

## Chapter 6: Example Problems

### Overview

TART95 is distributed with a number of example input problems that can be used to become familiar with preparing TART95 input parameters and interpreting output results. The examples presented here were selected to illustrate the use of as many different TART95 input options as possible, so that readers could see how these options are actually used. A second use of these problems is to allow users to understand what is important in each problem and how sensitive the results are to changes in input parameters. There are a number of suggested additional exercises that users can perform to learn more about input preparation and sensitivity of results to changes in input. We encourage users to run these example problems and to try the suggested exercises. Here we will discuss a number of these problems; additional problems are distributed with TART95.

### Running TART95 Problems

Before running any full TART95 problems it is strongly recommended that you first:

- 1) completely check your input using **TARTCHEK**. Use all of the option to check for empty zones, overlapping zones, etc. Check for both zones and materials in zones.
- 2) run a small preliminary problem to insure your problem can run to conclusion and the output contains what you want. To do this use **sentl 2** and **3** to minimize the number of batches and histories per batch. For example, for criticality problems run 10 batches (**sentl 2 10**) with 500 histories per batch (**sentl 3 500**). For source problems run 5 batches (**sentl 2 5**) with 500 histories per batch (**sentl 3 500**). With these parameters either criticality or source problems should run to completion in seconds.

All of the example input decks distributed with TART95 have filename extensions .IN, e.g., TART.IN, HEXAGON.IN, etc. To run any TART95 problem you have the option to either explicitly include on the execution line the names of the input parameter file and output report file, or you can omit these names, in which case it will use the default names TART.IN and TART.OUT. If you use the default names you should copy your input parameter file to TART.IN, and if you wish to prevent the output report file from being overwritten the next time you write the code, you should also rename or copy the output file TART.OUT to have some other name. Note, the option to include file names on the execution line is not currently available on an IBM-PC; you must use the default names. In general to run a problem the execution line is,

```
TART95 [input parameter file] [output report file]
```

For example,

tart95 README WRITEME

will read input parameters from a file named README and write the output report to a file WRITEME. TART95 is case sensitive, i.e., it distinguishes between upper and lower case characters. So that inputting README is not equivalent to inputting readme.

or to use the default file names,

tart95

Unless instructed to do otherwise by input parameters (**sentl 12**) TART95 will always start with exactly the same random number seed. This means that if you run the same problem more than once on the same computer you will get exactly the same answer - not statistically the same - exactly the same. Therefore do not try to improve the statistical results of a problem by running exactly the same problem a number of times, assuming that you can then add up the answers and thereby reduce the statistical uncertainty. If you do this all you will get back is an exact duplicate of your previous results, which obviously are completely correlated between the two runs, so that you cannot use the two results together to reduce uncertainty. As explained elsewhere in this report, if you want to attempt to do this, you should use **sentl 12** to input the ending random number seed from one run as input as the starting seed for the next run.

### Criticality Problems

#### TART95's Definition of $K_{\text{eff}}$

TART95 defines  $K_{\text{eff}}$  as the ratio of **ALL** neutrons produced divided by neutron disappearance (absorption plus leakage) per generation (static) or time interval (dynamic). This definition may differ from that used by other codes that define  $K_{\text{eff}}$  as the ratio of **FISSION** neutrons produced divided by neutron disappearance. For example, in defining the neutron balance between production and disappearance TART95 includes in the production, fission,  $(n,2n)$ ,  $(n,3n)$ , etc., i.e., any processes that produce a multiple number of neutrons. Other codes may only include fission neutrons in their definition of neutron production. The user should be aware of the different possible definitions used by various codes.

#### Static vs. Dynamic Criticality Problems

TART95 can perform static or dynamic criticality calculations. The basic assumption in either calculation is that if a system containing fissile material is pulsed with a source of neutrons, regardless of the initial energy, direction, and position of the source of neutrons, the system will eventually relax into a single fundamental mode corresponding to the critical distribution in energy, position and direction. If the system is exactly critical it will stay in this distribution at a constant flux, or power level. If the system is subcritical the distribution will decrease and eventually die away toward zero. If it is super critical the

distribution will increase and if no other effects are introduced to reduce the reactivity the system will eventually destroy itself.

In the static case it is assumed that everything is **time independent** and whether or not the system is exactly critical the system is forced into a pseudo-static, time independent condition by artificially changing the number of neutrons per fission by the ratio of the real number of fission per neutron to the reactivity of the system ( $K_{\text{eff}}$ ). If the system is exactly critical  $K_{\text{eff}}$  will be exactly unity and the calculation will use the real number of neutrons per fission. If it is sub critical it will use a number of neutrons per fission larger than the actual value, and if super critical it will use a value less than the actual value.

In the dynamic case it is assumed that everything is **time dependent** and whether or not the system is exactly critical the system is forced into a pseudo-static, time independent condition by assuming that the distribution at all points in space is changing exponentially as a function of time according to  $\text{Exp}(\alpha t)$ , so that by multiplying the distribution by a compensating factor  $\text{Exp}(-\alpha t)$  the resulting distribution will be static, time independent.

In static problems TART95 will start from a given number of neutrons (defined by input or default), and track neutrons through **successive generations**. At the end of each generation the multiplication of the system ( $K_{\text{eff}}$ ) is defined by the ratio of the number of neutrons at the end of the generation to the number at the beginning of the generation. At the end of each generation, in order to keep the cycle going, the number of neutrons is statistically reset to the original number by effectively multiplying  $\bar{\nu}$  by  $1/K_{\text{eff}}$ .

In dynamic problems TART95 will start from a given number of neutrons (defined by input or default), and track neutrons through **successive time intervals**. At the end of each time interval the time constant for the system,  $\alpha$ , and multiplication,  $K_{\text{eff}}$ , are defined by the ratio of the number of neutrons at the end of the time interval to the number at the beginning of the time interval. At the end of each time interval, in order to keep the cycle going, the number of neutrons is statistically reset to the original number by multiplying the distribution by  $\text{Exp}(-\alpha t)$ .

Note, that the static and dynamic calculations are using different approaches. For systems that are very close to critical the calculated static and dynamic  $K_{\text{eff}}$  will be the same, or very nearly the same. For systems that are not close to critical the results can differ by a significant amount; this will be discussed more below, when dynamic criticality calculations are covered.

There is a basic problem that is encountered when one tries to use the ENDL data and TART95 for any problem involving fission: ENDL does not contain any information defining the time dependent emission of delayed neutrons. Therefore the basic data file used by TARTNP and TART95 only uses the prompt number of neutrons per fission as  $\bar{\nu}$ ; not only is the time dependent emission of delayed neutrons not included, even the basic number of delayed neutrons is not included in the definition of  $\bar{\nu}$ . This means

that when you run criticality problems, static or dynamic, you should expect these codes to slightly underestimate the criticality of any system. Fortunately over a large energy range from thermal to well into the MeV range the ratio of the delayed to prompt number of neutrons per fission is virtually constant. Since the reactivity of any system is defined as the ratio of fission neutrons produced to neutron disappearance (absorption within the system plus leakage from the system), generally it is a good approximation to assume that the reactivity of the system is a linear function of the number of neutrons produced per fission (the numerator of the above mentioned ratio). So that when TARTNP or TART95 is used it is a very good approximation to assume that the calculated  $K_{\text{eff}}$  will be lower than the real value by the ratio of the delayed neutron fraction to the total number of delayed neutrons. For  $\text{U}^{235}$  this ratio is about 0.64 % and for  $\text{Pu}^{239}$  it is about 0.21 %. For example, if you have a system which is really exactly critical, if it is a  $\text{U}^{235}$  system you should expect these codes to calculate 0.9936, and for a  $\text{Pu}^{239}$  system about 0.9979. The lack data in ENDL to describe the time dependent emission of delayed neutrons presents additional problems when performing dynamic (time dependent) criticality calculations; this will be discussed in detail below under the section on dynamic criticality calculations.

### Example Criticality Input Parameters

Before looking at results, let's first go through a typical TART95 input parameter deck. The following deck corresponds to the first of 68 fast critical assemblies described below. This deck was automatically produced by the **TARTVIEW** code, which will be available to users in the near future. Note, the extensive use of comments to explain everything and an organized grouping of each type of input. It is suggested that rather than starting from nothing to create input decks, you should start from an existing input deck distributed with TART95 and modify it to meet your needs. Many of the input options are exactly the same for a wide variety of applications, so that this approach will both save you time in preparation and help to insure that you do not forget to include important options.

The following input deck is for a static criticality calculation, involving a simple spherically symmetric system, with Pu in the center, surrounded by a thin shell of Ni, followed by a Be reflector.

The deck starts with two lines to identify the problem and route the output to a computer output box. In this case the **name** option identifies this program is Pu surrounded by 5.222 cm of Be. The **box** option is included with TART95 merely for compatibility with TARTNP; TART95 leaves the output on disk and does not route it to an output box.

```
* =====
name      c10100  pu-a    be      5.222
box o84   pu239  spherical multi-band
* =====
*
* TART Input Deck Generated Using Program TARTVIEW (93-1)
*
* =====
```

The next section identifies this as a criticality problem, rather than a source problem. The **critcalc** input says to run 15 settle cycles, repeat (do) the calculation only once, run until the statistical uncertainty in  $K_{\text{eff}}$  is less than 3 %. Since no time step is included on the **critcalc** input line, this is a static criticality problem. After running the static calculation to define the neutron removal time, this same input deck could be used for a dynamic criticality calculations by merely adding a time step on the **critcalc** input line and running TART95 again.

```
* =====
*
* Criticality Problem
* 1) Number of Settle Cycles (should be 10 to 20)
* 2) Number of Repetitions after Settling (usually 1)
* 3) % Standard Deviation to Stop Repetition (default 3.0)
* 4) Time Step (shakes) - Only for Dynamic K Calculation
*
* =====
critcalc      15      1      3.000
```

The next section defines the surfaces. There are three spherical surfaces of 4.075, 4.088 and 9.31 cm radius, and all are centered on the origin. Note, in this case the zero input for the (x, y, z) center of the spheres could have been omitted, since the default is (0, 0, 0).

```
* =====
*
* Surface Definitions
* 1-D Spherical - Only Spherical Surfaces Required
* 1) Surface Number - used later to Define Zone Boundaries
* 2) Position - Radius Z0 X0 Y0
*
* =====
sphere      1  4.07500e+00  0.00000e+00  0.00000e+00  0.00000e+00
sphere      2  4.08800e+00  0.00000e+00  0.00000e+00  0.00000e+00
sphere      3  9.31000e+00  0.00000e+00  0.00000e+00  0.00000e+00
```

The next section defines the zones. We define each zone by defining the surfaces that bound it. To do this we use the surface numbers, defined above, and a sign to indicate whether the zone is inside or outside each surface. There are four zones. One first zone is inside the inner most sphere of radius 4.075 cm, and only requires one bounding surface. The second zone is between the first and second spheres of 4.075 and 4.088 cm radius, and requires two bounding surfaces. Note, the input says that zone 2 is outside of surface 1 (-1) and inside of surface 2 (2). Similarly the third zone is between the second and third spheres of 4.088 and 9.31 cm radius and requires two bounding surfaces. Finally the fourth zone is defined to be everything outside the third sphere of radius 9.31 cm and only requires one bounding zone, to define it to be outside surface 3 (-3 input).

```
* =====
*
* Zone Definitions
* 1) Zone Number
* 2) Bounding Surface Number and Sign of Vector Normal
```

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```

*   to the Surface for Particles Leaving the Zone.
*   2) Is repeated to include all Bounding Surfaces.
*
* =====
jb    1    1
jb    2   -1    2
jb    3   -2    3
jb    4   -3

```

The next section defines the materials to use in this problem. Three materials are defined. The first material has an overall density of 19.14 grams/cc, with relative atom fractions: Pu<sup>239</sup> of 93.7468, Pu<sup>240</sup> of 5.78658, and Pu<sup>241</sup> of 0.46666. Note, these fractions are relative and the sum need not be normalized; the code will normalize everything to obtain the correct atoms/cc to obtain the correct overall density. The second material is natural Ni at a density of 8.902 grams/cc. The third material is Be<sup>9</sup> and O<sup>16</sup>, at an overall density of 1.84 grams/cc, and relative atom fractions of 99.2756 and 0.724353, respectively.

```

* =====
*
* Material Definitions
* 1) Material Number
* 2) Density (grams/cc)
* 3) Atom %
* 4) Isotope I.D. (ZZZAAA), e.g., 92238 for U-238
*   3) and 4) can be repeated in pairs to define
*   composite materials
*
* =====
* 94-Pu-239
matl  1  1.91400e+01  9.37468e+01  94239  5.78658e+00  94240 &
      4.66660e-01  94241
* 28-Ni-Nat
matl  2  8.90200e+00  1.00000e+02  28000
* 4-Be-9
matl  3  1.84000e+00  9.92756e+01  4009  7.24353e-01  8016

```

The next section assigns materials to zones. The first assignment of material 0 (0 = no material = vacuum) is optional. TARTVIEW includes it in its output as a reminder to users of the zones that have not been assigned any material. In this case it is explicitly assigning no material to the outside of the largest radius sphere, zone 4 defined above. The other three assignments are, Pu in the inner most zone 1, surrounded by a thin sphere as Ni zone 2, and Be in zone 3. If you would like to see the effect of the thin Ni sphere you can simply comment out the assignment of material 2 to zone 2, \* **matz 2 2**, and assign Be to both zones 2 and 3, **mat 3 2 3**, and re-run the problem.

```

* =====
*
* Assignment of Materials to Zones
* 1) Material Number
* 2) Zone Containing Material - Can be Repeated
*
* =====
* Vacuum - ILLEGAL EXCEPT IN OUTER, NON-RE-ENTRANT ZONES
matz  0    4

```

```
* 94-Pu-239
matz  1    1
* 28-Ni-Nat
matz  2    2
* 4-Be-9
matz  3    3
```

The next section describes the initial source guess to use at the beginning of the settle cycles. In this case a spherical shell source has been specified (**source3**) in the middle two thirds of the inner most sphere of Pu. For static criticality calculations the results are very insensitive to the initial spatial distribution and we could have just as easily specified a point source at the origin. Dynamic criticality calculations can be more sensitive to the initial spatial distribution; this will be discussed below under the section on dynamic criticality calculations.

The default for the energy spectrum is a neutron induced fission spectrum, and the default angular distribution is isotropic. Since the following section does not explicitly define either the energy spectrum or angular distribution, the defaults will be used. There is no reason to use any other options for criticality problems, but if we had wanted to specify a different energy spectrum there are a number of possible options, the simplest being **sentl 4**, to define a monoenergetic neutron source. The default for this option is zero, indicating a neutron induced fission spectrum. Similarly for the angular distribution we could have used **sentl 6** and **7**, to define a uniform cosine range. These options default to 2.0 and -1.0, respectively, indicating an isotropic angular distribution.

```
* =====
*
* Definition of Sources
*
* For Criticality Problems a Uniform, Isotropic, Fission
* Spectrum is used in All Zones containing Fissile Material
*
* =====
source3  1  6.79167e-01  3.39583e+00  0.0  0.0  0.0
```

The last section defines input options, mostly in terms of the sentinels used in the problem. Hopefully all of the following are self explanatory and need not be described further here. The one option that should be mentioned is **sentl 24**. TARTNP and TART95 can be used to run any number of problems one after the other. For example, the 68 fast critical assemblies discussed below were all run during a single continuous execution of TART95. One TART95 problem extended from a starting **name** input line to an **end** input line. At the end of each problem, if **sentl 24** is not set in the input for the current problem, TART95 will terminate without looking for any following problems. Therefore if you wish to run a number of problems one after the other you MUST include **sentl 24 1** in your input, as is done below in this sample input.

It is worth mentioning that if you wish to allow yourself the flexibility to later combine input deck for a single run, you can always include **sent 24 1** in your input decks. When you run any of the these decks TART95 will always look for another following input

problem. When TART95 gets to the end of all of your input it will merely print an error message that no input was found and it will then terminate. This will have no effect on any results produced in the preceding calculations, and can be ignored.

```
* =====
*
* Definition of Running Conditions and Output Edit Options
*
* =====
* 1) Transport (neutrons and/or photons) (0)
sentl 1 1
* 2) Number of Samples (20)
sentl 2 2000
* 3) Histories per Sample (5000)
sentl 3 1000
* 8) Neutron Minimum Energy (2.53e-8 MeV)
sentl 8 2.53000e-08
* 20) Multi-Band Sentinel (0)
sentl 20 1
* 39) Thermal Scattering Sentinel (0)
sentl 39 1
* Thermal Scattering Temperature in All Zones (2.53e-8 MeV)
emin 2.53000e-08 1 thru 4
* 24) Continuation - another problem follows this one
sentl 24 1
end
```

### New TART95 Features

TART95 output has been designed to be as close of possible to TARTNP output, in order to allow TARTNP users to easily use TART95 and obtain results that are very similar to what they are used to.

In addition to the output normally obtained from TARTNP, TART95 contains a few new features that will be explained below. The following has been extracted from the TART95 output file, TART.OUT. Most of the following information will appear both in the TART95 output and on your screen, so that you can monitor the progress of calculations in real time.

The following output is for a static criticality calculation involving a bare (unreflected) sphere is enriched uranium. We will refer back to the output later when we discuss dynamic criticality calculations.

When TART95 starts it will identify itself by a version number and date, e.g., in the following listing the code is identified as the TART 95-1 version from January 1995. This information is important if you wish to report errors or any difficulties in using the code. Only by knowing what version of the code you are using will we be able to assist you.

TART95 next identifies all the files that it will be using. You should in particular check the names of the Input Parameter and Output Listing files, to insure that you are running the correct problem. It lists the dates of all four data files used by TART95, which allows you

to determine at any later date exactly what data you used in the calculation. TART95 then lists the time, date and type of computer it is running on, e.g., in this case it is running on an HP computer. This is important information that will allow you to determine at some later date exactly when and on what computer you ran this problem.

```
TART - Coupled Neutron-Photon Monte Carlo Transport (TART 95-1, Jan.95)
=====
I/O Files Opened for Entire Run
=====
Definition                               Filename  Unit   Date
=====
TART Input Parameters.....TART.IN          2
TART Output Listing.....TART.OUT           3
Neutron Interaction Data File.....TARTND      7   6/10/92
Photon Interaction Data File.....GAMDAT       8   3/19/93
Neutron Induced Photon Production File...TARTPPD  9   6/10/92
Multi-Band Parameter File.....NEWCROSS      10  7/19/90
=====
Start of Next Problem      11:57:46 May16'95 HP
=====
```

After reading and checking your input TART95 will list a summary of the running conditions. It will tell you the type of problem: static or dynamic reactivity, or source problem, what particles will be tracked: neutrons, photons, or neutrons and photons, whether the recommended options for self-shielding (Multi-band) and thermal scattering are on or off, and finally the number and size of batches. When the problem starts you should check this list (it will also appear on your screen), particularly to insure that the correct type of problem will be run, tracking the right particles, using the recommended options, with the correct number and size of batches.

```
=====
Summary of Running Conditions
=====
Type of Problem      : Static Reactivity
Particles Tracked   : Neutrons
Multi-band method   : On
Thermal scattering  : On
Repetitions         :          1
Settle Cycles       :          15
Batches             :         2000
Particles/Batch     :         1000
=====
```

After listing a detailed description of all input parameters, options, zone masses and volumes, cross sections and expected energy deposition for all materials in the problems (exactly the same as TARTNP, and described in detail below), the criticality calculation will start.

TART95 lists the results for every cycle, both in the output file and on your screen, so that you can monitor the results in real time as they are being produced. The output includes the  $K_{eff}$  calculated for each cycle, the  $K_{eff}$  averaged over all cycles (the sum for  $K_{eff}$  for all cycles divided by the number of cycles) and the time used in seconds. It also includes

the balance between absorption within the system and leakage from the system, so that you have some idea which is more important and what the system is sensitive to.  $K_{eff}$  is defined as the ratio of neutrons produced to neutrons disappearing, i.e., neutrons produced in fission divided by the sum of absorption and leakage (the only ways that neutrons can disappear). The output includes  $K_{eff}$ , but it also turns this ratio inside down so that we can separately define the number of neutrons absorbed per fission neutron and the number of neutrons that have leaked per fission neutron. If the system is exactly critical, for each neutron produced by fission the sum of absorption and leakage must be exactly unity. If the system is sub critical there is too much absorption and/or leakage and the sum will be greater than unity. Conversely, if the system is super critical there is not enough absorption and/or leakage and the sum will be less than unity.

First TART95 will run settle cycles, to allow your initial guess of the flux distribution (in energy, space and direction), to settle toward the critical distribution of the system. During the settle cycles the output will include all of the terms described above, such as the average  $K_{eff}$ , however once the settle cycles are over these settle cycle statistics are ignored; the real criticality calculation starts and the definition of the final averaged  $K_{eff}$  is only based on the statistics from this point on. The calculation will continue until either the maximum number of batches have been processed (**sentl 2**) or the results are within the requested accuracy. Periodically the code will notify you when it feels that 25 %, 50 %, etc. of the calculation has been completed. These are merely estimates, so don't get confused or think anything is wrong if the code says the calculation is 100 % completed and then runs a few more batches, as happens in this case.

-----  
Settle cycle k values  
-----

Cycle	Keff-Batch	Keff-Average	Absorbed Per Fission	Leaked Neutron	Time Used (Seconds)
1	1.034610	1.034610	.433020	.533528	1.00
.					
14	.939322	.986401	.435707	.578282	2.00
15	.984971	.986306	.435721	.578351	2.00

Batch	Keff-Batch	Keff-Average	Absorbed Per Fission	Leaked Neutron	Time Used (Seconds)
1	.948632	.948632	.434586	.619563	2.00
2	.955938	.952285	.434499	.615555	2.00
.					
128	1.000215	.990766	.434259	.575054	30.00
129	.979648	.990679	.434261	.575141	30.00
		.990679+/-	.001998	s.d. 25. %	Completed
130	.989146	.990668	.434270	.575144	30.00
.					
493	.984647	.993021	.434429	.572641	126.00
		.993021+/-	.001000	s.d. 100. %	Completed
494	1.019051	.993074	.434435	.572581	127.00

```

495      .958934      .993005      .434434      .572654      127.00
496      .991462      .993002      .434439      .572652      127.00
497      .970400      .992956      .434441      .572696      128.00

```

Once the calculation is completed the output will include a brief summary of the results, clearly identifying the recommended  $K_{\text{eff}}$  (this has been a problem in the past for TARTNP users, since TARTNP merely lists a number of possible values of  $K_{\text{eff}}$  without telling users what is recommended for their use). It also includes a variety of times to let you know the speed at which things are happening. Note, in particular the removal lifetime, which can be used to define the recommended time step for dynamic criticality calculations. Most of these quantities are also listed by TARTNP, but TART95 has added an estimate of the time constant, alpha, for the system; this approximation can often be used to avoid having to run a dynamic criticality calculation. Below, in the discussion of dynamic problems, we can compare the below estimate, based on this static calculation, to the value from a dynamic calculation.

```

Expected k = 9.92956E-01  Std dev = 9.964E-04 (Recommended)
Actual k   = 9.88887E-01  Std dev = 1.913E-03
Mixed k   = 9.95099E-01  Std dev = 1.290E-03
ekbar     = 9.92914E-01

```

All times are in microseconds.

```

Removal lifetime      6.03623E-03.  Std dev  1.146E-05
Leakage lifetime     1.05919E-02
Absorption lifetime  1.40341E-02
Production lifetime  5.42183E-03.  Std dev  1.899E-05
Time to prod. event  2.16328E-03.  Std dev  8.121E-06
Time to capt. event  4.27807E-04.  Std dev  4.379E-06
Time to leakage      3.44515E-03.  Std dev  8.566E-06
Approx.(alpha)/usec -1.16694E+00  (Expected k - 1)/(Removal lifetime)

Normalized leakage           5.69889E-01
Normalized census           9.88887E-01
Normalized net collision gain 5.58777E-01
Normalized absorption        4.57948E-02

```

One convenient new feature of TART95 is an independent analysis of the results using a completely different approach from that used by TARTNP. All calculated values of  $K_{\text{eff}}$  between 0.9 and 1.1 are binned using very fine increments in  $K_{\text{eff}}$  of 0.0001. The results are first shown using a coarser grouping of 0.01 bin widths; this presentation allows you to check for unreasonable outlying results, or a non-physical shape to the statistical results. The results are then averaged over all of the 0.0001 width bins, assuming that any results that fall in a bin have a  $K_{\text{eff}}$  corresponding to the center of the bin. In this case comparing the above Expected K of 0.9929 to the below bin averaged value of 0.9930 shows excellent agreement.

Frequency Distribution for Expected K Values from 0.9 to 1.1  
(0.01 bin width sums)

```

Expected K Range  Occurrences
.900 .910        0
.910 .920        0

```

.920	.930	3	X
.930	.940	1	
.940	.950	11	XXXXXX
.950	.960	20	XXXXXXXXXXXX
.960	.970	40	XXXXXXXXXXXXXXXXXXXXXXXXXXXX
.970	.980	68	XX
.980	.990	78	XX
.990	1.000	79	XX
1.000	1.010	84	XX
1.010	1.020	63	XX
1.020	1.030	25	XXXXXXXXXXXX
1.030	1.040	19	XXXXXXXXXXXX
1.040	1.050	4	XX
1.050	1.060	2	X
1.060	1.070	0	
1.070	1.080	0	
1.080	1.090	0	
1.090	1.100	0	

