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SAND2002-3868
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Printed November 2002

BURNCAL: A Nuclear Reactor Burnup Code Using MCNP Tallies

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BURNCAL: A Nuclear Reactor Burnup Code Using MCNP Tallies

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Abstract

BURNCAL is a Fortran computer code designed to aid in analysis, prediction, and optimization of fuel burnup performance in a nuclear reactor. The code uses output parameters generated by the Monte Carlo neutronics code MCNP to determine the isotopic inventory as a function of time and power density. The code allows for multiple fueled regions to be analyzed. The companion code, RELOAD, can be used to shuffle fueled regions or reload regions with fresh fuel. BURNCAL can be used to study the reactivity effects and isotopic inventory as a function of time for a nuclear reactor system. Neutron transmutation, fission, and radioactive decay are included in the modeling of the production and removal terms for each isotope of interest. For a fueled region, neutron transmutation, fuel depletion, fission-product poisoning, actinide generation, and burnable poison loading and depletion effects are included in the calculation. Fueled and un-fueled regions, such as cladding and moderator, can be analyzed simultaneously. The nuclides analyzed are limited only by the neutron cross section availability in the MCNP cross-section library. BURNCAL is unique in comparison to other burnup codes in that it does not use the calculated neutron flux as input to other computer codes to generate the nuclide mixture for the next time step. Instead, BURNCAL directly uses the neutron absorption tally/reaction information generated by MCNP for each nuclide of interest to determine the nuclide inventory for that region. This allows for the full capabilities of MCNP to be incorporated into the calculation and a more accurate and robust analysis to be performed.

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EXECUTIVE SUMMARY

There are many coupled neutronic/isotope-composition codes available to the nuclear community that are capable of performing burnup and depletion calculations on nuclear reactor fuels or fuel element assemblies. In general, these coupled codes all work in a similar fashion - a neutronics code performs a calculation to determine the multiplication factor, k_{eff}, and the energy dependent neutron flux for a given geometry and fuel loading; another code then uses the neutron flux, power density, and material composition to determine the new material composition after a given burnup time. The material composition can include transmuted isotopes, fission products, actinides, and burnable poisons, as well as the fissile component of the fuel. Many of these codes have been used extensively by the nuclear industry, perform adequately, and give perfectly acceptable answers for their intended purpose. However, using a neutronics code such as MCNP to directly determine the neutron absorption rate for each individual nuclide as part of the neutronics calculation, thereby eliminating the need to include neutron flux information in the burnup calculation, is a more straightforward and accurate solution method. To the author's knowledge, applying this type of a methodology on a modern neutronics analysis code has never been accomplished. The goal of this work was to develop a burnup analysis code that uses the full capability of the neutronics analysis code to provide the highest fidelity solution. BURNCAL uses the neutron absorption and fission information generated from the neutronics code in calculating the nuclide mixture for the next time step of interest. BURNCAL allows for multiple regions to be simultaneously analyzed and would track fissile and fissionable isotope burnup, fission product and actinide production, absorption and decay, and general isotope production by transmutation, absorption and decay. A companion post-processor code would allow for reloading and shuffling of fueled regions with fresh, spent, or reprocessed fuel.

MCNP was chosen as the neutronics code to be used in the analysis for a number of reasons:

- MCNP has been used extensively and with a large degree of success by the author in performing reactor design and safety analyses on research reactors;
- MCNP allows for neutron reaction information, referred to as "tallies", to be generated during the neutronics calculation, the results of which are output in a readable format;
- MCNP4B and later versions are distributed with a cross-section library that includes most of the important fission-product nuclides as continuous-energy cross sections.

BURNCAL, was written as a Fortran computer code that uses the output generated by MCNP version 4B. Later versions of MCNP should also work with BURNCAL with minor or no modifications. BURNCAL was developed to incorporate the tally information into a burnup/production calculative model. The model uses the nuclide inventory, MCNP tally information, power density, fission-product yield data, and half-life data to determine the new nuclide inventory for a given region at the next time step. The new inventories are then automatically placed back into the MCNP input file and the case run for the next time step.

At the completion of an analysis, the companion code, RELOAD, can be used to shuffle fuel regions or reload regions with fresh fuel at a specific time in the analysis. The analysis can then be repeated with the new loading scheme.

BURNCAL has been successfully used to analyze burnup effects for low-density fueled, large graphite cores and for small space-type reactor systems. Homogeneous and heterogeneous geometries have been studied. BURNCAL has been run successfully for both k infinity and k_{eff} geometries. For the k infinity cases, the outer surface of a fuel element/moderator model was maintained as a specular reflector in the MCNP input. This allowed the reactor to be modeled as an infinite core made up of one type of fuel element. The computer time for a k infinity analysis is small compared to an analysis on a complete reactor core. However, to perform more complex analyses using different regional loading schemes or fuel shuffling and reloading, a complete reactor geometry must be modeled. The computer time required to run a burnup analysis is dominated by the statistical accuracy desired in the results from MCNP.

1. INTRODUCTION

Burnup and depletion codes have been developed and used in the nuclear industry since the introduction of digital computing. Some of the original codes included CANDLE, DRACO, and TURBO (ANL-5800). These codes solved the diffusion equation in one to three dimensions using a few neutron-energy groups. Only a few of the major fission products were included in the calculation. As the digital computing capabilities and the neutron cross-section libraries developed, more sophisticated solution techniques, such as discrete-ordinates and Monte Carlo methods, were employed to solve for k_{eff} and the multigroup neutron flux.

Codes used today to calculate burnup reactivity effects include coupled neutronics/burnup codes and post-processing modules. The neutronics code is used to calculate k_{eff} and the energy-dependent neutron flux for a reactor or a region within the reactor. Typical neutronics codes include S_n-type codes like XSDRN (Greene and Craven, 1969) or Monte Carlo codes like MCNP (Briesmeister, 1997). The burnup code then uses the neutron flux information, loading composition, and power density to compute the nuclide composition for the next time step. The process is repeated for the duration of the burnup cycle. Usually a zero-dimensional, one-group, depletion/fission-product generation code is used to perform the burnup analysis. Typical depletion/fission-product generation codes include ORIGEN (Bell, 1973) (Croff, 1983) and CINDER (England, 1962).

There are many coupled burnup codes available to the nuclear community. Each is unique in its neutronic modeling technique or specific function. For example, a code may be designed to specifically analyze boiling-water reactor fuel assemblies. However, all of the codes follow a similar solution methodology. That is, k_{eff} and the neutron flux are calculated and input into a coupled depletion/fission-product generation code. Although not inclusive, some of these codes include SAS2H (Hermann and Parks, 2000) from the SCALE Modular Code System (Parks, 1987), VENTURE (Shapiro and Huria, 1990), (Vondy, et al., 1981), COSMO (Uchikawa and Takeda, 1974), LASER (Poncelet, 1966), EPRI-CELL (ARMP-02, 1988a), PDQ7 (ARMP-02, 1988b), CORSYS (Caner, et al., 1994), MOCUP (Moore, et al., 1995), and MONTEBURNS (Poston and Trellue, 1999). These and other codes have been used extensively and perform adequately for their intended purpose.

The highest fidelity approach simply uses neutron absorption and fission reaction information generated in the neutronics code to determine the nuclide composition at the next time step. This type of model allows for all of the neutron flux information to be integrated into the calculation without post-processing and additional manipulation of the neutron flux and cross-section set. Neutron absorption and fission reaction data for individual nuclides are available as output for Monte Carlo codes like MCNP through the use of "tallies". The only requirement is that an energy-dependent cross-section set is available for each nuclide of interest at the required material temperature.

The Fortran code, BURNCAL, was developed to incorporate the neutron absorption reaction output available using the MCNP tally specification cards into a burnup/production calculative model. The code uses the nuclide inventory, MCNP tally information, power density, fission-

product yield data, and half-life data to determine the new nuclide inventory for a given region at the new time. The new inventories are then automatically placed back into the MCNP input file and the case run for the next time step.

At the completion of an analysis, the companion code, RELOAD, can be used to shuffle fuel regions or reload regions with fresh fuel at a specific time in the analysis. The analysis can then be repeated with the new loading scheme.

2. MODELING

2.1 Code Overview

The BURNCAL code package consists of two independent parts: the burnup modules and the reload module. The overall flow of the burnup modules is shown in Figure 1. A more thorough description of the code is presented in a later section. The burnup modules use a script or batch file to run MCNP, MCNPDATA, and BURNCAL in an iterative fashion. The script file runs MCNP on an initial input deck, the output of which is reduced into a smaller and more manageable size using the developed Fortran module MCNPDATA. The script file then runs the developed Fortran module BURNCAL, which uses the reduced MCNP output file, the MCNP input file, a nuclide library data file, an isotope concentration file, and a power history file to generate a new isotopic inventory for each region for the next time step designated in the power history file. BURNCAL then places the new inventory for each region into the MCNP input file. The script file then repeats the steps until the last time step is reached. After the analysis is completed, regions of the core can be reloaded with fresh, burned, or reprocessed fuel using the developed Fortran module RELOAD. RELOAD uses the MCNP input module and the isotope concentration history output file generated by BURNCAL to generate the new reactor core inventory.

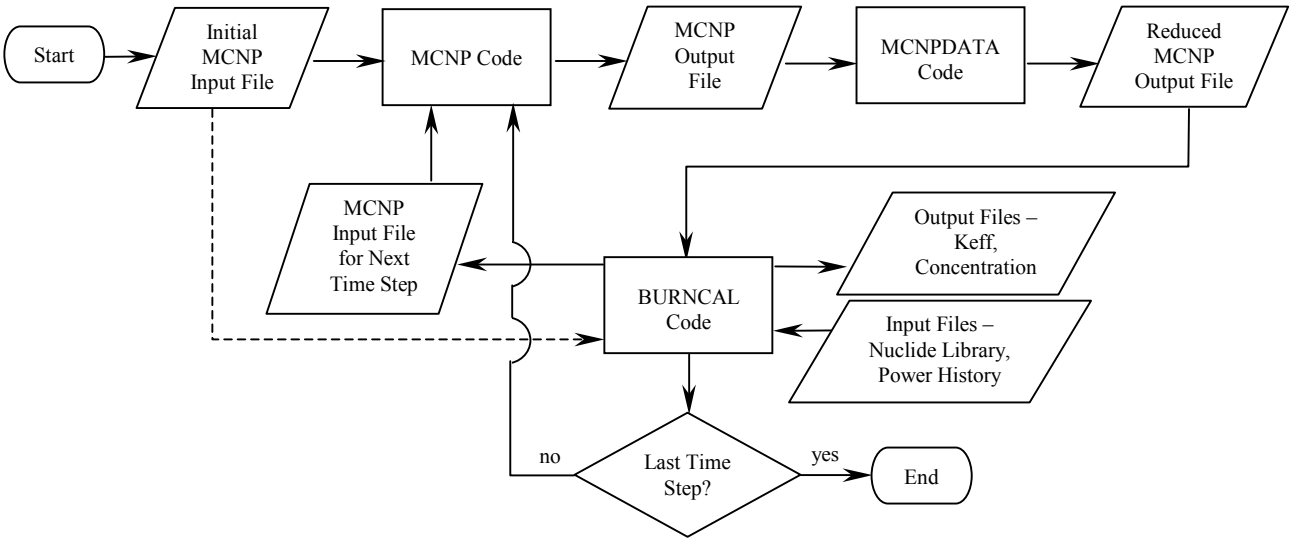

Figure 1. Code Flow for the Burnup Modules.

2.2 System Requirements

The BURNCAL code package, which includes BURNCAL, MCNPDATA, and RELOAD, were written using the Fortran 77 programming language. The script or batch file was written for a PC running Windows NT in a DOS Window. However, the code package should be able to run on any computer platform that runs MCNP. In order to use the package, the source modules must be compiled using a Fortran compiler, or the existing executable files may be used for a PC. The script file must be written to be compatible with the computer platform, or the batch file written for a PC may be used. MCNP version 4B is assumed to be installed and operating with the Kidman fission-product cross-section library or other cross-section libraries that include the transmuted or fission-product isotopes of interest.

2.3 Design Goals and Requirements

The main goal in the development of the BURNCAL code package was to construct a burnup code that would use only the neutron absorption tally/reaction information generated by MCNP version 4B in the calculation of fissioned or neutron-transmuted isotopes for different fuel-loaded regions. Accomplishing this goal would allow for a simple, straightforward, and accurate calculation to be performed without having to use the calculated group fluxes to perform transmutation analysis in a separate code. Using the MCNP tally information directly in the computation allows for the most information to be used in the analysis. Also, using MCNP and the continuous-energy cross sections allows for cross section manipulations to be avoided.

In order to accomplish this goal, the following requirements were imposed in the development of the code package:

- The code must be able to read and interpret an MCNP input file such that isotopic concentrations can be identified for different fueled and unfueled regions of a core.
- The code must be able to read and interpret an MCNP version 4B output file such that fission and transmutation reaction information can be resolved for each isotope and region of interest.
- The code must be able to numerically calculate the production and removal of a specific isotope through fission, transmutation, and radioactive decay.
- The code must be able to read, interpret, and write to an MCNP input file such that the resultant isotopic concentrations can be reinserted into the MCNP input file at the proper locations.

In order to fulfill the above requirements, the code package must be able to read and interpret an MCNP input file correctly. In order to perform this task for a multi-region problem, the input file must be structured in such a way that the code package can logically interpret the input information. This requires the use of specific cell numbers, material numbers, keywords, and a particular structure and flow to the input file. This unique structure is described in a later section.

2.4 Production/Removal Equations

The time rate of change in the concentration of a specific isotope is equal to its production rate per unit volume less its removal rate per unit volume. Production can be by 1) fission of a fissile isotope (e.g., U-235, U-233, Pu-239) or a fissionable isotope (e.g., U-238), which produces the isotope of interest as a fission product, 2) neutron capture of a parent isotope transmuted to the isotope of interest, or 3) decay of a parent isotope resulting in the isotope of interest. Removal can be by 1) fission, if the isotope of interest is fissile or fissionable, 2) neutron capture transmuting the isotope to another, or 3) decay to a daughter product.

Time Rate of Change in Concentration of Isotope i	=	Production Rate of Isotope i Per Unit Volume	–	Removal Rate of Isotope i Per Unit Volume

Production Rate =>
- Fission – isotope i generated as a fission product
- Neutron Capture – isotope i generated by neutron transmutation
- Decay – isotope i generated from decay of a parent isotope

Removal Rate =>
- Fission – isotope i removed by fission if fissile or fissionable
- Neutron Capture – isotope i removed by neutron transmutation
- Decay – isotope i removed by radioactive decay

Using the above rate information, a general coupled set of linear first-order differential equations can be developed for all of the isotopes of interest. These equations can be solved by using analytical techniques (e.g., Laplace transforms) or by numerical methods (e.g., Euler's method or Runge-Kutta method). Due to the fact that a large number of coupled equations can result, a numerical solution technique is the only practical solution method. For each isotope, the following general differential equation is valid:

$$\frac{dN_i}{dt} = \sum_j \gamma_{ji}\sigma_{f,j}N_j\phi + \sum_k \sigma_{c,k\to i}N_k\phi + \sum_l \lambda_{l\to i}N_l - \left(\sigma_{f,i}N_i\phi + \sigma_{c,i}N_i\phi + \lambda_i N_i\right) \quad (1)$$

where

$\frac{dN_i}{dt}$ = time rate of change in concentration of isotope i,

$\sum_j \gamma_{ji}\sigma_{f,j}N_j\phi$ = production rate per unit volume of isotope i from fission of all fissionable nuclides,

$\sum_k \sigma_{c,k\to i} N_k \phi$ = production rate per unit volume of isotope i from neutron transmutation of all isotopes including (n,γ), (n,α), etc.,

$\sum_l \lambda_{l\to i} N_l$ = production rate per unit volume of isotope i from decay of all isotopes including β^-, α, β^+, etc.,

$\sigma_{f,i} N_i \phi$ = removal rate per unit volume of isotope i by fission,

$\sigma_{c,i} N_i \phi$ = removal rate per unit volume of isotope i by neutron capture (excluding fission),

$\lambda_i N_i$ = removal rate per unit volume of isotope i by decay.

In the above equation, N_i represents the concentration of isotope i in atoms/cm^3, t is time in seconds, γ_{ji} is the fission fraction for the production of isotope i from fission of isotope j, $\sigma_{f,i}$ is the microscopic fission cross section for isotope i in cm^2, $\sigma_{c,i}$ is the microscopic total capture cross section (absorption minus fission) for isotope i in cm^2, ϕ is the neutron flux in n/cm^2-s, and λ_i is the decay constant (equal to ln(2)/half-life) for isotope i in 1/s.

Note that, although not explicitly defined as such, the neutron flux and cross sections are energy dependent. The neutron flux and isotope concentration are both position and time dependent. Hence, the flux, cross section, and concentration should really be defined as

$\phi \;\Rightarrow\; \phi(x,y,z,E,t)$,

$\sigma \;\Rightarrow\; \sigma(E)$, and

$N_i \;\Rightarrow\; N_i(x,y,z,t)$.

The energy dependence can be integrated out of the differential equation, since only the flux and cross sections are energy dependent. That is,

$$\sigma N \phi \;\Rightarrow\; N(x,y,z,t) \int_0^\infty \sigma(E)\, \phi(x,y,z,E,t)\, dE \quad .$$

Hence, the time- and space-dependent isotope concentrations can be solved from a set of coupled linear first-order differential equations in time.

Equation 1 may be simplified if it is assumed that isotope i is produced by fission of one fissile isotope, decay of one parent isotope, and absorption by one parent isotope. Including all neutron absorption (i.e., neutron capture and fission) of isotope i into one term results in the following equation:

$$\frac{dN_i}{dt} = \gamma_i F + \sigma_c Y\phi + \lambda_P P - \left(\sigma_{a,i} N_i \phi + \lambda_i N_i\right) \tag{2}$$

where

$\frac{dN_i}{dt}$ = time rate of change in concentration of isotope i,

$\gamma_i F$ = production rate per unit volume of isotope i from fission, where F is the fission rate,

$\sigma_c Y\phi$ = production rate per unit volume of isotope i from neutron transmutation of a parent isotope, where Y is the concentration of the parent isotope,

$\lambda_P P$ = production rate per unit volume of isotope i from decay of a parent isotope, where P is the concentration of the parent isotope,

$\sigma_{a,i} N_i \phi$ = removal rate per unit volume of isotope i by neutron capture (including fission), where $\sigma_{a,i}$ is the total microscopic absorption cross section for isotope i,

$\lambda_i N_i$ = removal rate per unit volume of isotope i by decay.

For Equation 2, N, Y, and P are all time-dependent quantities. The solution to Equation 2 can be found by using Laplace transforms.

$$\overline{N}_i(s) = \frac{N_i(0)}{s + (\sigma_{a,i}\phi + \lambda_i)} + \frac{\gamma_i F}{s\,(s + (\sigma_{a,i}\phi + \lambda_i))} + \left[\frac{\sigma_c \phi \overline{Y}(s) + \lambda_P \overline{P}(s)}{s + (\sigma_{a,i}\phi + \lambda_i)}\right] \tag{3}$$

$$N_i(t) = N_i(0)\, e^{-(\sigma_{a,i}\phi + \lambda_i)t} + \frac{\gamma_i F}{\sigma_{a,i}\phi + \lambda_i}\left(1 - e^{-(\sigma_{a,i}\phi + \lambda_i)t}\right) + L^{-1}\left[\frac{\sigma_c \phi \overline{Y}(s) + \lambda_P \overline{P}(s)}{s + (\sigma_{a,i}\phi + \lambda_i)}\right] \tag{4}$$

Equation 4 is not a simple solution for isotope chains involving more than a few isotopes. Hence, although not practical for application in a burnup code, Equation 4 may be used to check the accuracy of the code for isotope chains involving a few isotopes.

For example, Xe-135 and Sm-149 are two of the most important fission products due to their large absorption cross section and yield from fission. Both chains allow for relatively simple solutions to be determined. The 135 chain, excluding Te-135 and Xe-135m due to their short half-lives, is shown in Figure 2.

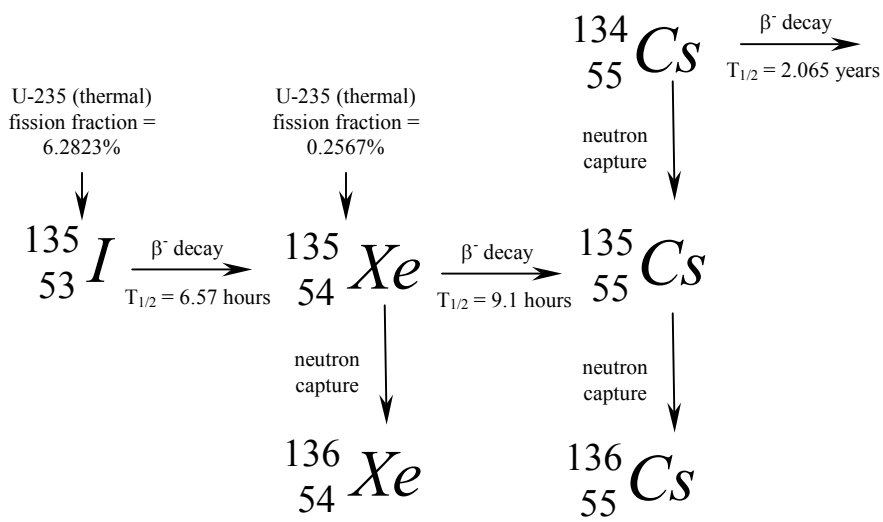

Figure 2. The 135 Fission Product Chain.

The solution for Cs-135 is complicated by the feed-in from Cs-134, Cs-133, and their respective chains. However, I-135 and Xe-135 can be solved analytically with the following result:

$$I^{135}(t) \;=\; I^{135}(0)\, e^{-\lambda_I t} \;+\; \frac{\gamma_I}{\lambda_I} F \left(1 - e^{-\lambda_I t}\right) \tag{5}$$

$$\begin{aligned} Xe^{135}(t) \;&= Xe^{135}(0)\, e^{-(\sigma_{Xe}\phi + \lambda_{Xe})\, t} + \frac{\gamma_{Xe} F}{\sigma_{Xe}\phi + \lambda_{Xe}} \left(1 - e^{-(\sigma_{Xe}\phi + \lambda_{Xe})\, t}\right) \\ &+ \lambda_I\, I^{135}(0) \left[\frac{e^{-\lambda_I t}}{\sigma_{Xe}\phi + \lambda_{Xe} - \lambda_I} + \frac{e^{-(\sigma_{Xe}\phi + \lambda_{Xe})t}}{\lambda_I - \sigma_{Xe}\phi - \lambda_{Xe}} \right] \\ &+ \lambda_I\, \gamma_I F \left[\frac{1}{\lambda_I\,(\sigma_{Xe}\phi + \lambda_{Xe})} + \frac{e^{-\lambda_I t}}{-\lambda_I\,(\sigma_{Xe}\phi + \lambda_{Xe} - \lambda_I)} + \frac{e^{-(\sigma_{Xe}\phi + \lambda_{Xe})t}}{-(\sigma_{Xe}\phi + \lambda_{Xe})(\lambda_I - \sigma_{Xe}\phi - \lambda_{Xe})} \right] \end{aligned} \tag{6}$$

where $I^{135}(t)$ and $Xe^{135}(t)$ are in atoms/cm^3, F is in fissions/cm^3-s, $\sigma_{Xe}\phi$ is in 1/s, λ_{Xe} and λ_I are in 1/s, and t is in seconds. The term $\sigma_{Xe}\phi$ is the neutron absorption reaction rate, derived from the power density and the neutron capture tally generated in the output of the MCNP run. $I^{135}(0)$ and

$Xe^{135}(0)$ can normally be set equal to zero. However, since $\sigma_{Xe}\phi$ and the power density can change at each time step, $I^{135}(0)$ and $Xe^{135}(0)$ represent the concentrations at the end of the previous time step.

The 149 chain is shown in Figure 3.

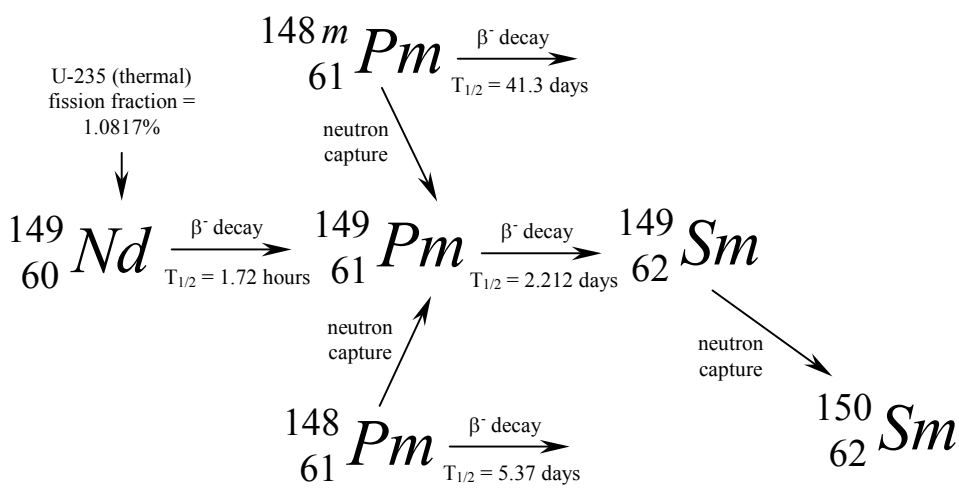

Figure 3. The 149 Fission Product Chain.

