DKR-1100: A Univac 1100 Version of the DKR Radioactivity Code

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FUSION TECHNOLOGY INSTITUTE

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PREFACE

Since the original writing of the DKR code by Tak Yun Sung [Ref. 1, Appendix A], a number of modifications to the code were made by several graduate students attending the University of Wisconsin-Madison (Kevin Okula, Andrew Klein, Andrew White, Augusto Morelli and myself). This naturally led to the existence of a number of different DKR versions on the NMFECC (National Magnetic Fusion Energy Computing Center) and on the MACC (Madison Academic Computing Center) networks. Unfortunately, many of the modifications were never documented, resulting in old modifications becoming obscure as time passed.

This particular revision of the DKR code, called DKR-1100, is an attempt to establish some coherence among the different versions that now exist. The DKR-1100 version, as the name implies, is the version of the DKR code that now exists at the MACC site. Although many of the main computational subroutines are nearly identical to the NMFECC network version, the input to the DKR-1100 version is considerably different than that of the DKR versions existing on the NMFECC network (which among other things, has been adapted to the Lawrence Livermore CRAY computers). Thus for all the new users of the DKR-1100 code, "Happy Computing."

ABSTRACT

DKR is a point activity calculation code which constructs linear decay chains using nuclear data from Decay Chain Data Library (DCDLIB) and solves them to compute the activity of a fusion reactor. Transmutation data in the DCDLIB and neutron fluxes of a system are the essential inputs for this program.

The calculation of radioactivity, biological hazard potential (BHP), afterheat due to β- and γ-rays, and that due to β-rays only, is performed with the DKR code. A decay γ-ray source may also be produced as one of the optional outputs from DKR.

The photon transport calculation is performed with decay γ-ray sources at times after shutdown, or with adjoint sources (kerma of tissue) at a specified position. Detailed spatial afterheat is obtained from the γ-ray heating rate in the photon forward calculation and β-ray heating in the DKR results.

1. INTRODUCTION

Calculations of radioactivity and afterheat due to neutron activation are of great concern in a fusion reactor design. Evaluation of environmental impact, accident analysis, maintenance procedures, and to some extent the choice of blanket and shield materials depend on determining the radioactivity and afterheat.

Previous activation calculations performed in conjunction with fusion reactor designs [1-5] agree on the order of magnitude in radioactivity and afterheat, in general, but show wide variation in the radiological hazard which is sensitive to the concentration of each isotope. The differences observed in activity for various calculations result primarily from design differences such as size and composition, and the choice of material. However, it should be pointed out that there are no well-established methods for calculating radioactivity and afterheat of fusion reactors, and some of the discrepancies between various calculations are due to inconsistent procedures in using the nuclear data.

The activation of a nuclide can be represented by linear decay chains, as will be discussed later, which are solved accurately and efficiently using the recursion coefficient formula [6].

Recently, nuclear data systems have been improved considerably, in particular the expanded and updated ENDF/B nuclear data library is now available [7,8]. The ENDF/B-IV library provides a unified data format and updated nuclear data which makes it possible to describe the neutron and photon interactions reasonably well. Also, dosimetry files in ENDF/B-IV for radioisotopes are very useful for radioactivity studies. Consequently, it is desirable to construct the linear decay chains from the nuclear data based on ENDF/B-IV.

Based on these concepts, the DKR code has been written and a Decay Chain Data Library (DCDLIB) [9] has been compiled. Figure 1.1 shows the flowchart for the complete activity calculational scheme.

The secondary libraries for the neutron and photon transport calculations were processed by the AMPX modular system [10]. The reaction cross sections in ENDF/B and other complimentary sources were processed by the MACK program [11]. Prep [9] is the program for generating the DCDLIB. The reaction cross sections and the radioactive decay data have been compiled into the DCDLIB under the transmutation types shown in Table 1.1. Once DCDLIB is completed, it may be used for activity calculation until improvements in ENDF/B or the other data sources warrant its revision or expansion.

DKR is the major program in the activity calculation and is designed to construct and then solve the linear decay chains using nuclear data from DCDLIB, leading to the activity of a fusion reactor. The neutron flux from ANISN [12] and transmutation data from DCDLIB are the essential inputs for the DKR program. The calculation of radioactivity, BHP, and afterheat due to β- and γ-rays and that due to β-rays only is performed with the DKR code. Decay γ-ray sources are also produced as one of the outputs from DKR.

The photon transport calculation is performed with decay γ-ray sources at times after shutdown, or with adjoint sources (kerma of tissue) at a specified position using the one-dimensional transport code ANISN. Detailed spatial afterheat is obtained from the γ-ray heating rate in the ANISN forward calculation and β-ray heating in the DKR results.

2. CALCULATIONAL METHODS

The activity calculation in a fusion reactor is based on the transmutations of nuclides which are determined by their decay rate and/or reaction

TABLE 1.1. Definition of Transmutation Types

Transmutation Types Identified by an Integer KT

KT	Reaction Type†	Change in†† KZA	Change in†† LIS	ENDF/B-IV
1	Total Reaction			
2	(n,γ)	+ 1		MT = 102
3	(n,p)	-1000		103
4	$(n,2n)$	- 1		16
5	(n,d), $(n,n')p$	-1001		28,104
6	(n,t)	-1002		32,105
7	(n,He^3)	-2002		106
8	(n,α)	-2003		107
9	$(n,n)\alpha$	-2004		22
10	$(n,2n)\alpha$	-2005		24
11	$(n,2\alpha)$	-4007		108
12	$(n,2\alpha)t$	-5010		113
13	$(n,3\alpha)$	-6011		109
14	$(n,3\alpha)n$	-6012		23
15	$(n,n')*$		+1	
16	$(n,\gamma)*$	+ 1	+1	
17	$(n,2n)*$	- 1	+1	26
18	$(n,p)*$	-1000	+1	
19	$(n,d)*$, $n(np)*$	-1001	+1	
20	(to be assigned)			
21	Total Decay			
22	β^-	+1000		RTYP = 1.0
23	β^+, EC (=3)	-1000		2.0
24	α (=9)	-2004		4.0
25	γ		-1	3.0
26	$(\beta^-)*$	+1000	+1	
27	$(\beta^+)*$, (EC)* (=18)	-1000	+1	
28	n	- 1		

†the reaction type with * lead to the isomeric state

††KZA: (Z,A) number of a nuclide (= 1000•Z + A)

LIS: isomeric state of a nuclide

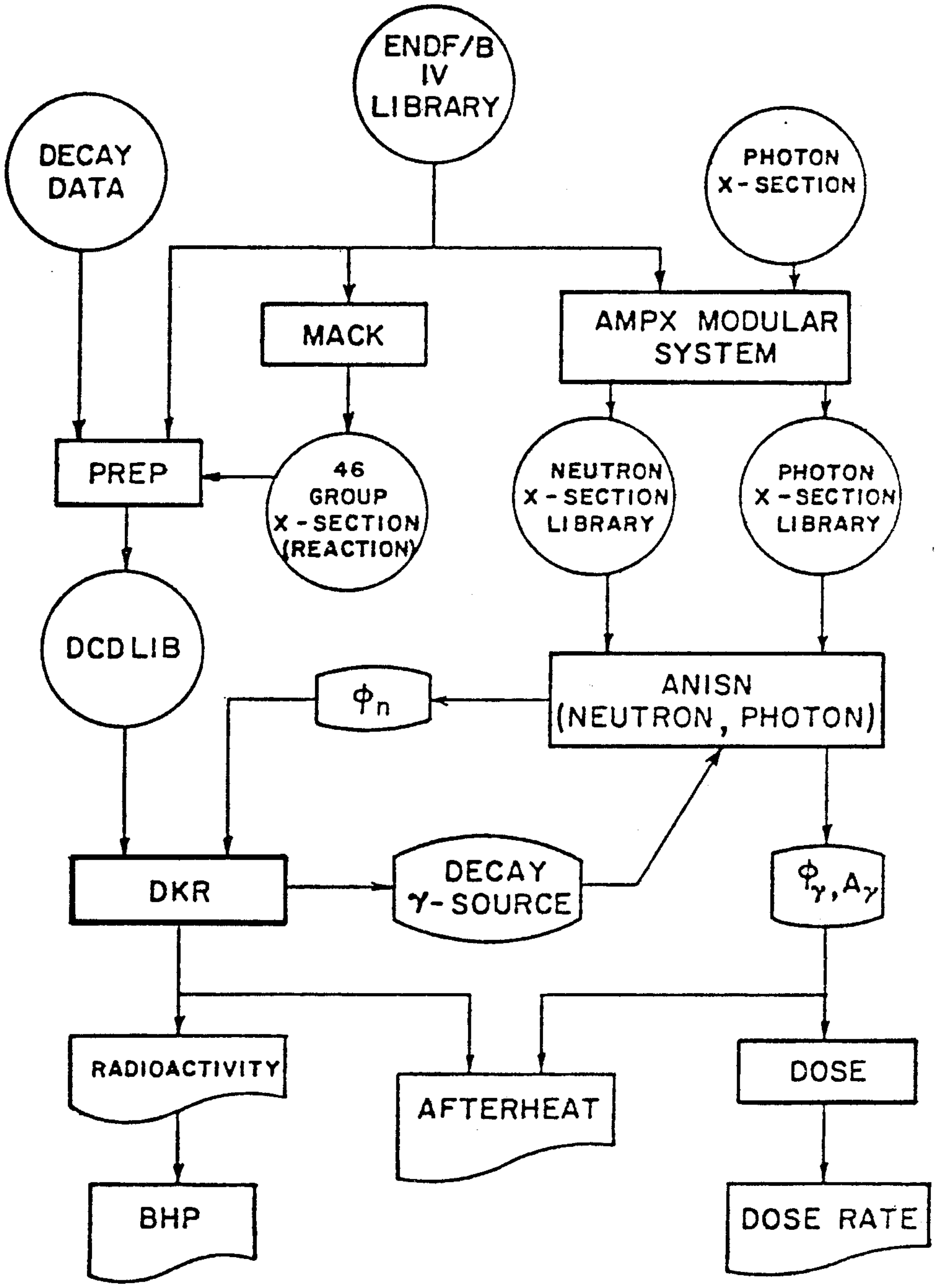


Fig. 1.1. Calculational scheme of activity in fusion reactor.

rate. A reaction rate is given by

$$A = (\sigma, \phi)$$

where σ is the reaction cross section which converts the scalar flux ϕ into a reaction rate of interest, and the symbol (,) indicates integration over all energies.

The reaction rate is spatially dependent while the decay constant of a nuclide is independent of any external influence. The concern in activity calculation is the transmutation of target nuclides. Most transmutation products do not move in the reactor blanket and shield, thus activation and transmutation are computed by a point-wise calculation while transport is calculated by a 1-D approximation. The main purpose of this section is to show calculational methods for determining the activity of a fusion reactor using linear decay chains.

2.1 Transmutation

Neutron induced reactions in the blanket and shield lead to transmutations of nuclides and some of the transmutation products are radioactive. The products, including radioactive ones, are also exposed to a neutron flux and they may transmute and/or decay out to result in other transmutations. Some may feed back to their precursors.

About twenty neutron reactions are possible when nuclei are bombarded by neutrons in the energy range below 20 MeV [13]. Among them, the (n,3n) reaction and (n,n't) reaction are energetically the most unfavorable and generally do not occur below about 15 MeV incident neutron energy, because of their high threshold energy. An important exception is the (n,n't) reaction of ^{7}Li which was treated as an $(n,n'\alpha)$ reaction in this work. Probable neutron reactions and radioactive decay processes of a nuclide in a fusion reactor are given in Fig. 2.1.

There are several reactions which result in the same transmutation product. For example, the nuclide with Z protons and N neutrons is transmuted into the nuclide with (Z - 1) protons and N neutrons by an (n,d), an (n,n'p), or an (n,p'n) reaction. The transmutation products cannot be distinguished by their transmutation history unless one reaction type is favored to form the product nuclei in a metastable state. Therefore, these reactions are the same from the transmutation point of view and they will be included in one transmutation process. Rearranged transmutation types were shown in Table 1.1.

Let N_k represent the number density of nuclide k and let the transmutation type be either an induced reaction or a radioactive decay; then the inventory of nuclide k can be calculated from the nuclear data information, i.e. reaction rates and decay rates.

The number density of nuclide k at one point is represented by a balance equation

$$\frac{dN_k}{dt} = \sum_j \gamma_j^k N_j - \lambda_k N_k - N_k \int_0^\infty \sigma_a^k \phi \, dE + Q \qquad (2.1)$$

where γ_j^k is the probability for a nuclide j forming a nuclide k per unit time, σ_a^k is the absorption cross section of nuclide k, and λ_k is the total decay constant of nuclide k.

It is necessary to construct reaction chains for every blanket material to solve the balance equations. A considerable amount of effort was made to construct chains which are important for the activity calculation, in order to

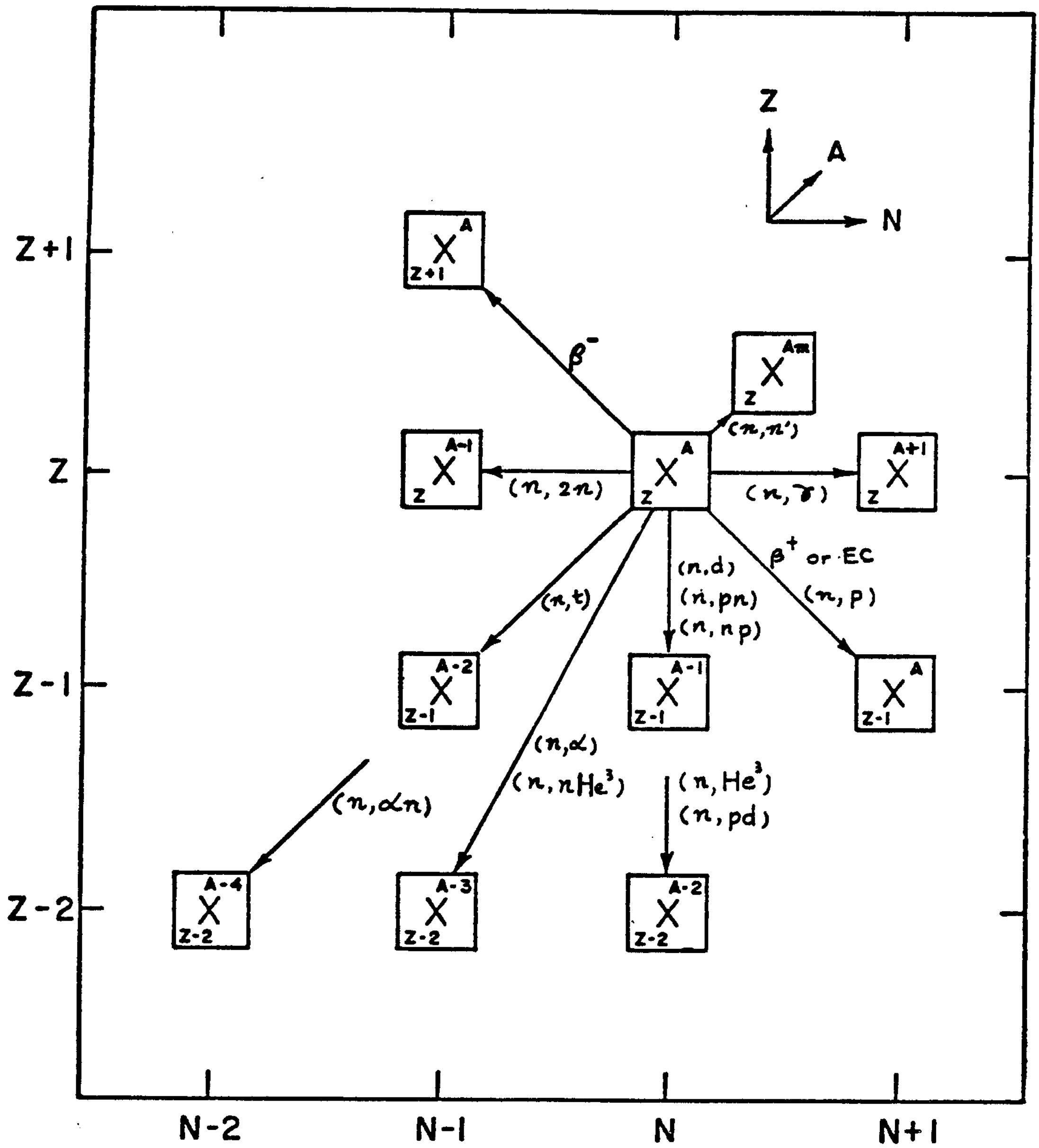


Fig. 2.1. Transmutation products by neutron reaction and by radioactive decay.

solve the corresponding balance equation efficiently. The activity has been calculated using the activation chains for each isotope assuming constant flux during the operating time. Also, special attention was given as to what branching ratio to use and the effect of these choices where large discrepancies exist in available data for cross sections and branching ratios.

The balance equation may be expressed by a matrix representation

$$\frac{d}{dt}\underline{N} = \underline{Q} - \underset{\sim}{B}\underline{N} \ . \tag{2.2}$$

Although this equation can be solved by the matrix exponential method [14,15], it is not always easy to solve, because there are several hundreds of nuclides, each of which can be produced and destroyed by one or more transmutation process. Thus a matrix whose order is sometimes unknown must be constructed to describe the problem exactly, and even if the order of matrix is known, it may be too large to be calculated efficiently.

Another problem is caused by a wide range in magnitude of coefficients of the matrix $\underset{\sim}{B}$, which can lead to meaningless solutions. The coefficient range must be restricted to make the matrix calculation possible and the time steps limited to those corresponding to the coefficient magnitude. Furthermore, too many zeroes in a matrix are undesirable because they cost computing time. To avoid these problems, a matrix $\underset{\sim}{B}$ which has finite dimensions and whose coefficients lie in a reasonable range must be constructed.

2.2 Linear Decay Chains

An effective method of calculating the inventory of nuclides using linear decay chains has been developed and applied in the DKR code [6].

The linear decay chains are constructed by taking all the possible linear paths so that the resolved chains show no branches. However, the work to construct the chains and to prepare chain data is too time-consuming to be repeated for each calculation.

Recently, nuclear data library systems have been improved considerably, especially that in ENDF/B-IV. Therefore, it is desirable to construct the linear decay chains from the nuclear data based on ENDF/B-IV. After the linear decay chains are constructed, they can be solved analytically. The DKR code has been developed with these concepts, and a Decay Chain Data Library (DCDLIB) for construction of chains has been compiled. The algorithm for formation of linear decay chains with nuclear data from DCDLIB will be discussed later.

In the linear decay chains, the balance equations become an ordered set of coupled differential equations. The number density of a nuclide is related only to that of a preceding nuclide, and can be written as

$$\frac{d}{dt} N_k = S_k + \gamma_{k-1}^{k} N_{k-1} - \beta_k N_k \qquad (2.3)$$

where S_k is the external source of the k^{th} nuclide, γ_{k-1}^{k} is the production rate of a nuclide from its precursor, and β_k is the destruction rate of the k^{th} nuclide in the linear decay chains. On the other hand, a loop is a chain where a nuclide leads to the production of itself by transmutation processes.

Each mode in a chain represents partial or whole nuclide concentrations. After the calculation of linear chains, the number density of a nuclide can be written as a sum of its partial concentrations.

The balance equation for the k^{th} nuclide in a linear decay chain without loops is

$$\frac{d}{dt} N_k = Q_k - \beta_k N_k \ , \tag{2.3}$$

where Q_k, the production rate of the k^{th} nuclide from the $(k - 1)^{th}$ nuclide and from the source, is given by

$$Q_k = \gamma_{k-1}^{k} N_{k-1} + S_k \ .$$

The solution of Eq. (2.3) is

$$N_k(t) = N_k(t_o) \ e^{-\beta_k (t-t_o)} + \int_{t_o}^{t} Q_k(t') \ e^{-\beta_k (t-t')} \ dt' \ . \tag{2.4}$$

The first term in Eq. (2.4) may be computed directly, but the second term which is related to the history of the transmutations and the source cannot. Since N_k and Q_k are a linear combination of exponentials, we can express the solution as

$$N_k(t) = \sum_{j=1}^{k} a_j^k \ e^{-\beta_j (t-t_o)} + \sum_{j=1}^{k} V_j(S_j, (t-t_o)) \tag{2.5}$$

where a_j^k is the coefficient associated with the exponential $e^{-\beta_j(t-t_o)}$ and $V_j(S_j,(t-t_o))$ is the term related to the external source.

Comparing Eqs. (2.4) and (2.5) for the case of no source, it is clear that

$$Q_k(t') = \gamma_{k-1}^{k} \sum_{j=1}^{k-1} a_j^{k-1} e^{-\beta_j(t'-t_o)} \tag{2.6}$$

where γ_{k-1}^{k} is the production rate of the k^{th} nuclide from the $(k - 1)^{th}$ nuclide. Substituting the expression for $Q_k(t')$ and $N_k(t)$ into the Eq. (2.5) with no source related terms,

$$N_k(t) = N_k(t_o) e^{-\beta_k(t-t_o)} + \gamma_{k-1}^{k} \sum_{j=1}^{k-1} \frac{a_j^{k-1}}{\beta_k - \beta_j} \left(e^{-\beta_j(t-t_o)} - e^{-\beta_k(t-t_o)}\right) . \tag{2.7}$$

Rearranging Eq. (2.7) yields

$$N_k(t) = \left(N_k(t_o) - \gamma_{k-1}^{k} \sum_{j=1}^{k-1} \frac{a_j^{k-1}}{\beta_k - \beta_j}\right) e^{-\beta_k(t-t_o)} + \gamma_{k-1}^{k} \sum_{j=1}^{k-1} \frac{a_j^{k-1}}{\beta_k - \beta_j} e^{-\beta_j(t-t_o)} \tag{2.8}$$

and

$$a_k^k = N_k(t_o) - \gamma_{k-1}^{k} \sum_{j=1}^{k-1} \frac{a_j^{k-1}}{\beta_k - \beta_j} , \tag{2.9}$$

$$a_j^k = \gamma_{k-1}^{k} \frac{a_j^{k-1}}{\beta_k - \beta_j} , \qquad j = 1, 2, \ldots, k-1 . \tag{2.10}$$

Therefore, the coefficients can be computed from the preceding coefficients successively.