```
-----
                Sum      497 (inside 0.9 to 1.1 Range)
                        0 (outside 0.9 to 1.1 Range)
-----
Average   .9930 +/-   .0222 s. d. (0.0001 bin width average)
```

Next, using the standard deviation calculated for the bin averages (0.0222 in this case) the output includes an analysis of how the results are distributed in terms of multiplies of the standard deviation from the average. Note, in this case even though we only have 497 results they are distributed almost exactly in a normal distribution: 66.8 % are within one standard deviation, and 95.7 % are within two standard deviations. The results are also very symmetric about the mean, e.g., 33.4 % are within one standard definition on either side of the mean. This is a fairly sensitive test for outlying or other problems in calculations and is something that you should check for each calculation.

Confidence Limits (Occurrences out to +/- 10 times s.d.)

s.d.	Range	Occurrences	Per-Cent
-4	-3	3	.604
-3	-2	9	1.811
-2	-1	69	13.883
-1	0	166	33.400
0	1	166	33.400
1	2	75	15.091
2	3	9	1.811
-----			
	Sum	497	

**Static Criticality Calculations**

**Fast Critical Assemblies**

TART95 is distributed with an example problem involving a set of 68 fast critical systems. These are all geometrically simple, involving only spherical or cylindrical fuel regions, surrounded by either nothing or some reflecting material. These examples have been used as one set of benchmarks to compare results obtained using the current CRAY production

version of TARTNP, to the results obtained using TART95 on CRAY, HP, SUN, SGI, IBM-RSIC, and IBM-PC.

Examples are included for a wide variety of reflecting materials. By comparing critical masses, etc., you will be able to determine the effectiveness of each of these reflecting materials. By turning self-shielding on or off (**sentl 20**) you will be able to determine whether or not self-shielding is important for fast systems. Generally people think of self-shielding as important only at lower energies, as in the keV resonance region of fissile isotopes. But many materials, particularly metals, have resonances well up into the MeV energy range, and you might be surprised to find out how important self-shielding can be even for these fast critical assemblies.

The below table shows the results obtained running all 68 assemblies on an HP-350 computer. The following table was prepared using the **CRITEDIT** utility code; this code reads the entire TART95 output file, extracts information that characterizes each problem, and producing a table defining the average and spread in the results. This table included a description of each problem, as far as the fuel and reflector, the calculated expected k value, and a few quantities that can be used to simply characterize the system, including the neutron removal life, median and average fission energy. The table also includes the time required to run each problem.

The results illustrate one of the powers of TARTNP and TART95: speed. In this case it took 6873 seconds to run all 68 problems; an average of only about 100 seconds per problem. The running time for specific problems varies, being longer from system with thick reflectors, and less than a minute for systems with no or a thin reflector. In no case, even for the problems involving the thickest reflectors, did any of these problems take more than about 7 minutes.

By examining the median and average fission energy we can see that all of these systems are fast in terms of neutron energy. By examining the removal lifetime we can see that there can be a great deal of variation as far as how fast (in terms of time) events are happening in these systems. For example, for the first three problems involving progressively thicker Be reflectors we can see that the removal lifetime varies from about 0.63 to 21.65 microseconds. Removal lifetime will be discussed in detail below under the section on dynamic criticality calculations. Note, the increase in running time for these three problems, as the Be reflector is made progressively thicker.

Crit. #	Fuel	Reflector	Expected K	Removal Lifetime (Microsec.)	Median Energy (MeV)	Average Energy (MeV)	Seconds	
c10100	pu-a	be	5.222	9.90524E-01	6.38582E-02	1.18878E+00	1.93752E+00	70.000
c20100	pu-a	be	8.170	9.99855E-01	1.37900E+00	1.17076E+00	1.94074E+00	188.000
c30100	pu-a	be	13.000	9.99346E-01	2.16577E+01	1.16526E+00	1.99200E+00	413.000
c40100	pu-d			1.00265E+00	3.91365E-03	1.19757E+00	1.84523E+00	31.000
c50100	pu-d	be	3.690	9.96745E-01	1.67315E-02	1.14493E+00	1.88450E+00	76.000
c60100	pu-d	be	5.250	9.97792E-01	8.37375E-02	1.15443E+00	1.91230E+00	112.000
c70100	pu-d	c	3.830	9.94610E-01	9.40696E-03	1.13822E+00	1.76072E+00	48.000
c80100	pu-d	ti	8.000	9.89739E-01	1.45612E-02	1.14674E+00	1.79309E+00	70.000
c90100	pu-d	w	4.700	9.86307E-01	1.39891E-02	1.01217E+00	1.69297E+00	84.000
c10010	pu-d	u--235	0.660	9.99326E-01	4.52963E-03	1.08553E+00	1.74508E+00	41.000
c11010	pu-d	u--238	1.930	9.92304E-01	6.22187E-03	1.22556E+00	1.90716E+00	44.000
c12010	pu-d	u--238	6.740	9.96307E-01	1.75296E-02	1.27721E+00	1.91966E+00	96.000

c13010	pu-d	u	4.130	9.96192E-01	1.03626E-02	1.24167E+00	1.92082E+00	92.000
c14010	pu-d	u	19.600	9.96127E-01	7.23795E-02	1.33102E+00	1.98171E+00	424.000
c10100	u--233			9.92621E-01	3.21781E-03	1.04182E+00	1.67076E+00	31.000
c20100	u--233	be	2.050	9.98352E-01	6.02942E-03	9.75421E-01	1.65853E+00	55.000
c30100	u--233	be	4.200	9.97651E-01	1.96502E-02	9.26651E-01	1.64984E+00	84.000
c40100	u--233	w	2.440	9.94453E-01	6.33914E-03	9.07984E-01	1.54268E+00	48.000
c50100	u--233	w	5.790	9.95594E-01	1.51061E-02	8.28507E-01	1.49093E+00	89.000
c60100	u--233	u--235	1.210	9.97302E-01	3.70799E-03	1.00492E+00	1.63540E+00	43.000
c70100	u--233	u--235	1.980	1.00102E+00	4.09850E-03	9.68922E-01	1.62458E+00	42.000
c80100	u--233	u--235	4.820	1.00269E+00	5.45698E-03	8.56758E-01	1.53686E+00	48.000
c90100	u--233	u	2.300	1.00084E+00	5.66585E-03	1.02412E+00	1.67370E+00	43.000
c10010	u--233	u	5.310	1.00387E+00	1.12457E-02	1.03701E+00	1.70311E+00	74.000
c11010	u--233	u	19.910	9.99824E-01	6.69586E-02	1.13604E+00	1.79585E+00	427.000
c001	be	1.27		9.84956E-01	7.92590E-03	7.66181E-01	1.50254E+00	63.000
c002	be	2.54		9.89349E-01	1.19728E-02	7.31254E-01	1.49590E+00	81.000
c003	c	1.27		9.97132E-01	7.52245E-03	7.80325E-01	1.45285E+00	60.000
c004	c	2.54		9.84808E-01	9.54820E-03	7.44669E-01	1.43082E+00	53.000
c005	mg	1.27		9.80226E-01	7.26601E-03	7.85811E-01	1.48546E+00	46.000
c006	mg	2.54		9.85306E-01	8.74325E-03	7.57234E-01	1.45128E+00	58.000
c007	al	1.27		9.84040E-01	7.30325E-03	7.89344E-01	1.48183E+00	49.000
c008	al	2.54		9.82376E-01	8.82979E-03	7.61154E-01	1.43757E+00	63.000
c009	ti	1.27		9.86470E-01	7.25457E-03	7.90573E-01	1.47590E+00	48.000
c010	ti	2.54		9.85082E-01	8.69680E-03	7.76932E-01	1.48943E+00	56.000
c011	fe	1.27		9.91683E-01	7.38926E-03	7.74572E-01	1.46834E+00	54.000
c012	fe	2.54		9.85022E-01	9.12293E-03	7.64133E-01	1.44124E+00	68.000
c013	ni	1.27		9.89787E-01	7.60197E-03	7.59230E-01	1.43730E+00	67.000
c014	ni	2.54		9.92284E-01	9.59394E-03	7.42961E-01	1.41780E+00	64.000
c015	cu	1.27		9.94020E-01	7.67577E-03	7.50467E-01	1.44277E+00	53.000
c016	cu	2.54		9.93251E-01	9.88651E-03	7.11314E-01	1.39312E+00	61.000
c017	mo	1.27		9.99787E-01	7.83534E-03	7.39537E-01	1.44241E+00	68.000
c018	mo	2.54		1.00349E+00	1.01786E-02	6.93400E-01	1.39208E+00	79.000
c019	mo-alloyh			9.98147E-01	7.64916E-03	7.27318E-01	1.40612E+00	54.000
c020	w	1.27		9.79927E-01	7.73420E-03	7.41982E-01	1.46242E+00	55.000
c021	w	2.54		9.78124E-01	1.03006E-02	7.08306E-01	1.40689E+00	82.000
c10100	u--235			9.87672E-01	6.00832E-03	8.02186E-01	1.49004E+00	56.000
c20100	u--235			1.00047E+00	6.01213E-03	8.09337E-01	1.51527E+00	53.000
c30100	u--235			9.97600E-01	6.32936E-03	8.02424E-01	1.48561E+00	44.000
c40100	u--235	be	2.222	9.92053E-01	1.01476E-02	7.28594E-01	1.45967E+00	71.000
c50100	u--235	be	3.260	9.91169E-01	1.54182E-02	7.02254E-01	1.47861E+00	89.000
c60100	u--235	be	4.710	9.93151E-01	4.04487E-02	6.79967E-01	1.49930E+00	89.000
c70100	u--235	be	5.440	9.88441E-01	8.70853E-02	6.86548E-01	1.50238E+00	137.000
c80100	u--235	be	9.270	9.96735E-01	3.22728E+00	6.05843E-01	1.49997E+00	220.000
c90100	u--235	be	11.790	9.96449E-01	1.20894E+01	5.67320E-01	1.49358E+00	328.000
c10010	u--235	c	10.160	9.93185E-01	6.12162E-02	6.63068E-01	1.33192E+00	101.000
c11010	u--235	c	15.240	9.83354E-01	8.31071E-01	6.25795E-01	1.29026E+00	214.000
c12010	u--235	ni	4.940	9.91062E-01	1.38437E-02	7.20833E-01	1.38303E+00	88.000
c13010	u--235	cu	5.030	1.00006E+00	1.48053E-02	6.60191E-01	1.36246E+00	92.000
c14010	u--235	cu	10.560	1.00144E+00	3.36150E-02	6.11069E-01	1.28480E+00	122.000
c15010	u--235	w	5.080	9.95629E-01	1.70971E-02	6.36561E-01	1.36370E+00	94.000
c16010	u--235	w	10.160	9.95321E-01	4.10885E-02	5.84957E-01	1.30232E+00	169.000
c17010	u--235	pb	8.990	1.01657E+00	1.96904E-02	7.16326E-01	1.42830E+00	83.000
c18010	u--235	pb	17.220	1.01366E+00	4.47035E-02	6.80386E-01	1.40573E+00	149.000
c19010	u--235	u	1.760	9.94353E-01	7.86457E-03	8.05291E-01	1.52775E+00	55.000
c20010	u--235	u	4.470	1.00270E+00	1.28804E-02	8.03924E-01	1.55049E+00	76.000
c21010	u--235	u	9.960	9.98858E-01	2.98260E-02	8.46800E-01	1.61144E+00	173.000
c22010	u--235	u	18.010	9.92314E-01	6.24729E-02	8.60911E-01	1.63460E+00	315.000
Average				9.94258E-01	+/- 5.62928E-03	Total	6873.000	
Lowest				9.78124E-01	-1.61341E-02 (from Average)			
Highest				1.01657E+00	2.23119E-02 (from Average)			

The following table presents results for the same 68 fast critical assemblies discussed above, comparing results using the production (floor) version of TARTNP on a Cray-YMP, and TART95 on six different types of computers. The calculations were all run using the default 3 % allowable uncertainty in  $K_{eff}$ . The results show some statistical variation, as we would expect, but in no case are the differences anywhere near 3 %. At the bottom of the table is the average value of  $K_{eff}$  for the 68 assemblies on each computer, followed by the difference of these averages from an average of the average values over the seven sets of results. Here we see excellent agreement, with no apparent bias on any of the computers, and differences of the averages only in the fourth digit, two orders of magnitude below the 3 % allowable uncertainty.

Description	TARTNP Cray-YMP	TART95 Cray-YMP	TART95 HP	TART95 SUN	TART95 SGI	TART95 IBM-RSIC	TART95 IBM-PC
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## Chapter 6: Example Problems

c10100	pu-a	be	5.222	1.00136	0.99921	0.99052	0.99611	0.99052	0.99291	0.99291
c20100	pu-a	be	8.170	0.99804	0.99980	0.99985	1.00286	0.99985	0.99985	0.99985
c30100	pu-a	be	13.00	1.00161	0.99856	0.99935	1.00182	0.99935	1.00000	1.00000
c40100	pu-d			1.00024	1.00082	1.00265	1.00265	1.00265	1.00265	1.00265
c50100	pu-d	be	3.690	0.99793	0.99835	0.99674	0.99655	0.99674	0.99674	0.99674
c60100	pu-d	be	5.250	0.99810	0.99611	0.99779	0.99673	0.99779	0.99779	0.99779
c70100	pu-d	c	3.830	0.99775	0.99570	0.99461	0.99461	0.99461	0.99461	0.99461
c80100	pu-d	ti	8.000	0.98752	0.99250	0.98974	0.98974	0.98974	0.98974	0.98974
c90100	pu-d	w	4.700	0.99377	0.99371	0.98631	0.98631	0.98631	0.98631	0.98631
c10010	pu-d	u--235	0.660	1.00023	0.99819	0.99933	0.99933	0.99933	0.99933	0.99933
c11010	pu-d	u--238	1.930	0.99679	0.99300	0.99230	0.99230	0.99230	0.99230	0.99230
c12010	pu-d	u--238	6.740	0.99436	0.99316	0.99631	0.99631	0.99631	0.99631	0.99631
c13010	pu-d	u	4.130	1.00050	0.99954	0.99619	0.99619	0.99619	0.99619	0.99619
c14010	pu-d	u	19.60	0.99538	0.99454	0.99613	0.99613	0.99613	0.99613	0.99613
c10100	u--233			0.99688	0.99107	0.99262	0.99262	0.99262	0.99262	0.99262
c20100	u--233	be	2.050	0.99856	1.00159	0.99835	0.99835	0.99835	0.99835	0.99835
c30100	u--233	be	4.200	0.99874	1.00221	0.99765	0.99959	0.99765	0.99765	0.99765
c40100	u--233	w	2.440	0.99967	0.99405	0.99445	0.99445	0.99445	0.99445	0.99445
c50100	u--233	w	5.790	0.99677	0.99549	0.99559	0.99559	0.99559	0.99559	0.99559
c60100	u--233	u--235	1.210	1.00130	1.00045	0.99730	0.99730	0.99730	0.99730	0.99730
c70100	u--233	u--235	1.980	1.00284	1.00146	1.00102	1.00102	1.00102	1.00102	1.00102
c80100	u--233	u--235	4.820	1.00563	1.00536	1.00269	1.00269	1.00269	1.00374	1.00374
c90100	u--233	u	2.300	1.00096	1.00016	1.00084	1.00084	1.00084	1.00084	1.00084
c10010	u--233	u	5.310	0.99608	1.00185	1.00387	1.00387	1.00387	1.00387	1.00387
c11010	u--233	u	19.91	1.00146	1.00335	0.99982	0.99982	0.99982	0.99982	0.99982
c001	be		1.27	0.99140	0.98541	0.98496	0.98496	0.98496	0.98496	0.98496
c002	be		2.54	0.98513	0.98623	0.98935	0.98935	0.98935	0.98935	0.98935
c003	c		1.27	0.99185	0.99355	0.99713	0.99713	0.99713	0.99713	0.99713
c004	c		2.54	0.99116	0.99230	0.98481	0.98481	0.98481	0.98481	0.98481
c005	mg		1.27	0.98429	0.98354	0.98023	0.98023	0.98023	0.98023	0.98023
c006	mg		2.54	0.98548	0.98762	0.98531	0.98531	0.98531	0.98531	0.98531
c007	al		1.27	0.98324	0.97631	0.98404	0.98404	0.98404	0.98404	0.98404
c008	al		2.54	0.98613	0.98334	0.98238	0.98238	0.98238	0.98238	0.98238
c009	ti		1.27	0.98214	0.98151	0.98647	0.98647	0.98647	0.98647	0.98647
c010	ti		2.54	0.98976	0.98429	0.98508	0.98508	0.98508	0.98508	0.98508
c011	fe		1.27	0.99314	0.99256	0.99168	0.99168	0.99168	0.99168	0.99168
c012	fe		2.54	0.98666	0.98236	0.98502	0.98502	0.98502	0.98502	0.98502
c013	ni		1.27	0.98362	0.98684	0.98979	0.98979	0.98979	0.98979	0.98979
c014	ni		2.54	0.98822	0.98646	0.99228	0.99228	0.99228	0.99228	0.99228
c015	cu		1.27	0.98929	0.98564	0.99402	0.99402	0.99402	0.99402	0.99402
c016	cu		2.54	0.99478	0.99520	0.99325	0.99325	0.99325	0.99325	0.99325
c017	mo		1.27	0.99931	0.99291	0.99979	0.99979	0.99979	0.99979	0.99979
c018	mo		2.54	1.00853	1.00774	1.00349	1.00349	1.00349	1.00349	1.00349
c019	mo-alloyh			0.99435	0.99671	0.99815	0.99815	0.99815	0.99815	0.99815
c020	w		1.27	0.98011	0.97990	0.97993	0.97993	0.97993	0.97993	0.97993
c021	w		2.54	0.98242	0.98141	0.97812	0.97812	0.97812	0.97812	0.97812
c10100	u--235			0.99384	0.99127	0.98767	0.98767	0.98767	0.98767	0.98767
c20100	u--235			1.00213	1.00140	1.00047	1.00047	1.00047	1.00047	1.00047
c30100	u--235			0.99634	0.99101	0.99760	0.99760	0.99760	0.99760	0.99760
c40100	u--235	be	2.222	0.98990	0.99267	0.99205	0.99205	0.99205	0.99205	0.99205
c50100	u--235	be	3.260	0.99593	0.99162	0.99117	0.99117	0.99117	0.99117	0.99117
c60100	u--235	be	4.710	0.99828	0.99559	0.99315	0.99473	0.99315	0.99315	0.99315
c70100	u--235	be	5.440	0.99265	0.99698	0.98844	0.99376	0.98844	0.98844	0.98844
c80100	u--235	be	9.270	0.99362	0.99281	0.99673	0.99500	0.99673	0.99673	0.99673
c90100	u--235	be	11.79	0.99358	0.99170	0.99645	0.99512	0.99645	0.99645	0.99645
c10010	u--235	c	10.16	0.99490	0.99531	0.99318	0.99417	0.99318	0.99318	0.99318
c11010	u--235	c	15.24	0.98799	0.98761	0.98335	0.98757	0.98335	0.98335	0.98335
c12010	u--235	ni	4.940	0.98832	0.99091	0.99106	0.99106	0.99106	0.99106	0.99106
c13010	u--235	cu	5.030	0.99559	1.00001	1.00006	1.00006	0.99899	0.99899	0.99899
c14010	u--235	cu	10.56	1.00216	0.99889	1.00144	1.00144	1.00144	1.00144	1.00144
c15010	u--235	w	5.080	0.99721	0.99305	0.99563	0.99563	0.99563	0.99563	0.99563
c16010	u--235	w	10.16	0.99726	0.99873	0.99532	0.99532	0.99532	0.99532	0.99532
c17010	u--235	pb	8.990	1.01844	1.01859	1.01657	1.01657	1.01657	1.01657	1.01657
c18010	u--235	pb	17.22	1.01279	1.01228	1.01366	1.01366	1.01460	1.01460	1.01460
c19010	u--235	u	1.760	0.99598	0.99983	0.99435	0.99435	0.99435	0.99435	0.99435
c20010	u--235	u	4.470	1.00048	0.99997	1.00270	1.00270	1.00270	1.00270	1.00270
c21010	u--235	u	9.960	0.99737	0.99184	0.99886	0.99886	0.99886	0.99720	0.99720
c22010	u--235	u	18.01	0.99336	0.99738	0.99231	0.99231	0.99231	0.99210	0.99210
Averages				0.99517	0.99443	0.99426	0.99454	0.99426	0.99429	0.99429
Differences from Averages				0.00071	-0.00003	-0.00020	0.00008	-0.00020	-0.00017	-0.00017

The following table again shows results for the 68 assemblies calculated with TARTNP and TART95, however in this case rather than presenting  $K_{\text{eff}}$ , for each of the 68 cases we have defined an average over the seven different results, and this table presents the difference between the calculated  $K_{\text{eff}}$  on each computer and the  $K_{\text{eff}}$  averaged over the

## Chapter 6: Example Problems

seven results. From these results we can more clearly see that in no case are the differences anywhere near 1 %, let alone near the allowable 3 % uncertainty.

Description			TARTNP	TART95	TART95	TART95	TART95	TART95	TART95	
			Cray-YMP	Cray-YMP	HP	SUN	SGI	IBM-RSIC	IBM-PC	
=====										
c10100	pu-a	be	5.222	0.00657	0.00442	-0.00427	0.00132	-0.00427	-0.00188	-0.00188
c20100	pu-a	be	8.170	-0.00197	-0.00022	-0.00016	0.00284	-0.00016	-0.00016	-0.00016
c30100	pu-a	be	13.00	0.00151	-0.00154	-0.00075	0.00172	-0.00075	-0.00010	-0.00010
c40100	pu-d			-0.00180	-0.00122	0.00061	0.00061	0.00061	0.00061	0.00061
c50100	pu-d	be	3.690	0.00081	0.00124	-0.00037	-0.00056	-0.00037	-0.00037	-0.00037
c60100	pu-d	be	5.250	0.00066	-0.00133	0.00035	-0.00071	0.00035	0.00035	0.00035
c70100	pu-d	c	3.830	0.00254	0.00048	-0.00060	-0.00060	-0.00060	-0.00060	-0.00060
c80100	pu-d	ti	8.000	-0.00230	0.00268	-0.00008	-0.00008	-0.00008	-0.00008	-0.00008
c90100	pu-d	w	4.700	0.00534	0.00528	-0.00212	-0.00212	-0.00212	-0.00212	-0.00212
c10010	pu-d	u--235	0.660	0.00094	-0.00110	0.00003	0.00003	0.00003	0.00003	0.00003
c11010	pu-d	u--238	1.930	0.00375	-0.00004	-0.00074	-0.00074	-0.00074	-0.00074	-0.00074
c12010	pu-d	u--238	6.740	-0.00122	-0.00242	0.00073	0.00073	0.00073	0.00073	0.00073
c13010	pu-d	u	4.130	0.00321	0.00225	-0.00109	-0.00109	-0.00109	-0.00109	-0.00109
c14010	pu-d	u	19.60	-0.00041	-0.00126	0.00033	0.00033	0.00033	0.00033	0.00033
c10100	u--233			0.00387	-0.00193	-0.00039	-0.00039	-0.00039	-0.00039	-0.00039
c20100	u--233	be	2.050	-0.00028	0.00275	-0.00049	-0.00049	-0.00049	-0.00049	-0.00049
c30100	u--233	be	4.200	0.00001	0.00348	-0.00108	0.00085	-0.00108	-0.00108	-0.00108
c40100	u--233	w	2.440	0.00453	-0.00109	-0.00069	-0.00069	-0.00069	-0.00069	-0.00069
c50100	u--233	w	5.790	0.00102	-0.00026	-0.00015	-0.00015	-0.00015	-0.00015	-0.00015
c60100	u--233	u--235	1.210	0.00298	0.00213	-0.00102	-0.00102	-0.00102	-0.00102	-0.00102
c70100	u--233	u--235	1.980	0.00150	0.00012	-0.00032	-0.00032	-0.00032	-0.00032	-0.00032
c80100	u--233	u--235	4.820	0.00184	0.00157	-0.00110	-0.00110	-0.00110	-0.00005	-0.00005
c90100	u--233	u	2.300	0.00020	-0.00060	0.00008	0.00008	0.00008	0.00008	0.00008
c10010	u--233	u	5.310	-0.00638	-0.00062	0.00140	0.00140	0.00140	0.00140	0.00140
c11010	u--233	u	19.91	0.00090	0.00279	-0.00074	-0.00074	-0.00074	-0.00074	-0.00074
c001	be	1.27		0.00546	-0.00053	-0.00099	-0.00099	-0.00099	-0.00099	-0.00099
c002	be	2.54		-0.00317	-0.00207	0.00105	0.00105	0.00105	0.00105	0.00105
c003	c	1.27		-0.00402	-0.00231	0.00127	0.00127	0.00127	0.00127	0.00127
c004	c	2.54		0.00438	0.00551	-0.00198	-0.00198	-0.00198	-0.00198	-0.00198
c005	mg	1.27		0.00301	0.00226	-0.00105	-0.00105	-0.00105	-0.00105	-0.00105
c006	mg	2.54		-0.00018	0.00196	-0.00035	-0.00035	-0.00035	-0.00035	-0.00035
c007	al	1.27		0.00042	-0.00651	0.00122	0.00122	0.00122	0.00122	0.00122
c008	al	2.54		0.00308	0.00029	-0.00067	-0.00067	-0.00067	-0.00067	-0.00067
c009	ti	1.27		-0.00300	-0.00363	0.00133	0.00133	0.00133	0.00133	0.00133
c010	ti	2.54		0.00412	-0.00135	-0.00055	-0.00055	-0.00055	-0.00055	-0.00055
c011	fe	1.27		0.00113	0.00054	-0.00033	-0.00033	-0.00033	-0.00033	-0.00033
c012	fe	2.54		0.00178	-0.00251	0.00015	0.00015	0.00015	0.00015	0.00015
c013	ni	1.27		-0.00487	-0.00165	0.00130	0.00130	0.00130	0.00130	0.00130
c014	ni	2.54		-0.00266	-0.00441	0.00141	0.00141	0.00141	0.00141	0.00141
c015	cu	1.27		-0.00286	-0.00651	0.00187	0.00187	0.00187	0.00187	0.00187
c016	cu	2.54		0.00104	0.00145	-0.00050	-0.00050	-0.00050	-0.00050	-0.00050
c017	mo	1.27		0.00057	-0.00583	0.00105	0.00105	0.00105	0.00105	0.00105
c018	mo	2.54		0.00371	0.00292	-0.00133	-0.00133	-0.00133	-0.00133	-0.00133
c019	mo-alloyh			-0.00305	-0.00069	0.00075	0.00075	0.00075	0.00075	0.00075
c020	w	1.27		0.00016	-0.00005	-0.00002	-0.00002	-0.00002	-0.00002	-0.00002
c021	w	2.54		0.00321	0.00220	-0.00108	-0.00108	-0.00108	-0.00108	-0.00108
c10100	u--235			0.00477	0.00221	-0.00140	-0.00140	-0.00140	-0.00140	-0.00140
c20100	u--235			0.00129	0.00056	-0.00037	-0.00037	-0.00037	-0.00037	-0.00037
c30100	u--235			-0.00014	-0.00547	0.00112	0.00112	0.00112	0.00112	0.00112
c40100	u--235	be	2.222	-0.00194	0.00084	0.00022	0.00022	0.00022	0.00022	0.00022
c50100	u--235	be	3.260	0.00422	-0.00010	-0.00055	-0.00193	-0.00055	-0.00055	-0.00055
c60100	u--235	be	4.710	0.00382	0.00114	-0.00131	0.00027	-0.00131	-0.00131	-0.00131
c70100	u--235	be	5.440	0.00163	0.00596	-0.00258	0.00273	-0.00258	-0.00258	-0.00258
c80100	u--235	be	9.270	-0.00186	-0.00267	0.00125	-0.00048	0.00125	0.00125	0.00125
c90100	u--235	be	11.79	-0.00159	-0.00347	0.00128	-0.00005	0.00128	0.00128	0.00128
c10010	u--235	c	10.16	0.00102	0.00143	-0.00069	0.00030	-0.00069	-0.00069	-0.00069
c11010	u--235	c	15.24	0.00276	0.00238	-0.00187	0.00235	-0.00187	-0.00187	-0.00187
c12010	u--235	ni	4.940	-0.00233	0.00026	0.00041	0.00041	0.00041	0.00041	0.00041
c13010	u--235	cu	5.030	-0.00351	0.00090	0.00095	0.00095	0.00095	-0.00012	-0.00012
c14010	u--235	cu	10.56	0.00098	-0.00229	0.00026	0.00026	0.00026	0.00026	0.00026
c15010	u--235	w	5.080	0.00172	-0.00244	0.00014	0.00014	0.00014	0.00014	0.00014
c16010	u--235	w	10.16	0.00117	0.00264	-0.00076	-0.00076	-0.00076	-0.00076	-0.00076
c17010	u--235	pb	8.990	0.00131	0.00146	-0.00056	-0.00056	-0.00056	-0.00056	-0.00056
c18010	u--235	pb	17.22	-0.00082	-0.00133	0.00005	0.00005	0.00005	0.00099	0.00099
c19010	u--235	u	1.760	0.00062	0.00446	-0.00102	-0.00102	-0.00102	-0.00102	-0.00102
c20010	u--235	u	4.470	-0.00151	-0.00202	0.00071	0.00071	0.00071	0.00071	0.00071
c21010	u--235	u	9.960	0.00020	-0.00533	0.00169	0.00169	0.00169	0.00003	0.00003
c22010	u--235	u	18.01	0.00024	0.00425	-0.00081	-0.00081	-0.00081	-0.00102	-0.00102
=====										
Averages			0.00071	-0.00003	-0.00021	0.00008	-0.00021	-0.00017	-0.00017	

There are a number of conclusions that can be reached based on the above results,

- 1) The results verify that at least for these problems the TARTNP and TART95 results are in excellent agreement - which was the primary reason for running these calculations.
- 2) Although not presented in the above results, we can say that for this set of 68 assemblies on a CRAY-YMP TART95 ran about 10 % faster than TARTNP. This illustrates that in converting TARTNP to create TART95 for use on a wide variety of computers, not only have we maintained the speed advantage of TART compared to other codes, we have improved it.
- 3) Of the results obtained using TART95 on six different computers, the fastest computer was the HP-350, not the Cray-YMP. TART95 on the HP-350 ran about 12 % faster than on the Cray-YMP. This illustrates how practical it is these days to use relatively inexpensive workstations instead of multi-million dollar central computers.
- 4) One interesting question is: On all of the workstations and IBM-PC exactly the same TART95 code was used, using exactly the same random number generator and initial random number seed. These are all 32 bit per word computers, so why aren't all of the results exactly the same? The differences have been traced to: a) differences in architecture as far as how the 32 bits per word are used to represent the exponent and mantissa, and how registers are used to perform arithmetic, and 2) more important is the use of different algorithms for functions, such as Exp, Log, Sin, Cos, etc. Remember that this is a Monte Carlo calculation and all you need is a small difference in one sampled result to send calculations off in completely different directions for the entire remainder of the calculation. Note, the two IBM results exactly agree; the results for IBM-RSIC and IBM-PC are exactly the same, indicating similar architecture and algorithms for functions. Similarly the HP and SGI results exactly agree; only the SUN results cannot be exactly matched to the results on another computer in this comparison.

### Slow Critical Assemblies

TART95 is distributed with an example problem involving a simple spherical, homogeneous mixture of water and uranium, for a number of different water to  $U^{235}$  ratios (the results are actually presented for various hydrogen to  $U^{235}$  ratios, as H/U Ratio). The problem only involves one spatial region containing the mixture. For each water to  $U^{235}$  ratio the radius of the spherical radius can be varied to make the system critical.

As distributed the results have not been optimized to produce exactly critical systems for each ratio. It has been left as an exercise for to user to try varying the radius of each system to make it critical, e.g., if the system presently has an effective multiplication ( $K_{eff}$ ) greater than unity try slightly reducing the radius and run the problem again; you will find that with just a few tries you can make the system critical. What does critical mean? Usually the uncertainty in measured critical systems is roughly 0.5 %, so that if you can get  $K_{eff}$  between about 0.995 and 1.005, you can stop; don't keep trying forever until you get exactly unity.

A second use of this example problem is to illustrate the effects of self-shielding and thermal scattering. The following table illustrates results obtained by performing the calculations four different ways. The columns in the table, from left to right, correspond to performing the calculations: 1) using the multi-band method to account for self-shielding (**sentl 20 1**), and thermal scattering (**sentl 39 1**), 2) NOT using the multi-band method, but using thermal scattering, 3) Using the multi-band method, but NOT thermal scattering, and 4) NOT using the multi-band method or thermal scattering. The **recommended** method is 1) with both the multi-band method to account for self-shielding and thermal scattering. Unfortunately, the **default** method is 4) without either the multi-band method or thermal scattering. Therefore, if you do not specify these options on input you will obtain the results corresponding to 4). As an exercise for users, you can try running this example with the same combinations of options described above.

H/U Ratio	Multi-Band Thermal 1)	Thermal 2)	Multi-Band 3)	4)
76	1.00225	0.91217	0.97698	
0.89233				
218	1.02157	0.98747	0.98052	
0.95041				
365	1.02821	1.02927	0.97640	
0.97524				
534	1.01645	1.00300	0.96578	
0.95956				
635	1.01863	1.01782	0.96072	
0.96424				
1037	1.01584	1.00946	0.97789	
0.97469				

As mentioned above, the results 1) have not been optimized to produce a  $K_{eff}$  of unity in all cases. In the results what we are interested in is not the absolute values of  $K_{eff}$ , but rather the changes in  $K_{eff}$  using different combinations of options. Another point to keep in mind is that we did not run these calculations to high precision, so that we should only consider large changes of  $K_{eff}$  to be real, rather than merely the effect of statistics.

If we compare results 1) and 2) we can see the effects of self-shielding, since this is the only difference between these two cases. Compared to the 1) results with self-shielding, the 2) results show that when the ratio of water to  $U^{235}$  is the smallest self-shielding has the greatest effect, and for the larger ratios it has almost no effect. This agrees with the

predictions of the narrow resonance and Bondarenko approximations, where a pure material will self-shield like  $1/\sigma_{\text{tot}}$  and an infinitely dilute amount of material in a mixture will be unshielded. Intermediate ratios of material will self-shield like  $1/[\sigma_{\text{tot}} + \sigma_0]$ , where  $\sigma_0$  varies from zero for a pure material to infinity for an infinitely dilute, unshielded case. In case 2), without self-shielding, the uranium cross sections will effectively be larger, which will have two effects. First, in terms of ratios of cross sections this effectively lowers the water to uranium ratio, decreasing slowing down and hardening the spectrum. Second, and more important, while all the uranium cross sections are increased this will be particularly true of the many, very narrow  $\text{U}^{238}$  capture resonances, leading to a large decrease in  $K_{\text{eff}}$ , with the largest decreases occurring for the smallest ratios; the effect is less for larger ratios and effectively disappears for the largest ratios.

If we compare results 1) and 3) we can see the effects of thermal scattering, since this is the only difference between these two cases. Compared to the 1) results with thermal scattering, the 3) results show that without thermal scattering  $K_{\text{eff}}$  is always smaller. With thermal scattering the neutrons will slow down and the slowest neutrons will go over into a Maxwellian distribution with a temperature corresponding to the temperature of the medium (in this case, room temperature). Without thermal scattering the neutrons will continue to slow down to the minimum energy of the TART95 multi-group structure and just pile up at this energy until they leak from the system or are absorbed, i.e., they will slow down to  $1.307\text{E-}09$  MeV ( $0.001307$  eV). The capture cross section of water is quite small, but it is increasing as  $1/V$  (one over the speed of the neutrons). Even though the water capture is quite small, by allowing the neutrons to go to much lower energies without thermal scattering, in all cases the capture in water is increased leading to a decrease in  $K_{\text{eff}}$ .

Finally comparing results 1) and 4) we can see the results obtained using the **recommended** and **default** input parameters. Compared to the 1) results the 4) results are in all case drastically smaller  $K_{\text{eff}}$ . Not only are the results 4) very poor, since they do not include the correct physics to model this problem, they are dangerous!!! Remember, what we are trying to do is predict the  $K_{\text{eff}}$  of mixtures of water and uranium in various ratios. If we were to believe results 4) we would conclude that for the mixtures and radii that we are using all of these cases would be safely subcritical with  $K_{\text{eff}}$  significantly less than unity. But in fact results 1) indicate that all of these systems really have a  $K_{\text{eff}}$  of unity or greater. Therefore, if anyone believed the results 4) and actually built a system corresponding to one of these cases, assuming it would be safely subcritical, the result could be a criticality accident.

The last use of this example problem that will be left as an exercise for users, is to vary the temperature of the medium to determine whether or not these systems are nuclear safe as far as potential temperature increases. If a system becomes slightly super critical, with  $K_{\text{eff}}$  greater than unity, the flux and power generated by the system will increase, as will the temperature. If the system is nuclear safe for temperature increases, an increase in temperature will tend to decrease  $K_{\text{eff}}$  and eventually return the system to a stable situation. Try increasing the temperature of the medium from its present value of  $2.53\text{E-}08$

MeV (room temperature) to twice this value, and re-run this example problem. Again, here we are looking for changes in  $K_{\text{eff}}$  rather than absolute values. What's important is: does increasing the temperature result in a smaller  $K_{\text{eff}}$ ?

### Hexagonal Repeating Cells

In order to demonstrate the use of general planes and reflecting zones, TART95 is distributed with a simple hexagonal infinitely repeating cell problem. This problem does not correspond to any real cell, and is included only to familiarize the user with these TART95 input options.

In this problem there is a cylindrical fuel region, surrounded by a cylindrical air gap, which is in turn surrounded by a cylindrical cladding. Water is assumed to fill the entire region outside the cladding. The cylindrical regions are assumed to be in a hexagonal, infinitely repeating array, and they are assumed to be infinitely long. This entire infinitely repeating system can be defined using: 1) three cylindrical surfaces whose axis lies along the Z axis, to define the fuel, air gap and cladding regions, 2) six planes perpendicular to the X, Y plane to define the six edges of the hexagonal, 3) two planes perpendicular to the Z plane to define the Z ends of the cell = a total of only 11 surfaces. The geometry is made infinite in extent by defining eight reflecting zones: six reflecting zones are outside the six edges of the hexagon perpendicular of the X, Y plane, and two reflecting zones are outside the ends of the cylinder perpendicular to the Z plane. In addition there are zones for fuel, air gap, cladding and water = a total of only 12 zones. The result is a single hexagonal cell that simulates an infinitely repeating array of hexagonal cells that exactly fill all space with no overlap, e.g., you cannot do this with cylindrical cells since they cannot fill all space without overlap. The six reflecting zones on the six sides of the hexagon perpendicular to the X, Y plane, exactly simulate neutrons entering an adjacent cell of the array. The two reflecting zones at the ends of the cell perpendicular to the Z plane, exactly simulate neutrons entering a portion of the cell outside the defined ends of the cell.

Below is the entire input deck for this problem. In this case we started from an existing TART95 input deck, which already included most of the input options that we need, and modified it to define a new geometry, and materials. That all that was required to produce a new input deck. Since much of the following deck is exactly the same as the input deck described above, here we will only discuss how it has been modified. There are minor changes at the beginning of the deck to describe the problem.

