.

3.1. Geometry and Materials

The irradiator design which is early developmental work, not the final architecture, employs a 100-cm spherical cask that is filled with polyethylene moderator (see Fig. 1). The cylindrical sample (see Figs. 3 and 4) is suspended by remote control at the center

of the cask while four Cf-252 sources (see Fig. 2), cylindrical in shape, are symmetrically positioned around the sample capsule and within the plane perpendicular to the sample guide tube. Due to the symmetric nature of this design, we can multiply the result from a single source by a factor of 4 to find the total absorbed flux in the sample from four equidistant sources. All the guide tubes and source capsules are made of aluminum.

Fig. 1. ^{252}Cf irradiator cask

3.2. Source Specification

There are four 50-mg ^{252}Cf sources in the form of sesquioxide (Cf_2O_3), each surrounded by a matrix of aluminum pellets (see Fig. 2) which in turn are doubly encapsulated in aluminum cylinders (Hampel 1968; Seaborg 1958).

Fig. 2. Oak Ridge National Laboratory (ORNL) model LSD neutron source.

3.3. Sample

Sodium is an important analyte in food products due to its potential for adverse health effects (Wang et al. 1975). In the irradiator design, a 30 ml cylindrical food sample (see Fig. 3) is positioned at the center of the cask, surrounded by polyethylene material.

Fig. 3. Cylindrical sample and capsule.

Fig. 4. Five-segment division of sample.

Table 1. Twenty -two group source distribution for ^{252}Cf .^a

Group	Energy interval (MeV)	Type	Source
		Radiation	Distribution (1/disintegration)
1	10.0 - 14.92	Neutron	0.0036
2	6.70 - 10.00		0.0279
3	5.49 - 6.70		0.0356
4	4.49 - 5.49		0.0547
5	3.68 - 4.49		0.0730
6	3.01 - 3.68		0.0869
7	2.02 - 3.01		0.1894
8	0.91 - 2.02		0.3033
9	0.41 - 0.91		0.1462
10	0.11 - 0.41		0.0676
11	0.015 - 0.11		0.0119
12	0.0000041 - 0.015		0.0
13 ^b	0.0 - 0.0000041		0.0
14	6.0 - 10.0	Gamma	0.000212
15	5.0 - 6.0		0.000991
16	4.0 - 5.0		0.00354
17	3.0 - 4.0		0.01
18	2.0 - 3.0		0.02
19	0.9 - 2.0		0.19
20	0.4 - 0.9		0.42
21	0.12 - 0.4		0.35
22	0.01 - 0.12		0.0

^a Stoddard (1971).

^b Called thermal neutrons.

CHART 1A

Chart 1a. Plot of ^{252}Cf neutron source energy distribution.

CHART 1B

Chart 1b. Plot of ^{252}Cf gamma source energy distribution.

Table 2. Neutrons from spontaneous fission of ^{252}Cf .^a

Energy (MeV)	(n / sec- g of nuclide)
0.0 - 0.5	2.8×10^{11}
0.5 - 1.0	3.7×10^{11}
1.0 - 2.0	7.6×10^{11}
2.0 - 3.0	4.6×10^{11}
3.0 - 4.0	2.8×10^{11}
4.0 - 5.0	1.6×10^{11}
5.0 - 6.0	5.6×10^{10}
6.0 - 7.0	4.0×10^{10}
7.0 - 8.0	1.3×10^{10}
8.0 - 10.0	9.9×10^9
10.0 - 13.0	2.2×10^9
Total	2.43×10^{12}

^a Stoddard (1971).

CHART 2

Chart 2. Plot of neutrons from spontaneous fission of ^{252}Cf vs energy. Unit in Y-axis is scaled down by 11 orders of magnitude.

Table 3. Gamma rays from spontaneous fission of ^{252}Cf .^a

Energy (MeV)	Abundance (p/sec-g of nuclide) Equilibrium Fission		
	Prompt Gammas	Product Gammas	Total
0.0 - 0.5	3.30E+12	1.30E+12	4.60E+12
0.5 - 1.0	1.70E+12	4.00E+12	5.70E+12
1.0 - 1.5	7.70E+11	9.10E+11	1.70E+12
1.5 - 2.0	4.20E+11	<u>3.50E+11</u>	7.70E+11
2.0 - 2.5	2.20E+11		2.20E+11
2.5 - 3.0	1.10E+11		1.10E+11
3.0 - 3.5	5.60E+10		5.60E+10
3.5 - 4.0	3.00E+10		3.00E+10
4.0 - 4.5	1.70E+10		1.70E+10
4.5 - 5.0	8.20E+09		8.20E+09
5.0 - 5.5	4.90E+09		4.90E+09
5.5 - 6.0	1.80E+09		1.80E+09
6.0 - 6.5	1.00E+09		1.00E+09

^a Stoddard (1971).

CHART 3

Chart 3. Plot of gamma rays from spontaneous fission of ^{252}Cf vs energy.

Unit in Y-axis is scaled down by 12 orders of magnitude.

II. RESULTS AND DISCUSSION

1. Maximizing the Fully Thermalized Flux over a 30 ml Cylindrical Sample Volume

To find the maximum neutron flux over the sample volume (Stoddard et al. 1971; Butler 1956), the relative position of the co-planar sample and sources was varied from 3.4 cm center-to-center, the shortest possible distance due to the physical sizes of the source and sample, to 9 cm, a reasonable cut-off for calculation within the irradiator cask (see Table 4). The chart of flux vs sample-source distance shows the neutron flux changes and the 3.4 cm optimum point (see Chart 4).

After selecting the 3.4 cm - point as the preferred radial separation (see Table 5), we gradually move the sample away from the horizontal source plane both upward and downward (due to asymmetry of the source capsule). The chart of flux vs vertical position for the [-5, 5] centimeter range (see Chart 5) shows the symmetric property of the tally for the vertical displacements and suggests the achievement of the maximum flux at zero vertical displacement (i.e., 3.4 cm). It also shows how the efficiency of the tally flux decreases as the sample position is displaced from the plane of the sources.

Table 4. Sample neutron flux vs sample - source distance.

Distance (cm)	(n /sec - cm ²)x10 ⁻⁶ / neutron emitted
3.40	3.841234
3.60	3.826590
3.80	3.643240
4.00	3.549300
4.50	3.455370
5.00	3.038410
5.95	2.736600
6.05	2.728550
6.15	2.586030
6.25	2.674410
6.35	2.594434
7.00	2.428620
8.00	2.534276
9.00	2.504645

CHART 4

Chart 4. Plot of neutron flux in sample vs sample-source distance. Unit in Y-axis is scaled up by 6 orders of magnitude.

Table 5. Sample neutron flux vs vertical shifting.

Shifting (cm)	(n / sec - cm ²)x10 ⁻⁶ / neutron emitted
-5	1.05
-4.5	1.20
-4	1.35
-3.5	1.43
-3	1.50
-2.5	1.80
-2	2.12
-1.5	2.20
-1	2.27
-0.5	3.22
0	3.84
0.5	3.25
1	2.27
1.5	2.18
2	2.10
2.5	1.80
3	1.51
3.5	1.47
4	1.31
4.5	1.14
5	1.01

CHART 5

Chart 5. Plot of neutron flux vs vertical shifting of sample. Unit in Y-axis is scaled up by 6 orders of magnitude.

2. Study of the Absorbed Neutron Distribution in the Sample

To study the neutron absorption gradient within the sample (Wang et al. 1975), first consider the axial distribution of absorption using a subroutine program called AXIAL (see App. V.1 for programs and App. V.2 for tally printout of IS program). In this MCNP program (Butler 1956; Kingman et al. 1966), the sample cylinder is divided into five smaller constituent disks (see Fig.4). Of course, increasing the number of the divisions provides better resolution at increased cost, and ultimately better estimation of minimum detectable concentrations; see Table 8. This program enables us find the amount of absorption by each cylinder.

The plot of neutron flux over individual disk vs position with respect to the central disk demonstrates the distribution density along the axis of symmetry of the sample (see Chart 6). This distribution is symmetric with respect to the plane perpendicular to the axis of the cylinder at its center. It also demonstrates the highest density for the central disk.

Table 6. Axial neutron flux distribution of cylindrical sample from single source.^a

Cylindrical position (cm)	(n / sec - cm ²) / neutron emitted
13	2.74E-06
14	2.91E-06
15	3.18E-06
16	2.91E-06
17	2.72E-06

^a The 15-cm position is the half-height of the cylinder which is used as the reference point for vertical shiftings.

CHART 6

Chart 6. Plot of axial distribution of neutron flux in sample vs axial distance for a single irradiator source. Unit in Y-axis is scaled up by 6 orders of magnitude and 15-cm, the half-height of the cylinder is chosen as the reference point for X-axis.

To study how the absorbed neutrons are distributed both radially and panoramically within the sample, pieces of the cylinder are cut parallel to the axial plane (see Fig. 5); see App. II. A program called AZIMUTHAL (Spanier et al. 1969) gives the density distribution over the axial slices (see Table 7); see App. V.1 for programs and App. V.2 for tally printout of IS program. The symmetry of the distribution of the neutron flux is then found on both sides of the central slice (see Chart 7). It also shows the highest density for the central slice. Combining Chart 7 for both the length and width (see App. II) will give the general absorption density distribution within the sample (see Chart 8).

Table 7. Panoramic neutron flux distribution of sample.

Slice position (cm)	(n / sec - cm ²)x10 ⁻⁶ / neutron emitted
-0.452550	3.042160
-0.226270	3.055700
0	3.063800
0.226270	3.056620
0.452548	3.044092

Fig. 5 Rectangular sample and divisions

CHART 7

Chart 7. Plot of panoramic neutron flux in sample vs slice position for a single irradiator source. Unit in Y-axis is scaled up by 6 orders of magnitude.

III. SUMMARY AND CONCLUSION

We began by introducing the purpose and the outline sections of this thesis work in the first chapter. The geometry, materials, source specification, radiation properties of Cf-252, and the sample were discussed. The second chapter is entirely devoted to the applications of the MCNP code as a design tool for simulation of Neutron Activation and its results and analyses. To study the possible effect of the source dimension on the output results, we wrote two different sets of input files called IS (for the point source) and FIT (for the real situation) which is discussed in Appendix V. The results showed no significant differences in the output tallies, which in turn demonstrates the plausibility of the point source approach to this problem.

A separation of 3.4 cm from the source center to the sample center provides the highest thermalized neutron flux of $3.84 \times 10^{-6} \pm 0.05$ n/sec-cm² per source (see II.1). This maximum flux occurs when the sources and samples are level. The collection of the axial distribution of the flux over the sample cylinder indicates the symmetric property of the distribution, while the central constituent cylinder has the highest flux density. The panoramic study is the last section of this chapter which considers the functionality of the neutron absorption density over the sample. To study the flux gradient within the sample the panoramic neutron flux was plotted by three-dimensional chart (see II.2, and App. III). The arrangement of four sources at the center of a square centered on the sample yields a neutron flux density that varies from $6.06 \times 10^{-6} \pm 0.05$ n/sec-cm² to $6.12 \times 10^{-6} \pm 0.05$ n/sec-cm² for each original source.

APPENDIX I

Monte Carlo Neutron and Photon Interaction Code (MCNP)

1.1. History and Structure

The Monte Carlo method as a radiation transport research tool came from work done at Los Alamos during World War II. Enrico Fermi, J. Von Neumann, and S. M. Ulam (Ulam et al. 1947; Ehrlich 1973) from Los Alamos National Laboratory are generally called the founding fathers of Monte Carlo techniques as a mathematical discipline. Many followers in various laboratories developed their ideas, but the name of Von Neumann is attached to many, if not most, of the fundamental ideas and techniques. Metropolis and Richtmyer at Los Alamos did some of the early part of the Monte Carlo development, and Enrico Fermi invented a mechanical device called FERMIAC (USDOE 1991) in 1947 while he was working at Los Alamos. The purpose of this machine was to trace neutron movements through fissionable materials by the Monte Carlo method. Since the inception of Monte Carlo at Los Alamos, it has had a number of devoted and well-known people who assisted in its development and establishment.

With the help and guidance of Ulam in early 1950s, a small band of scientists at Los Alamos became engaged in investigating transport methods, and using Monte Carlo (USDOE 1991) as a tool for solving transport problems. The rudimentary calculational effort finally gained momentum with the completion of the MANIAC computer, and rapidly became a factor in much of the research at Los Alamos (USDOE 1991; Parzen 1960).

In 1965, several more features were added to the Monte Carlo neutron code MCN (Spanier et al. 1969). These features included improved and expanded cross-section libraries, general sources, general physical geometries, standard variance reduction techniques, thermalization based on the free gas model, and general tallies and output.

The photon codes MCG and MCP were then added; these two both dealt with photon transport, MCG treating higher energy photons or gamma rays and MCP treating photons of energies down to 1keV (USDOE 1991).

In 1973 MCG and MCN were merged to form MCNG (USDOE 1991; Parzen 1960), which in turn was the final step towards the first version of MCNP, which was released in 1977. To enable the gamma ray produced by neutron interaction to be evaluated, photon production cross-sections were added to the cross-section libraries. MCNP is the MCNG form plus many additions in physics, mathematics, and user oriented features. Since the first version of MCNP was released, there have been several editions (Carter et al. 1975; Hammersley et al. 1964; Khan 1954; Schreider 1966; Loeve 1950).

MCNP can be run in one of three modes: neutron transport only, combined neutron/photon transport, and photon transport only. In the combined neutron/photon mode, photon histories are followed as part of their parent neutron histories. Finally MCNP has been made as system independent as possible to enhance its portability, and has been written to comply with the ANSI FORTRAN 77 standard. MCNP can be used on CRAY, CDC, IBM, VAX, PRIME, SIEMENS, RIDGE, APOLLO, and SUN computers.

1.2. Cross-Sections:

The MCNP code package is complete with the associated nuclear data tables. There are two objectives in preparing nuclear data tables for MCNP. First, it is essential that the data tables reproduce the original evaluated data as much as is practical. Second, new data should be incorporated in a timely manner so that calculations are based on the most recent evaluations.

The following are the five classes of nuclear data tables used in MCNP (USDOE 1991):

- (1) continuous-energy neutron interaction data,
- (2) discrete reaction neutron interaction data,
- (3) photon interaction data,
- (4) neutron dosimetry cross sections, and
- (5) neutron $S(\alpha, \beta)$ thermal data

where $S(\alpha, \beta)$ is the chemical binding and crystal structure effects.

Here the photon data are atomic rather than nuclear. In mode N and mode NP problems, for each isotope or element in the problem, one continuous-energy or discrete-reaction neutron interaction table is required. Similarly, one photon interaction table is required for each element in a mode NP or mode P problem.

Thermal $S(\alpha, \beta)$ tables may be appropriate if the neutrons are transported at sufficiently low energies that molecular binding effects cannot be ignored. Cross sections from dosimetry tables may be used as responding functions with the FM (format) card to determine reaction data. MCNP may read from data tables in any of three formats. Data tables are then transmitted between computer installations in 80-column card-image BCD format (Type-1 format). An auxiliary processing code, MAKXSF, will convert the BCD files to standard unformatted binary files (Type-2 format) allowing more economical access during MCNP execution.

1.3. Physics

There are two important concepts in the physics of MCNP for setting up the input and for understanding the output. These are particle weight and particle tracks.