The solution is complicated by the feed-in from Pm-148, Pm-148m, and the 147 chain that feeds these two isotopes. The feed-in is through Nd-147, which is produced as a fission product and decays by beta decay with a half-life of 10.98 days to Pm-147. Pm-147 can decay by beta decay with a half-life of 2.6234 years to Sm-147, or be transmuted to Pm-148m or Pm-148 by neutron capture. Since two neutron absorptions are required to reach Pm-149 (i.e., one absorption to reach Pm-148 or Pm-148m and another to reach Pm-149), this feed-in can be ignored for purposes of determining an analytical solution. Ignoring this feed-in, the solution for the 149 chain is as follows:

$$Nd^{149}(t) \quad = \quad Nd^{149}(0)\, e^{-\lambda_{Nd}\, t} \quad + \quad \frac{\gamma_{Nd}}{\lambda_{Nd}}\, F\left(1 - e^{-\lambda_{Nd}\, t}\right) \tag{7}$$

$$Pm^{149}(t) \quad = Pm^{149}(0)\, e^{-\lambda_{Pm}\, t} \quad + \quad \lambda_{Nd}\, Nd^{149}(0)\left[\frac{e^{-\lambda_{Nd}\, t}}{\lambda_{Pm} - \lambda_{Nd}} + \frac{e^{-\lambda_{Pm}\, t}}{\lambda_{Nd} - \lambda_{Pm}}\right]$$

$$+ \lambda_{Nd}\, \gamma_{Nd} F \left[\frac{1}{\lambda_{Nd}\, \lambda_{Pm}} + \frac{e^{-\lambda_{Nd} t}}{-\lambda_{Nd}\,(\lambda_{Pm} - \lambda_{Nd})} + \frac{e^{-\lambda_{Pm} t}}{-\lambda_{Pm}\,(\lambda_{Nd} - \lambda_{Pm})} \right] \tag{8}$$

$$Sm^{149}(t) \;\; = Sm^{149}(0)\, e^{-\sigma_{Sm}\phi\, t} \;\; + \;\; \lambda_{Pm} Pm^{149}(0) \left[\frac{e^{-\lambda_{Nd}\, t}}{\sigma_{Sm}\phi - \lambda_{Nd}} + \frac{e^{-\sigma_{Sm}\phi\, t}}{\lambda_{Nd} - \sigma_{Sm}\phi} \right]$$

$$+ \lambda_{Pm}\, \lambda_{Nd}\, Nd^{149}(0) \left[\frac{e^{-\sigma_{Sm}\phi t}}{(\lambda_{Pm} - \sigma_{Sm}\phi)(\lambda_{Nd} - \sigma_{Sm}\phi)} + \frac{e^{-\lambda_{Pm} t}}{(\sigma_{Sm}\phi - \lambda_{Pm})(\lambda_{Nd} - \lambda_{Pm})} + \frac{e^{-\lambda_{Nd} t}}{(\sigma_{Sm}\phi - \lambda_{Nd})(\lambda_{Pm} - \lambda_{Nd})} \right]$$

$$+ \lambda_{Pm}\, \lambda_{Nd}\, \gamma_{Nd} F \left[\frac{1}{\sigma_{Sm}\phi\, \lambda_{Pm}\, \lambda_{Nd}} + \frac{e^{-\sigma_{Sm}\phi\, t}}{-\sigma_{Sm}\phi\,(\lambda_{Pm} - \sigma_{Sm}\phi)(\lambda_{Nd} - \sigma_{Sm}\phi)} \right.$$

$$\left. + \frac{e^{-\lambda_{Pm} t}}{-\lambda_{Pm}\,(\sigma_{Sm}\phi - \lambda_{Pm})(\lambda_{Nd} - \lambda_{Pm})} + \frac{e^{-\lambda_{Nd} t}}{-\lambda_{Nd}\,(\sigma_{Sm}\phi - \lambda_{Nd})(\lambda_{Pm} - \lambda_{Nd})} \right] \tag{9}$$

where $Nd^{149}(t)$, $Pm^{149}(t)$, and $Sm^{150}(t)$ are in atoms/cm^3, F is in fissions/cm^3-s, $\sigma_{Sm}\phi$ is in 1/s, λ_{Nd} and λ_{Pm} are in 1/s, and t is in seconds. The term $\sigma_{Sm}\phi$ is the neutron absorption reaction rate, derived from the power density and the neutron capture tally generated in the output of the MCNP run. $Nd^{149}(0)$, $Pm^{149}(0)$, and $Sm^{150}(0)$ can normally be set equal to zero. However, since $\sigma_{Sm}\phi$ and the power density can change at each time step, $Nd^{149}(0)$, $Pm^{149}(0)$, and $Sm^{150}(0)$ represent the concentrations at the end of the previous time step.

2.5 Solution Technique

Equation 1 or the simplified Equation 2 version may be solved using a variety of numerical solution techniques. The method chosen for this code was to use a simple Euler's method (1st order Runge-Kutta) with sufficiently small time steps to ensure adequate accuracy in the results. Using this technique, Equation 2 becomes

$$N_i(t + \Delta t) \;\; = \;\; N_i(t) \;\; + \;\; \big(\gamma_i F + \sigma_c \phi Y(t) + \lambda_P P(t)\big)\Delta t \;\; - \;\; \big(\sigma_{a,i}\phi + \lambda_i\big) N_i(t)\, \Delta t \tag{10}$$

where $t + \Delta t$ is the time at the new time step. The code uses the tallies generated from MCNP to determine F, $\sigma_c\phi$ and $\sigma_{a,i}\phi$. The accuracy of this solution is dependent on the size of the time step. Numerical error associated with this solution technique will be discussed later for a sample problem.

Equation 10 is used in the code for each fission, actinide production, and transmutation chain of interest for each region analyzed. Regions do not have to maintain a fissile isotope in order to be evaluated. Hence, cladding, moderator, and other regions of a core or a fuel element can be evaluated to determine transmutation or poison burnup effects.

Although there are some exceptions, most fission products and short-lived actinides are found to decay by beta emission, and most transmutations are found to occur by radiative capture (i.e., n,γ). Under these assumptions, if $N_i(t)$ is isotope X(Z,A), where Z is the atomic number and A is the mass number, $Y(t)$ is equal to isotope X(Z,A-1), and $P(t)$ is equal to isotope X(Z-1,A). Equation 10 represents the solution for production by one type of fission, decay, and transmutation, and removal by absorption or decay.

In order to solve Equation 10, γ_i, λ_p, and $\lambda_{a,i}$ must be known, and F, $\sigma_c\phi$, and $\sigma_{a,i}\phi$ must be calculated. The values for F and ϕ are dependent on the average power density of the core and the region being analyzed. Determining F, $\sigma_c\phi$, and $\sigma_{a,i}\phi$ requires the use of the fission and absorption tally results for each region. The MCNP reaction tallies are in units of reactions/cm^3-source neutron or reactions/cm^3-source neutron per atom/barn-cm, depending on how the tally is specified. In order to use the tally value properly in Equation 10, the power density and unit conversion must be included in the calculation. The value for F, the number of fissions/cm^3-s, for a given region can be mathematically expressed as follows:

$$F_l = Tally(f,l) \frac{P'''V_c C}{\sum_n Tally(f,n)V_n} \tag{11}$$

where

F_l = fission rate for region l, in units of fissions/cm^3-s,

$Tally(f,l)$ = fission tally for region l, in units of fissions/cm^3-source neutron,

V_n = volume for region n, in units of cm^3,

$P'''V_c$ = core average power density times the core volume, in units of Watts (W),

C = power to fission rate conversion factor, equal to 3.285 x 10^{10} fissions/W-s, assuming 190 MeV per fission (excludes neutrino energy).

The value for $\sigma_c\phi$ or $\sigma_{a,i}\phi$, the transmutation or absorption rate, can be mathematically expressed as follows:

$$(\sigma_c\phi)_m = tally(c,m) \; D \frac{P'''V_c C}{\sum_n Tally(f,n)V_n} \tag{12}$$

where

$(\sigma_c\phi)_m$ = transmutation or absorption rate for region m, in units of transmutations/atom-s,

tally(*c,m*) = transmutation or absorption tally for region *m*, in units of transmutations/cm^3-source neutron per atom/b-cm,

D = barn conversion factor, equal to 1.0×10^{-24} cm^2/barn.

3. CODE DESCRIPTION

A general description of the BURNCAL code package was presented in Section 2.1. The code modules, input, output, and library files required to perform a burnup analysis were also presented along with a flow description. The intent of this section is to present a more in-depth description of the code package, with more emphasis on the details of the BURNCAL code. Limitations and potential future enhancements to the code will also be presented.

The code package and a sample problem are presented in the Appendices. The functionality of each of the code modules and files included in, or generated by, the BURNCAL code package is as follows:

- Script/Batch File for DOS – BURNM.BAT – Appendix A – Controls the overall code flow; starts a new job; cleans the run directory; runs MCNP; runs MCNPDATA; runs BURNCAL; stops job.

- Monte Carlo Neutronics Code – MCNP4B – Performs the neutronic analysis on the MCNP input file, calculates the effective multiplication factor (k_{eff}) and reaction tallies for each isotope and region of interest.

- Output File Manipulation - MCNPDATA.for – Appendix B - Reduces the size of the MCNP output file to include only the information required for the analysis.

- Burnup Analysis Code - BURNCAL.for – Appendix C - Performs the burnup analysis using the reduced size MCNP output file, MCNP input file, LIBDATA.IN, SETUP.IN, CELL.TMP, ISO.TMP, and generates files KEFF.OUT, and CONC.OUT.

- Data Library – LIBDATA.IN – Appendix D – Contains nuclide data required to perform analysis including the atomic number, mass number, fission fraction (U-235 thermal), and half-life.

- MCNP Input File – Appendix E – MCNP input file that is structured in such a way as to allow for a logical interpretation of the information.

- Reduced MCNP Output File – Appendix F – MCNP output file reduced using MCNPDATA that contains the required information for running BURNCAL.

- Power History Input File – SETUP.IN – Appendix G – Includes the power density history and time step requirements for the analysis.

- Volume and Cell Information File – CELL.TMP – Appendix H – Generated at the first time step by BURNCAL, this file is used in subsequent iterations to find the total core volume, the number of cells/regions to be analyzed, their identification number, and their volume.

- Tally Identifier File – TAL.TMP – Appendix I – Generated at the first time step by BURNCAL, this file is used in subsequent iterations to identify the tally numbers to be read from the reduced MCNP output file.

- Temporary Isotopic Inventory File – ISO.TMP – Appendix J– Regenerated by BURNCAL at the end of each iteration, this file is used to find the starting isotopic inventory for each region at the next time step.

- Isotopic Inventory History File – CONC.OUT – Appendix K– Same as ISO.TMP, but maintains the successive inventory for each iteration time step.

- Effective Multiplication Factor (k_{eff}) History File – KEFF.OUT – Appendix L– Updated by BURNCAL at each iteration, this file maintains the values for k_{eff} and the standard deviation generated by MCNP at each iteration time step.

- Isotopic Inventory/Region Manipulation – RELOAD.for – Appendix M – A separate code that uses some of the input and output subroutines of BURNCAL to allow for fuel reloading, shuffling, or reprocessing.

In order to run the code, the user must supply all of the files with a ".IN" designation (e.g., LIBDATA.IN and SETUP.IN) as well as the MCNP input file. The MCNP input file must have the correct structure such that the BURNCAL code may properly interpret the information in the file.

3.1 BURNCAL Code Description

As described in Section 2.1, the script file controls the overall flow of the code package and runs BURNCAL at every iteration in the burnup analysis. BURNCAL is a Fortran 77 program that is made up of a number of subprograms that perform the necessary data interpretation requirements, calculate the new nuclide inventory, and insert the new data into the MCNP input file. The subroutines and their function are as follows:

- Main Program– Controls overall flow in the code and determines when the last iteration is reached. Reads SETUP.IN to find the power density and time step values.

- routkef – reads the reduced MCNP output file and finds k_{eff} and the standard deviation.

- prtkeff – writes k_{eff} and the standard deviation to KEFF.OUT.

- rdprop – reads the LIBDATA.IN file for isotope identifiers, fission fractions, and half-lives.

- readin1 – reads the MCNP input file at the first iteration only. Finds the cell numbers, material numbers, and volumes to be used in the evaluation. Calculates the total core volume. Prints information to CELL.TMP. Finds the word "keyword" and "mcard tallies" in the input file. Finds the material identifier for each tally. Prints information to TAL.TMP.

- readinp – reads the MCNP input file at the first iteration only. Finds the initial isotope concentration for each region (i.e., cell number) and stores information in arrays.

- storeiso – writes the isotope concentrations for each region to ISO.TMP at the beginning of the first iteration and at the end of each subsequent iteration.

- prtconc – writes the isotope concentrations for each region to CONC.OUT at the beginning of the first iteration and at the end of each subsequent iteration.

- readinf – reads CELL.TMP and stores data.

- readiso – reads ISO.TMP and stores data.

- routtal – reads the reduced MCNP output file and finds the tally results for each tally identified in TAL.TMP.

- prtinp – writes the new nuclide inventory for each region to the MCNP input file.

- solve – uses all of the information (nuclides, inventories, power density, time step, fission fractions, half-lives, and MCNP tally results) to solve Equations 10, 11, and 12 for each region at each time step.

Most of the code involves reading, writing, and manipulating files and data in order to find the correct information to be used in the analysis, and then placing the data generated as a result of the analysis back into an MCNP input file. All of the burnup calculations are performed within the subroutine "solve."

The flow of the BURNCAL code is shown in Figure 2. The code reads the first line in SETUP.IN and ISO.TMP and records the iteration value. The subroutines "routkef" and "prtkeff" are then called that read k_{eff} and the standard deviation from the reduced MCNP output file and record the information in KEFF.OUT. A decision is made as to whether this is the last iteration or if k_{eff} is less than 0.61 (an arbitrary lower value for k_{eff}). If either is true, a new file is generated called STOP.OUT, and the code is terminated. The STOP.OUT file is used as an identifier in the script program to determine when to stop the iteration process. If either is false, the program continues.

The subroutine “rdprop” is called, which reads LIBDATA.IN and stores in an array the U-235 thermal fission fraction and the half-life for each isotope that is tracked in a chain. The data is stored for each nuclide by the Z (atomic number) and M (mass number) as the row and column identifiers of the array. Nuclides that are not depleted or produced in the calculation are also identified.

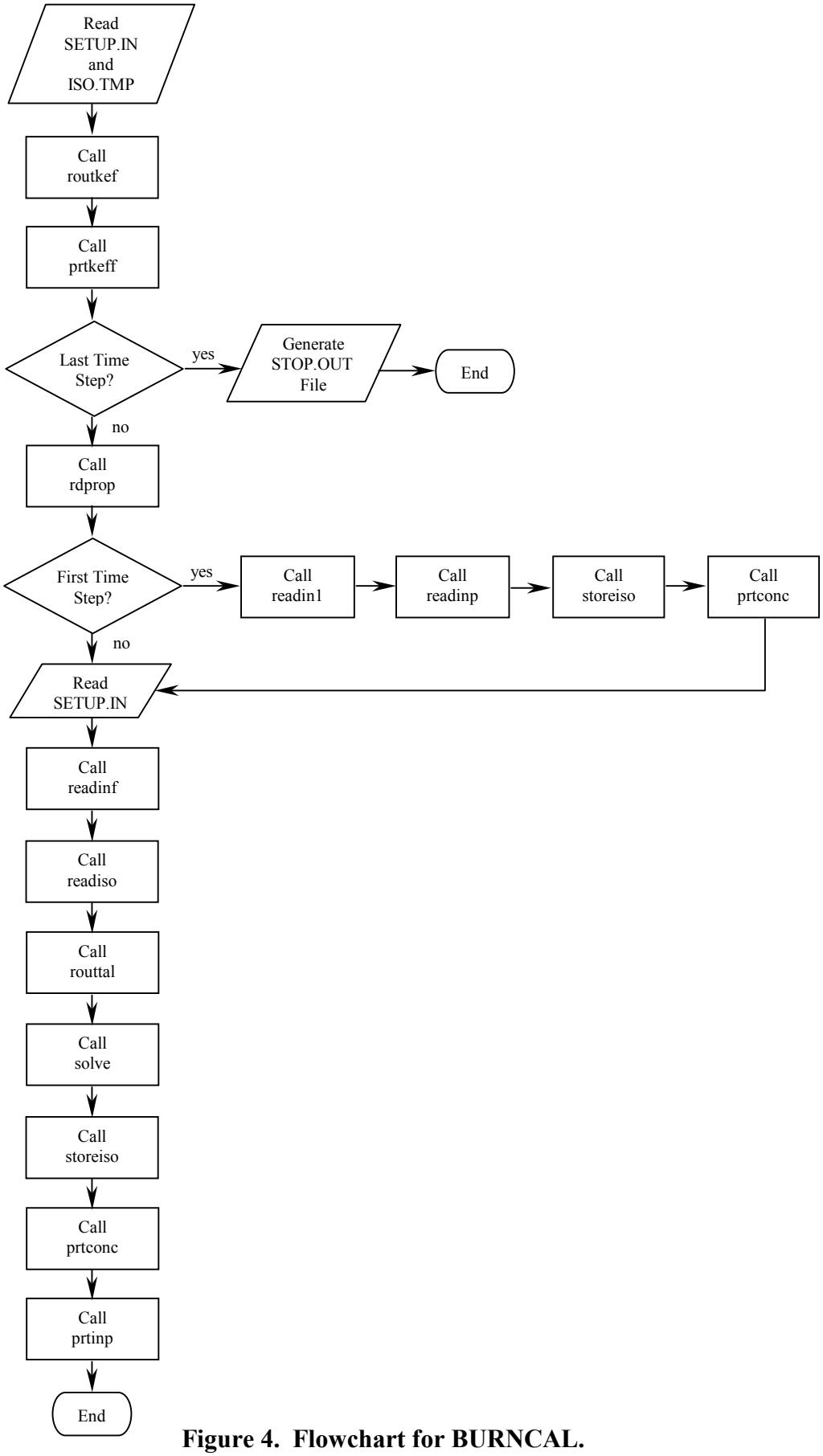

Figure 4. Flowchart for BURNCAL.

If this is the first iteration in the burnup analysis, the subroutines "readin1" and "readinp" are called. The subroutine "readin1" opens the MCNP input file and identifies the cell cards and their associated material numbers prior to a cell card identified as 200. The cells prior to cell card 200 represent the components for all of the fuel element assemblies and core regions. The "vol" card is then found in the input file and the total volume for each cell is read. The volumes are manually placed into the MCNP input file so that the correct total volume for the core or fueled region is used in association with the power density identified in SETUP.IN. The cell cards that are in the range of 100 to 199 are identified as the ones on which to perform the analysis. BURNCAL currently allows up to 30 different cells or regions to be analyzed. The program stores the total number of cells to be analyzed, the total core volume, and each individual cell number to be analyzed, along with its material number and volume, into the file CELL.TMP. The subroutine then looks for the word "keyword" and "mcard tallies" in the input file. The subsequent material cards are read and the material numbers and isotope Z,A values are stored for later use as tally identifiers. The tally identifier information is saved in the file TAL.TMP.

The subroutine "readinp" opens the MCNP input file and, using the material identifiers previously found for the cells to be analyzed, identifies the material cards associated with each cell. The isotopes and densities for each material are read and compared to the isotopes identified in LIBDATA.IN. The density for each isotope in each region is summed, converted to number density, and stored in an array.

The subroutines "storeiso" and "prtconc" are then called. These subroutines write, for each cell region identifier, the isotope and its concentration to ISO.TMP and CONC.OUT. These two files differ in that ISO.TMP only records the data for the current iteration. The values are used as the starting inventory for the subsequent iteration. CONC.OUT is a historical record of the inventory for each iteration and is used by RELOAD to access the inventory at a given time.

If this is the first iteration of the analysis, access now returns to the main code. If this is not the first iteration, the above subroutines are skipped. SETUP.IN is read to find the power density, incremental time step, and iteration time step for the current iteration. The incremental time step (in hours) is used in the subroutine "solve" as the Δt value for Equation 10. The global iteration time step (in days) is used to determine the number of passes through "solve" before exiting the program. The power density (in W/cm^3) is used in "solve" along with the tally information to determine the number of fissions per source neutron.

The subroutine "readinf" reads CELL.TMP and stores the data. The subroutine "readiso" reads ISO.TMP and stores the data.

The subroutine "routtal" opens the reduced MCNP output file and TAL.TMP. Using the cell identifier information and the material identifiers for the tallies, the fission and absorption tallies are found and stored in arrays. The total fission tally for each cell analyzed is also found and stored.

The subroutine "solve" is then called iteratively until the time increment equals the global time step for the next iteration. The subroutine uses all of the information derived (nuclides, inventories, power density, time step, fission fractions, half-lives, and MCNP tally results) to solve Equations 10, 11, and 12 for each region at each incremental time step.

The subroutines "storeiso" and "prtconc" then store the new nuclide inventory results for each region. The subroutine "prtinp" is called and places the new nuclide inventory into the proper locations of the MCNP input file.

The code then exits, allowing the script program to repeat the process by running MCNP for the new MCNP input file. The script program terminates the analysis when the STOP.OUT file is found.

3.2 MCNP Input Description

The MCNP input is identified as "atst.i" in Appendix E and in the script/batch file in Appendix A. Although the MCNP input file is required to be structured and formatted in such a way as to allow BURNCAL to properly resolve the information, the intent in developing the code was to minimize the requirements as much as possible. The user will find that the basic format for typical k_{eff}-type calculations exists with the ability to include lattices and universes. The sample problem included as Appendix E includes a header with details on the structure and format that should be rigorously followed.

The main structure and formatting requirements include proper identification of cell cards to be analyzed, the use of cell card 200 as an end identifier, correct column formatting for cells and material cards, isotope concentrations in density format, addition of keywords to identify the start of the materials for tally identifiers, the addition of a "vol" card, and the proper use of the tally multiplier cards. The isotopes and cross sections used in the analysis are completely user adjustable. The amount of detail in the geometry and the number of neutrons and histories used in the analysis are user specified.

3.3 SETUP.IN Description

The SETUP.IN file is a user-supplied file that allows the user to specify the number of burnup iterations to be performed, the time increment for solving the burnup equations, the burnup duration and the power density. A sample SETUP.IN file is included in Appendix G. The value on the first line represents the number of iterations to be performed. The following lines include the numbering up to the total number of iterations, the duration at the specified power density in days, the time increment for solving the equations in hours, and the power density in W/cm^3.

For the sample file, a one-hour time increment was chosen for all of the iteration time steps. However, this value can be changed to a smaller value depending on the accuracy required by the user for solving the burnup equations. Since running the BURNCAL code generally takes on the order of a minute or less for a 1 GHz processor, and MCNP generally takes tens of minutes

(depending on the complexity of the geometry and the accuracy desired for k_{eff}), the value of the time increment is not computer-time critical and can be set to a relatively small value.

The burnup time used at each iteration is an arbitrary value chosen by the user. However, it is recommended that a relatively small value be used for the initial calculations for thermal systems, since the Xe-135 reactivity effect can be relatively large. For the sample problem, the time increments used were 1, 7, 16, 36, and 60 days, building up to 180-day increments.

Decay with no power generation can be included at any point in the analysis by simply setting the power density to zero and setting the burnup duration time to the desired decay time. Including decay can be useful when the it is desired that the fuel be removed from a core and reloaded at a later date, or if the fission product or actinide inventory is required for a given decay time period after irradiation.

3.4 LIBDATA Description

The LIBDATA.IN file is a user-supplied file that allows the user to specify the isotopes and isotopic chains to be analyzed and tracked. A sample LIBDATA.IN file is included in Appendix D. This file can have more isotopes than are actually used in the MCNP input file. Isotopes can also be added to the library as needed.

The isotopes included in LIBDATA.IN are divided into two groups – stable isotopes that are used as bulk materials in which no depletion occurs, and all other isotopes in which removal and production are tracked. Both types of isotopes can be used in the material cards in the MCNP input file. The only difference is that the stable/bulk type isotopes are not depleted in the BURNCAL analysis. That is, the transmutation of these isotopes is calculated, but no change is made to the isotope density. Allowing for stable/bulk isotopes in the library permits the user to include specific isotopes or weighted, naturally-occurring isotopes (e.g., carbon or zirconium) in the analysis for which depletion is not a concern. Every isotope that is used in the MCNP input file, either as an isotope in one of the cells being analyzed or in one of the tally materials, must be identified in the LIBDATA.IN file. If an isotope is found by BURNCAL that is not included in LIBDATA.IN, BURNCAL is terminated with an error message.

The format for LIBDATA.IN is as follows. The integer value in the first line represents the number of stable/bulk type isotopes in the library. Following the first line are the stable/bulk isotopes. The data for each isotope is included on a single line. Following the stable/bulk isotope data is another integer value that represents the number of other isotopes in the library. For each isotope, the data included are the atomic number (Z), mass number (A), atomic weight as defined in the MCNP "XSDIR" file, the recommended individual fission yield for thermal U-235 fission (England and Rider, 1994) in percent, and the half-life of the isotope in hours. A comment may be added in the space following the half-life on the data card.