```
* =====
name      c10100  hexagonal cell
box g32   hexagonal cell
* =====
*
* TART Input Deck Generated Using Program TARTVIEW (93-1)
*
* =====
*
```

## Chapter 6: Example Problems

```
* PROBLEM: AN INFINITELY REPEATING ARRAY OF HEXAGON
* CELLS OF URANIUM, AIR GAP, CLADDING, SURROUNDED BY WATER
*
* =====
*
* Criticality Problem
* 1) Number of Settle Cycles (should be 10 to 20)
* 2) Number of Repetitions after Settling (usually 1)
* 3) % Standard Deviation to Stop Repetition (default 3.0)
* 4) Time Step (shakes) - Only for Dynamic K Calculation
*
* =====
critcalc 15 1 3.000
* =====
*
* Surface Definitions - For All Surfaces the first 2
* fields are,
* 1) A surface type keyword
* 2) An Assigned Surface Number
* For Cylinders
* 3) Radius and Position - Radius Z0 X0 Y0
* For Aligned Planes
* 3) Position - X0, Y0, OR Z0
* For General Planes
* 3) Position and Direction Cosines - X0 Y0 Z0 A B G
*
```

In the following section we defined 3 cylinders whose axes are parallel to the z axis (**cylz**), of radius 1.9, 2.0, and 2.2 cm, all centered on the origin (the optional center of the cylinder has not been explicitly defined, since it defaults to the origin). These 3 cylinders will be used to define the fuel, air gap and cladding zones.

There are 6 planes used to define the 6 edges of the hexagonal cell. All of these planes are parallel to the z axis, so we need only define x and y coordinates. The 6 planes include 2 **xplane** perpendicular to the x axis, and 4 general planes (**gpl**) perpendicular to the x, y plane.

The last 2 surfaces are perpendicular to the z axis (**zplane**) used to define the reflecting ends of the cell. Note, since these will be used to define reflecting zones, the position of the planes are completely arbitrary; you will get the same results regardless of where they are placed. In this case we have defined them to be at -10 and +10 cm. This was done to insure that they include the origin, where we will place our initial source guess, and symmetric about the initial source guess; the symmetry isn't necessary, but it makes physical sense to speed symmetric convergence toward the final distribution.

```
* =====
* cylindrical fuel - air gap - cladding surfaces
cylz      1      1.9
```

```

cylz      2    2.0
cylz      3    2.2
* hexagonal bounding planes
xplane    4   -3.400
xplane    5    3.400
gpl       6    0.0  4.0  0.0   1.0  1.773  0.0
gpl       7    0.0  4.0  0.0  -1.0  1.773  0.0
gpl       8    0.0 -4.0  0.0   1.0  1.773  0.0
gpl       9    0.0 -4.0  0.0  -1.0  1.773  0.0
* ends of cell along z axis
zplane   10   -10.0
zplane   11    10.0

```

In the next section we define our zones using combinations of the above defined surfaces. For example, zone 1, the fuel, is defined to be inside the inner most cylinder (surface 1), above the z plane at -10 cm (surface 10), and below the z plane at 10 cm (surface 11). Only 3 bounding surfaces are required to define this zone. The next 2 zones require 4 bounding surfaces. Because they are located between pairs of cylinders (an annulus), we have to identify each as outside one cylinder and inside another cylinder. Both are located above the z plane at -10 cm and below the z plane at 10 cm. These 2 zones define the air gap and cladding, respectively. Zone 4 defines the remaining inside of the hexagon cell; the water region. This region is located outside the largest radius cylinder (surface 3), and inside an hexagon region defined by the 6 bounding planes. It is also bounded by the z planes at -10 and 10 cm. That completes the definition of the inside of the cell.

The outside of the cell is defined by 8 zones; later these will be defined to all be reflecting zones. Each of the 8 zones is defined by a single bounding surface. There are 6 zones, one outside each of the 6 edges of the hexagon in the x, y plane, and 2 zones, one on either end of the cell in the z plane. Note, in this case for simplicity we have allowed the outside zones to overlap. Since no neutrons can enter these zones, this does not present a problem for this calculation. If you use **TARTCHEK** to see this geometry and run the overlap option, it will identify the overlapped regions. You should realize that there is no problem for overlapping outside, non-re-entrant zones.

```

* =====
*
* Zone Definitions
* 1) Zone Number
* 2) Bounding Surface Number and Sign of Vector Normal
*    to the Surface for Particles Leaving the Zone.
*    2) Is repeated to include all Bounding Surfaces.
*
* =====
* All regions inside cell are bound by surfaces 10, 11
* in the z plane = the ends of the cell = Reflectors
* fuel
jb      1      1          -10  11
* air gap

```

```

jb      2   -1    2                -10  11
* cladding
jb      3   -2    3                -10  11
* the rest of cell
jb      4   -3   -4   5   6   7  -8  -9  -10  11
* outside of cell
jb      5    4
jb      6   -5
jb      7   -6
jb      8   -7
jb      9    8
jb     10    9
* ends of cell along z axis
jb     11                10
jb     12               -11

```

In the next two sections we first define materials and then assign them to zones. In this case we define four materials: uranium, aluminum, water and air. Note, since we are using **matl** input, it is easy to define something like water (material 3). In this case we merely say it has a density of 1 gram/cc, and atom ratios of 2 atoms of hydrogen to 1 atom of oxygen; TART95 will correctly define the atoms/cc of each to obtain the correct overall density.

```

* =====
*
* Material Definitions
* 1) Material Number
* 2) Density (grams/cc)
* 3) Atom %
* 4) Isotope I.D. (ZZZAAA), e.g., 92238 for U-238
*    3) and 4) can be repeated in pairs to define
*    composite materials
*
* =====
* uranium - 0.932 % u-235
matl  1  1.91400e+01  0.932 92235   99.068  92238
* aluminum
matl  2  2.70000e+00  1.00000e+02  13027
* water
matl  3  1.00000e+00  2.0  1001  1.0  8016
* air
matl  4  1.0e-08  1.0  7014

```

In the following section we assign the above defined materials to specific zones. Note, you don't have to assign material 1 to zone 1, and material 2 to zone 2, etc. There is no direct correspondence between the material numbers defined above and their assignments to specific zones below. In general for a complicated problem involving many zones each material will be assigned to many zones, e.g., we could have **matz 3 4 7 19 21 3 12**, to assign material number 3 to zones 4, 7, 19, 21, 3, and 12. In the simplest case, below,

note we assign material 1(fuel) to zone 1 (fuel), material 4 (air) to zone 2 (the air gap), and material 2 (aluminum) to zone 3 (the cladding) and material 3 (water) to zone 4 (the water region).

As a convenience to insure that all zone numbers appear in the following section we have also included the definition of all reflection zones in this section. If we mistakenly leave any zone empty (i.e., do not assign a material to it or make it reflecting) the problem will still run, but TART95 will consider the zone to be completely absorbing; an exterior, non-re-entrant zone. For example, if we omit zones 11 and 12 in the list of reflecting zones, rather than being infinitely long along the z axis, the cell will be finite in length extending from -10 to 10 cm along the z axis. Similarly if we had omitted one of the 6 sides of the hexagon (zones 5 through 10), rather than reflecting, every neutron crossing the surface and entering the zone with no material assigned to it would be absorbed. By including all zones assigned materials and all reflecting zones in this sections it allows us to simply count and see if all the zone numbers being used in the problem are defined below. An easier way to be this is to use **TARTCHEK** and run the flood or holes option to determine if any zones are empty.

```
* =====
*
* Assignment of Materials to Zones
* 1) Material Number
* 2) Zone Containing Material - Can be Repeated
*
* =====
* fuel
matz  1    1
* air
matz  4    2
* aluminum
matz  2    3
* water
matz  3    4
* reflectors
reflgp 5  6  7  8  9 10 11 12
```

The remaining sections are so similar to the input parameters already discussed above that we will not discuss them in detail here. We will merely mention that in this case we use the simplest possible guess for the initial spatial distribution of neutrons, namely a point source at the origin.

```
* =====
*
* Definition of Sources
*
* A simple point source at the origin.
* Since it is not defined the spectrum will by default
* be an isotropic, neutron induced fission spectrum
```

```

*
* =====
source1  1  0.0  0.0  0.0
* =====
*
* Definition of Running Conditions and Output Edit Options
*
* =====
* 1) Transport (neutrons and/or photons) (0)
sent1  1  1
* 2) Number of Samples (20)
sent1  2  2000
* 3) Histories per sample (5000)
sent1  3  1000
* 8) Neutron Minimum Energy (2.53e-8 MeV)
sent1  8  1.30700e-09
* 20) Multi-Band for Self-Shielding (0)
sent1  20  1
* 39) Thermal Scattering Sentinels (0)
sent1  39  1
* Thermal Scattering Temperature in All Zones (2.53e-8 MeV)
emin  2.53000e-08  1 thru  4
end

```

Figs. 1 and 2 (see the appendix) are **TARTCHEK** plots for this problem. Fig. 1 illustrates the profile of the hexagonal cell in the x, y plane perpendicular to the z axis. In this case we have used the **TARTCHEK** option to show the material in each zone. The central fuel region is shown in red, the air gap in magenta (difficult to distinguish from the red fuel on this plot, but easy to see on a computer screen), the cladding is in green, the water in blue, and the outside of the cell is black, indicating no material has been assigned to this outside region. By comparing this figure to the above input definitions you should be able to better understand the geometry involved in this problem.

Fig. 2 illustrates the results obtained using the **TARTCHEK** 2-D Overlap option. This figure indicates that the reflecting outside zones overlap, and lists the numbers of the zones that overlap. Generally this list of overlapping zones will allow you to quickly identify which zone definitions you must change to eliminate overlap. In this case, since only exterior, reflecting zones are overlapped we need not worry, since particles cannot really enter these zones.

In this case even though we have made what might seem to be the worst possible initial guess for the spatial distribution of neutrons (all at the origin), TART95 can iterate to convergence and solve this problem extremely quickly. The below results, run on an HP-350, illustrate that this problem can be solved in 1 minute (60 seconds). Even on an IBM-PC it only takes a handful of minutes.

Note, in this case since all bounding zones are reflectors, the output indicates no leakage from the cell. If we had made an error in our input and not made all bounding zones

reflecting, we should have immediately noticed the error based on the output results; immediately, since these results also appear on our screen.

```

-----
Settle cycle k values
-----
Cycle   Keff-Batch   Keff-Average   Absorbed      Leaked      Time Used
          (Seconds)
-----
      1     1.021518     1.021518     .978935     .000000     3.00
      2     1.000909     1.011213     .988827     .000000     4.00
      3     .988851     1.003759     .995943     .000000     5.00
      .
      .
      .
     14     1.009243     1.003909     .995944     .000000     19.00
     15     .995729     1.003363     .996479     .000000     20.00
-----
Batch   Keff-Batch   Keff-Average   Absorbed      Leaked      Time Used
          (Seconds)
-----
      1     .981121     .981121     1.019242     .000000     21.00
      2     1.036502     1.008812     .991361     .000000     23.00
      .
      .
     10     1.020968     1.002499     .997440     .000000     32.00
     11     .993386     1.001670     .998266     .000000     34.00
          1.001670+/- .005763 s.d. 25. % Completed
     12     .971779     .999179     1.000743     .000000     35.00
      .
      .
     26     .988350     .999892     1.000026     .000000     52.00
     27     .991433     .999578     1.000341     .000000     54.00
     28     .996607     .999472     1.000449     .000000     55.00
          .999472+/- .002905 s.d. 100. % Completed
     29     1.006483     .999714     1.000212     .000000     56.00
     30     1.010435     1.000071     .999874     .000000     57.00
     31     .967308     .999015     1.000906     .000000     59.00
     32     1.019213     .999646     1.000284     .000000     60.00
-----

```

```

Expected k = 9.99646E-01 Std dev = 2.835E-03 (Recommended)
Actual k = 1.00512E+00 Std dev = 6.069E-03
Mixed k = 9.93413E-01 Std dev = 6.497E-03
ekbar = 9.99716E-01

```

All times are in microseconds.

```

Removal lifetime      4.48548E+01. Std dev 3.923E-01
Absorption lifetime  4.48548E+01
Production lifetime   4.98618E+01. Std dev 5.644E-01
Time to prod. event   2.08491E+01. Std dev 3.030E-01
Time to capt. event   2.40057E+01. Std dev 2.910E-01
Time to leakage       .00000E+00. Std dev .000E+00
Approx.(alpha)/usec  -7.89573E-06 (Expected k - 1)/(Removal lifetime)

Normalized leakage      .00000E+00
Normalized census      1.00513E+00
Normalized net collision gain 5.12500E-03
Normalized absorption   5.89750E-01

```

Another use of this example problem is to illustrate the reliability, or repeatability, of the results. In this example the **critcalc** input line says to run the calculations to an accuracy of 3 %. What TARTNP and TART95 do is to decrease the input value by a factor of 10, and perform the calculation using more and more batches until the standard deviation of  $K_{\text{eff}}$  summed over all batch is within 1/10 of the per-cent accuracy input on the **critcalc** line. Unfortunately over the years users have learned what TARTNP is doing internally and they now assume that the results are ALWAYS accurate to within 1/10 of the per-cent accuracy requested by input. What TARTNP is doing is trying to guarantee that the accuracy of the result is well within the accuracy requested by input, but there is no guarantee that in one single calculation the results are ALWAYS within 1/10 of this accuracy. In order to illustrate this we can use this example problem. The original **critcalc** input line reads **critcalc 15 1 3.0**, run a criticality problem using 15 settle cycles, repeat (do) the calculation only once, calculating  $K_{\text{eff}}$  to within an accuracy of 3 %. We can modify this line to read **critcalc 15 20 3.0**, which will repeat the calculation 20 times. When we run this calculation we will be able to determine how reliable, or repeatable, the results are.

The following results were produced on a SUN SPARC-2. The table was prepared using the **CRITEDIT** utility code; this code reads the entire TART95 output file, extracts information that characterizes each problem, and producing a table defining the average and spread in the results.

```

=====
Crit.  Fuel   Reflector  Expected K  Removal  Median  Average  Seconds
#      #      #          #          Lifetime Energy   Energy
          (Microsec.) (MeV)   (MeV)
=====
c10100 hexagonacell 1.01146E+00 4.48690E+01 3.07173E-08 4.12684E-01 189.000
c10100 hexagonacell 9.99646E-01 4.48548E+01 3.18998E-08 4.55825E-01 232.000
c10100 hexagonacell 9.97630E-01 4.51854E+01 3.15175E-08 4.59778E-01 346.000
c10100 hexagonacell 1.00091E+00 4.52525E+01 3.15889E-08 4.34258E-01 491.000
c10100 hexagonacell 1.00242E+00 4.51422E+01 3.21192E-08 4.23137E-01 644.000
c10100 hexagonacell 1.00376E+00 4.52141E+01 3.20109E-08 4.26076E-01 785.000
c10100 hexagonacell 1.00229E+00 4.56259E+01 3.13178E-08 4.48119E-01 900.000
c10100 hexagonacell 1.00200E+00 4.51377E+01 3.17724E-08 4.70400E-01 1033.000
c10100 hexagonacell 1.00399E+00 4.48284E+01 3.21142E-08 4.55341E-01 1161.000
c10100 hexagonacell 1.00492E+00 4.46737E+01 3.18787E-08 4.37399E-01 1294.000
c10100 hexagonacell 1.00393E+00 4.57122E+01 3.15362E-08 4.40011E-01 1423.000
c10100 hexagonacell 1.00355E+00 4.51710E+01 3.19133E-08 4.68045E-01 1514.000
c10100 hexagonacell 9.99794E-01 4.55576E+01 3.25166E-08 4.62646E-01 1663.000
c10100 hexagonacell 1.00541E+00 4.54093E+01 3.18271E-08 4.27971E-01 1805.000
c10100 hexagonacell 1.00313E+00 4.51594E+01 3.31041E-08 4.64794E-01 1939.000
c10100 hexagonacell 1.00913E+00 4.49929E+01 3.16040E-08 4.63431E-01 2062.000
c10100 hexagonacell 1.00156E+00 4.51276E+01 3.20585E-08 4.64839E-01 2228.000
c10100 hexagonacell 1.00398E+00 4.55599E+01 3.20174E-08 4.38860E-01 2378.000
c10100 hexagonacell 1.00457E+00 4.58368E+01 3.17587E-08 4.55708E-01 2522.000
c10100 hexagonacell 1.00349E+00 4.52696E+01 3.21480E-08 4.55034E-01 2690.000
=====
Average 1.00338E+00 +/- 2.10265E-03 Total 2690.000
Lowest 9.97630E-01 -5.74850E-03 (from Average)
Highest 1.01146E+00 8.08150E-03 (from Average)
=====

```

The results indicate that for these 20 calculations the standard deviation of the results is about 0.2 %, well within 1/10 of the 3 % requested by **critcalc** input. The results are approximately normally distributed with some results -0.57 to +0.8 % from the average, roughly 2 to 3 times 1/10 of the requested 3 % accuracy. All of these results are no big surprise; this is what we should expect from a Monte Carlo calculation. The one point to

note is that the first calculation, the only answer we would have obtained if we had only run this calculation once, just happened to be the farthest from the average of the 20 calculations. This result is about 0.8 % above the average of the 20 calculations, well within the requested 3 % accuracy, but well outside 1/10 of 3 %.

**Bottom line:** when interpreting the results of TARTNP and TART95 calculations, you can assume that the results will be well within the accuracy you request by input, but do not assume that just because internally these codes continue calculations until the standard deviation of  $K_{\text{eff}}$  is 1/10 that requested, that ALL results will be within 1/10 of the accuracy requested. Please remember how to interpret the TART95 output,

```
Expected k = 9.92956E-01 Std dev = 9.964E-04 (Recommended)
```

This doesn't guarantee that the answer is within plus or minus 9.964E-04 of 0.992956 (which is how people often misinterpret and misquote results). The results will be more or less normally distributed and in a single calculation you could just be unlucky and get results that are in the wings of the normal distribution, two, three, or even four standard deviations from the average of the distribution. The probability of obtaining results that are far from the mean of the normal distribution is quite small, so that in most cases the results will be within one standard deviation about 2/3 of the time and within two standard deviations 95 % of the time. But as we can see from the above results, selecting only one result from the normal distribution is a very small sample, and you can indeed be unlucky some times.

### Dynamic Criticality Calculations

This is an option of the code that seems to be generally not too well understood by users and is often misused and the results are often misinterpreted. Therefore it is worthwhile discussing this type of calculation.

There is a basic problem that is encountered when one tries to use the ENDL data and TART95 for any dynamic (time dependent) problem involving fission: ENDL does not contain any information defining the time dependent emission of delayed neutrons. In any real system that is close to, but not exactly critical, if we pulse the system the short time response can be related to the actual reactivity ( $K_{\text{eff}}$ ) of the system, but the longer time response will be defined by the time dependent emission of delayed neutrons. The time response will also be effected by feedback effects, such as Doppler broadening and expansion or contraction of the system due to temperature changes as the power level of the system changes; none of these effects are included in a TART95 dynamic reactivity calculation. The only situation in which the longer time response is not defined by the emission of delayed neutrons is for a system that is highly prompt critical, where the multiplication of the system can rapidly increase without the contribution of delayed neutrons. Except for the highly prompt super critical case you shouldn't expect the result of a TART95, or TARTNP, dynamic criticality calculation to bear any relationship to anything that can be measured in nature; even in the highly prompt super critical case you

might ask yourself how one would assemble such a system in nature without having it blow itself apart before being finally assembled.

Let's for a moment ignore all these problems and assume that you still want to use TART95 to run dynamic criticality problems. You still have to be very careful to correctly interpret the results. If you run a TART95 calculation and pulse a system with an initial source of neutrons, the distribution will eventually relax and you will get a time response of the form  $\text{Exp}(\alpha t)$ . But depending on what the system looks like, this could take a very long time to happen; long compared to how long neutrons and the system are willing to hang around. For example, if we have a simple spherical system involving fissile material surrounded by a large amount of Be, events can be happening inside the fissile material on a nanosecond time scale and inside the Be on a millisecond time scale. If the system is super critical it would probably blow itself apart before the time scale on which neutrons are effectively reflected back from the Be. If the system is sub critical the distribution inside the fissile material would effectively die out on a short time scale, except for the neutrons that are reflected back from the Be on a longer time scale. In the last case as a function of time all we would be seeing is how long it takes neutrons to leak out of a large volume of Be (either leak through the outer surface and disappear from the system or leak back into the fissile material and possibly cause more fissions to occur), which has little to do with the reactivity of the system. But if we try to use the time scale on which this is happening, and how many fissions are occurring per unit time, we could reach completely erroneous conclusions concerning the time constant and reactivity of the system.

About the only situation in which you may be able to effectively run dynamic criticality calculations is when events are happening in the entire system on roughly the same time scale, e.g., a simple sphere of fissile material with nothing around it to reflect neutrons. Even in this case you have to be careful to correctly interpret the results. For example, if you want to know what happens when you pulse this sub critical sphere with neutrons on its surface, the short time response will depend on how you pulsed it and the multiplication of the system, whereas the longer time response will be independent of how you pulsed it and only depend on the multiplication of the system. Since compared to a pulsed surface source of neutrons, the fundamental critical spatial mode is a more efficient distribution of neutrons as far as criticality, the short term response due to pulsing the surface will show more leakage and have a larger negative alpha, than the long time response which will have less leakage. Which of these are you interested in? TARTNP and TART95 will run for as long a period of time as necessary for the entire distribution to settle down into the fundamental mode and when everything finally converges the alpha and  $K_{\text{eff}}$  will be that corresponding to the long time response of the system. You won't find out anything about the short term response of the system as far as alpha; you can use the calculated  $K_{\text{eff}}$  for the first few time intervals to estimate the short time multiplication. Is this what you were expecting for an answer and does this meet your need? In most cases the answer is no. And, again, remember that even the long term response, the final TARTNP or TART95 answer, does not include the effect of the time dependent emission

of delayed neutrons, so that it will be difficult to relate the results to any real, physical situation.

Therefore, if you expect to use TART95 to run dynamic criticality calculations to define a realistic time constant, alpha, that is related to anything that you can experimentally measure, you are kidding yourself. The option to allow dynamic criticality calculations has been included in TART95 only to maintain compatibility with TARTNP, but this option is **Not Recommended**. Before investing your time and energy running such problems, it is strongly suggested that you contact the authors of this report and discuss your application.

With that said, since dynamic criticality is a TART95 option below we present results for a sample calculation. Above under **New TART95 Features** results are presented for a static criticality calculation involving a bare (unreflected) sphere of uranium. Below we present dynamic criticality results for the same system. Before running any dynamic criticality calculation you first have to determine a time step to use in the calculation. The recommended way to define the time step is to first run a static criticality calculation. One of the results of the static calculation will be the neutron removal time. It is recommended that for a dynamic criticality calculation for the same system you use 1/2 the removal lifetime.

Accurately selecting a time step for a dynamic criticality calculation is very important. Starting from an initial number of neutrons TART95 will track them for the time step indicated by **critcalc** input. If the time step is too large and the system isn't almost exactly critical, in one time step the initial neutrons may all disappear from the system if it is sub critical, or grow to too large a number of neutrons to contain in the memory of a computer if it is super critical. On the other hand if the time step is too small during a single time step you may prevent neutrons from interacting or leaking from the system. Accurately estimating the correct time step to use is almost impossible without first running a static criticality calculation, and then using a time step which is roughly equivalent to the time between generations of neutrons, i.e., the removal lifetime from the static calculation.

The results of the above static criticality calculation include the following information that we will refer to below. In particular note, the removal lifetime of 6.03623E-03 microseconds. For input to a dynamic criticality calculation we must define the time step in shakes (1 shake =  $10^{-8}$  seconds =  $10^{-2}$  microseconds). Therefore based on the static calculation the removal lifetime is about 0.6 shakes. If we use the recommendation to use 1/2 the static removal lifetime as the time step in the corresponding dynamic calculation, we should specify a time step of 0.3.