NOTE: The linear decay chains were originally solved by Eqs. (2.8) through (2.10). Due to numerical roundoff errors occurring whenever the magnitude of two destruction rates differed by 10^9 or more (which lead to some negative number densities in the third rank), the linear decay chains are now solved by Eq. (2.7). The numerical stability is improved but negative number densities may occasionally still be encountered in the fourth and higher ranks.

A special case occurs when β_k is equal to β_j, or β_k is close to β_j. The first case occurs frequently when a loop in a linear chain is expanded linearly and the other case occurs when the destruction rates of two or more nuclides in a chain are accidentally very close. However, both cases can be treated as one case $\beta_k \approx \beta_j$ to preserve the simplicity of the recursion coefficient formula by keeping k linear combination of exponentials. A destruction rate is either a reaction rate or a decay rate, or sometimes the sum of both.

The recursion formula in Eq. (2.9) or (2.10) cannot be used in this case because of the singularity. Going back to Eq. (2.7) and considering the case where β_k is very close to β_j $(j \neq k)$,

$$N_k(t) = N_k(t_o)\, e^{-\beta_k(t-t_o)} + \gamma_{k-1}^{k} \sum_{j=1}^{k-1} a_j^{k-1} \left[\frac{e^{-\beta_j(t-t_o)} - e^{-\beta_k(t-t_o)}}{\beta_k - \beta_j}\right] ; \qquad (2.11)$$

the quantity in the brackets becomes

$$\frac{e^{-\beta_j(t-t_o)} - e^{-\beta_k(t-t_o)}}{\beta_k - \beta_j} = (t - t_o)\cdot e^{-\beta_j(t-t_o)} \left[\frac{1 - e^{\beta_j(t-t_o)-\beta_k(t-t_o)}}{\beta_k(t - t_o) - \beta_j(t - t_o)}\right] \qquad (2.12)$$

$$= (t - t_o)\cdot e^{-\beta_j(t-t_o)} \sum_{n=1}^{\infty} \frac{[\beta_j(t - t_o) - \beta_k(t - t_o)]^{n-1}}{n!}$$

Substituting Eq. (2.12) into Eq. (2.11) results

$$N_k(t) = N_k(t_o)\, e^{-\beta_k(t-t_o)} + \gamma_{k-1}^{k} \sum_{\substack{j=1 \\ \beta_k \neq \beta_j}}^{k-1} a_j^{k-1} \left[\frac{e^{-\beta_j(t-t_o)} - e^{-\beta_k(t-t_o)}}{\beta_k - \beta_j}\right] \tag{2.13}$$

$$+ \gamma_{k-1}^{k} \sum_{\substack{k=1 \\ \beta_k \simeq \beta_j}}^{k-1} a_j^{k-1}(t - t_o)\cdot e^{-\beta_j(t-t_o)} \sum_{n=1}^{\infty} \frac{[\beta_j(t - t_o) - \beta_k(t - t_o)]^{n-1}}{n!} .$$

The modified recursion coefficients are

$$a_k^k = N_k(t_o) - \gamma_{k-1}^{k} \sum_{\substack{j=1 \\ \beta_k \neq \beta_j}}^{k-1} \frac{a_j^{k-1}}{\beta_k - \beta_j} \tag{2.14a}$$

$$a_j^k = \gamma_{k-1}^{k} \frac{a_j^{k-1}}{\beta_k - \beta_j} , \qquad \beta_k \neq \beta_j \tag{2.14b}$$

$$a_j^k = \gamma_{k-1}^{k}\, a_j^{k-1}\cdot(t - t_o) \sum_{n=1}^{\infty} \frac{[\beta_j(t - t_o) - \beta_k(t - t_o)]^{n-1}}{n!} , \; \beta_k \simeq \beta_j . \tag{2.14c}$$

When an external source is included in a chain, we have to compute the $V_j(S_j,(t-t_o))$ term in Eq. (2.5), which may be treated in the same way except S_j is assumed to come from another precursor. With the following convention $\beta_{j-1} = 0$, $\gamma_{j-1}^{j} = 1$, and

$$b_{j-1}^{j} = S_j ,$$

we get

$$V_j(S_j,(t-t_o)) = \sum_{i=j-1}^{k} b_i^k\, e^{-\beta_i(t-t_o)} \tag{2.15}$$

where the coefficients are

$$b_k^k = -\gamma_{k-1}^k \sum_{\substack{i=j-1 \\ \beta_k \neq \beta_i}}^{k-1} \frac{b_i^{k-1}}{\beta_k - \beta_i} \tag{2.16a}$$

$$b_i^k = \gamma_{k-1}^k \frac{b_i^{k-1}}{\beta_k - \beta_i} , \qquad \beta_k \neq \beta_i \tag{2.16b}$$

$$b_i^k = \gamma_{k-1}^k \, b_i^{k-1}(t - t_o) \sum_{n=1}^{\infty} \frac{[\beta_i(t - t_o) - \beta_k(t - t_o)]^{n-1}}{n!} , \; \beta_k \approx \beta_i . \tag{2.16c}$$

If β_k is very close to β_i, b_i^k is computed in the same way as a_j^k in Eq. (2.14). When solving the chains, it is not always easy to choose an appropriate form from Eqs. (2.14) and (2.16) for small $(\beta_k - \beta_j)$ or $(\beta_k - \beta_i)$, and the method of choosing a right form will be analyzed in the next section.

Thus, the solution of the balance equation (2.3) is

$$N_k(t) = \sum_{j=1}^{k} [a_j^k \, e^{-\beta_j(t-t_o)} + \sum_{i=j-1}^{k} b_i^k \, e^{-\beta_i(t-t_o)}] \tag{2.17}$$

where coefficients a_j^k and b_i^k are defined in Eqs. (2.14) and (2.16), respectively. The empty summation for i = 0 is defined as zero.

NOTE: The solutions to the various approximations given by Eqs. (2.11) through (2.17) are not programmed within the code.

In a fusion reactor, it is usual to have no external source in a chain or at most, a very few. Even considering external sources in the system, the recursion coefficient formula for a linear chain is effective in solving the linear chains and preserves the concise form of Eq. (2.17).

Either the decay rate λ_{k-1}^{k} for a radioisotope, or reaction rate $A_{k-1}^{k} = \int_{0}^{\infty} \sigma_{k-1}^{k} \phi \, dE$ for a stable nuclide dominates the production rate of the k^{th} nuclide in most cases. But it is not uncommon to observe the case of the two processes competing with each other.

A loop occurring in a chain may be solved by matrix transformation methods, or by Laplace transform methods. The recursion coefficient formula can also be used if the loop is expanded in a linear chain truncating the higher terms. A loop occurs when the $(k + n)^{th}$ nuclide feeds back to the k^{th} nuclide in a chain. Important cases frequently met in a fusion reactor are a (n,p) reaction followed by a β^- decay or a (n,2n) reaction followed by a (n,γ) reaction.

To be able to calculate the activation due to a small time pulse as encountered in Inertial Confinement Fusion (ICF) reactors, special treatment is given for the case where the irradiation time t is less than 10^{-3} seconds. For most nuclides considered, this corresponds to a fluence (destruction rate x time) of $(\sigma\phi)\cdot t \ll 10^{-3}$. Beginning with Eq. (2.7) and considering the case where both $\beta_k t$ and $\beta_j t$ are $\ll 1.0$,

$$N_k(t) = N_k(0)\, e^{-\beta_k t} + \gamma_{k-1}^{k} \sum_{j=1}^{k-1} a_j^{k-1} \left[\frac{e^{-\beta_j t} - e^{-\beta_k t}}{\beta_k - \beta_j}\right] , \qquad (2.18)$$

the exponential terms can be expanded as

$$e^{-x} = 1 - x + \frac{x^2}{2!} - \frac{x^3}{3!} \cdots . \qquad (2.19)$$

Substituting the expansion into Eq. (2.18) results in

$$N_k(t) = N_k(0)\ (1 - \beta_k t + \frac{1}{2!}\ (\beta_k t)^2 - \ldots) \tag{2.20}$$

$$+ \gamma_{k-1}^{k} \sum_{k=1}^{k-1} a_j^{k-1} \left[\frac{(1 - \beta_j t + \frac{1}{2!}\ (\beta_j t)^2 - \ldots) - (1 - \beta_k t + \frac{1}{2!} - \ldots)}{\beta_k - \beta_j}\right]$$

$$N_k(t) = N_k(0)(1 - \beta_k t + \frac{1}{2!}\ (\beta_k t)^2 - \ldots) \tag{2.21}$$

$$+ \gamma_{k-1}^{k} \sum_{j=1}^{k-1} a_j^{k-1} \left[\frac{(\beta_k - \beta_j)t - \frac{t^2}{2!}\ (\beta_k^2 - \beta_j^2) + \frac{t^3}{3!}\ (\beta_k^3 - \beta_j^3) - \ldots}{\beta_k - \beta_j}\right]$$

Assuming $N_k(0) = 0$ for $k = 2, n$, Eq. (2.21) reduces to

$$N_1(t) = N_1(0)\ (1 - \beta_1 t)$$

$$N_k(0) = N_1(0)\ (\prod_{n=2}^{k} \gamma_n^{n-1})(\frac{t^{k-1}}{(k-1)!})[1 - \frac{t}{k}\ (\sum_{m=1}^{k} \beta_m)] \tag{2.22}$$

2.3 Radioactivity

Once the number density of nuclides is calculated at shutdown, the number density of any radionuclide k is calculated again by a recursion coefficient formula

$$N_k(\underline{r},t) = \sum_{j=1}^{n} a_j^n\ e^{-\beta_j t} \tag{2.23}$$

where t is the after shutdown time and n is the counting number of successive radioactive steps to nuclide k in the chain under consideration. Now the time dependent radioactivity after shutdown is given by

$$R(t) = \int_{\underline{r}} \sum_{\substack{k:\text{all} \\ \text{radioisotopes}}} \lambda_k N_k(\underline{r},t)\, d\underline{r} \tag{2.24}$$

where the integration is over the volume of interest.

In fusion reactors the successive radioactive decay steps in a chain are fewer than those of fission reactors, because the neutron reaction products in fusion reactors are only slightly displaced from the stability line of nuclides.

2.4 Biological Hazard Potential (BHP)

It is well known that the radiological hazard from radioisotopes cannot be estimated by the number of disintegrations in a given time only. The half-life of radioisotope, the type of decay particle and its energy, the dispersion rate of decay particle through the environment and its biological effect to the critical organ in a human body is also important. Among the many quantities which have been used to try and estimate the radiological hazard more accurately, BHP has been widely used in fusion reactor studies.

The BHP is defined as the ratio of radioactivity to the maximum permissible concentration (MPC) for a single isotope, and is interpreted as the volume of air or water that would be required to dilute the given inventory of radionuclide to its MPC value with the assumption of total release and uniform dispersion from the reactor [16]. However, it would be sensible to use BHP with consideration of volatilities and solubilities of the material under various conditions, because MPC values are related to the internal radiation in the human body.

The BHP of a given system is

$$B(t) = \int_{\underline{r}} \sum_{\substack{k:\text{all} \\ \text{radioisotopes}}} \xi_k \lambda_k N_k(\underline{r},t)\, d\underline{r} \tag{2.25}$$

where ξ_k refers to a BHP weighting function for nuclide k, which is the inverse of MPC for radioisotope k.

2.5 Afterheat

The afterheat of a fusion reactor can be divided into two parts: one due to heating by gamma rays and the other due to heating by decaying particles other than gamma.

The major reasons for separating gamma ray heating from other contributors to decay heating are: first, to get a realistic spatial afterheat without assuming γ-ray energy deposition in its birthplace; and secondly to apply a decay γ-ray source to a dose rate calculation directly. However, it should be noted that a total afterheat treatment of blanket and shield without a gamma transport calculation will give a realistic value because the γ-ray leakage at the boundary is small. The assumption that the energy or particles other than γ-rays are deposited at the point of production is still valid because of their short range in reactor materials.

Thus, the afterheat is given by

$$H(t) = H_\gamma(t) + \int_{\underline{r}} \sum_{\substack{k:\text{all} \\ \text{radioisotopes}}} \bar{E}_k \lambda_k N_k(\underline{r},t)\, d\underline{r} \tag{2.26}$$

where $\bar{E}_k$ is the average energy of a decay particle, which is zero in an isomeric transition case. The gamma flux is computed from the gamma transport equation given by

$$L\phi_\gamma = \Omega \ . \tag{2.27}$$

Ω is the number of photons produced per second by radioactive decay, and in the multigroup approximation the group source Ω_g is

$$\Omega_g(\underline{r},t) = \sum_{\substack{k:\text{all} \\ \text{radioisotopes}}} y_g^k \lambda_k N_k(\underline{r},t) \tag{2.28}$$

where y_g^k is the gamma yield in the gth group by the decay of the nuclide k.

Decay gamma heating is given by

$$H_\gamma(t) = \int_{\underline{r}} \sum_{\ell} N_\ell(\underline{r}) \int_0^\infty K_\ell(\underline{r},E)\ \phi_\gamma(\underline{r},E,t)\ dEd\underline{r} \tag{2.29}$$

where K_ℓ is the fluence-to-kerma factor [17] and N_ℓ is the number density for element ℓ. In the gamma transport calculation, we need only nuclear data for each element, not for every isotope considered.

2.6 Dose Rate

The calculation of the dose rate is not unfamiliar in reactor engineering. The average dose rate is given by

$$D(t) = \frac{\int_{\underline{r}} \Gamma\phi_\gamma(\underline{r},t)\ d\underline{r}}{\int_{\underline{r}} d\underline{r}} \tag{2.30}$$

where Γ is the flux-to-dose conversion factor and the integral is over the tissue volume of interest. Only external γ-rays are considered when dose is calculated for whole body near or inside the reactor, because of the short range of other particle radiation. A calculation of dose rate is performed by substituting tissue equivalent material at a position of interest.

Using the variational method, a simple calculation for the dose rate functional I_d is

$$I_D = (\Gamma,\phi_\gamma) - (\phi^*_\gamma,\Delta L_{\phi_\gamma}) \tag{2.31}$$

where I_d is accurate to second order. ϕ^*_γ is calculated from

$$L^*\phi_\gamma = \Gamma \tag{2.32}$$

for tissue outside the reactor. ϕ_γ is extrapolated to the outside of the reactor, as is ϕ^*_γ.

3. COMPUTER IMPLEMENTATION AND NUCLEAR DATA

3.1 Computer Implementation

According to the computational scheme (Fig. 1.1), DKR constructs the linear decay chains and computes the activity with these chains. Several considerations in constructing the chains and in employing the computational methods in the algorithm of DKR are to be discussed in this section.

The establishment of the coupled decay chain data formats for the reaction cross sections and the radioactive decay data makes it possible to construct decay chains directly from the nuclear data in DCDLIB. A decay chain begins at a nuclide which is a constituent of the blanket or shield structure and it terminates at a nuclide which has no data, or whose contribution to the total activity of blanket or shield is negligible. Although it is very difficult to terminate the chains before calculating the activity, selecting and truncating the appropriate chains before a calculation is essential to save time and money.

After all input data is read and stored by the DKR program, the nuclear data from DCDLIB and the neutron flux are used to construct the chain data table which includes possible transmutation types and transmutation rates at each spatial point including a reference one.

The linear decay chains are constructed for each material of the system in the way shown in Fig. 3.1. The chain data table is searched to find the nuclear data for the isotope considered. If there is no such data, the chains initiated by that isotope do not exist. If data is found, the transmutation types and the reference transmutation rates are taken as well as reaction products, and temporary chains of two steps are constructed with them. Next, each temporary chain is examined to determine whether it should be continued or terminated. For the chain which is not terminated by the chain tests, another search for the last step nuclide is taken to add the new step to the chain. This procedure continues until all chains are terminated either by lack of data or by failing the tests. A terminated chain is catalogued in the decay chain file if it includes any radioisotopes. Although there are several ways to establish criteria of continuing or terminating chains, two schemes are applied in sequence in the DKR program.

The first test is the number of stable nuclides to be included in a chain. The chains initiated by a major component isotope in the blanket and shield are allowed to include more chain steps than the chains initiated by an impurity in the system. For an impurity only primary reactions are important, so the number of stable nuclides in a chain is usually restricted by two. For the more important nuclides in the system, the number of stable nuclides in a chain could be increased to as many as five. However, this number can be easily modified by simple correction statements in the program. This test is very simple to apply, but effective in saving computing time in constructing the chains.

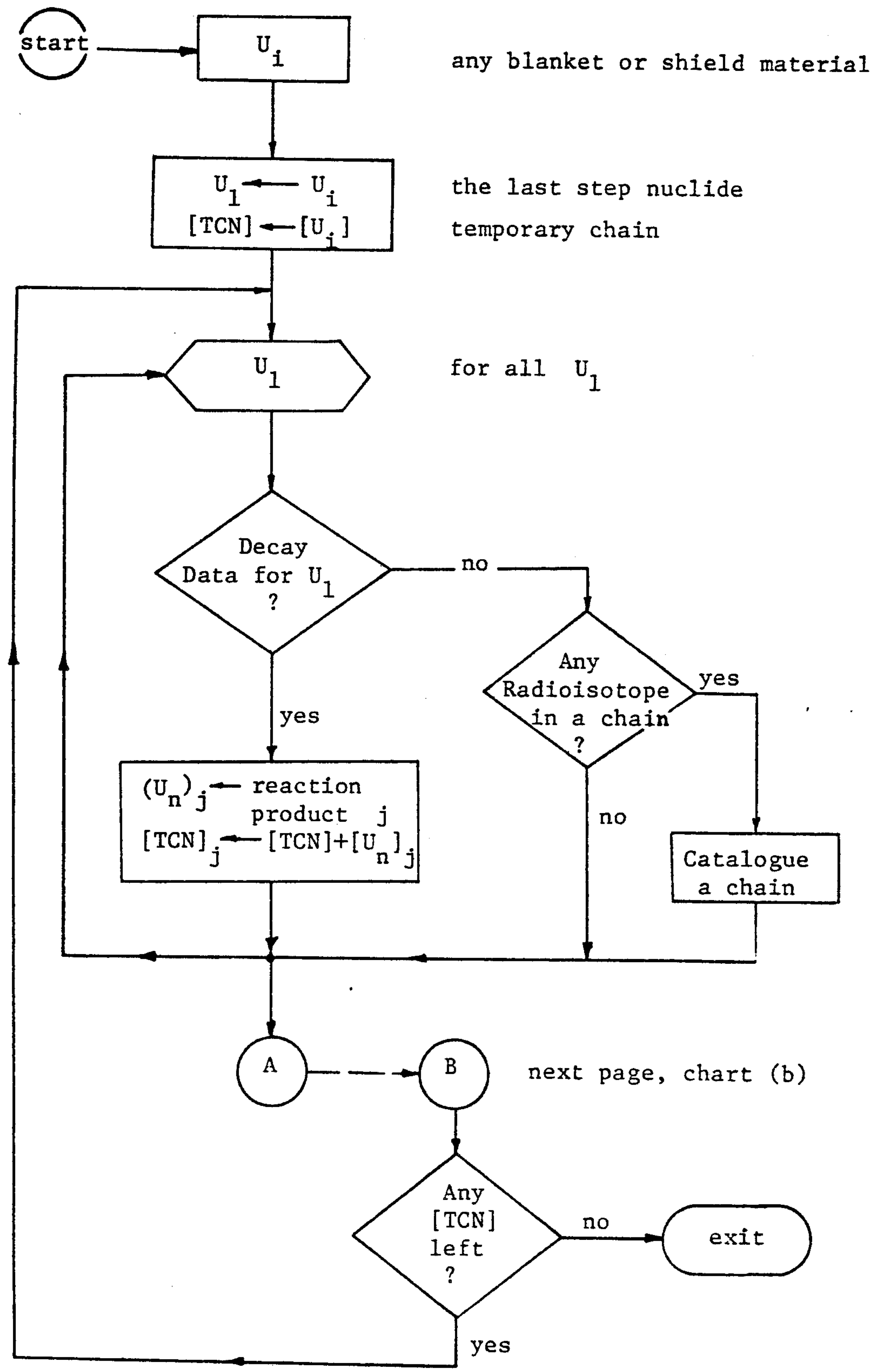
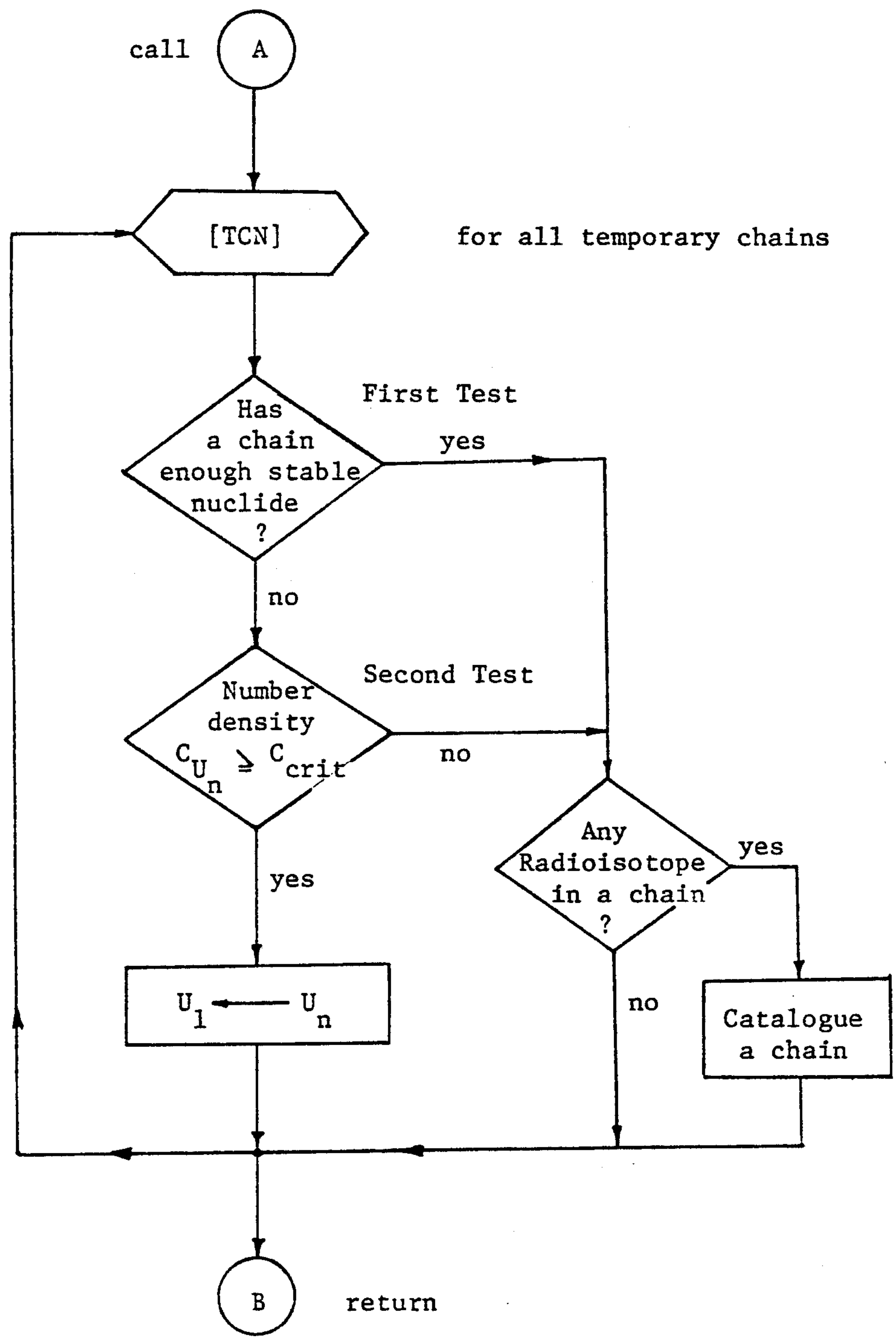


Fig. 3.1. Flowchart for calculating chains.