1.3.1. Particle Weight

If MCNP is used only for the purpose of simulation of the exact physical transport, then each MCNP particle would represent one real physical particle and would have a unit weight. But for computational efficiency, MCNP allows many techniques to be employed that do not simulate physical transport exactly. For example, each MCNP particle might represent a number W of particles emerging from a source. The number W is then the initial weight of the MCNP particle. The W physical particles all will have different random walks, but the one MCNP particle representing these W physical particles will only have one random walk.

For example, if the particle goes directly away from the region of interest, we assume that it will never be scattered back to the region of interest, and terminate that random walk. Obviously this is not an exact simulation; on the other hand, the real number of physical particles is preserved in MCNP in the sense of statistical averages and ultimately in the limit of large particle numbers. Of course, particle production or loss is included if it occurs.

Each MCNP particle result is multiplied by W so that the full results of W physical particles which are represented by each MCNP particle are exhibited in the final results (tallies). This procedure makes it possible for users to normalize their calculations to the desirable source strength. The default normalization is set to weight one per MCNP particle.

For the expected means to be independent of the number of source particles actually initiated in the MCNP calculation, a second normalization to the Monte Carlo histories is necessary. The application of particle weight, however, goes far beyond the simple normalization of the source. Every Monte Carlo biasing technique (Butler 1956;

USDOE 1991) alters the random walk paths executed by the particles. This biasing technique is used to increase the number of those particles that sample some part of the problem of special interest (1) without increasing the sampling of less interesting parts of the problem and (2) without erroneous effects on the expected mean physical result (see App. IV).

Although the proper application of this procedure increases the precision in the desired result compared to an unbiased calculation, the computation time is the same. For instance, if an event is made $\sqrt{2}$ times as likely to occur (without biasing), the tally ought to be multiplied by inverse of $\sqrt{2}$ i.e., $1/\sqrt{2}$ to keep the expected average tally unaffected. This tally multiplication will be accomplished by multiplying the particle weight by $1/\sqrt{2}$ because the tally contribution by a particle is always multiplied by the particle weight in MCNP.

In short, the particle weight W is simply a number carried along with each MCNP particle that represents that particle's relative contribution to the final tallies. The magnitude of W is checked and determined so as to ensure that whenever MCNP deviates from an exact simulation of the physics, the expected value of physical results are nonetheless preserved in the sense of statistical averages, and therefore in the limit of large MCNP particle numbers. Its utility is in the multiplication of the number of particles, sampling just a part of the problem to improve the precision of selected results without the need for a full unbiased calculation - with its added cost in computing time - to achieve the same results and precision.

1.3.2. Particle Tracks

When a source particle starts out, a particle track is created; a track refers to each component of a source particle during its history. On this track, if two particles are produced after an interaction, a second track is created and now there are two tracks from

the original source particle, each with one-half of the single track weight. If an $(n, 2n)$ reaction happens on one of the tracks, one more track is started for a total of three.

Some output tallies use the length of a track in a given cell to find out the quantity of interest; others tally at each collision point or surface crossing (see part III.1.1 for cell and surface definitions). Inside a given cell with fixed composition, the method of sampling the collision along a track is used.

The probability of a first collision for a particle positioned between x and $x + dx$ on its line of flight is given by (USDOE 1991)

$$P(x)dx = e^{-\Sigma_t x} \Sigma_t dx \quad (1)$$

where Σ_t is the microscopic total cross section of the medium, the probability per unit length of a collision. By defining ξ , the random number on the domain $[0, x]$, to be

$$\xi = \int_0^x e^{-\Sigma_t s} \Sigma_t ds = 1 - e^{-\Sigma_t x} \quad (2)$$

(s is used to in place of x to avoid confusion with the limit of integration x),

then

$$l = - (1 / \Sigma_t) \ln(1-\xi) \quad (3)$$

Because $1-\xi$ is distributed the same way as ξ and hence can be replaced by ξ , we obtain the well-known expression for the distance before collision.

$$l = (-1 / \Sigma_t) \ln \xi \quad (4)$$

1.3.3. Neutron Interactions

A collision between a target atom and a neutron is affected by the thermal motion of the atom; the collision is also influenced by the presence of other nearby atoms (Chilton 1984; Spanier et al. 1969; Attix et al. 1968; Knoll 1989). The thermal motion cannot be ignored in many applications of the MCNP code without serious error. The influences of the nearby atoms are important in some applications. MCNP uses the thermal treatment based on the free gas approximation model to account for the thermal motion. Although MCNP has an explicit $S(\alpha, \beta)$ capability that considers the effects of chemical binding and crystal structure, it is available for only a limited number of substances at ambient temperature.

Basically the regular thermal treatment in MCNP assumes that the medium is a free gas. It also assumes that within the range of the atomic weight and neutron energy (the region the thermal effects are significant), the elastic scattering cross section at zero temperature is independent of the energy of the neutron particle. This ultimately concludes that the reaction cross sections are nearly independent of temperature.

Based on these assumptions, MCNP then can have a thermal treatment of neutron collisions that runs almost as fast as a completely non-thermal treatment and that is sufficient for most practical problems.

With these assumptions, the thermal treatment of cross sections consists of raising the value of a zero-temperature elastic cross section by the following factor

$$F = (1 + 0.5 / a^2) \operatorname{erf}(a) + \exp(-a^2) / [a * (\pi)^{1/2}] \quad (5)$$

where $a = (AE/kT)^{1/2}$, A = atomic weight, E = neutron energy, and T = temperature.

For fast neutrons, F can be approximated by

$$F = 1 + 0.5 / a^2 \quad (6)$$

when $a \geq 2$, and by linear interpolation in a table of 51 values of $a \cdot F$ when $a < 2$. These approximations have relative errors less than 0.0001. The total cross section is increased by the amount of increase in the elastic cross section. The thermal treatment of cross sections is done partly in the setup of the problem and partly during the actual transport calculation. The thermal adjustment is ignored if the neutron energy exceeds $500kT / A$; for this energy the adjustment of elastic cross section would be less than 0.1%.

2.1. Monte Carlo Mean, Variance, and Standard Deviations

Monte Carlo results are obtained by sampling possible random walks and assigning a score $x(i)$ (for example, $x(i)$ = energy deposited by i -th. random walk) to each random walk (USDOE 1991; Knuth 1981; Fitzgerald et al. 1967). The mean is:

$$\bar{x} = (1/N) \sum_{i=1, \dots, N} x(i) \quad (7)$$

The estimated variance of $\bar{x}$ is given by

$$S^2(\bar{x}) = \sum_{i=1, \dots, N} [x(i) - \bar{x}]^2 / N \quad (8)$$

where S is estimated standard deviation. The relative error is defined as:

$$R = S(\bar{x}) / \bar{x} \quad (9)$$

and the empirical guideline for interpreting the relative error R is shown in Table 8.

2.2. Estimation of Monte Carlo Precision

The results of Monte Carlo are an average of the contributions from many histories calculated during running the problem (Knuth 1981; Shreider 1966; Fitzgerald et al. 1967). The tally (an important quantity equal in stature to the Monte Carlo answer) itself has statistical error or uncertainty associated with its results. This error not only gives the opportunity to the user to gain an insight into the quality of the results, but also can determine if a tally is well behaved (by examining error and uncertainty data).

If a tally is not well behaved, the estimated error associated with the result generally will not reflect the true confidence interval of the result sought, and thus, these

answers could be completely erroneous. The general rule is to carefully examine the MCNP estimated means and relative errors, and associated figure of merit tables (FOM tables) in the tally fluctuation charts. A constant FOM ensures a well-behaved and meaningful tally.

Table 4. Guidelines for interpreting the relative error R.

Range of R value	Quality of the Tally
0.5 to 1	Poor
0.2 to 0.5	Suitable for first approximation only
0.1 to 0.2	Moderately reliable
< 0.10	Generally reliable except for point detector
< 0.05	Generally reliable for point detector

3.1. The MCNP Figure Of Merit (FOM)

Three factors that affect the efficiency of a Monte Carlo solution are (1) history-scoring efficiency, (2) dispersions in nonzero history scores, and (3) computer time per history. All three factors are included in the FOM.

3.1.1. Definition

The figure of merit (FOM) of a tally is defined as:

$$\text{FOM} = 1/(R^2T) \quad (10)$$

where R is relative error and T is the computer time in minutes that is used in MCNP to run the problem (which is directly proportional to N, the running number of particles). For a well-sampled problem, the FOM will approach a constant as N becomes very large. It is a relative indication of the computational efficiency of an MCNP calculation - the larger the better (see ref.12 App. II).

The FOM is a very important statistic about an output tally bin and gives us useful information that should be studied by the user. If the tally is well behaved, the FOM should be approximately constant with the possible exception of statistical fluctuation at the very beginning of the problem.

The numerical value of the FOM is better appreciated by considering the relation

$$R = 1/(\text{FOM} \cdot T)^{1/2} \quad (11)$$

Table 9 shows the expected value of R which should be produced in a one-minute problem ($T = 1$) as a function of the value of the FOM.

Clearly it is advantageous to have a large FOM for a problem because the computer time required to reach a desired level of precision is proportionally reduced. The above equation (eqn. 11) shows that if the FOM for a problem is doubled, the computer time required to achieve the same R is halved.

TABLE 9. R values as a function of FOM for T=1 Minute.

FOM	1	10	100	1000	10000
R	1.0	0.32	0.10	0.032	0.010

3.1.2. Application

In summary, there are three possible uses for FOM. The most important one is its use as a tally reliability indicator. If FOM is not approximately a constant (except for the statistical fluctuations at the early part of the problem), the confidence intervals may not overlap the precise answer as expected. The second use of the FOM is to optimize the efficiency of the Monte Carlo calculation by making several short test runs with different variance reduction parameters and then selecting the method that gives the largest FOM; remember that the FOM for a small number of histories quite possibly could cloud the selection of techniques competing for the computer time which is required to reach a desired value of R by using T proportional to $1 / R^2 * FOM$. For more details about the variance reduction techniques, see ref. 12, pp 116-147 .

APPENDIX II

Neutron Activation Analysis (NAA)

One of the processes by which a neutron is attenuated is absorption by a nucleus (Moe et al. 1992). For many substances, this process produces a compound nucleus which is unstable. The substance then becomes radioactive. Many artificially radioactive substances are produced when other substances are exposed to the neutron fluence in a reactor. Absorption occurs mainly for thermal neutrons. For this reason, most artificial radionuclides have been produced in reactors. When a nucleus absorbs a neutron, the term activation is used to imply that the product becomes radioactive. The thermal activation cross section, σ_{act} , expresses the probability that a certain substance will be activated if it is exposed to a thermal neutron flux. However, activation by neutrons of energy greater than thermal energy also occurs.

Assume a sample containing N atoms is exposed to a thermal neutron fluence rate ϕ . If σ_{act} is the probability of activation per atom, the total probability is $N\sigma_{act}$. The rate of formation of radioactive atoms is then

$$\text{formation rate} = \phi \sigma_{act} N. \quad (1)$$

During the time that radioactive atoms are being formed, some of the atoms will decay. The decay rate of the radioactive atoms N^1 will be given by

$$\text{decay rate} = \lambda N^1, \quad (2)$$

where N^1 is the number of radioactive atoms present and λ is the decay constant. The net rate of growth is then

$$\text{rate of growth} = \text{formation rate} - \text{decay rate}.$$

The corresponding differential equation is

$$dN^1/dt = \phi \sigma_{act} N - \lambda N^1 \quad (3)$$

here N^1 gives the number of radioactive atoms present after an irradiation time t . The activity A_t (in dis/s) of the sample at the time t that the irradiation is stopped is expressed by

$$A_t = \lambda N^1 = \phi \sigma_{\text{act}} N (1 - e^{-\lambda t}) \quad (4)$$

The activity at time t_1 after a sample is removed, is given by:

$$\begin{aligned} A &= A_t e^{-\lambda t_1} \\ &= \phi \sigma_{\text{act}} N (1 - e^{-\lambda t}) e^{-\lambda t_1} \end{aligned} \quad (5)$$

The activity A_t of the sample in the neutron field will increase until the formation rate is equal to the decay rate. At this point, no further increase in activity will occur. For a given sample, the maximum value that the activity will reach is $\phi \sigma_{\text{act}} N$. This is called the saturation activity A_s . The term $(1 - e^{-\lambda t})$ varies from 0 for irradiation time t equal 0, to 1 for infinite irradiation time. For any given irradiation time t , this term gives the fraction of the saturation activity A_s which the irradiated sample attains.

The equation is valid if all target atoms are in the same neutron field. In effect, this means that the samples must be thin. Also, the relationship assumes that the number of target atoms remains about constant. This means that there should be no significant decrease in the number of target atoms during the irradiation.

For thick samples, the neutron field is reduced by the presence of a neutron absorber. This loss is called the flux depression. In addition, the fluence rate at the inner central portion of the sample will be reduced due to absorption of neutrons by the outer layers of atoms. This is referred to as the self-shielding effect. Both of these perturbing effects need to be accounted for in thermal neutron absorption. As the energy of the neutron increases, these perturbation effects become less severe and the corrections are reduced in magnitude.

APPENDIX III

Derivation of Neutron Absorption Function for Panoramic Case

To study how the absorbed neutron is distributed both radially and panoramically within the sample, pieces of the cylinder are cut parallel to the axial plane (see Fig. 5).

To simplify the calculation of neutron distribution in the sample we can approximate the sample cylinder with the closest rectangular parallelepiped. As is demonstrated in Fig. 6, we have divided the length and width of it into five identical increments, where l and w are the length and width of the rectangular sample.

Fig. 5. Rectangular sample and divisions.

To derive the formula for the neutron distribution in the sample, we first need to know the neutron distribution within each individual slice.

We then define the following functions:

$$f_{lr} =: \text{flux from left to right}^a \quad (\text{II.1})$$

$$\text{and} \quad f_{rl} =: \text{flux from right to left} \quad (\text{II.2})$$

for any xi slice.

But because of the following symmetries (see Fig. 5)

$$f_{lr}(xi) = f_{rl}(xi), \quad (\text{II.3})$$

$$f_{lr} = f_{lr}(x1) + f_{lr}(x2) + \dots \quad (\text{II.4})$$

$$\text{and} \quad f_{rl} = f_{rl}(x1) + f_{rl}(x2) + \dots \quad (\text{II.5})$$

$$\text{We can get} \quad f(xi) = f_{lr}(xi) + f_{rl}(xi), \quad (\text{II.6})$$

$$\text{or} \quad f(xi) = 2f_{lr}(xi) \quad (\text{II.7})$$

Similarly for slice yi we get

$$f(yi) = 2f_{fb}(yi) \quad (\text{II.8})$$

where the f and b indices stand for forwards and backwards directions respectively.

But the flux density is defined as:

$$F(xi) =: f(xi) / Vi = f(xi) / S * dxi = f(xi) / l * w \quad (\text{II.9})$$

where the Vi, S, and dxi stand for volume, surface, and a change in xi respectively for the ith. slice.

In a similar manner we get

$$F(yj) =: f(yj) / l * w \quad (\text{II.10})$$

Therefore, for the total flux density we get

$$F(xi, yj) =: F(xi) + F(yj) = (f(xi) + f(yj)) / l * w \quad (\text{II.11})$$

with the following plot:

^a Here the symbol "=: " is used for "definition"

CHART 8

CHART of $F(X_i, Y_j)$

Chart 8. Plot of panoramic neutron flux distribution in sample. Unit in Z-axis is scaled up by 6 orders of magnitude.