```

Expected k = 9.92956E-01  Std dev = 9.964E-04 (Recommended)
Actual k   = 9.88887E-01  Std dev = 1.913E-03
Mixed k    = 9.95099E-01  Std dev = 1.290E-03
ekbar     = 9.92914E-01
    
```

All times are in microseconds.

```

Removal lifetime      6.03623E-03.  Std dev  1.146E-05
Leakage lifetime     1.05919E-02
Absorption lifetime  1.40341E-02
Production lifetime  5.42183E-03.  Std dev  1.899E-05
Time to prod. event  2.16328E-03.  Std dev  8.121E-06
Time to capt. event  4.27807E-04.  Std dev  4.379E-06
Time to leakage      3.44515E-03.  Std dev  8.566E-06
Approx.(alpha)/usec -1.16694E+00  (Expected k - 1)/(Removal lifetime)
    
```

For the dynamic calculation we can use essentially the same input deck that we used for the static calculation. In the static case the **critcalc** input specifies **critcalc 15 1 1.0**. For the corresponding dynamic calculation we need merely change this to **critcalc 15 1 1.0 0.3**, adding our step time to indicate a dynamic, rather than static, criticality calculation. No other changes need be made to the input deck. Below we present the results for the dynamic calculation, which are very similar to the static results, and as such need not be discussed in detail here. There are a few points to note, such as the summary of running conditions that now indicates a dynamic reactivity calculation.

```

=====
Summary of Running Conditions
=====
Type of Problem      : Dynamic Reactivity
Particles Tracked    : Neutrons
Multi-band method    : On
Thermal scattering   : On
Repetitions          :          1
Settle Cycles        :          15
Batches              :         2000
Particles/Batch      :         1000
=====
    
```

The cycle by cycle results of the calculation are similar to those in the static case. However, in this case 1524 cycles were run, compared to the 497 in the static case, merely indicating that in this case the step time could have been made somewhat larger; say be a factor of two. The large number of cycles required to converge in both the static and dynamic cases indicates the importance of specifying by input (**sentl 2**) that it is o.k. to run a large number of batches. In this case we allowed up to 2,000 batches. If we had not defined this by input TART95 would have used the default of 20 batches and terminated well before convergence was reached in either the static or dynamic calculation. If TART95 terminates due to using the maximum number of batches allowed it does print a warning message that this has occurred. Be sure to check the number of batches actually used in your calculations to insure that the results are converged and reliable.

```

-----
Settle cycle k values
-----
Cycle   Keff-Cycle   Keff-Average   Absorbed      Leaked      Time Used
          (Seconds)
-----
      1   1.381106   1.381106     .405045     .319013     .00
      .
      .
     14   .927358   1.033493     .428420     .535637     1.00
     15   .999838   1.031249     .429141     .537082     1.00
    
```

Cycle	Keff-Cycle	Keff-Average	Absorbed Per Fission	Leaked Neutron	Time Used (Seconds)
1	1.009240	1.009240	.435507	.555338	1.00
2	1.023396	1.016318	.435113	.548885	1.00
.					
1505	1.007624	.993288	.434821	.572716	294.00
1506	1.033653	.993314	.434820	.572690	294.00
		.993314+/-	.001000	s.d. 100. % Completed	
1507	1.059657	.993358	.434818	.572648	295.00
.					
1522	1.032658	.993639	.434806	.572374	297.00
1523	.996851	.993641	.434804	.572373	298.00
1524	.934096	.993602	.434802	.572417	298.00

It is interesting to compare the static and dynamic results. For  $K_{eff}$  the dynamic result is 0.9936, and the static result is 0.9929. Since they only differ by 0.0007 (well below the requested accuracy of the calculations), statistically they are essentially identical. For the system time constant, alpha, the dynamic result is -1.1979 per microsecond, and the static result is -1.1669 per microsecond. These result are not in as good agreement as  $K_{eff}$ , but still in only differing by about 2.5 % this is an indication that the simple method used in the static calculation to estimate alpha is adequate for use in many applications. For systems that are very close to critical, such as this one, accurately estimating the system time constant, alpha, is very difficult. For example, from the method used in the static calculation is estimate alpha as  $(K_{eff} - 1)$  divided by the removal lifetime, we can see that for  $K_{eff}$  close to unity the uncertainty in alpha can be quite large.

```

(alpha)/usec. = -1.19791E+00 Std dev = 2.759E-01
ex(alpha)/usec. = -1.18519E+00 Std dev = 1.648E-01
n(alpha)/usec. = -1.24771E+00 Std dev = 1.655E-01
Expected k = 9.93602E-01 Std dev = 9.982E-04 (Recommended)
Actual k = 9.91547E-01 Std dev = 1.680E-03
ekbar = 9.92833E-01

```

All times are in microseconds.

```

Removal lifetime 6.05487E-03 Std dev = 6.160E-06
Leakage lifetime 2.15135E-02
Absorption lifetime 6.07667E-03

Normalized leakage 2.81445E-01
Normalized census 9.96413E-01
Normalized net collision gain 2.77858E-01
Normalized absorption 2.25328E-02

```

In this case, with more cycles, the below analysis indicates an excellent approximation to a normal distribution. Note, however that in this case the distribution is wider than in the static case, and of the 1524 values of  $K_{eff}$  sampled 1502 are in the range from 0.9 to 1.1, but 22 are outside this range. This compares to the static case where all of the samples were within the 0.9 to 1.1 range, and almost all were in the range 0.95 to 1.05. Usually wide distributions such as this one merely indicates that we could have statistically



## Chapter 6: Example Problems

In the case using 1,000 histories per batch TART95 ran 1,524 batches in 298 seconds. Using 5,000 histories per batch it ran 301 batches in 318 seconds; less than 7 % increase in the running time.

2)  $K_{\text{eff}}$  is very insensitive to batch size. In the case using 1,000 histories per batch the calculated  $K_{\text{eff}}$  was 0.9936, and using 5,000 it was 0.9945, a difference of 0.0009; this difference is even within 1/10 of the requested 1 % accuracy.

3) For  $K_{\text{eff}}$  close to unity the system time constant, alpha, is very sensitive to the results. Comparing the 1,000 and 5,000 cases, even though the  $K_{\text{eff}}$  only changed from 0.9936 to 0.9945, alpha changed from -1.1979 to -0.876 per microsecond; a change of about 27 %. The user should be aware of this fact and not place too much confidence in the accuracy of small alpha.

4) Using larger batches eliminates outlying statistical results. Below we can see that all 301 sampled  $K_{\text{eff}}$  values are in the 0.9 to 1.1 range, and most are in the 0.95 to 1.03 range. This can be compared to the above results where 22 or the 1524 samples were completely outside the 0.9 to 1.1 range.

```

-----
Settle cycle k values
-----
Cycle   Keff-Cycle   Keff-Average   Absorbed      Leaked      Time Used
          Per Fission Neutron      (Seconds)
-----
    1    1.375135    1.375135      .403032      .324169      1.00
    2    1.013059    1.194097      .410350      .425428      2.00
      .
    15    .973207     1.000480      .429715      .565764     10.00
-----
Cycle   Keff-Cycle   Keff-Average   Absorbed      Leaked      Time Used
          Per Fission Neutron      (Seconds)
-----
    1    .978974     .978974      .433867      .587610     11.00
    2    .982995     .980985      .433392      .585981     12.00
      .
   299    1.020386     .994357      .434828      .570996    316.00
   300    1.012895     .994419      .434825      .570936    317.00
   301    1.019099     .994501      .434822      .570856    318.00

(alpha)/usec. = -8.76012E-01  Std dev = 2.837E-01
ex(alpha)/usec. = -9.33033E-01  Std dev = 1.652E-01
n(alpha)/usec. = -9.45434E-01  Std dev = 1.662E-01

Expected k      = 9.94501E-01  Std dev = 9.952E-04 (Recommended)
Actual k        = 9.94435E-01  Std dev = 1.719E-03
ekbar           = 9.94354E-01
-----

```

All times are in microseconds.

Removal lifetime                      6.05525E-03    Std dev = 6.045E-06

## Chapter 6: Example Problems

```

Leakage lifetime          2.15477E-02
Absorption lifetime      6.07118E-03
  
```

```

Normalized leakage       2.81015E-01
Normalized census        9.97375E-01
Normalized net collision gain 2.78391E-01
Normalized absorption     2.29050E-02
  
```

Frequency Distribution for Expected K Values from 0.9 to 1.1  
(0.01 bin width sums)

Expected K Range	Occurrences
.900 .910	0
.910 .920	0
.920 .930	2 X
.930 .940	0
.940 .950	2 X
.950 .960	4 XX
.960 .970	20 XXXXXXXXXXXXXXXXX
.970 .980	25 XXXXXXXXXXXXXXXXX
.980 .990	53 XXX
.990 1.000	71 XXX
1.000 1.010	73 XXX
1.010 1.020	37 XXXXXXXXXXXXXXXXXXXXXXX
1.020 1.030	10 XXXXXX
1.030 1.040	3 XX
1.040 1.050	1
1.050 1.060	0
1.060 1.070	0
1.070 1.080	0
1.080 1.090	0
1.090 1.100	0

```

-----
Sum          301 (inside 0.9 to 1.1 Range)
              0 (outside 0.9 to 1.1 Range)
-----
  
```

Average .9945 +/- .0172 s. d. (0.0001 bin width average)

Confidence Limits (Occurrences out to +/- 10 times s.d.)

s.d.	Range	Occurrences	Per-Cent
-4	-3	3	.997
-3	-2	5	1.661
-2	-1	34	11.296
-1	0	93	30.897
0	1	123	40.864
1	2	38	12.625
2	3	5	1.661

```

-----
Sum          301
  
```

### Sensitivity of Criticality Results to the Initial Flux Guess

Generally static criticality calculations for very insensitive to the spatial distribution of the initial flux guess. The initial guess will somewhat effect the time for the problem to converge, but not excessively so. The simplest guess is a point source at the origin. The

other extreme that can be used for systems with thick reflectors is, instead of a point source, fill the entire system with the initial guess. If you use either simple guess, for really complicated geometries you should run more settle cycles before the actual calculations begins, to allow the distribution to become properly distributed in space. Whatever you decide to do generally the generation by generation method used in static criticality calculations will always converge, as long as you allow the problem to run enough batches (**sentl 2**).

In contrast dynamic criticality calculations can be sensitive to the spatial distribution of the initial flux guess. This is because of the time step by time step approach used in the calculation. The problem is that if you specify an initial spatial flux guess to be a point source at the origin, the most appropriate time step to use in a calculation will itself be a very strong function of time; short at the beginning and much longer later.

Consider the case of a spherical system of fuel surrounded by a thick Be reflector. If we first run a static criticality calculation to determine the removal lifetime and then use this to estimate the time step for the dynamic criticality problem there can be a problem. The removal lifetime calculated by the static calculation characterizes the long time behavior of the system where the distribution of neutrons is distributed through the system, in particular the distribution will extend well out into the reflector where events will be happening on a much longer time scale.

In contrast, starting from an initial flux guess as a point source at the origin the appropriate time step for the first few time steps, as the distribution spreads out from the origin, can be orders of magnitude smaller than the appropriate time step when the distribution has spread out from the fuel into the thick Be reflector. During the first few time steps the distribution will only be interacting with the fuel, resulting in a very small removal time, and rapid changes in the population of neutrons (increasing if  $K_{eff} > 1$ , and decreasing if  $K_{eff} < 1$ ). During later time steps the spatial distribution will extend out into the reflector and everything will be happening on a much longer time scale. It is only during these later time steps that the removal lifetime calculated in the static criticality calculation is a good estimate of the time step to use in the calculation.

Assuming that the distribution will eventually vary as  $\text{Exp}[\alpha t]$ , and that a good approximation is that,

$$\alpha \sim (K_{eff} - 1)/[\text{neutron removal lifetime}]$$

During the first few time steps when the distribution is only interacting with the fuel and the removal lifetime is very small, the magnitude of the distribution can vary very rapidly. In contrast during the later time steps when the distribution is interacting with the reflector and the removal lifetime is long, the magnitude of the distribution can vary very slowly. In order to reach convergence the calculation has to reach these later time steps in order to characterize the stable, final  $\text{Exp}[\alpha t]$  variation. Unfortunately, due to the different time scales involved early and later in the calculation, it may not be able to get through the

initial time steps, and even if it does it may then converge very slowly because of the method used.

As it effects TART95 dynamic criticality calculations, during the first time step the initial distribution can completely disappear (if  $K_{\text{eff}} < 1$ ), or multiply so much that all the neutrons cannot be contained in memory (if  $K_{\text{eff}} > 1$ ). If either of these extremes occur TART95 will assume that the input time step was too large and it will decrease the time step by a factor of 2 and restart the calculation. It will continue to half the time step until it obtains what it considers to be reasonable results at the end of the first time step. It will then continue the calculation through successive time steps to convergence. The problem is that we know that the static removal life time is a good estimate of the long time behavior of the system. However, once TART95 has initially decreased the time step to get through the initial time step it continues to use this time step throughout the remainder of the calculation. It may have initially reduced the time step so much that it would take an enormous number of additional time steps to reach and characterize the long time behavior of the system, in which case you will probably exceed the maximum number of batches (**sentl 2**) before reaching convergence.

This problem can be avoided by realizing what the problem is and insuring that your initial spatial flux guess doesn't result in TART95 having to drastically reduce the time step. The way to avoid this is to simply use a distributed source, i.e., an initial flux guess that extends well out into thick reflectors. As pointed out above, a simple guess is to fill the entire geometry.

As an example of how to recognize and avoid this problem, consider the third of the 84 fast critical assemblies described above. This is a Pu sphere reflected by 13 cm of Be. The static removal lifetime is 21.65 microseconds. Based on this removal lifetime for a dynamic calculation we should use a time step of about 11 microseconds (1100 shakes as input).

Below we show the results of running this calculation starting from an initial flux guess as a point at the origin; we only show the cycles. Once you see the message about overlapping storage and reducing time step you should interpret this to mean that there is a problem due to your initial spatial flux guess. In the below case starting from an input time step of 1100 shakes, the code has reduced the time step to 550, and since it still had problem it reduced it again to 275 shakes, before proceeding with the calculation. Note, by the time it had run enough calculations to decide to reduce the time step twice it took 406 seconds of computer time just to produce results for the first settle cycle. Most importantly after running 2000 cycles it still hadn't reached convergence and it terminated with the warning message that all batches were processed before convergence. Therefore, the results are completely unreliable and the entire run was a waste of time.

In any case where you start to see the message about reducing time step you should strongly consider stopping the run, improving your initial spatial flux guess, and then

starting the run again. You can do this immediately because the message about reducing time step will appear on your screen to immediately let you know there is a problem.

```
-----
Settle cycle k values
-----
Cycle   Keff-Cycle   Keff-Average   Absorbed      Leaked      Time Used
          Per Fission Neutron      (Seconds)
-----
alphak ERROR aflag=3841
Neutron census storage overlapped. Will try to expand.

Reduced time step to 5.50000E+02
pborn storage size is 500 particles.
```

```
-----
Settle cycle k values
-----
Cycle   Keff-Cycle   Keff-Average   Absorbed      Leaked      Time Used
          Per Fission Neutron      (Seconds)
-----
alphak ERROR aflag=3841
Neutron census storage overlapped. Will try to expand.

Reduced time step to 2.75000E+02
pborn storage size is 500 particles.
```

```
-----
Settle cycle k values
-----
Cycle   Keff-Cycle   Keff-Average   Absorbed      Leaked      Time Used
          Per Fission Neutron      (Seconds)
-----
      1      1.212036      1.212036      .402615      .422443      406.00
      2      .937424      1.074730      .402908      .427461      411.00
          .
          .
      14      .950596      .896596      .404216      .442332      438.00
      15      .969109      .901430      .404289      .443044      440.00
-----
Cycle   Keff-Cycle   Keff-Average   Absorbed      Leaked      Time Used
          Per Fission Neutron      (Seconds)
-----
      1      .556849      .556849      .464431      1.331388      440.00
      2      1.001231      .779040      .424441      .716530      441.00
          .
          .
     1902      .751690      .859743      .417038      .582863      1831.00
     1903      .636885      .859626      .417041      .582922      1832.00
          .
          .
     1904      .881554      .859638      .417043      .582950      1832.00
     1905      .781918      .859597      .417047      .582988      1833.00
          .
          .
     1998      .804688      .861722      .416925      .581956      1941.00
```

## Chapter 6: Example Problems

```

1999    1.068703    .861825    .416919    .581918    1943.00
2000    .535239      .861662    .416922    .581988    1943.00

```

15 settle batches processed.

2000 batches processed.

Warning - All batches processed before convergence

To illustrate the sensitivity to the initial source and how to avoid the problem, for exactly the same problem as discussed above we have changed from an initial spatial flux guess of a point source at the origin to a distributed spherical source throughout the entire system and we ran the problem again. The below results illustrate that compared to the above results where it took 406 seconds to produce results for the first settle cycle, here the same results are produced in 30 seconds, and by the time required to produce the first settle cycle results in the above case, in this case the problem is well on its way toward convergence.

```

-----
Settle cycle k values
-----

```

Cycle	Keff-Cycle	Keff-Average	Absorbed Per Fission	Leaked Neutron	Time Used (Seconds)
1	.970750	.970750	.418466	.611666	30.00
2	.982917	.976833	.417504	.610058	39.00
.	.	.	.	.	.
14	1.066591	.998534	.415699	.581222	101.00
15	.873415	.990193	.415958	.583241	103.00

```

-----

```

Cycle	Keff-Cycle	Keff-Average	Absorbed Per Fission	Leaked Neutron	Time Used (Seconds)
1	1.065743	1.065743	.414341	.523972	108.00
2	.934038	.999891	.417609	.567481	111.00
.	.	.	.	.	.
68	1.019486	.944928	.415910	.570731	376.00
69	1.090038	.947031	.415698	.569080	382.00
70	1.035750	.948299	.415637	.568598	390.00
71	.993714	.948938	.415655	.568978	395.00
72	.938745	.948797	.415754	.569686	399.00
73	.847507	.947409	.415852	.570613	401.00
74	1.061718	.948954	.415799	.570201	404.00

```

-----

```

**Bottom line:** concerning dynamic criticality calculations, this type of calculation does not include the effect of delayed neutrons, either in the basic definition of  $\bar{\nu}$ , nor in the time dependent emission of delayed neutrons. All delayed neutrons are completely ignored in the calculation. Therefore it is difficult to design and correctly interpret the results of dynamic criticality calculations that bear any relationship to a real system. The option to perform dynamic criticality calculations has been included in TART95 only to maintain compatibility with TARTNP. However, this type of calculation is **not recommended**. If

you feel that you have an application that requires this type of calculation, we strongly urge you to consult the authors of this report before performing the calculation.

For  $K_{eff}$  close to unity the uncertainty in the calculated system time constant, alpha, will be very large and very sensitive to running conditions, e.g., note the large change in alpha for the above results using 1000 and 5000 histories per batch. Due to this large uncertainty, for many applications the estimated alpha from a static calculation is sufficiently accurate and you may not have to run a dynamic calculation.

## Source Problems

### 14.1 MeV Neutrons or Photons in Water

The first example source is intentionally simple and hopefully easy to understand. This problem is designed to illustrate the speed of TART95 and to introduce users to the types of output that they will receive.

The problem involves a source of 14.1 MeV neutrons or photons at the origin of a sphere of water 10 cm in radius. The sphere has been divided into 10 zones each 1 cm in radius. The input deck includes options to run either neutrons or photons; to switch from one to the other we need merely comment out a few lines and activate a few other lines. In each case we will run 10 batches of 100,000 histories (neutrons or photons) = a total of 1,000,000 histories. We will run three different calculations: 1) neutron source, only track neutrons, 2) neutron source, track neutrons and induced photons, 3) photon source, only track photons. The neutron source calculations include all of the options that will make the calculation as accurate as possible and take the longest time to run. These options include the multi-band method to handle resonance self-shielding and thermal scattering.

Let's first look at the timing results as they appear on your screen. At the end of each batch (in this case 10 batches) TART95 will tell you how much time it has used so far and estimate how long it still has to run in order to complete the program. In simple geometry, as in this case, the estimated time to completion is usually a good estimate, even after only the first batch of particles has been run. In more complicated geometry TART95 has a built in accelerator that learns progressively more what the geometry looks like as the calculation proceeds, allowing it to reduce the total running time from the initial (first batch) estimate. As mentioned above, in each case we ran 1,000,000 histories. The timing results on an HP-350 are,

Neutron Source, Only Track Neutrons.....	347 Seconds
Neutron source, track neutrons and induced photons...	424 Seconds
Photon source, only track photons.....	93 Seconds

If you compare these running times to those of other codes you will begin to appreciate just how fast TART95 is. Below we show the batch by batch results that appear on your screen for each of these three cases.

Neutron Source, Only Track Neutrons

Time to Completion (Seconds)	Per-cent Completed	Time Used (Seconds)
306.00	10.00	34.00
276.00	20.00	69.00
242.67	30.00	104.00
208.50	40.00	139.00
173.00	50.00	173.00
138.67	60.00	208.00
104.14	70.00	243.00
69.50	80.00	278.00
34.67	90.00	312.00
-----		
347.00 seconds - Finished run		
-----		

Neutron source, track neutrons and induced photons

Time to Completion (Seconds)	Per-cent Completed	Time Used (Seconds)
378.00	10.00	42.00
340.00	20.00	85.00
296.33	30.00	127.00
259.50	40.00	173.00
215.00	50.00	215.00
171.33	60.00	257.00
127.71	70.00	298.00
85.00	80.00	340.00
42.44	90.00	382.00
-----		
424.00 seconds - Finished run		
-----		

Photon source, only track photons

Time to Completion (Seconds)	Per-cent Completed	Time Used (Seconds)
81.00	10.00	9.00
72.00	20.00	18.00
63.00	30.00	27.00
54.00	40.00	36.00
46.00	50.00	46.00
36.67	60.00	55.00

```

27.43      70.00      64.00
18.25      80.00      73.00
 9.22      90.00      83.00

```

```

-----
93.00 seconds - Finished run
-----

```

Next we can look at the output report. Because of the length of the actual output report, we have edited it down to a much smaller size in order to present here, and we have included only enough information to allow us to explain the results. The following output is from the neutron source, track neutrons and induced photons problem. This is the only problem of the three that includes results for both neutrons and photons, so that the output for this case can serve to illustrate what you should generally obtain as output. In this problem we have only requested standard output; special options to obtain additional output were not used.

```

TART - Coupled Neutron-Photon Monte Carlo Transport (TART 95-1, Jan. '95)
=====
I/O Files Opened for Entire Run
=====
Definition                               Filename  Unit  Date
=====
TART Input Parameters.....TART.IN        2
TART Output Listing.....TART.OUT         3
Neutron Interaction Data File.....TARTND    7   6/10/92
Photon Interaction Data File.....GAMDAT     8   3/19/93
Neutron Induced Photon Production File...TARTPPD  9   6/10/92
Multi-Band Parameter File.....NEWCROSS    10  7/19/90
=====
Start of Next Problem      11:56:01 May23'95 HP
=====
Current Dynamic Memory Used      0 Words
=====

```

The following section is a listing of the input deck as read. Generally this listing will include warnings or error messages if there appears to be anything wrong with the input. In this case there are no warnings or error messages. Hopefully based on the above example input for criticality problems and the above description of this problem the reader is now familiar enough with TART95 input to understand the following deck.

```

=====
Problem Input Parameters
=====
* =====
name      Water spheres
box o84   Water spheres
* =====
*
* TART Input Deck Generated Using Program TARTVIEW (93-1)
*
* =====
* 14.1 MeV Neutron or photon source at the origin of
* Concentric spheres out to 10 cm in radius
*
* see below comments on how to switch between neutrons

```

## Chapter 6: Example Problems

```

* and photons
*
* =====
*
* Surface Definitions
* 1-D Spherical - Only Spherical Surfaces Required
* 1) Surface Number - used later to Define Zone Boundaries
* 2) Position - Radius Z0 X0 Y0
*
* =====
sphere    1    1.0
sphere    2    2.0
sphere    3    3.0
sphere    4    4.0
sphere    5    5.0
sphere    6    6.0
sphere    7    7.0
sphere    8    8.0
sphere    9    9.0
sphere   10   10.0
* =====
*
* Zone Definitions
* 1) Zone Number
* 2) Bounding Surface Number and Sign of Vector Normal
*    to the Surface for Particles Leaving the Zone.
*    2) Is repeated to include all Bounding Surfaces.
*
* =====
jb        1     1
jb        2    -1     2
jb        3    -2     3
jb        4    -3     4
jb        5    -4     5
jb        6    -5     6
jb        7    -6     7
jb        8    -7     8
jb        9    -8     9
jb       10    -9    10
jb       11   -10
* =====
*
* Material Definitions
* 1) Material Number
* 2) Density (grams/cc)
* 3) Atom %
* 4) Isotope I.D. (ZZZAAA), e.g., 92238 for U-238
*    3) and 4) can be repeated in pairs to define
*    composite materials
*
* =====
* Water
matl    1  1.0  2.0  1001  1.0  8016
* =====
*
* Assignment of Materials to Zones
* 1) Material Number
* 2) Zone Containing Material - Can be Repeated
*
* =====
* Vacuum - ILLEGAL EXCEPT IN OUTER, NON-REENTRANT ZONES
matz    0    11
* Water
matz    1    1 thru 10
* =====
*
* Definition of Sources

```

```

*
* A point source at the origin
* Monoenergetic 14.1 MeV
* Isotropic = default sentl 6 and 7
*
* =====
* use this for a neutron source
source1  1  0.0  0.0  0.0
sentl    4   14.1
* use this for a photon source
* slg    1  0.0  0.0  0.0
* sentl   17   14.1
* =====
*
* Definition of Running Conditions and Output Edit Options
*
* =====
* 1) Transport (neutrons and/or photons) (0)
* use this for a neutron source
* sentl  1    1
* use this for a neutron source + induced photons
sentl  1    0
* use this for a photon source
* sentl  1    2
* 2) Number of Samples (20)
sentl  2    10
* 3) Histories per Sample (5000)
sentl  3   10000
* 8) Neutron Minimum Energy (2.53e-8 MeV)
sentl  8  1.307E-09
* 20) Multi-Band (0)
sentl  20    1
* 39) Thermal Scattering Sentinels (0)
sentl  39    1
* Thermal Scattering Temperature in All Zones (2.53e-8 MeV)
emin  2.53000e-08  1 thru  4
* Score and output fluence per zone
ltype  2  1 thru 10
end

```

Next we have the summary of running conditions that we can check to insure that we are running the correct problem using the recommended options.

```

=====
Summary of Running Conditions
=====
Type of Problem      : Source
Particles Tracked   : Neutrons and induced photons
Multi-band method   : On
Thermal scattering  : On
Batches              :          10
Particles/Batch     :       100000
=====

```

Then there is a summary of all materials used in the problem; in this case only one material. This is followed by the volume (in cc) and mass (in grams) of each zone. Note, TART95 can analytically calculate the volume of a zone only if it is symmetric about the z axis and finite in volume. In this case all 10 interior spherical shells are symmetric about the z axis, so that their volumes has been correctly calculated. Zone 11, everything outside the outer most spherical surface, is not finite in volume, so its volume cannot be correctly calculated. In general if TART95 cannot correctly calculate the volume of a zone it is defined to have a volume of 1 cc. Slightly further down in the report is an additional table

defining the mass of each constituent of the material in each zone; in this case only hydrogen and oxygen.