For the stable isotopes, a mass number equal to zero indicates that the naturally occurring isotope cross section is used in the MCNP input file. A value of –9999.0 for the half-life indicates that it is a stable isotope.

The code package is currently not designed to handle isotopes where both a metastable and ground state exist simultaneously. For most cases involving fission product chains, one state is found to dominate, or the longer half-life value may be used to ensure the k_{eff} calculation is conservative.

The code package is currently not designed to handle fission yields for multiple fissile or fissionable isotopes or neutron energies. Only thermal U-235 fission is included in the fission-yield data. Hence, although fission and depletion of all fissile and fissionable isotopes are calculated and traced in the code, the fission products that are generated are assumed to come from thermal fission of U-235. However, the user may have multiple LIBDATA.IN libraries that include fission-product yield data for fast U-235 fission, fast or thermal U-233 fission, fast or thermal Pu-239 fission, etc. The proper library can then be used for the main fissile component of the system.

3.5 RELOAD Code Description

RELOAD is a separate Fortran code that allows for the isotope concentrations of an MCNP input file to be adjusted using the CONC.OUT file generated from a prior burnup analysis. The code uses some of the read and write subprograms of BURNCAL, combined with a series of user-defined parameters, to determine the new inventory mixture. The primary usefulness of RELOAD is in shuffling reactor fuel from one region to another or for adding fresh fuel to a region of the core. The user identifies the time, for a prior burnup analysis, at which the inventory from one cell is to be moved into another. The code also allows the user to specify the fractional quantity of the initial inventory to remain in the region, resulting in a homogeneous mixture of different quantities of burned and/or fresh fuel.

3.6 Possible Future Work and Code Enhancements

Although there may exist a number of potential code enhancements and improvements, the most important are in those areas that would allow the code to become even more robust as an analysis tool. One important overall enhancement would be to expand the number of fission products and actinides included in the analysis. However, this enhancement is more an MCNP cross-section library issue, as opposed to those affecting the BURNCAL code package.

Another enhancement would be to include fission yields for mixtures of fissile and fissionable isotopes as well as the effects due to fission from fast and thermal neutrons. Since the code already determines the number of fissions for every fissile or fissionable isotope for every cell region, combining the isotope-specific yield data with this information would allow a more correct fission product inventory to be generated. Including the neutron energy dependence of the fission yield would be more difficult. This could be accomplished by tallying the fast to thermal fission ratio for each fissile or fissionable isotope during the MCNP run, the result of which would be used in generating the fission yield.

To date, no in-depth comparative or benchmark analyses have been performed using the BURNCAL code package. Hence, future work would include comparative studies to other

burnup-type codes using benchmark problems. Future work would also include analyses on light-water power reactor cores or assemblies. Code verification has been performed by analytic comparison of selected isotopes using sample case studies. These results are presented below.

4. SAMPLE CASE RESULTS

The sample case presented in the appendices represents a simple reactor system that demonstrates the capabilities of the BURNCAL code package. The sample case represents a good starting point for understanding the code and input/output features. The sample case problem is a conceptual design for a high-temperature power reactor using particle fuel in a carbide matrix, a molten salt coolant, and a graphite moderator. The sample case was part of a larger study to determine the burnup reactivity effects and conversion possibilities for a low-enriched, high-temperature, graphite-moderated reactor system. The overall reactor concept is shown in Figure 5. The details of the design are not important for this discussion.

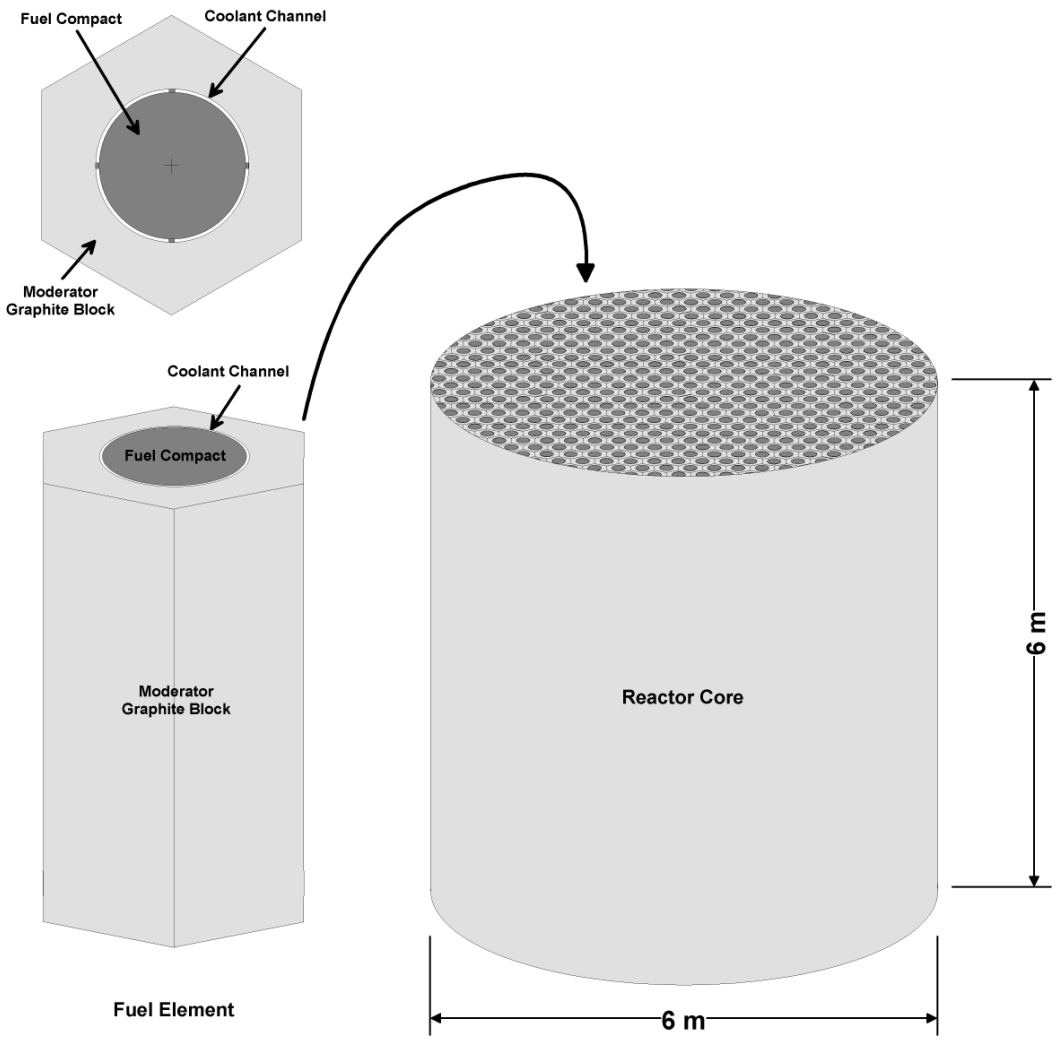

Figure 5. Sample Case Study Problem for a High Temperature Reactor.

The desired goal for the sample problem study was to calculate k_{eff} as a function of time for the reactor operating at an average power density of 5 W/cm^3. The reactor core was broken into three radial regions identified as outer, middle, and inner, each with equal areas. The MCNP input file for the initial case study problem is presented in Appendix E. This case study started with all fresh fuel and no burnable poisons or other reactor poisons. The fuel loading used 10% enriched uranium with no Th-232 additive. The case study was run using the LIBDATA.IN file presented in Appendix D, and SETUP.IN presented in Appendix G. A sample reduced MCNP output file is presented in Appendix F for one of the runs during the analysis.

The sample case study problem used 31 fission-product isotopes and 18 actinides in the MCNP input file. These isotopes and their respective chains were tracked using the data from the LIBDATA.IN file. The fission products used in the analysis are shown in Table 1. The actinides used are shown in Table 2. These isotopes were chosen as a result of cross sections that were available from the MCNP4B cross-section library, and from prior experience using the SAS2H (Hermann, et al., 2000) code from the SCALE Modular Code System (Parks, 1987).

Table 1. Fission Products Used in the Sample Case Study Problem.

1 - Xe-135	11 - Nd-145	21 - Sm-147
2 - Sm-149	12 - Eu-153	22 - Ag-109
3 - Nd-143	13 - Eu-155	23 - Cs-135
4 - Rh-103	14 - Eu-154	24 - I-129
5 - Pm-147	15 - Mo-95	25 - Zr-93
6 - Sm-151	16 - Sm-150	26 - Pd-105
7 - Xe-131	17 - Cs-134	27 - Pd-108
8 - Cs-133	18 - Pr-141	28 - Gd-157
9 - Tc-99	19 - Ru-101	29 - Pm-148
10 - Sm-152	20 - Kr-83	30 - Gd-156
		31 - Rh-105

Table 2. Actinide Isotopes Used in the Sample Case Study Problem.

1 - U-235	11 - Pu-240
2 - U-238	12 - Pu-241
3 - Th-232	13 - Am-241
4 - Pa-233	14 - Am-242m
5 - U-233	15 - Pu-242
6 - U-234	16 - Am-243
7 - U-236	17 - Cm-244
8 - Np-237	18 - Cm-245
9 - Pu-238	
10 - Pu-239	

The analysis used 50,000 neutron histories per MCNP run resulting in a standard deviation of approximately 0.004 in k_{eff}. Since this particular reactor system was large, used a graphite moderator, and was dilute in fuel concentration, the neutron fission generation times were somewhat large – on the order of about 0.7 ms. This resulted in MCNP run times on the order of 30 minutes per iteration using a 400 MHz processor. A 2 GHz processor resulted in a run time per iteration of less than 8 minutes.

The results of the analysis are shown in Figures 6 and 7. Figure 7 is a continuation in time of Figure 6. The curve identified as the "initial loading," starting at time equal to zero, represents the first series of iterations using the MCNP input file in Appendix E as the starting file. Smaller time steps between iterations were used for the first few iterations of the series to ensure that the Xe-135 reactivity effects were incorporated into the analysis. After a burnup time period of about 1260 days (~3.5 years), k_{eff} drops to a value below one. The analysis continued for a total of a 6-year burnup period, even though k_{eff} is below one for much of the time.

After the initial analysis, the RELOAD code was run using the MCNP input file and the CONC.OUT file. At the time of 1260 days, the fuel in the middle ring of the reactor core was moved to the inner ring, the fuel in the outer ring was moved to the middle ring, and fresh fuel was placed in the outer ring. The analysis was then repeated using this new MCNP input file as the starting condition. For this case, the reactor ran for about 540 days (1.5 years) before k_{eff} dropped below a value of one. RELOAD was again run with the shuffling and fresh fuel addition at 540 days. The same study was repeated twice more resulting in the other two curves.

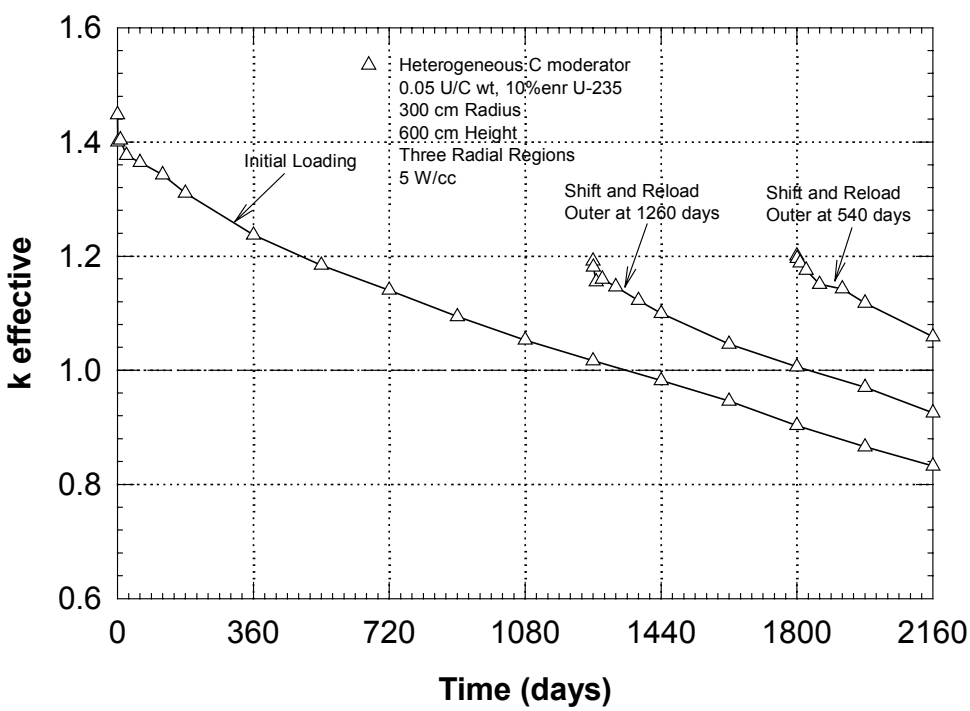

Figure 6. k_{eff} Versus Burnup Time for the Sample Case Study.

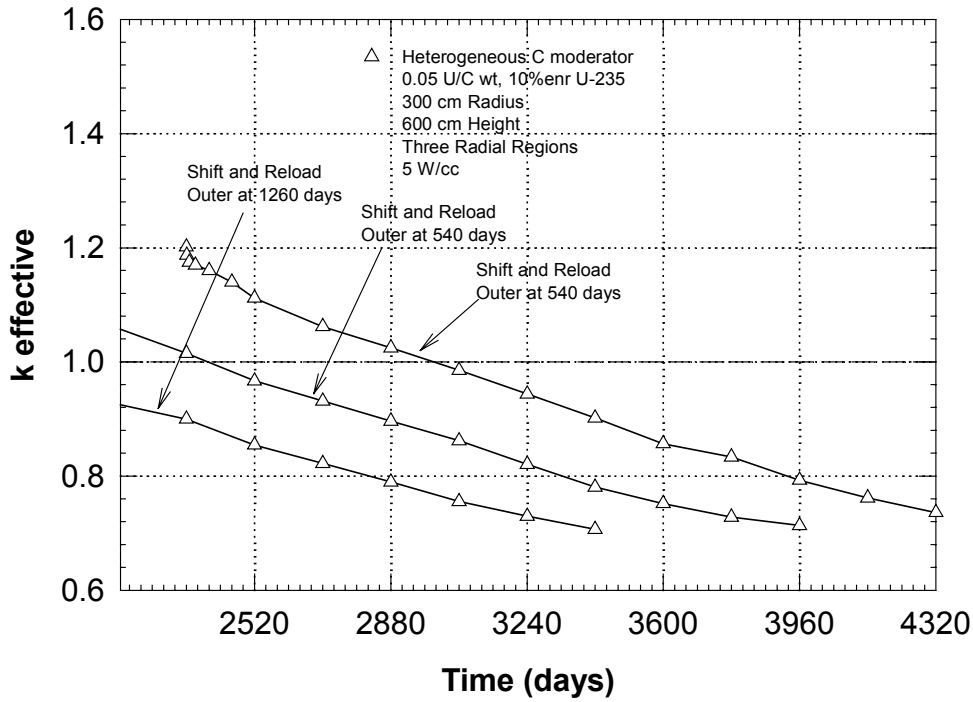

Figure 7. k_{eff} Versus Burnup Time for the Sample Case Study (Cont.).

Although not presented here, other valuable information can be attained from the analyses results, including actinide and fissile inventory as a function of time. Other types of analyses can also be performed including poisoning the fuel or moderator with a burnable poison such as boron, gadolinium, or erbium; adding Th-232 to generate the fissile isotope U-233; or removing fission products during reloading to simulate reprocessing.

5. CODE VERIFICATION

Verification calculations were performed for the BURNCAL code using the sample case study problem and comparing the results to analytical solutions for U-235 depletion and for the 135 and 149 isotope chains presented in Section 2.4. The sample problem was run using one-hour time increments. Power densities of both 5 and 50 W/cm^3 were analyzed. The isotopes for which analytical solutions were derived included I-135, Xe-135, Nd-149, Pm-149, Sm-149, and U-235. The three fueled regions for the sample problem were each analyzed. In order to calculate results for the 149 chain that could be compared to the analytical solution, the feed-in from Pm-148 to Pm-149 was removed from the analysis by setting the half-life in the LIBDATA.IN file for the Pm-148 to a small value. The iteration times used were those presented in SETUP.IN of Appendix G (1 day, 7 days, 16 days, 36 days, etc.).

Comparison of the analytical solutions to the BURNCAL values were made using a spreadsheet. The tally results for fission and radiative capture for Xe-135, Sm-149, and U-235 for the MCNP run at each iteration time were used in the spreadsheet along with the power density to determine the analytical results. The BURNCAL results were retrieved from the CONC.OUT file. The difference in the results were found to be less than 1% for the three regions for each of the six isotopes, at each iteration time. The only exception was for Xe-135 at a 50 W/cm^3 power density when the k_{eff} value dropped below 0.9. For this particular case, two of the regions overestimated the Xe-135 concentration by about a factor of two due to the rapid change in the concentration. Decreasing the calculation time step from one hour to 0.1 hours resulted in differences of less than 1%.

The results of the verification study show that the solution technique used in BURNCAL is accurate and robust. Situations may occur where the calculation time step may be required to be less than one hour in duration in order to generate the correct inventory. These situations are possible when the power density is relatively large and the value for k_{eff} drops well below one. These conditions are not likely to be encountered for real problems where k_{eff} is required to be greater than one during the burnup period.

6. SUMMARY AND CONCLUSIONS

The BURNCAL code package is a unique reactor burnup simulation code that uses neutron absorption and fission tally information provided by the Monte Carlo neutronics code MCNP directly in the calculative model. This allows for a more direct solution technique to be employed without the use of the spatial and energy-dependent neutron flux results. The code has been used successfully to study burnup reactivity effects and actinide inventories in advanced-type nuclear power reactor systems and small space-type reactor systems.

The code can be easily expanded to include other isotopes and chains, depending on the availability of MCNP cross sections for the isotopes of interest. Currently, 31 fission products and 18 actinides are used in the analysis. A sample problem is included in the appendices that will allow the user to become familiar with the code features and allow for more complicated systems to be studied.

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APPENDIX A – DOS Batch File BURNM.BAT

```
rem 1
rem  delete any old output or interface files
rem
    if not exist atst1.i goto next1
    copy atst1.i atst.i
    del iso.tmp
    del cell.tmp
    del tal.tmp
    del keff.out
    del conc.out
    del stop.out
rem
:rept1
    del atst.out
    del runtp*
    del outp*
    del mctl*
    del mctal*
    del srctp*
    del wssa*
    del inp*
rem
g:\MCNP4B\DOS\MCNP7 inp=atst.i mctal=mctal01 outp=atst.out
    del runtp*
    del srctp*
    del atstr.out
MCNPDATA
BURNCAL
    if not exist stop.out goto rept1
    echo normal stop of burn
    del atst.out
    copy keff.out keff1.out
    copy conc.out conc1.out
    copy atst.i atstl1.i
    del atst.i
    copy atstr.out atstr1.out
:next1
rem
rem 2
rem  delete any old output or interface files
rem
    if not exist atst2.i goto next2
    copy atst2.i atst.i
    del iso.tmp
    del cell.tmp
    del tal.tmp
    del keff.out
    del conc.out
    del stop.out
rem
:rept2
    del atst.out
    del runtp*
    del outp*
    del mctl*
    del mctal*
    del srctp*
    del wssa*
    del inp*
rem
g:\MCNP4B\DOS\MCNP7 inp=atst.i mctal=mctal01 outp=atst.out
    del runtp*
    del srctp*
    del atstr.out
MCNPDATA
BURNCAL
    if not exist stop.out goto rept2
    echo normal stop of burn
    del atst.out
    copy keff.out keff2.out
    copy conc.out conc2.out
    copy atst.i atstl2.i
    del atst.i
    copy atstr.out atstr2.out
:next2
rem
```

APPENDIX B – Fortran Code MCNPDATA.FOR

```
C
C
C
C  TAKES DATA FROM ATST.OUT FILES FROM MCNP AND
C  TRANFORMS TO A REDUCED FILE
C
C
      PROGRAM MCNPDATA
      IMPLICIT REAL*8(A-H,O-Z)
      CHARACTER*120 C$
      CHARACTER*10 A$,B$
      CHARACTER*4 D$
C      WRITE(*,10)' '
C      WRITE(*,10)'FILE TO STRIP? '
C      READ(*,10)A$
      A$='ATST.OUT'
C      WRITE(*,10)' '
C      WRITE(*,10)'NEW FILE NAME? '
C      READ(*,10)B$
      B$='ATSTR.OUT'
      OPEN(2,FILE=A$,STATUS='OLD')
      OPEN(3,FILE=B$,STATUS='UNKNOWN')
      WRITE(3,10)' '
C
C
C  FIND THE CYCLE NUMBER
C
  12  READ(2,10)C$
  10  FORMAT(A120)
      DO 20 I=1,90
       IF(C$(I:I+5).EQ.'kcode')THEN
        WRITE(*,10)C$
         DO 21 J=1,90
          IF(C$(J:J).NE.' '.AND.ICO.EQ.0)THEN
           ICO=1
           ICOUNT=ICOUNT+1
            IF(ICOUNT.EQ.6)THEN
            WRITE(*,10)C$(J:J+3)
            D$=C$(J:J+3)
            WRITE(*,10)D$
            GO TO 13
            ENDIF
          ENDIF
          IF(C$(J:J).EQ.' ')ICO=0
  21     CONTINUE
       ENDIF
  20  CONTINUE
      GO TO 12
C
  13  READ(2,10)C$
      DO 220 K=1,90
      IF(C$(K:K+4).EQ.'cycle')GO TO 221
  220 CONTINUE
      GO TO 13
C
  221   DO 22 I=1,90
        IF(C$(I:I).EQ.' ')THEN
        IF(C$(I+1:I+5).EQ.D$) THEN
        WRITE(3,10)C$
        WRITE(*,10)C$
        DO 23 J=1,44
        READ(2,10)C$
        WRITE(3,10)C$
  23    WRITE(*,10)C$
        GO TO 24
        ENDIF
        ENDIF
  22    CONTINUE
        GO TO 13
C
  24  CONTINUE
      WRITE(3,10)' '
C
  25  READ(2,10)C$
      DO 26 I=1,90
      IF(C$(I:I+16).EQ.' keff estimator ')THEN
      WRITE(3,10)C$
      WRITE(*,10)C$
      DO 27 J=1,41
      READ(2,10)C$
      WRITE(3,10)C$
```

```
      WRITE(*,10)c$
   27 CONTINUE
      GO TO 28
      ENDIF
   26 CONTINUE
      GO TO 25
C
   28 CONTINUE
      WRITE(3,10)' '
C
   29 READ(2,10)C$
      DO 30 I=1,90
      IF(C$(I:I+4).EQ.'tally')THEN
      WRITE(3,10)C$
      WRITE(*,10)C$
      GO TO 31
      ENDIF
   30 CONTINUE
      GO TO 29
C
   31 CONTINUE
   32 READ(2,10)C$
      DO 33 I=1,90
      IF(C$(I:I+5).EQ.'status')THEN
      GO TO 34
      ENDIF
   33 CONTINUE
      WRITE(3,10)C$
      WRITE(*,10)C$
      GO TO 32
C
   34 CONTINUE
      WRITE(3,10)' '
C
   35 READ(2,10)C$
      DO 36 I=1,90
      IF(C$(I:I+5).EQ.'******')THEN
      GO TO 37
      ENDIF
   36 CONTINUE
      GO TO 35
   37 CONTINUE
      WRITE(3,10)C$
      WRITE(*,10)C$
      DO 38 I=1,11
      READ(2,10)C$
      WRITE(3,10)C$
      WRITE(*,10)C$
   38 CONTINUE
C
      CLOSE(3)
      CLOSE(2)
      STOP
      END
C
```