Fig. 3-1. (b)



call
A
[TCN]
for all temporary chains
First Test
Has a chain enough stable nuclide ?
yes
no
Second Test
Number density $C_{U_n} \geq C_{crit}$
no
yes
Any Radioisotope in a chain ?
yes
no
$U_1 \leftarrow U_n$
Catalogue a chain
B
return

If a chain is not terminated by the first test, another test is made, in which the number density of the last nuclide in a chain is calculated. If the last nuclide in a chain is radioactive, the chain is exempted from this test. One advantage of testing the number densities of stable nuclides instead of those of radioisotopes is the ease of applying the criteria and the flexibility of changing the test criteria.

The reference flux used in this test was taken from the first wall flux of UWMAK-I [1], which is stored in the program. There is also an option to use this flux or to supply another set of reference fluxes fit for the problem. Usually the first wall and other blanket structures are designed to be replaced after a few years because of radiation damage to material. Consequently, a reference operating time is set to 10^8 seconds, about 3.2 years. (NOTE: This has been changed. Users can now specify the reference operation time.) If the number density of a stable reaction product after continuous operation of 10^8 seconds at a UWMAK-I first wall flux is less than a preset number density criterion, the chain is terminated. 10 appm of the product nuclide is set as the number density criterion, but this can also be modified easily.

Unlike the radioactive chains in a fission reactor, the chains in a fusion reactor are relatively short, even if we try to keep all consecutive radioisotopes in the chains. This is another reason to apply the number density test only to a stable isotope in the chain. Also, it should be noted that radioisotopes with large cross sections build up their activity after shutdown. But applying the test to stable nuclides only eliminates this kind of difficulty.

The next problem encountered in constructing the linear decay chain is how to treat a loop in a chain. A loop may be linearized, but the resulting infinite series in the chain must be terminated and a truncation error occurs. A loop may also be solved exactly by matrix transformation methods, which may affect the solution of other chains which share the same initial isotope as in the loop chain because the number density of the initial nuclide does not depend on its destruction rate only. If the exact solution of a loop is fed back into other chains, time intervals for a solution feedback should be considered. However, considering computing time and effort in calculation, the feedback of a loop solution need not be necessary in an activity calculation, because the transmutation of the original component in the system does not exceed a few percent as shown in the UWMAK-I study [18]. Thus when the next nuclide directly feeds back to its precursor, the chain takes the exact loop solution. Otherwise the loop is expanded as a linear decay chain and solved by a recursion coefficient formula. In either case, no feedback solution to other chains is considered.

Another problem considered is the loss of accuracy in calculations. It is well known that great loss of accuracy usually occurs when two numbers close to one another in size are subtracted. In a linear chain, the difference in destruction rates may be large enough to allow one destruction rate to be neglected compared to another, as in the case of the short half-life radioisotope preceded by a stable isotope. However, destruction rates of the same order of magnitude are not rare, and they not only lead to a loss of accuracy, but sometimes make it impossible to get a solution because of the singularity.

During formulation of the recursion formula, the recursion coefficients were modified to take care of this loss of accuracy. However, the infinite series in Eqs. (2.14c) and (2.16c) should be truncated after a few terms for an efficient computation. The error involved in cutting the series after the first three terms will be analyzed by using Taylor's formula with remainder [19]. The expression for an exponential is

$$\frac{1 - e^{-x}}{x} = 1 - \frac{x}{2!} + \frac{x^2}{3!} + \frac{(-x)^n e^{-\xi}}{(n + 1)!} , \qquad n = 3 \tag{3.1}$$

for some ξ between 0 and x, where the $\frac{x^n e^{-\xi}}{(n + 1)!}$ term represents the error bound. If only the first three terms were kept to have an error less than 10^{-6}, then x must be less than 0.02885. Thus, whenever $(t - t_0)(\beta_k - \beta_j)$ is less than 0.01, the expressions of Eqs. (2.14c) and (2.16c) are applied to the computer algorithm instead those of Eqs. (2.14b) and (2.16b) in Section 2. Table 3.1 shows the error range due to this approximation.

NOTE: The approximation given in Eq. (3.1) is not programmed within the code.

To keep the necessary accuracy in the calculation, double precision arithmetic was applied to the translation of the formulae in the program. Initial number densities of nuclides, destruction rates and production rates as well as recursion coefficients are defined and computed in double precision and final computed number densities are converted into single precision numbers. However, it is unnecessary to use double precision in all calculations, because of machine time costs and the fact that other calculations in the program are straightforward arithmetic computations.

TABLE 3.1. Error Range Due to Approximation

$$\frac{1 - e^{-x}}{x} = 1 - \frac{x}{2!} + \frac{x^2}{3!} + \ldots + (-1)^n \frac{x^n}{(n + 1)!} = \sum_{n=1}^{\infty} \frac{(-x)^{n-1}}{n!}$$

x	$A = \frac{1 - e^{-x}}{x}$	$B = 1 - \frac{x}{2!} + \frac{x^2}{3!}$	$\frac{B - A}{A} \times 100(\%)$
0.1	0.95162582	$\underline{0.9516}6667$	0.43×10^{-2}
0.05	0.97541151	$\underline{0.97541}667$	0.53×10^{-3}
0.01	0.99501663	$\underline{0.9950166}7$	0.40×10^{-5}
0.005	0.99750416	$\underline{0.9975041}7$	0.10×10^{-5}
0.001	0.99950020	$\underline{0.999500}17$	-0.30×10^{-5}
0.0005	0.99975020	$\underline{0.999750}04$	-0.16×10^{-5}
0.0001	0.99995000	$\underline{0.99995000}$	---

3.2 Decay Chain Decay Library (DCDLIB)

The DCDLIB [9] is a concise library containing necessary nuclear data information for use in the fusion activity studies.

Reaction cross sections obtained from ENDF/B-IV and the calculated cross section library [20] were processed into 25 group cross sections using the MACK program. The calculated BNL cross section library is especially helpful because it includes cross section data for many isotopes including radioactive ones. BNL-325 [21] was also referred to when no reaction cross sections were available elsewhere.

The radioactive decay data was taken either from ENDF/B-IV, or from the Table of Isotopes [22] for nuclides not in the ENDF/B-IV library. The maximum permissible concentration (MPC) values for radioisotopes not covered in the NRC regulations [23] were estimated based on the decay mode, decay energies and intensities [24,25].

One of the quantities needed to calculate the transmutation process is the branching ratio to a metastable state of a reaction product. Reactions whose branching ratios are not well known may introduce significant uncertainties in activity calculations. The branching ratios in the DCDLIB are generally taken from previous work [24-26] or from estimates based on the isomeric energy state in a nuclide and the threshold energy of reactions.

The DCDLIB formats were designed to include all the information for activation studies, and the reaction cross sections and radioactive decay data were processed into the DCDLIB in couple transmutation types of Table 1.1. All the data necessary for constructing decay chains in the DKR code is stored in DCDLIB. It can also be used as a tentative reaction data library. The list of data in the DCDLIB is given below.

The data for all stable and radioactive nuclides include,

initial number densities,

reaction cross sections,

reaction products, and

branching ratios to isomeric states.

The data for radioactive nuclides include,

decay constants,

decay modes and decay products,

average energies of emitted particles, and

MPC values.

4. DESCRIPTION OF INPUT AND OUTPUT

4.1 Input Data

A brief description of the input data is given below. It is intended to serve as a guide for input data preparation.

Card No. 1 (18A4)

title card

Card No. 2 (12I6)

LID	1-6	identification number
LNK	7-12	program execution option 0: construction of linear decay chains only 1: calculation of radioactivity related parameters 2: same as LNK=1 except the decay chains and destruction data tables from preceding runs are used 3: generation of decay gamma source with the calculation of radioactivity related parameters 4: same as LNK=3 except the destruction table and decay chains from preceding runs are used
LGE	13-18	geometry 1: slab 2: cylinder 3: sphere 4: torus

LFX	19-24	flux format description 1: DKR format flux 2: ANISN scalar format flux
IZM	25-30	number of zones
INT	31-36	number of intervals
NOP	37-42	number of operating times 0: nine built-in operations times are used (Table 4.2) n: user supplied operations times on card 9 ($n \leq 9$)
NAS	43-48	number of after shutdown times 0: twelve built-in after shutdown times are used (Table 4.2) n: user supplied after shutdown times on card 10 ($n \leq 12$)
NNC	49-54	number of nuclides in the system
NCMP	55-60	number of composition tables
IGN	61-66	number of neutron energy groups
IGG	67-72	number of gamma energy groups

Card No. 3 (8I6)

LPRT1	1-6	print option for radioactivity 0: no effect 1: print zonewise radioactivity, afterheat, and BHP; print specific radioactivity of first interval in the first zone (usually first wall)
LPRT2	7-12	print option for index file 0: no effect 1: print index file for nuclides
LPRT3	13-18	print option for radioactivity of each interval 0: no effect 1: print radioactivity for each interval
LPRT4	19-24	print option for chain results 0: no effect 1: print chain results
LFLX	25-30	reference flux option for chain calculation 0: uniform flat flux of 10^{14} n/cm^2-s is used 1: the first wall flux of the UWMAK-I design is used 2: reference flux set is supplied on cards 11 and 12 by user

LFCF	31-36	flag for FFCF (Flux Conversion Factor) 0: FCF is calculated within the code 1: FCF is supplied on card 4 by user
LCLPS	37-42	flag for flux collapsing 0: no collapsing 1: flux collapsing
KZEND	43-48	last interval of plasma; used by code when LCLPS=1

<u>Card No. 4 (5F12.3)</u>

WLLD	1-12	neutron wall loading in MW/m^2
HTN	13-24	neutron heating in MeV
HTG	25-36	gamma ray heating in MeV
HTT	37-48	total nuclear heating in MeV
FCF	49-60	flux conversion factor; if LFCF=1, FCF value other than zero must be given

<u>Card No. 5 (5F12.3)</u>

RRP	1-12	plasma radius in cm
RRW	13-24	first wall radius in cm
RRT	25-36	torus radius in cm, if LGE=4
ACUT	37-48	ratio of stable nuclide in the decay chain to the initial input nuclide; used in the chain construction procedures
TIR	49-60	time of irradiation in seconds; used in the chain construction procedures

<u>Card No. 6 (3I6,6X,F9.2,3X,F9.2,I6)</u>

As many cards as IZM are required

IZ	1-6	zone number
NZI	7-12	number of intervals in a zone
LCAL	13-18	flag for zone radioactivity calculation 0: no radioactivity calculation 1: radioactivity calculation
RRI	25-33	inner radius of zone in cm

RRO	34-45	outer radius of zone in cm
LCP	46-51	number of intervals collapsed into one (NOTE: only input if LCLPS=1)

Card No. 7 (12F6.2)

As many cards as NCMP are required

CMP(1)	1-6	first zone composition
CMP(2)	7-12	second zone composition
CMP(IZM)		last zone composition

Card No. 8 (3I6,E12.3)

As many cards as NNC are required

LCMP	1-6	composition table number to be referred
KZA	7-12	nuclide ID number
LKUT	13-18	priority number of a nuclide 1: primary 2: auxiliary 3: impurities 4: negligible impurities
WND	19-30	number density of a nuclide

Card No. 9 (A6,E12.3)

As many cards as NOP are required only if NOP≠0

BOP	1-6	alphanumeric expression for an operating time
TOP	7-18	operating time in seconds

Card No. 19 (A6,E12.3)

As many cards as NAS are required only if NAS≠0

BAS	1-6	alphanumeric expression for an after shutdown time
TAS	7-18	after shutdown time in seconds

Card No. 11 (18A4)

Title card for reference flux and is given only if LFLX=2

Card No. 12 (6E12.3)

This is a reference flux set for constructing chains and required only if LFLX=2

PHI(1)	1-12	reference flux for the first group
PHI(2)	13-24	reference flux for the second group
PHI(IGN)		reference flux for the last group

4.2 Detailed Data Notes

More detailed information for some parameters, variables, and arrays is described below. The parameter variables used as dimension limits are given in Table 4.1.

LID. Program run identification number which is used for bookkeeping purposes.

LNK. The options of the program that are available for various calculational purposes.

If LNK=0, input data flux file and DCDLIB are read to make an index file and interval cross section table. Linear Decay Chains are constructed using the index file and these are printed along with the index file. Errors in input data may be detected in this calculation and it is recommended to put LNK=0, for the first run, or test run.

If LNK=1, in addition to the work for LNK=0 case, the program calculates the radioactivity, BHP, and afterheat which includes average decay particle energy and gamma energy. Zonewise radioactivity, BHP, and afterheat for each radioisotope are printed with the totals of that zone. Finally total blanket radioactivity, BHP, and afterheat are summarized.

If LNK=2, same as LNK=1 case, but the chain construction procedure is saved and the destruction data tables from preceding runs are used. With

TABLE 4.1. Dimension Parameters

MZN	Number of zones (< 17)
MRG	Number of intervals (< 100)
MRZ	Maximum numberof intervals in a zone (< 20)
MOP	Number of operating times (< 9)
MAS	Number of after shutdown times (< 12)
MKT	Number of transmutation types (= 29)
MXN	Number of neutron reaction types (= 19)
MCP	Number of composition tables (< 11)
MNN	Number of nuclides in the system (< 39)
MRD	Number of radioactive reaction products (< 61)
MPX	Number of radioisotopes for which data is given in BLOCK DATA (< 190)
MGX	Number of radioisotopes for which decay γ-ray data is given in BLOCK DATA (< 140)
MNG	Number of neutron energy groups (= 25)
MGG	Number of γ-ray energy groups (= 21)
MND	Number of nuclides for which data is given in DCDLIB (< 280)
MC	Number of chains from one nuclide (< 44)
MK	Number of steps in a chain (< 9)

this option, the segment PICKUP in the program is bypassed (option not working -- same as LNK=1).

If LNK=3, in addition to the calculations for LNK=1, the decay gamma ray data is stored in the file as a decay gamma source for the ANISN gamma transport calculation.

If LNK=4, same as LNK=3 case, except it uses the decay chains and destruction data table from the preceding run (option not working -- same as LNK=3).

LFX

Neutron flux is provided in either a DKR format or an ANISN scalar flux format.

In DKR format, LFX=1, and the flux set begins with a title card. This is followed by the flux for each interval in which the first card shows the interval number followed by the 25 group neutron flux for that interval. In ANISN scalar flux format, LFX=2, first title card and second flux data array identification card (=' 3*') are followed by neutron fluxes of the intervals, group by group.

Usually, the ANISN calculation is done on the basis of a normalized source, 10^{15} n/s, and the real fluxes are calculated by multiplying the flux conversion factor (FCF) to normalize the flux.

FCF is either supplied as input data (LFCF=1) or computed by the formula (LFCF=0)

$$FCF = W_L \times A_W \times 4.43 \times 10^{13}/10^{15} \qquad (4.1)$$

where W_L is the wall loading and A_W is the first wall area. 4.43×10^{13}

$n/s\text{-}cm^2$ is equivalent to a wall loading of $1\ MW/m^2$, and the factor of 10^{-24} is multiplied to simplify the activity calculation later, because σ is given in barns. If the neutron flux is based on other than a 10^{15} n/s strength, the flux conversion factor must be adjusted.

LGE. This gives the geometry of a reactor.

LGE = 1, slab.

LGE = 2, infinite cylinder, and the volume and area of first wall are computed for a 1 cm thick slice of cylinder.

The radial dimension is usually taken as the distance from the plasma center. But in a cylindrical shell calculation for tokamak reactors, this dimension is measured from the torus center.

LGE=3, sphere.

LGE=4, torus, but treated as same as LGE=2.

NOP. NOP represents total number of operation times.

If NOP > 0, Card No. 9 should be given, and if NOP = 0, a set of nine built-in operating times are used (Table 4.2).

NAS. NAS is total number of after shutdown times to be considered.

If NAS > 0, Card No. 10 should be given, but if NAS=0, twelve built-in after shutdown times are used (Table 4.2).

WLLD. Wall loading should be given in the unit of MW/m^2.

HTN. Neutron heating per fusion reaction in MeV.

HTG. Gamma ray heating per fusion reaction in MeV.

HTT. Total nuclear heating which includes neutron, gamma ray, and alpha particle heating in MeV.

LCLPS. In many fusion reactor neutron transport calculations, the neutron sources (emanating from a plasma or ICF target) are assumed to originate

TABLE 4.2. Built-In Times

1. 1 day = 8.640×10^4 s
2. 2 wk = 1.315×10^6 s
3. 1 mo = 2.630×10^6 s
4. 6 mo = 1.578×10^7 s
5. 1 yr = 3.156×10^7 s
6. 2 yr = 6.312×10^7 s
7. 4 yr = 1.262×10^8 s
8. 8 yr = 2.525×10^8 s
9. 16 yr = 5.050×10^8 s

1. 0
2. 1 m = 6.000×10 s
3. 10 m = 6.000×10^2 s
4. 1 h = 3.600×10^3 s
5. 6 h = 2.160×10^4 s
6. 1 d = 8.640×10^4 s
7. 1 wk = 6.048×10^5 s
8. 1 mo = 2.630×10^6 s
9. 1 yr = 3.156×10^7 s
10. 10 yr = 3.156×10^8 s
11. 100 yr = 3.156×10^9 s
12. 1000 yr = 3.156×10^{10} s

in a vacuum or void surrounded by the first wall and blanket materials. This usually calls for the presence of a number of mesh cells within the vacuum (void) to be able to transport the neutrons to the first wall and blanket. Since these zones effectively contain no materials, they are considered superfluous for the activation calculation. Thus the LCLPS parameter provides a means to eliminate these superfluous zone flux values. The KZEND parameter indicates the last interval of the neutron transport calculation which is considered part of the plasma or void.

ACUT. DKR terminates chains by several criteria (see Section 3.1). One criterion used examines the number density of the last stable nuclide or radionuclide with cross sections and compares this number density to the original input nuclide density. If this ratio is less than a given limit (< ACUT), the chain is terminated. A value of ACUT on the order of 5×10^{-4} is recommended. Lower values can also be used as long as the maximum number of booked chains, parameter MC, per unit nuclide is not exceeded.

TIR. The DKR code needs to know the time of irradiation to be able to construct and terminate chains. This irradiation time is given by the parameter TIR. This option was included to be able to take very small irradiation times (pulses, $t < 10^{-3}$ s) into account. Thus for irradiation times less than 1 year, specify the irradiation time. For times greater than 1 year, use 1 or 2 years. The TIR parameter should be specified in units of seconds.