```
matl  atom/cm-b  fractions  isotopes  1
1
1  3.34367E-02  2.00000E+00  1001.  1.00000E+00  8016.
```

zone	volume	s.d.	pct.	mass
1	4.188790E+00	.00E+00	.00	4.188790E+00
2	2.932153E+01	.00E+00	.00	2.932153E+01
3	7.958701E+01	.00E+00	.00	7.958701E+01
4	1.549852E+02	.00E+00	.00	1.549852E+02
5	2.555162E+02	.00E+00	.00	2.555162E+02
6	3.811799E+02	.00E+00	.00	3.811799E+02
7	5.319764E+02	.00E+00	.00	5.319764E+02
8	7.079055E+02	.00E+00	.00	7.079055E+02
9	9.089675E+02	.00E+00	.00	9.089675E+02
10	1.135162E+03	.00E+00	.00	1.135162E+03
11	1.000000E+00	.00E+00	.00	1.000000E+00

Total volume = 4.18879020E+03 cc.      Total weight = 4.18879020E+03 grams

material	mass	density
1	4.188790E+03	1.000000E+00

```
Library  6/10/92
Sizes of iwl and w1 are  353  3897
iso      age      size
1  1001  9202  1240
2  8016  9112  2657
```

```
Zonal weight in grams of each isotope.
1 4.688E-01  1001  3.720E+00  8016
2 3.282E+00  1001  2.604E+01  8016
3 8.907E+00  1001  7.068E+01  8016
4 1.735E+01  1001  1.376E+02  8016
5 2.860E+01  1001  2.269E+02  8016
6 4.266E+01  1001  3.385E+02  8016
7 5.954E+01  1001  4.724E+02  8016
8 7.923E+01  1001  6.287E+02  8016
9 1.017E+02  1001  8.072E+02  8016
10 1.270E+02  1001  1.008E+03  8016
```

Summary of multi-band data being used

```
-----
zaaaa  temperature  temperature Bands
      (Kelvin)      (MeV)
-----
1001  2.9949E+02  2.5808E-08  2
   za group  1 band      2 bands  % difference
1001  175  5.040E-01  5.189E-01  -2.875E+00
1001  1 Total cross sections differed by more than 0.1 %
8016  2.9949E+02  2.5808E-08  2
   za group  1 band      2 bands  % difference
8016  175  1.597E+00  1.594E+00  1.781E-01
8016  1 Total cross sections differed by more than 0.1 %
iwl size  493  w1 size  4775  pband size  700
```

Next the neutron mean free path (in cm) is listed for each material used in the problem (in this case only one) for each of the 175 neutron multi-groups. The number 1.0000E+00 at the top of the table is the density (grams/cc) of each material. This is followed by neutron energy deposits (in MeV/cm) for each material. This is followed by a summary of

important individual cross sections expressed are a fraction of the total cross section in each group. The neutron angular and energy tally bins are then defined.

Neutron mean free paths for each material

energy	1
1	1.0000E+00
1	1.307E-09
2	2.0023E-01
2	5.227E-09
3	3.5555E-01
3	2.091E-08
4	4.6283E-01
4	3.267E-08
	5.1055E-01
.	
173	1.752E+01
173	1.1259E+01
174	1.813E+01
174	1.1551E+01
175	1.875E+01
175	1.1364E+01
176	2.000E+01
176	1.0000E+20

Neutron energy deposits for each material

energy	1
1	1.7018E-05
1	1.307E-09
2	1.5114E-05
2	5.227E-09
3	1.3125E-05
3	2.091E-08
4	1.1855E-05
4	3.267E-08
.	
173	4.8488E+00
173	1.752E+01
174	4.9815E+00
174	1.813E+01
175	5.0842E+00
175	1.875E+01
176	.0000E+00
176	2.000E+01

mat	tally	group	energy	elastic	non-elastic minus absorption	fission	absorption minus fission
1	1	1	1.307E-09	9.902E-01	.000E+00	.000E+00	9.819E-03
1	2	2	8.322E-08	9.953E-01	.000E+00	.000E+00	4.736E-03
1	3	3	4.234E-07	9.978E-01	.000E+00	.000E+00	2.220E-03
1	4	4	2.091E-06	9.988E-01	.000E+00	.000E+00	1.169E-03
.							
1	48	48	1.413E+01	7.385E-01	2.193E-01	.000E+00	4.219E-02
1	49	49	1.441E+01	7.329E-01	2.252E-01	.000E+00	4.186E-02
1	50	50	1.519E+01	7.243E-01	2.300E-01	.000E+00	4.566E-02
			2.000E+01				

Cosines of angular tally groups.

1	1.00	5	.82	9	.20	13	-.40	17	-.90
2	.98	6	.72	10	.05	14	-.60	18	-.95
3	.95	7	.60	11	-.05	15	-.72	19	-.98
4	.90	8	.40	12	-.20	16	-.82	20	-1.00

Energy of neutron tally groups.

1	1.307E-09	11	6.097E-05	21	5.763E-03	31	2.091E+00	41	8.322E+00
2	8.322E-08	12	7.155E-05	22	2.646E-02	32	2.530E+00	42	9.177E+00
3	4.234E-07	13	8.431E-05	23	7.002E-02	33	3.011E+00	43	1.012E+01
4	2.091E-06	14	9.811E-05	24	2.075E-01	34	3.533E+00	44	1.101E+01
5	6.737E-06	15	1.338E-04	25	3.777E-01	35	4.069E+00	45	1.199E+01
6	1.468E-05	16	1.789E-04	26	5.123E-01	36	4.704E+00	46	1.307E+01
7	2.277E-05	17	3.267E-04	27	7.527E-01	37	5.353E+00	47	1.386E+01
8	2.940E-05	18	6.042E-04	28	1.025E+00	38	6.042E+00	48	1.413E+01
9	4.048E-05	19	1.058E-03	29	1.338E+00	39	6.737E+00	49	1.441E+01
10	4.918E-05	20	2.561E-03	30	1.694E+00	40	7.548E+00	50	1.519E+01
									2.000E+01

Similar information is next presented for photons, including the mean free path (in cm) and energy deposition (MeV/cm) for each material, followed by the angular and energy photon tally bins.

Photon mean free paths, for each material

	energy	1
		1.0000E+00
1	1.000E-04	1.6894E-05
2	1.089E-04	2.0236E-05
3	1.103E-04	2.0790E-05
4	1.466E-04	3.9354E-05
.		
173	1.186E+01	4.8157E+01
174	1.335E+01	5.0024E+01
175	2.000E+01	5.5431E+01
176	3.000E+01	5.8888E+01

Probable photon energy deposition per collision for each material

	energy	1
1	1.000E-04	9.9999E-05
2	1.089E-04	1.0890E-04
3	1.103E-04	1.1030E-04
4	1.466E-04	1.4659E-04
.		
173	1.186E+01	8.9644E+00
174	1.335E+01	1.0305E+01
175	2.000E+01	1.6490E+01
176	3.000E+01	2.6184E+01

Energy of photon tally groups		cosines of angular tally groups	
output	normalized per source neutron.		
1	.00010	1	1.00
2	.00136	2	.98
3	.00184	3	.95
4	.00224	4	.90
.			
47	6.58300		
48	8.33200		
49	10.55000		
50	13.35000		
	30.00000		

The next section presents the neutron induced photon production cross sections for each isotope used in the problem; in this case for hydrogen and oxygen.

Gamma production cross sections for each isotope

	energy	1001	8016
1	1.307E-09	9.7757E-01	.0000E+00
2	5.227E-09	4.8878E-01	.0000E+00
3	2.091E-08	3.2587E-01	.0000E+00
4	3.267E-08	2.6668E-01	.0000E+00
.			
173	1.752E+01	3.4136E-05	1.1672E+00
174	1.813E+01	3.1840E-05	1.2625E+00
175	1.875E+01	2.8362E-05	1.4385E+00
176	2.000E+01	.0000E+00	.0000E+00

At this point the actual Monte Carlo calculation begins. At the end of each batch of source particles the code will print results for photons (listed first) and neutrons. The following

results are from the end of the first batch, after 100,000 neutron histories have been followed. First results are printed for the actual (analog) and expected energy deposition in each zone. These results are integrated over energy; later results will be presented for each energy tally bin. Then integrated over energy results are presented for the tally type for each zone. Note, for photons we used the default tally type 3 (particles entering each zone) and for neutrons all interior zones used tally type 2 (pathlength = fluence), and the exterior zone used tally type 3 (particles entering the zone). Since the exterior zone is using tally type 3, the results are the leakage from the outer most sphere. The integrated energy and tally type results indicated that for each 14.1 MeV source neutron, about 8.76 MeV and 89.2 % of the neutrons leaked from the outer most sphere. These results are following by detailed results for each zone, for each energy tally bin. Note, in order to present results in a limited space the detailed result table has been edited down to only include 5 zones; the actual output listing contains results for all 11 zones.

Photon edit is per source neutron unless noted otherwise.

Error in photon energy balance is -5.004952E-13 percent.

Actual photon energy deposition  
in MeV/zone by time step.

zone	total	Std dev	pct	1.000E+08
1	2.288E-03	.00E+00	.0	2.29E-03
2	6.763E-03	.00E+00	.0	6.76E-03
3	1.026E-02	.00E+00	.0	1.03E-02
4	1.488E-02	.00E+00	.0	1.49E-02
5	1.782E-02	.00E+00	.0	1.78E-02
6	1.990E-02	.00E+00	.0	1.99E-02
7	2.245E-02	.00E+00	.0	2.25E-02
8	2.428E-02	.00E+00	.0	2.43E-02
9	2.362E-02	.00E+00	.0	2.36E-02
10	2.335E-02	.00E+00	.0	2.33E-02
11	8.229E-01	.00E+00	.0	8.23E-01

Expected value photon energy depositions.  
in MeV/zone by time step.

zone	total	Std dev	pct	1.000E+08
1	2.502E-03	.00E+00	.0	2.50E-03
2	6.947E-03	.00E+00	.0	6.95E-03
3	1.091E-02	.00E+00	.0	1.09E-02
4	1.444E-02	.00E+00	.0	1.44E-02
5	1.792E-02	.00E+00	.0	1.79E-02
6	2.084E-02	.00E+00	.0	2.08E-02
7	2.255E-02	.00E+00	.0	2.25E-02
8	2.383E-02	.00E+00	.0	2.38E-02
9	2.385E-02	.00E+00	.0	2.38E-02
10	2.284E-02	.00E+00	.0	2.28E-02
11	8.229E-01	.00E+00	.0	8.23E-01

Photon tally type totals, by zone and time step.

zone	tt	total	dev.	pct.	1.00E+08
1	3	9.200E-03	.0E+00	.0	9.20E-03
2	3	5.598E-02	.0E+00	.0	5.60E-02
3	3	1.003E-01	.0E+00	.0	1.00E-01
4	3	1.436E-01	.0E+00	.0	1.44E-01
5	3	1.821E-01	.0E+00	.0	1.82E-01
6	3	2.188E-01	.0E+00	.0	2.19E-01
7	3	2.505E-01	.0E+00	.0	2.51E-01
8	3	2.755E-01	.0E+00	.0	2.75E-01

## Chapter 6: Example Problems

```

 9  3  2.915E-01  .0E+00  .0  2.91E-01
10  3  2.965E-01  .0E+00  .0  2.97E-01
11  3  3.071E-01  .0E+00  .0  3.07E-01

Census time is 1.000E+08  photon tally type  Water spheres
                    type      3          3          3          3          3
n  energy      zone 1  zone 2  zone 3  zone 4  zone 5
 1  1.00E-04    .000E+00 .000E+00 .000E+00 .000E+00 .000E+00
 2  1.36E-03    .000E+00 .000E+00 .000E+00 .000E+00 .000E+00
 3  1.84E-03    .000E+00 .000E+00 .000E+00 .000E+00 .000E+00

43  2.03E+00    1.440E-03 7.350E-03 1.458E-02 2.403E-02 3.240E-02
44  2.89E+00    1.000E-03 7.200E-03 1.221E-02 1.684E-02 2.044E-02
45  4.11E+00    3.500E-04 2.230E-03 3.970E-03 5.330E-03 6.740E-03
46  5.20E+00    1.650E-03 1.048E-02 1.749E-02 2.349E-02 2.807E-02
47  6.58E+00    7.300E-04 5.640E-03 9.460E-03 1.282E-02 1.560E-02
48  8.33E+00    2.000E-05 1.200E-04 2.300E-04 3.400E-04 4.400E-04
49  1.06E+01    .000E+00 .000E+00 .000E+00 .000E+00 .000E+00
50  1.34E+01    .000E+00 .000E+00 .000E+00 .000E+00 .000E+00

```

Neutron edit.

Error in total energy balance is -4.90100E-12 percent.

All energies are in MeV.

```

Total source energy from mass difference q is..... -5.83457E-01
Average energy leaked per source neutron is..... 8.66036E+00
Average energy leaked per neutron leaking is..... 9.70664E+00
Energy gained or lost by splitting at a boundary is. .00000E+00
Energy gained or lost by thermal treatment is..... 1.16943E-14
Energy deposits from neutrons only.
Endothermic energy from mass difference q is..... 7.38608E-01
Analog energy per source neutron deposited is..... 3.86763E+00
Expected energy per source neutron deposited is..... 3.81802E+00

```

Local expected value energy depositions from neutrons only, in MeV/zone, by time step.

```

zone total      Std dev pct  1.000E+08
 1  4.1043E-01  .00E+00  .0  4.104E-01
 2  4.0907E-01  .00E+00  .0  4.091E-01
 3  4.0619E-01  .00E+00  .0  4.062E-01
 4  4.0044E-01  .00E+00  .0  4.004E-01
 5  3.9440E-01  .00E+00  .0  3.944E-01
 6  3.8495E-01  .00E+00  .0  3.849E-01
 7  3.7406E-01  .00E+00  .0  3.741E-01
 8  3.6178E-01  .00E+00  .0  3.618E-01
 9  3.4733E-01  .00E+00  .0  3.473E-01
10  3.2937E-01  .00E+00  .0  3.294E-01
11  8.6604E+00  .00E+00  .0  8.660E+00

```

Neutron tally type totals by zone and time step.

```

zone tt total      dev.      pct.  1.00E+08
 1  2  1.029E+00  .00E+00  .0  1.03E+00
 2  2  1.108E+00  .00E+00  .0  1.11E+00
 3  2  1.207E+00  .00E+00  .0  1.21E+00
 4  2  1.305E+00  .00E+00  .0  1.30E+00
 5  2  1.420E+00  .00E+00  .0  1.42E+00
 6  2  1.499E+00  .00E+00  .0  1.50E+00
 7  2  1.520E+00  .00E+00  .0  1.52E+00
 8  2  1.483E+00  .00E+00  .0  1.48E+00
 9  2  1.389E+00  .00E+00  .0  1.39E+00
10  2  1.215E+00  .00E+00  .0  1.22E+00
11  3  8.922E-01  .00E+00  .0  8.92E-01

```

## Chapter 6: Example Problems

Photon source / zone in MeV.

zone	total	Std dev	pct	1.000E+08
1	1.0338E-01	.00E+00	.0	1.034E-01
2	9.8052E-02	.00E+00	.0	9.805E-02
3	9.6626E-02	.00E+00	.0	9.663E-02
4	9.2144E-02	.00E+00	.0	9.214E-02
5	1.0609E-01	.00E+00	.0	1.061E-01
6	1.0854E-01	.00E+00	.0	1.085E-01
7	1.0599E-01	.00E+00	.0	1.060E-01
8	1.0329E-01	.00E+00	.0	1.033E-01
9	9.4534E-02	.00E+00	.0	9.453E-02
10	7.9909E-02	.00E+00	.0	7.991E-02

Local neutron energy deposition plus photon deposition (total energy deposited)

zone	total	Std dev	pct	1.000E+08
1	4.1294E-01	.00E+00	.0	4.129E-01
2	4.1602E-01	.00E+00	.0	4.160E-01
3	4.1711E-01	.00E+00	.0	4.171E-01
4	4.1487E-01	.00E+00	.0	4.149E-01
5	4.1232E-01	.00E+00	.0	4.123E-01
6	4.0578E-01	.00E+00	.0	4.058E-01
7	3.9660E-01	.00E+00	.0	3.966E-01
8	3.8561E-01	.00E+00	.0	3.856E-01
9	3.7118E-01	.00E+00	.0	3.712E-01
10	3.5221E-01	.00E+00	.0	3.522E-01
11	9.4833E+00	.00E+00	.0	9.483E+00

Multi-band n-gammas by zone and time step.

zone	total	1.000E+08
1	9.22236E-05	9.222E-05
2	5.41933E-04	5.419E-04
3	1.30391E-03	1.304E-03
4	2.58039E-03	2.580E-03
5	8.67447E-03	8.674E-03
6	1.26867E-02	1.269E-02
7	1.39864E-02	1.399E-02
8	1.31462E-02	1.315E-02
9	1.09624E-02	1.096E-02
10	6.54105E-03	6.541E-03

Census time is 1.000E+08 neutron tally type Water spheres

n	energy	zone 1	zone 2	zone 3	zone 4	zone 5
1	1.31E-09	3.435E-03	2.116E-02	5.171E-02	9.224E-02	1.733E-01
2	8.32E-08	7.607E-04	5.141E-03	1.196E-02	1.873E-02	1.529E-02
3	4.23E-07	3.224E-04	2.339E-03	6.207E-03	9.957E-03	1.367E-02
4	2.09E-06	2.690E-04	1.793E-03	4.899E-03	8.421E-03	1.044E-02
42	9.18E+00	1.589E-03	4.565E-03	7.569E-03	9.777E-03	1.207E-02
43	1.01E+01	2.767E-03	6.870E-03	9.663E-03	1.197E-02	1.405E-02
44	1.10E+01	5.392E-03	1.307E-02	2.008E-02	2.384E-02	2.696E-02
45	1.20E+01	5.611E-03	1.535E-02	2.363E-02	3.006E-02	3.571E-02
46	1.31E+01	4.901E-03	1.346E-02	2.048E-02	2.676E-02	3.147E-02
47	1.39E+01	9.585E-01	8.813E-01	8.104E-01	7.449E-01	6.850E-01
48	1.41E+01	.000E+00	.000E+00	.000E+00	.000E+00	.000E+00
49	1.44E+01	.000E+00	.000E+00	.000E+00	.000E+00	.000E+00
50	1.52E+01	.000E+00	.000E+00	.000E+00	.000E+00	.000E+00

Average particle collision gain by zone and time step.

1.00000E+08  
 1-4.71000E-03  
 2-4.71000E-03  
 3-5.16000E-03

```

4-6.69000E-03
5-1.23700E-02
6-1.63400E-02
7-1.77700E-02
8-1.73300E-02
9-1.36800E-02
10-9.03000E-03

```

The above output is repeated at the end of each batch. In this case there will 10 sets of similar results, until we reach the final results shown below. By checking the results at the end of each batch you can check on the convergence of the calculations. For example based on the results of the first batch (presented above) for every 14.1 MeV source neutron 8.76 MeV and 89.2 % of the neutrons leaked. Based on the below results after ten batches these numbers have changed only slightly to 8.63 MeV and 89.1 %.

**Warning** - the standard deviation in the results are defined by statistically comparing the results of successive batches. Note, at the end of the first batch (the results presented above) there are no other batches to compare and the code lists standard deviation as zero. This doesn't mean everything has converged; it merely indicates that the statistical comparison hasn't started yet. Once you understand this point you will be able to answer the question: why run 10 batches each of a given number of particles? Why not simply run one batch with 10 times as many particles? The answer is that with one batch the output will not include any meaningful statistical information concerning the accuracy of the results.

There is a number, 1.000E+08, that keeps appearing in the output results. This is because in this case we ran a time independent calculation and the default time step is 1.000E+08 shakes (1 second). If we had run a time dependent calculation (see, **centim**) we would have obtained results integrated between each of the time steps that we specified. In this case we only obtain result at the default time step and the 1.000E+08 indicates the time at the end of our default time step. Therefore in this case this number can be ignored.

When interpreting these results remember that they are integral results; all are integrated over the volume of each zone, and the energy range of each tally bin, and some are also integrated over the entire energy range to present a single result for each zone. **sentl 35** can be used to obtain results per unit volume or mass; these results will only be correct for zones that are symmetric about the z axis and finite in volume, to allow the code to correctly calculate the volume and mass of each zone. An alternative approach is to use the utility code **FLUXEDIT** that will divide the results by both the volume of each zone (assuming it is correct) and the energy width of each energy tally bin, to obtain results per cc and per MeV. This latter form is particularly useful if you want to see the real energy dependence of the fluence or flux; the shape is not at all obvious when integrated over each energy bin (as appears in the normal output). The **FLUXEDIT** results per cc are also useful if you want to use **TARTCHEK** to display results overlaid on your geometry.

Photon edit is per source neutron unless noted otherwise.

```
Error in photon energy balance is -9.385938E-14 percent.
```

```
Actual photon energy deposition
```

## Chapter 6: Example Problems

in MeV/zone by time step.

zone	total	Std dev	pct	1.000E+08
1	2.461E-03	7.13E-05	2.9	2.46E-03
2	7.248E-03	2.00E-04	2.8	7.25E-03
3	1.092E-02	2.29E-04	2.1	1.09E-02
4	1.491E-02	9.97E-05	.7	1.49E-02
5	1.788E-02	2.04E-04	1.1	1.79E-02
6	2.054E-02	2.98E-04	1.4	2.05E-02
7	2.237E-02	1.76E-04	.8	2.24E-02
8	2.363E-02	2.00E-04	.8	2.36E-02
9	2.406E-02	3.84E-04	1.6	2.41E-02
10	2.306E-02	2.38E-04	1.0	2.31E-02
11	8.252E-01	1.38E-03	.2	8.25E-01

  

	e leak	ebound	egain	edep	xdep
/neutron	8.251672E-01	.000000E+00	.000000E+00	1.670859E-01	1.674213E-01
/ photon	2.697286E+00	.000000E+00	.000000E+00	5.461664E-01	5.472627E-01

Expected value photon energy depositions.  
in MeV/zone by time step.

zone	total	Std dev	pct	1.000E+08
1	2.494E-03	7.88E-06	.3	2.49E-03
2	6.972E-03	3.38E-05	.5	6.97E-03
3	1.099E-02	4.29E-05	.4	1.10E-02
4	1.458E-02	7.28E-05	.5	1.46E-02
5	1.806E-02	7.22E-05	.4	1.81E-02
6	2.078E-02	5.46E-05	.3	2.08E-02
7	2.277E-02	5.45E-05	.2	2.28E-02
8	2.392E-02	4.80E-05	.2	2.39E-02
9	2.399E-02	4.25E-05	.2	2.40E-02
10	2.285E-02	3.01E-05	.1	2.29E-02
11	8.252E-01	1.38E-03	.2	8.25E-01

Photon tally type totals, by zone and time step.

zone	tt	total	dev.	pct.	1.00E+08
1	3	8.929E-03	5.7E-05	.6	8.93E-03
2	3	5.578E-02	2.2E-04	.4	5.58E-02
3	3	1.009E-01	3.7E-04	.4	1.01E-01
4	3	1.444E-01	5.6E-04	.4	1.44E-01
5	3	1.838E-01	6.0E-04	.3	1.84E-01
6	3	2.213E-01	7.7E-04	.3	2.21E-01
7	3	2.530E-01	7.2E-04	.3	2.53E-01
8	3	2.776E-01	6.6E-04	.2	2.78E-01
9	3	2.930E-01	6.2E-04	.2	2.93E-01
10	3	2.977E-01	5.1E-04	.2	2.98E-01
11	3	3.077E-01	4.7E-04	.2	3.08E-01

Census time is	1.000E+08	photon tally type				Water spheres
type	3	3	3	3	3	3
n	energy	zone 1	zone 2	zone 3	zone 4	zone 5
1	1.00E-04	.000E+00	.000E+00	.000E+00	.000E+00	.000E+00
2	1.36E-03	.000E+00	.000E+00	.000E+00	.000E+00	.000E+00
3	1.84E-03	.000E+00	.000E+00	.000E+00	.000E+00	.000E+00
43	2.03E+00	1.268E-03	7.594E-03	1.504E-02	2.424E-02	3.301E-02
44	2.89E+00	1.094E-03	7.383E-03	1.246E-02	1.687E-02	2.060E-02
45	4.11E+00	3.540E-04	2.391E-03	4.082E-03	5.607E-03	6.870E-03
46	5.20E+00	1.626E-03	1.007E-02	1.707E-02	2.296E-02	2.769E-02
47	6.58E+00	8.300E-04	5.585E-03	9.572E-03	1.304E-02	1.578E-02
48	8.33E+00	2.500E-05	1.950E-04	3.100E-04	4.110E-04	5.070E-04
49	1.06E+01	.000E+00	.000E+00	.000E+00	.000E+00	.000E+00
50	1.34E+01	.000E+00	.000E+00	.000E+00	.000E+00	.000E+00

## Chapter 6: Example Problems

Neutron edit.

Error in total energy balance is -3.61790E-10 percent.

All energies are in MeV.

Total source energy from mass difference q is..... -5.96650E-01  
Average energy leaked per source neutron is..... 8.63490E+00  
Average energy leaked per neutron leaking is..... 9.68630E+00  
Energy gained or lost by splitting at a boundary is. .00000E+00  
Energy gained or lost by thermal treatment is..... 8.91208E-15  
Energy deposits from neutrons only.  
Endothermic energy from mass difference q is..... 7.54642E-01  
Analog energy per source neutron deposited is..... 3.87620E+00  
Expected energy per source neutron deposited is..... 3.81326E+00

Local expected value energy depositions from neutrons only, in MeV/zone, by time step.

zone	total	Std dev	pct	1.000E+08
1	4.1035E-01	4.00E-05	.0	4.104E-01
2	4.0923E-01	1.06E-04	.0	4.092E-01
3	4.0591E-01	1.74E-04	.0	4.059E-01
4	4.0045E-01	2.32E-04	.1	4.004E-01
5	3.9345E-01	1.72E-04	.0	3.934E-01
6	3.8442E-01	2.17E-04	.1	3.844E-01
7	3.7352E-01	1.77E-04	.0	3.735E-01
8	3.6109E-01	1.62E-04	.0	3.611E-01
9	3.4657E-01	2.80E-04	.1	3.466E-01
10	3.2828E-01	2.58E-04	.1	3.283E-01
11	8.6349E+00	5.37E-03	.1	8.635E+00

Neutron tally type totals by zone and time step.

zone	tt	total	dev.	pct.	1.00E+08
1	2	1.028E+00	2.13E-04	.0	1.03E+00
2	2	1.109E+00	5.61E-04	.1	1.11E+00
3	2	1.209E+00	1.01E-03	.1	1.21E+00
4	2	1.308E+00	1.41E-03	.1	1.31E+00
5	2	1.429E+00	1.77E-03	.1	1.43E+00
6	2	1.503E+00	1.15E-03	.1	1.50E+00
7	2	1.520E+00	1.85E-03	.1	1.52E+00
8	2	1.488E+00	1.31E-03	.1	1.49E+00
9	2	1.393E+00	1.30E-03	.1	1.39E+00
10	2	1.211E+00	1.05E-03	.1	1.21E+00
11	3	8.915E-01	3.34E-04	.0	8.91E-01

Photon source / zone in MeV.

zone	total	Std dev	pct	1.000E+08
1	1.0222E-01	3.28E-04	.3	1.022E-01
2	9.8022E-02	6.82E-04	.7	9.802E-02
3	9.7500E-02	7.18E-04	.7	9.750E-02
4	9.5401E-02	8.55E-04	.9	9.540E-02
5	1.0671E-01	7.77E-04	.7	1.067E-01
6	1.0857E-01	6.96E-04	.6	1.086E-01
7	1.0798E-01	4.66E-04	.4	1.080E-01
8	1.0263E-01	3.50E-04	.3	1.026E-01
9	9.4419E-02	4.56E-04	.5	9.442E-02
10	7.8810E-02	5.99E-04	.8	7.881E-02

Local neutron energy deposition plus photon deposition (total energy deposited)

zone	total	Std dev	pct	1.000E+08
1	4.1285E-01	3.68E-05	.0	4.128E-01

```

2 4.1620E-01 9.51E-05 .0 4.162E-01
3 4.1690E-01 1.50E-04 .0 4.169E-01
4 4.1503E-01 2.01E-04 .0 4.150E-01
5 4.1151E-01 1.56E-04 .0 4.115E-01
6 4.0520E-01 2.20E-04 .1 4.052E-01
7 3.9629E-01 1.72E-04 .0 3.963E-01
8 3.8501E-01 1.72E-04 .0 3.850E-01
9 3.7057E-01 2.64E-04 .1 3.706E-01
10 3.5113E-01 2.51E-04 .1 3.511E-01
11 9.4601E+00 5.17E-03 .1 9.460E+00

```

Multi-band n-gammas by zone and time step.

```

zone total 1.000E+08
1 9.30150E-05 9.301E-05
2 5.56694E-04 5.567E-04
3 1.35772E-03 1.358E-03
4 2.65147E-03 2.651E-03
5 9.06167E-03 9.062E-03
6 1.28056E-02 1.281E-02
7 1.39546E-02 1.395E-02
8 1.33785E-02 1.338E-02
9 1.10020E-02 1.100E-02
10 6.33878E-03 6.339E-03

```

```

Census time is 1.000E+08 neutron tally type Water spheres
type 2 2 2 2 2
n energy zone 1 zone 2 zone 3 zone 4 zone 5
1 1.31E-09 3.574E-03 2.192E-02 5.339E-02 9.452E-02 1.808E-01
2 8.32E-08 7.812E-04 5.290E-03 1.241E-02 1.909E-02 1.583E-02
3 4.23E-07 3.862E-04 2.342E-03 6.138E-03 1.002E-02 1.378E-02
4 2.09E-06 2.593E-04 1.843E-03 4.661E-03 7.976E-03 1.085E-02

43 1.01E+01 2.617E-03 6.550E-03 9.385E-03 1.160E-02 1.357E-02
44 1.10E+01 5.111E-03 1.320E-02 1.874E-02 2.274E-02 2.542E-02
45 1.20E+01 5.628E-03 1.545E-02 2.382E-02 3.035E-02 3.560E-02
46 1.31E+01 4.710E-03 1.325E-02 2.044E-02 2.650E-02 3.131E-02
47 1.39E+01 9.592E-01 8.819E-01 8.108E-01 7.459E-01 6.857E-01
48 1.41E+01 .000E+00 .000E+00 .000E+00 .000E+00 .000E+00
49 1.44E+01 .000E+00 .000E+00 .000E+00 .000E+00 .000E+00
50 1.52E+01 .000E+00 .000E+00 .000E+00 .000E+00 .000E+00

```

Average particle collision gain by zone and time step.