APPENDIX C – Fortran Code BURNCAL.FOR

```
c
c
c
c
c  Performs burnup calculation using MCNP data
c
c
      program burncal
      implicit real*4(A-H,O-Z)
c
      dimension ids(100),ida(100),icell(30),imat(30),vol(30),
     1 zaic(2,30,100,250),zari(100,250),
     2 zalam(100,250), zasigg(30,100,250),zasigf(30,100,250),
     3 zawt(100,250)
c
      open(21,file='setup.in',status='old')
      open(11,file='iso.tmp',status='unknown')
c
      read(21,*)ihist
c
      read(11,*,err=200,end=200)ihcurnt,time
  200 continue
c  read keff from output
      call routkef(akeff,akerr)
      call prtkeff(time,akeff,akerr)
c
      if(ihist.eq.0.or.ihcurnt.eq.ihist.or.akeff.lt.0.61)then
      write(*,*)' '
      write(*,*)'** stop - last period = ', ihist, time,' ** '
      open(16,file='stop.out',status='new')
      write(16,*)ihcurnt,ihist,time
      write(16,*)'stop'
      close(16)
      stop
      endif
c
      call rdprop(ids,ida,zari,zalam,zawt)
c
c if 1st pass, write initial concentraions
      if(ihcurnt.eq.0)then
c read cell ids and corresponding material ids to calculate
c  and tally mcards
      call readin1(idnum,icell,imat,vol,ids,ida)
      call readinp(idnum,icell,imat,ids,ida,zawt,zaic)
      call storeiso(id,idnum,imat,ids,ida,ihcurnt,time,zawt,zaic)
      call prtconc(id,idnum,imat,ids,ida,ihcurnt,time,zawt,zaic)
      endif
c
c read time and power info for next period
      if(ihcurnt.lt.ihist)then
       do 201 i=1,ihist
c delt=days,ddelt=hrs, pden=W/cc (core average - all regions)
       read(21,*)ih,delt,ddelt,pden
       if(ih.eq.ihcurnt+1)go to 202
  201  continue
  202  ihcurnt=ihcurnt+1
       time=time+delt
      else
       stop
      endif
c
c get all data and perform calculation to determine
c  istotope concentrations for next time period
c
      call readinf(idnum,icell,imat,vol,voltot)
      call readiso(idnum,imat,zaic)
      call routtal(idnum,icell,imat,zasigg,zasigf)
c
c calculate new concentrations
      numpas=(delt*24.0)/ddelt
      write(*,*)numpas
      id=1
      do 251 i=1,numpas
      call solve(id,ddelt,pden,idnum,vol,voltot,
     1  ids,ida,zaic,zari,zalam,zasigg,zasigf)
      if(id.eq.1)then
      id=2
      go to 251
      endif
      if(id.eq.2)then
      id=1
      go to 251
```

```
      endif
  251 continue
c
c
c    copy all info into conc.out
c    copy all info (ihcurnt,time,conc) into iso.tmp
      call storeiso(id,idnum,imat,ids,ida,ihcurnt,time,zawt,zaic)
      call prtconc(id,idnum,imat,ids,ida,ihcurnt,time,zawt,zaic)
c
      call prtinp(id,idnum,imat,ids,ida,zawt,zaic)
c
      close(21)
      close(11)
c
      stop
      end
c
cccccccccccccccccccccccccccccccccccccccccccc
cccccccccccccccccccccccccccccccccccccccccccc
c
c
c  subroutine routkef
c   reads atstr.out (unit=24) to find keff and kerr in MCNP output
c
      subroutine routkef(akeff,akerr)
      implicit real*4(a-h,o-z)
      character*120 c$
c
      open(24,file='atstr.out',status='old')
c
 1001 read(24,1002)c$
 1002 format(a120)
      do 1003 i=1,90
       if(c$(i:i+14).eq.'keff estimator')then
c
 1005   read(24,1002)c$
         do 1006 j=1,90
          if(c$(j:j+11).eq.'col/abs/trk')then
           backspace (24)
           read(24,1100)akeff,akerr
 1100 format(t28,f7.5,t44,f7.5)
           go to 1007
          endif
 1006    continue
        go to 1005
 1007   continue
c
        go to 1008
       endif
 1003 continue
      go to 1001
 1008 continue
c
      close(24)
      return
      end
c
cccccccccccccccccccccccccccccccccccccccccccc
c
c
c  subroutine prtkeff
c  writes time, keff, and kerr to keff.out (unit=12)
      subroutine prtkeff(time,akeff,akerr)
      implicit real*4(a-h,o-z)
c
      open(12,file='keff.out',status='unknown',access='append')
c
      write(*,*)time,akeff,akerr
      write(12,*)time,akeff,akerr
c
      close(12)
      return
      end
c
cccccccccccccccccccccccccccccccccccccccccccc
c
c
c  subroutine rdprop
c  reads libdata.in (unit=22) z, a, awt, ri, T-half(hrs)
      subroutine rdprop(ids,ida,zari,zalam,zawt)
      implicit real*4(a-h,o-z)
c
      dimension ids(100),ida(100),zari(100,250),
     1  zalam(100,250),zawt(100,250)
c
      open(22,file='libdata.in',status='old')
c
```

```
c  read constant isotopes data
      read(22,*)ids(1)
      if(ids(1).eq.0)go to 2001
      do 2000 i=1,ids(1)
c  read in z, a, wt(neu=1.0),ri (%) for U-235 thermal
      read(22,*)iz,ia,awt,ri,thalf
      ids(i+1)=iz*1000 + ia
      if(ia.eq.0)ia=250
      zari(iz,ia)=ri/100.0
       if(thalf.ne.-9999.0)then
c calculate lambda in 1/s (thalf is in hours)
       zalam(iz,ia)=alog(2.0)/(3600.0*thalf)
       else
c a stable isotope will have thalf = inf, i.e. lambda=0.0
       zalam(iz,ia)=0.0
       endif
      zawt(iz,ia)=awt/5.97134e23
 2000 continue
 2001 continue
c
c  read active isotopes data
      read(22,*)ida(1)
      if(ida(1).eq.0)go to 2003
      do 2002 i=1,ida(1)
c  read in z, a, wt(neu=1.0),ri (%)
      read(22,*)iz,ia,awt,ri,thalf
      ida(i+1)=iz*1000 + ia
      if(ia.eq.0)ia=250
      zari(iz,ia)=ri/100.0
       if(thalf.ne.-9999.0)then
c calculate lambda in 1/s (thalf is in hours)
       zalam(iz,ia)=alog(2.0)/(3600.0*thalf)
       else
c a stable isotope will have thalf = inf, i.e. lambda=0.0
       zalam(iz,ia)=0.0
       endif
      zawt(iz,ia)=awt/5.97134e23
 2002 continue
 2003 continue
c
c
      close(22)
      return
      end
c
cccccccccccccccccccccccccccccccccccccccccccc
c
c
c  subroutine readin1
c  reads atst.i (unit=23) to determine cell and material numbers
c   to perform calculations on and identify tally material cards
c  write cell and material ids in cell.tmp (unit=15)
c  write material id for tally specification to tal.tmp (unit=13)
      subroutine readin1(idnum,icell,imat,vol,ids,ida)
      implicit real*4(a-h,o-z)
c
      dimension icell(30),imat(30),vol(30),ids(100),ida(100),
     1  icell0(100),imat0(100),vol0(100),itallyt(2,300)
      character*120 b$
      character*3 a$,t$
c
      open(23,file='atst.i',status='old')
c
c  find cell cards and mat number to be used in calculation cell 100-199
c
      idnum=0
      idnumt=0
      voltot=0.0
c determine if cell card is in range 100-199
c for cell card 200 stop
c
 3001 read(23,3002,end=3099)a$
 3002 format(a3)
       if(a$(1:1).eq.'c'.or.a$(1:1).eq.' ')go to 3001
       if(a$(1:1).eq.'1'.or.a$(1:1).eq.'2')then
       backspace(23)
       read(23,*)jcell,jmat
c for cell card 200 stop
       if(jcell.eq.200)go to 3099
       idnumt=idnumt+1
       icell0(idnumt)=jcell
       imat0(idnumt)=jmat
       endif
      go to 3001
c
 3099 continue
c
```

```
c next find volumes
 3011 read(23,3012,end=3199)a$
 3012 format(a3)
      if(a$(1:3).ne.'vol')go to 3011
c
      backspace(23)
      read(23,*)a$,(vol0(i),i=1,idnumt)
c
      do 3013 i=1,idnumt
      voltot=voltot+vol0(i)
       if(icell0(i).ge.100.and.icell0(i).lt.200)then
       idnum=idnum+1
       icell(idnum)=icell0(i)
       imat(idnum)=imat0(i)
       vol(idnum)=vol0(i)
       endif
 3013 continue
c
 3199 continue
c
c  write cell and material card id to cell.tmp (unit=15)
      open(15,file='cell.tmp',status='new')
c
      write(15,*)idnum,voltot
      do 3005 j=1,idnum
      write(15,*)icell(j),imat(j),vol(j)
 3005 continue
c
c  find m cards for tallies
c  put mcard number at location in array itallyt
c
      rewind(23)
      italls=0
c  find correct m card
 7001 read(23,7002,end=7099)b$
 7002 format(a120)
      if(b$(8:14).eq.'keyword'.and.
     1 b$(17:30).eq.'mcard tallies')then
c
 7004 continue
      read(23,7003,end=7099)t$
 7003 format(a3)
c
      if(t$(1:1).eq.'m'.and.t$(2:2).ne.'t'.and.t$(2:2).ne.'o')then
      backspace(23)
      read(23,7050)j,izacomp
 7050 format(t2,i3,t8,i5)
c
      do 7101 k=1,ids(1)
      if(ids(k+1).eq.izacomp)then
      italls=italls+1
      itallyt(1,italls)=izacomp
      itallyt(2,italls)=j
      go to 7110
      endif
 7101 continue
      do 7102 k=1,ida(1)
      if(ida(k+1).eq.izacomp)then
      italls=italls+1
      itallyt(1,italls)=izacomp
      itallyt(2,italls)=j
      go to 7110
      endif
 7102 continue
c  if can't find in ids or ida - stop
      write(*,*)' '
      write(*,*)'error tally specification'
      write(*,*)'no z,a identifed in libdata.in'
      write(*,*)'za = ',izacomp
      stop
 7110 continue
c
       endif
       go to 7004
      endif
       go to 7001
c
 7099 continue
c
c
c  write material id for tally specification to tal.tmp (unit=13)
      open(13,file='tal.tmp',status='new')
c
      do 7500 i=1,italls
      iz=itallyt(1,i)/1000
      ia=itallyt(1,i) - 1000*(itallyt(1,i)/1000)
      write(13,*)i,iz,ia,itallyt(2,i)
```

```
 7500 continue
c
      close(15)
      close(13)
      close(23)
      return
      end
c
cccccccccccccccccccccccccccccccccccccccccccccc
c
c
c  subroutine readinp
c  reads atst.i (unit=23) initial isotope concentrations to zaic
      subroutine readinp(idnum,icell,imat,ids,ida,zawt,zaic)
      implicit real*4(a-h,o-z)
c
      dimension icell(30),imat(30),ids(100),ida(100),
     1  density(3),zawt(100,250),zaic(2,30,100,250)
      character*120 b$
      character*3 a$,t$
c
      open(23,file='atst.i',status='old')
c
c  find isotopes and gm/cc for each material card associated with
c   cell to be calculated 100-199
      do 3099 i=1,idnum
       rewind(23)
c  find correct m card
 3001 read(23,3002,end=3099)a$
 3002 format(a3)
       if(a$(1:1).eq.'m'.and.a$(2:2).ne.'t'.and.a$(2:2).ne.'o')then
       backspace(23)
       read(23,3050)j
 3050  format(t2,i3)
       if(j.eq.imat(i))go to 3003
       endif
      go to 3001
c
 3003 continue
      backspace(23)
 3004 read(23,3002)t$
c if first column in card is not a "c" or " ", then
c  go to 3099 and look for next m card
      if(t$.eq.a$.or.t$.eq.'c  '.or.t$.eq.'  ')then
      if(t$.eq.'c  ')go to 3004
      backspace(23)
      read(23,3051)izacomp,zadens
 3051 format(t8,i5,t18,g14.7)
      else
      go to 3099
      endif
c
c
      do 3101 k=1,ids(1)
      if(ids(k+1).eq.izacomp)go to 3110
 3101 continue
      do 3102 k=1,ida(1)
      if(ida(k+1).eq.izacomp)go to 3110
 3102 continue
c  if can't find in ids or ida - stop
      write(*,*)' '
      write(*,*)'error - no z a identifed in libdata.in'
      write(*,*)'za = ',izacomp
      stop
 3110 continue
      iz=izacomp/1000
      ia=izacomp - 1000*(izacomp/1000)
      if(ia.eq.0)ia=250
      zaic(1,i,iz,ia)=-zadens/zawt(iz,ia)
c      write(*,*)izacomp,iz,ia,i,zaic(1,i,iz,ia)
      density(i)=density(i)+zadens
c
      go to 3004
c
 3099 continue
c
      do 3020 i=1,idnum
      write(*,*)icell(i),imat(i),density(i)
 3020 continue
c
      close(23)
      return
      end
c
cccccccccccccccccccccccccccccccccccccccccccccc
c
c
```

```
c      subroutine storeiso
c  writes time, ihcurnt, and zaic to iso.tmp (unit=11)
      subroutine storeiso(id,idnum,imat,ids,ida,ihcurnt,time,zawt,zaic)
      implicit real*4(a-h,o-z)
c
      dimension imat(30),ids(100),ida(100),zawt(100,250),
     1 zaic(2,30,100,250)
c
      rewind(11)
      write(11,*)ihcurnt,time
c
      if(id.eq.0)then
      ib=1
      go to 4111
      endif
      ib=id
 4111 continue
c
      do 4050 j=1,idnum
c  write constant isotopes
      if(ids(1).eq.0)go to 4001
      do 4000 i=1,ids(1)
      iz=ids(i+1)/1000
      ia=ids(i+1) - 1000*(ids(i+1)/1000)
      iaa=ia
      if(ia.eq.0)iaa=250
      write(11,*)imat(j),iz,ia,zaic(ib,j,iz,iaa),
     1   -zaic(ib,j,iz,iaa)*zawt(iz,iaa)
 4000 continue
 4001 continue
c
c  write active isotopes
      if(ida(1).eq.0)go to 4021
      do 4020 i=1,ida(1)
      iz=ida(i+1)/1000
      ia=ida(i+1) - 1000*(ida(i+1)/1000)
      iaa=ia
      if(ia.eq.0)iaa=250
      write(11,*)imat(j),iz,ia,zaic(ib,j,iz,iaa),
     1   -zaic(ib,j,iz,iaa)*zawt(iz,iaa)
 4020 continue
 4021 continue
c
 4050 continue
c
      return
      end
c
cccccccccccccccccccccccccccccccccccccccccccccccccccccccc
c
c
c      subroutine prtconc
c  writes time, and zaic to conc.out for each time period (unit=13)
      subroutine prtconc(id,idnum,imat,ids,ida,ihcurnt,time,zawt,zaic)
      implicit real*4(a-h,o-z)
c
      dimension imat(30),ids(100),ida(100),zawt(100,250),
     1 zaic(2,30,100,250)
c
      open(14,file='conc.out',status='unknown',access='append')
c
      write(14,*)' '
      write(14,*)ihcurnt, time
c
      if(id.eq.0)then
      ib=1
      go to 5111
      endif
      ib=id
 5111 continue
c
      do 5050 j=1,idnum
c  write constant isotopes
      if(ids(1).eq.0)go to 5001
      do 5000 i=1,ids(1)
      iz=ids(i+1)/1000
      ia=ids(i+1) - 1000*(ids(i+1)/1000)
      iaa=ia
      if(ia.eq.0)iaa=250
      write(14,*)imat(j),iz,ia,zaic(ib,j,iz,iaa),
     1   -zaic(ib,j,iz,iaa)*zawt(iz,iaa)
 5000 continue
 5001 continue
c
c  write active isotopes
      if(ida(1).eq.0)go to 5021
      do 5020 i=1,ida(1)
```

```
      iz=ida(i+1)/1000
      ia=ida(i+1) - 1000*(ida(i+1)/1000)
      iaa=ia
      if(ia.eq.0)iaa=250
      write(14,*)imat(j),iz,ia,zaic(ib,j,iz,iaa),
     1   -zaic(ib,j,iz,iaa)*zawt(iz,iaa)
 5020 continue
 5021 continue
c
 5050 continue
c
      close(14)
      return
      end
c
c
cccccccccccccccccccccccccccccccccccccccccc
c
c
c      subroutine readinf
c  reads cell.tmp (unit=15) into idnum,icell,imat,vol,voltot
      subroutine readinf(idnum,icell,imat,vol,voltot)
      implicit real*4(a-h,o-z)
c
      dimension icell(30),imat(30),vol(30)
      open(15,file='cell.tmp',status='old')
      read(15,*)idnum,voltot
c
c  read cell,mat,vol
      do 6051 j=1,idnum
      read(15,*)icell(j),imat(j),vol(j)
 6051 continue
c
      close(15)
      return
      end
c
c
cccccccccccccccccccccccccccccccccccccccccc
c
c
c      subroutine readiso
c  reads iso.tmp (unit=11) into zaic
      subroutine readiso(idnum,imat,zaic)
      implicit real*4(a-h,o-z)
c
      dimension imat(30),zaic(2,30,100,250)
c
      kuse=1
      rewind(11)
      read(11,*)ix,ti
c
c  read moderator,clad,fuel isotopes
 6050 read(11,*,end=6051)j,iz,ia,conc,dens
      if(j.eq.imat(kuse))go to 6053
      do 6052 k=1,idnum
      if(j.eq.imat(k))then
      kuse=k
      go to 6053
      endif
 6052 continue
 6053 continue
c
      if(ia.eq.0)ia=250
      zaic(1,kuse,iz,ia)=conc
c      write(*,*)j,iz,ia,zaic(1,kuse,iz,ia),conc
      go to 6050
 6051 continue
c
      return
      end
c
cccccccccccccccccccccccccccccccccccccccccccccccccccccccccc
c
c
c      subroutine routtal
c  reads atstr.out (unit=24) and tal.tmp (unit=13) to find the material id
c   for each tally specification and n,gamma, fission, or other tally value
      subroutine routtal(idnum,icell,imat,zasigg,zasigf)
      implicit real*4(a-h,o-z)
c
      dimension icell(30),imat(30),zasigg(30,100,250),zasigf(30,100,250)
      character*120 c$
      character*10 a$
      character*4 b$
c
      open(24,file='atstr.out',status='old')
```

```
      open(13,file='tal.tmp',status='old')
c
 8001 read(13,*,end=8000)j,iz,ia,idmt
      if(idmt.eq.0)go to 8001
      italred=0
      rewind(24)
 8002 read(24,8003,end=8005)c$
 8003 format(a120)
      if(c$(2:16).eq.'multiplier bin:')then
      backspace(24)
      read(24,*)a$,b$,x,idcomp,itype
c
       if(idcomp.eq.idmt)then
       italred=1
       backspace(24)
       backspace(24)
c
       do 8021 jk=1,idnum
       read(24,*)b$,iceltst
       read(24,*)a$,b$,x,idcomp,itype
       read(24,*)tally,talerr
       jj=0
        if(itype.eq.-6)then
         do 8022 k=1,idnum
         if(icell(k).eq.iceltst)jj=k
 8022    continue
        zasigf(jj,iz,ia)=tally
        endif
        if(itype.eq.-2.or.itype.eq.102.or.itype.eq.107)then
         do 8023 k=1,idnum
         if(icell(k).eq.iceltst)jj=k
 8023    continue
        zasigg(jj,iz,ia)=tally
        endif
 8021  continue
c
       endif
c
      endif
      go to 8002
c
 8005 if(italred.eq.0)then
      write(*,*)'error multiplier read'
      write(*,*)'no multiplier card in atstr.out for',iz,ia,idmt
      stop
      endif
      go to 8001
 8000 continue
c
c  read total fission tally
      do 8031 jk=1,idnum
      italred=0
      rewind(24)
 8012 read(24,8013,end=8015)c$
 8013 format(a120)
      if(c$(2:16).eq.'multiplier bin:')then
      backspace(24)
      read(24,*)a$,b$,x,idcomp,itype
       if(idcomp.eq.imat(jk))then
       italred=1
       read(24,*)tally,talerr
        if(itype.eq.-6)then
        zasigf(jk,1,1)=tally
c        write(*,*)'fission',idcomp,itype,tally
        endif
       endif
      endif
      go to 8012
 8015 if(italred.eq.0)then
      write(*,*)'error multiplier read'
      write(*,*)'no total fission multiplier card in atstr.out'
      stop
      endif
c
 8031 continue
c
      close(13)
      close(24)
c
      return
      end
c
cccccccccccccccccccccccccccccccccccccccccccccccccccccccc
c
c
c  subroutine prtinp
c  writes new isotope densities to atst.i (unit=23)
```

```
      subroutine prtinp(id,idnum,imat,ids,ida,zawt,zaic)
c
      implicit real*4(a-h,o-z)
c
      dimension ids(100),ida(100),imat(30),
     1  density(3),zawt(100,250),zaic(2,30,100,250)
      character*80 c$
      character*64 d$
      character*17 e$
      character*3 a$,t$
c
      open(23,file='atst.i',status='unknown')
      open(31,status='scratch',
     1 access='direct',form='formatted',recl=80)
c
c  copy atst.i to direct scratch file for processing
c   direct file can be changed per record without
c   loosing all subsequent data
c
      ic=0
 9200 read(23,9201,end=9202)c$
 9201 format(a80)
      ic=ic+1
      write(31,9201,rec=ic)c$
      go to 9200
 9202 continue
c
c  find isotopes and gm/cc for mod, clad, fuel
      do 9099 i=1,idnum
      rewind(31)
      ic=0
c  find correct m card
 9001 ic=ic+1
      read(31,9002,rec=ic,err=9099)a$
 9002 format(a3)
       if(a$(1:1).eq.'m'.and.a$(2:2).ne.'t'.and.a$(2:2).ne.'o')then
       read(31,9050,rec=ic)j
 9050  format(t2,i3)
       if(j.eq.imat(i))go to 9003
       endif
      go to 9001
c
 9003 continue
 9004 read(31,9002,rec=ic)t$
      if(t$.eq.a$.or.t$.eq.'c  '.or.t$.eq.'  ')then
       if(t$.eq.'c  ')then
       ic=ic+1
       go to 9004
       endif
      read(31,9051,rec=ic)izacomp,zadens
 9051 format(t8,i5,t18,g14.7)
      read(31,9053,rec=ic)e$,x
 9053 format(t1,a17,t18,g14.7)
c      write(*,*)izacomp,zadens
      else
      go to 9099
      endif
c
c
      do 9101 k=1,ids(1)
      if(ids(k+1).eq.izacomp)go to 9110
 9101 continue
      do 9102 k=1,ida(1)
      if(ida(k+1).eq.izacomp)go to 9110
 9102 continue
c  if can't find in ids or ida - stop
      write(*,*)' '
      write(*,*)'error - no z a identifed in libdata.in'
      write(*,*)'za = ',izacomp
      stop
 9110 continue
      iz=izacomp/1000
      ia=izacomp - 1000*(izacomp/1000)
      if(ia.eq.0)ia=250
      zadens=-zaic(id,i,iz,ia)*zawt(iz,ia)
      if(zadens.eq.0.0.or.abs(zadens).lt.1.0e-24)zadens=-1.0e-24
      write(31,9052,rec=ic)e$,zadens
 9052 format(t1,a17,t18,g14.7)
c      write(*,*)t$,izacomp,zadens
      density(i)=density(i)+zadens
c
      ic=ic+1
      go to 9004
c
 9099 continue
c
c  change density value in cell card
```

```
c
      do 9070 i=1,idnum
      rewind(31)
      ic=0
c  find correct cell card with material id
 9061 ic=ic+1
      read(31,9002,rec=ic)a$
       if(a$(1:3).eq.'c  '.or.a$(1:3).eq.'   ')then
       go to 9061
       endif
      read(31,9064,rec=ic)j,k
 9064 format(t1,i3,t6,i2)
c      write(*,*)j
      if(j.eq.200)go to 9070
      if(k.eq.0)go to 9061
      read(31,9063,rec=ic)j,k,oldens,d$
 9063 format(t1,i3,t6,i2,t9,f7.3,a64)
       if(k.eq.imat(i))then
       write(31,9063,rec=ic)j,k,density(i),d$
c       write(*,*)i,ic
c       write(*,*)j,k,oldens,density(i),d$
       endif
      go to 9061
c
 9070 continue
c
c  copy scratch file to atst.i
c
      rewind(31)
      rewind(23)
      ic=0
 9600 ic=ic+1
      read(31,9601,rec=ic,err=9602)c$
 9601 format(a80)
      write(23,9601)c$
c      write(*,*)c$
      go to 9600
 9602 continue
c
      close(23)
      close(31)
c
      return
      end
c
cccccccccccccccccccccccccccccccccccccccccccccccccccccccccc
c
c  subroutine solve
c  writes new isotope densities to atst.i (unit=23)
      subroutine
     1 solve(id,ddelt,pden,idnum,vol,voltot,
     1  ids,ida,zaic,zari,zalam,zasigg,zasigf)
c
      implicit real*4(a-h,o-z)
c
      dimension ids(100),ida(100),vol(30),
     1 zaic(2,30,100,250),zari(100,250),
     2 zalam(100,250), zasigg(30,100,250),zasigf(30,100,250)
c
c   keep track of fissions
      addfiss=0.0
      totfis=0.0
c
      if(id.eq.1)ic=2
      if(id.eq.2)ic=1
      do 7 i=1,idnum
c  totfis units fission/source-n
      totfis=totfis+zasigf(i,1,1)*vol(i)
      do 8 j=1,100
      do 9 k=1,250
      zaic(ic,i,j,k)=zaic(id,i,j,k)
    9 continue
    8 continue
    7 continue
c
c  change time increment to sec
      timeinc=ddelt*3600.0
c
c  power density is in watts/cc core average
c   change to fissions/s (core tot) (pwra)
c             fissions-cm**2/s-b (core tot) (pwrb)
c   190 MeV/fission -no neutrino
      pwra=pden * voltot * 3.285e10
      pwrb=pwra * 1.0e-24
c
c  fismult units are f-cm**2/s-b / f/source-n
c  i.e. number of source neutrons per second * cm**2/b
```

```
      fismult=pwrb/totfis
c
c      write(*,*)id,ic,ddelt,pden
c      write(*,*)pwra,pwr,totfis
      do 10 i=1,idnum
      do 11 ii=1,2
      if(ii.eq.1)irep=ids(1)
      if(ii.eq.2)irep=ida(1)
      if(irep.eq.0)go to 11
c
      do 12 j=1,irep
      if(ii.eq.1)then
      iz=ids(j+1)/1000
      ia=ids(j+1) - 1000*(ids(j+1)/1000)
      if(ia.eq.0)ia=250
      endif
      if(ii.eq.2)then
      iz=ida(j+1)/1000
      ia=ida(j+1) - 1000*(ida(j+1)/1000)
      if(ia.eq.0)ia=250
      endif
c
c      write(*,*)id,i,iz,ia
      sigf=zasigf(i,iz,ia)*fismult
cc      if(sigf.ne.0.0)write(*,*)i,iz,ia,fismult,zasigf(i,iz,ia)
      siga=zasigg(i,iz,ia)*fismult
      alamb=zalam(iz,ia)
c
c  simple solution - use production and removal rates times time
c   solve using previous time step concentrations
c   put removals into proper next time step for next istope
c
c
      if(ii.eq.2)then
c for isotope iz,ia
c   production (iz,ia) from fission/cc in region
      proizia=pwra*zasigf(i,1,1)/totfis
      proizia=proizia*zari(iz,ia)*timeinc
      zaic(ic,i,iz,ia)=zaic(ic,i,iz,ia) + proizia
c   removal (iz,ia) from fission
      fisizia=sigf*zaic(id,i,iz,ia)*timeinc
cc      if(sigf.ne.0.0)write(*,*)i,iz,ia,fisizia,zaic(id,i,iz,ia)
      if(fisizia.ge.zaic(id,i,iz,ia))
     1  fisizia=zaic(id,i,iz,ia)
      zaic(ic,i,iz,ia)=zaic(ic,i,iz,ia) - fisizia
      addfiss=addfiss+fisizia*vol(i)
cc      if(sigf.ne.0.0)write(*,*)addfiss
      endif
c
c   removal (iz,ia) from decay
      decizia=alamb*zaic(id,i,iz,ia)*timeinc
      if(decizia.ge.zaic(id,i,iz,ia))
     1  decizia=zaic(id,i,iz,ia)
      if(ii.eq.2)zaic(ic,i,iz,ia)=zaic(ic,i,iz,ia) - decizia
c   removal (iz,ia) from absorption (n,gamma or other)
      absizia=siga*zaic(id,i,iz,ia)*timeinc
      if(absizia.ge.zaic(ic,i,iz,ia))
     1  absizia=zaic(ic,i,iz,ia)
      if(ii.eq.2)zaic(ic,i,iz,ia)=zaic(ic,i,iz,ia) - absizia
c   production of iz,ia+1, i.e. (n,gamma) of iz,ia
c   He-6 production from n,alpha of be-9 (4009)
c   Li-7 production from n,alpha of b-10 (5010)
      if(iz.eq.4.and.ia.eq.9.or.iz.eq.5.and.iz.eq.10)then
      if(iz.eq.4.and.ia.eq.9)zaic(ic,i,2,6)=zaic(ic,i,2,6)+absizia
      if(iz.eq.5.and.ia.eq.10)zaic(ic,i,3,7)=zaic(ic,i,3,7)+absizia
      else
      if(ia.ne.250)zaic(ic,i,iz,ia+1)=zaic(ic,i,iz,ia+1)+absizia
      endif
c   production of iz+1,ia, i.e. beta decay of iz,ia
c   pu-240 (94240) is alpha decay from cm-244 (96244)
      if(iz.eq.96.and.ia.eq.244)then
      zaic(ic,i,94,240)=zaic(ic,i,94,240) + decizia
      else
      zaic(ic,i,iz+1,ia)=zaic(ic,i,iz+1,ia) + decizia
      endif
c
c
 12   continue
 11   continue
 10   continue
c
      write(*,*)pwra*timeinc,addfiss
c
      return
      end
c
cccccccccccccccccccccccccccccccccccccccccccccccccccccccc
```