4.3 Output

The first output section is an editing of input data with several calculated parameter values, e.g., operating power, first wall area, zone volume, and nuclide number densities by zone. Flux data is summarized to show the number of intervals and neutron groups, and the flux data title is also printed out.

The second part of output is the nuclear data library which is DCDLIB itself or a part of it. The nuclear data table follows to show available nuclides in the library and reveals the content of decay chain data.

If LPRT2=1, the nuclear data index table for the chain construction is printed out next. The reference flux is used to produce this table, which can be used as the table for reaction rates or transmutation rates.

The next section shows the procedure for chain construction. The existence of chains corresponding to each nuclide and the constructed chain information are printed out.

Then for each zone, zonewise radioactivity, BHP, and afterheat, both total $\beta + \gamma$ and β particle only for each radionuclide are presented, if LPRT1=1. If LPRT4=1, each linear decay chain is presented with its solution for each interval and for each operating time. Although this option is essential to check the solution of each chain, it should be used only when necessary, because it significantly increases the bulk of output. If LPRT3=1, radioactivities for whole intervals are printed out. Otherwise, only the activities of the intervals in the first zone are presented.

After the last zone activity is presented, final summary tables for the entire system are shown. For each operating time, normalized activities at each after shutdown time are in a concise form. The more important quantities in the summary are given in the units of [km^3 of air/kW_{th}] for BHP, [Ci/W_{th}] for radioactivity, and [% of operating power] for afterheat.

When LNK=3 or 4, the decay γ-ray sources for the ANISN transport calculation are stored in the γ-ray source array file (='17*').

Zonewise radioactivity, BHP, and afterheat can be written in a file or tape, as well as the specific activity of each interval in the zone. By applying the punch card unit number, these can be easily converted into card punched outputs. An example of output is given in Appendix B with input data.

5. USER'S GUIDE

5.1 Program Features

This section gives an outline of the DKR code and various information about the program for a user. All the routines in DKR have been written in standard FORTRAN-IV and considerable effort was made for generality in the program so that it can be run on most computers with minimal work in modification. DKR needs 65K words of core memory, and can be operable in any FORTRAN compiler. In addition to the standard input and output units, several logical units are required. The most unique feature of DKR is its construction of the linear decay chains with nuclear data from DCDLIB.

The program adopts the simple overlay structure for saving core storage. The segment in the overlay structures is shown in Fig. 5.1.

5.2 Subroutines

Various subroutines are described to show their major functions and relation to the other subroutines.

MAIN. Supervises the execution of other routines, and defines the logical units. The logical units, including standard input and output units, are specified as follows (numbers in parentheses correspond to the UNIVAC 1110 at MACC):

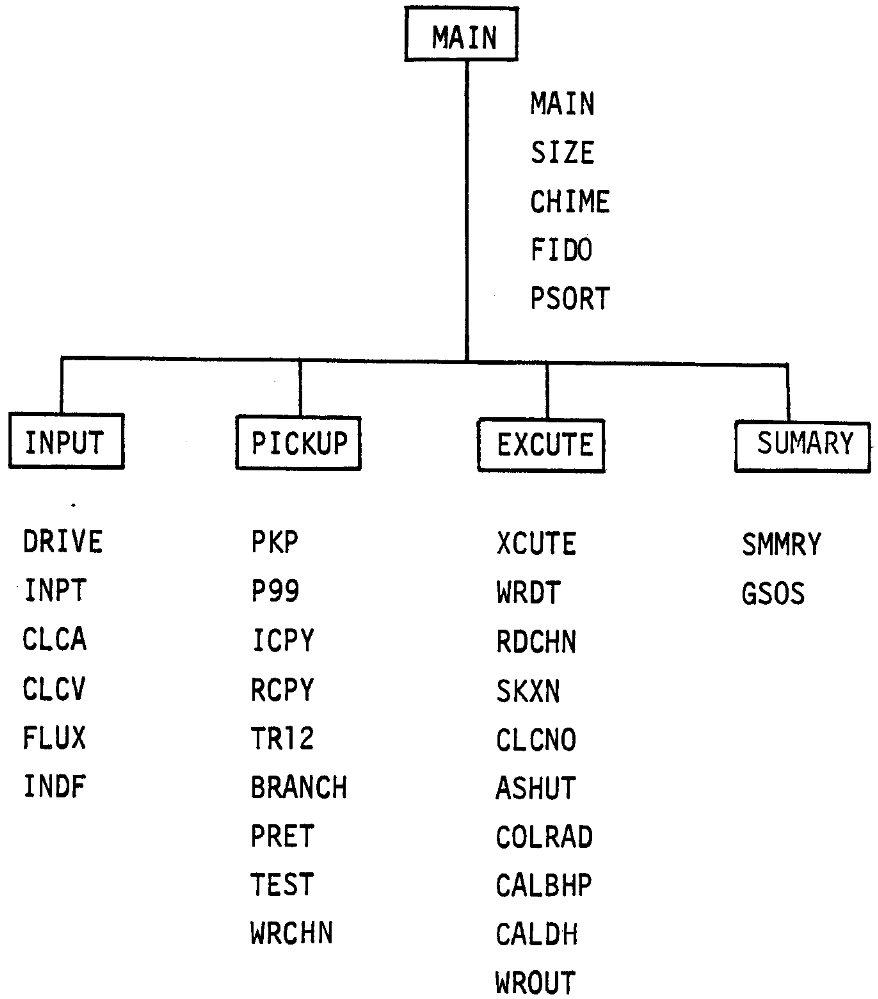


Fig. 5.1. Overlay structure.

I/O Units

N5	(5)	Standard input unit from which the basic data cards are read
N6	(6)	Standard output unit for printing
NT1	(1)	Punched output unit
NT2	(2)	Linear Decay chain file unit
NT3	(3)	Cross section X flux ($\sigma\phi$) table unit
NT7	(17)	Radioactivity file unit
NT8	(18)	Gamma-ray source file unit
NT9	(9)	Decay Chain Data Library unit

BLOK. BLOK is the BLOCK DATA subroutine. Miscellaneous nuclear data including the radioactive decay data for DKR are stored in BLOK.

SIZE. Approximate core size is estimated in subroutine SIZE based on the parameters given for array dimensions.

CHIME. Subroutine CHIME sets a clock at the beginning of a job and follows the collapsed time thereafter in units of seconds. Since this routine is from the UNIVAC 1110 at MACC of the University of Wisconsin, one can change this subroutine into a dummy routine or an equivalent time recording routine without affecting other other parts of the program.

CHIME (1) is the clock setting, and CHIME (2) is for the collapsed time after CHIME (1).

NOTE: This subroutine has been deactivated.

INPT, CLCA and CLCV. Input data are read in the subroutine INPT which edits and prints out the data and calls CLCA to calculate the first wall area. Also INPT calls subroutine CLCV for the calculation of each interval and each zone. The volume and area are calculated from the formula based on the geometry of reactor as shown in Table 5.1.

TABLE 5.1

Geometry	First Wall Area	Zone Volume
slab	1	$(R_o - R_1)\cdot 1$
cylinder	$2\pi R_W \cdot 1$	$\pi(R_o^2 - R_i^2)\cdot 1$
sphere	$4\pi R_W^2$	$\frac{4}{3}\pi (R_o^3 - R_i^3)$
torus	$4\pi^2 R_W R_T$	$2\pi^2(R_o^2 - R_i^2)\cdot R_T$
point	given	given

*All dimensions are measured in cm

R_W: first wall radius
R_o: outer radius of a zone
R_i: inner radius of a zone
R_T: major radius of torus

FLUX and FIDO. Subroutine FLUX reads neutron flux either in DKR format, or in ANISN scalar flux format read by a simplified FIDO subroutine.

INDF. Subroutine INDF processes the nuclear data from DCDLIB and the neutron flux into a general transmutation rate table and a reference data table. The main transmutation rate table is stored for later use in the EXCUTE segment. The reference table is used in the subroutines which construct the linear decay chains.

PKP and P99. These subroutines construct the decay chains. Subroutine PKP initiates the chain construction and assigns the maximum number of steps in each chain according to the importance of the initiating nuclide in the system.

P99 is the subroutine which actually constructs linear decay chains with nuclear data from DCDLIB. It calls subroutines such as BRANCH, ICPY, RCPY, TR12, PRET, and TEST, to gather together information and to decide whether the chain continues. After constructing a chain, it calls WRCHN to copy each chain into the decay chain file.

BRANCH. BRANCH retrieves and arranges the transmutation information for each nuclide in the last step of a chain, if data for it is stored in the DCDLIB.

ICPY, RCPY and TR12. During the chain construction, these subroutines are used to transfer information for each chain.

PRET and TEST. PRET is the subroutine which checks the maximum number of stable nuclides in the chains. TEST is the subroutine to check whether the chains continue according to their importance in the system.

WRCHN. WRCHN is used to write each constructed chain into a radioactive decay chain file.

XCUTE, RDCHN and SKXN. Subroutine XCUTE is the administration subroutine for calculating the radioactivity, biological hazard potential (BHP), and afterheat. RDCHN retrieves the chains and SKXN retrieves the corresponding destruction and production table. Then XCUTE calls CLCNO and ASHUT to solve the chains, calculate the radioactivity, and transfer the result to COLRAD.

CLCNO and ASHUT. Each decay chain is solved in the subroutine CLCNO to get the number density of nuclides at designated operating times. The number densities corresponding to various after shutdown times are calculated in the subroutine ASHUT.

COLRAD, CALBHP and CALDH. When the radioactivities of one zone are found, they are transferred to COLRAD, which edits them for each interval, and for each after shutdown time. It calls subroutines CALBHP and CALDH to compute corresponding BHP and afterheat, respectively. Also, COLRAD assembles the decay γ-ray source for each interval according to the program execution option.

WRDT and WROUT. These subroutines are for the printing of output. WRDT is called, if LPRT=4, to write information for each chain with its solution.

WROUT prints out the activity results of each zone. For each operating time, and after shutdown time, the radioactivity, BHP_{air}, and afterheat of each nuclide are printed out with their sums.

SMMRY. Subroutine SMMRY summarizes the radioactivity, BHP, and afterheat of the system in a concise form. Also normalized radioactivity and afterheat are presented for a comparison with the results of other systems.

5.3 Error Messages

This section contains error messages due to inconsistent input data and computational inconsistancies.

Error Messages

Error	Subroutine	Remarks
121	INPT	Inconsistent number of intervals
131	FLUX	Inconsistent number of intervals
132	FLUX	Incorrect flux format, ANISN format flux should be read group by group
141	FIDO	Incorrect input data array
211	INDF	Error in DCDLIB format
231	P99	Error in the chain sorting
233	P99	Error in chain construction procedures. Maximum number of chains in a step (rank) exceeded (> MK)
235	P99	Error in chain construction procedures. Maximum number of booked chains exceeded (> MC)
321	RDCHN	Incorrect transfer of chain information
331	SKXN	Incorrect number of intervals
332	SKXN	Error in the transmutation rate table

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APPENDIX A. COMPUTER CODE ABSTRACT OF DKR

1. Name of Code: DKR: A radioactivity calculation code for fusion reactors [1].

2. Coding Language and Computer: FORTRAN IV; UNIVAC 1100.

3. Description of Problem: The major purpose of DKR is to compute the activity of a fusion reactor by constructing the linear decay chains, and then solving them using nuclear data from Decay Chain Data Library (DCDLIB) [2]. The activity due to neutron activation is of great concern in a fusion reactor design for the choice of blanket and shield materials, accident analysis, maintenance procedures, and the evaluation of environmental impact. Special attention is given to developing an effective method of solving the activation of a nuclide as well as consistent procedures in using the existing nuclear data.

4. Method of Solution: The activation of a nuclide can be represented by linear decay chains, which in turn can be solved by a recursion coefficient formula [3]. The solution of chains is used to compute radioactivity, biological hazard potential (BHP), and afterheat by applying appropriate weighting functions. Decay γ-ray sources generated by DKR can be used in a γ-ray transport calculation for dose rate and a better estimation of afterheat.

5. Restrictions on the Complexity of the Problem: The following limits are noted for the core memories of less than 65K words. DKR accommodates nuclear data in DCDLIB, which are in 29 transmutation types, with 25 neutron energy groups.

6. Typical Running Time: Running time depends on the number of initial nuclides, and the number of intervals in the system. For 30 nuclides present in the first wall, the typical running time on the UNIVAC 1110 is approximately 10 to 15 seconds.

7. Unique Features of the Program: DKR constructs the linear decay chains by itself with nuclear data from DCDLIB. This is the first code to compute radioactivity, biological hazard potential (BHP), and afterheat without input of chain data. Also DKR generates decay γ-ray sources for use in the γ-ray transport calculation.

8. Related and Auxiliary Programs: DOSE is the auxiliary program to DKR in calculating dose rate due to decay γ-ray sources.

9. Machine Requirements: DKR was written in FORTRAN-IV for the UNIVAC 1110, and 65K words of core memory are needed. It can be run by most computers with minimal work in modification, and is operable in any FORTRAN compiler. In addition to the standard input and output unit, several logical units are required.

10. References:

[1] T.Y. Sung and W.F. Vogelsang, "DKR: A Radioactivity Calculation Code for Fusion Reactors," University of Wisconsin Fusion Technology Institute Report UWFDM-170 (September 1976).

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APPENDIX B. SAMPLE PROBLEM #1

A cylindrical model calculation of UWMAK-I radioactivity is described here as a sample problem.

The UWMAK-I blanket consists of first wall, homogenized breeding zone of liquid lithium (95%) and structure (5%), basic structure zone, another homogenized zone of liquid lithium (95%) and structure (5%), and final structure zone. 316 type stainless steel was chosen for the first wall material and structural materials, but in this sample problem Ti alloy (Ti-6Al-4V) is substituted on a volume basis.

The following pages contain the input data and the output for this sample problem.

Table B-1 Input for Sample Problem #1

```
                                                                                          CARD NO.
  UWMAK-I  < FIRST WALL + BLANKET >  *** TI-6AL-4V ***  TAK YUN SUNG                        1

 111      1      2      1      5     26      1     12     10      2     46     43            2

   1      1      0      1      1      1                                                     3

   1.25        12.43        4.13        20.060        1.00-24                               4

 500.         550.        1300.                                                             5

   1      1      1          550.        550.4                                               6.1
   2     17      0          550.4       601.4                                               6.2
   3      5      0          601.4       616.4                                               6.3
   4      2      0          616.4       621.4                                               6.4
   5      1      0          621.4       623.4                                               6.5

0.00   0.95   0.00   0.95   0.00                                                            7.1
1.00   0.05   1.00   0.05   1.00                                                            7.2

   1   3006      1    3.401E+21                                                             8.1
   1   3007      1   42.429E+21                                                             8.2
   2  13027      1    6.039E+21                                                             8.3
   2  22046      1    4.043E+21                                                             8.4
   2  22047      1    3.712E+21                                                             8.5
   2  22048      1   37.709E+21                                                             8.6
   2  22049      1    2.869E+21                                                             8.7
   2  22050      1    2.723E+21                                                             8.8
   2  23050      3     .005E+21                                                             8.9
   2  23051      2    2.128E+21                                                             8.10

 2 YRS     6.312E+07                                                                        9.1

 0         0.000E+00                                                                        10.1
 1 M       6.000E+01                                                                        10.2
 1 H       3.600E+03                                                                        10.3
 1 D       8.640E+04                                                                        10.4
 1 MO      2.630E+06                                                                        10.5
 1 YR      3.156E+07                                                                        10.6
10 YR      3.156E+08                                                                        10.7
100 Y      3.156E+09                                                                        10.8
 1 KY      3.156E+10                                                                        10.9
10 KY      3.156E+11                                                                        10.10
100KY      3.156E+12                                                                        10.11
 1 MY      3.156E+13                                                                        10.12
```

Table B-2 Output for Sample Problem #1

```
UWMAK-I  < FIRST WALL + BLANKET >  *** TI-6AL-4V ***  TAK YUN SUNG

                   PROBLEM RUN ID        111
LNK    LINK TO THE OTHER SOLUTION               1
LGE    1/2/3     = SLAB/CYL/SPH                 2
LFX    1/2       = TK3/SCALAR(ANISN)            1
IZM    NUMBER OF ZONES                          5
INT    NUMBER OF INTERVALS                     26
NOP    NUMBER OF OPERATING TIMES                1
NAS    NUMBER OF AFTER SHUTDOWN TIMES          12
NNC    NUMBER OF MATERIALS(NUCLIDES)           10
NCMP   NUMBER OF COMPOSITION TABLE              2
IGN    NUMBER OF NEUTRON GROUPS              * 46
IGG    NUBER OF PHOTON GRUOPS                * 43

                ZONE   1   2   3   4   5
 LI     3006               *       *
 LI     3007               *       *
 AL    13027           *   *   *   *   *
 TI    22046           *   *   *   *   *
 TI    22047           *   *   *   *   *
 TI    22048           *   *   *   *   *
 TI    22049           *   *   *   *   *
 TI    22050           *   *   *   *   *
 V     23050           *   *   *   *   *
 V     23051           *   *   *   *   *
```

Table B-2. Sample Output (Continued)

```
                    REACTOR SYSTEM PARAMETERS

RADIUS OF THE PLASMA          500.00        CM
RADIUS OF THE FIRST WALL      550.00        CM
RADIUS  OF THE TORUS         1300.00        CM

FIRST WALL AREA                 3.456-01    M2
NEUTRON WALL LOADING            1.250+00    MW/M2
TOTAL OPERATING POWER           6.146-01    MW
FLUX CONVERSION FACTOR          1.000-24

OPERATING TIME     1              AFTER SHUTDOWN TIME   12

 2 YRS     6.312+07   SECOND          0       0.000      SECOND
                                      1 M     6.000+01   SECOND
                                      1 H     3.600+03   SECOND
                                      1 D     8.640+04   SECOND
                                      1 MO    2.630+06   SECOND
                                      1 YR    3.156+07   SECOND
                                     10 YR    3.156+08   SECOND
                                    100 Y     3.156+09   SECOND
                                      1 KY    3.156+10   SECOND
                                     10 KY    3.156+11   SECOND
                                    100KY     3.156+12   SECOND
                                      1 MY    3.156+13   SECOND

              VOLUME OF ZONE

 ZONE    1    1.383+03   CM3
 ZONE    2    1.845+05   CM3
 ZONE    3    5.739+04   CM3
 ZONE    4    1.944+04   CM3
 ZONE    5    7.821+03   CM3
```

Table B-2. Sample Output (Continued)

NUCLIDE NO. DENSITY(10**18)

KZA	ZONE 1	2	3	4	5
3006	.0	3230.9	.0	3230.9	.0
3007	.0	40307.5	.0	40307.5	.0
13027	6039.0	301.9	6039.0	301.9	6039.0
22046	4043.0	202.1	4043.0	202.1	4043.0
22047	3712.0	185.6	3712.0	185.6	3712.0
22048	37709.0	1885.4	37709.0	1885.4	37709.0
22049	2869.0	143.4	2869.0	143.4	2869.0
22050	2723.0	136.1	2723.0	136.1	2723.0
23050	5.0	.2	5.0	.2	5.0
23051	2128.0	106.4	2128.0	106.4	2128.0

```
REFERENCE FLUX
          FIRST WALL  FLUX OF UWMAK-I

FLUX READING

     26 INTERVALS READ FROM FLUX ( 26, 46)
          NEUTRON FLUX UWMAK-I FIRST WALL


TIME RECORD
**** TIME AT  NOW   IS          .977 SECONDS *****
```

Table B-2. Sample Output (Continued)

NUCLEAR DATA TABLE

LKZA ①	NKT ②	③	KT= ④ 1 21	2 22	3 23	4 24	5 25	6 26	7 27	8 28	9 29	10	11	12	13	14	15	16	17	18	19
20060	2	R	O	O																	
20080	2	R	O	O																	
30060	5	S	X	X	X					X		X									
30070	6	S	X	X		X	X				X	X									
30080	2	R	O	O																	
30090	2	R	O	O																	
40090	5	S	X	X	X			X		X											
40100	2	R	O	O																	
50100	5	S	X		X		X			X			X								
50110	6	S	X	X	X	X		X		X											
60120	4	S	X	X						X					X						
60130	2	S	X	X																	
60140	2	R	O	O																	
70140	8	S	X	X	X	X	X	X		X			X								
70160	2	R	O	O																	
80160	5	S	X	X	X		X			X											
80190	2	R	O	O																	
90180	2	R	O		O																
90190	8	S	X	X	X	X	X	X		X	X										
90200	2	R	O	O																	
100230	2	R	O	O																	
110240	2	R	O	O																	
110250	2	R	O	O																	
110260	2	R	O	O																	
120240	3	S	X	X	X																
120250	3	S	X	X	X																
120260	4	S	X	X	X					X											
120270	2	R	O	O																	
130260	2	R	O		O																
130270	8	S	X	X	X	X	X	X		X									X		
130280	2	R	O	O																	
130290	2	R	O	O																	
130300	2	R	O	O																	
130301	2	R	O				O														
140280	4	S	X	X	X					X											
140290	4	S	X	X	X					X											
140300	4	S	X	X	X					X											
140310	2	R	O	O																	
170360	3	R	O	O	O																
170380	2	R	O	O																	

*** Note ***

① LKZA: Nuclide Identification number (= 10·KZA + LIS)
KZA: 1000·Z + A
LIS: Isomeric State of a Nuclide