```

1.00000E+08
1-4.65800E-03
2-4.85900E-03
3-5.53200E-03
4-6.64900E-03
5-1.27840E-02
6-1.64630E-02
7-1.77410E-02
8-1.65910E-02
9-1.40870E-02
10-9.18100E-03

```

-----  
424.00 seconds - Finished run  
-----

### Water Phantom

The next example is for a water phantom, represented by a cylinder 30 cm long and 20 cm in radius. The cylinder is divided into zones by planes each cm of length and cylinders each cm of radius. The result is a problem involving 600 interior zones, plus 3 exterior non-re-entrant zones; one zone outside the 20 cm cylinder, and one zone at each end of the

cylinder at 0 and 30 cm. The zoning for this problem was done interactively using the **TARTVIEW** code, that will soon be available for your use. Using **TARTVIEW** interactively this problem can be completely zoned in just a few minutes. If you tried to do this by hand it would be very difficult and extremely error prone.

You may ask: why did we use a cylindrical R-Z geometry, instead of a rectilinear grid in X, Y, Z? Three reasons: 1) finely zoning 3-D (X, Y, Z) space would require many more zones than we have used to model our 2-D R-Z geometry, 2) in this case we will use the interactive graphics program **TARTCHEK** to analyze the output results and 2-D results can all be presented on a single plane in one picture, so that we can see all of the results immediately; with 3-D (X, Y, Z) results we would have to scan many views before being able to see overall trends, 3) for this case we do not need 3-D results, so why complicate the problem. In any problem that you set up look for symmetries and decide how much detail you really need to achieve your goal, and if at all possible avoid 3-D calculations. Today we really have a big advantage as far as using interactive graphics to analyze output results, but this is really hard to take advantage of if your results are in 3-D. Here we discussed 2-D R-Z geometry, but in general any 2-D results can all be presented on a single plane, so that you can see and try to understand them based on just one single picture.

In our phantom problem there is a 14.1 MeV neutron or photon source incident on the axis of the cylinder at the 30 cm end of the cylinder. The source can either be monodirectional, straight down the central axis of the cylinder, or it can be spread over a narrow angular interval to simulate the distribution from an accelerator. We could also vary the spectrum of neutrons or photons. The most difficult part of preparing the TART95 input for this problem was involved in correctly defining the geometry. Once this is done it is very simple to modify the source definition (in particle type, energy, direction, and position) to allow this input deck to be used to study many different effects. We can also easily modify surface locations (to change dimension), material, or any other input parameter, to see the effects of neutrons or photons incident on a cylinder of material of any dimensions.

The objective is to define the particle fluence (flux) and energy deposition within the phantom. In particular we wish to examine the results along, or near, the axis of the cylinder, as well as the spread in the beam in space as it penetrates progressively further into the phantom. The results near the axis will tell us how much flux or deposition can be delivered to a hypothetical tumor located some distance from the source, inside the phantom. The spread in the beam in space will allow us to estimate the collateral damage throughout the phantom. By varying the angular interval of the beam we can define the sensitivity of the results to the beam's angular distribution.

The below input deck for this problem illustrates additional features of TART95. The deck has been edited, as explained below, to only include the information that we will discuss here. The deck has been designed to allow you to easily switch back and forth between

neutron and photon sources, and the deck illustrates how to define source energies and angular distributions, which were not discussed in the above applications.

```
* =====
name Water Phantom
box t32 Water Phantom
* =====
*
* TART Input Deck Generated Using Program TARTVIEW (93-1)
*
* =====
*
* PROBLEM: WATER PHANTOM = CYLINDER, 30 CM LONG, 15 CM
RADIUS
*
* See below on how to switch between neutron and photon
source
*
* =====
*
* Surface Definitions
* 2-D (R-Z) - Planes, Cylinders, Spheres and Cones Allowed
* 1) Surface Number - used later to Define Zone Boundaries
* 2) Position - Radius Z0 X0 Y0
*
* =====
```

This section defines the bounding surfaces: 31 planes, in 1 cm steps, between 0 and 30 cm, and 20 cylinders, in 1 cm steps in radius between 1 and 20 cm.

```
zplane      1  0.00000e+00
zplane      2  1.00000e+00
zplane      3  2.00000e+00
zplane      4  3.00000e+00
zplane      5  4.00000e+00
.
.
zplane     26  2.50000e+01
zplane     27  2.60000e+01
zplane     28  2.70000e+01
zplane     29  2.80000e+01
zplane     30  2.90000e+01
zplane     31  3.00000e+01
cylz       32  1.00000e+00  0.00000e+00  0.00000e+00
cylz       33  2.00000e+00  0.00000e+00  0.00000e+00
cylz       34  3.00000e+00  0.00000e+00  0.00000e+00
cylz       35  4.00000e+00  0.00000e+00  0.00000e+00
.
.
cylz       48  1.70000e+01  0.00000e+00  0.00000e+00
```

```

cylz      49  1.80000e+01  0.00000e+00  0.00000e+00
cylz      50  1.90000e+01  0.00000e+00  0.00000e+00
cylz      51  2.00000e+01  0.00000e+00  0.00000e+00
* =====
*
* Zone Definitions
* 1) Zone Number
* 2) Bounding Surface Number and Sign of Vector Normal
*    to the Surface for Particles Leaving the Zone.
*    2) Is repeated to include all Bounding Surfaces.
*
* =====

```

The next section is very long, since it defines the bounding surfaces for each of the 603 zones. Only a few of the zone definitions are included here. Each zone on the axis of the cylinders requires 3 bounding surfaces to define it: the inner 1 cm radius cylinder and 2 bounding planes. Each off axis zone requires 4 bounding surfaces to define it: an inner and outer cylinder and 2 bounding planes. The last three, exterior non-re-entrant, zones only require one bounding surface. The following sections define water and assign it to all 600 interior zones; this input is similar to what was described above, so we will not discuss it in detail here.

```

jb      1  -30   31   32
jb      2  -30   31  -32   33
jb      3  -30   31  -33   34
jb      4  -30   31  -34   35
jb      5  -30   31  -35   36
jb      6  -30   31  -36   37
jb      7  -30   31  -37   38
jb      8  -30   31  -38   39
jb      9  -30   31  -39   40
.
.
jb     590   -1    2  -40   41
jb     591   -1    2  -41   42
jb     592   -1    2  -42   43
jb     593   -1    2  -43   44
jb     594   -1    2  -44   45
jb     595   -1    2  -45   46
jb     596   -1    2  -46   47
jb     597   -1    2  -47   48
jb     598   -1    2  -48   49
jb     599   -1    2  -49   50
jb     600   -1    2  -50   51
jb     601    1
jb     602  -31
jb     603  -51
* =====
*

```

```

* Material Definitions
* 1) Material Number
* 2) Density (grams/cc)
* 3) Atom %
* 4) Isotope I.D. (ZZZAAA), e.g., 92238 for U-238
* 3) and 4) can be repeated in pairs to define
* composite materials
*
* =====
* Water
matl 1 1.00000e+00 6.66667e+01 1001 3.33333e+01
8016
* =====
*
* Assignment of Materials to Zones
* 1) Material Number
* 2) Zone Containing Material - Can be Repeated
*
* =====
* Water
matz 1 1 thru 600
* =====
*
* Definition of Running Conditions and Output Edit Options
*
* =====
* Same for Neutrons or Photons
*
* =====
* 2) Number of batches
sentl 2 10
* 3) Histories per Batch
sentl 3 100000

```

The follow sections allow us to switch back and forth between neutron and photon sources. The deck is now set up to run neutrons. To switch to photons merely comment out the options in the first second and activate the photon options in the following section.

These sections include options that we did not encounter in the above examples including:

1) The use of **sentl 1** to explicitly specify the particles to be tracked. For neutrons we indicate **sentl 1 0**, which means neutrons and neutron induced photon production. For photons we indicate **sentl 1 2**, which means only photons.

2) In each case we want to tally pathlength (fluence) in each zone. For neutrons this is done by specifying **ltype 2 1 thru 600**, indicating fluence in zones 1 through 600. For photons we specify **ltypeg 2 1 thru 600**, to obtain similar results.

3) In each case we will use a monoenergetic 14.1 MeV source energy. For neutrons we specify **sentl 4 14.1**, and for photons **sentl 17 14.1**.

4) In each case we will use a point source located on the axis of the cylinders, just inside the 30 cm end of the cylinder. For neutrons we specify **source1 0.0 0.0 29.9**, and for photons **s1g 0.0 0.0 29.9**. Both specify  $(x, y) = (0, 0)$  = on the central axis, and  $z = 29.9$ , just inside the 30 cm end of the cylinders.

5) The angular distribution will initially be monodirectional, along the z axis of the cylinders. For neutrons we specify **sentl 6 0.0** and **sentl 7 -1.0**, and for photons **sentl 41 0.0** and **sentl 42 -1.0**. **sentl 7** or **42** defines one end of the angular distribution and **sentl 6** or **41** defines the cosine width of the distribution. Therefore in this case both specify a monodirectional angular distribution, directed straight down the z axis of the cylinders.

Later we will introduce an angular spread in the source by changing **sentl 6** or **41** from 0.0 to 0.01, which will give us an angular distribution that is uniformly distributed between  $\text{Cos} = -1.0$  and  $-0.99$ .

6) For neutrons there are additional specifications, all of which have already been discussed in the problems covered above, and as such will not be discussed in detail here.

```
* =====
*
* Neutrons Incident
*
* =====
* 1) Transport (neutrons + induced photons) (0)
sentl 1 0
* Scalar Flux output
ltype 2 1 thru 600
* Source energy
sentl 4 14.1
* Source position = point source just inside end of cylinder
source1 1 0.0 0.0 29.9
* Source angular distribution
sentl 6 0.0
sentl 7 -1.0
* minimum energy
sentl 8 1.307e-09
* Thermal Scattering and temperature (MeV)
sentl 39 1
* room temperature in all zones
emin 2.53e-08 1 thru 600
* =====
*
* Photons Incident
*
* =====
```

```

* 1) Transport (photons) (0)
* sentl 1 2
* Scalar Flux output
* ltypeg 2 1 thru 600
* Source energy
* sentl 17 14.1
* Source position = point source just inside end of cylinder
* slg 1 0.0 0.0 29.9
* Source angular distribution
* sentl 41 0.0
* sentl 42 -1.0
end

```

Performing any number of calculations is straightforward and very fast. The problem is that the output results are so voluminous that it is very difficult to analyze the results. Consider the size of the output results of the simple 11 zone water sphere problem discussed above. For this problem we will have results for 603 zones and the output file is extremely large; it is so large we will not even consider presenting a portion of it here. In any of the 600 interior zones we can define the integral fluence and deposition as well as the energy dependent results based on the tally bins. For example, zone 300 is located on the axis of the cylinder 15 cm from the source. We can compare neutron and photon energy deposition, with and without an angular interval for the source. The results are,

```

neutron, monodirectional beam..... 0.0364 MeV/cc
photon, monodirectional beam..... 0.0531 MeV/cc
neutron, uniform in cos = 1.0 to 0.99..... 0.0103 MeV/cc
photon, uniform in cos = 1.0 to 0.99..... 0.0130 MeV/cc

```

From the results we can see that the photon deposition is higher than the neutron deposition, and that compared to the hypothetical monodirectional source results, introducing an angular spread in the beam causes a rather large decrease in the energy deposition; for the spread used here the deposition decreased by a factor of 3 to 4. We can obtain simple results like these from the output listing, but it is very difficult to easily see overall trends in the results.

This is where the codes **FLUXEDIT** and **TARTCHEK** come to the rescue. **FLUXEDIT** can be used to read the TART95 output results and extract results for fluence and energy deposition for each zone, normalize per cc of volume of each zone, and put these results in a simple format that can be read by **TARTCHEK** to overlay the results onto the phantom geometry. With this approach instead of spending days or even weeks trying to figure out what the results mean, you can immediately "see" the results, and be able to understand the trends in fluence and deposition.

Figs. 3 through 6 (see the appendix) present neutron deposition and fluence results, without and with an angular spread in the source. Figs. 7 through 10 present similar results for photons.

Figs. 3 and 4 present neutron results without any angular spread in the source. Fig. 3 illustrates the deposition and fig. 4 the fluence. From fig. 4 we can see that neutrons are quite invasive, in that the neutron fluence spreads out over a considerable portion of the phantom. In contrast from fig. 3 we can see that the deposition is much more closely grouped about the z axis. Figs. 5 and 6 show similar neutron results for deposition and fluence, respectively, with an angular source spread uniform between  $\text{Cos} = -1.0$  and  $-0.99$ . Comparing the results without (figs. 3 and 4) and with (figs. 5 and 6) a spread in the angular source we can see the large decrease in both deposition and fluence along the z axis when the source has an angular spread.

Figs. 7 and 8 present photon results without any angular spread in the source. Fig. 7 illustrates the deposition and fig. 8 the fluence. From fig. 8 we can see that compared to neutrons fluence (fig. 2), photons are much less invasive and much more tend to go in the direction that you point them. From fig. 8 we can see that the photon deposition hardly spreads out from the z axis at all. By comparing the neutron and photon results you will begin to understand why photons are used much more often than neutrons in radiation therapy; photons are much less invasive, tend to go where you point them, and as such cause much less collateral damage. Figs. 9 and 10 show similar photon results for deposition and fluence, respectively, with an angular source spread uniform between  $\text{Cos} = -1.0$  and  $-0.99$ . Comparing the results without (figs. 7 and 8) and with (figs. 9 and 10) a spread in the angular source we can see the large decrease in both deposition and fluence along the z axis when the source has an angular spread.

Both neutron and photon results illustrate the importance of realistically modeling the angular distribution of the source, since the results can be very sensitive to this angular distribution.

Hopefully just these few figures illustrate the usefulness of interactive graphics to analyze TART95 output results. Using **TARTCHEK** to overlay your results on your geometry can provide more insight in a few minutes than you could gain in days or even weeks trying to see overall trends based on the TART95 output listing.

However, we should point out that there is a penalty that you may have to pay if you wish to use this approach. TART95 only produces results on a zone by zone basis. Therefore if you want to see smoothly varying results you will have to finely zone your problem, as we have done in the above problem using 600 zones.

### **Pulsed Spheres**

Livermore had an extensive experimental measurement program to measure the time dependent leakage of neutrons from spheres of various materials. In each case the sphere of material had a hole bored in it to allow a neutron generator to generate neutrons in a small region close to the origin of the sphere. Neutrons were generated in time dependent pulses; hence the name "Pulsed Spheres". TART95 does not describe the generation of neutrons by the generator, it starts from a given neutron measured source distribution and

transports the neutrons from the position where they are generated. The neutron generator is included in the geometric description because it can scatter or absorb neutrons, which could effect measured results. Detectors were placed at different angles around the sphere, some distance from the spheres, to measure the time dependent response to each pulse of neutrons. These measurements were very useful to allow us to gain insight into the physical processes involved, the relative importance of each, and how to model them both in our evaluated data and transport calculations.

Figs. 11 and 12 (see appendix) illustrate a typical pulsed sphere based on **TARTCHEK** output plots; in this case we show the Pb, 1 mean free path (to 14.1 MeV neutron) thick pulsed sphere. From fig. 11 we can see a sphere of Pb with a hole bored into it to accommodate the neutron generator, which extends well outside the sphere along the negative z axis. Fig. 11 shows the zones in the problem, where each different color corresponds to a different zone. We have used the **TARTCHEK** option What Zone? to indicate a few zone numbers (the white boxes with enclosed zone numbers), and the option What Material? to indicate the two zones containing the Pb (material 8 in the yellow boxes). Comparing zone numbers and material we can see that the Pb (material 8) is in zones 19 and 20. In fig. 12 we show a different view of exactly the same problem. In going from fig. 11 to 12 we have used the **TARTCHEK** option Zone/Matter, to switch between showing zones as different colors (fig. 11) to showing materials as different colors (fig. 12). Again we use the option What Material? to indicate a few materials: material 8 is the Be, 6 is air, 5 and 7 are different parts of the neutron generator.

This is a good example to examine because it uses some of the more complicated input options, particularly to describe the neutron source distribution and output tallies. It will also illustrate what the input decks used to look like before we started using **TARTVIEW** to interactively prepare and annotate the input decks. The following input deck has almost no comments describing the meaning of the input.

This deck is for a 0.8 mean free path (to 14.1 MeV neutrons) sphere of Be<sup>9</sup>. This is one of the more complicated pulsed sphere input decks, because it includes a description of the entire accelerator that was used to generate the neutron source; some of the other pulsed sphere decks do not describe the accelerator at all. Here we need not be concerned about the detailed description of the geometry and materials. Here we will concentrate mostly on describing the source which is a function of space, energy, direction and time. This is about as general as it gets, so that if you can understand this source, you can probably understand any TART95 source description. We will also discuss a few tally options. We will only discuss the overall problem in sufficient detail to understand the objective of the measurements.

The central region includes a spherical shell of Be<sup>9</sup> with inner radius 8.001 cm and outer radius of 12.573 cm. There is a hole in the spherical shell along the negative z axis to allow the neutron generator to be inserted. The regions outside the Be<sup>9</sup> shell contain only air (except for the accelerator assembly along the negative z axis). The entire geometry is surrounded by spherical surfaces 722.0 and 722.5 cm from the origin (surfaces 33 and 34).

## Chapter 6: Example Problems

There are also cones at 35 and 11 degrees from the z axis (surfaces 35 and 37). These are used to define zone 26, our detector zone. The neutron source is a function of space, energy, direction, and time, and will be discussed in detail below. The source is on a disk 0.6 cm in radius symmetric about the z axis, close to the origin of the spheres, at  $z = -0.4756$ . Pulses of neutrons are generated on this disk and the neutrons are tracked. Every time a neutron enters the detector zone its coordinates are written to a binary output file. Included in the input are (x, y, z, alpha, beta, gamma, speed, time). After the completion of the TART95 run, this binary file is read by a utility code to fold in a detector response and compare the calculated results to the experimental results. These are the final results that we want.

Because of the large difference in size between the central region out to 12.573 cm and the detector region out to 722.5 cm if you use **TARTCHEK** to view this problem, you will initially see very little; basically just the large sphere of air. To see the central region you should zoom in until you can see the Be<sup>9</sup> shell and the accelerator assembly.

```
name    be-9                0.8 mfp
box t64
zplane 1 0
zplane 2 -3.58800e-01
zplane 3 -3.98300e-01
zplane 4 -4.74500e-01
zplane 5 -4.75500e-01
zplane 6 -1.19000e+01
zplane 7 -2.49000e+01
zplane 8 -2.74000e+01
zplane 9 -7.478
zplane 11 -13.3858
zplane 12 -17.0688
zplane 13 -25.4508
zplane 14 -29.2608
surf 15 2.85496 0 0 0 0 1 1
surf 16 2.1345
surf 17 3.81 0 0 0 0 1 1
surf 18 2.49088 0 0 0 0 1 1
surf 19 2.8575 0 0 0 0 1 1
surf 20 3.4925 0 0 0 0 1 1
surf 21 3.6513 0 0 0 0 1 1
surf 22 1.03100e+00 0. 0. 0. 0. 1.e+00 1.e+00
sphere 23 8.001 0 0 0
sphere 24 12.573 0 0 0
sphere 25 20.066
sphere 26 27.94
surf 27 0. -4.76000e-03 1.19300e+01 0. 0. 1.e+00 1.e+00
surf 28 0. -4.76000e-03 1.11890e+01 0. 0. 1.e+00 1.e+00
surf 29 7.50000e-01 0. 0. 0. 0. 1.e+00 1.e+00
surf 30 6.30000e-01 0. 0. 0. 0. 1.e+00 1.e+00
sphere 31 721 0 0 0
sphere 32 721.5 0 0 0
sphere 33 722 0 0 0
sphere 34 722.5 0 0 0
cone 35 35 0
cone 37 11 0
cone 38 139 0
jb 1 -5 4 22
```

## Chapter 6: Example Problems

```
jb 2 -4 3 22
jb 3 -3 2 22
jb 4 28 5 -6
jb 5 27 -28 5 -7
jb 6 28 -7 -29 6
jb 7 29 -30 6 -7
jb 8 30 6 -7
jb 9 27 7 -8
jb 10 15 -16 9 24
jb 11 17 -18 -24 1 -11
jb 12 17 -19 11 -12
jb 13 17 -20 12 -13
jb 14 17 -21 13 -14
jb 15 24 -15 -23 1
jb 16 -23 24 -1
jb 17 -24 25 1 -17
jb 18 -24 25 -1
jb 19 -17 1 -25 26
jb 20 -25 26 -1
jb 21 -31 32 37 -1
jb 22 -32 33 37 -1
jb 23 -33 34 37 -1
jb 24 -31 32 -37 35 -1
jb 25 -32 33 -37 35 -1
jb 26 -33 34 -37 35 -1
jb 30 -31 32 -1 -35
jb 31 -32 33 -1 -35
jb 32 -33 34 -1 -35
jb 33 -31 32 -38 1
jb 34 -32 33 -38 1
jb 35 -33 34 -38 1
jb 36 -31 32 38 1
jb 37 -32 33 38 1
jb 38 1 -33 34 38 1
jb 39 -26 31 -1
jb 40 31 1 -17 -26
jb 41 -14 21 8
jb 42 -8 21 13 -27
jb 43 -13 20 12 -27
jb 44 -12 19 11 -27
jb 45 -11 18 -24 1 -27
jb 46 24 16 9 -27
jb 47 -15 23 5
jb 48 -5 2 23 -22
jb 49 -2 1 23
jb 50 -1 23
jb 52 -9 15 5 -27
jb 53 14 17 31
jb 54 -34
matl 1 1.6 5e-01 1003 5e-01 22000
matl 2 18.8 1 74000
matl 3 8.0000e+00 4.5000e-01 26000 3.2000e-01 29000 8.0000e-02 8016
&
2.0000e-02 13027 6.0000e-02 6012 1.3000e-01 1001
matl 5 2.7000e+00 1.0000e+00 13027
matl 6 1.2880e-03 7.8850e-01 7014 2.1150e-01 8016
matl 7 8.0000e+00 5.0000e-01 13027 5.0000e-01 29000
matl 8 1.84 1 4009
matz 1 1
matz 2 2
```

```

matz 3 3
matz 5 5 7 10 11 12 13 14
matz 6 4 6 8 17 21 22 23 24 25 26 30 31 49 50 52 53 18 19 20
matz 6 32 33 34 35 36 37 38 39 40 41 42 43 44 45 46 47 48
matz 7 9
matz 8 16 15
matz 0 54
sentl 1 1 2 20 3 50000 22 1 23 1 28 1 8 -1.0e-05

```

Zones 15 and 16 are the  $\text{Be}^9$  shell, divided into two hemispheres; zone 15 is the hemisphere in the negative z direction and 16 is the hemisphere in the positive z direction. The following **reacted** input requests output for all reactions that occur within the  $\text{Be}^9$ . The above **sentl 23** input specifies that the reaction output will only be the total reactions, not by energy tally group.

```

reacted 15
reacted 16
source2 1 0. 6.00000e-01 -4.756e-01 0. 0. 0.

```

The following **ltype** input specifies that every time a neutron enters zone 26 it should be tallied using tally type 12, which means write its coordinates to a binary file. The **centim** input sets a maximum time of  $2.0\text{E}+4$  shakes (200 microseconds) for tracking neutrons. The **cphotal** input changes from the standard 50 neutron tally bins, to a new set of bins clustered at high energy between about 1 keV (group 91) and 5.0 MeV (group 144), to obtain better resolution in the higher energy results.

```

ltype 12 26
centim 2.00000e+04
cphotal 1 1 91 96 97 98 99 100 101 102 103 104 105 106 107 108
109 110
cphotal 18 111 112 113 114 115 116 117 118 119 120 121 122 123 124 125
126 127
cphotal 35 128 129 130 131 132 133 134 135 136 137 138 139 140 141 142
143 144

```

The source description for this problem is quite complicated. The source is a function of position, direction, energy and time. The above **source2** input defines the spatial distribution to be uniform over a cylindrical disk centered on the z axis, 0.6 cm in radius, located at  $z = -0.4756$  cm. The following section defines the direction and energy dependence. The source is correlated in direction and energy.

The following **maec** input defines the relative strength of the source in cosine intervals. For example, the first **maec** input says that in the first interval, the strength is 0.0319, in the cosine interval 1.0 to 0.93969, the second **maec** input has strength 0.09091 in the cosine interval 0.93969 to 0.76604, etc., until the ninth **maec** input has strength 0.02804 in the cosine interval -0.93969 to -1.0. In combination the nine **maec** inputs define the strength of the source over the entire cosine range +1 to -1.

The **maeeh** input defines the energy spectrum in each cosine interval. For example, the first two **maeeh** input lines refers to **maec** input 1, the cosine interval 1.0 to 0.93969, and

defines a spectrum between 14.3 and 15.56 MeV in terms of a table of seven (strength, energy) pairs. Similarly, each successive **maeeh** input refers to the **maec** input just preceding it, and defines the spectrum in the corresponding cosine interval. In combination the nine **maec** and corresponding **maeeh** inputs define the strength of the source over the entire cosine range +1 to -1, and energy range.

Note, the strength of the source is NOT defined by integrating over all cosine and energy intervals. The strength is defined by the relative weight assigned to each cosine interval by **maec** input. The distribution in cosines is integrated, and normalized for sampling a cosine. Once a cosine interval has been selected the spectrum in that interval, defined by **maeeh** input, is sampled as a normalized distribution.

```
* seagrave source spectrum
maec 1 .03199 1.0 .93969
maeeh 1 1 1.63 14.3 17.94 14.5 30.19 14.7 22.84 14.9
maeeh 1 5 14.68 15.1 12.72 15.3 0.0 15.56
maec 2 .09091 .93969 .76604
maeeh 2 1 1.45 14.2 14.49 14.4 30.43 14.6 23.19 14.8
maeeh 2 5 13.77 15.0 9.42 15.2 7.25 15.4 0.0 15.6
maec 3 .13834 .76604 .5
maeeh 3 1 5.41 14.2 40.15 14.4 30.89 14.6 17.76 14.8
maeeh 3 5 5.79 15.0 0.0 15.1
maec 4 .16644 .5 .17365
maeeh 4 1 57.97 14.196 42.03 14.396 0.0 14.596
maec 5 .17365 .17365 -.17365
maeeh 5 1 100. 14.085 0. 14.095
maec 6 .15942 -.17365 -.5
maeeh 6 1 100. 13.805 0. 13.815
maec 7 .12743 -.5 -.76604
maeeh 7 1 100. 13.555 0. 13.565
maec 8 .08179 -.76604 -.93969
maeeh 8 1 100. 13.375 0. 13.385
maec 9 .02804 -.93969 -1.000
maeeh 9 1 100. 13.265 0. 13.275
```

The neutron source distribution is also time dependent, as defined by the following **timspec** input. The time distribution starts with zero strength at time zero, increases to a maximum at .746 shakes (0.01746 microseconds) later, and then decreases to zero at 7.459 shakes (0.07459 microseconds). This table defines the strength of the distribution at 22 times between zero and 7.459 shakes. The input is a table index, e.g., 1, 3, 5, etc., followed by pairs of (neutron source strength, time), with two pairs per line; generally a **timspec** input line can contain any number of pairs. The neutron source time distribution is sampled from this table assuming linear interpolation between the tables values.

```
timspec 1 0. 0. 1.26000e+01 3.17400e-01
timspec 3 3.78000e+01 7.93500e-01 5.67000e+01 9.52200e-01
timspec 5 9.45000e+01 1.11100e+00 2.39400e+02 1.27000e+00
timspec 7 3.49090e+03 1.42800e+00 3.70825e+04 1.58700e+00
timspec 9 1.54228e+05 1.74600e+00 1.26963e+05 1.90400e+00
timspec 11 2.45054e+04 2.06300e+00 7.46060e+03 2.22200e+00
timspec 13 3.88780e+03 2.38100e+00 2.15500e+03 2.53900e+00
timspec 15 1.17200e+03 2.69800e+00 6.61600e+02 3.01500e+00
timspec 17 3.52900e+02 3.49100e+00 1.57500e+02 4.28500e+00
```

```
timspec 19 7.56000e+01 5.07800e+00 3.15000e+01 5.87200e+00
timspec 21 6.30000e+00 6.66500e+00 0. 7.45900e+00
end
```

Much of the output is similar to what has already been described above for other problems, so we will not discuss it here. The only major difference in the output is the list of reactions in the Be<sup>9</sup> that was requested by **reacted** input. The results are included below. The titles for the columns refer to n,pglv = (n, proton and gamma emitted) to a level, similarly for deuteron, triton and alpha emission. 4009 refers to Be<sup>9</sup>. Since this is the only material in these zones this is the only output. Generally there would be output for each constituent of the material, with each identified by its ZA (1000\*Z + A, Be<sup>9</sup> = 4009). The results include both the expected number of reactions and the standard deviation (dev in the below table).

Non-elastic minus absorption, by zone and time step.

```
zone total
      2.00E+04
15 1.64957E-01 1.65E-01
16 1.56935E-01 1.57E-01
```

Absorption minus fission, by zone and time step.