APPENDIX D – Library File LIBDATA.IN

```
8
 4   9     8.934763    0.0      -9999.0       n,alpha to He-6 (2006)
 6  12    11.896914    0.0      -9999.0
 7  14    13.882780    0.0      -9999.0
 7  15    14.871250    0.0      -9999.0
 8  16    15.857510    0.0      -9999.0
 9  19    18.835197    0.0      -9999.0
14   0    27.844241    0.0      -9999.0
40   0    90.439990    0.0      -9999.0

98
 2   6     5.963       0.0          0.00022
 3   6     5.963450    0.0      -9999.0
 3   7     6.955732    0.0      -9999.0
 5  10     9.926922    0.0      -9999.0
 5  11    10.914730    0.0      -9999.0
90 232   230.044724    0.0      -9999.0
90 233   231.039630    0.0          0.3717
91 233   231.038304    0.0        648.0
92 233   231.037695    0.0      -9999.0
91 234   232.03        0.0          6.69
92 234   232.030412    0.0      -9999.0
92 235   233.024773    0.0      -9999.0
92 236   234.017806    0.0      -9999.0
92 237   235.012352    0.0        162.0
93 237   235.011799    0.0      -9999.0
93 238   236.005958    0.0         50.808
94 238   236.004583    0.0      -9999.0
92 238   236.005803    0.0      -9999.0
92 239   236.999       0.0          0.3917
93 239   236.999341    0.0         56.52
94 239   236.998573    0.0      -9999.0
94 240   237.991619    0.0      -9999.0
94 241   238.986041    0.0     126230.4
95 241   238.986019    0.0      -9999.0
95 242   239.980173    0.0      -9999.0
94 242   239.979326    0.0      -9999.0
94 243   240.973968    0.0          4.956
95 243   240.973348    0.0      -9999.0
95 244   241.967640    0.0          0.4333
96 244   241.966119    0.0     158664.6       alpha decay to pu-240 (94240)
96 245   242.960245    0.0      -9999.0

35  83   82.2          0.5362       2.4
36  83   82.201858     0.0      -9999.0

39  93   92.111437     6.3466      10.2
40  93   92.108361     0.0      -9999.0

40  95   94.092727     6.5032    1536.48
41  95   94.091532     0.0        839.28
42  95   94.090546     0.0      -9999.0

40  97   96.078429     5.9840      16.8
41  97   96.075601     0.0          1.23
42  97   96.073544     0.0      -9999.0

42  99   98.058039     6.1092      65.942
43  99   98.056595     0.0      -9999.0

44 101   100.038748    5.1729   -9999.0

44 103   102.022302    3.0312     942.48
45 103   102.021490    0.0      -9999.0

44 105   104.0         0.9642       4.44
45 105   104.004489    0.0         35.4
46 105   104.003885    0.0      -9999.0

46 108   106.976942    0.0541   -9999.0

46 109   107.969       0.0312      13.5
47 109   107.969204    0.0      -9999.0

51 129   127.798       0.5434       4.40
52 129   127.798       0.0          1.16
53 129   127.798000    0.0077   -9999.0

52 131   129.780       0.4123      32.4
53 131   129.780       2.4786     192.96
54 131   129.780532    0.0      -9999.0
```

```
53 133   131.763       6.6970      20.8
54 133   131.763       0.0        125.832
55 133   131.763705    0.0      -9999.0

55 134   132.756370    0.0      18101.79

53 135   133.751093    6.2823       6.57
54 135   133.748208    0.2567       9.1
55 135   133.746975    0.0      -9999.0

56 139   137.713       6.4139       1.396
57 139   137.713076    0.0      -9999.0

57 141   139.697       5.8474       3.90
58 141   139.697       0.0        780.0
59 141   139.697185    0.0      -9999.0

58 143   141.682       5.9562      33.12
59 143   141.682       0.0        325.68
60 143   141.682152    0.0      -9999.0

58 144   142.673       5.5000    6830.4
59 144   142.673       0.0          0.288
60 144   142.673832    0.0      -9999.0

59 145   143.667       3.9336       5.98
60 145   143.667706    0.0      -9999.0

60 147   145.654023    2.2469     263.52
61 147   145.653069    0.0      22996.72

61 148   146.646787    0.0        228.101       half-life is inv avg of 5.37d and 41.3d(m)

62 147   145.652830    0.0      -9999.0

60 149   147.639       1.0817       1.72
61 149   147.639055    0.0         53.088
62 149   147.637915    0.0      -9999.0

62 150   148.629416    0.0      -9999.0

61 151   149.623       0.4188      28.392
62 151   149.623460    0.0     788940.0

62 152   150.614670    0.2669   -9999.0

62 153   151.607       0.1583      46.296
63 153   151.607568    0.0      -9999.0

63 154   152.600719    0.0      75299.94

63 155   153.592023    0.0321   41287.86

62 156   154.582       0.0148       9.4
63 156   154.582       0.0        364.8
64 156   154.582676    0.0      -9999.0

63 157   155.575       0.0062      15.13
64 157   155.575907    0.0      -9999.0
```

APPENDIX E – MCNP Input – atst.i

```
      MCNP4b burnup
c     Use with burncal.for, burn.bat
c     libdata.in, setup.in
c     all cells ids must start in column one
c     all material card ids must start in column one
c     all isotope identifiers (i.e. 92235.60c) in material cards
c      must start in column eight.  Four digit identifiers
c      (i.e. 6012.60c) must start in column nine
c     all isotope densities must start with a negative sign in
c      column 18.  If a density starts out as zero use the value
c      -1.0000e-24.  The densities in the material cards must add
c      up to the value in the cell card.  The values in the material
c      cards and cell card will change for cells in the range
c      100 to 199.  All other cell and material card will not change.
c     cell ids must be 100 to 199 to perform depletion,
c      burnup, production, decay, or activation - must start
c      in column one
c     cells not requiring calculation should be outside
c      of this range (e.g. clad or moderator regions that are part
c      of the fuel but do not require a depletion or activation calc)
c      but must be identified before cell 200
c     material cards must be provided and individualized
c      for each cell card 100 to 199
c     tally f914+ is used to determine total fission and
c      fission in each region.  One tally must be identified
c      for each calculated region, whether or not there are fission
c      in that region
c     all tallies must include all cells identified as 100 to 199,
c      even if the tallies will not be used in a region
c     last cell card must have id 200 and start in column one
c     cell cards for regions outside of core must follow card 200
c     also fill and repeated structure cards must follow card 200
c     other regions outside of the core can also follow 200
c      this allows the volume for the core region to be found
c     volumes for core regions must be calculated manually
c      and place in vol card.  All volumes need to be included
c      for all regions - i.e. no jump cards. For volumes outside
c      of core region can use jump cards.
c     volumes need to be accurately calculated
c      and included for all of the core regions since
c      the power density is for the core average and
c      MCNP has trouble calculating total volumes for
c      repeated structures
c     all volumes added must equal the core volume - this is used
c      with the average power density to determine
c      the fuel power density
c     one keyword is used by the code to locate a particular
c      place in the input deck - the position prior to the
c      material cards for the tallies.  On the line prior to
c      these materail cards place the card
c      c      keyword- mcard tallies
c      with the c in column one.
c
c
c     K eff study - UO2/ThO2/C fuel, FLIBE mod
c     PBR particle fuel dimensions and materials
c     with interstitual C
c     FLIBE (LiFBeF2) salt ortho= 66m/o LiF, 34m/o BeF2
c
c     ----------
c     cell cards
c     ----------
c
c     fuel lattice - fuel is a cylinder, mod in hex
c     fuel pin 10%packing in C
101   2 -2.09349   -107       u=1  imp:n=1 $ U/Th/C/Zr
c     moderator
c     must input volume for a hexagon in vol A=sqrt(3)/2*P**2
11    1 -2.050      107 -108      u=1  imp:n=1 $ FLIBE
12    5 -1.800      108           u=1  imp:n=1 $ C
c     fuel lattice - fuel is a cylinder, mod in hex
c     fuel pin 10%packing in C - use same load as fuel
111   3 -2.09349   -107       u=2  imp:n=1 $ U/Th/C/Zr
c     moderator
c     must input volume for a hexagon in vol A=sqrt(3)/2*P**2
13    1 -2.050      107 -108      u=2  imp:n=1 $ FLIBE
14    5 -1.800      108           u=2  imp:n=1 $ C
c     fuel lattice - fuel is a cylinder, mod in hex
c     fuel pin 10%packing in C - use same load as fuel
121   4 -2.09349   -107       u=3  imp:n=1 $ U/Th/C/Zr
c     moderator
```

```
c     must input volume for a hexagon in vol A=sqrt(3)/2*P**2
15    1 -2.050      107 -108      u=3  imp:n=1 $ FLIBE
16    5 -1.800      108           u=3  imp:n=1 $ C
c     last core region cell - card 200 must follow
c     external void
200   0            310:211:-212    imp:n=0
31    0            -110 115 -111 112 fill=4 imp:n=1
32    1 -2.050     -101 102 -103 104 -105 106 lat=2 u=4
                   fill -80:80 -80:80 0:0
                   1 12959r 1 1 12959r  imp:n=1
33    0            -115 114 -111 112 fill=5  imp:n=1 $center
34    1 -2.050     -101 102 -103 104 -105 106 lat=2 u=5
                   fill -80:80 -80:80 0:0
                   2 25920r  imp:n=1
35    0            -114 -111 112 fill=6  imp:n=1 $center
36    1 -2.050     -101 102 -103 104 -105 106 lat=2 u=6
                   fill -80:80 -80:80 0:0
                   3 25920r  imp:n=1
37    6 -2.050     -210 -211 212 (110:111:-112) imp:n=1 $salt ref
38    5 -1.800     -310 -211 212 (210:211:-212) imp:n=1 $c ref
c

c     -------------
c     surface cards
c     -------------
c
c    pitch value based on %vol coolant
101    px  7.620
102    px -7.620
103    p   1 1.7320508076 0  15.24
104    p   1 1.7320508076 0 -15.24
105    p  -1 1.7320508076 0  15.24
106    p  -1 1.7320508076 0 -15.24
c
c   fuel element is solid cylinder in
c   a hex for coolant - 10%vol coolant
c   fuel radius
107    cz   4.3079
108    cz   4.4912
c   core
110    cz  300.0
c
111    pz  300.0
112    pz -300.0
c
114    cz  173.205
115    cz  244.949
c
210    cz  400.0
211    pz  400.0
212    pz -400.0
c
310    cz  500.0
c

c     ---------
c     materials
c     ---------
c
c   m1 Moderator
c    FLIBE (LiFBeF2) salt ortho= 66m/o LiF, 34m/o BeF2
m1      3007.60c 0.282
        4009.60c 0.145
        9019.60c 0.573
c     Li-6
c              3006.60c -1.0000e-24
c   m5 graphite (fuel and reflector region)
m5      6012.50c 1.0
c   m4 salt reflector
c    FLIBE (LiFBeF2) salt ortho= 66m/o LiF, 34m/o BeF2
m6      3007.60c 0.282
        4009.60c 0.145
        9019.60c 0.573
c
c   m2 Fuel for cell 101
c     10% enriched in 235, no Th
c     1200K Sab, Fuel at 900K (54c)
c
m2      6012.50c -1.7354
       14000.60c -0.0392
       40000.60c -1.0000e-24
        8016.60c -0.0196
       92235.54c -0.029929
       92238.54c -0.269359
       90232.60c -1.0000e-24
c
c    Th-232 family
```

```
c     Pa-233
       91233.50c -1.0000e-24
c     U-233
       92233.60c -1.0000e-24
c     U-234
       92234.60c -1.0000e-24
c
c    U-235 family
c     U-236
       92236.60c -1.0000e-24
c     Np-237
       93237.60c -1.0000e-24
c     Pu-238
       94238.60c -1.0000e-24
c
c    U-238 family
c     Pu-239
       94239.60c -1.0000e-24
c     Pu-240
       94240.60c -1.0000e-24
c     Pu-241
       94241.60c -1.0000e-24
c     Am-241
       95241.60c -1.0000e-24
c     Am-242m
       95242.50c -1.0000e-24
c     Pu-242
       94242.60c -1.0000e-24
c     Am-243
       95243.60c -1.0000e-24
c     Cm-244
       96244.60c -1.0000e-24
c     Cm-245
       96245.60c -1.0000e-24
c
c    fission products
c     Xe-135
       54135.54c -1.0000e-24
c     Sm-149
       62149.50c -1.0000e-24
c     Nd-143
       60143.50c -1.0000e-24
c     Rh-103
       45103.50c -1.0000e-24
c     Pm-147
       61147.50c -1.0000e-24
c     Sm-151
       62151.50c -1.0000e-24
c     Xe-131
       54131.50c -1.0000e-24
c     Cs-133
       55133.60c -1.0000e-24
c     Tc-99
       43099.60c -1.0000e-24
c     Sm-152
       62152.50c -1.0000e-24
c     Pm-148m
c      no mcnp xsec
c     Nd-145
       60145.50c -1.0000e-24
c     Eu-153
       63153.60c -1.0000e-24
c     Eu-155
       63155.50c -1.0000e-24
c     Eu-154
       63154.50c -1.0000e-24
c     Mo-95
       42095.50c -1.0000e-24
c     Sm-150
       62150.50c -1.0000e-24
c     Cs-134
       55134.60c -1.0000e-24
c     Pr-141
       59141.50c -1.0000e-24
c     Ru-101
       44101.50c -1.0000e-24
c     Kr-83
       36083.50c -1.0000e-24
c     La-139
c      no mcnp xsec
c     Sm-147
       62147.50c -1.0000e-24
c     Ag-109
       47109.60c -1.0000e-24
c     Cs-135
       55135.60c -1.0000e-24
c     I-129
```

```
       53129.60c -1.0000e-24
c     Zr-93
       40093.50c -1.0000e-24
c     Nd-144
c      no mcnp xsec
c     Pd-105
       46105.50c -1.0000e-24
c     Pd-108
       46108.50c -1.0000e-24
c     Mo-97
c      no mcnp xsec
c     Gd-157
       64157.60c -1.0000e-24
c     Pm-148
       61148.50c -1.0000e-24
c     Gd-156
       64156.60c -1.0000e-24
c     Rh-105
       45105.50c -1.0000e-24
c
c
c   m3 Fuel for cell 111
c     10% enriched in 235, no Th
c     1200K Sab, Fuel at 900K (54c)
c
m3      6012.50c -1.7354
       14000.60c -0.0392
       40000.60c -1.0000e-24
        8016.60c -0.0196
       92235.54c -0.029929
       92238.54c -0.269359
       90232.60c -1.0000e-24
c
c    Th-232 family
c     Pa-233
       91233.50c -1.0000e-24
c     U-233
       92233.60c -1.0000e-24
c     U-234
       92234.60c -1.0000e-24
c
c    U-235 family
c     U-236
       92236.60c -1.0000e-24
c     Np-237
       93237.60c -1.0000e-24
c     Pu-238
       94238.60c -1.0000e-24
c
c    U-238 family
c     Pu-239
       94239.60c -1.0000e-24
c     Pu-240
       94240.60c -1.0000e-24
c     Pu-241
       94241.60c -1.0000e-24
c     Am-241
       95241.60c -1.0000e-24
c     Am-242m
       95242.50c -1.0000e-24
c     Pu-242
       94242.60c -1.0000e-24
c     Am-243
       95243.60c -1.0000e-24
c     Cm-244
       96244.60c -1.0000e-24
c     Cm-245
       96245.60c -1.0000e-24
c
c    fission products
c     Xe-135
       54135.54c -1.0000e-24
c     Sm-149
       62149.50c -1.0000e-24
c     Nd-143
       60143.50c -1.0000e-24
c     Rh-103
       45103.50c -1.0000e-24
c     Pm-147
       61147.50c -1.0000e-24
c     Sm-151
       62151.50c -1.0000e-24
c     Xe-131
       54131.50c -1.0000e-24
c     Cs-133
       55133.60c -1.0000e-24
c     Tc-99
```

```
       43099.60c -1.0000e-24
c     Sm-152
       62152.50c -1.0000e-24
c     Pm-148m
c      no mcnp xsec
c     Nd-145
       60145.50c -1.0000e-24
c     Eu-153
       63153.60c -1.0000e-24
c     Eu-155
       63155.50c -1.0000e-24
c     Eu-154
       63154.50c -1.0000e-24
c     Mo-95
       42095.50c -1.0000e-24
c     Sm-150
       62150.50c -1.0000e-24
c     Cs-134
       55134.60c -1.0000e-24
c     Pr-141
       59141.50c -1.0000e-24
c     Ru-101
       44101.50c -1.0000e-24
c     Kr-83
       36083.50c -1.0000e-24
c     La-139
c      no mcnp xsec
c     Sm-147
       62147.50c -1.0000e-24
c     Ag-109
       47109.60c -1.0000e-24
c     Cs-135
       55135.60c -1.0000e-24
c     I-129
       53129.60c -1.0000e-24
c     Zr-93
       40093.50c -1.0000e-24
c     Nd-144
c      no mcnp xsec
c     Pd-105
       46105.50c -1.0000e-24
c     Pd-108
       46108.50c -1.0000e-24
c     Mo-97
c      no mcnp xsec
c     Gd-157
       64157.60c -1.0000e-24
c     Pm-148
       61148.50c -1.0000e-24
c     Gd-156
       64156.60c -1.0000e-24
c     Rh-105
       45105.50c -1.0000e-24
c
c
c
c   m4 Fuel for cell 121
c     10% enriched in 235, no Th
c     1200K Sab, Fuel at 900K (54c)
c
m4      6012.50c -1.7354
       14000.60c -0.0392
       40000.60c -1.0000e-24
        8016.60c -0.0196
       92235.54c -0.029929
       92238.54c -0.269359
       90232.60c -1.0000e-24
c
c    Th-232 family
c     Pa-233
       91233.50c -1.0000e-24
c     U-233
       92233.60c -1.0000e-24
c     U-234
       92234.60c -1.0000e-24
c
c    U-235 family
c     U-236
       92236.60c -1.0000e-24
c     Np-237
       93237.60c -1.0000e-24
c     Pu-238
       94238.60c -1.0000e-24
c
c    U-238 family
c     Pu-239
       94239.60c -1.0000e-24
```

```
c     Pu-240
       94240.60c -1.0000e-24
c     Pu-241
       94241.60c -1.0000e-24
c     Am-241
       95241.60c -1.0000e-24
c     Am-242m
       95242.50c -1.0000e-24
c     Pu-242
       94242.60c -1.0000e-24
c     Am-243
       95243.60c -1.0000e-24
c     Cm-244
       96244.60c -1.0000e-24
c     Cm-245
       96245.60c -1.0000e-24
c
c    fission products
c     Xe-135
       54135.54c -1.0000e-24
c     Sm-149
       62149.50c -1.0000e-24
c     Nd-143
       60143.50c -1.0000e-24
c     Rh-103
       45103.50c -1.0000e-24
c     Pm-147
       61147.50c -1.0000e-24
c     Sm-151
       62151.50c -1.0000e-24
c     Xe-131
       54131.50c -1.0000e-24
c     Cs-133
       55133.60c -1.0000e-24
c     Tc-99
       43099.60c -1.0000e-24
c     Sm-152
       62152.50c -1.0000e-24
c     Pm-148m
c      no mcnp xsec
c     Nd-145
       60145.50c -1.0000e-24
c     Eu-153
       63153.60c -1.0000e-24
c     Eu-155
       63155.50c -1.0000e-24
c     Eu-154
       63154.50c -1.0000e-24
c     Mo-95
       42095.50c -1.0000e-24
c     Sm-150
       62150.50c -1.0000e-24
c     Cs-134
       55134.60c -1.0000e-24
c     Pr-141
       59141.50c -1.0000e-24
c     Ru-101
       44101.50c -1.0000e-24
c     Kr-83
       36083.50c -1.0000e-24
c     La-139
c      no mcnp xsec
c     Sm-147
       62147.50c -1.0000e-24
c     Ag-109
       47109.60c -1.0000e-24
c     Cs-135
       55135.60c -1.0000e-24
c     I-129
       53129.60c -1.0000e-24
c     Zr-93
       40093.50c -1.0000e-24
c     Nd-144
c      no mcnp xsec
c     Pd-105
       46105.50c -1.0000e-24
c     Pd-108
       46108.50c -1.0000e-24
c     Mo-97
c      no mcnp xsec
c     Gd-157
       64157.60c -1.0000e-24
c     Pm-148
       61148.50c -1.0000e-24
c     Gd-156
       64156.60c -1.0000e-24
c     Rh-105
```

```
       45105.50c -1.0000e-24
c
c
c     Material Cards for Tallies
c      keyword- mcard tallies
c
c     Be-9
c     m5      4009.60c 1.0
c     Li-6
c     m6      3006.60c 1.0
c
c     U-235
m7     92235.54c 1.0
c     U-238
m8     92238.54c 1.0
c     Th-232
m9     90232.50c 1.0
c
c    Th-232 family
c     Pa-233
m11    91233.50c 1.0
c     U-233
m12    92233.60c 1.0
c     U-234
m13    92234.60c 1.0
c
c    U-235 family
c     U-236
m21    92236.60c 1.0
c     Np-237
m22    93237.60c 1.0
c     Pu-238
m23    94238.60c 1.0
c
c    U-238 family
c     Pu-239
m31    94239.60c 1.0
c     Pu-240
m32    94240.60c 1.0
c     Pu-241
m33    94241.60c 1.0
c     Am-241
m34    95241.60c 1.0
c     Am-242m
m35    95242.50c 1.0
c     Pu-242
m36    94242.60c 1.0
c     Am-243
m37    95243.60c 1.0
c     Cm-244
m38    96244.60c 1.0
c     Cm-245
m39    96245.60c 1.0
c
c    fission products
c     Xe-135
m51    54135.54c 1.0
c     Sm-149
m52    62149.50c 1.0
c     Nd-143
m53    60143.50c 1.0
c     Rh-103
m54    45103.50c 1.0
c     Pm-147
m55    61147.50c 1.0
c     Sm-151
m56    62151.50c 1.0
c     Xe-131
m57    54131.50c 1.0
c     Cs-133
m58    55133.60c 1.0
c     Tc-99
m59    43099.60c 1.0
c     Sm-152
m60    62152.50c 1.0
c     Pm-148m
c       no mcnp xsec
c     Nd-145
m61    60145.50c 1.0
c     Eu-153
m62    63153.60c 1.0
c     Eu-155
m63    63155.50c 1.0
c     Eu-154
m64    63154.50c 1.0
c     Mo-95
m65    42095.50c 1.0
```

```
c     Sm-150
m66    62150.50c 1.0
c     Cs-134
m67    55134.60c 1.0
c     Pr-141
m68    59141.50c 1.0
c     Ru-101
m69    44101.50c 1.0
c    Kr-83
m70    36083.50c 1.0
c     La-139
c       no mcnp xsec
c     Sm-147
m71    62147.50c 1.0
c     Ag-109
m72    47109.60c 1.0
c     Cs-135
m73    55135.60c 1.0
c     I-129
m74    53129.60c 1.0
c     Zr-93
m75    40093.50c 1.0
c     Nd-144
c       no mcnp xsec
c     Pd-105
m76    46105.50c 1.0
c     Pd-108
m77    46108.50c 1.0
c     Mo-97
c       no mcnp xsec
c     Gd-157
m78    64157.60c 1.0
c     Pm-148
m79    61148.50c 1.0
c     Gd-156
m80    64156.60c 1.0
c     Rh-105
m81    45105.50c 1.0
c
c
c     s(a,b) identifiers
c      06 is 1200K, 01 is 300K
c      07 is 1600K,
mt1     be.06t
mt2     grph.06t
mt3     grph.06t
mt4     grph.06t
mt6     be.06t
mt5     grph.06t
c
c     Tallies
c
fc914    tally fissions 101 m2
f914:n  101
fm914   -1 2 -6
c
fc924    tally fissions 111 m3
f924:n  111
fm924   -1 3 -6
c
fc934    tally fissions 111 m3
f934:n  121
fm934   -1 4 -6
c
c
c     fc34    Be-9 n,alpha
c     f34:n  12
c     fm34    1 5 107
c     fc44    Li-6 n,abs
c     f44:n  12
c     fm44    1 6 -2
c
fc54    U-235 n,gamma
f54:n  101 111 121
fm54    1 7 102
fc64    U-235 n,fission
f64:n 101 111 121
fm64   1 7 -6
fc74    U-238 n,gamma
f74:n  101 111 121
fm74    1 8 102
fc84    U-238 n,fission
f84:n 101 111 121
fm84   1 8 -6
fc94    Th-232 n,gamma
f94:n  101 111 121
fm94    1 9 102
```

```
fc104   Pa-233 n,gamma
f104:n 101 111 121
fm104   1 11 102
fc114   U-233 n,gamma
f114:n 101 111 121
fm114   1 12 102
fc124   U-233 n,fission
f124:n 101 111 121
fm124   1 12 -6
fc134   U-234 n,gamma
f134:n 101 111 121
fm134   1 13 102
fc144   U-236 n,gamma
f144:n 101 111 121
fm144   1 21 102
fc154   Np-237 n,gamma
f154:n 101 111 121
fm154   1 22 102
fc164   Pu-238 n,gamma
f164:n 101 111 121
fm164   1 23 102
fc174   Pu-238 n,fission
f174:n 101 111 121
fm174   1 23 -6
c
fc184   Pu-239 n,gamma
f184:n 101 111 121
fm184   1 31 102
fc194   Pu-239 n,fission
f194:n 101 111 121
fm194   1 31 -6
fc204   Pu-240 n,gamma
f204:n 101 111 121
fm204   1 32 102
fc214   Pu-241 n,gamma
f214:n 101 111 121
fm214   1 33 102
fc224   Pu-241 n,fission
f224:n 101 111 121
fm224   1 33 -6
fc234   Am-241 n,gamma
f234:n 101 111 121
fm234   1 34 102
fc244   Am-241 n,fission
f244:n 101 111 121
fm244   1 34 -6
fc254   Am-242m n,gamma
f254:n 101 111 121
fm254   1 35 102
fc264   Am-242m n,fission
f264:n 101 111 121
fm264   1 35 -6
fc274   Pu-242 n,gamma
f274:n 101 111 121
fm274   1 36 102
fc284   Pu-242 n,fission
f284:n 101 111 121
fm284   1 36 -6
fc294   Am-243 n,gamma
f294:n 101 111 121
fm294   1 37 102
fc304   Cm-244 n,gamma
f304:n 101 111 121
fm304   1 38 102
fc314   Cm-244 n,fission
f314:n 101 111 121
fm314   1 38 -6
fc324   Cm-245 n,gamma
f324:n 101 111 121
fm324   1 39 102
fc334   Cm-245 n,fission
f334:n 101 111 121
fm334   1 39 -6
c
fc514   Xe-135 n,gamma
f514:n 101 111 121
fm514   1 51 102
fc524   Sm-149 n,gamma
f524:n 101 111 121
fm524   1 52 102
fc534   Nd-143 n,gamma
f534:n 101 111 121
fm534   1 53 102
fc544   Rh-103 n,gamma
f544:n 101 111 121
fm544   1 54 102
fc564   Pm-147 n,gamma
```

```
f564:n 101 111 121
fm564   1 55 102
fc574   Sm-151 n,gamma
f574:n 101 111 121
fm574   1 56 102
fc584   Xe-131 n,gamma
f584:n 101 111 121
fm584   1 57 102
fc594   Cs-133 n,gamma
f594:n 101 111 121
fm594   1 58 102
fc604   Tc-99 n,gamma
f604:n 101 111 121
fm604   1 59 102
fc614   Sm-152 n,gamma
f614:n 101 111 121
fm614   1 60 102
fc624   Nd-145 n,gamma
f624:n 101 111 121
fm624   1 61 102
fc634   Eu-153 n,gamma
f634:n 101 111 121
fm634   1 62 102
fc644   Eu-155 n,gamma
f644:n 101 111 121
fm644   1 63 102
fc654   Eu-154 n,gamma
f654:n 101 111 121
fm654   1 64 102
fc664   Mo-95 n,gamma
f664:n 101 111 121
fm664   1 65 102
fc674   Sm-150 n,gamma
f674:n 101 111 121
fm674   1 66 102
fc684   Cs-134 n,gamma
f684:n 101 111 121
fm684   1 67 102
fc694   Pr-141 n,gamma
f694:n 101 111 121
fm694   1 68 102
fc704   Ru-101 n,gamma
f704:n 101 111 121
fm704   1 69 102
fc714   Kr-83 n,gamma
f714:n 101 111 121
fm714   1 70 102
fc724   Sm-147 n,gamma
f724:n 101 111 121
fm724   1 71 102
fc734   Ag-109 n,gamma
f734:n 101 111 121
fm734   1 72 102
fc744   Cs-135 n,gamma
f744:n 101 111 121
fm744   1 73 102
fc754   I-129 n,gamma
f754:n 101 111 121
fm754   1 74 102
fc764   Zr-93 n,gamma
f764:n 101 111 121
fm764   1 75 102
fc774   Pd-105 n,gamma
f774:n 101 111 121
fm774   1 76 102
fc784   Pd-108 n,gamma
f784:n 101 111 121
fm784   1 77 102
fc794   Gd-157 n,gamma
f794:n 101 111 121
fm794   1 78 102
fc804   Pm-148 n,gamma
f804:n 101 111 121
fm804   1 79 102
fc814   Gd-156 n,gamma
f814:n 101 111 121
fm814   1 80 102
fc824   Rh-105 n,gamma
f824:n 101 111 121
fm824   1 81 102
c
c
c     volumes need to be accurately calculated
c      and included for all of the core regions since
c      the power density is for the core average and
c      MCNP has trouble calculating total volumes for
c      repeated structures
```

```
c     all volumes added must equal the core volume
c
vol          1.6391e7 1.4245e6 3.8733e7
             1.6391e7 1.4245e6 3.8733e7
             1.6391e7 1.4245e6 3.8733e7  8j 0
mode   n
kcode  500 1.0 3 100
ksrc   0 0 0  220 0 0 0 220 0
print  10 60 100 110
prdmp  50 50 0 1
```