APPENDIX IV

Tallies

The results of Monte Carlo are an average of the contributions from many histories calculated during running the problem (Knuth 1981; Shreider 1966; Fitzgerald et al. 1967). The expected mean physical result is called a tally. A tally can be defined as: surface current, surface flux, track length of cell flux, flux at a point, track length of energy deposition, or track estimate of fission energy deposition. The tally, an important quantity equal in stature to the Monte Carlo answer, itself has statistical error or uncertainty associated with its results; see ref. 12 for more details. This error not only gives the user the opportunity to gain an insight into the quality of the results, but also provides the basis for determining if a tally is well behaved based on examination of error and uncertainty data.

If a tally is not well behaved, the estimated error associated with the result generally will not reflect the true confidence interval of the result sought, and thus, these answers could be completely erroneous. The general rule is to carefully examine the MCNP estimated means and relative errors, and associated figure of merit (FOM) tables in the tally fluctuation charts to ensure a well-behaved and meaningful tally.

MCNP provides six standard neutron tallies and five standard photon tallies, but these basic tallies can be easily modified by the user in almost any way desired (see Table 10). All tallies are normalized to the number of starting particles. The following six tally categories represent the basic MCNP tally types; F1 (surface current), F2 (surface flux), F4 (track length of cell flux), F5 (flux at a point), F6 (track length of energy deposition), and F7 (track length of fission energy deposition). To have many tallies of a given type, add multiples of 10 to the tally number. For example, F1, F11, F21...F971, F981 are all type F1 tallies. Particle type is specified by appending a colon

and the particle type. For example, F11:N and F86:N are neutron tallies and F2:P and F25:P are photon tallies. F6 tallies may be for both neutrons and photons – F16:N,P.

Table 10. Six standard neutron tallies and five standard photon tallies.

Tallies Mnemonic	Description
F1 : N or F1 : P	Surface current
F2 : N or F2 : P	Surface flux
F4 : N or F4 : P	Track length estimate of cell flux ^a
F5a : N or F5a : P	Flux at a point (point detector)
F6 : N or F6 : N,P or F6 : P	Track length estimate of energy deposition
F7	Track length estimate of fission energy deposition

^a This is the tally we are using in this project.

APPENDIX V

The MCNP Input files and FIT Output printout and Tallies

The MCNP requires mathematical descriptions of cells, sources, materials, tallies, and miscellaneous sections of sources and regions of interest. These INPUT MCNP files were used in this project to study the possible effect of the source dimension on the output results. The two different sets of input files defined here are called IS (for point source) and FIT (for the real non-point source that will be used). The results showed no significant differences in the output tallies, which in turn demonstrates the plausibility of the point source approach to this problem. This simplifies the structure for the input files.

Another set of input files, TANK, demonstrate the effects of cask material, the surrounding water tank, and the concrete walls on the output results.

I. INPUT MCNP

There are several ways of writing an MCNP input file, depending on the nature of a program, its size, need for accuracy and precision, and time constraints. The following is a review of the structure of the medium size input file such as IS (see part L3 for IS input file) which was written for this work.

1. The IS Input File Structure

1.1. Cell Description

The first part of the IS input file is the cell description, where each line defines the boundaries of each cell (a cell is a volume of interest that is defined by its boundary surfaces) in conjunction with the neighboring cells, including the boundary of the cask with the outside world (see Column 1, lines 4-40).

For example line 4 shows the material code 2 (which is defined in materials section) followed by density code of -.961 (- symbol is used for g/cc units) and the intersections of the boundary surfaces 90 and 19 for the polyethylene cell material filling the tank.

1.2. Cell Boundaries

Cell boundary surfaces are defined in part two by their code numbers and the following surface mnemonic (see column 1, lines 1-94): each code line is assigned for only one surface. For example the line "19 cx 10.15" means: surface 19 which is a cylindrical surface (cx code) with radius 10.15 cm.

1.3. Source Definition

Source definition (Sdef) is used to specify the radiant particle characteristics at the start of the problem. Each source particle is defined by the cell location, position in the cell, energy of the particle, axis of the cell, distance from axis, extension within the cell, and its direction of motion (see column 1, the lines following "Sdef" up to "Phys:n" line). For example line " Sdef par =1 erg d1 cel 30 pos 15.0 -3.4 0 axs 1 0 0 rad d3 ext d2 vec 100" means: Sdef (source definition) is defined by par = 1 (particle of kind 1, i.e., neutron), erg d1 (energy of neutrons which is defined by the following si1 and sp1), cel (cell number 30), pos 15.0 -3.4 0 (position of the cell center with (x,y,z) coordinates (15, -3.4, 0)), axs 1 0 0 (axis of cylinder with (x,y,z) coordinates (1, 0, 0)), rad d3 (radius of cylinder is defined by si3 and sp3), ext d2 (extension of cylinder along the axis which is defined by si2, sp2, and sb2), and vec 1 0 0 (shows the cosine directors of the unit vector along the axis of the cylinder). The mnemonics si, pi, and sbi stand respectively for source information, source bias, and source probability for distribution i . The line "phys:n 20 1.8e-8" means: those neutron particles are considered in this run that have maximum energy of 20 MeV and minimum energy of 1.8 e-8 MeV.

1.4. Material Section

The material section is used to introduce the chemical composition of the materials through which the particle is transported. These material codes are recognized by the Cross-Section directory and the library section of MCNP (see column 1, m1-m4). For example line " m1 11023.35c 1" is read by the cross-section library as : material with atomic number 11, mass number 23 (i.e., sodium), and cross-section code 35 (for the corresponding energy bins of the crossing neutrons), and letter "c" stands for the continuous cross-section.

1.5. Tallies

This part defines the format and content of the print out. For example line "f4:n 8" stands for tally 4 (neutron flux in cell 8), and "fs4 -91 -92 -93 -94 t" line instructs tally 4 to print the neutron fluxes of the divisions of cell 8 which are bounded by surfaces -91, -92, -93, -94, including the flux of the whole cell. Finally line "fm4 0.02542 1 102" is a tally multiplier card which states tally 4 should be multiplied by 0.02542, unit 1 is a flag number to indicate the multiplier rather than attenuator, and number 102 is a reaction number for microscopic radiative capture cross-section. The line "e4 2.5e-8 5.0e-7 .1 .25 .8 1.5 12i 14.5" instructs how the energy bins of tally 4 (e4) should be selected from 2.5e-8 MeV (below the thermal neutron energy) to 14.5 MeV.

1.6. Miscellaneous

This section is for the rest of the input commands such as "vol" (volume), "imp" (important for the geometry splitting and Russian roulette) and so on (USDOE 1991). For example line "imp:n 1 32 2 2 16 8 1 2 2 2m 0" and following describe: how a single

neutron should be weighted (statistical weight) while it is crossing through eleven different media, with energy cut (cut) which puts a limit on the lower energy of the particles in MCNP run. The "nps" is the number of particles coming from the source, "prtmp" is used for print dump, and finally, "print" (print) command controls the printing of various tables to the output file (see column 1, miscellaneous section) .

2. IS, FIT and TANK Input Files

Here we have created three sets of MCNP INPUTS (see the following part I.3 for programs). The IS set stands for point source, to differentiate from the FIT set which is for an encapsulated source. The output results of these two sets show almost no significant differences. To study the possible changes in the sample absorbed flux due to the changes in a specific parameter, we first let the parameter take a new value in the input file. Then MCNP RUN produces the output tally, and thereby the plot of output values vs parameter values. These results are shown in Tables 4 through 8, and correspondingly in Graphs 4 through 8. The three TANK input files also show the sensitivity of the output results to the water and concrete surrounding the cask, and the materials filling the space within the cask (see Programs).

Except for the panoramic part of the sample neutron absorption (see App.IV), everywhere else in this problem we have considered only the output tallies for a single source. Due to the symmetric nature of the problem, we can multiply the tally result by a factor of 4 to calculate the total absorbed flux in the sample from four equidistant sources.

3. Programs

Since the output results of IS and FIT showed no significant differences, we then use the IS program as an input file. It has a simpler structure and therefore saves a considerable amount of the computer time, as long as the running time (or time sharing) of the input files are our main concern too (see part V.II for more details of the FIT output results and tallies). Then the set of input files TANKA, TANKP, and TANKW follow, in order to demonstrate the effects of cask material, and the surrounding water tank and concrete walls on the output results (see the following pages for input files).

MCNP Program for Cf-252 Irradiator IS INPUT RUN

Cf-252 Irradiator and Na-23 Sample (ISO, starts from center to center)

c It includes the AXIAL and AZIMUTHAL programs
4 2 -.961 -90 19 S polyethylene within tank & tubes
8 1 -.971 14 -15 -13 S sodium sample
9 3 -2.7 -7 -1 2 #30 S aluminum encapsulate of volume source
10 3 -2.7 -18 1 -90 S aluminum tube for source replacement
11 2 -.961 6 -8 -16 #8 S sodium sample encapsulate
27 3 -2.7 -17 16 -90 S aluminum tube for sample replacement
30 4 -4.18 -5 -4 3 S volume source(AL, O, Cf)
35 3 -2.7 -90 -19 20 S main cylinder to separate tank & irradiation zone
37 0 -20 -90 17 18
40 0 (90 : -1 : -16)#9 #30 #11 #8 S zero-importance outside world

1 c/x -3.4 0 .8
2 px 13
3 px 14.68
4 c/x -3.4 0 .32
5 px 15.32
6 px 10
7 px 17
8 px 20
13 cx .8
14 px 12.5
15 px 17.5
16 cx 2.2
17 cx 2.4
18 c/x -3.4 0 1
19 cx 10.15
20 cx 10
90 sx 15 100
91 px 13.5
92 px 14.5
93 px 15.5
94 px 16.5

```

sdef par=1 erg d1 cel 30 pos 15.0 -3.4 0 axs 1 0 0 rad d3 ext d2
vec 1 0 0
si1 2.5e-8 5.0e-7 .1 .25 .8 1 12i 14 16 20
sp1 -3 1.025 2.926 Sfor cf-252
sc2 the symmetry makes it a fair game.
si2 -.32 4i .32
sp2 0 1 4r
sb2 0 1 3i 5
si3 0 .32
sp3 -21 1
phys:n 20 1.8e-8
c materials
m1 11023.35c 1
m2 1001.35c .666666 6012.10c .333333
m3 13027.04c 1
m4 98252.35c .058 8016.35c .44548 13027.04c .4965
drxs
c tally
f4:n 8
fs4 -91 -92 -93 -94 t
fm4 0.02542 1 102
e4 2.5e-8 5.0e-7 .1 .25 .8 1.5 12i 14.5
f14:n 8
fs14 -91 -92 -93 -94 t
fm14 0.02542 1 102
e14 5.0e-7
c miscellaneous
vol 3j 112.858164516 4j 376.37910175 j
imp:n 1 32 2 2 16 8 1 2 2m 0
cut:n j 1.8e-8 .18 .009
nps 100000
prdmp 2j -1
print

```

MCNP Program for Cf-252 Irradiator FIT INPUT RUN

Inp101.a – Cf-252 and Na-23(center to center)

```

4  2 -.961 -90 19   S polyethylene within tank & tubes
8  1 -.971 14 -15 -13 S sodium sample
9  3 -2.7 -7 -1 2 #30 S aluminum encapsulate of volume source
10 3 -2.7 -18 1 -90 S aluminum tube for source replacement
11 2 -.961 6 -8 -16 #8 S sodium sample encapsulate
27 3 -2.7 -17 16 -90 S aluminum tube for sample replacement
30 4 -4.18 -5 -4 3   S volume source(AL,O, Cf)
35 3 -2.7 -90 -19 20 S main cylinder to seperate tank & irradi. zone
37 0 -20 -90 17 18
40 0 (90 : -1 : -16)#9 #30 #11 #8 S zero-importance outside world

1  c/x -3.4 0 .8
2  px 13
3  px 14.68
4  c/x -3.4 0 .32
5  px 15.32
6  px 10
7  px 17
8  px 20
13 cx .8
14 px 12.5
15 px 17.5
16 cx 2.2
17 cx 2.4
18 c/x -3.4 0 1
19 cx 10.15
20 cx 10
90 sx 15 100
91 px 13.5
92 px 14.5
93 px 15.5
94 px 16.5

```



```

sdef par=1 erg d1 cel 30 pos 15.0 -3.4 0 axs 1 0 0 rad d3 ext d2
vec 1 0 0 dir d5
si1 2.5e-8 5.0e-7 .1 .25 .8 1 12i 14 16 20
sp1 -3 1.025 2.926 Sfor cf-252
sc2 the symmetry makes it a fair game.
si2 -.32 4i .32
sp2 0 1 4r
sb2 0 1 3i 5
si3 0 .32
sp3 -21 1
si5 -1 -.99 -.9 4i -.4 i -.36 36i .38 .4 4i .9 .99 1
sp5 0 10 53r
sb5 0 1 3.25 51r 1
phys:n 20 1.8e-8
c materials
m1 11023.35c 1
m2 1001.35c .666666 6012.10c .333333
m3 13027.04c 1
m4 98252.35c .058 8016.35c .44548 13027.04c .4965
drxs
c tally
f4:n 8
fs4 -91 -92 -93 -94 t
fm4 0.02542 1 102
e4 2.5e-8 5.0e-7 .1 .25 .8 1.5 12i 14.5
f14:n 8
fs14 -91 -92 -93 -94 t
fm14 0.02542 1 102
e14 5.0e-7
c miscellaneous
vol 3j 112.858164516 4j 376.37910175 j
imp:n 1 32 2 2 16 8 1 2 2m 0
cut:n j 1.8e-8 .18 .009
nps 100000
prdmp 2j -1
print

```

MCNP Program for Cf-252 Irradiator TANKA INPUT RUN

Inp101.a -- Cf-252 and Na-23(center to center)

c It includes the "axial" and "azimuthal" programs

4 2 -.961 -90 19 S polyethylene within tank & tubes

8 1 -.971 14 -15 -13 S sodium sample

9 3 -2.7 -7 -1 2 #30 S aluminum encapsulate of volume source

10 3 -2.7 -18 1 -90 S aluminum tube for source replacement

11 2 -.961 6 -8 -16 #8 S sodium sample encapsulate

27 3 -2.7 -17 16 -90 S aluminum tube for sample replacement

30 4 -4.18 -5 -4 3 S volume source(AL,O, Cf)

35 3 -2.7 -90 -19 20 S main cylinder to separate tank & irradiation zone

37 0 -20 -90 17 18 S main cyl. empty

40 5 -1 (101 -102 103 -104 109 -110)(90 : -1 -2: -1 7: -16 -6: -16 8) S

45 5 -1 (111 -112 113 -114 115 -116)(-101 102 -103 104 -109 110) S concrete ???