② NKT: Total Number of Transmutation types

③ R: Radioisotope
S: Stable Nuclide

④ KT: Transmutation type Number
1~19, Reaction type ('X')
21~29, Radioactive Decay Type ('O')

Table B-2. Sample Output (Continued)

INDEX TABLE

FIRST WALL FLUX OF UWMAK-I

LKZA	SR	NKT	PRODUCT	T. RATE	KT		
	flag for an isotope			transmutation rate			
20060	10	2					
	radioisotope		20060	8.664-01	21	*TOT	
			30060	8.664-01	22	*B-	← λ_{β^-}
20080	10	2					
			20080	5.682+00	21	*TOT	
			30080	5.682+00	22	*B-	
30060	1	5					
			30060	3.626-10	1	TOTX	
	stable nuclide		30070	1.093-14	2	G	← (n,γ)
	(11 ← radioisotope		20060	1.487-12	3	P	
	which has cross sections)		10030	3.528-10	8	A	← (n,α)
			10010	6.229-12	10	A2N	
30070	1	6					
			30070	5.816-11	1	TOTX	
			30080	1.351-14	2	G	
			30060	2.603-12	4	2N	
			20060	1.153-12	5	NP	
			10030	5.060-11	9	NA	
			10020	3.786-12	10	A2N	
30080	10	2					
			30080	8.232-01	21	*TOT	
			40080	8.232-01	22	*B-	
30090	10	2					
			30090	4.077+00	21	*TOT	
			40090	4.077+00	22	*B-	
40090	1	5					
			40090	8.153-12	1	TOTX	
			40100	4.844-14	2	G	
			30090	6.643-15	3	P	
			30070	2.945-12	6	T	
			20060	5.151-12	8	A	

Table B-2. Sample Output (Continued)

```
                    CHAIN CONSTRUCTION PROCEDURES

3006 CHAIN
   6/i
                                                          Reaction     30070 - 4
                                                          Product  ->  20060 - 3
                                                                       10030 - 2
                                                                       10010 - 1
  Chain No.     Combined ID number        20060  *PASS  2
                of SR and KT              30070  TEST   0
                                                                       30060 - 1
 101                (σφ+λ)i-1    (σφ+λ)T       λi-1          λT
     30060   103    .8664+00     .3626-09     .8664+00      .0000          101  1
     20060  1022    .1487-11     .6664+00     .0000         .8664+00       101  2

 $ YES              NUMBER OF CHAINS =  1

3007 CHAIN
                                                                       30080 - 5
                                                                       30060 - 4
                                                                       20060 - 3
                                                                       10030 - 2
                                                                       10020 - 1
                                          20060  *PASS  2
                                          30060  TEST   0
                                          30080  *PASS  2
                                                                       30060 - 1
                                                                       40080 - 2
 201
     30070   105    .0000        .5816-10     .0000         .0000          201  1
     20060  1022    .1153-11     .8664+00     .0000         .8664+00       201  2
 202
     30070   102    .0000        .5816-10     .0000         .0000          202  1
     30080  1022    .1351-13     .8232+00     .0000         .8232+00       202  2

 $ YES              NUMBER OF CHAINS =  2

13027 CHAIN
                                                                       130280 - 7
                                                                       120270 - 4
                                                                       130260 - 5
                                                                       120260 - 3
                                                                       120250 - 2
                                                                       110240 - 1
                                                                       130261 - 6
                                          110240  *PASS  2
                                          120250  TEST   0
                                          120260  TEST   0
                                          120270  *PASS  2
                                          130260  *PASS  2
                                          130280  *PASS  2
```

Table B-2. Sample Output (Continued)

EXECUTING PROCEDURES FOR ZONE 1

LKZA	LRX	AI	AK	BI	BK	YO	YT	Z, I	OP
	(=100·SR+KT)	$(\sigma\varphi+\lambda)^{i}_{i-1}$	$(\sigma\varphi+\lambda)_{T}$	λ^{i}_{i-1}	λ_{T}	$y(0)$	$y(t)$	Zone Interval	Operation Time
MXC = 4									
130270	108	0.000	3.511-11	0.000	0.000	6.039+21	6.026+21	< 1, 1>	2 YRS
110240	1022	1.523-11	1.287-05	0.000	1.287-05	0.000	7.129+15		
130270	103	1.221-03	3.511-11	1.221-03	0.000	6.039+21	6.026+21	< 1, 1>	2 YRS
120270	1022	1.107-11	1.221-03	0.000	1.221-03	0.000	5.437+13		
130270	104	0.000	3.511-11	0.000	0.000	6.039+21	6.026+21	< 1, 1>	2 YRS
130260	1023	2.442-12	2.968-14	0.000	2.968-14	0.000	9.297+17		
130270	102	0.000	3.511-11	0.000	0.000	6.039+21	6.026+21	< 1, 1>	2 YRS
130280	1022	9.322-13	5.020-03	0.000	5.020-03	0.000	1.119+12		
MXC = 2									
220460	103	9.430-08	6.642-11	9.430-08	0.000	4.043+21	4.026+21	< 1, 1>	2 YRS
210460	1022	4.070-11	9.430-08	0.000	9.430-08	0.000	1.726+18		
220460	104	0.000	6.642-11	0.000	0.000	4.043+21	4.026+21	< 1, 1>	2 YRS
220450	1023	3.310-12	6.340-05	0.000	6.340-05	0.000	2.102+14		
MXC = 2									
220470	105	0.000	9.990-11	0.000	0.000	3.712+21	3.689+21	< 1, 1>	2 YRS
210460	1022	4.391-12	9.430-08	0.000	9.430-08	0.000	1.715+17		
220470	103	2.338-06	9.990-11	2.338-06	0.000	3.712+21	3.689+21	< 1, 1>	2 YRS
210470	1022	2.192-11	2.338-06	0.000	2.338-06	0.000	3.442+16		
MXC = 3									
220480	108	0.000	6.946-11	0.000	0.000	3.771+22	3.754+22	< 1, 1>	2 YRS
200450	1022	3.352-12	4.850-08	0.000	4.850-08	0.000	2.476+18		
220480	105	0.000	6.946-11	0.000	0.000	3.771+22	3.754+22	< 1, 1>	2 YRS
210470	1022	3.143-12	2.338-06	0.000	2.338-06	0.000	5.047+16		
220480	103	4.366-06	6.946-11	4.386-06	0.000	3.771+22	3.754+22	< 1, 1>	2 YRS
210480	1022	9.860-12	4.386-06	0.000	4.386-06	0.000	8.398+16		

Table B-2. Sample Output (Continued)

INTERVAL ACTIVITY

1 - 1 (ZONE-INT)   2 YRS OPERATING

NUCLIDE	0	1 M	1 H	1 D	1 MO	1 YR	10 YR	100 Y	1 KY	10 KY	100KY	1 MY	
110240	9.174+10	9.167+10	8.769+10	3.018+10	1.830-04	0.000	0.00	0.00	0.00	0.00	0.00	0.00	
120270	6.639+10	6.170+10	8.186+08	1.015-35	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00	
130260	2.759+04	2.759+04	2.759+04	2.759+04	2.759+04	2.759+04	2.76+04	2.76+04	2.76+04	2.73+04	2.51+04	1.08+04	
130280	5.617+09	4.156+09	7.960+01	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00	
210460	1.789+11	1.789+11	1.780+11	1.774+11	1.396+11	9.122+09	2.13-02	0.00	0.00	0.00	0.00	0.00	
220450	1.333+10	1.328+10	1.061+10	5.569+07	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00	
210470	2.013+11	2.012+11	1.996+11	1.640+11	4.928+08	4.685-15	0.00	0.00	0.00	0.00	0.00	0.00	
200450	1.201+11	1.201+11	1.201+11	1.196+11	1.058+11	2.600+10	2.70+04	0.00	0.00	0.00	0.00	0.00	
210480	3.602+11	3.801+11	3.742+11	2.603+11	3.718+06	0.000	0.00	0.00	0.00	0.00	0.00	0.00	
210490	1.987+10	1.963+10	9.639+09	5.751+02	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00	
210500	6.278+09	4.196+09	1.992-01	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00	
200470	2.072+09	2.072+09	2.059+09	1.778+09	1.971+07	1.138-15	0.00	0.00	0.00	0.00	0.00	0.00	
230490	3.280+08	3.280+08	3.280+08	3.274+08	3.077+08	1.524+08	1.53+05	1.62-25	0.00	0.00	0.00	0.00	
220510	1.345+10	1.194+10	1.041+07	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00	
230520	1.157+10	9.620+09	1.770+05	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00	
TOTAL	1.111+12	1.099+12	9.838+11	7.545+11	2.462+11	3.527+10	2.08+05	2.76+04	2.76+04	2.73+04	2.51+04	1.08+04	DPS/CM3

Table B-2. Sample Output (Continued)

ZONE 1 ACTIVITY 2 YRS OPERATING (IN CURIE)

	NUCLIDE	0	1 M	1 H	1 D	1 MO	1 YR	10 YR	100 Y	1 KY	10 KY	100KY	1 MY
110240	NA 24	3.429+03	3.426+03	3.273+03	1.128+03	6.841-12	0.000	0.00	0.00	0.00	0.00	0.00	0.00
120270	MG 27	2.481+03	2.306+03	3.059+01	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
130260	AL 26	1.031-03	1.031-03	1.031-03	1.031-03	1.031-03	1.031-03	1.03-03	1.03-03	1.03-03	1.02-03	9.39-04	4.04-04
130280	AL 28	2.099+02	1.553+02	2.975-06	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
210460	SC 46	6.685+03	6.685+03	6.683+03	6.631+03	5.217+03	3.409+02	7.94-10	0.00	0.00	0.00	0.00	0.00
220450	TI 45	4.981+02	4.962+02	3.964+02	2.061+00	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
210470	SC 47	7.522+03	7.520+03	7.459+03	6.159+03	1.842+01	1.751-22	0.00	0.00	0.00	0.00	0.00	0.00
200450	CA 45	4.490+03	4.490+03	4.489+03	4.471+03	3.952+03	9.716+02	1.01-03	0.00	0.00	0.00	0.00	0.00
210480	SC 48	1.421+04	1.420+04	1.399+04	9.727+03	1.390-01	0.000	0.00	0.00	0.00	0.00	0.00	0.00
210490	SC 49	7.425+02	7.336+02	3.602+02	2.149-05	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
210500	SC 50	2.346+02	1.568+02	7.443-09	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
200470	CA 47	7.744+01	7.743+01	7.695+01	6.646+01	7.367-01	4.253-23	0.00	0.00	0.00	0.00	0.00	0.00
230490	V 49	1.226+01	1.226+01	1.226+01	1.223+01	1.150+01	5.694+00	5.73-03	6.05-33	0.00	0.00	0.00	0.00
220510	TI 51	5.027+02	4.461+02	3.890-01	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
230520	V 52	4.325+02	3.595+02	6.615-03	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL	4.153+04	4.107+04	3.677+04	2.820+04	9.200+03	1.318+03	7.77-03	1.03-03	1.03-03	1.02-03	9.39-04	4.04-04 CI

Table B-2. Sample Output (Continued)

ZONE 1 BHP 2 YRS OPERATING (IN KM3/KW)

	NUCLIDE	0	1 M	1 H	1 D	1 MO	1 YR	10 YR	100 Y	1 KY	10 KY	100KY	1 MY
110240	NA 24	5.579+01	5.575+01	5.327+01	1.835+01	1.113-13	0.000	0.00	0.00	0.00	0.00	0.00	0.00
120270	MG 27	1.346-01	1.251-01	1.659-03	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
130260	AL 26	1.678-05	1.678-05	1.678-05	1.678-05	1.678-05	1.678-05	1.68-05	1.68-05	1.68-05	1.66-05	1.53-05	6.58-06
130280	AL 28	1.139-02	8.425-03	1.614-10	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
210460	SC 46	1.360+01	1.360+01	1.359+01	1.349+01	1.061+01	6.934-01	1.62-12	0.00	0.00	0.00	0.00	0.00
220450	TI 45	2.384-03	2.375-03	1.897-03	9.961-06	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
210470	SC 47	6.119-01	6.119-01	6.069-01	5.011-01	1.498-03	1.424-26	0.00	0.00	0.00	0.00	0.00	0.00
200450	CA 45	7.306+00	7.306+00	7.305+00	7.276+00	6.431+00	1.581+00	1.64-06	0.00	0.00	0.00	0.00	0.00
210480	SC 48	4.624+00	4.623+00	4.552+00	3.165+00	4.522-05	0.000	0.00	0.00	0.00	0.00	0.00	0.00
210490	SC 49	9.293-04	9.182-04	4.509-04	2.690-11	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
210500	SC 50	3.471-03	2.320-03	1.101-13	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
200470	CA 47	2.100-02	2.100-02	2.087-02	1.802-02	1.998-04	1.154-26	0.00	0.00	0.00	0.00	0.00	0.00
230490	V 49	7.980-05	7.980-05	7.979-05	7.963-05	7.486-05	3.706-05	3.73-08	3.94-38	0.00	0.00	0.00	0.00
220510	TI 51	9.623-04	8.540-04	7.447-07	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
230520	V 52	2.011-03	1.672-03	3.075-08	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL	8.211+01	8.205+01	7.935+01	4.280+01	1.704+01	2.274+00	1.85-05	1.68-05	1.68-05	1.66-05	1.53-05	6.58-06 KM3/KW

Table B-2. Sample Output (Continued)

ZONE 1 AFTERHEAT 2 YRS OPERATING (IN MW)

	NUCLIDE	0	1 M	1 H	1 D	1 MO	1 YR	10 YR	100 Y	1 KY	10 KY	100KY	1 MY
110240	NA 24	9.517-05	9.509-05	9.086-05	3.130-05	1.899-19	0.000	0.00	0.00	0.00	0.00	0.00	0.00
120270	MG 27	2.322-05	2.158-05	2.864-07	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
130260	AL 26	2.026-11	2.026-11	2.026-11	2.026-11	2.026-11	2.026-11	2.03-11	2.03-11	2.02-11	2.01-11	1.84-11	7.94-12
130280	AL 28	3.613-06	2.673-06	5.120-14	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
210460	SC 46	8.288-05	8.287-05	8.285-05	8.220-05	6.467-05	4.226-06	9.85-18	0.00	0.00	0.00	0.00	0.00
220450	TI 45	1.125-06	1.121-06	8.954-07	4.701-09	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
210470	SC 47	1.227-05	1.227-05	1.217-05	1.005-05	3.004-08	2.856-31	0.00	0.00	0.00	0.00	0.00	0.00
200450	CA 45	1.712-06	1.712-06	1.711-06	1.704-06	1.507-06	3.704-07	3.85-13	0.00	0.00	0.00	0.00	0.00
210480	SC 48	2.965-04	2.964-04	2.919-04	2.030-04	2.900-09	0.000	0.00	0.00	0.00	0.00	0.00	0.00
210490	SC 49	3.650-06	3.606-06	1.771-06	1.057-13	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
210500	SC 50	6.640-06	4.438-06	2.106-16	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
200470	CA 47	6.083-07	6.082-07	6.044-07	5.220-07	5.786-09	3.341-31	0.00	0.00	0.00	0.00	0.00	0.00
230490	V 49	1.308-08	1.308-08	1.308-08	1.305-08	1.227-08	6.076-09	6.11-12	0.00	0.00	0.00	0.00	0.00
220510	TI 51	2.607-06	2.314-06	2.018-09	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
230520	V 52	6.413-06	5.331-06	9.808-11	0.000	0.000	0.000	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL	5.364-04	5.301-04	4.830-04	3.288-04	6.623-05	4.602-06	2.68-11	2.03-11	2.02-11	2.01-11	1.84-11	7.94-12 MW

TIME RECORD
**** TIME AT NOW IS 21.592 SECONDS *****

Table B-2. Sample Output (Continued)

SUMMARY OF UWMAK-I < FIRST WALL + BLANKET > *** TI-6AL-4V *** TAK YUN SUNG

2 YRS OPERATION 6.312+07 SEC

TIME AFTER SHUTDOWN SEC		TOTAL ACT CI	TOTAL BHP KM3/KW	TOTAL AHT MW	BETA AHT MW	PER ACT CI/W	% AHT %
0.000	0	4.153+04	8.211+01	5.364-04	6.221-05	6.757-02	8.729-02
6.000+01	1 M	4.107+04	8.205+01	5.301-04	5.959-05	6.683-02	8.625-02
3.600+03	1 H	3.677+04	7.935+01	4.830-04	4.040-05	5.983-02	7.860-02
8.640+04	1 D	2.820+04	4.260+01	3.268-04	2.487-05	4.588-02	5.350-02
2.630+06	1 MO	9.200+03	1.704+01	6.623-05	4.343-06	1.497-02	1.078-02
3.156+07	1 YR	1.318+03	2.274+00	4.602-06	5.545-07	2.145-03	7.489-04
3.156+08	10 YR	7.766-03	1.846-05	2.675-11	3.222-12	1.264-08	4.353-09
3.156+09	100 Y	1.031-03	1.678-05	2.026-11	2.836-12	1.678-09	3.296-09
3.156+10	1 KY	1.030-03	1.676-05	9.627-02	2.670-08	1.676-09	1.566+01
3.156+11	10 KY	1.027-03	1.667-05	2.007-11	2.810-12	1.662-09	3.266-09
3.156+12	100KY	9.390-04	1.528-05	1.845-11	2.583-12	1.528-09	3.002-09
3.156+13	1 MY	4.042-04	6.576-06	1.901-11	1.112-12	6.576-10	3.094-09

APPENDIX B. SAMPLE PROBLEM #2

This problem is the first part of an activation calculation which models a spherical reactor cavity consisting of a 10 cm thick H-451 graphite ISSEC (Internal Spectrum Shifter and Energy Converter)/moderator (0.4 void fraction) and a 4 cm thick aluminum-6061-T6 first wall sandwiched between two 1 cm thick Boral sheets. This structure is surrounded by a 300 cm thick borated water shield containing 2000 wppm of boron.

Table B-3 Input for Sample Problem #2

```
TDF ALUMINUM: CARBON + BORAL + F.W. + BORAL + WATER + BORON   PART-I
      1      3      3      2      9     80      1      0     35     10     25     21
      1      0      0      0      2      1      1      1
  1.341E-4       11.322         3.570        14.893       7.012E-9
     289.0        289.0           0.0        4.0E-4       3.156E+7
      1      5      1           289.00      299.00      1
      2      2      1           299.00      300.00      1
      3      4      1           300.00      304.00      1
      4      2      1           304.00      305.00      1
      5     10      1           305.00      325.00      1
      6     20      1           325.00      425.00      1
      7     20      1           425.00      525.00      1
      8     16      1           525.00      605.00      1
      9      1      1           605.00      606.00      1
 1.04-3  198.6    0.0  198.6   1.00   1.00   1.00   1.00    0.0
    0.0    0.0    0.0    0.0   1.00   1.00   1.00   1.00    0.0
  121.4  12.65   1.00  12.65    0.0    0.0    0.0    0.0    0.0
 1.60-6  .7764   1.00  .7764    0.0    0.0    0.0    0.0    0.0
 1.35-3    0.0   1.00    0.0    0.0    0.0    0.0    0.0    0.0
 2.58-4    0.0   1.00    0.0    0.0    0.0    0.0    0.0    0.0
   1.00    0.0    0.0    0.0    0.0    0.0    0.0    0.0    0.0
 1.66-4    0.0   1.00    0.0    0.0    0.0    0.0    0.0    0.0
    0.0    0.0   1.00    0.0    0.0    0.0    0.0    0.0    0.0
 3.87-5    0.0   1.00    0.0    0.0    0.0    0.0    0.0    0.0
      1   5010      1    2.229E+19
      1   5011      1    8.914E+19
      3   6012      1    4.266E+20
      3   6013      1    4.788E+18
```

Table B-3 continued

```
29        2  8016     1    3.335E+22
30        2  8017     1    1.270E+19
31        2  8018     1    6.820E+19
32       10 12024     1    5.284E+20
33       10 12025     1    6.689E+19
34       10 12026     1    7.365E+19
35        4 13027     1    5.819E+22
36        5 14028     1    3.204E+20
37        5 14029     1    1.622E+19
38        5 14030     1    1.077E+19
39        7 16032     1    1.863E+16
40        7 16033     1    1.471E+14
41        7 16034     1    8.256E+14
42        9 19039     1    1.435E+15
43        9 19041     1    1.036E+14
44        6 22046     1    4.201E+18
45        6 22047     1    3.794E+18
46        6 22048     1    3.753E+19
47        6 22049     1    2.750E+18
48        6 22050     1    2.648E+18
49        7 23050     1    3.086E+13
50        7 23051     1    1.231E+16
51        9 24050     1    4.082E+18
52        9 24052     1    7.862E+19
53        9 24053     1    8.914E+18
54        9 24054     1    2.214E+18
55        9 25055     1    4.440E+19
56        8 26054     1    1.182E+19
57        8 26056     1    1.871E+20
58        8 26057     1    4.280E+18
59        8 26058     1    6.115E+17
60     1 YR    3.156E+7
61     CARBON - BORAL - ALUMINUM - BORAL - BORATED WATER SHIELD  STEADY STATE
62        6.835E-7    1.065E-7    1.195E-7    9.732E-8    4.330E-8    5.219E-8
63        3.872E-8    6.500E-8    6.198E-8    5.472E-8    1.033E-7    8.804E-8
64        7.886E-8    1.115E-7    4.498E-7    6.505E-7    2.776E-7    3.457E-7
65        5.032E-7    5.443E-7    4.168E-7    3.051E-7    2.109E-7    1.414E-7
66        4.513E-7
@XQT DKRMAIN11.ABS4
```