APPENDIX F – MCNP Output (Reduced) - atstr.out

```
 estimator     cycle   100   ave of    97 cycles      combination         simple average    combined average     corr
 k(collision)      1.484162       1.453593 0.0050     k(col/abs)          1.450423 0.0032     1.447705 0.0025   0.3736
 k(absorption)     1.422357       1.447254 0.0025     k(abs/tk ln)        1.449021 0.0032     1.447474 0.0025   0.3908
 k(trk length)     1.466247       1.450789 0.0050     k(tk ln/col)        1.452191 0.0050     1.452209 0.0051   0.9777
 rem life(col)   2.9137E+04     3.1581E+04 0.0112     k(col/abs/tk ln)    1.450545 0.0037     1.448087 0.0025
 rem life(abs)   3.0265E+04     3.1605E+04 0.0106     life(col/abs/tl)  3.1574E+04 0.0101   3.1498E+04 0.0092
 source points generated    510

 source distribution written to file srctp           cycle =   100
1problem summary (active cycles only)              source particle weight for summary table normalization =    48500.00

      run terminated when   100 kcode cycles were done.
+                                                                                                          08/08/02 06:30:28
            MCNP4b burnup                                                                          probid =   08/08/02 06:22:19
0
 neutron creation    tracks      weight        energy            neutron loss        tracks      weight        energy
                                 (per source particle)                                                (per source particle)

 source               48515    1.0000E+00    2.0470E+00          escape                 123    9.8147E-04    1.9547E-10
                                                                  energy cutoff            0    0.            0.
                                                                  time cutoff              0    0.            0.
 weight window            0    0.            0.                  weight window            0    0.            0.
 cell importance          0    0.            0.                  cell importance          0    0.            0.
 weight cutoff            0    5.5289E-02    4.4922E-05          weight cutoff        48446    5.5758E-02    5.7124E-07
 energy importance        0    0.            0.                  energy importance        0    0.            0.
 dxtran                   0    0.            0.                  dxtran                   0    0.            0.
 forced collisions        0    0.            0.                  forced collisions        0    0.            0.
 exp. transform           0    0.            0.                  exp. transform           0    0.            0.
 upscattering             0    0.            1.4137E-06          downscattering           0    0.            2.0164E+00
                                                                  capture                  0    4.0603E-01    1.8794E-02
 (n,xn)                 108    2.0707E-03    1.7868E-03          loss to (n,xn)          54    1.0353E-03    5.2008E-03
 fission                  0    0.            0.                  loss to fission          0    5.9355E-01    8.4459E-03
     total            48623    1.0574E+00    2.0489E+00              total            48623    1.0574E+00    2.0489E+00

   number of neutrons banked                      55        average time of (shakes)              cutoffs
   neutron tracks per source particle     1.0025E+00          escape            6.4176E+05          tco   1.0000E+34
   neutron collisions per source particle 1.7664E+02          capture           2.5960E+04          eco   0.0000E+00
   total neutron collisions                  8567119          capture or escape 2.7445E+04          wc1  -5.0000E-01
   net multiplication              1.0010E+00 0.0004          any termination   3.3058E+04          wc2  -2.5000E-01

 computer time so far in this run     8.16 minutes            maximum number ever in bank         1
 computer time in mcrun               7.63 minutes            bank overflows to backup file       0
 source particles per minute            6.5781E+03            dynamic storage   2751440 words,   11005760 bytes.
 random numbers generated                 76319886            most random numbers used was    47675 in history     1629


        keff estimator       keff     standard deviation      68% confidence        95% confidence        99% confidence

             collision     1.45359         0.00725          1.44633 to 1.46085    1.43915 to 1.46804    1.43443 to 1.472
            absorption     1.44725         0.00362          1.44363 to 1.45087    1.44005 to 1.45446    1.43770 to 1.456
          track length     1.45079         0.00725          1.44353 to 1.45805    1.43634 to 1.46524    1.43162 to 1.469
            col/absorp     1.44770         0.00362          1.44408 to 1.45133    1.44050 to 1.45491    1.43814 to 1.457
           abs/trk len     1.44747         0.00362          1.44386 to 1.45109    1.44027 to 1.45468    1.43792 to 1.457
           col/trk len     1.45221         0.00738          1.44483 to 1.45959    1.43752 to 1.46690    1.43272 to 1.471
       col/abs/trk len     1.44809         0.00368          1.44440 to 1.45178    1.44075 to 1.45543    1.43835 to 1.457


 if the largest of each keff occurred on the next cycle, the keff results and 68, 95, and 99 percent confidence interval

        keff estimator       keff     standard deviation      68% confidence        95% confidence        99% confidence

             collision     1.45558         0.00745          1.44812 to 1.46303    1.44074 to 1.47041    1.43590 to 1.475
            absorption     1.44813         0.00368          1.44444 to 1.45181    1.44079 to 1.45547    1.43839 to 1.457
          track length     1.45257         0.00740          1.44517 to 1.45998    1.43784 to 1.46731    1.43303 to 1.472
       col/abs/trk len     1.44871         0.00379          1.44491 to 1.45250    1.44116 to 1.45625    1.43869 to 1.458


 the estimated average prompt removal lifetimes, one standard deviations, and 68, 95, and 99 percent confidence interval

       estimator      lifetime     std. dev.         68% confidence           95% confidence           99% confidence

       collision   3.15814E-04   3.53635E-06    3.1227E-04 to 3.1935E-04 3.0877E-04 to 3.2286E-04 3.0647E-04 to 3.2516E-
      absorption   3.16055E-04   3.35623E-06    3.1270E-04 to 3.1941E-04 3.0937E-04 to 3.2274E-04 3.0719E-04 to 3.2492E-
    track length   3.15341E-04   2.91853E-06    3.1242E-04 to 3.1826E-04 3.0953E-04 to 3.2115E-04 3.0763E-04 to 3.2305E-
      col/absorp   3.16035E-04   3.37293E-06    3.1266E-04 to 3.1941E-04 3.0932E-04 to 3.2275E-04 3.0712E-04 to 3.2495E-
     abs/trk len   3.14981E-04   2.87882E-06    3.1210E-04 to 3.1786E-04 3.0925E-04 to 3.2072E-04 3.0737E-04 to 3.2259E-
     col/trk len   3.15234E-04   2.91081E-06    3.1232E-04 to 3.1815E-04 3.0944E-04 to 3.2103E-04 3.0754E-04 to 3.2292E-
 col/abs/trk len   3.14975E-04   2.89435E-06    3.1208E-04 to 3.1787E-04 3.0921E-04 to 3.2074E-04 3.0733E-04 to 3.2262E-

 absorption estimates of prompt lifetimes (sec):

                            escape        capture        fission        removal

            lifespan    6.41757E-03    2.59598E-04    3.44646E-04    3.16090E-04
            fraction    9.80914E-04    4.05804E-01    5.93215E-01    1.00000E+00
       lifetime(abs)    3.22205E-01    7.78837E-04    5.32783E-04    3.16055E-04
     lifetime(c/a/t)    3.21104E-01    7.76176E-04    5.30963E-04    3.14975E-04


1tally  54        nps =    50168
+                                  U-235 n,gamma
```

```
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00    7        102
                 1.85626E-05 0.0129

 cell  111
 multiplier bin:   1.00000E+00    7        102
                 3.44570E-05 0.0092

 cell  121
 multiplier bin:   1.00000E+00    7        102
                 5.30040E-05 0.0074

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.508E-01)**2 = (6.454E+03)*(1.230E-01) = 7.94
1tally  64        nps =    50168
+                                   U-235 n,fission
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00    7         -6
                 8.34337E-05 0.0140

 cell  111
 multiplier bin:   1.00000E+00    7         -6
                 1.52744E-04 0.0101

 cell  121
 multiplier bin:   1.00000E+00    7         -6
                 2.35509E-04 0.0082

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.245E-01)**2 = (6.454E+03)*(1.053E-01) = 6.79
1tally  74        nps =    50168
+                                   U-238 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00    8        102
                 3.43211E-06 0.0201

 cell  111
 multiplier bin:   1.00000E+00    8        102
                 6.69358E-06 0.0142

 cell  121
 multiplier bin:   1.00000E+00    8        102
                 1.01998E-05 0.0112

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.258E-01)**2 = (6.454E+03)*(5.100E-02) = 3.29
1tally  84        nps =    50168
+                                   U-238 n,fission
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00    8         -6
                 3.26439E-08 0.0176

 cell  111
 multiplier bin:   1.00000E+00    8         -6
                 6.24290E-08 0.0126

 cell  121
 multiplier bin:   1.00000E+00    8         -6
                 9.79062E-08 0.0100

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.586E-01)**2 = (6.454E+03)*(6.686E-02) = 4.31
1tally  94        nps =    50168
+                                   Th-232 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
```

```
 multiplier bin:   1.00000E+00    9        102
                 4.22461E-06 0.0312

 cell  111
 multiplier bin:   1.00000E+00    9        102
                 7.97934E-06 0.0214

 cell  121
 multiplier bin:   1.00000E+00    9        102
                 1.18146E-05 0.0165

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 1.458E-01)**2 = (6.454E+03)*(2.125E-02) = 1.37
1tally 104        nps =    50168
+                                   Pa-233 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   11        102
                 3.47073E-05 0.0177

 cell  111
 multiplier bin:   1.00000E+00   11        102
                 6.73129E-05 0.0125

 cell  121
 multiplier bin:   1.00000E+00   11        102
                 1.01664E-04 0.0100

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.572E-01)**2 = (6.454E+03)*(6.615E-02) = 4.27
1tally 114        nps =    50168
+                                   U-233 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   12        102
                 1.20760E-05 0.0130

 cell  111
 multiplier bin:   1.00000E+00   12        102
                 2.26021E-05 0.0092

 cell  121
 multiplier bin:   1.00000E+00   12        102
                 3.46292E-05 0.0073

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.487E-01)**2 = (6.454E+03)*(1.216E-01) = 7.84
1tally 124        nps =    50168
+                                   U-233 n,fission
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   12         -6
                 1.01881E-04 0.0131

 cell  111
 multiplier bin:   1.00000E+00   12         -6
                 1.88742E-04 0.0093

 cell  121
 multiplier bin:   1.00000E+00   12         -6
                 2.90212E-04 0.0075

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.454E-01)**2 = (6.454E+03)*(1.193E-01) = 7.69
1tally 134        nps =    50168
+                                   U-234 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   13        102
                 3.22290E-05 0.0296

 cell  111
 multiplier bin:   1.00000E+00   13        102
                 6.21524E-05 0.0232

 cell  121
 multiplier bin:   1.00000E+00   13        102
```

```
                 8.91726E-05 0.0175

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 1.534E-01)**2 = (6.454E+03)*(2.354E-02) = 1.51
1tally 144        nps =    50168
+                                    U-236 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   21        102
                 1.13821E-05 0.0494

 cell  111
 multiplier bin:   1.00000E+00   21        102
                 2.11504E-05 0.0363

 cell  121
 multiplier bin:   1.00000E+00   21        102
                 3.30093E-05 0.0292

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 9.185E-02)**2 = (6.454E+03)*(8.437E-03) = 5.44
1tally 154        nps =    50168
+                                    Np-237 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   22        102
                 7.76341E-05 0.0142

 cell  111
 multiplier bin:   1.00000E+00   22        102
                 1.45714E-04 0.0100

 cell  121
 multiplier bin:   1.00000E+00   22        102
                 2.24181E-04 0.0080

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.206E-01)**2 = (6.454E+03)*(1.028E-01) = 6.63
1tally 164        nps =    50168
+                                    Pu-238 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   23        102
                 5.54470E-05 0.0148

 cell  111
 multiplier bin:   1.00000E+00   23        102
                 1.00535E-04 0.0108

 cell  121
 multiplier bin:   1.00000E+00   23        102
                 1.55509E-04 0.0087

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.067E-01)**2 = (6.454E+03)*(9.407E-02) = 6.07
1tally 174        nps =    50168
+                                    Pu-238 n,fission
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   23         -6
                 2.70403E-06 0.0126

 cell  111
 multiplier bin:   1.00000E+00   23         -6
                 5.02537E-06 0.0088

 cell  121
 multiplier bin:   1.00000E+00   23         -6
                 7.73459E-06 0.0070

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.618E-01)**2 = (6.454E+03)*(1.309E-01) = 8.44
1tally 184        nps =    50168
+                                    Pu-239 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00
```

```
          volumes
                  cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   31        102
                 2.13046E-04 0.0152

 cell  111
 multiplier bin:   1.00000E+00   31        102
                 3.94514E-04 0.0110

 cell  121
 multiplier bin:   1.00000E+00   31        102
                 6.10159E-04 0.0089

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.982E-01)**2 = (6.454E+03)*(8.890E-02) = 5.73
1tally 194        nps =    50168
+                                   Pu-239 n,fission
          tally type 4    track length estimate of particle flux.
          tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

          volumes
                  cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   31         -6
                 3.45449E-04 0.0150

 cell  111
 multiplier bin:   1.00000E+00   31         -6
                 6.38257E-04 0.0109

 cell  121
 multiplier bin:   1.00000E+00   31         -6
                 9.86832E-04 0.0088

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.018E-01)**2 = (6.454E+03)*(9.108E-02) = 5.87
1tally 204        nps =    50168
+                                   Pu-240 n,gamma
          tally type 4    track length estimate of particle flux.
          tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

          volumes
                  cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   32        102
                 4.12650E-04 0.0299

 cell  111
 multiplier bin:   1.00000E+00   32        102
                 7.75347E-04 0.0211

 cell  121
 multiplier bin:   1.00000E+00   32        102
                 1.17800E-03 0.0168

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 1.521E-01)**2 = (6.454E+03)*(2.313E-02) = 1.49
1tally 214        nps =    50168
+                                   Pu-241 n,gamma
          tally type 4    track length estimate of particle flux.
          tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

          volumes
                  cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   33        102
                 1.03012E-04 0.0147

 cell  111
 multiplier bin:   1.00000E+00   33        102
                 1.88665E-04 0.0107

 cell  121
 multiplier bin:   1.00000E+00   33        102
                 2.91417E-04 0.0086

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.080E-01)**2 = (6.454E+03)*(9.486E-02) = 6.12
1tally 224        nps =    50168
+                                   Pu-241 n,fission
          tally type 4    track length estimate of particle flux.
          tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

          volumes
                  cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   33         -6
                 2.84988E-04 0.0146

 cell  111
```

```
 multiplier bin:   1.00000E+00   33         -6
                 5.20272E-04 0.0106

 cell  121
 multiplier bin:   1.00000E+00   33         -6
                 8.04165E-04 0.0085

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.111E-01)**2 = (6.454E+03)*(9.679E-02) = 6.24
1tally 234        nps =    50168
+                                   Am-241 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   34        102
                 3.39104E-04 0.0147

 cell  111
 multiplier bin:   1.00000E+00   34        102
                 6.35174E-04 0.0105

 cell  121
 multiplier bin:   1.00000E+00   34        102
                 9.75558E-04 0.0085

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.094E-01)**2 = (6.454E+03)*(9.573E-02) = 6.17
1tally 244        nps =    50168
+                                   Am-241 n,fission
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   34         -6
                 2.19882E-06 0.0140

 cell  111
 multiplier bin:   1.00000E+00   34         -6
                 4.13120E-06 0.0099

 cell  121
 multiplier bin:   1.00000E+00   34         -6
                 6.35445E-06 0.0080

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.245E-01)**2 = (6.454E+03)*(1.053E-01) = 6.79
1tally 254        nps =    50168
+                                   Am-242m n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   35        102
                 2.88264E-04 0.0149

 cell  111
 multiplier bin:   1.00000E+00   35        102
                 5.21675E-04 0.0108

 cell  121
 multiplier bin:   1.00000E+00   35        102
                 8.05263E-04 0.0088

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.053E-01)**2 = (6.454E+03)*(9.319E-02) = 6.01
1tally 264        nps =    50168
+                                   Am-242m n,fission
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   35         -6
                 1.42132E-03 0.0148

 cell  111
 multiplier bin:   1.00000E+00   35         -6
                 2.57544E-03 0.0107

 cell  121
 multiplier bin:   1.00000E+00   35         -6
                 3.97481E-03 0.0087

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.077E-01)**2 = (6.454E+03)*(9.467E-02) = 6.11
1tally 274        nps =    50168
```

```
+                                   Pu-242 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   36        102
                 4.71445E-05 0.0474

 cell  111
 multiplier bin:   1.00000E+00   36        102
                 8.78309E-05 0.0347

 cell  121
 multiplier bin:   1.00000E+00   36        102
                 1.26884E-04 0.0269

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 9.576E-02)**2 = (6.454E+03)*(9.171E-03) = 5.91
1tally 284        nps =    50168
+                                   Pu-242 n,fission
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   36         -6
                 1.71710E-07 0.0134

 cell  111
 multiplier bin:   1.00000E+00   36         -6
                 3.28005E-07 0.0093

 cell  121
 multiplier bin:   1.00000E+00   36         -6
                 5.11558E-07 0.0073

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.395E-01)**2 = (6.454E+03)*(1.152E-01) = 7.43
1tally 294        nps =    50168
+                                   Am-243 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   37        102
                 7.95391E-05 0.0223

 cell  111
 multiplier bin:   1.00000E+00   37        102
                 1.53998E-04 0.0158

 cell  121
 multiplier bin:   1.00000E+00   37        102
                 2.29211E-04 0.0127

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.039E-01)**2 = (6.454E+03)*(4.156E-02) = 2.68
1tally 304        nps =    50168
+                                   Cm-244 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   38        102
                 2.33712E-05 0.0518

 cell  111
 multiplier bin:   1.00000E+00   38        102
                 4.02479E-05 0.0354

 cell  121
 multiplier bin:   1.00000E+00   38        102
                 6.33249E-05 0.0295

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 8.767E-02)**2 = (6.454E+03)*(7.686E-03) = 4.96
1tally 314        nps =    50168
+                                   Cm-244 n,fission
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07
```

```
 cell  101
 multiplier bin:   1.00000E+00   38         -6
                 7.06700E-07 0.0263

 cell  111
 multiplier bin:   1.00000E+00   38         -6
                 1.29622E-06 0.0172

 cell  121
 multiplier bin:   1.00000E+00   38         -6
                 2.02210E-06 0.0142

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 1.728E-01)**2 = (6.454E+03)*(2.987E-02) = 1.92
1tally 324        nps =    50168
+                                     Cm-245 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   39        102
                 3.21012E-05 0.0146

 cell  111
 multiplier bin:   1.00000E+00   39        102
                 5.85232E-05 0.0106

 cell  121
 multiplier bin:   1.00000E+00   39        102
                 8.99827E-05 0.0086

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.101E-01)**2 = (6.454E+03)*(9.618E-02) = 6.20
1tally 334        nps =    50168
+                                     Cm-245 n,fission
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   39         -6
                 2.31770E-04 0.0144

 cell  111
 multiplier bin:   1.00000E+00   39         -6
                 4.23003E-04 0.0104

 cell  121
 multiplier bin:   1.00000E+00   39         -6
                 6.51037E-04 0.0084

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.159E-01)**2 = (6.454E+03)*(9.979E-02) = 6.44
1tally 514        nps =    50168
+                                     Xe-135 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   51        102
                 3.23716E-01 0.0162

 cell  111
 multiplier bin:   1.00000E+00   51        102
                 5.79165E-01 0.0120

 cell  121
 multiplier bin:   1.00000E+00   51        102
                 8.95102E-01 0.0098

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.808E-01)**2 = (6.454E+03)*(7.887E-02) = 5.09
1tally 524        nps =    50168
+                                     Sm-149 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   52        102
                 1.09488E-02 0.0163

 cell  111
 multiplier bin:   1.00000E+00   52        102
                 1.95492E-02 0.0120

 cell  121
```

```
 multiplier bin:   1.00000E+00   52        102
                 3.02552E-02 0.0098

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.789E-01)**2 = (6.454E+03)*(7.776E-02) = 5.01
1tally 534        nps =    50168
+                                     Nd-143 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101         111         121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   53        102
                 4.87169E-05 0.0143

 cell  111
 multiplier bin:   1.00000E+00   53        102
                 8.89611E-05 0.0103

 cell  121
 multiplier bin:   1.00000E+00   53        102
                 1.36989E-04 0.0084

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.183E-01)**2 = (6.454E+03)*(1.013E-01) = 6.54
1tally 544        nps =    50168
+                                     Rh-103 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101         111         121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   54        102
                 6.79507E-05 0.0165

 cell  111
 multiplier bin:   1.00000E+00   54        102
                 1.28806E-04 0.0118

 cell  121
 multiplier bin:   1.00000E+00   54        102
                 1.94548E-04 0.0092

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.747E-01)**2 = (6.454E+03)*(7.543E-02) = 4.86
1tally 564        nps =    50168
+                                     Pm-147 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101         111         121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   55        102
                 9.00299E-05 0.0274

 cell  111
 multiplier bin:   1.00000E+00   55        102
                 1.66849E-04 0.0195

 cell  121
 multiplier bin:   1.00000E+00   55        102
                 2.60320E-04 0.0162

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 1.660E-01)**2 = (6.454E+03)*(2.755E-02) = 1.77
1tally 574        nps =    50168
+                                     Sm-151 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101         111         121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   56        102
                 8.47465E-04 0.0157

 cell  111
 multiplier bin:   1.00000E+00   56        102
                 1.54037E-03 0.0115

 cell  121
 multiplier bin:   1.00000E+00   56        102
                 2.36234E-03 0.0092

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.898E-01)**2 = (6.454E+03)*(8.399E-02) = 5.42
1tally 584        nps =    50168
+                                     Xe-131 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00
```

```
          volumes
                  cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   57        102
                 4.64930E-05 0.0281

 cell  111
 multiplier bin:   1.00000E+00   57        102
                 8.64147E-05 0.0197

 cell  121
 multiplier bin:   1.00000E+00   57        102
                 1.37739E-04 0.0167

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 1.614E-01)**2 = (6.454E+03)*(2.604E-02) = 1.68
1tally 594        nps =    50168
+                                     Cs-133 n,gamma
          tally type 4    track length estimate of particle flux.
          tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

          volumes
                  cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   58        102
                 1.61366E-05 0.0272

 cell  111
 multiplier bin:   1.00000E+00   58        102
                 3.07924E-05 0.0187

 cell  121
 multiplier bin:   1.00000E+00   58        102
                 4.63099E-05 0.0147

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 1.668E-01)**2 = (6.454E+03)*(2.783E-02) = 1.79
1tally 604        nps =    50168
+                                     Tc-99 n,gamma
          tally type 4    track length estimate of particle flux.
          tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

          volumes
                  cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   59        102
                 1.23724E-05 0.0271

 cell  111
 multiplier bin:   1.00000E+00   59        102
                 2.44870E-05 0.0202

 cell  121
 multiplier bin:   1.00000E+00   59        102
                 3.73136E-05 0.0164

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 1.675E-01)**2 = (6.454E+03)*(2.807E-02) = 1.81
1tally 614        nps =    50168
+                                     Sm-152 n,gamma
          tally type 4    track length estimate of particle flux.
          tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

          volumes
                  cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   60        102
                 1.32353E-04 0.0309

 cell  111
 multiplier bin:   1.00000E+00   60        102
                 2.45183E-04 0.0214

 cell  121
 multiplier bin:   1.00000E+00   60        102
                 3.82628E-04 0.0175

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 1.469E-01)**2 = (6.454E+03)*(2.157E-02) = 1.39
1tally 624        nps =    50168
+                                     Nd-145 n,gamma
          tally type 4    track length estimate of particle flux.
          tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

          volumes
                  cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   61        102
                 1.37152E-05 0.0196
```

```
 cell  111
 multiplier bin:   1.00000E+00   61        102
                 2.64493E-05 0.0142

 cell  121
 multiplier bin:   1.00000E+00   61        102
                 3.96567E-05 0.0112

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.314E-01)**2 = (6.454E+03)*(5.353E-02) = 3.45
1tally 634        nps =    50168
+                                    Eu-153 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   62        102
                 8.53161E-05 0.0137

 cell  111
 multiplier bin:   1.00000E+00   62        102
                 1.60458E-04 0.0095

 cell  121
 multiplier bin:   1.00000E+00   62        102
                 2.45346E-04 0.0075

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.326E-01)**2 = (6.454E+03)*(1.106E-01) = 7.13
1tally 644        nps =    50168
+                                    Eu-155 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   63        102
                 6.35347E-04 0.0142

 cell  111
 multiplier bin:   1.00000E+00   63        102
                 1.15992E-03 0.0103

 cell  121
 multiplier bin:   1.00000E+00   63        102
                 1.78800E-03 0.0083

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.197E-01)**2 = (6.454E+03)*(1.022E-01) = 6.59
1tally 654        nps =    50168
+                                    Eu-154 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   64        102
                 1.79986E-04 0.0135

 cell  111
 multiplier bin:   1.00000E+00   64        102
                 3.37695E-04 0.0094

 cell  121
 multiplier bin:   1.00000E+00   64        102
                 5.15111E-04 0.0074

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.364E-01)**2 = (6.454E+03)*(1.132E-01) = 7.30
1tally 664        nps =    50168
+                                    Mo-95 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   65        102
                 5.90465E-06 0.0363

 cell  111
 multiplier bin:   1.00000E+00   65        102
                 1.14488E-05 0.0272

 cell  121
 multiplier bin:   1.00000E+00   65        102
                 1.85199E-05 0.0209

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 1.251E-01)**2 = (6.454E+03)*(1.566E-02) = 1.01
```

```
1tally 674        nps =    50168
+                                   Sm-150 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   66        102
                 1.96925E-05 0.0166

 cell  111
 multiplier bin:   1.00000E+00   66        102
                 3.69595E-05 0.0125

 cell  121
 multiplier bin:   1.00000E+00   66        102
                 5.60875E-05 0.0096

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.730E-01)**2 = (6.454E+03)*(7.454E-02) = 4.81
1tally 684        nps =    50168
+                                   Cs-134 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   67        102
                 2.21726E-05 0.0139

 cell  111
 multiplier bin:   1.00000E+00   67        102
                 4.05323E-05 0.0100

 cell  121
 multiplier bin:   1.00000E+00   67        102
                 6.25438E-05 0.0081

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.260E-01)**2 = (6.454E+03)*(1.063E-01) = 6.85
1tally 694        nps =    50168
+                                   Pr-141 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   68        102
                 2.40536E-06 0.0150

 cell  111
 multiplier bin:   1.00000E+00   68        102
                 4.46210E-06 0.0106

 cell  121
 multiplier bin:   1.00000E+00   68        102
                 6.78854E-06 0.0083

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.019E-01)**2 = (6.454E+03)*(9.112E-02) = 5.88
1tally 704        nps =    50168
+                                   Ru-101 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   69        102
                 3.92393E-06 0.0252

 cell  111
 multiplier bin:   1.00000E+00   69        102
                 7.44438E-06 0.0177

 cell  121
 multiplier bin:   1.00000E+00   69        102
                 1.17610E-05 0.0144

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 1.803E-01)**2 = (6.454E+03)*(3.249E-02) = 2.09
1tally 714        nps =    50168
+                                   Kr-83 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07
```

```
 cell  101
 multiplier bin:   1.00000E+00   70        102
                 3.38056E-05 0.0154

 cell  111
 multiplier bin:   1.00000E+00   70        102
                 6.26393E-05 0.0112

 cell  121
 multiplier bin:   1.00000E+00   70        102
                 9.64941E-05 0.0089

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.955E-01)**2 = (6.454E+03)*(8.733E-02) = 5.63
1tally 724        nps =    50168
+                                     Sm-147 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   71        102
                 3.46873E-05 0.0208

 cell  111
 multiplier bin:   1.00000E+00   71        102
                 6.57961E-05 0.0145

 cell  121
 multiplier bin:   1.00000E+00   71        102
                 1.01432E-04 0.0116

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.188E-01)**2 = (6.454E+03)*(4.787E-02) = 3.09
1tally 734        nps =    50168
+                                     Ag-109 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   72        102
                 5.41898E-05 0.0277

 cell  111
 multiplier bin:   1.00000E+00   72        102
                 1.08034E-04 0.0208

 cell  121
 multiplier bin:   1.00000E+00   72        102
                 1.56000E-04 0.0159

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 1.639E-01)**2 = (6.454E+03)*(2.686E-02) = 1.73
1tally 744        nps =    50168
+                                     Cs-135 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   73        102
                 3.36665E-06 0.0116

 cell  111
 multiplier bin:   1.00000E+00   73        102
                 6.49452E-06 0.0079

 cell  121
 multiplier bin:   1.00000E+00   73        102
                 9.98102E-06 0.0061

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.902E-01)**2 = (6.454E+03)*(1.522E-01) = 9.82
1tally 754        nps =    50168
+                                     I-129 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   74        102
                 5.15943E-06 0.0133

 cell  111
 multiplier bin:   1.00000E+00   74        102
                 9.60266E-06 0.0097
```

```
 cell  121
 multiplier bin:   1.00000E+00   74        102
                 1.48428E-05 0.0078

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.403E-01)**2 = (6.454E+03)*(1.158E-01) = 7.47
1tally 764        nps =    50168
+                                     Zr-93 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   75        102
                 1.48602E-06 0.0135

 cell  111
 multiplier bin:   1.00000E+00   75        102
                 2.83423E-06 0.0092

 cell  121
 multiplier bin:   1.00000E+00   75        102
                 4.39995E-06 0.0072

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.375E-01)**2 = (6.454E+03)*(1.139E-01) = 7.35
1tally 774        nps =    50168
+                                     Pd-105 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   76        102
                 5.41105E-06 0.0159

 cell  111
 multiplier bin:   1.00000E+00   76        102
                 1.02117E-05 0.0110

 cell  121
 multiplier bin:   1.00000E+00   76        102
                 1.55650E-05 0.0087

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.848E-01)**2 = (6.454E+03)*(8.109E-02) = 5.23
1tally 784        nps =    50168
+                                     Pd-108 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   77        102
                 8.68511E-06 0.0425

 cell  111
 multiplier bin:   1.00000E+00   77        102
                 1.89474E-05 0.0323

 cell  121
 multiplier bin:   1.00000E+00   77        102
                 2.86311E-05 0.0253

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 1.068E-01)**2 = (6.454E+03)*(1.140E-02) = 7.36
1tally 794        nps =    50168
+                                     Gd-157 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                        1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   78        102
                 1.24873E-02 0.0173

 cell  111
 multiplier bin:   1.00000E+00   78        102
                 2.24072E-02 0.0130

 cell  121
 multiplier bin:   1.00000E+00   78        102
                 3.45417E-02 0.0105

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 2.624E-01)**2 = (6.454E+03)*(6.886E-02) = 4.44
1tally 804        nps =    50168
+                                     Pm-148 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
```

```
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   79        102
                 2.68808E-03 0.0148

 cell  111
 multiplier bin:   1.00000E+00   79        102
                 5.12499E-03 0.0104

 cell  121
 multiplier bin:   1.00000E+00   79        102
                 7.83227E-03 0.0082

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.062E-01)**2 = (6.454E+03)*(9.377E-02) = 6.05
1tally 814        nps =    50168
+                                     Gd-156 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   80        102
                 3.93675E-06 0.0377

 cell  111
 multiplier bin:   1.00000E+00   80        102
                 7.92504E-06 0.0291

 cell  121
 multiplier bin:   1.00000E+00   80        102
                 1.20982E-05 0.0227

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 1.204E-01)**2 = (6.454E+03)*(1.449E-02) = 9.35
1tally 824        nps =    50168
+                                     Rh-105 n,gamma
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101          111          121
                         1.63910E+07  1.63910E+07  1.63910E+07

 cell  101
 multiplier bin:   1.00000E+00   81        102
                 2.97672E-03 0.0139

 cell  111
 multiplier bin:   1.00000E+00   81        102
                 5.48077E-03 0.0099

 cell  121
 multiplier bin:   1.00000E+00   81        102
                 8.43166E-03 0.0080

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.278E-01)**2 = (6.454E+03)*(1.075E-01) = 6.93
1tally 914        nps =    50168
+                                     tally fissions 101 m2
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      101
                         1.63910E+07

 cell  101
 multiplier bin:  -1.00000E+00    2         -6
                 6.42002E-09 0.0140

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 3.252E-01)**2 = (6.454E+03)*(1.058E-01) = 6.82
1tally 924        nps =    50168
+                                     tally fissions 111 m3
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00

           volumes
                   cell:      111
                         1.63910E+07

 cell  111
 multiplier bin:  -1.00000E+00    3         -6
                 1.17551E-08 0.0101

 fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 4.515E-01)**2 = (6.454E+03)*(2.038E-01) = 1.31
1tally 934        nps =    50168
+                                     tally fissions 111 m3
           tally type 4    track length estimate of particle flux.
           tally for  neutrons
 number of histories used for normalizing tallies =    48500.00
```

```
          volumes
                  cell:      121
                        1.63910E+07

cell  121
multiplier bin:  -1.00000E+00    4         -6
                 1.81258E-08 0.0081

fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (6.454E+03)*( 5.578E-01)**2 = (6.454E+03)*(3.111E-01) = 2.00

************************************************************************************************************************
dump no.    3 on file runtpe       nps =     50168     coll =        8567119     ctm =    7.63     nrn =       76319886


       58 warning messages so far.


run terminated when   100 kcode cycles were done.

computer time =    8.16 minutes

mcnp     version 4b    02/04/97                      08/08/02 06:30:29                    probid =   08/08/02 06:22:19
```