46 0 -111 112 -113 114 -115 116 S outside world

1 c/x -3.4 0 .8

2 px 13

3 px 14.68

4 c/x -3.4 0 .32

5 px 15.32

6 px 10

7 px 17

8 px 20

13 cx .8

14 px 12.5

15 px 17.5

16 cx 2.2

17 cx 2.4

18 c/x -3.4 0 1

19 cx 10.15

20 cx 10

90 sx 15 100

91 px 13.5

92 px 14.5

93 px 15.5

94 px 16.5

101 py -150

102 py 150

103 pz -150

104 pz 150

```

109 px -165
110 px 165
111 py -210
112 py 210
113 pz -210
114 pz 210
115 pz -225
116 pz 225

sdef par=i erg d1 cei 30 pos 15.0 -3.4 0 axs 1 0 0 rad d3 ext d2
    vec 1 0 0 dir d5
si1 2.5e-8 5.0e-7 .1 .25 .8 1 12i 14 16 20
sp1 -3 1.025 2.926 Sfor cf-252
sc2 the symmetry makes it a fair game.
si2 -.32 4i .32
sp2 0 1 4r
sb2 0 1 3i 5
si3 0 .32
sp3 -21 1
si5 -1 -.99 -.9 4i -.4 i -.36 36i .38 .4 4i .9 .99 1
sp5 0 10 53r
sb5 0 1 3.25 51r 1
phys:n 20 1.8e-8
c materials
m1 11023.35c 1
m2 1001.35c .666666 6012.10c .333333
m3 13027.04c 1
m4 98252.35c .058 8016.35c .44548 13027.04c .4965
m5 1001.35c .666666 8016.30c .333333
drxs
c tally
f4:n 8
fs4 -91 -92 -93 -94 t
fm4 0.02542 1 102
e4 2.5e-8 5.0e-7 .1 .25 .8 1.5 12i 14.5
fl4:n 8
fs14 -91 -92 -93 -94 t
fm14 0.02542 1 102
e14 5.0e-7
c miscellaneous
vol 3j 112.858164516 4j 376.37910175 j
imp:n 1 32 2 2 16 8 1 2 2m 4 4 0
cut:n j 1.8e-8 .18 .009
nps 30000
prdmp 2j -1
print

```

MCNP Program for Cf-252 Irradiator TANKP INPUT RUN

Inp101.a – Cf-252 and Na-23(center to center)

c It includes the "axial" and "azimuthal" programs

4 2 -.961 -90 19 S polyethylene within tank & tubes

8 1 -.971 14 -15 -13 S sodium sample

9 3 -2.7 -7 -1 2 #30 S aluminum encapsulate of volume source

10 3 -2.7 -18 1 -90 S aluminum tube for source replacement

11 2 -.961 6 -8 -16 #8 S sodium sample encapsulate

27 3 -2.7 -17 16 -90 S aluminum tube for sample replacement

30 4 -4.18 -5 -4 3 S volume source(AL,O, Cf)

35 3 -2.7 -90 -19 20 S main cylinder to seperate tank & irradi. zone

37 5 -.961 -20 -90 17 18 S main cyl. filled with polyethylene

40 5 -1 (101 -102 103 -104 109 -110)(90 : -1 -2: -1 7: -16 -6: -16 8) S

45 5 -1 (111 -112 113 -114 115 -116)(-101 102 -103 104 -109 110) S concrete ???

46 0 -111 112 -113 114 -115 116 S outside world

1 c/x -3.4 0 .8

2 px 13

3 px 14.68

4 c/x -3.4 0 .32

5 px 15.32

6 px 10

7 px 17

8 px 20

13 cx .8

14 px 12.5

15 px 17.5

16 cx 2.2

17 cx 2.4

18 c/x -3.4 0 1

19 cx 10.15

20 cx 10

90 sx 15 100

91 px 13.5

92 px 14.5

93 px 15.5

94 px 16.5

101 py -150

102 py 150

103 pz -150

104 pz 150

109 px -165

110 px 165

```

111 py -210
112 py 210
113 pz -210
114 pz 210
115 pz -225
116 pz 225

sdef par=1 erg d1 cel 30 pos 15.0 -3.4 0 axs 1 0 0 rad d3 ext d2
    vec 1 0 0 dir d5
si1 2.5e-8 5.0e-7 .1 .25 .8 1 12i 14 16 20
sp1 -3 1.025 2.926 Sfor cf-252
sc2 the symmetry makes it a fair game.
si2 -.32 4i .32
sp2 0 1 4r
sb2 0 1 3i 5
si3 0 .32
sp3 -21 1
si5 -1 -.99 -.9 4i -.4 i -.36 36i .38 .4 4i .9 .99 1
sp5 0 10 53r
sb5 0 1 3.25 51r 1
phys:n 20 1.8e-8
c materials
m1 11023.35c 1
m2 1001.35c .666666 6012.10c .333333
m3 13027.04c 1
m4 98252.35c .058 8016.35c .44548 13027.04c .4965
m5 1001.35c .666666 8016.30c .333333
drxs
c tally
f4:n 8
fs4 -91 -92 -93 -94 t
fm4 0.02542 1 102
e4 2.5e-8 5.0e-7 .1 .25 .8 1.5 12i 14.5
f14:n 8
fs14 -91 -92 -93 -94 t
fm14 0.02542 1 102
e14 5.0e-7
c miscellaneous
vol 3j 112.858164516 4j 376.37910175 j
imp:n 1 32 2 2 16 8 1 2 2m 4 4 0
cut:n j 1.8e-8 .18 .009
nps 25000
prdump 2j -1
print

```

MCNP Program for Cf-252 Irradiator TANKW INPUT RUN

Inp101.a – Cf-252 and Na-23(center to center)

c It includes the "axial" and "azimuthal" programs

4 2 -.961 -90 19 S polyethylene within tank & tubes

8 1 -.971 14 -15 -13 S sodium sample

9 3 -2.7 -7 -1 2 #30 S aluminum encapsulate of volume source

10 3 -2.7 -18 1 -90 S aluminum tube for source replacement

11 2 -.961 6 -8 -16 #8 S sodium sample encapsulate

27 3 -2.7 -17 16 -90 S aluminum tube for sample replacement

30 4 -4.18 -5 -4 3 S volume source(AL,O, Cf)

35 3 -2.7 -90 -19 20 S main cylinder to separate tank & irradiation zone

37 5 -1 -20 -90 17 18 S main cyl. filled with stilwater

40 5 -1 (101 -102 103 -104 109 -110)(90 : -1 -2: -1 7: -16 -6: -16 8) S

45 5 -1 (111 -112 113 -114 115 -116)(-101 102 -103 104 -109 110) S concrete ???

46 0 -111 112 -113 114 -115 116 S outside world

1 c/x -3.4 0 .8

2 px 13

3 px 14.68

4 c/x -3.4 0 .32

5 px 15.32

6 px 10

7 px 17

8 px 20

13 cx .8

14 px 12.5

15 px 17.5

16 cx 2.2

17 cx 2.4

18 c/x -3.4 0 1

19 cx 10.15

20 cx 10

90 sx 15 100

91 px 13.5

92 px 14.5

93 px 15.5

94 px 16.5

101 py -150

102 py 150

103 pz -150

104 pz 150

109 px -165

110 px 165

```

111 py -210
112 py 210
113 pz -210
114 pz 210
115 pz -225
116 pz 225

sdef par=1 erg d1 cel 30 pos 15.0 -3.4 0 axs 1 0 0 rad d3 ext d2
    vec 1 0 0 dir d5
si1 2.5e-8 5.0e-7 .1 .25 .8 1 12i 14 16 20
sp1 -3 1.025 2.926 $for cf-252
sc2 the symmetry makes it a fair game.
si2 -.32 4i .32
sp2 0 1 4r
sb2 0 1 3i 5
si3 0 .32
sp3 -21 1
si5 -1 -.99 -.9 4i -.4 i -.36 36i .38 .4 4i .9 .99 1
sp5 0 10 53r
sb5 0 1 3.25 51r 1
phys:n 20 1.8e-8
c materials
m1 11023.35c 1
m2 1001.35c .666666 6012.10c .333333
m3 13027.04c 1
m4 98252.35c .058 8016.35c .44548 13027.04c .4965
m5 1001.35c .666666 8016.30c .333333
drxs
c tally
f4:n 8
fs4 -91 -92 -93 -94 t
fm4 0.02542 1 102
e4 2.5e-8 5.0e-7 .1 .25 .8 1.5 12i 14.5
f14:n 8
fs14 -91 -92 -93 -94 t
fm14 0.02542 1 102
e14 5.0e-7
c miscellaneous
vol 3j 112.858164516 4j 376.37910175 j
imp:n 1 32 2 2 16 8 1 2 2m 4 4 0
cut:n j 1.8e-8 .18 .009
nps 50000
prdmp 2j -1
print

```

II. The FIT Output Printout and Tallies

1mcnp version 4.2 ld=05/01/94 I 05/01/94 14:56:53

* probid = I 05/01/94 14:56:53

inp=fit1

```

1-  Inp101.a - Cf-252 and Na-23(center to center)
2-  4  2 -.961 -90 19  $ polyethylene within tank & tubes
3-  8  1 -.971 14 -15 -13 $ sodium sample
4-  9  3 -2.7 -7 -1 2 #30 $ aluminum encapsulate of volume source
5-  10 3 -2.7 -18 1 -90 $ aluminum tube for source replacement
6-  11 2 -.961 6 -8 -16 #8 $ sodium sample encapsulate
7-  27 3 -2.7 -17 16 -90 $ aluminum tube for sample replacement
8-  30 4 -4.18 -5 -4 3  $ volume source(AL,O, Cf)
9-  35 3 -2.7 -90 -19 20 $ main cylinder to seperate tank & irradi. zone
10- 37 0 -20 -90 17 18
11- 40 0 (90 : -1 : -16)#9 #30 #11 #8 $ zero-importance outside world
12-
13- 1  c/x -3.4 0 .8
14- 2  px 13
15- 3  px 14.68
16- 4  c/x -3.4 0 .32
17- 5  px 15.32
18- 6  px 10
19- 7  px 17
20- 8  px 20
21- 13 cx .8
22- 14 px 12.5
23- 15 px 17.5
24- 16 cx 2.2
25- 17 cx 2.4
26- 18 c/x -3.4 0 1
27- 19 cx 10.15
28- 20 cx 10
29- 90 sx 15 100
30- 91 px 13.5
31- 92 px 14.5
32- 93 px 15.5
33- 94 px 16.5
34-

```



```

35-   sdef par=1 erg d1 cel 30 pos 15.0 -3.4 0 axs 1 0 0 rad d3 ext d2
36-       vec 1 0 0 dir d5
37-   si1 2.5e-8 5.0e-7 .1 .25 .8 1 12i 14 16 20
38-   sp1 -3 1.025 2.926 $for Cf-252
39-   sc2 the symmetry makes it a fair game.
40-   si2 -.32 4i .32
41-   sp2 0 1 4r
42-   sb2 0 1 3i 5
43-   si3 0 .32
44-   sp3 -21 1
45-   si5 -1 -.99 -.9 4i -.4 i -.36 36i .38 .4 4i .9 .99 1
46-   sp5 0 10 53r
47-   sb5 0 1 3.25 51r 1
48-   phys:n 20 1.8e-8
49-   c materials
50-   m1 11023.35c 1
51-   m2 1001.35c .666666 6012.10c .333333
52-   m3 13027.04c 1
53-   m4 98252.35c .058 8016.35c .44548 13027.04c .4965
54-   drxs
55-   c tally
56-   f4:n 8
57-   fs4 -91 -92 -93 -94 t
58-   fm4 0.02542 1 102
59-   e4 2.5e-8 5.0e-7 .1 .25 .8 1.5 12i 14.5
60-   fl4:n 8
61-   fs14 -91 -92 -93 -94 t
62-   fm14 0.02542 1 102
63-   e14 5.0e-7
64-   c miscellaneous
65-   vol 3j 112.858164516 4j 376.37910175 j
66-   imp:n 1 32 2 2 16 8 1 2 2m 0
67-   cut:n j 1.8e-8 .18 .009
68-   nps 100000
69-   prdmp 2j -1
70-   print

```

1source

print table 10

values of defaulted or explicitly defined source variables

```

cel      3.0000E+01
sur      0.0000E+00
tme      0.0000E+00
pos      1.5000E+01 -3.4000E+00 0.0000E+00
x        0.0000E+00

```

```

y      0.0000E+00
z      0.0000E+00
axs    1.0000E+00  0.0000E+00  0.0000E+00
vec    1.0000E+00  0.0000E+00  0.0000E+00
ccc    0.0000E+00
nrm    1.0000E+00
ara    0.0000E+00
wgt    1.0000E+00
eff    1.0000E-02
par    1.0000E+00

```

probability distribution 1 for source variable erg
energy function 3: watt (fission) spectrum (endf law 10)

$f(e)=c*\exp(-e/a)*\sinh(\sqrt{b}*e)$
a = 1.0250E+00 b = 2.9260E+00 c = 3.0033E-01

equiprobable bin distribution used to model source distribution 1

```

1 2.5000E-08 9.0559E-08 1.3671E-07 1.7605E-07 2.1137E-07 2.4394E-07
2.7447E-07 3.0337E-07 3.3097E-07
3.5745E-07 3.8299E-07 4.0770E-07 4.3169E-07 4.5502E-07 4.7778E-07
5.0000E-07

```

```

2 5.0000E-07 1.6154E-02 2.5718E-02 3.3757E-02 4.0951E-02 4.7582E-02
5.3798E-02 5.9690E-02 6.5319E-02
7.0729E-02 7.5954E-02 8.1018E-02 8.5939E-02 9.0735E-02 9.5418E-02
1.0000E-01

```

```

3 1.0000E-01 1.1228E-01 1.2398E-01 1.3522E-01 1.4607E-01 1.5659E-01
1.6681E-01 1.7678E-01 1.8653E-01
1.9607E-01 2.0543E-01 2.1462E-01 2.2367E-01 2.3257E-01 2.4134E-01
2.5000E-01

```

```

4 2.5000E-01 2.9366E-01 3.3515E-01 3.7501E-01 4.1362E-01 4.5123E-01
4.8804E-01 5.2418E-01 5.5978E-01
5.9492E-01 6.2970E-01 6.6417E-01 6.9838E-01 7.3240E-01 7.6626E-01
8.0000E-01

```

```

5 8.0000E-01 8.1336E-01 8.2671E-01 8.4005E-01 8.5338E-01 8.6670E-01
8.8002E-01 8.9334E-01 9.0666E-01

```

9.1998E-01 9.3330E-01 9.4663E-01 9.5996E-01 9.7330E-01 9.8664E-01
1.0000E+00

6 1.0000E+00 1.0608E+00 1.1219E+00 1.1836E+00 1.2457E+00 1.3086E+00
1.3723E+00 1.4368E+00 1.5024E+00
1.5691E+00 1.6370E+00 1.7063E+00 1.7771E+00 1.8496E+00 1.9238E+00
2.0000E+00

7 2.0000E+00 2.0554E+00 2.1119E+00 2.1696E+00 2.2287E+00 2.2892E+00
2.3512E+00 2.4148E+00 2.4802E+00
2.5474E+00 2.6167E+00 2.6882E+00 2.7620E+00 2.8384E+00 2.9177E+00
3.0000E+00

8 3.0000E+00 3.0531E+00 3.1075E+00 3.1634E+00 3.2210E+00 3.2802E+00
3.3412E+00 3.4042E+00 3.4692E+00
3.5366E+00 3.6063E+00 3.6787E+00 3.7540E+00 3.8324E+00 3.9143E+00
4.0000E+00

9 4.0000E+00 4.0517E+00 4.1049E+00 4.1598E+00 4.2163E+00 4.2747E+00
4.3351E+00 4.3976E+00 4.4624E+00
4.5297E+00 4.5997E+00 4.6726E+00 4.7488E+00 4.8285E+00 4.9121E+00
5.0000E+00

10 5.0000E+00 5.0508E+00 5.1032E+00 5.1573E+00 5.2131E+00 5.2709E+00
5.3309E+00 5.3930E+00 5.4576E+00
5.5249E+00 5.5950E+00 5.6683E+00 5.7450E+00 5.8256E+00 5.9104E+00
6.0000E+00

11 6.0000E+00 6.0501E+00 6.1019E+00 6.1554E+00 6.2108E+00 6.2681E+00
6.3277E+00 6.3896E+00 6.4540E+00
6.5212E+00 6.5914E+00 6.6650E+00 6.7422E+00 6.8234E+00 6.9092E+00
7.0000E+00