Table B-4 Output for Sample Problem #2

```
TDF ALUMINUM: CARBON + BORAL + F.W. + BORAL + WATER + BORON    PART-I

A ONE HERE INDICATES COLLAPSING OF FLUXES  1  1

                  PROBLEM RUN ID              1
LNK    LINK TO THE OTHER SOLUTION                3
LGE    1/2/3   = SLAB/CYLINDER/SPHERE            3
LFX    1/2     = DKR/ANISN (FORMATTED            2
IZM    NUMBER OF ZONES                           9
INT    NUMBER OF INTERVALS                      80
NOP    NUMBER OF OPERATING TIMES                 1
NAS    NUMBER OF AFTER SHUTDOWN TIMES           12
NNC    NUMBER OF MATERIALS(NUCLIDES)            35
NCMP   NUMBER OF COMPOSITION TABLE              10
IGN    NUMBER OF NEUTRON GROUPS               * 25
IGG    NUMBER OF PHOTON GROUPS                * 43
```

	ZONE	1	2	3	4	5	6	7	8	9
B	5010	*	*		*	*	*	*	*	
B	5011	*	*		*	*	*	*	*	
C	6012	*	*	*	*					
C	6013	*	*	*	*					
O	8016					*	*	*	*	
O	8017					*	*	*	*	
O	8018					*	*	*	*	
MG	12024	*		*						
MG	12025	*		*						
MG	12026	*		*						
AL	13027	*	*	*	*					
SI	14028	*		*						
SI	14029	*		*						
SI	14030	*		*						
S	16032	*								
S	16033	*								
S	16034	*								
K	19039			*						
K	19041			*						
TI	22046	*		*						
TI	22047	*		*						
TI	22048	*		*						
TI	22049	*		*						
TI	22050	*		*						
V	23050	*								
V	23051	*								
CR	24050			*						
CR	24052			*						
CR	24053			*						
CR	24054			*						
MN	25055			*						
FE	26054	*		*						
FE	26056	*		*						
FE	26057	*		*						
FE	26058	*		*						

Table B-4 continued

REACTOR SYSTEM PARAMETERS

RADIUS OF THE PLASMA	289.00	CM
RADIUS OF THE FIRST WALL	289.00	CM
RADIUS OF THE TORUS	.00	CM

FIRST WALL AREA	1.050+002	M2
NEUTRON WALL LOADING	1.341-004	MW/M2
TOTAL OPERATING POWER	1.487-002	MW
FLUX CONVERSION FACTOR	7.012-009	
ACCURCY LIMIT	4.000-004	
TEST IRRADIATION TIME	3.156+007	

OPERATING TIME 1 AFTER SHUTDOWN TIME 12

1 YR	3.156+007 SECOND	0	.000	SECOND
		1 M	6.000+001	SECOND
		10 M	6.000+002	SECOND
		1 H	3.600+003	SECOND
		6 H	2.160+004	SECOND
		1 D	8.640+004	SECOND
		1 W	6.048+005	SECOND
		1 MO	2.630+006	SECOND
		1 YR	3.156+007	SECOND
		10 YR	3.156+008	SECOND
		100 YR	3.156+009	SECOND
		1000 Y	3.156+010	SECOND

VOLUME OF ZONE

ZONE 1	1.086+007	CM3
ZONE 2	1.127+006	CM3
ZONE 3	4.584+006	CM3
ZONE 4	1.165+006	CM3
ZONE 5	2.495+007	CM3
ZONE 6	1.778+008	CM3
ZONE 7	2.846+008	CM3
ZONE 8	3.215+008	CM3
ZONE 9	4.607+006	CM3

Table B-4 Continued

NUCLIDE NO. DENSITY(10**18)

KZA ZONE	1	2	3	4	5	6	7	8	9
5010	.023	4426.794	.000	4426.794	22.290	22.290	22.290	22.290	.000
5011	.093	17703.204	.000	17703.204	89.140	89.140	89.140	89.140	.000
6012	51789.239	5396.490	426.600	5396.490	.000	.000	.000	.000	.000
6013	581.263	60.568	4.788	60.568	.000	.000	.000	.000	.000
8016	.000	.000	.000	.000	33350.000	33350.000	33350.000	33350.000	.000
8017	.000	.000	.000	.000	12.700	12.700	12.700	12.700	.000
8018	.000	.000	.000	.000	68.200	68.200	68.200	68.200	.000
12024	.020	.000	528.400	.000	.000	.000	.000	.000	.000
12025	.003	.000	66.890	.000	.000	.000	.000	.000	.000
12026	.003	.000	73.650	.000	.000	.000	.000	.000	.000
13027	.093	45178.715	58189.999	45178.715	.000	.000	.000	.000	.000
14028	.433	.000	320.400	.000	.000	.000	.000	.000	.000
14029	.022	.000	16.220	.000	.000	.000	.000	.000	.000
14030	.015	.000	10.770	.000	.000	.000	.000	.000	.000
16032	.019	.000	.000	.000	.000	.000	.000	.000	.000
16033	.000	.000	.000	.000	.000	.000	.000	.000	.000
16034	.001	.000	.000	.000	.000	.000	.000	.000	.000
19039	.000	.000	.001	.000	.000	.000	.000	.000	.000
19041	.000	.000	.000	.000	.000	.000	.000	.000	.000
22046	.001	.000	4.201	.000	.000	.000	.000	.000	.000
22047	.001	.000	3.794	.000	.000	.000	.000	.000	.000
22048	.010	.000	37.530	.000	.000	.000	.000	.000	.000
22049	.001	.000	2.750	.000	.000	.000	.000	.000	.000
22050	.001	.000	2.648	.000	.000	.000	.000	.000	.000
23050	.000	.000	.000	.000	.000	.000	.000	.000	.000
23051	.012	.000	.000	.000	.000	.000	.000	.000	.000
24050	.000	.000	4.062	.000	.000	.000	.000	.000	.000
24052	.000	.000	78.620	.000	.000	.000	.000	.000	.000
24053	.000	.000	8.914	.000	.000	.000	.000	.000	.000
24054	.000	.000	2.214	.000	.000	.000	.000	.000	.000
25055	.000	.000	44.400	.000	.000	.000	.000	.000	.000
26054	.002	.000	11.820	.000	.000	.000	.000	.000	.000
26056	.031	.000	187.100	.000	.000	.000	.000	.000	.000
26057	.001	.000	4.280	.000	.000	.000	.000	.000	.000
26058	.000	.000	.611	.000	.000	.000	.000	.000	.000

REFERENCE FLUX
CARBON - BORAL - ALUMINUM - BORAL - BORATED WATER SHIELD   STEADY STATE
@ADD,PD alum*flux.

FLUX READING

TITLE OF FLUX FILE : TDF ALUM F.W.+CARBON+BORAL+Fw+BORAL+WATER+BORON

81 INTERVALS READ FROM FLUX (81, 25)
81 INTERVALS HAVE BEEN COLLAPSED TO 80 INTERVALS

Table B-4 Continued

NUCLEAR DATA TABLE

LKZA	NKT	KT=	1 21	2 22	3 23	4 24	5 25	6 26	7 27	8 28	9 29	10	11	12	13	14	15	16	17	18	19
20060	2	R	0	0																	
20080	2	R	0	0																	
30060	5	S	X	X	X					X		X									
30070	6	S	X	X		X	X				X	X									
30080	2	R	0	0																	
30090	2	R	0	0																	
40090	5	S	X	X	X			X		X											
40100	2	R	0	0																	
50100	5	S	X		X		X			X			X								
50110	6	S	X	X	X	X		X		X											
60120	4	S	X	X						X					X						
60130	2	S	X	X																	
60140	2	R	0	0																	
70140	8	S	X	X	X	X	X	X		X			X								
70160	2	R	0	0																	
80160	5	S	X	X	X		X			X											
80190	2	R	0	0																	
90180	2	R	0		0																
90190	8	S	X	X	X	X	X	X		X	X										
90200	2	R	0	0																	
100230	2	R	0	0																	
110240	2	R	0	0																	
110250	2	R	0	0																	
110260	2	R	0	0																	
120240	3	S	X	X	X																
120250	3	S	X	X	X																
120260	4	S	X	X	X					X											
120270	2	R	0	0																	
130260	2	R	0		0																
130270	8	S	X	X	X	X	X	X		X									X		
130280	2	R	0	0																	
130290	2	R	0	0																	
130300	2	R	0	0																	
130301	2	R	0				0														
140280	4	S	X	X	X					X											
140290	4	S	X	X	X					X											
140300	4	S	X	X	X					X											
140310	2	R	0	0																	
170360	3	R	0	0	0																
170380	2	R	0	0																	

Table B-4 Continued

```
CHAIN CONSTRUCTION PROCEDURES

5010 CHAIN
                                                        40100 - 4
                                                        40090 - 3
                                                        30070 - 2
                                                        10030 - 1
                                  30070 TEST    1
                                  40090 TEST    0
                                  40100  *PASS  2
                                                        30080 - 5
                                                        30060 - 4
                                                        20060 - 3
                                                        10030 - 2
                                                        10020 - 1
                                                        50100 - 6
                                  20060  *PASS  2
                                  30060 TEST    0
                                  30080  *PASS  2
  101
      50100   103   .1357-013   .2017-010   .1357-013   .0000          101  1
      40100  1022   .4139-015   .1357-013   .0000       .1357-013      101  2
                                                        30060 - 1
                                                        40080 - 2
                                  30060 TEST    0
  102
      50100   108   .0000       .2017-010   .0000       .0000          102  1
      30070   105   .2017-010   .4360-014   .0000       .0000          102  2
      20060  1022   .6040-016   .8557+000   .0000       .8557+000      102  3
  103
      50100   108   .0000       .2017-010   .0000       .0000          103  1
      30070   102   .2017-010   .4360-014   .0000       .0000          103  2
      30080  1022   .1846-015   .8232+000   .0000       .8232+000      103  3

 $ YES           NUMBER OF CHAINS =  3

5011 CHAIN
                                                        50120 - 5
                                                        40110 - 3
                                                        50100 - 4
                                                        40090 - 2
                                                        30080 - 1
                                  30080  *PASS  2
                                  40090 TEST    0
                                  50100 TEST    0
                                                        40080 - 1
  201
      50110   108   .0000       .4189-015   .0000       .0000          201  1
      30080  1022   .2059-015   .8232+000   .0000       .8232+000      201  2

 $ YES           NUMBER OF CHAINS =  1

6012 CHAIN
                                                        60130 - 3
                                                        40090 - 2
                                                           10 - 1
```

Table B-4 Continued

INTERVAL ACTIVITY

1 - 1 (ZONE-INT)   1 YR OPERATING

NUCLIDE	0	1 M	10 M	1 H	6 H	1 D	1 W	1 MO	1 YR	10 YR	100 YR	1000 Y
40100	4.008-006	4.008-006	4.008-006	4.008-006	4.008-006	4.008-006	4.01-006	4.01-006	4.01-006	4.01-006	4.01-006	4.01-006
20060	5.263-004	2.653-026	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
30080	1.877+001	6.649-021	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
60140	7.324-002	7.324-002	7.324-002	7.324-002	7.324-002	7.324-002	7.32-002	7.32-002	7.32-002	7.32-002	7.24-002	6.49-002
110240	1.109+002	1.108+002	1.100+002	1.059+002	8.396+001	3.647+001	4.62-002	2.21-013	.00	.00	.00	.00
110250	7.816-001	3.908-001	7.644-004	6.639-019	.000	.000	.00	.00	.00	.00	.00	.00
100230	4.781-001	1.582-001	7.535-006	7.327-030	.000	.000	.00	.00	.00	.00	.00	.00
110260	3.688-001	5.414-018	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
120270	7.375+001	6.854+001	3.545+001	9.094-001	2.593-010	.000	.00	.00	.00	.00	.00	.00
130260	1.015-005	1.015-005	1.015-005	1.015-005	1.015-005	1.015-005	1.02-005	1.02-005	1.02-005	1.02-005	1.02-005	1.01-005
130280	1.202+003	8.820+002	5.446+001	1.040-005	.000	.000	.00	.00	.00	.00	.00	.00
130290	1.070+001	9.631+000	3.721+000	1.689-002	3.236-016	.000	.00	.00	.00	.00	.00	.00
130300	1.254+001	1.574-004	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
140310	1.085+001	1.081+001	1.038+001	8.331+000	2.220+000	1.902-002	5.50-019	.00	.00	.00	.00	.00
210460	2.906+000	2.906+000	2.906+000	2.905+000	2.900+000	2.882+000	2.74+000	2.26+000	1.42-001	2.20-013	.00	.00
220450	1.656-001	1.650-001	1.595-001	1.322-001	4.292-002	7.473-004	6.31-018	.00	.00	.00	.00	.00
210470	3.000+000	3.000+000	2.996+000	2.975+000	2.854+000	2.455+000	7.38-001	7.11-003	5.94-026	.00	.00	.00
200450	1.336+000	1.336+000	1.336+000	1.336+000	1.335+000	1.330+000	1.30+000	1.17+000	2.83-001	2.40-007	.00	.00
210480	7.154+000	7.152+000	7.135+000	7.041+000	6.504+000	4.888+000	4.97-001	6.60-005	.00	.00	.00	.00
210490	3.056-001	3.019-001	2.709-001	1.483-001	3.986-003	8.846-009	.00	.00	.00	.00	.00	.00
210500	6.726-002	5.820-002	1.519-003	2.423-012	.000	.000	.00	.00	.00	.00	.00	.00
200470	2.647-002	2.646-002	2.644-002	2.630-002	2.547-002	2.271-002	9.07-003	2.52-004	1.45-026	.00	.00	.00
230490	7.529-002	7.529-002	7.529-002	7.529-002	7.525-002	7.513-002	7.42-002	7.06-002	3.50-002	3.51-005	3.60-035	.00
220510	4.972+000	4.412+000	1.506+000	3.846-003	1.069-018	.000	.00	.00	.00	.00	.00	.00
230520	8.781+001	7.300+001	1.384+001	1.343-003	1.124-027	.000	.00	.00	.00	.00	.00	.00
240510	1.893+000	1.893+000	1.893+000	1.891+000	1.881+000	1.846+000	1.59+000	8.85-001	2.07-004	.00	.00	.00
250530	2.636-007	2.636-007	2.636-007	2.636-007	2.636-007	2.636-007	2.64-007	2.64-007	2.64-007	2.64-007	2.64-007	2.64-007
250540	3.985+000	3.985+000	3.985+000	3.985+000	3.983+000	3.977+000	3.92+000	3.72+000	1.77+000	1.17-003	1.96-035	.00
260550	2.061+001	2.061+001	2.061+001	2.061+001	2.061+001	2.060+001	2.05+001	2.02+001	1.59+001	1.56+000	1.24-010	.00
260530	1.459-001	1.346-001	6.477-002	1.115-003	2.901-014	.000	.00	.00	.00	.00	.00	.00
250560	2.835+001	2.822+001	2.711+001	2.167+001	5.651+000	4.474-002	6.91-019	.00	.00	.00	.00	.00
250570	4.134-001	2.639-001	4.656-003	8.440-013	.000	.000	.00	.00	.00	.00	.00	.00
240550	9.324-003	7.659-003	1.304-003	6.966-008	1.622-033	.000	.00	.00	.00	.00	.00	.00
260590	2.477-001	2.477-001	2.477-001	2.476-001	2.468-001	2.439-001	2.22-001	1.55-001	9.00-004	9.92-026	.00	.00
TOTAL	1.605+003	1.230+003	2.983+002	1.782+002	1.324+002	7.492+001	3.17+001	2.85+001	1.82+001	1.63+000	7.24-002	6.49-002 DPS/CM3

1 - 2 (ZONE-INT)   1 YR OPERATING

NUCLIDE	0	1 M	10 M	1 H	6 H	1 D	1 W	1 MO	1 YR	10 YR	100 YR	1000 Y
40100	3.627-006	3.627-006	3.627-006	3.627-006	3.627-006	3.627-006	3.63-006	3.63-006	3.63-006	3.63-006	3.63-006	3.63-006
20060	4.185-004	2.109-026	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
30080	1.619+001	5.734-021	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
60140	6.810-002	6.810-002	6.810-002	6.810-002	6.810-002	6.810-002	6.81-002	6.81-002	6.81-002	6.80-002	6.73-002	6.03-002
110240	9.716+001	9.708+001	9.641+001	9.276+001	7.358+001	3.196+001	4.05-002	1.94-013	.00	.00	.00	.00
110250	6.474-001	3.238-001	6.332-004	5.665-019	.000	.000	.00	.00	.00	.00	.00	.00
100230	3.960-001	1.311-001	6.242-006	6.069-030	.000	.000	.00	.00	.00	.00	.00	.00
110260	3.055-001	4.485-018	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
120270	6.559+001	6.096+001	3.153+001	8.089-001	2.306-010	.000	.00	.00	.00	.00	.00	.00
130260	8.410-006	8.410-006	8.410-006	8.410-006	8.410-006	8.410-006	8.41-006	8.41-006	8.41-006	8.41-006	8.41-006	8.40-006

Table B-4 Continued

ZONE 1 ACTIVITY 1 YR OPERATING (IN CURIES)

	NUCLIDE	0	1 M	10 M	1 H	6 H	1 D	1 W	1 MO	1 YR	10 YR	100 YR	1000 Y
40100	BE 10	9.547-010	9.547-010	9.547-010	9.547-010	9.547-010	9.547-010	9.55-010	9.55-010	9.55-010	9.55-010	9.55-010	9.54-010
20060	HE 6	9.063-008	4.568-030	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
30080	LI 8	4.201-003	1.468-024	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
60140	C 14	1.570-005	1.570-005	1.570-005	1.570-005	1.570-005	1.570-005	1.57-005	1.57-005	1.57-005	1.57-005	1.55-005	1.39-005
110240	NA 24	2.536-002	2.534-002	2.517-002	2.422-002	1.921-002	8.343-003	1.06-005	5.06-017	.00	.00	.00	.00
110250	NA 25	1.661-004	8.308-005	1.625-007	1.454-022	.000	.000	.00	.00	.00	.00	.00	.00
100230	NE 23	1.016-004	3.363-005	1.602-009	1.558-033	.000	.000	.00	.00	.00	.00	.00	.00
110260	NA 26	7.840-005	1.151-021	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
120270	MG 27	1.716-002	1.595-002	8.249-003	2.116-004	6.035-014	.000	.00	.00	.00	.00	.00	.00
130260	AL 26	2.158-009	2.158-009	2.158-009	2.158-009	2.158-009	2.158-009	2.16-009	2.16-009	2.16-009	2.16-009	2.16-009	2.16-009
130280	AL 28	2.816-001	2.067-001	1.276-002	2.437-009	.000	.000	.00	.00	.00	.00	.00	.00
130290	AL 29	2.275-003	2.047-003	7.910-004	4.016-006	6.879-020	.000	.00	.00	.00	.00	.00	.00
130300	AL 30	2.666-003	3.345-008	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
140310	SI 31	2.320-003	2.309-003	2.220-003	1.781-003	4.746-004	4.064-006	1.16-022	.00	.00	.00	.00	.00
210460	SC 46	6.702-004	6.702-004	6.702-004	6.700-004	6.688-004	6.647-004	6.33-004	5.21-004	3.27-005	5.07-017	.00	.00
220450	TI 45	3.521-005	3.508-005	3.392-005	2.812-005	9.126-006	1.589-007	1.34-021	.00	.00	.00	.00	.00
210470	SC 47	6.806-004	6.805-004	6.797-004	6.750-004	6.473-004	5.568-004	1.67-004	1.61-006	1.31-029	.00	.00	.00
200450	CA 45	2.975-004	2.975-004	2.975-004	2.974-004	2.972-004	2.962-004	2.89-004	2.61-004	6.29-005	5.34-011	.00	.00
210480	SC 48	1.628-003	1.627-003	1.623-003	1.602-003	1.480-003	1.112-003	1.13-004	1.50-008	.00	.00	.00	.00
210490	SC 49	7.008-005	6.924-005	6.213-005	3.400-005	9.142-007	2.029-012	.00	.00	.00	.00	.00	.00
210500	SC 50	1.959-005	1.306-005	3.409-007	5.439-016	.000	.000	.00	.00	.00	.00	.00	.00
200470	CA 47	5.841-006	5.840-006	5.835-006	5.804-006	5.622-006	5.012-006	2.00-006	5.56-008	3.21-030	.00	.00	.00
230490	V 49	1.682-005	1.662-005	1.682-005	1.682-005	1.681-005	1.678-005	1.66-005	1.58-005	7.81-006	7.83-009	5.44-039	.00
220510	TI 51	1.147-003	1.018-003	3.476-004	8.880-007	2.466-022	.000	.00	.00	.00	.00	.00	.00
230520	V 52	1.904-002	1.563-002	3.000-003	2.912-007	2.437-031	.000	.00	.00	.00	.00	.00	.00
240510	CR 51	4.273-004	4.273-004	4.272-004	4.268-004	4.246-004	4.167-004	3.59-004	2.00-004	4.67-008	.00	.00	.00
250530	MN 53	5.802-011	5.802-011	5.802-011	5.802-011	5.802-011	5.802-011	5.80-011	5.80-011	5.80-011	5.80-011	5.80-011	5.80-011
250540	MN 54	9.270-004	9.270-004	9.270-004	9.269-004	9.265-004	9.249-004	9.13-004	8.66-004	4.11-004	2.73-007	.00	.00
260550	FE 55	4.525-003	4.525-003	4.525-003	4.525-003	4.524-003	4.522-003	4.50-003	4.43-003	3.49-003	3.42-004	2.73-014	.00
260530	FE 53	3.102-005	2.860-005	1.377-005	2.370-007	6.168-018	.000	.00	.00	.00	.00	.00	.00
250560	MN 56	6.459-003	6.430-003	6.176-003	4.937-003	1.267-003	1.019-005	1.57-022	.00	.00	.00	.00	.00
250570	MN 57	9.483-005	6.055-005	1.068-006	1.936-016	.000	.000	.00	.00	.00	.00	.00	.00
240550	CR 55	2.075-006	1.704-006	2.901-007	1.550-011	3.610-037	.000	.00	.00	.00	.00	.00	.00
260590	FE 59	5.297-005	5.297-005	5.296-005	5.294-005	5.277-005	5.216-005	4.76-005	3.32-005	1.92-007	2.12-029	.00	.00
	TOTAL	3.721-001	2.852-001	6.807-002	4.043-002	3.004-002	1.694-002	7.07-003	6.34-003	4.03-003	3.58-004	1.55-005	1.39-005