APPENDIX G – Input File SETUP.IN

```
17

1     1  1 5.000
2     7  1 5.000
3    16  1 5.000
4    36  1 5.000
5    60  1 5.000
6    60  1 5.000
7   180  1 5.000
8   180  1 5.000
9   180  1 5.000

10 180  1 5.000
11 180  1 5.000
12 180  1 5.000
13 180  1 5.000
14 180  1 5.000
15 180  1 5.000

16 180  1 5.000
17 180  1 5.000
```

APPENDIX H – Output File CELL.TMP

```
  3   0.169646E+09
101           2   0.163910E+08
111           3   0.163910E+08
121           4   0.163910E+08
```

APPENDIX I – Output File TAL.TMP

1	92	235	7
2	92	238	8
3	90	232	9
4	91	233	11
5	92	233	12
6	92	234	13
7	92	236	21
8	93	237	22
9	94	238	23
10	94	239	31
11	94	240	32
12	94	241	33
13	95	241	34
14	95	242	35
15	94	242	36
16	95	243	37
17	96	244	38
18	96	245	39
19	54	135	51
20	62	149	52
21	60	143	53
22	45	103	54
23	61	147	55
24	62	151	56
25	54	131	57
26	55	133	58
27	43	99	59
28	62	152	60
29	60	145	61
30	63	153	62
31	63	155	63
32	63	154	64
33	42	95	65
34	62	150	66
35	55	134	67
36	59	141	68
37	44	101	69
38	36	83	70
39	62	147	71
40	47	109	72
41	55	135	73
42	53	129	74
43	40	93	75
44	46	105	76
45	46	108	77
46	64	157	78
47	61	148	79
48	64	156	80
49	45	105	81