12 7.0000E+00 7.0496E+00 7.1009E+00 7.1540E+00 7.2089E+00 7.2660E+00
7.3252E+00 7.3869E+00 7.4512E+00
7.5183E+00 7.5886E+00 7.6623E+00 7.7399E+00 7.8216E+00 7.9082E+00
8.0000E+00

13 8.0000E+00 8.0492E+00 8.1001E+00 8.1528E+00 8.2074E+00 8.2642E+00
8.3232E+00 8.3847E+00 8.4489E+00
8.5160E+00 8.5863E+00 8.6602E+00 8.7380E+00 8.8202E+00 8.9073E+00
9.0000E+00

14 9.0000E+00 9.0489E+00 9.0994E+00 9.1518E+00 9.2062E+00 9.2627E+00
9.3215E+00 9.3829E+00 9.4469E+00

9.5140E+00 9.5843E+00 9.6583E+00 9.7364E+00 9.8189E+00 9.9066E+00
1.0000E+01

15 1.0000E+01 1.0049E+01 1.0099E+01 1.0151E+01 1.0205E+01 1.0261E+01
1.0320E+01 1.0381E+01 1.0445E+01
1.0512E+01 1.0583E+01 1.0657E+01 1.0735E+01 1.0818E+01 1.0906E+01
1.1000E+01

16 1.1000E+01 1.1048E+01 1.1098E+01 1.1150E+01 1.1204E+01 1.1260E+01
1.1319E+01 1.1380E+01 1.1444E+01
1.1511E+01 1.1581E+01 1.1655E+01 1.1734E+01 1.1817E+01 1.1905E+01
1.2000E+01

17 1.2000E+01 1.2048E+01 1.2098E+01 1.2150E+01 1.2203E+01 1.2259E+01
1.2318E+01 1.2379E+01 1.2443E+01
1.2510E+01 1.2580E+01 1.2654E+01 1.2733E+01 1.2816E+01 1.2905E+01
1.3000E+01

18 1.3000E+01 1.3048E+01 1.3098E+01 1.3149E+01 1.3203E+01 1.3259E+01
1.3317E+01 1.3378E+01 1.3442E+01
1.3508E+01 1.3579E+01 1.3653E+01 1.3732E+01 1.3815E+01 1.3905E+01
1.4000E+01

19 1.4000E+01 1.4071E+01 1.4146E+01 1.4225E+01 1.4309E+01 1.4399E+01
1.4495E+01 1.4599E+01 1.4712E+01
1.4835E+01 1.4971E+01 1.5122E+01 1.5293E+01 1.5489E+01 1.5720E+01
1.6000E+01

20 1.6000E+01 1.6086E+01 1.6177E+01 1.6275E+01 1.6382E+01 1.6498E+01
1.6625E+01 1.6765E+01 1.6923E+01
1.7103E+01 1.7311E+01 1.7560E+01 1.7867E+01 1.8272E+01 1.8865E+01
2.0000E+01

fraction of distribution below source range = 1.35377E-12

fraction of distribution above source range = 1.34045E-06

source weight multiplied by 9.99999E-01 = fraction of distribution within range.

total distribution integral = 0.99999999

	cumulative weight	
probability	bias	probability multiplier

2.5000E-08	0.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00
5.0000E-07	1.1974E-10	0.0000E+00	1.1974E-10	1.0000E+00

1.0000E-01	1.0517E-02	0.0000E+00	1.0517E-02	1.0000E+00
2.5000E-01	2.9246E-02	0.0000E+00	3.9762E-02	1.0000E+00
8.0000E-01	1.5305E-01	0.0000E+00	1.9281E-01	1.0000E+00
1.0000E+00	6.0714E-02	0.0000E+00	2.5352E-01	1.0000E+00
2.0000E+00	2.7563E-01	0.0000E+00	5.2915E-01	1.0000E+00
3.0000E+00	1.9552E-01	0.0000E+00	7.2467E-01	1.0000E+00
4.0000E+00	1.2198E-01	0.0000E+00	8.4665E-01	1.0000E+00
5.0000E+00	7.0887E-02	0.0000E+00	9.1754E-01	1.0000E+00
6.0000E+00	3.9301E-02	0.0000E+00	9.5684E-01	1.0000E+00
7.0000E+00	2.1054E-02	0.0000E+00	9.7789E-01	1.0000E+00
8.0000E+00	1.0985E-02	0.0000E+00	9.8888E-01	1.0000E+00
9.0000E+00	5.6107E-03	0.0000E+00	9.9449E-01	1.0000E+00
1.0000E+01	2.8160E-03	0.0000E+00	9.9730E-01	1.0000E+00
1.1000E+01	1.3926E-03	0.0000E+00	9.9870E-01	1.0000E+00
1.2000E+01	6.7994E-04	0.0000E+00	9.9938E-01	1.0000E+00
1.3000E+01	3.2833E-04	0.0000E+00	9.9970E-01	1.0000E+00
1.4000E+01	1.5700E-04	0.0000E+00	9.9986E-01	1.0000E+00
1.6000E+01	1.0943E-04	0.0000E+00	9.9997E-01	1.0000E+00
2.0000E+01	2.9044E-05	0.0000E+00	1.0000E+00	1.0000E+00

the mean of source distribution 1 is 2.3061E+00

probability distribution 3 for source variable rad

power law 21: $f(x)=c*abs(x)**k$ $k = 1.0000E+00$

probability distribution 2 for source variable ext

the symmetry makes it a fair game.

biased histogram distribution

source weight entry multiplier	source value	cumulative probability	biased cumulative	probability of bin	biased probability
1	-3.20000E-01	0.000000E+00	0.000000E+00	0.000000E+00	
0.000000E+00	1.000000E+00				
2	-1.92000E-01	2.000000E-01	6.666667E-02	2.000000E-01	
6.666667E-02	3.000000E+00				
3	-6.40000E-02	4.000000E-01	2.000000E-01	2.000000E-01	
1.333333E-01	1.500000E+00				

4	6.40000E-02	6.000000E-01	4.000000E-01	2.000000E-01
	2.000000E-01	1.000000E+00		
5	1.92000E-01	8.000000E-01	6.666667E-01	2.000000E-01
	2.666667E-01	7.500000E-01		
6	3.20000E-01	1.000000E+00	1.000000E+00	2.000000E-01
	3.333333E-01	6.000000E-01		

the mean of source distribution 2 is 1.4535E-18

probability distribution 5 for source variable dir
biased histogram distribution

source weight entry multiplier	source value	cumulative probability	biased cumulative	probability of bin	biased probability
1	-1.00000E+00	0.000000E+00	0.000000E+00	0.000000E+00	
	0.000000E+00	1.000000E+00			
2	-9.90000E-01	1.851852E-02	5.847953E-03	1.851852E-02	
	5.847953E-03	3.166667E+00			
3	-9.00000E-01	3.703704E-02	2.485380E-02	1.851852E-02	
	1.900585E-02	9.743590E-01			
4	-8.00000E-01	5.555556E-02	4.385965E-02	1.851852E-02	
	1.900585E-02	9.743590E-01			
5	-7.00000E-01	7.407407E-02	6.286550E-02	1.851852E-02	
	1.900585E-02	9.743590E-01			
6	-6.00000E-01	9.259259E-02	8.187135E-02	1.851852E-02	
	1.900585E-02	9.743590E-01			
7	-5.00000E-01	1.111111E-01	1.008772E-01	1.851852E-02	
	1.900585E-02	9.743590E-01			
8	-4.00000E-01	1.296296E-01	1.198830E-01	1.851852E-02	
	1.900585E-02	9.743590E-01			
9	-3.80000E-01	1.481481E-01	1.388889E-01	1.851852E-02	
	1.900585E-02	9.743590E-01			
10	-3.60000E-01	1.666667E-01	1.578947E-01	1.851852E-02	
	1.900585E-02	9.743590E-01			
11	-3.40000E-01	1.851852E-01	1.769006E-01	1.851852E-02	
	1.900585E-02	9.743590E-01			
12	-3.20000E-01	2.037037E-01	1.959064E-01	1.851852E-02	
	1.900585E-02	9.743590E-01			
13	-3.00000E-01	2.222222E-01	2.149123E-01	1.851852E-02	
	1.900585E-02	9.743590E-01			

14	-2.80000E-01	2.407407E-01	2.339181E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
15	-2.60000E-01	2.592593E-01	2.529240E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
16	-2.40000E-01	2.777778E-01	2.719298E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
17	-2.20000E-01	2.962963E-01	2.909357E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
18	-2.00000E-01	3.148148E-01	3.099415E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
19	-1.80000E-01	3.333333E-01	3.289474E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
20	-1.60000E-01	3.518519E-01	3.479532E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
21	-1.40000E-01	3.703704E-01	3.669591E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
22	-1.20000E-01	3.888889E-01	3.859649E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
23	-1.00000E-01	4.074074E-01	4.049708E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
24	-8.00000E-02	4.259259E-01	4.239766E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
25	-6.00000E-02	4.444444E-01	4.429825E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
26	-4.00000E-02	4.629630E-01	4.619883E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
27	-2.00000E-02	4.814815E-01	4.809942E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
28	6.24500E-17	5.000000E-01	5.000000E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
29	2.00000E-02	5.185185E-01	5.190058E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
30	4.00000E-02	5.370370E-01	5.380117E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
31	6.00000E-02	5.555556E-01	5.570175E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
32	8.00000E-02	5.740741E-01	5.760234E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
33	1.00000E-01	5.925926E-01	5.950292E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
34	1.20000E-01	6.111111E-01	6.140351E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
35	1.40000E-01	6.296296E-01	6.330409E-01	1.851852E-02
	1.900585E-02	9.743590E-01		

36	1.60000E-01	6.481481E-01	6.520468E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
37	1.80000E-01	6.666667E-01	6.710526E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
38	2.00000E-01	6.851852E-01	6.900585E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
39	2.20000E-01	7.037037E-01	7.090643E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
40	2.40000E-01	7.222222E-01	7.280702E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
41	2.60000E-01	7.407407E-01	7.470760E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
42	2.80000E-01	7.592593E-01	7.660819E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
43	3.00000E-01	7.777778E-01	7.850877E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
44	3.20000E-01	7.962963E-01	8.040936E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
45	3.40000E-01	8.148148E-01	8.230994E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
46	3.60000E-01	8.333333E-01	8.421053E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
47	3.80000E-01	8.518519E-01	8.611111E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
48	4.00000E-01	8.703704E-01	8.801170E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
49	5.00000E-01	8.888889E-01	8.991228E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
50	6.00000E-01	9.074074E-01	9.181287E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
51	7.00000E-01	9.259259E-01	9.371345E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
52	8.00000E-01	9.444444E-01	9.561404E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
53	9.00000E-01	9.629630E-01	9.751462E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
54	9.90000E-01	9.814815E-01	9.941520E-01	1.851852E-02
	1.900585E-02	9.743590E-01		
55	1.00000E+00	1.000000E+00	1.000000E+00	1.851852E-02
	5.847953E-03	3.166667E+00		

the mean of source distribution 5 is 5.1405E-17

order of sampling source variables.

cel axs rad ext pos vec dir erg tme

1tally 4

print table 30

tally type 4 track length estimate of particle flux.
tally for neutrons

cells 8

segments -91 -92 -93 -94 rest whole

multiplier bins

att constant material reactions or material-rho*x pairs

2.54200E-02 1

102

energy bins

1.80000E-08 to 2.50000E-08 mev

2.50000E-08 to 5.00000E-07 mev

5.00000E-07 to 1.00000E-01 mev

1.00000E-01 to 2.50000E-01 mev

2.50000E-01 to 8.00000E-01 mev

8.00000E-01 to 1.50000E+00 mev

1.50000E+00 to 2.50000E+00 mev

2.50000E+00 to 3.50000E+00 mev

3.50000E+00 to 4.50000E+00 mev

4.50000E+00 to 5.50000E+00 mev

5.50000E+00 to 6.50000E+00 mev

6.50000E+00 to 7.50000E+00 mev

7.50000E+00 to 8.50000E+00 mev

8.50000E+00 to 9.50000E+00 mev

9.50000E+00 to 1.05000E+01 mev

1.05000E+01 to 1.15000E+01 mev

1.15000E+01 to 1.25000E+01 mev

1.25000E+01 to 1.35000E+01 mev

1.35000E+01 to 1.45000E+01 mev

total bin

1tally 14

print table 30

tally type 4 track length estimate of particle flux.
tally for neutrons

cells 8

segments -91 -92 -93 -94 rest whole

multiplier bins

att constant material reactions or material-rho*x pairs

2.54200E-02 1

102

energy bins

1.80000E-08 to 5.00000E-07 mev

warning. continuous-energy cross-section table used for 1001.35d

warning. continuous-energy cross-section table used for 8016.35d

warning. continuous-energy cross-section table used for 11023.35d

warning. continuous-energy cross-section table used for 98252.35d

1material composition

print table 60

the sum of the fractions of material 4 was 9.999800E-01

material

number component nuclide, atom fraction

1	11023, 1.00000		
2	1001, 0.66667	6012, 0.33333	
3	13027, 1.00000		
4	98252, 0.05800	8016, 0.44549	13027, 0.49651

material

number component nuclide, mass fraction

1	11023, 1.00000		
2	1001, 0.14381	6012, 0.85619	
3	13027, 1.00000		
4	98252, 0.41604	8016, 0.20276	13027, 0.38120

warning. 1 of the materials had unnormalized fractions.