```
zone total
      2.00E+04
15 1.42285E-02 1.42E-02
16 1.40027E-02 1.40E-02
```

Census time is 2.000E+04 be-9 0.8 mfp

Reaction edit totals by zone and time step.

zone	elastic	n,2ng	n,pglv	n,dglv	n,tglv	n,tglv	n,aglv	n,g
15	4009	4009	4009	4009	4009	4009	4009	4009
	1.874E+00	1.650E-01	3.081E-08	.000E+00	2.546E-03	1.246E-03	1.037E-02	6.822E-05
dev	5.293E-03	1.692E-04	1.115E-09	.000E+00	1.567E-06	7.692E-07	2.573E-05	1.196E-07
zone	elastic	n,2ng	n,pglv	n,dglv	n,tglv	n,tglv	n,aglv	n,g
16	4009	4009	4009	4009	4009	4009	4009	4009
	1.875E+00	1.569E-01	2.585E-05	.000E+00	3.181E-03	1.377E-03	9.352E-03	6.765E-05
dev	6.216E-03	1.338E-04	2.075E-08	.000E+00	2.110E-06	9.288E-07	2.409E-05	1.407E-07

## Lead Shield

The next example is intended to demonstrate the use of weights. It is intentionally geometrically simple, so that we can focus on the results rather than the geometry. The problem involves a neutron induced fission spectrum incident normally (monodirectional) on a slab of lead 150 cm (5 feet) thick. What we are interested in is the transmission of neutrons and neutron induced photons through the lead.

The lead slab has been divided into 10 zones, each 15 cm thick. To close the geometry in 3-D we surround the lead by a cylinder 500 cm in radius. The source will be a point source on the axis of the cylinder just inside the first zone.

The entire problem involves 13 zones: 1 through 10 are the layers of lead, 11 is the zone outside the lead near the source (the reflection zone), 12 is the zone outside the lead away from the source (the transmission zone), and 13 is the zone outside the bounding cylinder.

To determine the transmission we need merely look at the results for zone 12, the transmission zone. We should also check the results in zone 13, to insure that the radius of the bounding cylinder is large enough to prevent any significant amount of radial leakage.

When neutron or photon statistical weights (**weight** or **wgtgam**) are used in TART95, the weights of adjacent zones can only be in the ratios 1/2, 1, or 2. As a particle crosses the boundary between two adjacent zones, the product of the weight and number of particles is conserved. As particles transport from one zone into another, if the weight of the last and new zone are the same transport continues without any changes. If the weight of the new zone is 1/2 times the weight of the previous zone, the particle is split into two particles each of half the weight of the original particle, and each is then independently transported. If the weight of the new zone is 2 times the weight of the previous zone, with probability 1/2 the neutron history will be terminated (Russian roulette) or its weight will be doubled. If the particle survives Russian roulette transport of the particle continues. Note, when entering a zone of 1/2 the weight of the previous zone, by creating 2 times as many particles each of 1/2 their original weight this method exactly conserves the product of particles times weight. When entering a zone of 2 times the weight of the previous zone, by statistically allowing 1/2 the particles to survive with 2 times their original weight this method statistically conserves the product of particles times weight.

When multipliers are used for neutron induced photon production (**gpwgt**) the multiplier in any zone can be any positive value, i.e., unlike **weight** and **wgtgam** it need not be a power of 2. For **gpwgt** other than 1 (the default value), following each neutron collision the weight,  $w$ , multiplies the number of photons generated, and each photon is tracked and scored with weight  $1/w$  (note, the multiplier,  $w$ , must be positive to avoid infinite weight).

Weights **weight** and **wgtgam** are used to improve statistics by encouraging particles to move from zones of larger weight toward zones of lower weight. For example, in our lead slab we would like to encourage particles (both neutrons and neutron induced photons) to move deeper into the slab toward the transmission zone. In this problem if we use weights that decrease for layers of lead successively further from the source, the splitting/Russian roulette used by TART95 will both: 1) create additional histories of lower weight as long as the particles keep moving deeper into the lead, and 2) kill off histories that start moving away from the transmission zone, thereby spending less computer time tracking particles that have a smaller chance of reaching the transmission zone. The result should be that more histories, of smaller weight, reach the transmission zone and improve the statistical accuracy of our results.

As used by TART95, weights **weight** and **wgtgam** are only relative. For example, if we want to encourage particles to move from a source in zone 1, through zones 2, 3, 4, and into zone 5, we can define weights that differ by a factor of 2, becoming progressively smaller from zone 1 through 5. To accomplish this the weights of zone 1 through 5 could be defined as 16, 8, 4, 2, 1, or equivalently we could have used weights, 1.0, 0.5, 0.25, 0.125, 0.0625; the choice is yours. In contrast, the neutron induced photon production multiplier **gpwgt** is absolute. For example, if the multiplier in a zone is 10, following every

neutron collision 10 times as many photons are generated as would have been with the default multiplier of 1.

The below deck will be used to perform three different calculations: 1) a preliminary calculation, without weights, of 10 batches of 1000 histories each, 2) without weights, same number of batches with 10,000 histories per batch, 3) with weight, 10 batches, 1000 histories per batch. The below deck as shown is set up to run case 1). To switch to case 2) we need merely change **sentl 3 1000** to **sentl 3 10000**. To switch to case 3) we activate the presently commented out **weight** input and insure we have **sentl 3 1000**.

```

* =====
name      Lead Shield
box o84   Lead Shield
* =====
*
* TART Input Deck Generated Using Program TARTVIEW (93-1)
*
* =====
*
* Neutron Fission spectrum incident on lead planes
*
* =====
*
* Surface Definitions
* Planes
* 1) Surface Number - used later to Define Zone Boundaries
* 2) Z Position - Z0
* Cylinder
* 1) Surface Number - used later to Define Zone Boundaries
* 2) Radius
*
* =====
zplane   1   0.0
zplane   2  15.0
zplane   3  30.0
zplane   4  45.0
zplane   5  60.0
zplane   6  75.0
zplane   7  90.0
zplane   8 105.0
zplane   9 120.0
zplane  10 135.0
zplane  11 150.0
cylz     12 500.0
* =====
*
* Zone Definitions - each zone is a lead layer
*
* 1) Zone Number
* 2) Bounding Surface Number and Sign of Vector Normal
*    to the Surface for Particles Leaving the Zone.
*    2) Is repeated to include all Bounding Surfaces.
*
* =====
* 10 Layers of Lead
jb       1  -1   2   12

```

```

jb      2   -2    3    12
jb      3   -3    4    12
jb      4   -4    5    12
jb      5   -5    6    12
jb      6   -6    7    12
jb      7   -7    8    12
jb      8   -8    9    12
jb      9   -9   10    12
jb     10  -10   11    12
* surface toward source
jb     11    1
* transmission surface
jb     12  -11
* outside surrounding cylinder
jb     13  -12
* =====
*
* Weights
*
* =====
* weight  1.0      1  2
* weight  0.5      3  4
* weight  0.25     5  6
* weight  0.125    7  8
* weight  0.0625   9 10
* =====
*
* Material Definitions
* 1) Material Number
* 2) Density (grams/cc)
* 3) Atom %
* 4) Isotope I.D. (ZZZAAA), e.g., 92238 for U-238
* 3) and 4) can be repeated in pairs to define
* composite materials
*
* =====
* Lead
matl  1  11.36  1.0 82000
* =====
*
* Assignment of Materials to Zones
* 1) Material Number
* 2) Zone Containing Material - Can be Repeated
*
* =====
* Vacuum - ILLEGAL EXCEPT IN OUTER, NON-REentrant ZONES
matz  0    11 thru 13
* Water
matz  1     1 thru 10
* =====
*
* Neutron source, track neutrons and induced photons
*
* =====
* 1) Transport (neutrons and induced photons) (0)
sentl  1     0
* point source just inside first layer of Lead
sourcel  1  0.0  0.0  0.01
* 4) Neutron induced fission spectrum
sentl    4    0.0

```

```

* monodirectional straight up z axis
sentl      6  0.0
sentl      7  1.0
* 8) Neutron Minimum Energy (2.53e-8 MeV)
sentl      8  1.307E-09
* 20) Multi-Band (0)
sentl     20   1
* 39) Thermal Scattering Sentinels (0)
sentl     39   1
* Thermal Scattering Temperature in All Zones (2.53e-8 MeV)
emin     2.53000e-08  1 thru   10
* Score and output fluence per zone
ltype     2   1 thru 10
* =====
*
* Definition of Running Conditions and Output Edit Options
*
* the following are the same in all 3 cases
*
* =====
* 2) Number of Samples (20)
sentl      2   10
* 3) Histories per Sample (5000)
sentl      3  1000
end

```

The results for the first calculation are shown below for photon and neutron numbers and energy deposition. From the end of these results we can see that this calculation took 132 seconds. When we look at the results for zone 12 (the transmission zone), for photons none of them got through the lead, and for neutrons 0.0265 per initial neutron got through (+/- 5.1 %), depositing 6.95E-04 MeV (+/- 22.5 %).

Actual photon energy deposition  
in MeV/zone by time step.

zone	total	Std dev	pct	1.000E+08
1	7.467E-01	1.28E-02	1.7	7.47E-01
2	3.147E-01	1.68E-02	5.3	3.15E-01
3	1.631E-01	1.12E-02	6.8	1.63E-01
4	1.282E-01	7.76E-03	6.1	1.28E-01
5	1.072E-01	1.07E-02	10.0	1.07E-01
6	6.876E-02	8.81E-03	12.8	6.88E-02
7	7.111E-02	7.52E-03	10.6	7.11E-02
8	3.257E-02	3.77E-03	11.6	3.26E-02
9	2.265E-02	4.30E-03	19.0	2.26E-02
10	8.284E-03	3.03E-03	36.6	8.28E-03
11	3.497E-02	3.05E-03	8.7	3.50E-02
12	.000E+00	.00E+00	.0	.00E+00
13	.000E+00	.00E+00	.0	.00E+00

Expected value photon energy depositions.  
in MeV/zone by time step.

zone	total	Std dev	pct	1.000E+08
1	7.461E-01	1.71E-02	2.3	7.46E-01
2	3.186E-01	1.41E-02	4.4	3.19E-01

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3	1.615E-01	1.51E-02	9.3	1.62E-01
4	1.230E-01	1.15E-02	9.4	1.23E-01
5	1.087E-01	1.14E-02	10.5	1.09E-01
6	7.314E-02	1.26E-02	17.3	7.31E-02
7	7.932E-02	9.84E-03	12.4	7.93E-02
8	4.302E-02	8.87E-03	20.6	4.30E-02
9	2.286E-02	5.58E-03	24.4	2.29E-02
10	9.914E-03	3.08E-03	31.1	9.91E-03
11	3.497E-02	3.05E-03	8.7	3.50E-02
12	.000E+00	.00E+00	.0	.00E+00
13	.000E+00	.00E+00	.0	.00E+00

Photon tally type totals, by zone and time step.

zone	tt	total	dev.	pct.	1.00E+08
1	3	1.120E-02	1.3E-03	11.6	1.12E-02
2	3	1.560E-02	1.5E-03	9.8	1.56E-02
3	3	5.100E-03	8.9E-04	17.4	5.10E-03
4	3	3.000E-03	5.4E-04	17.9	3.00E-03
5	3	2.000E-03	5.6E-04	27.9	2.00E-03
6	3	1.300E-03	3.0E-04	23.1	1.30E-03
7	3	8.000E-04	2.5E-04	31.2	8.00E-04
8	3	4.000E-04	2.2E-04	55.3	4.00E-04
9	3	9.000E-04	2.8E-04	30.8	9.00E-04
10	3	2.000E-04	1.3E-04	66.7	2.00E-04
11	3	2.700E-02	1.6E-03	5.8	2.70E-02
12	3	.000E+00	.0E+00	.0	.00E+00
13	3	.000E+00	.0E+00	.0	.00E+00

Local expected value energy depositions from neutrons only, in MeV/zone, by time step.

zone	total	Std dev	pct	1.000E+08
1	6.7197E-02	5.55E-04	.8	6.720E-02
2	4.1993E-02	3.62E-04	.9	4.199E-02
3	2.4315E-02	4.81E-04	2.0	2.431E-02
4	1.3641E-02	2.44E-04	1.8	1.364E-02
5	7.5131E-03	2.78E-04	3.7	7.513E-03
6	3.7819E-03	1.86E-04	4.9	3.782E-03
7	1.9659E-03	1.01E-04	5.1	1.966E-03
8	1.0291E-03	7.03E-05	6.8	1.029E-03
9	4.7181E-04	4.37E-05	9.3	4.718E-04
10	1.6199E-04	2.14E-05	13.2	1.620E-04
11	7.3660E-01	7.75E-03	1.1	7.366E-01
12	6.9583E-04	1.57E-04	22.5	6.958E-04
13	.0000E+00	.00E+00	.0	.000E+00

Neutron tally type totals by zone and time step.

zone	tt	total	dev.	pct.	1.00E+08
1	2	7.089E+01	9.68E-01	1.4	7.09E+01
2	2	8.071E+01	1.60E+00	2.0	8.07E+01
3	2	7.376E+01	1.73E+00	2.3	7.38E+01
4	2	6.291E+01	1.31E+00	2.1	6.29E+01
5	2	4.965E+01	1.00E+00	2.0	4.97E+01
6	2	3.717E+01	1.25E+00	3.4	3.72E+01
7	2	2.835E+01	1.35E+00	4.8	2.83E+01
8	2	1.914E+01	7.88E-01	4.1	1.91E+01
9	2	1.124E+01	5.62E-01	5.0	1.12E+01

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10	2	4.224E+00	2.79E-01	6.6	4.22E+00
11	3	8.889E-01	2.51E-03	.3	8.89E-01
12	3	2.650E-02	1.34E-03	5.1	2.65E-02
13	3	.000E+00	.00E+00	.0	.00E+00

Photon source / zone in MeV.

zone	total	Std dev	pct	1.000E+08
1	7.8066E-01	1.46E-02	1.9	7.807E-01
2	3.1992E-01	1.63E-02	5.1	3.199E-01
3	1.6152E-01	1.20E-02	7.4	1.615E-01
4	1.2552E-01	6.09E-03	4.9	1.255E-01
5	1.0790E-01	1.15E-02	10.6	1.079E-01
6	6.8550E-02	9.10E-03	13.3	6.855E-02
7	6.9644E-02	6.32E-03	9.1	6.964E-02
8	3.6584E-02	3.77E-03	10.3	3.658E-02
9	1.8357E-02	3.84E-03	20.9	1.836E-02
10	9.6293E-03	2.70E-03	28.0	9.629E-03

Local neutron energy deposition plus photon deposition (total energy deposited)

zone	total	Std dev	pct	1.000E+08
1	8.1325E-01	1.74E-02	2.1	8.132E-01
2	3.6063E-01	1.42E-02	3.9	3.606E-01
3	1.8583E-01	1.53E-02	8.2	1.858E-01
4	1.3665E-01	1.15E-02	8.4	1.367E-01
5	1.1624E-01	1.14E-02	9.8	1.162E-01
6	7.6923E-02	1.26E-02	16.4	7.692E-02
7	8.1282E-02	9.82E-03	12.1	8.128E-02
8	4.4050E-02	8.88E-03	20.2	4.405E-02
9	2.3332E-02	5.58E-03	23.9	2.333E-02
10	1.0076E-02	3.09E-03	30.6	1.008E-02
11	7.7157E-01	9.27E-03	1.2	7.716E-01
12	6.9583E-04	1.57E-04	22.5	6.958E-04
13	.0000E+00	.00E+00	.0	.000E+00

Multi-band n-gammas by zone and time step.

zone	total	1.000E+08
1	6.95849E-03	6.958E-03
2	1.21535E-02	1.215E-02
3	1.36747E-02	1.367E-02
4	1.33715E-02	1.337E-02
5	1.22217E-02	1.222E-02
6	9.70276E-03	9.703E-03
7	8.04383E-03	8.044E-03
8	5.75589E-03	5.756E-03
9	3.31265E-03	3.313E-03
10	1.69837E-03	1.698E-03

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-  
132.00 seconds - Finished run  
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## Chapter 6: Example Problems

The results for the second calculation are shown below for photon and neutron numbers and energy deposition. From the end of these results we can see that this calculation took 1178 seconds. When we look at the results for zone 12 (the transmission zone), for photons only 3.0E-05 of them got through the lead and deposited 2.21E-04 MeV (+/- 50.9 %), and for neutrons 0.02898 per initial neutron got through (+/- 1.7 %), depositing 6.1E-04 MeV (+/- 7.8 %). Since in case 1) no photons were transmitted we have nothing to compare to. For the neutrons the results are about what we would expect: using 10 times as many histories the standard deviation has been reduced by a factor of about 3, in line with  $1/\sqrt{\text{histories}}$  variation.

Actual photon energy deposition  
in MeV/zone by time step.

zone	total	Std dev	pct	1.000E+08
1	7.331E-01	3.51E-03	.5	7.33E-01
2	3.021E-01	3.65E-03	1.2	3.02E-01
3	1.522E-01	3.59E-03	2.4	1.52E-01
4	1.176E-01	3.83E-03	3.3	1.18E-01
5	9.572E-02	3.85E-03	4.0	9.57E-02
6	7.953E-02	3.37E-03	4.2	7.95E-02
7	6.526E-02	3.33E-03	5.1	6.53E-02
8	4.495E-02	2.48E-03	5.5	4.50E-02
9	2.813E-02	1.35E-03	4.8	2.81E-02
10	1.021E-02	6.63E-04	6.5	1.02E-02
11	3.411E-02	6.76E-04	2.0	3.41E-02
12	2.210E-04	1.13E-04	50.9	2.21E-04
13	.000E+00	.00E+00	.0	.00E+00

Expected value photon energy depositions.  
in MeV/zone by time step.

zone	total	Std dev	pct	1.000E+08
1	7.317E-01	5.20E-03	.7	7.32E-01
2	3.028E-01	3.08E-03	1.0	3.03E-01
3	1.540E-01	3.97E-03	2.6	1.54E-01
4	1.215E-01	4.00E-03	3.3	1.21E-01
5	9.736E-02	3.12E-03	3.2	9.74E-02
6	8.174E-02	5.28E-03	6.5	8.17E-02
7	6.716E-02	4.36E-03	6.5	6.72E-02
8	4.736E-02	3.96E-03	8.4	4.74E-02
9	2.711E-02	2.02E-03	7.4	2.71E-02
10	9.403E-03	8.47E-04	9.0	9.40E-03
11	3.411E-02	6.76E-04	2.0	3.41E-02
12	2.210E-04	1.13E-04	50.9	2.21E-04
13	.000E+00	.00E+00	.0	.00E+00

Photon tally type totals, by zone and time step.

zone	tt	total	dev.	pct.	1.00E+08
1	3	1.146E-02	2.5E-04	2.2	1.15E-02
2	3	1.573E-02	3.8E-04	2.4	1.57E-02
3	3	5.860E-03	2.2E-04	3.8	5.86E-03
4	3	2.570E-03	1.4E-04	5.4	2.57E-03

## Chapter 6: Example Problems

5	3	1.870E-03	1.8E-04	9.5	1.87E-03
6	3	1.540E-03	1.4E-04	9.3	1.54E-03
7	3	1.270E-03	8.0E-05	6.3	1.27E-03
8	3	8.900E-04	8.0E-05	8.9	8.90E-04
9	3	6.000E-04	9.5E-05	15.9	6.00E-04
10	3	2.000E-04	4.2E-05	21.1	2.00E-04
11	3	2.633E-02	5.6E-04	2.1	2.63E-02
12	3	3.000E-05	1.5E-05	50.9	3.00E-05
13	3	.000E+00	.0E+00	.0	.00E+00

Local expected value energy depositions from neutrons only, in MeV/zone, by time step.

zone	total	Std dev	pct	1.000E+08
1	6.7273E-02	2.31E-04	.3	6.727E-02
2	4.3074E-02	3.23E-04	.8	4.307E-02
3	2.4671E-02	2.00E-04	.8	2.467E-02
4	1.4040E-02	1.39E-04	1.0	1.404E-02
5	7.9126E-03	6.62E-05	.8	7.913E-03
6	4.2559E-03	6.68E-05	1.6	4.256E-03
7	2.2206E-03	5.82E-05	2.6	2.221E-03
8	1.1388E-03	2.74E-05	2.4	1.139E-03
9	5.2891E-04	1.72E-05	3.3	5.289E-04
10	1.6942E-04	5.87E-06	3.5	1.694E-04
11	7.3895E-01	2.56E-03	.3	7.390E-01
12	6.1077E-04	4.76E-05	7.8	6.108E-04
13	.0000E+00	.00E+00	.0	.000E+00

Neutron tally type totals by zone and time step.

zone	tt	total	dev.	pct.	1.00E+08
1	2	7.097E+01	2.76E-01	.4	7.10E+01
2	2	8.140E+01	5.01E-01	.6	8.14E+01
3	2	7.432E+01	5.39E-01	.7	7.43E+01
4	2	6.380E+01	5.08E-01	.8	6.38E+01
5	2	5.203E+01	5.48E-01	1.1	5.20E+01
6	2	4.058E+01	5.79E-01	1.4	4.06E+01
7	2	2.973E+01	3.95E-01	1.3	2.97E+01
8	2	2.057E+01	2.70E-01	1.3	2.06E+01
9	2	1.215E+01	1.98E-01	1.6	1.22E+01
10	2	4.473E+00	8.16E-02	1.8	4.47E+00
11	3	8.852E-01	1.36E-03	.2	8.85E-01
12	3	2.898E-02	4.84E-04	1.7	2.90E-02
13	3	.000E+00	.00E+00	.0	.00E+00

Photon source / zone in MeV.

zone	total	Std dev	pct	1.000E+08
1	7.6883E-01	3.96E-03	.5	7.688E-01
2	3.0165E-01	3.46E-03	1.1	3.017E-01
3	1.5121E-01	3.64E-03	2.4	1.512E-01
4	1.1669E-01	4.20E-03	3.6	1.167E-01
5	9.6883E-02	3.48E-03	3.6	9.688E-02
6	7.9352E-02	3.82E-03	4.8	7.935E-02
7	6.4634E-02	3.38E-03	5.2	6.463E-02
8	4.6170E-02	2.84E-03	6.2	4.617E-02
9	2.7128E-02	1.25E-03	4.6	2.713E-02
10	1.0642E-02	8.99E-04	8.4	1.064E-02

## Chapter 6: Example Problems

Local neutron energy deposition plus photon deposition (total energy deposited)

zone	total	Std dev	pct	1.000E+08
1	7.9894E-01	5.28E-03	.7	7.989E-01
2	3.4592E-01	3.32E-03	1.0	3.459E-01
3	1.7866E-01	4.03E-03	2.3	1.787E-01
4	1.3553E-01	3.97E-03	2.9	1.355E-01
5	1.0528E-01	3.13E-03	3.0	1.053E-01
6	8.5999E-02	5.32E-03	6.2	8.600E-02
7	6.9385E-02	4.40E-03	6.3	6.939E-02
8	4.8498E-02	3.97E-03	8.2	4.850E-02
9	2.7636E-02	2.01E-03	7.3	2.764E-02
10	9.5721E-03	8.48E-04	8.9	9.572E-03
11	7.7306E-01	2.93E-03	.4	7.731E-01
12	8.3182E-04	1.31E-04	15.8	8.318E-04
13	.0000E+00	.00E+00	.0	.000E+00

Multi-band n-gammas by zone and time step.

zone	total	1.000E+08
1	7.24026E-03	7.240E-03
2	1.16603E-02	1.166E-02
3	1.33167E-02	1.332E-02
4	1.32834E-02	1.328E-02
5	1.20188E-02	1.202E-02
6	1.05764E-02	1.058E-02
7	8.44655E-03	8.447E-03
8	6.29990E-03	6.300E-03
9	3.93076E-03	3.931E-03
10	1.49574E-03	1.496E-03

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-  
1178.00 seconds - Finished run  
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The results for the third calculation are shown below for photon and neutron numbers and energy deposition. From the end of these results we can see that this calculation took 392 seconds. When we look at the results for zone 12 (the transmission zone), for photons only 9.50E-06 of them got through the lead and deposited 7.8E-05 MeV (+/- 62.1 %), and for neutrons 0.02812 per initial neutron got through (+/- 3.1 %), depositing 6.16E-04 MeV (+/- 6.2 %). Since in case 1) no photon were transmitted we have nothing to compare to.

Actual photon energy deposition  
in MeV/zone by time step.

zone	total	Std dev	pct	1.000E+08
1	7.273E-01	1.10E-02	1.5	7.27E-01
2	2.995E-01	1.17E-02	3.9	2.99E-01
3	1.541E-01	9.44E-03	6.1	1.54E-01
4	1.196E-01	7.06E-03	5.9	1.20E-01
5	9.643E-02	4.66E-03	4.8	9.64E-02

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6	7.771E-02	5.61E-03	7.2	7.77E-02
7	6.585E-02	4.13E-03	6.3	6.58E-02
8	4.713E-02	2.90E-03	6.2	4.71E-02
9	2.963E-02	2.00E-03	6.8	2.96E-02
10	9.718E-03	5.28E-04	5.4	9.72E-03
11	3.209E-02	2.01E-03	6.3	3.21E-02
12	7.829E-05	4.86E-05	62.1	7.83E-05
13	.000E+00	.00E+00	.0	.00E+00

Expected value photon energy depositions.  
in MeV/zone by time step.

zone	total	Std dev	pct	1.000E+08
1	7.362E-01	1.88E-02	2.6	7.36E-01
2	3.065E-01	1.15E-02	3.7	3.06E-01
3	1.586E-01	1.03E-02	6.5	1.59E-01
4	1.093E-01	8.76E-03	8.0	1.09E-01
5	9.709E-02	5.05E-03	5.2	9.71E-02
6	7.995E-02	7.40E-03	9.3	8.00E-02
7	6.844E-02	5.54E-03	8.1	6.84E-02
8	4.882E-02	3.49E-03	7.2	4.88E-02
9	2.947E-02	2.51E-03	8.5	2.95E-02
10	8.593E-03	4.63E-04	5.4	8.59E-03
11	3.209E-02	2.01E-03	6.3	3.21E-02
12	7.829E-05	4.86E-05	62.1	7.83E-05
13	.000E+00	.00E+00	.0	.00E+00

Photon tally type totals, by zone and time step.

zone	tt	total	dev.	pct.	1.00E+08
1	3	9.500E-03	9.2E-04	9.7	9.50E-03
2	3	1.580E-02	1.1E-03	6.8	1.58E-02
3	3	5.600E-03	6.8E-04	12.2	5.60E-03
4	3	2.750E-03	4.2E-04	15.4	2.75E-03
5	3	2.375E-03	4.5E-04	19.0	2.38E-03
6	3	1.850E-03	1.5E-04	8.1	1.85E-03
7	3	1.088E-03	1.4E-04	13.2	1.09E-03
8	3	8.625E-04	8.0E-05	9.3	8.63E-04
9	3	4.875E-04	8.7E-05	17.8	4.88E-04
10	3	1.375E-04	3.3E-05	24.2	1.38E-04
11	3	2.530E-02	2.0E-03	8.0	2.53E-02
12	3	1.875E-05	9.5E-06	50.9	1.88E-05
13	3	.000E+00	.0E+00	.0	.00E+00

Local expected value energy depositions from neutrons only, in MeV/zone, by time step.

zone	total	Std dev	pct	1.000E+08
1	6.6311E-02	6.92E-04	1.0	6.631E-02
2	4.1892E-02	9.97E-04	2.4	4.189E-02
3	2.4194E-02	6.57E-04	2.7	2.419E-02
4	1.3980E-02	5.49E-04	3.9	1.398E-02
5	7.8593E-03	3.67E-04	4.7	7.859E-03
6	4.3141E-03	2.13E-04	4.9	4.314E-03
7	2.2185E-03	1.30E-04	5.9	2.219E-03
8	1.1396E-03	5.93E-05	5.2	1.140E-03
9	5.3098E-04	2.88E-05	5.4	5.310E-04

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10	1.6977E-04	1.12E-05	6.6	1.698E-04
11	7.4443E-01	6.87E-03	.9	7.444E-01
12	6.1613E-04	3.82E-05	6.2	6.161E-04
13	.0000E+00	.00E+00	.0	.000E+00

Neutron tally type totals by zone and time step.

zone	tt	total	dev.	pct.	1.00E+08
1	2	6.908E+01	1.11E+00	1.6	6.91E+01
2	2	7.984E+01	2.53E+00	3.2	7.98E+01
3	2	7.398E+01	2.13E+00	2.9	7.40E+01
4	2	6.021E+01	1.64E+00	2.7	6.02E+01
5	2	4.855E+01	1.55E+00	3.2	4.85E+01
6	2	3.876E+01	1.39E+00	3.6	3.88E+01
7	2	2.878E+01	1.02E+00	3.5	2.88E+01
8	2	1.970E+01	6.40E-01	3.2	1.97E+01
9	2	1.155E+01	3.90E-01	3.4	1.16E+01
10	2	4.230E+00	1.45E-01	3.4	4.23E+00
11	3	8.754E-01	5.57E-03	.6	8.75E-01
12	3	2.812E-02	8.65E-04	3.1	2.81E-02
13	3	.000E+00	.00E+00	.0	.00E+00

Photon source / zone in MeV.

zone	total	Std dev	pct	1.000E+08
1	7.6349E-01	1.00E-02	1.3	7.635E-01
2	2.9689E-01	1.08E-02	3.6	2.969E-01
3	1.5252E-01	9.32E-03	6.1	1.525E-01
4	1.1887E-01	6.96E-03	5.9	1.189E-01
5	9.7944E-02	4.72E-03	4.8	9.794E-02
6	7.6200E-02	5.20E-03	6.8	7.620E-02
7	6.5838E-02	4.68E-03	7.1	6.584E-02
8	4.6690E-02	2.59E-03	5.5	4.669E-02
9	2.9919E-02	1.94E-03	6.5	2.992E-02
10	9.6766E-03	4.83E-04	5.0	9.677E-03

Local neutron energy deposition plus photon deposition (total energy deposited)

zone	total	Std dev	pct	1.000E+08
1	8.0247E-01	1.88E-02	2.3	8.025E-01
2	3.4838E-01	1.20E-02	3.5	3.484E-01
3	1.8274E-01	1.05E-02	5.7	1.827E-01
4	1.2331E-01	8.85E-03	7.2	1.233E-01
5	1.0495E-01	5.18E-03	4.9	1.050E-01
6	8.4266E-02	7.43E-03	8.8	8.427E-02
7	7.0661E-02	5.48E-03	7.8	7.066E-02
8	4.9964E-02	3.49E-03	7.0	4.996E-02
9	3.0003E-02	2.50E-03	8.3	3.000E-02
10	8.7623E-03	4.66E-04	5.3	8.762E-03
11	7.7652E-01	7.86E-03	1.0	7.765E-01
12	6.9442E-04	4.54E-05	6.5	6.944E-04
13	.0000E+00	.00E+00	.0	.000E+00

Multi-band n-gammas by zone and time step.

zone	total	1.000E+08
1	7.05654E-03	7.057E-03

2	1.15665E-02	1.157E-02
3	1.37822E-02	1.378E-02
4	1.32077E-02	1.321E-02
5	1.20984E-02	1.210E-02
6	1.06424E-02	1.064E-02
7	8.41231E-03	8.412E-03
8	6.15564E-03	6.156E-03
9	3.83307E-03	3.833E-03
10	1.43349E-03	1.433E-03

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 392.00 seconds - Finished run  
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When we finally compare the results the conclusions are,

1) For neutrons the case 2) and 3) results are in excellent agreement. The major difference between these two cases is that case 2) took 3 times as long to run as case 3) (see the below summary table). The primary objective of this example was to demonstrate the use of weights and the advantage that they have in applications.

Cases 2) and 3) illustrate two different approaches to trying to improve the statistical accuracy of results: case 2) tries the brute force approach to simply use more histories; this approach converges very slowly, approximately as  $1/\sqrt{\text{number of histories}}$ . For example, to reduce the statistical uncertainty in the results by a factor of 10 would require approximately 100 times as many histories, case 3) uses weight and in this case produces answers which are reliable as those of case 2) in 1/3 the time. If we wanted a completely fair comparison we could have run 3 times as many histories in case 3) to have the same running time as case 2), and produce results to much better statistics than case 2).

One last point concerning the neutron results: the results clearly indicate that this is a completely hypothetical example. In this example about 2.8 % of the incident neutrons penetrate 150 cm (5 feet) of lead. Based on these results, nobody would build a purely lead shield to shield a neutron source. The photon results indicate lead is good to use if you want to shield against photons, but it isn't very good against neutrons. An excellent exercise to try is to replace the lead layers nearest the source by a good neutron attenuator and let the remaining lead layers deal with the photons. With this approach you should be able to reduce both neutron and photon transmission to a very low level. The above deck can be simply modified to perform this calculation by: 1) define your neutron attenuating material using **matl** input to define a second material, 2) use **matz** input to assign this new material to the first few zones nearest the source, e.g., zones 1, 2, 3, etc.. Be sure that you modify the existing **matz** input to insure that you do not assign lead to the same zones as your new material. For example, if you assign your new material to zones 1 through 3, the existing **matz** input should be modified to assign material 1 (the lead) to zones 4 through 10.

Case	1)	2)	3)
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Batches	10	10	10
Histories/Batch	1000	10000	1000
Weights	No	No	Yes
Running time (Seconds)	132	1178	392
Neutrons Transmitted	.0265	.02898	.02812
Average Energy	6.94E-4	6.10E-4	6.16E-4

---

2) For photons we still haven't run enough histories to get an accurate estimate of the transmission; all we can say that it is quite small in the neighborhood of  $1.0\text{E-}5$  photons per source neutron. Since obtaining accurate results for the photons was not the primary objective of this example we will not run additional cases to improve the accuracy of the photon results; we will leave this as an exercise for the reader.

If you are interested in learning more about the use of weights you can try to improve the accuracy of the photon transmission results by using **wgtgam** and **gpwgt** input. By default if **wgtgam** input is not used the statistical weight of the photons is the same as the neutrons and the same splitting and Russian roulette rules apply to both. Therefore in case 3) by default we used the same weights for both. If you want to improve the photon results without investing a great deal more time in calculations, the above input deck need only to slightly modified to include **wgtgam** input. Note, the neutron weights in the above deck are the same in pairs of zones, so that you have room to change the photon weights faster than the neutron weights. For example, in the most extreme case the **wgtgam** input for zones 1, 2, 3, 4, etc., could be 512, 256, 128, 64, etc., out to zone 10 with a weight of 1.

Another consideration besides using **wgtgam** to encourage photon flow toward the transmission surface, is to generate more photons using **gpwgt**; in this problem there just aren't too many photons being generated, which is leading to the poor statistical results.

If you want to use a different approach to improve the photon results, we will merely mention that most photons that penetrate 150 cm of lead will be produced fairly deep in the lead, so that the distance that they must transport to be transmitted through the lead is small. To account for this effect you can divide the last few layers of lead into finer zoning and change weights **wgtgam** even faster in these layers and use **gpwgt** to generate more photons in these zones.

**Bottom line** as far as using weights, is that they can be used to great advantage in many problems and is something that users should become familiar with. Remember that the neutron and photon statistical weights, **weight** and **wgtgam** are relative from one zone to the adjacent zones, whereas the neutron induced photon production multiplier **gpwgt** is

absolute. Used in combination weights and multipliers can be used to minimize the time required to obtain answers to within a given statistical uncertainty.

### Firstwall

The next example is a geometrically complicated mock up of an inertial confinement facility. TART95 cannot calculate what happens in a plasma, but it can calculate the effects of the spectrum leaking from the plasma. This example is designed to allow prediction of radiation damage to the first wall of the facility. This example is included here to demonstrate the use of reflecting zones and to illustrate that a problem that is geometrically very complicated can be reduced to much simpler problem by using reflection.

Figs. 13 through 15 show the **TARTCHEK** output for this problem. From fig. 13 not too much can be seen, since most of the detail is in a small region near the origin. This figure is included to identify the two reflecting zones, zone 103 and 104. The problem is rotationally symmetric about the z axis, except for a series of 144 tubes that are equally spaced around the z axis. In this case the reflecting zones, 103 and 104, have been used to allow a one quarter mock up only involving modeling 36 of the 144 tubes. Fig. 14 shows the position of the 36 tubes, with cylindrical axis parallel to the z axis, spaced around the z axis, all an equal distance from the z axis. By zooming in even further in fig. 15 we can see the details of each tube; each tube consists of two zones, an inner cylindrical zone and an outer cylindrical annulus surrounding each tube. From the below input we can identify these tubes as zones 20 through 91; a total of 72 zones, two per tube.

When using reflectors remember that as far as the code is concerned only a fraction of the geometry is actually present; in this case only one quarter of the actual geometry. In this case since the particles are constrained to stay within one quarter of the geometry and still have the same number of events per history, the resulting tallies will be four times larger per unit volume or mass than they would have been if the entire geometry were described. For example, statistically the events that really happen within the 144 tubes are constrained by reflectors to now happen within 36 tubes. So that in this problem involving 36 tubes, per tube the pathlength, deposition, etc. will be four times larger than in the real, complete, system. By knowing the ratio of the real part of the system to that described using reflectors, it is usually very simple to correctly define results per unit volume or mass. For example, in this case we are mocking up exactly one quarter of the system, which therefore contains exactly one quarter of the volume and mass of the entire real system. Therefore the real results per unit volume or mass will be exactly one quarter of the results obtained for the system using reflectors. Do not overlook this effect when interpreting your output results.

Since this problem is designed to predict radiation damage it is important to determine reaction rates for a number of reactions in many zones. Note, the use of **reactall** input to accomplish this. For specific zones we could add input to write a file tracing each history in specific zones. This output file can later be read and folded together with activation

cross sections to define more details of reaction rates. Since we are interested in reaction rates, note, the detailed definition of materials must include all, even minor, impurities, to insure that nothing is overlooked.

Note also the use of **eta** input. In this case a number of zones are initially defined to contain air at STP density. The **eta** input is then used to reduce the overall density by a factor of  $10^{-8}$ , so that these zones essentially contain only vacuum. Remember that in order to transport through any zone it must contain some material. So that in this case there is an important difference between assigning air at a very low density to simulate vacuum, and leaving the zones empty.