APPENDIX J – Output File ISO.TMP

```
7    360.000
2          4          9   0.000000       0.000000
2          6         12   0.871038E+23   -1.73540
2          7         14   0.000000       0.000000
2          7         15   0.000000       0.000000
2          8         16   0.738062E+21  -0.196000E-01
2          9         19   0.000000       0.000000
2         14          0   0.840664E+21  -0.392000E-01
2         40          0   0.660254E-02  -0.999999E-24
2          2          6   0.000000       0.000000
2          3          6   0.000000       0.000000
2          3          7   0.000000       0.000000
2          5         10   0.000000       0.000000
2          5         11   0.000000       0.000000
2         90        232   0.257718E-02  -0.992853E-24
2         90        233   0.198840E-08  -0.769339E-30
2         91        233   0.208835E-05  -0.808009E-27
2         92        233   0.435747E-02  -0.168595E-23
2         91        234   0.128520E-09  -0.499394E-31
2         92        234   0.254199E-02  -0.987751E-24
2         92        235   0.653930E+20  -0.255189E-01
2         92        236   0.206228E+19  -0.808211E-03
2         92        237   0.954248E+15  -0.375561E-06
2         93        237   0.179137E+17  -0.705021E-05
2         93        238   0.179007E+14  -0.707492E-08
2         94        238   0.712005E+15  -0.281405E-06
2         92        238   0.677699E+21  -0.267848
2         92        239   0.441471E+15  -0.175217E-06
2         93        239   0.359999E+17  -0.142882E-04
2         94        239   0.262651E+19  -0.104244E-02
2         94        240   0.388318E+18  -0.154767E-03
2         94        241   0.788925E+17  -0.315745E-04
2         95        241   0.956499E+15  -0.382812E-06
2         95        242   0.727748E+14  -0.292472E-07
2         94        242   0.340653E+16  -0.136903E-05
2         94        243   0.206144E+12  -0.831894E-10
2         95        243   0.602519E+14  -0.243146E-07
2         95        244   0.887265E+09  -0.359533E-12
2         96        244   0.136906E+13  -0.554759E-09
2         96        245   0.707795E+10  -0.287986E-11
2         35         83   0.206158E+14  -0.283792E-08
2         36         83   0.525215E+17  -0.723014E-05
2         39         93   0.103706E+16  -0.159972E-06
2         40         93   0.636712E+18  -0.982133E-04
2         40         95   0.160125E+18  -0.252315E-04
2         41         95   0.869589E+17  -0.137023E-04
2         42         95   0.405686E+18  -0.639240E-04
2         40         97   0.161051E+16  -0.259129E-06
2         41         97   0.117912E+15  -0.189714E-07
2         42         97   0.600341E+18  -0.965896E-04
2         42         99   0.645373E+16  -0.105980E-05
2         43         99   0.601987E+18  -0.988534E-04
2         44        101   0.518710E+18  -0.869003E-04
2         44        103   0.460697E+17  -0.787115E-05
2         45        103   0.247104E+18  -0.422183E-04
2         44        105   0.685823E+14  -0.119446E-07
2         45        105   0.532788E+15  -0.927972E-07
2         46        105   0.933705E+17  -0.162625E-04
2         46        108   0.539699E+16  -0.966874E-06
2         46        109   0.694625E+13  -0.125597E-08
2         47        109   0.303965E+16  -0.549606E-06
2         51        129   0.383031E+14  -0.819759E-08
2         52        129   0.100981E+14  -0.216118E-08
2         53        129   0.551737E+17  -0.118082E-04
2         52        131   0.214003E+15  -0.465110E-07
2         53        131   0.893646E+16  -0.194223E-05
2         54        131   0.271155E+18  -0.589325E-04
2         53        133   0.223154E+16  -0.492410E-06
2         54        133   0.135001E+17  -0.297893E-05
2         55        133   0.649825E+18  -0.143391E-03
2         55        134   0.727507E+16  -0.161741E-05
2         53        135   0.661220E+15  -0.148106E-06
2         54        135   0.555436E+15  -0.124408E-06
2         55        135   0.372209E+18  -0.833680E-04
2         56        139   0.143440E+15  -0.330805E-07
2         57        139   0.645152E+18  -0.148787E-03
2         57        141   0.365333E+15  -0.854681E-07
2         58        141   0.733835E+17  -0.171678E-04
2         59        141   0.513691E+18  -0.120176E-03
2         58        143   0.316025E+16  -0.749833E-06
2         59        143   0.310766E+17  -0.737355E-05
2         60        143   0.545369E+18  -0.129400E-03
2         58        144   0.366998E+18  -0.876867E-04
2         59        144   0.372404E+14  -0.889784E-08
2         60        144   0.205912E+18  -0.491987E-04
2         59        145   0.376837E+15  -0.906647E-07
2         60        145   0.390908E+18  -0.940507E-04
2         60        147   0.948587E+16  -0.231381E-05
2         61        147   0.179398E+18  -0.437588E-04
2         61        148   0.222063E+16  -0.545352E-06
2         62        147   0.234997E+17  -0.573205E-05
2         60        149   0.298055E+14  -0.736930E-08
2         61        149   0.100193E+16  -0.247724E-06
2         62        149   0.706670E+16  -0.174720E-05
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2	62	150	0.103802E+18	-0.258368E-04
2	61	151	0.190487E+15	-0.477301E-07
2	62	151	0.243428E+17	-0.609957E-05
2	62	152	0.418007E+17	-0.105434E-04
2	62	153	0.183382E+15	-0.465589E-07
2	63	153	0.186263E+17	-0.472907E-05
2	63	154	0.107530E+16	-0.274799E-06
2	63	155	0.201219E+16	-0.517566E-06
2	62	156	0.222870E+13	-0.576950E-09
2	63	156	0.197412E+15	-0.511047E-07
2	64	156	0.243336E+16	-0.629933E-06
2	63	157	0.150277E+13	-0.391526E-09
2	64	157	0.333602E+14	-0.869159E-08
3	4	9	0.000000	0.000000
3	6	12	0.871038E+23	-1.73540
3	7	14	0.000000	0.000000
3	7	15	0.000000	0.000000
3	8	16	0.738062E+21	-0.196000E-01
3	9	19	0.000000	0.000000
3	14	0	0.840664E+21	-0.392000E-01
3	40	0	0.660254E-02	-0.999999E-24
3	2	6	0.000000	0.000000
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3	3	7	0.000000	0.000000
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3	90	232	0.256422E-02	-0.987862E-24
3	90	233	0.342688E-08	-0.132591E-29
3	91	233	0.340372E-05	-0.131694E-26
3	92	233	0.388416E-02	-0.150283E-23
3	91	234	0.363857E-09	-0.141385E-30
3	92	234	0.250937E-02	-0.975073E-24
3	92	235	0.587736E+20	-0.229357E-01
3	92	236	0.327896E+19	-0.128503E-02
3	92	237	0.277372E+16	-0.109164E-05
3	93	237	0.515403E+17	-0.202845E-04
3	93	238	0.859642E+14	-0.339757E-07
3	94	238	0.344977E+16	-0.136345E-05
3	92	238	0.674532E+21	-0.266596
3	92	239	0.749582E+15	-0.297505E-06
3	93	239	0.611275E+17	-0.242612E-04
3	94	239	0.364754E+19	-0.144768E-02
3	94	240	0.846665E+18	-0.337444E-03
3	94	241	0.287250E+18	-0.114964E-03
3	95	241	0.356418E+16	-0.142646E-05
3	95	242	0.394449E+15	-0.158524E-06
3	94	242	0.220066E+17	-0.884412E-05
3	94	243	0.233796E+13	-0.943485E-09
3	95	243	0.710594E+15	-0.286760E-06
3	95	244	0.175618E+11	-0.711632E-11
3	96	244	0.288997E+14	-0.117105E-07
3	96	245	0.263188E+12	-0.107085E-09
3	35	83	0.336288E+14	-0.462926E-08
3	36	83	0.861050E+17	-0.118533E-04
3	39	93	0.169167E+16	-0.260950E-06
3	40	93	0.106166E+19	-0.163762E-03
3	40	95	0.260601E+18	-0.410639E-04
3	41	95	0.141424E+18	-0.222844E-04
3	42	95	0.685433E+18	-0.108004E-03
3	40	97	0.262709E+16	-0.422696E-06
3	41	97	0.192340E+15	-0.309465E-07
3	42	97	0.100187E+19	-0.161192E-03
3	42	99	0.105274E+17	-0.172875E-05
3	43	99	0.996676E+18	-0.163666E-03
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3	44	103	0.749795E+17	-0.128105E-04
3	45	103	0.400876E+18	-0.684905E-04
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3	45	105	0.855203E+15	-0.148953E-06
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3	58	143	0.515507E+16	-0.122314E-05
3	59	143	0.506917E+17	-0.120276E-04
3	60	143	0.890751E+18	-0.211349E-03
3	58	144	0.606141E+18	-0.144825E-03
3	59	144	0.615072E+14	-0.146959E-07
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3	61	148	0.493242E+16	-0.121133E-05
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3	62	150	0.185179E+18	-0.460918E-04
3	61	151	0.310725E+15	-0.778578E-07
3	62	151	0.312335E+17	-0.782615E-05
3	62	152	0.749740E+17	-0.189106E-04
3	62	153	0.396615E+15	-0.100697E-06
3	63	153	0.354984E+17	-0.901277E-05
3	63	154	0.323470E+16	-0.826643E-06
3	63	155	0.286143E+16	-0.736005E-06
3	62	156	0.363549E+13	-0.941131E-09
3	63	156	0.397637E+15	-0.102938E-06
3	64	156	0.485460E+16	-0.125673E-05
3	63	157	0.245134E+13	-0.638664E-09
3	64	157	0.354511E+14	-0.923634E-08
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4	7	14	0.000000	0.000000
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4	40	0	0.660254E-02	-0.999999E-24
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4	3	7	0.000000	0.000000
4	5	10	0.000000	0.000000
4	5	11	0.000000	0.000000
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4	91	233	0.565798E-05	-0.218914E-26
4	92	233	0.329930E-02	-0.127654E-23
4	91	234	0.109801E-08	-0.426655E-30
4	92	234	0.246113E-02	-0.956331E-24
4	92	235	0.506146E+20	-0.197518E-01
4	92	236	0.473014E+19	-0.185375E-02
4	92	237	0.665668E+16	-0.261985E-05
4	93	237	0.106935E+18	-0.420861E-04
4	93	238	0.319267E+15	-0.126184E-06
4	94	238	0.113067E+17	-0.446873E-05
4	92	238	0.670793E+21	-0.265118
4	92	239	0.133604E+16	-0.530266E-06
4	93	239	0.108960E+18	-0.432456E-04
4	94	239	0.430276E+19	-0.170774E-02
4	94	240	0.137982E+19	-0.549938E-03
4	94	241	0.694739E+18	-0.278050E-03
4	95	241	0.750701E+16	-0.300447E-05
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4	96	244	0.313221E+15	-0.126921E-06
4	96	245	0.431554E+13	-0.175589E-08
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4	36	83	0.130178E+18	-0.179203E-04
4	39	93	0.298688E+16	-0.460743E-06
4	40	93	0.164312E+19	-0.253452E-03
4	40	95	0.443602E+18	-0.699001E-04
4	41	95	0.232446E+18	-0.366270E-04
4	42	95	0.100552E+19	-0.158441E-03
4	40	97	0.463848E+16	-0.746329E-06
4	41	97	0.339603E+15	-0.546403E-07
4	42	97	0.155223E+19	-0.249740E-03
4	42	99	0.185874E+17	-0.305233E-05
4	43	99	0.152669E+19	-0.250701E-03
4	44	101	0.133327E+19	-0.223365E-03
4	44	103	0.130756E+18	-0.223402E-04
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4	44	105	0.197527E+15	-0.344023E-07
4	45	105	0.146196E+16	-0.254634E-06
4	46	105	0.229568E+18	-0.399843E-04
4	46	108	0.137577E+17	-0.246470E-05
4	46	109	0.209563E+14	-0.378915E-08
4	47	109	0.744957E+16	-0.134697E-05
4	51	129	0.110318E+15	-0.236102E-07
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4	53	129	0.141738E+18	-0.303347E-04
4	52	131	0.616360E+15	-0.133959E-06
4	53	131	0.257377E+17	-0.559379E-05
4	54	131	0.655249E+18	-0.142411E-03
4	53	133	0.642716E+16	-0.141821E-05
4	54	133	0.388815E+17	-0.857956E-05
4	55	133	0.163793E+19	-0.361425E-03
4	55	134	0.523095E+17	-0.116296E-04
4	53	135	0.190441E+16	-0.426565E-06
4	54	135	0.901823E+15	-0.201994E-06
4	55	135	0.578828E+18	-0.129647E-03
4	56	139	0.413127E+15	-0.952767E-07
4	57	139	0.166858E+19	-0.384813E-03
4	57	141	0.105221E+16	-0.246161E-06
4	58	141	0.209603E+18	-0.490358E-04
4	59	141	0.130518E+19	-0.305341E-03
4	58	143	0.910195E+16	-0.215962E-05
4	59	143	0.895012E+17	-0.212360E-04
4	60	143	0.132430E+19	-0.314218E-03
4	58	144	0.963589E+18	-0.230230E-03
4	59	144	0.977768E+14	-0.233618E-07

4	60	144	0.594920E+18	-0.142145E-03
4	59	145	0.108534E+16	-0.261127E-06
4	60	145	0.992044E+18	-0.238681E-03
4	60	147	0.273189E+17	-0.666368E-05
4	61	147	0.412591E+18	-0.100639E-03
4	61	148	0.914635E+16	-0.224620E-05
4	62	147	0.527112E+17	-0.128573E-04
4	60	149	0.858443E+14	-0.212247E-07
4	61	149	0.368159E+16	-0.910258E-06
4	62	149	0.908145E+16	-0.224534E-05
4	62	150	0.310412E+18	-0.772629E-04
4	61	151	0.548628E+15	-0.137469E-06
4	62	151	0.358842E+17	-0.899149E-05
4	62	152	0.119239E+18	-0.300755E-04
4	62	153	0.934167E+15	-0.237177E-06
4	63	153	0.643242E+17	-0.163314E-04
4	63	154	0.824848E+16	-0.210794E-05
4	63	155	0.408358E+16	-0.105036E-05
4	62	156	0.641897E+13	-0.166170E-08
4	63	156	0.890639E+15	-0.230563E-06
4	64	156	0.889733E+16	-0.230329E-05
4	63	157	0.432819E+13	-0.112765E-08
4	64	157	0.365396E+14	-0.951993E-08

APPENDIX K – Output File CONC.OUT

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2           8          16   0.738062E+21  -0.196000E-01
2           9          19   0.000000       0.000000
2          14           0   0.840664E+21  -0.392000E-01
2          40           0   0.660254E-02  -0.100000E-23
2           2           6   0.000000       0.000000
2           3           6   0.000000       0.000000
2          90         232   0.259573E-02  -0.100000E-23
2          90         233   0.000000       0.000000
2          91         233   0.258457E-02  -0.100000E-23
2          92         233   0.258457E-02  -0.100000E-23
2          91         234   0.000000       0.000000
2          92         234   0.257352E-02  -0.100000E-23
2          92         235   0.766941E+20  -0.299290E-01
2          92         236   0.255166E-02  -0.100000E-23
2          92         237   0.000000       0.000000
2          93         237   0.254087E-02  -0.100000E-23
2          93         238   0.000000       0.000000
2          94         238   0.253018E-02  -0.100000E-23
2          92         238   0.681523E+21  -0.269359
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2          93         239   0.000000       0.000000
2          94         239   0.251957E-02  -0.100000E-23
2          94         240   0.250905E-02  -0.100000E-23
2          94         241   0.249861E-02  -0.100000E-23
2          95         241   0.249861E-02  -0.100000E-23
2          95         242   0.248826E-02  -0.100000E-23
2          94         242   0.248827E-02  -0.100000E-23
2          94         243   0.000000       0.000000
2          95         243   0.247801E-02  -0.100000E-23
2          95         244   0.000000       0.000000
2          96         244   0.246784E-02  -0.100000E-23
2          96         245   0.245774E-02  -0.100000E-23
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2          45         103   0.585302E-02  -0.100000E-23
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2          45         105   0.574143E-02  -0.100000E-23
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2          60         144   0.000000       0.000000
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2          62         149   0.404458E-02  -0.100000E-23
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3	3	6	0.000000	0.000000
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3	90	233	0.000000	0.000000
3	91	233	0.258457E-02	-0.100000E-23
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3	93	238	0.000000	0.000000
3	94	238	0.253018E-02	-0.100000E-23
3	92	238	0.681523E+21	-0.269359
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3	94	239	0.251957E-02	-0.100000E-23
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3	94	241	0.249861E-02	-0.100000E-23
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3	62	150	0.154755E+19	-0.385191E-03
3	61	151	0.326456E+15	-0.817997E-07
3	62	151	0.457758E+17	-0.114700E-04
3	62	152	0.364238E+18	-0.918716E-04
3	62	153	0.222219E+16	-0.564195E-06
3	63	153	0.457768E+18	-0.116224E-03
3	63	154	0.170155E+18	-0.434841E-04
3	63	155	0.422478E+17	-0.108668E-04
3	62	156	0.348324E+13	-0.901717E-09
3	63	156	0.100049E+17	-0.259000E-05
3	64	156	0.314125E+18	-0.813189E-04
3	63	157	0.272081E+13	-0.708870E-09
3	64	157	0.986345E+14	-0.256980E-07
4	3	7	0.000000	0.000000
4	4	9	0.000000	0.000000
4	6	12	0.871038E+23	-1.73540
4	8	16	0.738062E+21	-0.196000E-01
4	9	19	0.000000	0.000000
4	14	0	0.840664E+21	-0.392000E-01
4	40	0	0.660254E-02	-0.999999E-24
4	2	6	0.000000	0.000000
4	3	6	0.000000	0.000000
4	90	232	0.229804E-02	-0.885317E-24
4	90	233	0.853958E-08	-0.330408E-29
4	91	233	0.777443E-05	-0.300802E-26
4	92	233	0.281079E-03	-0.108752E-24
4	91	234	0.225085E-08	-0.874618E-30
4	92	234	0.132605E-02	-0.515269E-24
4	92	235	0.422641E+19	-0.164931E-02
4	92	236	0.110740E+20	-0.433991E-02
4	92	237	0.222931E+17	-0.877385E-05
4	93	237	0.106942E+19	-0.420888E-03
4	93	238	0.581748E+16	-0.229925E-05
4	94	238	0.766899E+18	-0.303101E-03
4	92	238	0.613898E+21	-0.242632
4	92	239	0.200888E+16	-0.797314E-06
4	93	239	0.163851E+18	-0.650316E-04
4	94	239	0.358725E+19	-0.142376E-02
4	94	240	0.318157E+19	-0.126804E-02
4	94	241	0.198294E+19	-0.793617E-03
4	95	241	0.375164E+17	-0.150149E-04
4	95	242	0.718702E+16	-0.288837E-05
4	94	242	0.334884E+19	-0.134585E-02
4	94	243	0.849913E+15	-0.342983E-06
4	95	243	0.100754E+19	-0.406594E-03
4	95	244	0.679079E+14	-0.275173E-07
4	96	244	0.605737E+18	-0.245452E-03
4	96	245	0.303387E+17	-0.123441E-04
4	35	83	0.489335E+14	-0.673606E-08
4	36	83	0.371565E+18	-0.511499E-04
4	39	93	0.249506E+16	-0.384877E-06
4	40	93	0.741742E+19	-0.114414E-02
4	40	95	0.369319E+18	-0.581950E-04
4	41	95	0.195399E+18	-0.307894E-04
4	42	95	0.648915E+19	-0.102250E-02
4	40	97	0.384681E+16	-0.618948E-06
4	41	97	0.281641E+15	-0.453145E-07
4	42	97	0.711652E+19	-0.114498E-02
4	42	99	0.155374E+17	-0.255147E-05
4	43	99	0.599878E+19	-0.985072E-03
4	44	101	0.563697E+19	-0.944369E-03
4	44	103	0.112491E+18	-0.192195E-04
4	45	103	0.146726E+19	-0.250684E-03
4	44	105	0.168311E+15	-0.293140E-07
4	45	105	0.116131E+16	-0.202269E-06
4	46	105	0.994590E+18	-0.173230E-03
4	46	108	0.753529E+17	-0.134995E-04
4	46	109	0.293283E+14	-0.530291E-08
4	47	109	0.243223E+17	-0.439777E-05
4	51	129	0.107681E+15	-0.230457E-07
4	52	129	0.283885E+14	-0.607568E-08
4	53	129	0.729007E+18	-0.156021E-03
4	52	131	0.441146E+15	-0.958780E-07
4	53	131	0.210239E+17	-0.456930E-05
4	54	131	0.170938E+19	-0.371515E-03
4	53	133	0.541318E+16	-0.119447E-05
4	54	133	0.327474E+17	-0.722602E-05
4	55	133	0.624917E+19	-0.137894E-02
4	55	134	0.896226E+18	-0.199251E-03
4	53	135	0.161695E+16	-0.362179E-06
4	54	135	0.397745E+15	-0.890883E-07
4	55	135	0.259312E+19	-0.580811E-03
4	56	139	0.347648E+15	-0.801758E-07
4	57	139	0.774132E+19	-0.178533E-02
4	57	141	0.882072E+15	-0.206357E-06
4	58	141	0.175588E+18	-0.410780E-04
4	59	141	0.659435E+19	-0.154272E-02
4	58	143	0.759516E+16	-0.180210E-05
4	59	143	0.746849E+17	-0.177205E-04
4	60	143	0.336843E+19	-0.799229E-03
4	58	144	0.122647E+19	-0.293039E-03

4	59	144	0.124459E+15	-0.297370E-07
4	60	144	0.896893E+19	-0.214296E-02
4	59	145	0.905426E+15	-0.217840E-06
4	60	145	0.377392E+19	-0.907988E-03
4	60	147	0.231127E+17	-0.563769E-05
4	61	147	0.563075E+18	-0.137345E-03
4	61	148	0.149749E+17	-0.367759E-05
4	62	147	0.503672E+18	-0.122855E-03
4	60	149	0.722922E+14	-0.178739E-07
4	61	149	0.490671E+16	-0.121316E-05
4	62	149	0.523757E+16	-0.129496E-05
4	62	150	0.164041E+19	-0.408306E-03
4	61	151	0.460313E+15	-0.115340E-06
4	62	151	0.462424E+17	-0.115869E-04
4	62	152	0.390311E+18	-0.984479E-04
4	62	153	0.330150E+16	-0.838222E-06
4	63	153	0.483640E+18	-0.122792E-03
4	63	154	0.185588E+18	-0.474279E-04
4	63	155	0.458509E+17	-0.117935E-04
4	62	156	0.491147E+13	-0.127145E-08
4	63	156	0.162293E+17	-0.420133E-05
4	64	156	0.391607E+18	-0.101377E-03
4	63	157	0.383644E+13	-0.999530E-09
4	64	157	0.111162E+15	-0.289620E-07

APPENDIX L – Output File KEFF.OUT

```
0.0000000E+00   1.448090      3.6800001E-03
 1.000000       1.400020      4.1399999E-03
 8.000000       1.396120      3.4500000E-03
 24.00000       1.384960      3.3499999E-03
 60.00000       1.365530      3.6299999E-03
 120.0000       1.342850      3.4900000E-03
 180.0000       1.317340      3.6400000E-03
 360.0000       1.241950      3.3199999E-03
 540.0000       1.188870      3.5100000E-03
 720.0000       1.139180      4.1200002E-03
 900.0000       1.101380      3.1600001E-03
 1080.000       1.053000      3.0600000E-03
 1260.000       1.027410      3.9800000E-03
 1440.000      0.9847800      3.2899999E-03
 1620.000      0.9530200      3.1999999E-03
 1800.000      0.9086000      3.6500001E-03
 1980.000      0.8723300      3.9499998E-03
 2160.000      0.8360900      2.9800001E-03
```

APPENDIX M – Fortran Code RELOAD.FOR

```
c
c
c
c  Performs manual adjustment (dilution) in fuel density
c  plus reload of same quantity of fresh fuel
c  for multiregion problem
c
c
      program reload
      implicit real*4(A-H,O-Z)
c
      dimension ilftza(2,100),ida1(100),
     1  zadenf1(100,250),zadenf0(100,250)
c
      write(*,*)'MANUAL Fuel Adjustment - With Reload'
      write(*,*)' - uses atst.i and conc.out, changes atst.i'
      write(*,*)' '
c
 121  continue
      write(*,*)'input time (days) for fresh fuel reloading'
      write(*,*)' must be one identified in conc.out'
      write(*,*)' if 999, assumes no reload = '
      read(*,*)adjtim0
c
      if(adjtim0.eq.999)go to 101
      write(*,*)'input material number for fresh fuel reloading'
      write(*,*)' must be one identified in conc.out = '
      read(*,*)imat0
c
 101  write(*,*)'input time (days) for burned fuel'
      write(*,*)' must be one identified in conc.out'
      write(*,*)' if 999, assumes complete reload in same material = '
      read(*,*)adjtim1
c
      if(adjtim1.eq.999)go to 102
      write(*,*)'input material number for burned fuel'
      write(*,*)' must be one identified in conc.out = '
      read(*,*)imat1
c
      write(*,*)'input adjustment factor'
      write(*,*)'quantity of burned fuel to be left (0.0 to 1.0) = '
      read(*,*)factor
 102  if(adjtim1.eq.999)factor=0.0
      write(*,*)'input number of isotopes that will not be adjusted = '
      read(*,*)isonum
      if(isonum.eq.0)go to 10
      do 11 ij=1,isonum
      write(*,*)'input Z for isotope = ',ij,' = '
      read(*,*)ilftza((2*ij)-1,ij)
      write(*,*)'input A for isotope = ',ij,' = '
      read(*,*)ilftza((2*ij),ij)
  11  continue
  10  continue
c
c    zero all
      do 131 i=1,100
      ida1(i)=0
      do 132 j=1,250
      zadenf0(i,j)=0.0
      zadenf1(i,j)=0.0
 132  continue
 131  continue
c
c    read info from conc.out
      if(adjtim0.ne.999)call routcon(adjtim0,imat0,ida1,zadenf0)
      if(adjtim1.ne.999)call routcon(adjtim1,imat1,ida1,zadenf1)
      write(*,*)ida1(1)
      do 21 ij=1,ida1(1)
      iz=ida1(ij+1)/1000
      ia=ida1(ij+1) - 1000*(ida1(ij+1)/1000)
      write(*,*)ida1(ij+1),zadenf1(iz,ia),zadenf0(iz,ia)
      do 20 itst=1,isonum
      if(ilftza((2*itst-1),itst).eq.iz.and.ilftza((2*itst),itst).eq.ia)
     1 goto 21
  20  continue
      if(ia.eq.0)ia=250
      zadenf1(iz,ia)=zadenf1(iz,ia)*factor
     1    + zadenf0(iz,ia)*(1.0d0-factor)
      write(*,*)zadenf1(iz,ia)
  21  continue
c      copy results to prtinp
```

```
      if(adjtim1.eq.999)imat1=imat0
      call prtinp(imat1,ida1,zadenf1)
c
      write(*,*)' '
      write(*,*)' Repeat calculation for another material?'
      write(*,*)' (yes=1, no=0) = '
      read(*,*)igo
      if(igo.eq.1)go to 121
      stop
      end
c
cccccccccccccccccccccccccccccccccccccccccc
cccccccccccccccccccccccccccccccccccccccccc
c
c      subroutine routcon
c  for time adjtime and material imat,
c   reads zaic*zawt=zadenf from conc.out (unit=14)
      subroutine routcon(adjtime,imat,ida,zadenf)
      implicit real*4(a-h,o-z)
c
      dimension ida(100),zadenf(100,250)
      character*20 c$
c
c    zero all
      iswitch=0
      do 103 i=1,100
      ida(i)=0
      do 104 j=1,250
      zadenf(i,j)=0.0
 104  continue
 103  continue
c
      open(14,file='conc.out',status='old')
c
 101  continue
      read(14,111)c$
 111  format(a20)
      if(c$.ne.'                    ')go to 101
      read(14,*)ihcurnt, time
      if(time.eq.adjtime)then
c
 102  continue
      read(14,*,end=201)j,iz,ia,ac,ad
       if(j.eq.imat)then
       iswitch=1
       ida(1)=ida(1)+1
       ida(ida(1)+1)=iz*1000 + ia
       if(ia.eq.0)ia=250
       zadenf(iz,ia)=ad
c
       go to 102
       else
       if(iswitch.eq.1)go to 201
       go to 102
       endif
c
      else
c      ih=ih+1
      goto 101
      endif
c
 201  continue
      close(14)
      return
      end
c
cccccccccccccccccccccccccccccccccccccccccccccccccccccccccc
c
c  subroutine prtinp
c  writes new isotope densities to atst.i (unit=23)
      subroutine prtinp(imat,ida,zadenf)
c
      implicit real*4(a-h,o-z)
c
      dimension ida(100),zadenf(100,250)
      character*80 c$
      character*64 d$
      character*17 e$
      character*3 a$,t$
c
      open(23,file='atst.i',status='unknown')
      open(31,status='scratch',
     1 access='direct',form='formatted',recl=80)
c
c  copy atst.i to direct scratch file for processing
c   direct file can be changed per record without
c   loosing all subsequent data
```

```
c
      ic=0
 9200 read(23,9201,end=9202)c$
 9201 format(a80)
      ic=ic+1
      write(31,9201,rec=ic)c$
      go to 9200
 9202 continue
c
c  find isotopes and gm/cc for mod, clad, fuel
      rewind(31)
      density=0.0
      ic=0
c  find correct m card
 9001 ic=ic+1
      read(31,9002,rec=ic,err=9099)a$
 9002 format(a3)
       if(a$(1:1).eq.'m'.and.a$(2:2).ne.'t'.and.a$(2:2).ne.'o')then
       read(31,9050,rec=ic)j
 9050  format(t2,i3)
       if(j.eq.imat)go to 9003
       endif
      go to 9001
c
 9003 continue
 9004 read(31,9002,rec=ic)t$
      if(t$.eq.a$.or.t$.eq.'c  '.or.t$.eq.'  ')then
       if(t$.eq.'c  ')then
       ic=ic+1
       go to 9004
       endif
      read(31,9051,rec=ic)izacomp,zadens
 9051 format(t8,i5,t18,g14.7)
      read(31,9053,rec=ic)e$,x
 9053 format(t1,a17,t18,g14.7)
c      write(*,*)izacomp,zadens
      else
      go to 9099
      endif
c
c
      do 9102 k=1,ida(1)
      if(ida(k+1).eq.izacomp)go to 9110
 9102 continue
c  if can't find in ids or ida - stop
      write(*,*)' '
      write(*,*)'error - no z a identifed in libdata.in'
      write(*,*)'za = ',izacomp
      stop
 9110 continue
      iz=izacomp/1000
      ia=izacomp - 1000*(izacomp/1000)
      if(ia.eq.0)ia=250
      zadens=zadenf(iz,ia)
      if(zadens.eq.0.0.or.abs(zadens).lt.1.0e-24)zadens=-1.0e-24
      write(31,9052,rec=ic)e$,zadens
 9052 format(t1,a17,t18,g14.7)
c      write(*,*)t$,izacomp,zadens
      density=density+zadens
c
      ic=ic+1
      go to 9004
c
 9099 continue
c
c
c  change density value in cell card
c
      rewind(31)
      ic=0
c  find correct cell card with material id
 9061 ic=ic+1
      read(31,9002,rec=ic)a$
c      write(*,*)a$
       if(a$(1:3).eq.'c  '.or.a$(1:3).eq.'   ')then
       go to 9061
       endif
      read(31,9064,rec=ic)j,k
 9064 format(t1,i3,t6,i2)
c      write(*,*)j
      if(j.eq.200)go to 9070
      if(k.eq.0)go to 9061
      read(31,9063,rec=ic)j,k,oldens,d$
 9063 format(t1,i3,t6,i2,t9,f7.3,a64)
       if(k.eq.imat)then
       write(31,9063,rec=ic)j,k,density,d$
c       write(*,*)i,ic
```

```
c       write(*,*)j,k,oldens,density,d$
       endif
      go to 9061
c
 9070 continue
c
c  copy scratch file to atst.i
c
      rewind(31)
      rewind(23)
      ic=0
 9600 ic=ic+1
      read(31,9601,rec=ic,err=9602)c$
 9601 format(a80)
      write(23,9601)c$
c      write(*,*)c$
      go to 9600
 9602 continue
c
      close(23)
      close(31)
c
      return
      end
c
cccccccccccccccccccccccccccccccccccccccccccccccccccccccc
```

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