1cell volumes and masses

print table

70

	cell	atom	gram	input	calculated		reason	volume
		density	density	volume	volume	mass	pieces	not
calculated								

1	4	1.23873E-01	9.61000E-01	0.00000E+00	4.12423E+06	3.96338E+06	1
---	---	-------------	-------------	-------------	-------------	-------------	---

```

2  8 2.54348E-02 9.71000E-01 0.00000E+00 1.00531E+01 9.76156E+00 1
3  9 6.02616E-02 2.70000E+00 0.00000E+00 7.83659E+00 2.11588E+01
1
4 10 6.02616E-02 2.70000E+00 1.12858E+02 0.00000E+00 3.04717E+02
0 asymmetric
5 11 1.23873E-01 9.61000E-01 0.00000E+00 1.42000E+02 1.36462E+02
1
6 27 6.02616E-02 2.70000E+00 0.00000E+00 5.77900E+02 1.56033E+03
1
7 30 7.16274E-02 4.18000E+00 0.00000E+00 2.05887E-01 8.60609E-01 1
8 35 6.02616E-02 2.70000E+00 0.00000E+00 1.88943E+03 5.10146E+03
1
9 37 0.00000E+00 0.00000E+00 3.76379E+02 0.00000E+00 0.00000E+00
0 asymmetric
10 40 0.00000E+00 0.00000E+00 0.00000E+00 0.00000E+00 0.00000E+00
0 asymmetric
1surface areas
print table 770

```

	surface	input	calculated	reason	area
	area	area	area	not	calculated
1	1	0.00000E+00	0.00000E+00	asymmetric	
2	2	0.00000E+00	2.01062E+00		
3	3	0.00000E+00	0.00000E+00	asymmetric	
4	4	0.00000E+00	0.00000E+00	asymmetric	
5	5	0.00000E+00	0.00000E+00	asymmetric	
6	6	0.00000E+00	1.52053E+01		
7	7	0.00000E+00	2.01062E+00		
8	8	0.00000E+00	1.52053E+01		
9	13	0.00000E+00	0.00000E+00	asymmetric	
10	14	0.00000E+00	0.00000E+00	asymmetric	
11	15	0.00000E+00	0.00000E+00	asymmetric	
12	16	0.00000E+00	0.00000E+00	asymmetric	
13	17	0.00000E+00	3.01506E+03		
14	18	0.00000E+00	0.00000E+00	asymmetric	
15	19	0.00000E+00	1.26890E+04		
16	20	0.00000E+00	1.25034E+04		
17	90	0.00000E+00	0.00000E+00	asymmetric	
18	91	0.00000E+00	0.00000E+00	not a boundary	
19	92	0.00000E+00	0.00000E+00	not a boundary	
20	93	0.00000E+00	0.00000E+00	not a boundary	
21	94	0.00000E+00	0.00000E+00	not a boundary	

1 tally segment areas, volumes, or masses
table 770

print

tally calculated	tally cell	segment surface	segment surface	input	calculated	used	reason not
---------------------	---------------	--------------------	--------------------	-------	------------	------	------------

4	8	1	-91	0.00000E+00	2.01062E+00	2.01062E+00	
		2	-92	0.00000E+00	2.01062E+00	2.01062E+00	
		3	-93	0.00000E+00	2.01062E+00	2.01062E+00	
		4	-94	0.00000E+00	2.01062E+00	2.01062E+00	
		5		0.00000E+00	2.01062E+00	2.01062E+00	

total of segment volumes					1.00531E+01		
--------------------------	--	--	--	--	-------------	--	--

volume of whole cell					1.00531E+01		
----------------------	--	--	--	--	-------------	--	--

14	8	1	-91	0.00000E+00	2.01062E+00	2.01062E+00	
		2	-92	0.00000E+00	2.01062E+00	2.01062E+00	
		3	-93	0.00000E+00	2.01062E+00	2.01062E+00	
		4	-94	0.00000E+00	2.01062E+00	2.01062E+00	
		5		0.00000E+00	2.01062E+00	2.01062E+00	

total of segment volumes					1.00531E+01		
--------------------------	--	--	--	--	-------------	--	--

volume of whole cell					1.00531E+01		
----------------------	--	--	--	--	-------------	--	--

1cells

print table 80

	atom	gram		neutron			
cell	mat	density	density	volume	mass	pieces	importance
1	4	2	1.23873E-01	9.61000E-01	4.12423E+06	3.96338E+06	1 1.00000E+00
2	8	1	2.54348E-02	9.71000E-01	1.00531E+01	9.76156E+00	1 3.20000E+01
3	9	3	6.02616E-02	2.70000E+00	7.83659E+00	2.11588E+01	1 2.00000E+00
4	10	3	6.02616E-02	2.70000E+00	1.12858E+02	3.04717E+02	0 2.00000E+00
5	11	2	1.23873E-01	9.61000E-01	1.42000E+02	1.36462E+02	1 1.60000E+01
6	27	3	6.02616E-02	2.70000E+00	5.77900E+02	1.56033E+03	1 8.00000E+00
7	30	4	7.16274E-02	4.18000E+00	2.05887E-01	8.60609E-01	1 1.00000E+00
8	35	3	6.02616E-02	2.70000E+00	1.88943E+03	5.10146E+03	1 2.00000E+00
9	37	0	0.00000E+00	0.00000E+00	3.76379E+02	0.00000E+00	0
4.00000E+00							
10	40	0	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	0
0.00000E+00							

total	4.12734E+06	3.97052E+06
-------	-------------	-------------

1surfaces

print table 90

surface	trans	type	surface coefficients
1	1	c/x	-3.4000000E+00 0.0000000E+00 8.0000000E-01
2	2	px	1.3000000E+01

```

3 3 px 1.4680000E+01
4 4 c/x -3.4000000E+00 0.0000000E+00 3.2000000E-01
5 5 px 1.5320000E+01
6 6 px 1.0000000E+01
7 7 px 1.7000000E+01
8 8 px 2.0000000E+01
9 13 cx 8.0000000E-01
10 14 px 1.2500000E+01
11 15 px 1.7500000E+01
12 16 cx 2.2000000E+00
13 17 cx 2.4000000E+00
14 18 c/x -3.4000000E+00 0.0000000E+00 1.0000000E+00
15 19 cx 1.0150000E+01
16 20 cx 1.0000000E+01
17 90 sx 1.5000000E+01 1.0000000E+02
18 91 px 1.3500000E+01
19 92 px 1.4500000E+01
20 93 px 1.5500000E+01
21 94 px 1.6500000E+01

```

1 temperatures of the cells.

print table

92

time: 0.0000E+00

```

cell tmp 1
4 2.5300E-08
8 2.5300E-08
9 2.5300E-08
10 2.5300E-08
11 2.5300E-08
27 2.5300E-08
30 2.5300E-08
35 2.5300E-08

```

minimum source weight = 5.8461E-01 maximum source weight = 9.5000E+00

5 warning messages so far.

1cross-section tables

print table 100

table length

tables from file c:\mcnp\lib\endl852

1001.35c 1565 za= 1001 endl-85 85/04/24 t=0k
87/12/07

(1)

8016.35c 4831 za=8016 endl-85 85/04/24 t=0k (16)
86/01/31

11023.35c 11932 za=11023 endl-85 85/04/24 t=0k (18)
86/01/31

warning. nubar of 98252.35c may be either prompt or total.
98252.35c 9592 za=98252 endl-85 85/04/24 t=0k (101)
86/02/01

tables from file c:\mncpl\lib\d92

6012.10d 4904 c-12 lasl sublibrary mat 102 13 apr 1976 pointwise t=0.0
(102) 15 apr 76
13027.04d 7996 al-27 endf/b-iv new gamma production format
(1193) 14 oct 75

total 40820

any neutrons with energy greater than $\text{emax} = 2.00000\text{E}+01$ from the source or
from a collision will be resampled.

neutron cross sections outside the range from $1.8000\text{E}-08$ to $2.0000\text{E}+01$ mev are
expunged.

use of dynamically allocated storage

general	16004
tallies	1560
bank	6403
cross sections	40820
total	64787

*

dump no. 1 on file runtpf nps = 0 ctm = 0.00

6 warning messages so far.

1 starting mcrun. field length = 0 cp0 = 0.13 print
table 110

Inp101.a -- Cf-252 and Na-23(center to center)

nps x y z cell surf u v w time weight
energy

1 1.527E+01 -3.671E+00 -4.477E-02 30 0 2.993E-01 8.887E-01 3.474E-01
0.000E+00 5.846E-01 3.809E+00

warning. tally score not made because it is beyond the last energy bin of tally 14

2 1.510E+01 -3.437E+00 -1.157E-02 30 0 -4.704E-01 -7.576E-01 4.526E-01
0.000E+00 7.308E-01 4.826E-01

3 1.471E+01 -3.209E+00 1.910E-01 30 0 1.885E-01 3.709E-01 -9.093E-01
0.000E+00 2.923E+00 1.409E+00

4 1.471E+01 -3.350E+00 1.916E-01 30 0 -1.486E-01 -1.011E-01 -9.837E-01
0.000E+00 2.923E+00 5.036E-01

5 1.532E+01 -3.457E+00 9.984E-02 30 0 -7.025E-01 -6.858E-01 1.900E-01
0.000E+00 5.846E-01 2.887E-01

6 1.526E+01 -3.552E+00 1.499E-01 30 0 8.054E-02 9.919E-01 -9.833E-02
0.000E+00 5.846E-01 2.516E+00

7 1.490E+01 -3.566E+00 -1.570E-01 30 0 3.702E-01 7.714E-02 -9.258E-01
0.000E+00 1.462E+00 9.692E-02

8 1.498E+01 -3.169E+00 1.111E-01 30 0 -1.946E-01 -7.339E-01 6.508E-01
0.000E+00 9.744E-01 1.409E+00

9 1.507E+01 -3.303E+00 2.679E-01 30 0 6.592E-01 -6.345E-02 -7.493E-01
0.000E+00 7.308E-01 1.449E+00

10 1.527E+01 -3.245E+00 1.195E-01 30 0 -2.142E-01 9.475E-01 -2.375E-01
0.000E+00 5.846E-01 2.138E-01

11 1.509E+01 -3.533E+00 -1.156E-01 30 0 -9.352E-01 -1.877E-01 3.004E-01
0.000E+00 7.308E-01 2.648E+00

12 1.516E+01 -3.562E+00 -1.183E-01 30 0 3.539E-02 3.457E-02 9.988E-01
0.000E+00 7.308E-01 5.381E+00

13 1.519E+01 -3.505E+00 -1.073E-01 30 0 8.621E-01 5.064E-01 -1.749E-02
0.000E+00 7.308E-01 1.020E+00

14 1.496E+01 -3.353E+00 1.800E-01 30 0 6.798E-01 2.501E-01 6.895E-01
0.000E+00 9.744E-01 6.136E-01

15 1.532E+01 -3.526E+00 2.882E-01 30 0 3.044E-01 -1.438E-01 9.416E-01
0.000E+00 5.846E-01 6.069E-01

16 1.528E+01 -3.173E+00 -2.257E-01 30 0 4.821E-02 -4.076E-01 -9.119E-01
0.000E+00 5.846E-01 1.282E+01

17 1.515E+01 -3.337E+00 3.052E-02 30 0 1.833E-01 -1.036E-01 9.776E-01
0.000E+00 7.308E-01 1.464E+00

18	1.503E+01	-3.608E+00	1.510E-01	30	0	2.260E-01	9.033E-01	-3.647E-01
	0.000E+00	9.744E-01	8.314E-01					
19	1.511E+01	-3.186E+00	4.330E-02	30	0	2.883E-01	2.171E-01	9.326E-01
	0.000E+00	7.308E-01	4.381E+00					
20	1.492E+01	-3.108E+00	2.573E-02	30	0	5.214E-02	-2.951E-01	9.540E-01
	0.000E+00	1.462E+00	5.294E-01					
21	1.512E+01	-3.667E+00	1.119E-01	30	0	-1.520E-01	-8.926E-01	-4.243E-01
	0.000E+00	7.308E-01	1.248E-01					
22	1.509E+01	-3.440E+00	-2.001E-01	30	0	-1.531E-01	7.811E-01	6.053E-01
	0.000E+00	7.308E-01	2.827E+00					
23	1.506E+01	-3.166E+00	1.483E-01	30	0	-1.256E-01	-1.802E-01	-9.756E-01
	0.000E+00	9.744E-01	5.647E+00					
24	1.489E+01	-3.582E+00	1.761E-01	30	0	8.540E-01	5.065E-01	-1.191E-01
	0.000E+00	1.462E+00	5.622E-01					
25	1.530E+01	-3.396E+00	2.719E-01	30	0	-1.109E-01	9.936E-01	2.190E-02
	0.000E+00	5.846E-01	6.037E-01					
26	1.498E+01	-3.511E+00	-1.349E-01	30	0	2.745E-02	9.947E-01	-9.866E-02
	0.000E+00	9.744E-01	5.370E+00					
27	1.519E+01	-3.630E+00	1.552E-01	30	0	9.949E-01	-1.752E-02	9.953E-02
	0.000E+00	2.375E+00	7.592E-01					
28	1.478E+01	-3.310E+00	-7.869E-02	30	0	1.090E-01	6.859E-01	-7.195E-01
	0.000E+00	2.923E+00	3.520E+00					
29	1.509E+01	-3.232E+00	3.322E-02	30	0	-1.939E-01	1.653E-01	9.670E-01
	0.000E+00	7.308E-01	3.899E+00					
30	1.498E+01	-3.430E+00	1.897E-02	30	0	-1.770E-01	-6.705E-01	-7.205E-01
	0.000E+00	9.744E-01	2.319E+00					
31	1.504E+01	-3.421E+00	2.566E-01	30	0	7.514E-01	-5.232E-01	-4.021E-01
	0.000E+00	9.744E-01	3.105E+00					
32	1.492E+01	-3.232E+00	-1.012E-01	30	0	2.907E-01	8.973E-01	3.322E-01
	0.000E+00	1.462E+00	2.973E+00					
33	1.494E+01	-3.375E+00	-2.004E-01	30	0	1.355E-01	9.874E-01	8.171E-02
	0.000E+00	1.462E+00	5.377E-01					
34	1.503E+01	-3.632E+00	1.934E-01	30	0	-5.005E-01	7.246E-01	4.738E-01
	0.000E+00	9.744E-01	1.994E+00					
35	1.519E+01	-3.427E+00	-2.861E-01	30	0	-4.686E-02	-7.542E-01	6.550E-01
	0.000E+00	7.308E-01	3.959E+00					
36	1.508E+01	-3.378E+00	9.738E-02	30	0	9.698E-01	9.584E-02	-2.245E-01
	0.000E+00	7.308E-01	2.168E+00					
37	1.529E+01	-3.385E+00	6.641E-03	30	0	9.657E-01	1.797E-01	1.872E-01
	0.000E+00	5.846E-01	3.400E+00					
38	1.494E+01	-3.185E+00	-1.118E-01	30	0	1.903E-01	1.161E-01	9.748E-01
	0.000E+00	9.744E-01	1.206E+00					
39	1.475E+01	-3.469E+00	-1.668E-01	30	0	3.205E-01	8.818E-01	-3.459E-01
	0.000E+00	2.923E+00	5.284E-01					


```

40 1.523E+01 -3.300E+00 -1.073E-01 30 0 6.314E-02 3.510E-01 9.342E-01
0.000E+00 5.846E-01 1.631E+00
41 1.502E+01 -3.236E+00 -7.946E-03 30 0 5.685E-01 -3.340E-01 7.518E-01
0.000E+00 9.744E-01 1.730E+00
42 1.486E+01 -3.351E+00 9.797E-02 30 0 6.485E-01 6.524E-01 3.922E-01
0.000E+00 1.462E+00 9.203E-01
43 1.525E+01 -3.248E+00 2.446E-01 30 0 -5.438E-02 1.911E-02 -9.983E-01
0.000E+00 5.846E-01 7.435E-01
44 1.508E+01 -3.622E+00 -1.773E-01 30 0 5.958E-01 3.155E-01 -7.386E-01
0.000E+00 7.308E-01 4.709E+00
45 1.507E+01 -3.200E+00 -2.231E-02 30 0 -2.468E-01 -7.400E-01 6.257E-01
0.000E+00 7.308E-01 1.687E+00
46 1.504E+01 -3.289E+00 -5.824E-02 30 0 6.478E-01 7.363E-01 1.952E-01
0.000E+00 9.744E-01 5.836E+00
47 1.523E+01 -3.673E+00 1.269E-01 30 0 -3.613E-01 -3.776E-01 8.526E-01
0.000E+00 5.846E-01 8.139E-01
48 1.511E+01 -3.664E+00 1.574E-01 30 0 1.686E-01 -8.637E-01 4.750E-01
0.000E+00 7.308E-01 2.300E+00
49 1.486E+01 -3.136E+00 -6.764E-02 30 0 -2.559E-01 -1.844E-01 9.490E-01
0.000E+00 1.462E+00 1.504E+00
50 1.514E+01 -3.524E+00 -4.419E-02 30 0 -9.651E-01 -2.572E-01 4.995E-02
0.000E+00 7.308E-01 8.363E-01

```

1problem summary

run terminated when 100000 particle histories were done.