Table B-4 Continued

ZONE 1 BHP 1 YR OPERATING (IN KM3/KW)

	NUCLIDE	0	1 M	10 M	1 H	6 H	1 D	1 W	1 MO	1 YR	10 YR	100 YR	1000 Y
40100	BE 10	6.422-010	6.422-010	6.422-010	6.422-010	6.422-010	6.422-010	6.42-010	6.42-010	6.42-010	6.42-010	6.42-010	6.42-010
20060	HE 6	2.032-010	1.024-032	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
30080	LI 8	9.420-006	3.337-027	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
60140	C 14	1.056-008	1.056-008	1.056-008	1.056-008	1.056-008	1.056-008	1.06-008	1.06-008	1.06-008	1.05-008	1.04-008	9.35-009
110240	NA 24	3.412-004	3.410-004	3.386-004	3.258-004	2.584-004	1.122-004	1.42-007	6.81-019	.00	.00	.00	.00
110250	NA 25	3.725-007	1.863-007	3.643-010	3.260-025	.000	.000	.00	.00	.00	.00	.00	.00
100230	NE 23	2.279-007	7.541-008	3.591-012	3.492-036	.000	.000	.00	.00	.00	.00	.00	.00
110260	NA 26	1.758-007	2.561-024	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
120270	MG 27	3.848-005	3.576-005	1.850-005	4.745-007	1.353-016	.000	.00	.00	.00	.00	.00	.00
130260	AL 26	1.452-009	1.452-009	1.452-009	1.452-009	1.452-009	1.452-009	1.45-009	1.45-009	1.45-009	1.45-009	1.45-009	1.45-009
130280	AL 28	6.315-004	4.634-004	2.861-005	5.465-012	.000	.000	.00	.00	.00	.00	.00	.00
130290	AL 29	5.102-006	4.590-006	1.774-006	9.005-009	1.542-022	.000	.00	.00	.00	.00	.00	.00
130300	AL 30	5.978-006	7.501-011	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
140310	SI 31	5.201-006	5.178-006	4.977-006	3.992-006	1.064-006	9.113-009	2.64-025	.00	.00	.00	.00	.00
210460	SC 46	5.635-005	5.635-005	5.635-005	5.633-005	5.624-005	5.589-005	5.32-005	4.38-005	2.75-006	4.26-018	.00	.00
220450	TI 45	6.966-009	6.940-009	6.710-009	5.563-009	1.806-009	3.144-011	2.65-025	.00	.00	.00	.00	.00
210470	SC 47	2.289-006	2.289-006	2.286-006	2.270-006	2.177-006	1.873-006	5.63-007	5.41-009	4.41-032	.00	.00	.00
200450	CA 45	2.001-005	2.001-005	2.001-005	2.001-005	1.999-005	1.993-005	1.94-005	1.76-005	4.23-006	3.59-012	.00	.00
210480	SC 48	2.190-005	2.189-005	2.184-005	2.155-005	1.991-005	1.496-005	1.52-006	2.02-010	.00	.00	.00	.00
210490	SC 49	3.626-009	3.563-009	3.215-009	1.759-009	4.730-011	1.050-016	.00	.00	.00	.00	.00	.00
210500	SC 50	1.198-008	7.968-009	2.084-010	3.326-019	.000	.000	.00	.00	.00	.00	.00	.00
200470	CA 47	6.548-008	6.547-008	6.541-008	6.507-008	6.303-008	5.620-008	2.24-008	6.23-010	3.60-032	.00	.00	.00
230490	V 49	4.525-009	4.525-009	4.525-009	4.525-009	4.523-009	4.516-009	4.46-009	4.25-009	2.10-009	2.11-012	.00	.00
220510	TI 51	9.079-008	8.058-008	2.751-008	7.027-011	1.951-026	.000	.00	.00	.00	.00	.00	.00
230520	V 52	3.659-006	3.042-006	5.765-007	5.597-011	4.684-035	.000	.00	.00	.00	.00	.00	.00
240510	CR 51	3.593-007	3.593-007	3.592-007	3.589-007	3.570-007	3.504-007	3.02-007	1.68-007	3.93-011	.00	.00	.00
250530	MN 53	3.903-011	3.903-011	3.903-011	3.903-011	3.903-011	3.903-011	3.90-011	3.90-011	3.90-011	3.90-011	3.90-011	3.90-011
250540	MN 54	6.236-005	6.235-005	6.235-005	6.235-005	6.232-005	6.222-005	6.14-005	5.83-005	2.77-005	1.84-008	.00	.00
260550	FE 55	1.015-005	1.015-005	1.015-005	1.015-005	1.014-005	1.014-005	1.01-005	9.93-006	7.84-006	7.66-007	6.11-017	.00
260530	FE 53	6.956-008	6.413-008	3.087-008	5.314-010	1.383-020	.000	.00	.00	.00	.00	.00	.00
250560	MN 56	2.172-005	2.163-005	2.077-005	1.660-005	4.330-006	3.428-008	5.30-025	.00	.00	.00	.00	.00
250570	MN 57	2.126-007	1.358-007	2.395-009	4.341-019	.000	.000	.00	.00	.00	.00	.00	.00
240550	CR 55	4.652-009	3.821-009	6.505-010	3.476-014	.000	.000	.00	.00	.00	.00	.00	.00
260590	FE 59	1.782-006	1.782-006	1.781-006	1.780-006	1.775-006	1.754-006	1.60-006	1.12-006	6.47-009	7.14-031	.00	.00
	TOTAL	1.239-003	1.050-003	5.891-004	5.218-004	4.368-004	2.795-004	1.48-004	1.31-004	4.25-005	7.97-007	1.26-008	1.15-008

Table B-4 Continued

ZONE 1 AFTERHEAT 1 YR OPERATING (IN MW)

	NUCLIDE	0	1 M	10 M	1 H	6 H	1 D	1 W	1 MO	1 YR	10 YR	100 YR	1000 Y
40100	BE 10	1.019-018	1.019-018	1.019-018	1.019-018	1.019-018	1.019-018	1.02-018	1.02-018	1.02-018	1.02-018	1.02-018	1.02-018
20060	HE 6	8.446-016	4.257-038	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
30080	LI 8	1.586-010	5.618-032	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
60140	C 14	4.373-015	4.373-015	4.373-015	4.373-015	4.373-015	4.373-015	4.37-015	4.37-015	4.37-015	4.37-015	4.32-015	3.87-015
110240	NA 24	7.040-010	7.035-010	6.986-010	6.721-010	5.331-010	2.316-010	2.93-013	1.40-024	.00	.00	.00	.00
110250	NA 25	2.072-012	1.036-012	2.026-015	1.813-030	.000	.000	.00	.00	.00	.00	.00	.00
100230	NE 23	1.284-012	4.248-013	2.023-017	.000	.000	.000	.00	.00	.00	.00	.00	.00
110260	NA 26	2.300-012	3.377-029	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
120270	MG 27	1.607-010	1.493-010	7.722-011	1.981-012	5.649-022	.000	.00	.00	.00	.00	.00	.00
130260	AL 26	4.240-017	4.240-017	4.240-017	4.240-017	4.240-017	4.240-017	4.24-017	4.24-017	4.24-017	4.24-017	4.24-017	4.24-017
130280	AL 28	4.847-009	3.557-009	2.196-010	4.195-017	.000	.000	.00	.00	.00	.00	.00	.00
130290	AL 29	3.212-011	2.890-011	1.116-011	5.669-014	9.710-028	.000	.00	.00	.00	.00	.00	.00
130300	AL 30	4.309-011	5.406-016	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
140310	SI 31	6.779-012	6.749-012	6.487-012	5.204-012	1.387-012	1.188-014	3.44-031	.00	.00	.00	.00	.00
210460	SC 46	8.308-012	8.308-012	8.308-012	8.305-012	8.291-012	8.240-012	7.84-012	6.46-012	4.05-013	6.29-025	.00	.00
220450	TI 45	7.953-014	7.923-014	7.660-014	6.350-014	2.061-014	3.589-016	3.03-030	.00	.00	.00	.00	.00
210470	SC 47	1.110-012	1.110-012	1.109-012	1.101-012	1.056-012	9.084-013	2.73-013	2.62-015	2.14-038	.00	.00	.00
200450	CA 45	1.134-013	1.134-013	1.134-013	1.134-013	1.133-013	1.129-013	1.10-013	9.96-014	2.40-014	2.03-020	.00	.00
210480	SC 48	3.397-011	3.396-011	3.388-011	3.343-011	3.088-011	2.321-011	2.36-012	3.14-016	.00	.00	.00	.00
210490	SC 49	3.446-013	3.404-013	3.054-013	1.672-013	4.494-015	9.974-021	.00	.00	.00	.00	.00	.00
210500	SC 50	5.544-013	3.697-013	9.647-015	1.539-023	.000	.000	.00	.00	.00	.00	.00	.00
200470	CA 47	4.588-014	4.587-014	4.583-014	4.559-014	4.416-014	3.937-014	1.57-014	4.36-016	2.52-038	.00	.00	.00
230490	V 49	1.795-014	1.795-014	1.795-014	1.795-014	1.794-014	1.791-014	1.77-014	1.68-014	8.33-015	8.36-018	.00	.00
220510	TI 51	5.951-012	5.261-012	1.803-012	4.606-015	1.279-030	.000	.00	.00	.00	.00	.00	.00
230520	V 52	2.823-010	2.347-010	4.447-011	4.317-015	3.614-039	.000	.00	.00	.00	.00	.00	.00
240510	CR 51	7.346-014	7.345-014	7.344-014	7.338-014	7.300-014	7.164-014	6.17-014	3.44-014	8.03-018	.00	.00	.00
250530	MN 53	6.123-020	6.123-020	6.123-020	6.123-020	6.123-020	6.123-020	6.12-020	6.12-020	6.12-020	6.12-020	6.12-020	6.12-020
250540	MN 54	4.589-012	4.589-012	4.588-012	4.588-012	4.586-012	4.576-012	4.52-012	4.29-012	2.04-012	1.35-015	.00	.00
260550	FE 55	1.878-012	1.878-012	1.878-012	1.878-012	1.877-012	1.876-012	1.87-012	1.84-012	1.45-012	1.42-013	1.13-023	.00
260530	FE 53	4.171-013	3.846-013	1.851-013	3.167-015	8.293-026	.000	.00	.00	.00	.00	.00	.00
250560	MN 56	9.687-011	9.644-011	9.263-011	7.404-011	1.931-011	1.529-013	2.36-030	.00	.00	.00	.00	.00
250570	MN 57	6.173-013	3.941-013	6.952-015	1.260-024	.000	.000	.00	.00	.00	.00	.00	.00
240550	CR 55	1.310-014	1.076-014	1.832-015	9.787-020	.000	.000	.00	.00	.00	.00	.00	.00
260590	FE 59	4.082-013	4.062-013	4.082-013	4.080-013	4.066-013	4.020-013	3.67-013	2.56-013	1.48-015	1.64-037	.00	.00
	TOTAL	6.395-009	4.835-009	1.203-009	8.036-010	6.012-010	2.712-010	1.77-011	1.30-011	3.93-012	1.48-013	4.36-015	3.92-015

Table B-4 Continued

ZONE 1 BETA HEAT 1 YR OPERATING (IN MW)

NUCLIDE		0	1 M	10 M	1 H	6 H	1 D	1 W	1 MO	1 YR	10 YR	100 YR	1000 Y
40100	BE 10	1.019-018	1.019-018	1.019-018	1.019-018	1.019-018	1.019-018	1.02-018	1.02-018	1.02-018	1.02-018	1.02-018	1.02-018
20060	HE 6	8.446-016	4.257-038	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
30080	LI 8	1.564-010	5.540-032	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
60140	C 14	4.373-015	4.373-015	4.373-015	4.373-015	4.373-015	4.373-015	4.37-015	4.37-015	4.37-015	4.37-015	4.32-015	3.87-015
110240	NA 24	8.420-011	8.414-011	8.356-011	8.039-011	6.377-011	2.770-011	3.51-014	1.68-025	.00	.00	.00	.00
110250	NA 25	1.698-012	8.491-013	1.661-015	1.486-030	.000	.000	.00	.00	.00	.00	.00	.00
100230	NE 23	1.187-012	3.950-013	1.871-017	.000	.000	.000	.00	.00	.00	.00	.00	.00
110260	NA 26	1.454-012	2.135-029	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
120270	MG 27	7.000-011	6.505-011	3.365-011	8.632-013	2.461-022	.000	.00	.00	.00	.00	.00	.00
130260	AL 26	5.936-018	5.936-018	5.936-018	5.936-018	5.936-018	5.936-018	5.94-018	5.94-018	5.94-018	5.94-018	5.94-018	5.93-018
130280	AL 28	1.875-009	1.376-009	8.496-011	1.623-017	.000	.000	.00	.00	.00	.00	.00	.00
130290	AL 29	1.242-011	1.118-011	4.319-012	2.193-014	3.756-028	.000	.00	.00	.00	.00	.00	.00
130300	AL 30	3.619-011	4.541-016	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
140310	SI 31	6.779-012	6.749-012	6.487-012	5.204-012	1.367-012	1.188-014	3.44-031	.00	.00	.00	.00	.00
210460	SC 46	3.619-013	3.619-013	3.619-013	3.618-013	3.612-013	3.590-013	3.42-013	2.81-013	1.76-014	2.74-026	.00	.00
220450	TI 45	7.953-014	7.923-014	7.660-014	6.350-014	2.061-014	3.589-016	3.03-030	.00	.00	.00	.00	.00
210470	SC 47	6.391-013	6.390-013	6.382-013	6.338-013	6.078-013	5.229-013	1.57-013	1.51-015	1.23-038	.00	.00	.00
200450	CA 45	1.134-013	1.134-013	1.134-013	1.134-013	1.133-013	1.129-013	1.10-013	9.96-014	2.40-014	2.03-020	.00	.00
210480	SC 48	1.662-012	1.662-012	1.658-012	1.636-012	1.511-012	1.136-012	1.16-013	1.53-017	.00	.00	.00	.00
210490	SC 49	3.441-013	3.400-013	3.051-013	1.670-013	4.489-015	9.962-021	.00	.00	.00	.00	.00	.00
210500	SC 50	1.839-013	1.227-013	3.201-015	5.107-024	.000	.000	.00	.00	.00	.00	.00	.00
200470	CA 47	1.011-014	1.011-014	1.010-014	1.005-014	9.731-015	8.677-015	3.47-015	9.62-017	5.55-039	.00	.00	.00
230490	V 49	.000	.000	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
220510	TI 51	5.951-012	5.261-012	1.803-012	4.606-015	1.279-030	.000	.00	.00	.00	.00	.00	.00
230520	V 52	1.187-010	9.870-011	1.871-011	1.816-015	1.520-039	.000	.00	.00	.00	.00	.00	.00
240510	CR 51	.000	.000	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
250530	MN 53	.000	.000	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
250540	MN 54	.000	.000	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
260550	FE 55	.000	.000	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
260530	FE 53	2.074-013	1.913-013	9.206-014	1.585-015	4.124-026	.000	.00	.00	.00	.00	.00	.00
250560	MN 56	2.876-011	2.863-011	2.750-011	2.198-011	5.732-012	4.538-014	7.01-031	.00	.00	.00	.00	.00
250570	MN 57	.000	.000	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
240550	CR 55	1.310-014	1.076-014	1.832-015	9.787-020	.000	.000	.00	.00	.00	.00	.00	.00
260590	FE 59	3.705-014	3.705-014	3.705-014	3.703-014	3.691-014	3.649-014	3.33-014	2.32-014	1.35-016	1.48-038	.00	.00
	TOTAL	2.402-009	1.661-009	2.643-010	1.115-010	7.356-011	2.993-011	8.01-013	4.10-013	4.61-014	4.37-015	4.33-015	3.88-015

NO. OF ISOTOPES IN 2 IS 5 2 7

Table B-4 Continued

SUMMARY OF TDF ALUMINUM: CARBON + BORAL + F.W. + BORAL + WATER + BORON PART-I

1 YR OPERATION 3.156+007 SEC

TIME AFTER SHUTDOWN SEC		TOTAL ACT CI	TOTAL BHP KM3/KW	TOTAL AHT MW	BETA AHT MW	PER ACT CI/W	% AHT %
.000	0	8.569+003	6.197+001	2.031-004	4.800-005	5.764-001	1.366+000
6.000+001	1 M	7.068+003	5.862+001	1.377-004	2.667-005	4.768-001	9.265-001
6.000+002	10 M	5.362+003	5.446+001	1.191-004	1.899-005	3.607-001	8.010-001
3.600+003	1 H	3.679+003	4.910+001	1.006-004	1.219-005	2.475-001	6.765-001
2.160+004	6 H	2.880+003	3.895+001	7.933-005	9.496-006	1.937-001	5.336-001
8.640+004	1 D	1.261+003	1.722+001	3.445-005	4.115-006	8.479-002	2.318-001
6.048+005	1 W	2.084+001	5.486-001	8.859-008	5.829-009	1.402-003	5.959-004
2.630+006	1 MO	1.582+001	4.910-001	3.973-008	3.367-010	1.064-003	2.672-004
3.156+007	1 YR	7.379+000	2.163-001	1.693-008	3.743-011	4.964-004	1.139-004
3.156+008	10 YR	4.213-001	1.358-003	1.916-010	1.134-012	2.834-005	1.289-006
3.156+009	100 YR	4.400-004	2.855-004	7.919-012	1.133-012	2.960-008	5.327-008
3.156+010	1000 Y	4.375-004	2.849-004	7.911-012	1.131-012	2.943-008	5.322-008

APPENDIX B. SAMPLE PROBLEM #3

Sample problem number 3 consists of calculating the activity of a LiPb-Pb-DT inertial confinement fusion (ICF) target. The target consists of 4 material zones with the inner zone containing a 50-50 mixture of deuterium and tritium. The next two zones contain a LiPb compound (natural Li and Pb), both zones having different densities, and the outer zone contains natural Pb. Since DCDLIB does not contain any hydrogen isotopes, the first 11 original mesh cells are collapsed (using LCLPS) and the zone is treated as a void. Also, as can be seen from the input, this case uses the pulse mode approximation within the code.