```

name Firstwall
box r75 Firstwall
*
*
cyl 1 20.0
cyl 2 34.0
cyl 3 42.0
cyl 4 49.0
cyl 5 56.0
cyl 6 63.0
cyl 7 70.0
*
sphere 130 20.0 400.0 0.0 0.0
sphere 131 34.0 400.0 0.0 0.0
sphere 132 42.0 400.0 0.0 0.0
sphere 133 49.0 400.0 0.0 0.0
sphere 134 56.0 400.0 0.0 0.0
sphere 135 63.0 400.0 0.0 0.0
sphere 136 70.0 400.0 0.0 0.0
*
cyl 8 300.0
cyl 9 300.32
cyl 10 350.0
cyl 11 352.5
cyl 12 355.0
cyl 13 357.5
cyl 14 367.66
cyl 15 367.82
*
zplane 16 -150.0
zplane 17 400.0
*
xplane 18 0.0
yplane 19 0.0
*
*
cone 100 14.04 1050.00
cone 101 14.04 1053.09
cone 102 14.04 1259.19
cone 103 14.04 1269.50
cone 104 14.04 1279.81
cone 105 14.04 1290.12
cone 106 14.04 1332.00
cone 107 14.04 1332.66

```

## Chapter 6: Example Problems

```
*
zplane 108 -600.0
*
sphere 110 350.00 400.0 0.0 0.0
sphere 111 352.50 400.0 0.0 0.0
sphere 112 355.00 400.0 0.0 0.0
sphere 113 357.50 400.0 0.0 0.0
sphere 114 367.66 400.0 0.0 0.0
sphere 115 367.82 400.0 0.0 0.0
*
*
cyl 20 4.76 299.93 6.54
cyl 21 4.76 299.36 19.62
cyl 22 4.76 298.22 32.66
cyl 23 4.76 296.51 45.64
cyl 24 4.76 294.24 58.53
cyl 25 4.76 291.40 71.31
cyl 26 4.76 288.01 83.95
cyl 27 4.76 284.08 96.43
cyl 28 4.76 279.60 108.73
cyl 29 4.76 274.59 120.82
cyl 30 4.76 269.06 132.69
cyl 31 4.76 263.02 144.30
cyl 32 4.76 256.47 155.63
cyl 33 4.76 249.44 166.67
cyl 34 4.76 241.93 177.39
cyl 35 4.76 233.97 187.78
cyl 36 4.76 225.55 197.80
cyl 37 4.76 216.71 207.45
cyl 38 4.76 207.45 216.71
cyl 39 4.76 197.80 225.55
cyl 40 4.76 187.78 233.97
cyl 41 4.76 177.39 241.93
cyl 42 4.76 166.67 249.44
cyl 43 4.76 155.63 256.47
cyl 44 4.76 144.30 263.02
cyl 45 4.76 132.69 269.06
cyl 46 4.76 120.82 274.59
cyl 47 4.76 108.73 279.60
cyl 48 4.76 96.43 284.08
cyl 49 4.76 83.95 288.01
cyl 50 4.76 71.31 291.40
cyl 51 4.76 58.53 294.24
cyl 52 4.76 45.64 296.51
cyl 53 4.76 32.66 298.22
cyl 54 4.76 19.62 299.36
cyl 55 4.76 6.54 299.93
*
cyl 60 5.08 299.93 6.54
cyl 61 5.08 299.36 19.62
cyl 62 5.08 298.22 32.66
cyl 63 5.08 296.51 45.64
cyl 64 5.08 294.24 58.53
cyl 65 5.08 291.40 71.31
cyl 66 5.08 288.01 83.95
cyl 67 5.08 284.08 96.43
cyl 68 5.08 279.60 108.73
cyl 69 5.08 274.59 120.82
cyl 70 5.08 269.06 132.69
cyl 71 5.08 263.02 144.30
```

## Chapter 6: Example Problems

```
cyl 72 5.08 256.47 155.63
cyl 73 5.08 249.44 166.67
cyl 74 5.08 241.93 177.39
cyl 75 5.08 233.97 187.78
cyl 76 5.08 225.55 197.80
cyl 77 5.08 216.71 207.45
cyl 78 5.08 207.45 216.71
cyl 79 5.08 197.80 225.55
cyl 80 5.08 187.78 233.97
cyl 81 5.08 177.39 241.93
cyl 82 5.08 166.67 249.44
cyl 83 5.08 155.63 256.47
cyl 84 5.08 144.30 263.02
cyl 85 5.08 132.69 269.06
cyl 86 5.08 120.82 274.59
cyl 87 5.08 108.73 279.60
cyl 88 5.08 96.43 284.08
cyl 89 5.08 83.95 288.01
cyl 90 5.08 71.31 291.40
cyl 91 5.08 58.53 294.24
cyl 92 5.08 45.64 296.51
cyl 93 5.08 32.66 298.22
cyl 94 5.08 19.62 299.36
cyl 95 5.08 6.54 299.93
*
*
sphere 99 1000.0
*
*   inner pocket
jb 1 1 -16 17 -18 -19
*   six layers of flibe (total of 50 cm)
jb 2 -1 2 -16 17 -18 -19
jb 3 -2 3 -16 17 -18 -19
jb 4 -3 4 -16 17 -18 -19
jb 5 -4 5 -16 17 -18 -19
jb 6 -5 6 -16 17 -18 -19
jb 7 -6 7 -16 17 -18 -19
*
*   void between inner flibe & 1st wall
jb 8 -7 8 -16 17 -18 -19 -60 -61 -62 -63 -64 -65 -66 -67 &
-68 -69 -70 -71 -72 -73 -74 -75 -76 -77 -78 -79 -80 -81 -82 &
-83 -84 -85 -86 -87 -88 -89 -90 -91 -92 -93 -94 -95
*
*   36 tubes: interior flibe, structure
jb 20 20 -16 17 -18 -19
jb 21 -20 60 -16 17 -18 -19
jb 22 21 -16 17 -18 -19
jb 23 -21 61 -16 17 -18 -19
jb 24 22 -16 17 -18 -19
jb 25 -22 62 -16 17 -18 -19
jb 26 23 -16 17 -18 -19
jb 27 -23 63 -16 17 -18 -19
jb 28 24 -16 17 -18 -19
jb 29 -24 64 -16 17 -18 -19
jb 30 25 -16 17 -18 -19
jb 31 -25 65 -16 17 -18 -19
jb 32 26 -16 17 -18 -19
jb 33 -26 66 -16 17 -18 -19
jb 34 27 -16 17 -18 -19
jb 35 -27 67 -16 17 -18 -19
```

## Chapter 6: Example Problems

jb 36 28 -16 17 -18 -19  
jb 37 -28 68 -16 17 -18 -19  
jb 38 29 -16 17 -18 -19  
jb 39 -29 69 -16 17 -18 -19  
jb 40 30 -16 17 -18 -19  
jb 41 -30 70 -16 17 -18 -19  
jb 42 31 -16 17 -18 -19  
jb 43 -31 71 -16 17 -18 -19  
jb 44 32 -16 17 -18 -19  
jb 45 -32 72 -16 17 -18 -19  
jb 46 33 -16 17 -18 -19  
jb 47 -33 73 -16 17 -18 -19  
jb 48 34 -16 17 -18 -19  
jb 49 -34 74 -16 17 -18 -19  
jb 50 35 -16 17 -18 -19  
jb 51 -35 75 -16 17 -18 -19  
jb 52 36 -16 17 -18 -19  
jb 53 -36 76 -16 17 -18 -19  
jb 54 37 -16 17 -18 -19  
jb 55 -37 77 -16 17 -18 -19  
jb 56 38 -16 17 -18 -19  
jb 57 -38 78 -16 17 -18 -19  
jb 58 39 -16 17 -18 -19  
jb 59 -39 79 -16 17 -18 -19  
jb 60 40 -16 17 -18 -19  
jb 61 -40 80 -16 17 -18 -19  
jb 62 41 -16 17 -18 -19  
jb 63 -41 81 -16 17 -18 -19  
jb 64 42 -16 17 -18 -19  
jb 65 -42 82 -16 17 -18 -19  
jb 66 43 -16 17 -18 -19  
jb 67 -43 83 -16 17 -18 -19  
jb 68 44 -16 17 -18 -19  
jb 69 -44 84 -16 17 -18 -19  
jb 70 45 -16 17 -18 -19  
jb 71 -45 85 -16 17 -18 -19  
jb 72 46 -16 17 -18 -19  
jb 73 -46 86 -16 17 -18 -19  
jb 74 47 -16 17 -18 -19  
jb 75 -47 87 -16 17 -18 -19  
jb 76 48 -16 17 -18 -19  
jb 77 -48 88 -16 17 -18 -19  
jb 78 49 -16 17 -18 -19  
jb 79 -49 89 -16 17 -18 -19  
jb 80 50 -16 17 -18 -19  
jb 81 -50 90 -16 17 -18 -19  
jb 82 51 -16 17 -18 -19  
jb 83 -51 91 -16 17 -18 -19  
jb 84 52 -16 17 -18 -19  
jb 85 -52 92 -16 17 -18 -19  
jb 86 53 -16 17 -18 -19  
jb 87 -53 93 -16 17 -18 -19  
jb 88 54 -16 17 -18 -19  
jb 89 -54 94 -16 17 -18 -19  
jb 90 55 -16 17 -18 -19  
jb 91 -55 95 -16 17 -18 -19  
\*

\* 4 support brackets

jb 100 -8 9 -16 17 -18 -19 -60 -61 -62 -63 -64 -65 -66 -67 &  
-68 -69 -70 -71 -72 -73 -74 -75 -76 -77 -78 -79 -80 -81 -82 &

## Chapter 6: Example Problems

```
-83 -84 -85 -86 -87 -88 -89 -90 -91 -92 -93 -94 -95
*
* 50 cm of flibe
jb 101 -9 10 -16 17 -18 -19 -60 -61 -62 -63 -64 -65 -66 -67 &
-68 -69 -70 -71 -72 -73 -74 -75 -76 -77 -78 -79 -80 -81 -82 &
-83 -84 -85 -86 -87 -88 -89 -90 -91 -92 -93 -94 -95
*
* cylindrical leakage
jb 102 -15 99 -16 17 -18 -19
*
* the next 4 "leakage" zone are actually reflection zones -> do NOT
* assign materials to these!!!!
*
* -x leakage
jb 103 18 99
* -y leakage that is also +x
jb 104 19 -18 99
*
* master leakage
jb 107 -99
*
* shell #2 (SS #304)
jb 108 -10 11 -16 17 -18 -19
* flibe between shell #2 and shell #3
jb 109 -11 12 -16 17 -18 -19
* shell #3 (SS #304)
jb 110 -12 13 -16 17 -18 -19
* insulation that allows purge gas between shell #3 and shell #4
jb 111 -13 14 -16 17 -18 -19
* shell #4 (thin SS #304)
jb 112 -14 15 -16 17 -18 -19
*
*
* void inside upper hemisphere
jb 120 -136 110 -17 -18 -19
* shell #2 hemisphere
jb 121 -110 111 -17 -18 -19
* flibe between shell #2 and shell #3 - upper hemisphere
jb 122 -111 112 -17 -18 -19
* shell #3 hemisphere
jb 123 -112 113 -17 -18 -19
* insulation upper hemisphere
jb 124 -113 114 -17 -18 -19
* shell #4 hemisphere
jb 125 -114 115 -17 -18 -19
* upper hemisphere leakage
jb 126 -115 99 -17 -18 -19
*
* void inside flibe hemisphere, 6 flibe zones
jb 150 130 -17 -18 -19
jb 151 -130 131 -17 -18 -19
jb 152 -131 132 -17 -18 -19
jb 153 -132 133 -17 -18 -19
jb 154 -133 134 -17 -18 -19
jb 155 -134 135 -17 -18 -19
jb 156 -135 136 -17 -18 -19
*
* void inside lower cones
jb 130 -7 100 16 -108 -18 -19
* 1st wall lower cone
```

```

jb 131 -100 101 16 -108 -18 -19
* 50 cm flibe lower cone
jb 132 -101 102 16 -108 -18 -19
* shell #2 lower cone
jb 133 -102 103 16 -108 -18 -19
* flibe between shell #2 and shell #3 - lower cone
jb 134 -103 104 16 -108 -18 -19
* shell #3 lower cone
jb 135 -104 105 16 -108 -18 -19
* insulation lower cone
jb 136 -105 106 16 -108 -18 -19
* shell #4 lower cone
jb 137 -106 107 16 -108 -18 -19
* lower cone leakage
jb 138 -107 99 16 -108 -18 -19
* leakage below lower cone
jb 139 108 99 -18 -19
*
*
* void, 6 layers of flibe
jb 140 1 16 -108 -18 -19
jb 141 -1 2 16 -108 -18 -19
jb 142 -2 3 16 -108 -18 -19
jb 143 -3 4 16 -108 -18 -19
jb 144 -4 5 16 -108 -18 -19
jb 145 -5 6 16 -108 -18 -19
jb 146 -6 7 16 -108 -18 -19
*
reflx 103
refly 104
*
*
* Material #1 is air @ STP
matl 1 1.29e-3 0.784 7014 0.211 8016 0.005 18000
*
* Material #2 is pure flibe
matl 2 2.00 0.02115 3006 0.26085 3007 0.145 4009 0.573 9019
*
* Material #3 is SS #304 (Se is in As)
matlwp 3 7.86 0.00000923 5010 0.00004084 5011 0.00197899 6012 &
.00002385 6013 .00060085 7014 .00030043 13027 .00400568 14000 &
.00040057 15031 .00016023 16032 .00005007 23051 .19026961 24000 &
.70099331 26000 .00016023 27059 .10014190 28000 .00030043 29000 &
.00003004 30000 .00025035 33075 .00005007 40000 .00000050 47107 &
.00000050 47109 .00000200 48000 .00005007 50000 .00001001 51000 &
.00005007 73181 .00010014 74000 .00001001 82000 .00001001 83209
*
matz 1 1 8 102 111 120 124 126 130 136 138 139 140 150
matz 2 2 thru 7
matz 2 141 thru 146
matz 2 151 thru 156
matz 2 20 22 24 26 28 30 32 34 36 38 40 42 44 46 48 50 52 54 56 58
matz 2 60 62 64 66 68 70 72 74 76 78 80 82 84 86 88 90
matz 2 101 109 122 132 134
matz 3 100
matz 3 21 23 25 27 29 31 33 35 37 39 41 43 45 47 49 51 53 55 57 59
matz 3 61 63 65 67 69 71 73 75 77 79 81 83 85 87 89 91
matz 3 108 110 112 121 123 125 131 133 135 137
*
ltype 2 20 thru 90

```

```

ltype 2 2 thru 7
ltype 2 101 thru 109
ltype 2 122 thru 123
ltype 2 151 thru 156
ltype 2 132 thru 134
ltype 2 141 thru 146
*
reactall 2 thru 7
reactall 20 thru 90
reactall 101 thru 109
reactall 122 thru 123
reactall 132 thru 134
reactall 141 thru 146
reactall 151 thru 156
*
source1 1 0.0001 0.0001 0.0
*
sent1 1 1 2 5 3 200000 4 14.1 5 2 20 1 23 1 39 1 46 1 49 3
*
eta 1.0e-8 1 8 120 130 140 150
*
end

```

As an exercise see if you can use reflecting zones to reduce this problem to be much simpler. For example, can you set up this problem to only include one of the 144 tubes? As presently set up we will obtain results independently for each of the 36 tubes, each with its own statistical uncertainty. These results can be combined to define improved results, since we know that the system is rotationally symmetric and therefore the results for each of the 36 tubes should be identical. This could be done, but it is inconvenient since we would have to do it by hand, based on the 36 independent output results in the TART95 output file. If you can reduce this to a problem only involving one of the 144 tubes, then the output file will contain the results that you want directly, with the minimum statistical uncertainty already calculated for you. Try this exercise and compare the results for the 36 tube mock up to your one tube mock up.

### TARTCHEK Problems

The above example problems discussed a variety of criticality and source problems and used **TARTCHEK** output figures to illustrate these problems. Below we will discuss a number of problems in order to illustrate some features of **TARTCHEK** that were not covered above.

### Neutron Detector

Figs. 16 through 20 illustrate a large spherical inertial confinement cavity and a complicated 3-D detector. The overall geometry is spherical with a radius of over 15 meters (1585 cm). Fig. 16 shows the geometry. Since the detector on the right hand side is 3-D, the initial display shows three 2-D slices through the geometry.

The first test we ran was **2-D Track** and fig. 17 illustrates the results. This test involves starting randomly directed trajectories at the center of the displayed geometry and tracking

in a straight-line through the geometry until: 1) the track reaches the edge of the display, 2) the track enters a terminus zone, i.e., an area that is not defined to be included in any zone, or a zone that does not contain any material. In this case we have decreased the number of particles tracked to allow us to see individual tracks radiating out from the center of the display.

For any properly defined TART95 geometry starting a source anywhere inside the geometry it should be able to reach every other part of the geometry. Therefore if the geometry is correct we expect the entire interior of the geometry to eventually fill with tracks reaching all the way to the border of the geometry. In this case from fig. 17 we can see that there is a black area to the right of the displays of the (X, Z) and (Y, Z) 2-D slices, right in the middle of the detector. This indicates that there is something wrong in this area and if we had run this problem our detector response (the answer we want) would have been incorrect.

Starting from the display shown in fig. 17 we used the **Z-X View** option to increase our visual resolution, and then the **Center** option to move the center of the display to the center of the detector. Then we used the **10 Zoom** to zoom into the area around the detector and re-ran the **2-D Track** option. Fig. 18 illustrates the results. Note the large black area to the right, indicating an area that particles cannot pass through. In this case the test results say that zone 62 does not contain any material and we have found the source of the problem. But we will continue with this example in order to illustrate some other ways that we could have found the source of the problem.

In fig. 19 we illustrate the effect of next using the **2-D Re-Entry** test. The results indicate that there is an area interior to the problem that is identified as a terminus, void, zone that TART95 would not be able to transport particles through. With the **2-D Re-Entry** test **TARTCHEK** has continued tracking the trajectories through these terminus zones and found that the trajectories re-enter the interior geometry on the far side of the terminus zone, indicating a definite error. By comparing the results shown in figs. 18 and 19 we can see that the problem area is a small region between where the tracks were terminated in fig. 18 and where they re-entered the geometry in fig. 19.

Finally if fig. 20 we use the **What Zone?** and **What Material?** options to identify a number of zones and the materials that they contain. Note, that zone number 62 contains material number 0, indicating a terminus, void zone; this is exactly the same error indicated in fig. 18, but this figure illustrates a different way to come to the same conclusion.

Once this error was found it was simple to decide to modify the TART95 input deck to define zone number 62 to contain material number 1. After this change to the input deck, re-running **TARTCHEK** demonstrated that the deck is now correct and we could then start TART95 calculations. Note, the importance of finding this error. The answer that we were looking for was the detector response and with this error it was impossible for

neutrons to reach much of the detector and our results would have been completely in error and misleading.

### Canned Fuel

The next example illustrates how to use **TARTCHEK** to look inside your geometry and display 3-D results. In this example we used a can of fuel, defined by a cylindrical stack of uranium disks inside a double layer of cladding; the inner layer is silicon and the outer layer is aluminum. Fig. 21 illustrates three 2-D slices through the geometry. In the Z-X View we can see the two outer layers of cladding outside the cylinders of fuel as well as at both ends. It also shows the stack of 10 disks of fuel in the center. From the Y-X View it can be seen that we have divided the disks of fuel into four quadrants.

In fig. 22 we have switched to **Page 2: Surface** and we indicate the zones that we will make transparent in order to look inside the can. Fig. 23 shows the result after we have made one quadrant of the can transparent. Note, that we have left the top and bottom cladding on the can. In fig. 24 we have rotated our eye 45 degrees to the left and down 45 degrees (the -45 shown on the figure) and used the **Show Surface** to display a 3-D image allowing us to look into the interior of the can.

In fig. 25, the final figure for this example, we have removed the top and bottom of the can and display all three views. This is usually not recommended because each view is independently rotated the same 45 degrees left and down starting from its initial position, and usually this doesn't produce interesting results in all three views. Note, in the Z-X and Y-X Views this rotation positions us to a point where we can see inside the can, but the Z-Y View has rotated away from the removed quadrant and doesn't show very much. What isn't immediately obvious is that the Z-X and Y-X views may look similar, but they are actually views of opposite ends of the can. Note, the colors of the interior fuel disks, pink is on top in the Y-X View and on the bottom in the Z-X View. Similarly the orange on top in the Z-X View is on bottom in the Y-X View.

### Torus

The following examples are coming attractions to show what future versions of TART will be able to do. In these examples we illustrate that **TARTCHEK** has already been updated to handle cubic and quartic surfaces. Figs. 26 through 29 illustrate that sometimes for complicated geometry based on the three 2-D slices through the geometry it may still be difficult to visual what the 3-D object looks like.

In figs. 26 and 27 we illustrate the three 2-D slices and a 3-D perspective display of a case involving an accelerator beam tube and its supporting structure and insulators. In this case it is very difficult to tell from fig. 26 what the object looks like, but from fig. 27 it is obvious. Figs. 28 and 29 illustrate similar results for a different object; again, it is much more obvious what the object looks like from the 3-D display in fig. 29 than it is from the

2-D slices in fig. 28. These are but a few examples of the use of the **TARTCHEK** options to display 3-D objects.

The above example involve a torus, which is a specialized, rather simple form of a quartic (fourth order) surface. The final figure, fig. 30, illustrates a more general quartic surface. This is what we call our "Hour Glass", where the zoning has been defined to make it look like an hour glass with sand slowly flowing from the upper half to the lower half of the hour glass. The point to note here is that the entire surface of the hour glass has been defined by one single quartic surface, greatly simplifying the zoning. In this simple example we can look at the entire geometry and see the hour glass, or we can look at parts of it and see other possibilities. For example, the green zone is a good approximation to a nozzle; by changing only one of the quartic parameters we can describe an entire family of nozzles. Similarly the combination of red and green zones looks like a water tower. There are lots of possibilities involving quartic surfaces, limited only by one's imagination.