I

05/01/94 16:18:36

Inp101.a - Cf-252 and Na-23(center to center)

probid = I

05/01/94 14:56:53

0

neutron creation	tracks	weight	energy	neutron loss	tracks
weight	energy				
		(per source particle)			(per source particle)

source	100000	1.0006E+00	2.3023E+00	escape	80127
1.4712E-01	1.7604E-01				

	energy cutoff	99699	8.0186E-01
8.4710E-09			

weight window	0	0.	0.	weight window	0	0.	0.
cell importance	637023	9.1675E-01	1.4640E+00	cell importance			
557934	9.2127E-01	1.4799E+00					
weight cutoff	0	0.	0.	weight cutoff	0	0.	0.
energy importance	0	0.	0.	energy importance	0	0.	0.
0.							

dxtran	0	0.	0.	dxtran	0	0.	0.
forced collisions	0	0.	0.	forced collisions	0	0.	0.
exp. transform	0	0.	0.	exp. transform	0	0.	0.
upscattering	0	0.	5.1140E-08	downscattering	0	0.	
2.1178E+00							
			capture	0	5.4767E-02	1.0825E-	
02							
(n,xn)	8	4.4405E-05	2.6730E-05	loss to (n,xn)	4	2.2202E-	
05	1.9659E-04						
fission	968	1.0094E-02	2.4413E-02	loss to fission	235		
2.4346E-03	6.0494E-03						
total	737999	1.9275E+00	3.7908E+00	total	737999		
1.9275E+00	3.7908E+00						

number of neutrons banked	511008	average lifetime, shakes	
cutoffs			
neutron tracks per source particle	7.3800E+00	escape	5.7551E+02
tco 1.0000E+34			
neutron collisions per source particle	3.4517E+01	capture	1.1343E+03
eco 1.8000E-08			
total neutron collisions	3451738	capture or escape	7.2709E+02
wc1 1.8000E-01			
net multiplication	1.0077E+00 0.0022	any termination	7.7497E+02
wc2 9.0000E-03			

computer time so far in this run	81.71 minutes	maximum number ever in	
bank 11			
computer time in mcrun	81.58 minutes	bank overflows to backup file	
0			
source particles per minute	1.2258E+03	field length	0
random numbers generated	55924966	most random numbers used	
was 7005 in history	39350		

range of sampled source weights = 5.8461E-01 to 9.5000E+00

source efficiency = 1.0000 in cell 30
 1 source distribution frequency tables
 table 170

print

the expected values below do not include the effect of the rejection loop which samples position.

source distribution 2 for ext
 the symmetry makes it a fair game.

source		number		weight		
n	value	sampld	expected	sampld/expected	sampld	expected
sampld/expected						
1	-1.92000E-01	6.6830E-02	6.6667E-02	1.0025E+00	2.0049E-01	
	2.0000E-01	1.0025E+00				
2	-6.40000E-02	1.3391E-01	1.3333E-01	1.0043E+00	2.0087E-01	
	2.0000E-01	1.0043E+00				
3	6.40000E-02	2.0021E-01	2.0000E-01	1.0010E+00	2.0021E-01	
	2.0000E-01	1.0010E+00				
4	1.92000E-01	2.6543E-01	2.6667E-01	9.9536E-01	1.9907E-01	
	2.0000E-01	9.9536E-01				
5	3.20000E-01	3.3362E-01	3.3333E-01	1.0009E+00	2.0017E-01	
	2.0000E-01	1.0009E+00				
total		1.0000E+00	1.0000E+00	1.0000E+00	1.0008E+00	
	1.0000E+00	1.0008E+00				

source distribution 5 for dir

source		number		weight		
n	value	sampld	expected	sampld/expected	sampld	expected
sampld/expected						
1	-9.90000E-01	5.7900E-03	5.8480E-03	9.9009E-01	1.8007E-02	
	1.8519E-02	9.7239E-01				
2	-9.00000E-01	1.9550E-02	1.9006E-02	1.0286E+00	1.9058E-02	
	1.8519E-02	1.0292E+00				
3	-8.00000E-01	1.8680E-02	1.9006E-02	9.8286E-01	1.7880E-02	
	1.8519E-02	9.6554E-01				
4	-7.00000E-01	1.9160E-02	1.9006E-02	1.0081E+00	1.8857E-02	
	1.8519E-02	1.0183E+00				
5	-6.00000E-01	1.8780E-02	1.9006E-02	9.8812E-01	1.8608E-02	
	1.8519E-02	1.0048E+00				
6	-5.00000E-01	1.8350E-02	1.9006E-02	9.6549E-01	1.8257E-02	
	1.8519E-02	9.8588E-01				
7	-4.00000E-01	1.9070E-02	1.9006E-02	1.0034E+00	1.8361E-02	
	1.8519E-02	9.9148E-01				
8	-3.80000E-01	1.8940E-02	1.9006E-02	9.9654E-01	1.8814E-02	
	1.8519E-02	1.0160E+00				
9	-3.60000E-01	1.8470E-02	1.9006E-02	9.7181E-01	1.8309E-02	
	1.8519E-02	9.8867E-01				

10	-3.40000E-01	1.8630E-02	1.9006E-02	9.8022E-01	1.8166E-02
	1.8519E-02	9.8099E-01			
11	-3.20000E-01	1.9120E-02	1.9006E-02	1.0060E+00	1.8600E-02
	1.8519E-02	1.0044E+00			
12	-3.00000E-01	1.9720E-02	1.9006E-02	1.0376E+00	1.9211E-02
	1.8519E-02	1.0374E+00			
13	-2.80000E-01	1.9680E-02	1.9006E-02	1.0355E+00	1.8894E-02
	1.8519E-02	1.0203E+00			
14	-2.60000E-01	1.8410E-02	1.9006E-02	9.6865E-01	1.7710E-02
	1.8519E-02	9.5634E-01			
15	-2.40000E-01	1.8770E-02	1.9006E-02	9.8759E-01	1.8383E-02
	1.8519E-02	9.9269E-01			
16	-2.20000E-01	1.9060E-02	1.9006E-02	1.0028E+00	1.8309E-02
	1.8519E-02	9.8870E-01			
17	-2.00000E-01	1.8620E-02	1.9006E-02	9.7970E-01	1.7974E-02
	1.8519E-02	9.7062E-01			
18	-1.80000E-01	1.8950E-02	1.9006E-02	9.9706E-01	1.8676E-02
	1.8519E-02	1.0085E+00			
19	-1.60000E-01	1.8930E-02	1.9006E-02	9.9601E-01	1.8146E-02
	1.8519E-02	9.7988E-01			
20	-1.40000E-01	1.8670E-02	1.9006E-02	9.8233E-01	1.8685E-02
	1.8519E-02	1.0090E+00			
21	-1.20000E-01	1.9740E-02	1.9006E-02	1.0386E+00	1.9132E-02
	1.8519E-02	1.0331E+00			
22	-1.00000E-01	1.9220E-02	1.9006E-02	1.0113E+00	1.8974E-02
	1.8519E-02	1.0246E+00			
23	-8.00000E-02	1.9610E-02	1.9006E-02	1.0318E+00	1.9010E-02
	1.8519E-02	1.0265E+00			
24	-6.00000E-02	1.8840E-02	1.9006E-02	9.9127E-01	1.8234E-02
	1.8519E-02	9.8464E-01			
25	-4.00000E-02	1.9210E-02	1.9006E-02	1.0107E+00	1.8984E-02
	1.8519E-02	1.0251E+00			
26	-2.00000E-02	1.8520E-02	1.9006E-02	9.7444E-01	1.8404E-02
	1.8519E-02	9.9380E-01			
27	6.24500E-17	1.8380E-02	1.9006E-02	9.6707E-01	1.8302E-02
	1.8519E-02	9.8830E-01			
28	2.00000E-02	1.8690E-02	1.9006E-02	9.8338E-01	1.8511E-02
	1.8519E-02	9.9959E-01			
29	4.00000E-02	1.9020E-02	1.9006E-02	1.0007E+00	1.8838E-02
	1.8519E-02	1.0173E+00			
30	6.00000E-02	1.9250E-02	1.9006E-02	1.0128E+00	1.8452E-02
	1.8519E-02	9.9643E-01			
31	8.00000E-02	1.8730E-02	1.9006E-02	9.8549E-01	1.8412E-02
	1.8519E-02	9.9425E-01			

32	1.00000E-01	1.9050E-02	1.9006E-02	1.0023E+00	1.8394E-02
	1.8519E-02	9.9330E-01			
33	1.20000E-01	1.8820E-02	1.9006E-02	9.9022E-01	1.8736E-02
	1.8519E-02	1.0117E+00			
34	1.40000E-01	1.8900E-02	1.9006E-02	9.9443E-01	1.8141E-02
	1.8519E-02	9.7962E-01			
35	1.60000E-01	2.0040E-02	1.9006E-02	1.0544E+00	1.9746E-02
	1.8519E-02	1.0663E+00			
36	1.80000E-01	1.8490E-02	1.9006E-02	9.7286E-01	1.8056E-02
	1.8519E-02	9.7504E-01			
37	2.00000E-01	1.8830E-02	1.9006E-02	9.9075E-01	1.8230E-02
	1.8519E-02	9.8441E-01			
38	2.20000E-01	1.9230E-02	1.9006E-02	1.0118E+00	1.8794E-02
	1.8519E-02	1.0149E+00			
39	2.40000E-01	1.8470E-02	1.9006E-02	9.7181E-01	1.7965E-02
	1.8519E-02	9.7012E-01			
40	2.60000E-01	1.9050E-02	1.9006E-02	1.0023E+00	1.8500E-02
	1.8519E-02	9.9901E-01			
41	2.80000E-01	1.8820E-02	1.9006E-02	9.9022E-01	1.8223E-02
	1.8519E-02	9.8404E-01			
42	3.00000E-01	1.8800E-02	1.9006E-02	9.8917E-01	1.8314E-02
	1.8519E-02	9.8893E-01			
43	3.20000E-01	1.9280E-02	1.9006E-02	1.0144E+00	1.8919E-02
	1.8519E-02	1.0216E+00			
44	3.40000E-01	1.8830E-02	1.9006E-02	9.9075E-01	1.8293E-02
	1.8519E-02	9.8783E-01			
45	3.60000E-01	1.9530E-02	1.9006E-02	1.0276E+00	1.8976E-02
	1.8519E-02	1.0247E+00			
46	3.80000E-01	1.9500E-02	1.9006E-02	1.0260E+00	1.8837E-02
	1.8519E-02	1.0172E+00			
47	4.00000E-01	1.8740E-02	1.9006E-02	9.8601E-01	1.7974E-02
	1.8519E-02	9.7062E-01			
48	5.00000E-01	1.9030E-02	1.9006E-02	1.0013E+00	1.8209E-02
	1.8519E-02	9.8328E-01			
49	6.00000E-01	1.9190E-02	1.9006E-02	1.0097E+00	1.8664E-02
	1.8519E-02	1.0078E+00			
50	7.00000E-01	1.8790E-02	1.9006E-02	9.8864E-01	1.8156E-02
	1.8519E-02	9.8041E-01			
51	8.00000E-01	1.9550E-02	1.9006E-02	1.0286E+00	1.9220E-02
	1.8519E-02	1.0379E+00			
52	9.00000E-01	1.9590E-02	1.9006E-02	1.0307E+00	1.8986E-02
	1.8519E-02	1.0253E+00			
53	9.90000E-01	1.8930E-02	1.9006E-02	9.9601E-01	1.8359E-02
	1.8519E-02	9.9138E-01			

```

54      1.00000E+00      5.9800E-03      5.8480E-03      1.0226E+00      1.8731E-02
1.8519E-02      1.0115E+00
total      1.0000E+00      1.0000E+00      1.0000E+00      1.0006E+00
1.0000E+00      1.0006E+00
1neutron activity in each cell
print table
126

```

average	cell	entering	tracks	population	collisions	collisions	number	flux	average
track	mfp				* weight	weighted	weighted	track	weight
					(per history)	energy	energy	(relative)	(cm)
1	4	121610	108471	2269261	2.1864E+01	8.7965E-04	8.9443E-01		
9.7021E-01	1.5697E+00								
2	8	233388	223934	32248	9.6520E-03	8.3590E-03	1.5774E+00		
9.6839E-01	1.3285E+01								
3	9	208099	204450	27586	1.4084E-01	1.6645E-01	2.1643E+00		
1.0309E+00	5.7954E+00								
4	10	206337	200607	9743	4.7847E-02	2.4200E-02	2.0246E+00		
9.8202E-01	6.0519E+00								
5	11	503138	440707	1064205	6.4230E-01	6.8454E-03	1.5206E+00		
9.7309E-01	2.3098E+00								
6	27	424063	349750	30078	3.6763E-02	5.5740E-03	1.6093E+00		
9.7643E-01	6.8608E+00								
7	30	101276	101417	6543	6.5909E-02	8.3199E-01	2.2593E+00		
1.0031E+00	4.8107E+00								
8	35	314169	226918	12074	5.9418E-02	1.7854E-03	1.1200E+00		
9.7357E-01	7.9783E+00								
9	37	463777	398979	0	0.0000E+00	2.6114E-03	1.3547E+00		
1.3366E+00	0.0000E+00								

```
total 2575857 2255233 3451738 2.2867E+01
1neutron weight balance in each cell - external events
print table 130
```

	cell	entering	source	energy	time	exiting	total
			cutoff	cutoff			
1	4	1.1897E+00	0.0000E+00	-7.9446E-01	0.0000E+00	-3.3889E-01	5.6343E-02
2	8	7.0901E-02	0.0000E+00	-3.3185E-06	0.0000E+00	-7.0864E-02	3.3746E-05
3	9	1.0399E+00	0.0000E+00	0.0000E+00	0.0000E+00	-1.0396E+00	3.2301E-04

```

  4 10 1.0113E+00 0.0000E+00 0.0000E+00 0.0000E+00 -1.0112E+00
7.7694E-05
  5 11 3.0582E-01 0.0000E+00 -7.3103E-03 0.0000E+00 -2.9788E-01
6.2781E-04
  6 27 5.1645E-01 0.0000E+00 -1.2058E-05 0.0000E+00 -5.1582E-01
6.0992E-04
  7 30 1.2454E-02 1.0006E+00 0.0000E+00 0.0000E+00 -1.0203E+00 -
7.2840E-03
  8 35 1.5324E+00 0.0000E+00 -6.7751E-05 0.0000E+00 -1.5314E+00
8.7232E-04
  9 37 1.5479E+00 0.0000E+00 0.0000E+00 0.0000E+00 -1.5479E+00 -
6.4185E-17

```

```

total 7.2268E+00 1.0006E+00 -8.0186E-01 0.0000E+00 -7.3740E+00
5.1603E-02

```

1neutron weight balance in each cell – variance reduction events
print table 130

cell	weight	cell	weight	energy	dxtran	forced	exponential
total							
	window	importance	cutoff	importance		collision	
transform							

1	4	0.0000E+00	-3.4681E-03	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	-3.4681E-03					
2	8	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	0.0000E+00					
3	9	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	0.0000E+00					
4	10	0.0000E+00	5.1852E-05	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	5.1852E-05					
5	11	0.0000E+00	-5.0406E-05	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	-5.0406E-05					
6	27	0.0000E+00	-4.5899E-04	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	-4.5899E-04					
7	30	0.0000E+00	-1.5991E-04	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	-1.5991E-04					
8	35	0.0000E+00	-4.3327E-04	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	-4.3327E-04					
9	37	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	0.0000E+00					

```

total 0.0000E+00 -4.5189E-03 0.0000E+00 0.0000E+00 0.0000E+00
0.0000E+00 0.0000E+00 -4.5189E-03