Table B-5 Input for Sample Problem #3

```
     HIBALL TARGET(WIS) PB-TAMFER,LIPB-PUSHER
           1     3     3     2     3    25     1     0     6     2    25    21
           1     0     C     1     2     1     1    11
               5.0         14.1         0.0         14.1    3.55E+1
        0.010925    0.010925            0.0       4.0E-4    1.00E-6
           1    11     1        0.010925    0.01636     1
           2     3     1         0.01636    0.22360     2
           3    11     1          0.2236    0.29780     1
        100.  .168  0.00
        100.  .168 1.021
           1  3006     1   4.656E+21
           1  3007     1   5.809E+22
           2 82204     1   4.568E+19
           2 82206     1   7.752E+20
           2 82207     1   7.109E+20
           2 82208     1   1.685E+21
     10-6 s     1.000E-6
      LIPB-PB HIBALL TARGET PULSE
        7.701E+1    5.358E+1    3.138E+1    2.068E+1    1.456E+1    1.058E+1
        7.840E+0    6.041E+0    5.419E+0    5.324E+0    1.145E+0    1.334E+0
        1.142E+1    1.042E+1    2.790E+1    1.906E+1    1.374E+1    1.354E+1
        1.167E+1    4.388E+0    1.132E+0    2.886E-1    7.300E-2    1.810E-2
        5.399E-3
@XQT DKRMAIN11.ABS4
```

Table B-6 Output for Sample Probem #3

```
HIBALL TARGET(WIS) PB-TAMPER,LIPB-PUSHER

 A ONE HERE INDICATES COLLAPSING OF FLUXES  1 11

                    PROBLEM RUN ID          1
       LNK    LINK TO THE OTHER SOLUTION              3
       LGE    1/2/3  = SLAB/CYLINDER/SPHERE           3
       LFX    1/2    = DKR/ANISN (FORMATTED           2
       IZM    NUMBER OF ZONES                         3
       INT    NUMBER OF INTERVALS                    25
       NOP    NUMBER OF OPERATING TIMES               1
       NAS    NUMBER OF AFTER SHUTDOWN TIMES         12
       NNC    NUMBER OF MATERIALS(NUCLIDES)           6
       NCMP   NUMBER OF COMPOSITION TABLE             2
       IGN    NUMBER OF NEUTRON GROUPS             * 25
       IGG    NUMBER OF PHOTON GROUPS              * 43

                 ZONE    1   2   3
         LI    3006      *   *
         LI    3007      *   *
         PB   82204      *   *   *
         PB   82206      *   *   *
         PB   82207      *   *   *
         PB   82208      *   *   *
```

Table B-6 Continued

REACTOR SYSTEM PARAMETERS

RADIUS OF THE PLASMA	.01	CM
RADIUS OF THE FIRST WALL	.01	CM
RADIUS OF THE TORUS	.00	CM

FIRST WALL AREA	1.500-007	M2
NEUTRON WALL LOADING	5.000+000	MW/M2
TOTAL OPERATING POWER	7.499-007	MW
FLUX CONVERSION FACTOR	3.550+001	
ACCURCY LIMIT	4.000-004	
TEST IRRADIATION TIME	1.000-006	

OPERATING TIME	1		AFTER SHUTDOWN TIME	12	
10-6 s	1.000-006	SECOND	0	.000	SECOND
			1 M	6.000+001	SECOND
			10 M	6.000+002	SECOND
			1 H	3.600+003	SECOND
			6 H	2.160+004	SECOND
			1 D	8.640+004	SECOND
			1 W	6.048+005	SECOND
			1 MO	2.630+006	SECOND
			1 YR	3.156+007	SECOND
			10 YR	3.156+008	SECOND
			100 YR	3.156+009	SECOND
			1000 Y	3.156+010	SECOND

VOLUME OF ZONE

ZONE	1	1.288-005	CM3
ZONE	2	4.681-002	CM3
ZONE	3	6.380-002	CM3

Table B-6 Continued

NUCLIDE NO. DENSITY(10**18)

KZA	ZONE 1	2	3
3006	465599.992	782.208	.000
3007	5808999.938	9759.120	.000
82204	4568.000	7.674	46.639
82206	77519.999	130.234	791.479
82207	71089.999	119.431	725.829
82208	168499.998	283.080	1720.385

```
                REFERENCE FLUX
        LIPB-PB HIBALL TARGET PULSE
@ADD,PD TEST.FLUX1

                FLUX READING

        TITLE OF FLUX FILE :   WIS TARGET 1MG DT RFFUEL=2 RRPUSH=1


                   39 INTERVALS READ FROM FLUX ( 39, 25)
    39 INTERVALS HAVE BEEN COLLAPSED TO  25 INTERVALS
```

Table B-6 Continued

NUCLEAR DATA TABLE

LKZA	NKT	KT=	1 21	2 22	3 23	4 24	5 25	6 26	7 27	8 28	9 29	10	11	12	13	14	15	16	17	18	19
20060	2	R	0	0																	
20080	2	R	0	0																	
30060	5	S	X	X	X					X		X									
30070	6	S	X	X		X	X				X	X									
30080	2	R	0	0																	
30090	2	R	0	0																	
40090	5	S	X	X	X			X		X											
40100	2	R	0	0																	
50100	5	S	X		X		X			X			X								
50110	6	S	X	X	X	X		X		X											
60120	4	S	X	X						X					X						
60130	2	S	X	X																	
60140,	2	R	0	0																	
70140	8	S	X	X	X	X	X	X		X			X								
70160	2	R	0	0																	
80160	5	S	X	X	X		X			X											
80190	2	R	0	0																	
90180	2	R	0		0																
90190	8	S	X	X	X	X	X	X		X	X										
90200	2	R	0	0																	
100230	2	R	0	0																	
110240	2	R	0	0																	
110250	2	R	0	0																	
110260	2	R	0	0																	
120240	3	S	X	X	X																
120250	3	S	X	X	X																
120260	4	S	X	X	X					X											
120270	2	R	0	0																	
130260	2	R	0		0																
130270	8	S	X	X	X	X	X	X		X									X		
130280	2	R	0	0																	
130290	2	R	0	0																	
130300	2	R	0	0																	
130301	2	R	0				0														
140280	4	S	X	X	X					X											
140290	4	S	X	X	X					X											
140300	4	S	X	X	X					X											
140310	2	R	0	0																	
170360	3	R	0	0	0																
170380	2	R	0	0																	

Table B-6 Continued
CHAIN CONSTRUCTION PROCEDURES

```
3006 CHAIN
                                                      30070 - 4
                                                      20060 - 3
                                                      10030 - 2
                                                      10010 - 1
                                    20060  *PASS  2
                                    30070 TEST    0
                                                      30060 - 1
  101
       30060   103   .8557+000   .3628+004   .8557+000   .0000         101  1
       20060  1022   .8482+002   .8557+000   .0000       .8557+000     101  2

  $ YES          NUMBER OF CHAINS =  1

3007 CHAIN
                                                      30080 - 5
                                                      30060 - 4
                                                      20060 - 3
                                                      10030 - 2
                                                      10020 - 1
                                    20060  *PASS  2
                                    30060 TEST    0
                                    30080  *PASS  2
                                                      30060 - 1
                                                      40080 - 2
                                    30060 TEST    0
  201
       30070   105   .0000       .3393+004   .0000       .0000         201  1
       20060  1022   .4902+002   .8557+000   .0000       .8557+000     201  2
  202
       30070   102   .0000       .3393+004   .0000       .0000         202  1
       30080  1022   .1662+000   .8232+000   .0000       .8232+000     202  2

  $ YES          NUMBER OF CHAINS =  2

82204 CHAIN
                                                     812040 - 5
                                                     822030 - 6
                                                     812030 - 4
                                                     812020 - 3
                                                     802010 - 2
                                                     802000 - 1
                                   812040  *PASS  2
                                   822030  *PASS  2
                                                     822040 - 1
```

Table B-6 Continued
EXECUTING PROCEDURES FOR ZONE 1

LKZA	LRZ	AI	AK	BI	BK	YO	YT	Z,I	OP
MXC = 1									
								< 1, 1>	10-6 s
30060	103	8.557-001	6.257+003	8.557-001	.000	4.656+023	4.627+023		
20060	1022	2.829+002	8.557-001	.000	8.557-001	.000	1.313+020		
								< 1, 2>	10-6 s
30060	103	8.557-001	5.390+003	8.557-001	.000	4.656+023	4.631+023		
20060	1022	2.457+002	8.557-001	.000	8.557-001	.000	1.141+020		
								< 1, 3>	10-6 s
30060	103	8.557-001	4.728+003	8.557-001	.000	4.656+023	4.634+023		
20060	1022	2.164+002	8.557-001	.000	8.557-001	.000	1.005+020		
								< 1, 4>	10-6 s
30060	103	8.557-001	4.203+003	8.557-001	.000	4.656+023	4.636+023		
20060	1022	1.930+002	8.557-001	.000	8.557-001	.000	8.965+019		
								< 1, 5>	10-6 s
30060	103	8.557-001	3.775+003	8.557-001	.000	4.656+023	4.638+023		
20060	1022	1.737+002	8.557-001	.000	8.557-001	.000	8.072+019		
								< 1, 6>	10-6 s
30060	103	8.557-001	3.417+003	8.557-001	.000	4.656+023	4.640+023		
20060	1022	1.576+002	8.557-001	.000	8.557-001	.000	7.325+019		
								< 1, 7>	10-6 s
30060	103	8.557-001	3.113+003	8.557-001	.000	4.656+023	4.642+023		
20060	1022	1.439+002	8.557-001	.000	8.557-001	.000	6.690+019		
								< 1, 8>	10-6 s
30060	103	8.557-001	2.850+003	8.557-001	.000	4.656+023	4.643+023		
20060	1022	1.321+002	8.557-001	.000	8.557-001	.000	6.143+019		
								< 1, 9>	10-6 s
30060	103	8.557-001	2.618+003	8.557-001	.000	4.656+023	4.644+023		
20060	1022	1.218+002	8.557-001	.000	8.557-001	.000	5.664+019		
								< 1,10>	10-6 s
30060	103	8.557-001	2.412+003	8.557-001	.000	4.656+023	4.645+023		
20060	1022	1.127+002	8.557-001	.000	8.557-001	.000	5.240+019		
								< 1,11>	10-6 s
30060	103	8.557-001	2.226+003	8.557-001	.000	4.656+023	4.646+023		
20060	1022	1.045+002	8.557-001	.000	8.557-001	.000	4.861+019		
MXC = 2									
								< 1, 1>	10-6 s
30070	105	.000	1.340+004	.000	.000	5.809+024	5.731+024		
20060	1022	3.047+002	8.557-001	.000	8.557-001	.000	1.758+021		
								< 1, 2>	10-6 s
30070	105	.000	1.158+004	.000	.000	5.809+024	5.742+024		
20060	1022	2.623+002	8.557-001	.000	8.557-001	.000	1.515+021		
								< 1, 3>	10-6 s

Table B-6 Continued

INTERVAL ACTIVITY

1 - 1 (ZONE=INT) 10-6 s OPERATING

NUCLIDE	0	1 M	10 M	1 H	6 H	1 D	1 W	1 MO	1 YR	10 YR	100 YR	1000 Y	
20060	1.617+021	8.148-002	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00	
30080	1.996+018	7.071-004	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00	
812040	1.257+009	1.257+009	1.257+009	1.257+009	1.256+009	1.256+009	1.25+009	1.24+009	1.05+009	2.00+008	1.32+001	.00	
822030	1.257+015	1.256+015	1.254+015	1.240+015	1.160+015	9.128+014	1.34+014	7.47+010	2.43-036	.00	.00	.00	
802030	4.052+011	4.052+011	4.051+011	4.049+011	4.037+011	3.992+011	3.65+011	2.58+011	1.78+009	1.08-012	.00	.00	
822050	9.336+006	9.336+006	9.336+006	9.336+006	9.336+006	9.336+006	9.34+006	9.34+006	9.34+006	9.34+006	9.34+006	9.34+006	
802050	5.798+015	5.074+015	1.529+015	1.947+012	8.303-006	.000	.00	.00	.00	.00	.00	.00	
822090	1.509+014	1.504+014	1.457+014	1.222+014	4.247+013	9.465+011	5.76-002	.00	.00	.00	.00	.00	
TOTAL	1.619+021	6.481+015	2.928+015	1.365+015	1.203+015	9.141+014	1.34+014	3.34+011	2.83+009	2.10+008	9.34+006	9.34+006	DPS/CM3

1 - 2 (ZONE=INT) 10-6 s OPERATING

NUCLIDE	0	1 M	10 M	1 H	6 H	1 D	1 W	1 MO	1 YR	10 YR	100 YR	1000 Y	
20060	1.394+021	7.026-002	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00	
30080	1.722+018	6.100-004	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00	
812040	1.083+009	1.083+009	1.083+009	1.083+009	1.082+009	1.082+009	1.08+009	1.07+009	9.01+008	1.72+008	1.14+001	.00	
822030	1.090+015	1.090+015	1.088+015	1.075+015	1.006+015	7.917+014	1.16+014	6.46+010	2.11-036	.00	.00	.00	
802030	3.503+011	3.503+011	3.502+011	3.500+011	3.490+011	3.451+011	3.16+011	2.23+011	1.54+009	9.32-013	.00	.00	
822050	8.061+006	8.061+006	8.061+006	8.061+006	8.061+006	8.061+006	8.06+006	8.06+006	8.06+006	8.06+006	8.06+006	8.06+006	
802050	5.017+015	4.391+015	1.323+015	1.684+012	7.185-006	.000	.00	.00	.00	.00	.00	.00	
822090	1.339+014	1.334+014	1.293+014	1.084+014	3.768+013	8.397+011	5.11-002	.00	.00	.00	.00	.00	
TOTAL	1.396+021	5.615+015	2.540+015	1.186+015	1.044+015	7.929+014	1.17+014	2.89+011	2.45+009	1.81+008	8.06+006	8.06+006	DPS/CM3

1 - 3 (ZONE=INT) 10-6 s OPERATING

NUCLIDE	0	1 M	10 M	1 H	6 H	1 D	1 W	1 MO	1 YR	10 YR	100 YR	1000 Y	
20060	1.221+021	6.152-002	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00	
30080	1.510+018	5.351-004	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00	
812040	9.471+008	9.471+008	9.471+008	9.470+008	9.469+008	9.466+008	9.44+008	9.33+008	7.88+008	1.51+008	9.98+000	.00	
822030	9.588+014	9.586+014	9.567+014	9.461+014	8.852+014	6.965+014	1.02+014	5.70+010	1.85-036	.00	.00	.00	
802030	3.072+011	3.072+011	3.072+011	3.071+011	3.061+011	3.027+011	2.77+011	1.95+011	1.35+009	8.18-013	.00	.00	
822050	7.067+006	7.067+006	7.067+006	7.067+006	7.067+006	7.067+006	7.07+006	7.07+006	7.07+006	7.07+006	7.07+006	7.07+006	
802050	4.405+015	3.855+015	1.161+015	1.479+012	6.308-006	.000	.00	.00	.00	.00	.00	.00	
822090	1.200+014	1.196+014	1.159+014	9.716+013	3.378+013	7.528+011	4.58-002	.00	.00	.00	.00	.00	
TOTAL	1.222+021	4.933+015	2.234+015	1.045+015	9.193+014	6.975+014	1.03+014	2.53+011	2.14+009	1.58+008	7.07+006	7.07+006	DPS/CM3

1 - 4 (ZONE=INT) 10-6 s OPERATING

NUCLIDE	0	1 M	10 M	1 H	6 H	1 D	1 W	1 MO	1 YR	10 YR	100 YR	1000 Y
20060	1.082+021	5.456-002	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
30080	1.342+018	4.755-004	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
812040	8.390+008	8.390+008	8.390+008	8.390+008	8.389+008	8.386+008	8.36+008	8.26+008	6.98+008	1.34+008	8.85+000	.00
822030	8.536+014	8.534+014	8.517+014	8.423+014	7.880+014	6.200+014	9.11+013	5.07+010	1.65-036	.00	.00	.00
802030	2.728+011	2.728+011	2.728+011	2.727+011	2.718+011	2.688+011	2.46+011	1.74+011	1.20+009	7.26-013	.00	.00
822050	6.273+006	6.273+006	6.273+006	6.273+006	6.273+006	6.273+006	6.27+006	6.27+006	6.27+006	6.27+006	6.27+006	6.27+006

Table B-6 Continued

ZONE 1 ACTIVITY 10-6 s OPERATING (IN CURIES)

	NUCLIDE		0	1 M	10 M	1 H	6 H	1 D	1 W	1 MO	1 YR	10 YR	100 YR	1000 Y
20060	HE	6	3.103+005	1.564-017	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
30080	LI	8	3.850+002	1.364-019	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
812040	TL	204	2.402-007	2.402-007	2.402-007	2.402-007	2.401-007	2.401-007	2.39-007	2.37-007	2.00-007	3.83-008	2.53-015	.00
822030	PB	203	2.454-001	2.454-001	2.449-001	2.422-001	2.266-001	1.783-001	2.62-002	1.46-005	.00	.00	.00	.00
802030	HG	203	7.826-005	7.826-005	7.826-005	7.822-005	7.797-005	7.711-005	7.05-005	4.98-005	3.44-007	2.08-028	.00	.00
822050	PB	205	1.800-009	1.800-009	1.800-009	1.800-009	1.800-009	1.800-009	1.80-009	1.80-009	1.80-009	1.80-009	1.80-009	1.80-009
802050	HG	205	1.123+000	9.829-001	2.961-001	3.770-004	1.608-021	.000	.00	.00	.00	.00	.00	.00
822090	PB	209	3.132-002	3.121-002	3.024-002	2.536-002	8.315-003	1.965-004	1.20-017	.00	.00	.00	.00	.00
	TOTAL		3.107+005	1.260+000	5.713-001	2.680-001	2.355-001	1.785-001	2.63-002	6.46-005	5.45-007	4.01-008	1.80-009	1.80-009

Table B-6 Continued

ZONE 1 BHP 10-6 s OPERATING (IN KM3/KW)

	NUCLIDE	0	1 M	10 M	1 H	6 H	1 D	1 W	1 MO	1 YR	10 YR	100 YR	1000 Y
20060	HE 6	1.379+007	6.952-016	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
30080	LI 8	1.711+004	6.063-018	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
812040	TL 204	3.559-004	3.559-004	3.558-004	3.558-004	3.558-004	3.557-004	3.55-004	3.50-004	2.96-004	5.67-005	3.75-012	.00
822030	PB 203	5.454+000	5.453+000	5.442+000	5.382+000	5.035+000	3.962+000	5.82-001	3.24-004	.00	.00	.00	.00
802030	HG 203	5.218-002	5.218-002	5.217-002	5.215-002	5.199-002	5.141-002	4.70-002	3.32-002	2.29-004	1.39-025	.00	.00
822050	PB 205	2.400-005	2.400-005	2.400-005	2.400-005	2.400-005	2.400-005	2.40-005	2.40-005	2.40-005	2.40-005	2.40-005	2.40-005
802050	HG 205	4.992+001	4.369+001	1.316+001	1.676-002	7.148-020	.000	.00	.00	.00	.00	.00	.00
822090	PB 209	4.177+002	4.162+002	4.032+002	3.381+002	1.175+002	2.620+000	1.59-013	.00	.00	.00	.00	.00
	TOTAL	1.381+007	4.654+002	4.219+002	3.436+002	1.226+002	6.633+000	6.29-001	3.39-002	5.49-004	8.07-005	2.40-005	2.40-005

Table B-6 Continued

ZONE 1 AFTERHEAT 10-6 s OPERATING (IN MW)

NUCLIDE			0	1 M	10 M	1 H	6 H	1 D	1 W	1 MO	1 YR	10 YR	100 YR	1000 Y
20060	HE	6	2.892-003	1.457-025	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
30080	LI	8	1.453-005	5.149-027	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
812040	TL	204	5.126-016	5.126-016	5.126-016	5.126-016	5.125-016	5.123-016	5.11-016	5.05-016	4.27-016	8.17-017	5.40-024	.00
822030	PB	203	1.803-009	1.802-009	1.799-009	1.779-009	1.664-009	1.309-009	1.92-010	1.07-013	.00	.00	.00	.00
802030	HG	203	1.809-013	1.809-013	1.809-013	1.808-013	1.803-013	1.783-013	1.63-013	1.15-013	7.94-016	4.82-037	.00	.00
822050	PB	205	.000	.000	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
802050	HG	205	6.078-009	5.320-009	1.602-009	2.041-012	8.704-030	.000	.00	.00	.00	.00	.00	.00
822090	PB	209	.000	.000	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
TOTAL			2.906-003	7.122-009	3.401-009	1.781-009	1.664-009	1.310-009	1.92-010	2.23-013	1.22-015	8.17-017	5.40-024	.00

Table B-6 Continued

ZONE 1 BETA HEAT & 10-6 s OPERATING (IN MW)

	NUCLIDE	0	1 M	10 M	1 H	6 H	1 D	1 W	1 MO	1 YR	10 YR	100 YR	1000 Y
20060	HE 6	2.892-003	1.457-025	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
30080	LI 8	1.433-005	5.078-027	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
812040	TL 204	5.126-016	5.126-016	5.126-016	5.126-016	5.125-016	5.123-016	5.11-016	5.05-016	4.27-016	8.17-017	5.40-024	.00
822030	PB 203	1.397-009	1.396-009	1.394-009	1.378-009	1.289-009	1.015-009	1.49-010	8.30-014	.00	.00	.00	.00
802030	HG 203	5.150-014	5.150-014	5.149-014	5.147-014	5.131-014	5.074-014	4.64-014	3.28-014	2.26-016	1.37-037	.00	.00
822050	PB 205	.000	.000	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
802050	HG 205	4.727-009	4.137-009	1.246-009	1.587-012	6.769-030	.000	.00	.00	.00	.00	.00	.00
822090	PB 209	.000	.000	.000	.000	.000	.000	.00	.00	.00	.00	.00	.00
	TOTAL	2.906-003	5.533-009	2.640-009	1.380-009	1.289-009	1.015-009	1.49-010	1.16-013	6.53-016	8.17-017	5.40-024	.00

NO. OF ISOTOPES IN 2 IS 6 3 14

Table B-6 Continued

SUMMARY OF HIBALL TARGET(WIS) PB-TAMPER,LIPB-PUSHER

10-6 s OPERATION 1.000-006 SEC

TIME AFTER SHUTDOWN SEC		TOTAL ACT CI	TOTAL BHP KM3/KW	TOTAL AHT MW	BETA AHT MW	PER ACT CI/W	% AHT %
.000	0	3.423+005	1.522+007	3.202-003	3.202-003	4.565+005	4.270+005
6.000+001	1 M	1.541+000	5.670+002	8.715-009	6.771-009	2.055+000	1.162+000
6.000+002	10 M	6.990-001	5.138+002	4.163-009	3.231-009	9.320-001	5.551-001
3.600+003	1 H	3.279-001	4.184+002	2.180-009	1.689-009	4.373-001	2.907-001
2.160+004	6 H	2.882-001	1.494+002	2.038-009	1.579-009	3.843-001	2.717-001
8.640+004	1 D	2.186-001	8.104+000	1.603-009	1.242-009	2.915-001	2.138-001
6.048+005	1 W	3.214-002	7.704-001	2.357-010	1.825-010	4.286-002	3.143-002
2.630+006	1 MO	7.899-005	4.142-002	2.724-013	1.422-013	1.053-004	3.633-005
3.156+007	1 YR	6.660-007	6.705-004	1.491-015	7.967-016	8.881-007	1.989-007
3.156+008	10 YR	4.887-008	9.843-005	9.961-017	9.961-017	6.517-008	1.328-008
3.156+009	100 YR	2.195-009	2.927-005	6.591-024	6.591-024	2.927-009	8.788-016
3.156+010	1000 Y	2.195-009	2.927-005	.000	.000	2.927-009	.000