```

1neutron weight balance in each cell — physical events
print table 130

cell	(n,xn)	fission	capture (n,xn)	loss to fission	loss to	total
1	4	0.0000E+00	0.0000E+00	-5.2875E-02	0.0000E+00	0.0000E+00 - 5.2875E-02
2	8	0.0000E+00	0.0000E+00	-3.3746E-05	0.0000E+00	0.0000E+00 - 3.3746E-05
3	9	6.4202E-06	0.0000E+00	-3.2622E-04	-3.2101E-06	0.0000E+00 - 3.2301E-04
4	10	0.0000E+00	0.0000E+00	-1.2955E-04	0.0000E+00	0.0000E+00 - 1.2955E-04
5	11	0.0000E+00	0.0000E+00	-5.7740E-04	0.0000E+00	0.0000E+00 - 5.7740E-04
6	27	0.0000E+00	0.0000E+00	-1.5093E-04	0.0000E+00	0.0000E+00 - 1.5093E-04
7	30	3.7985E-05	1.0094E-02	-2.3482E-04	-1.8992E-05	-2.4346E-03 7.4439E- 03
8	35	0.0000E+00	0.0000E+00	-4.3905E-04	0.0000E+00	0.0000E+00 - 4.3905E-04
9	37	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00 0.0000E+00
total		4.4405E-05	1.0094E-02	-5.4767E-02	-2.2202E-05	-2.4346E-03 - 4.7085E-02

1neutron activity of each nuclide in each cell, per source particle
print table 140

cell	nuclides	atom	total	collisions	weight lost	weight gain
weight gain						
(n,xn)	fraction	collisions	* weight	to capture	by fission	by
1	4	1001.35c	6.6667E-01	1993175	1.9189E+01	5.1996E-02
		0.0000E+00	0.0000E+00			
		6012.10d	3.3333E-01	276086	2.6753E+00	8.7914E-04
		0.0000E+00	0.0000E+00			
2	8	11023.35c	1.0000E+00	32248	9.6520E-03	3.3746E-05
		0.0000E+00	0.0000E+00			
3	9	13027.04d	1.0000E+00	27586	1.4084E-01	3.2622E-04
		0.0000E+00	3.2101E-06			

4 10 13027.04d 1.0000E+00 9743 4.7847E-02 1.2955E-04
0.0000E+00 0.0000E+00

5 11 1001.35c 6.6667E-01 894879 5.3970E-01 5.3178E-04
0.0000E+00 0.0000E+00

6012.10d 3.3333E-01 169326 1.0260E-01 4.5621E-05
0.0000E+00 0.0000E+00

6 27 13027.04d 1.0000E+00 30078 3.6763E-02 1.5093E-04
0.0000E+00 0.0000E+00

7 30 98252.35c 5.8001E-02 915 9.3920E-03 4.9380E-05 7.6598E-03
1.8992E-05
8016.35c 4.4549E-01 2448 2.4553E-02 1.0168E-04 0.0000E+00
0.0000E+00

13027.04d 4.9651E-01 3180 3.1964E-02 8.3759E-05
0.0000E+00 0.0000E+00

8 35 13027.04d 1.0000E+00 12074 5.9418E-02 4.3905E-04
0.0000E+00 0.0000E+00

total 3451738 2.2867E+01 5.4767E-02 7.6598E-03
2.2202E-05

total over all cells for each nuclide

1001.35c 2888054 1.9729E+01 5.2527E-02 0.0000E+00
0.0000E+00

6012.10d 445412 2.7780E+00 9.2476E-04 0.0000E+00
0.0000E+00

8016.35c 2448 2.4553E-02 1.0168E-04 0.0000E+00
0.0000E+00

11023.35c 32248 9.6520E-03 3.3746E-05 0.0000E+00
0.0000E+00

13027.04d 82661 3.1683E-01 1.1295E-03 0.0000E+00
3.2101E-06

98252.35c 915 9.3920E-03 4.9380E-05 7.6598E-03
1.8992E-05

1 number of discrete reaction rejections in acecol.

13027.04d 3

1tally 4 nps = 100000

tally type 4 track length estimate of particle flux.

tally for neutrons

volumes

cell: 8

segment

1	2.01062E+00
2	2.01062E+00
3	2.01062E+00
4	2.01062E+00
5	2.01062E+00
whole	1.00531E+01

cell 8

segment: -91

multiplier bin: 2.54200E-02 1 102

energy

2.5000E-08	1.82140E-07	0.1604
5.0000E-07	1.82750E-06	0.0554
1.0000E-01	6.35723E-07	0.0371
2.5000E-01	1.04540E-08	0.0391
8.0000E-01	1.16137E-08	0.0389
1.5000E+00	5.49793E-09	0.0583
2.5000E+00	4.80849E-09	0.0590
3.5000E+00	2.62652E-09	0.0937
4.5000E+00	1.14418E-09	0.1282
5.5000E+00	5.89630E-10	0.2156
6.5000E+00	2.39276E-10	0.2334
7.5000E+00	1.73951E-10	0.3560
8.5000E+00	6.70995E-11	0.6868
9.5000E+00	2.18598E-11	0.5146
1.0500E+01	7.79427E-12	0.8926
1.1500E+01	4.82873E-12	1.0000
1.2500E+01	0.00000E+00	0.0000
1.3500E+01	0.00000E+00	0.0000
1.4500E+01	0.00000E+00	0.0000
total	2.68261E-06	0.0426

cell 8

segment: 91 -92

multiplier bin: 2.54200E-02 1 102

energy

2.5000E-08	1.22789E-07	0.1685
5.0000E-07	1.73835E-06	0.0515
1.0000E-01	7.43030E-07	0.0335
2.5000E-01	1.65588E-08	0.0369

8.0000E-01	2.10293E-08	0.0314
1.5000E+00	1.16717E-08	0.0411
2.5000E+00	1.17594E-08	0.0456
3.5000E+00	6.01070E-09	0.0680
4.5000E+00	2.78753E-09	0.0920
5.5000E+00	1.78714E-09	0.1156
6.5000E+00	7.03841E-10	0.1763
7.5000E+00	5.90004E-10	0.1802
8.5000E+00	2.16343E-10	0.2947
9.5000E+00	1.98431E-10	0.3425
1.0500E+01	2.19954E-11	0.9433
1.1500E+01	0.00000E+00	0.0000
1.2500E+01	6.78758E-12	0.7564
1.3500E+01	0.00000E+00	0.0000
1.4500E+01	0.00000E+00	0.0000
total	2.67751E-06	0.0376

cell 8

segment: 91 92 -93

multiplier bin: 2.54200E-02 1 102

energy

2.5000E-08	2.11139E-07	0.2086
5.0000E-07	1.97818E-06	0.0541
1.0000E-01	7.54443E-07	0.0311
2.5000E-01	1.93918E-08	0.0321
8.0000E-01	2.71227E-08	0.0281
1.5000E+00	1.36847E-08	0.0357
2.5000E+00	1.37030E-08	0.0441
3.5000E+00	7.89827E-09	0.0550
4.5000E+00	3.61436E-09	0.0819
5.5000E+00	1.79848E-09	0.1128
6.5000E+00	9.45322E-10	0.1446
7.5000E+00	6.84195E-10	0.1751
8.5000E+00	1.52150E-10	0.2845
9.5000E+00	1.68163E-10	0.3211
1.0500E+01	2.60667E-11	0.6092
1.1500E+01	9.79323E-11	0.7024
1.2500E+01	6.44010E-11	0.7550
1.3500E+01	0.00000E+00	0.0000
1.4500E+01	0.00000E+00	0.0000
total	3.03312E-06	0.0419

cell 8

segment: 91 92 93 -94

multiplier bin: 2.54200E-02 1 102

energy

2.5000E-08	1.86153E-07	0.1877
5.0000E-07	1.89312E-06	0.0576
1.0000E-01	7.21663E-07	0.0326
2.5000E-01	1.76593E-08	0.0358
8.0000E-01	2.18033E-08	0.0285
1.5000E+00	1.18903E-08	0.0381
2.5000E+00	1.06420E-08	0.0433
3.5000E+00	6.40164E-09	0.0530
4.5000E+00	2.95468E-09	0.0782
5.5000E+00	1.84821E-09	0.1083
6.5000E+00	9.75950E-10	0.1506
7.5000E+00	5.72503E-10	0.2386
8.5000E+00	2.40275E-10	0.3036
9.5000E+00	7.31705E-11	0.4290
1.0500E+01	1.60635E-10	0.3696
1.1500E+01	2.38340E-12	1.0000
1.2500E+01	1.30856E-11	0.6494
1.3500E+01	7.10387E-11	0.5993
1.4500E+01	0.00000E+00	0.0000
total	2.87625E-06	0.0432

cell 8

segment: 91 92 93 94
multiplier bin: 2.54200E-02 1 102

energy

2.5000E-08	1.66256E-07	0.1777
5.0000E-07	1.78835E-06	0.0557
1.0000E-01	6.26302E-07	0.0353
2.5000E-01	1.11997E-08	0.0357
8.0000E-01	1.25662E-08	0.0337
1.5000E+00	5.30756E-09	0.0470
2.5000E+00	4.11226E-09	0.0601
3.5000E+00	2.20186E-09	0.0741
4.5000E+00	9.22521E-10	0.1047
5.5000E+00	4.17559E-10	0.1517
6.5000E+00	2.73372E-10	0.2527
7.5000E+00	1.13565E-10	0.3016
8.5000E+00	4.45025E-11	0.6134
9.5000E+00	6.80037E-11	0.6017
1.0500E+01	1.03873E-10	0.8453
1.1500E+01	2.37384E-11	0.7561
1.2500E+01	1.36128E-11	1.0000
1.3500E+01	2.15137E-11	0.7682
1.4500E+01	0.00000E+00	0.0000

total 2.61830E-06 0.0431

cell 8

segment: whole cell

multiplier bin: 2.54200E-02 1 102

energy

2.5000E-08	1.73695E-07	0.1044
5.0000E-07	1.84510E-06	0.0370
1.0000E-01	6.96232E-07	0.0208
2.5000E-01	1.50527E-08	0.0205
8.0000E-01	1.88270E-08	0.0176
1.5000E+00	9.61042E-09	0.0220
2.5000E+00	9.00502E-09	0.0247
3.5000E+00	5.02780E-09	0.0327
4.5000E+00	2.28465E-09	0.0458
5.5000E+00	1.28820E-09	0.0640
6.5000E+00	6.27552E-10	0.0862
7.5000E+00	4.26844E-10	0.1095
8.5000E+00	1.44074E-10	0.1966
9.5000E+00	1.05926E-10	0.2071
1.0500E+01	6.40728E-11	0.3568
1.1500E+01	2.57766E-11	0.5550
1.2500E+01	1.95774E-11	0.5340
1.3500E+01	1.85105E-11	0.5761
1.4500E+01	0.00000E+00	0.0000
total	2.77756E-06	0.0287

for the tally-fluctuation-chart bin:

average tally per history = 2.79230E-05
1.76708E-02

largest tally in one history =

(largest tally)/(average tally) = 6.32838E+02 nps of largest tally = 38542

1tally 14 nps = 100000

tally type 4 track length estimate of particle flux.

tally for neutrons

volumes

cell: 8

segment

1	2.01062E+00
2	2.01062E+00
3	2.01062E+00
4	2.01062E+00
5	2.01062E+00

whole 1.00531E+01

cell 8
 segment: -91
 multiplier bin: 2.54200E-02 1 102
 energy
 5.0000E-07 2.00964E-06 0.0536

cell 8
 segment: 91 -92
 multiplier bin: 2.54200E-02 1 102
 energy
 5.0000E-07 1.86113E-06 0.0501

cell 8
 segment: 91 92 -93
 multiplier bin: 2.54200E-02 1 102
 energy
 5.0000E-07 2.18932E-06 0.0553

cell 8
 segment: 91 92 93 -94
 multiplier bin: 2.54200E-02 1 102
 energy
 5.0000E-07 2.07928E-06 0.0567

cell 8
 segment: 91 92 93 94
 multiplier bin: 2.54200E-02 1 102
 energy
 5.0000E-07 1.95460E-06 0.0548

cell 8
 segment: whole cell
 multiplier bin: 2.54200E-02 1 102
 energy
 5.0000E-07 2.01879E-06 0.0366

for the tally-fluctuation-chart bin:
 average tally per history = 2.02951E-05 largest tally in one history =
 1.76708E-02
 (largest tally)/(average tally) = 8.70689E+02 nps of largest tally = 38542
 1 warning. some tally scores were not made for various peculiar reasons:

	beyond last bin	not in
tally	angle	energy time user
14	0	258464 0 0

tally fluctuation charts

tally 4			tally 14			
nps	mean	error	fom	mean	error	fom
8000	2.94948E-06	0.0924	18	2.16978E-06	0.1164	11
16000	2.99259E-06	0.0688	16	2.22582E-06	0.0864	10
24000	2.92335E-06	0.0573	15	2.13828E-06	0.0722	9.8E+00
32000	2.93016E-06	0.0499	15	2.15459E-06	0.0626	9.7E+00
40000	2.85589E-06	0.0465	14	2.09453E-06	0.0589	8.8E+00
48000	2.87279E-06	0.0417	15	2.11420E-06	0.0527	9.2E+00
56000	2.81518E-06	0.0380	15	2.05985E-06	0.0482	9.4E+00
64000	2.85945E-06	0.0361	15	2.09546E-06	0.0457	9.1E+00
72000	2.86453E-06	0.0342	15	2.10065E-06	0.0434	9.0E+00
80000	2.84759E-06	0.0325	14	2.07933E-06	0.0414	8.9E+00
88000	2.84684E-06	0.0307	15	2.08697E-06	0.0389	9.2E+00
96000	2.79347E-06	0.0293	15	2.04167E-06	0.0373	9.2E+00
100000	2.77756E-06	0.0287	15	2.01879E-06	0.0366	9.2E+00

*

dump no. 2 on file runtpf nps = 100000 ctm = 81.58

tally data written to file mctam

7 warning messages so far.

run terminated when 100000 particle histories were done.

mcnp version 4.2 05/01/94
05/01/94 14:56:53

I 05/01/94 16:18:37

probid = I

lmcnp version 4.2 ld=05/01/94

l 05/01/94 14:56:53

probid = l 05 01/94 14:56:53

mp=fit1

```

1  lnp101.a -- Cf-252 and Na-23(center to center)
2  4  2 -.961 -90 19  S polyethylene within tank & tubes
3  8  1 -.971 14 -15 -13 S sodium sample
4  9  3 -2.7 -7 -1 2 #30 S aluminum encapsulate of volume source
5  10 3 -2.7 -18 1 -90 S aluminum tube for source replacement
6  11 2 -.961 6 -8 -16 #8 S sodium sample encapsulate
7  27 3 -2.7 -17 16 -90 S aluminum tube for sample replacement
8  30 4 -4.18 -5 -4 3 S volume source(Al.,O, Cf)
9  35 3 -2.7 -90 -19 20 S main cylinder to sepearate tank & irradi. zone
10 37 0 -20 -90 17 18
11 40 0 (90 : -1 : -16)#9 #30 #11 #8 S zero-importance outside world
12
13 1 c/x -3.4 0 .8
14 2 px 13
15 3 px 14.68
16 4 c/x -3.4 0 .32
17 5 px 15.32
18 6 px 10
19 7 px 17
20 8 px 20
21 13 cx .8
22 14 px 12.5
23 15 px 17.5
24 16 cx 2.2
25 17 cx 2.4
26 18 c/x -3.4 0 1
27 19 cx 10.15
28 20 cx 10
29 90 sx 15 100
30 91 px 13.5
31 92 px 14.5
32 93 px 15.5
33 94 px 16.5

```

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