DKR Radioactivity Calculation Code for Fusion Reactors

T.Y. Sung and W.F. Vogelsang

September 1976

UWFDM-170

FUSION TECHNOLOGY INSTITUTE

UNIVERSITY OF WISCONSIN

MADISON WISCONSIN

DKR Radioactivity Calculation Code for Fusion Reactors

T.Y. Sung and W.F. Vogelsang

Fusion Technology Institute
University of Wisconsin
1500 Engineering Drive
Madison, WI 53706

http://fti.neep.wisc.edu

September 1976

UWFDM-170

DKR: A Radioactivity Calculation Code for Fusion Reactors

Tak Yun Sung

William F. Vogelsang

September 1976

UWFDM-170

Fusion Technology Program
Nuclear Engineering Department
University of Wisconsin
Madison, Wisconsin 53706

This research was supported by a grant from the Energy Research and Development Administration.

Abstract

DKR is a point activity calculation code which constructs the 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. DOSE is an auxiliary program to DKR; with either forward or adjoint γ-ray flux, it computes the spatially dependent or time dependent dose rates, respectively.

Table of Contents

DKR

A Radioactivity Calculation Code for Fusion Reactors

T. Y. Sung and W. F. Vogelsang

1. Introduction

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

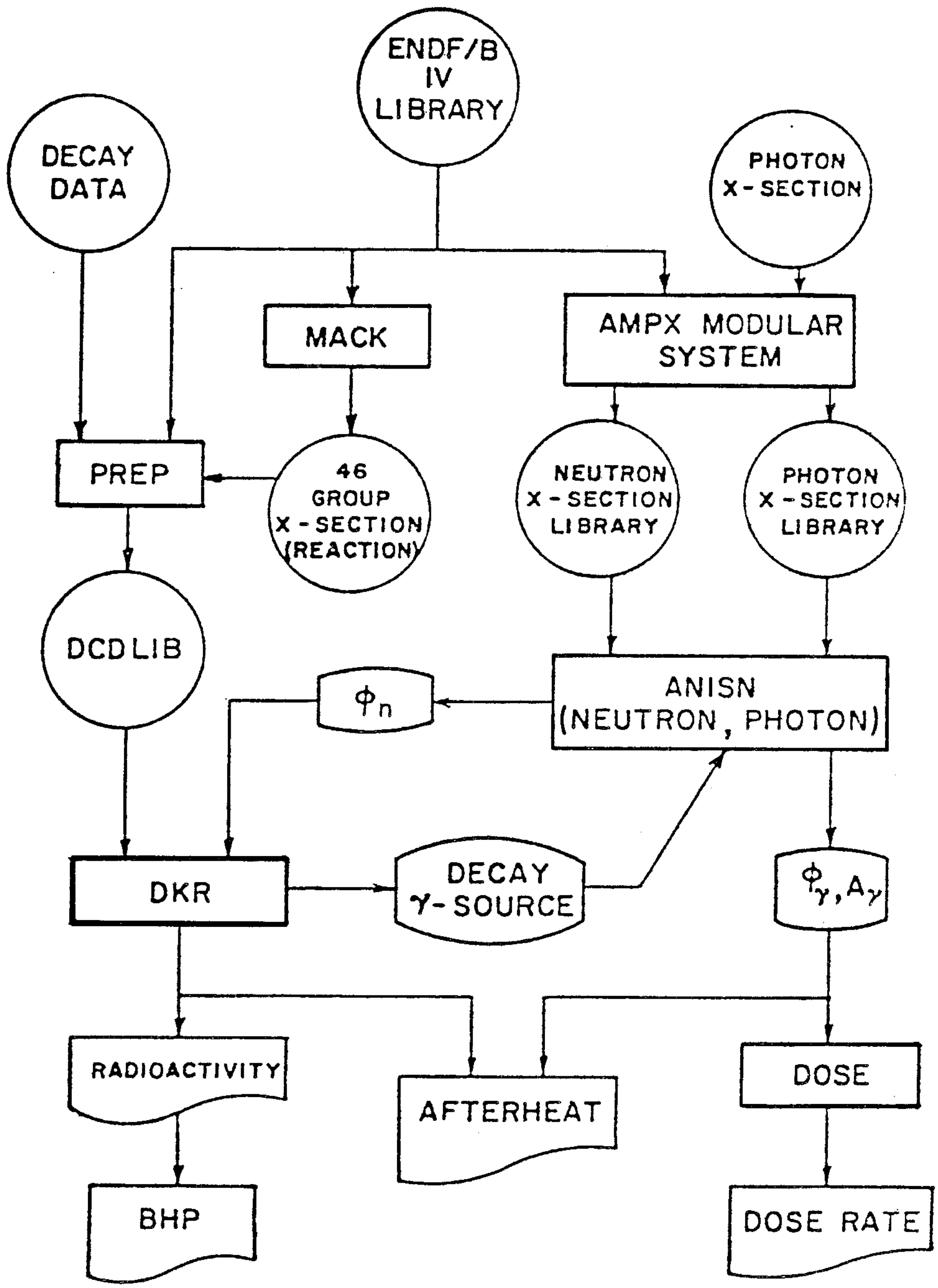
Previous activation calculations performed in conjunction with fusion reactor designs[1-5] agree on the order of magnitude in radioactivity and afterheat, in general, but show wide variation in the radiological hazard which is sensitive to the concentration of each isotope. The differences observed in 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

Fig. 1-1.

CALCULATIONAL SCHEME OF ACTIVITY IN FUSION REACTOR



ACTL

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 radiosiotopes 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. Fig. 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. 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 are performed with DKR code. Decay γ-ray sources are also produced as one of the outputs from DKR.

Table 1-1. Definition of Transmutation Types

Transmutation types identified by an integer KT

KT	REACTION TYPE†	CHANGE IN†† KZA,	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)α	-2004		22
10	(n,2n)α	-2005		24
11	(n,2α)	-4007		108
12	(n,2α)t	-5010		113
13	(n,3α)	-6011		109
14	(n,3α)n	-6012		23
15	(n,n')*		+ 1	
16	(n,γ)*	+ 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	(β^-)*	+1000	+ 1	
27	(β^+)*, (EC)* (=18)	-1000	+ 1	
28	n (=4)	- 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

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 ANISN forward calculation and β-ray heating in the DKR results.

DOSE is an auxiliary program to DKR, which reads the dimensions and compositions of the system considered, and the γ-ray flux. Either forward flux or adjoint flux from the ANISN calculation is an input to the DOSE program which is used to compute spatially dependent or time dependent dose rates, respectively.

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 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.

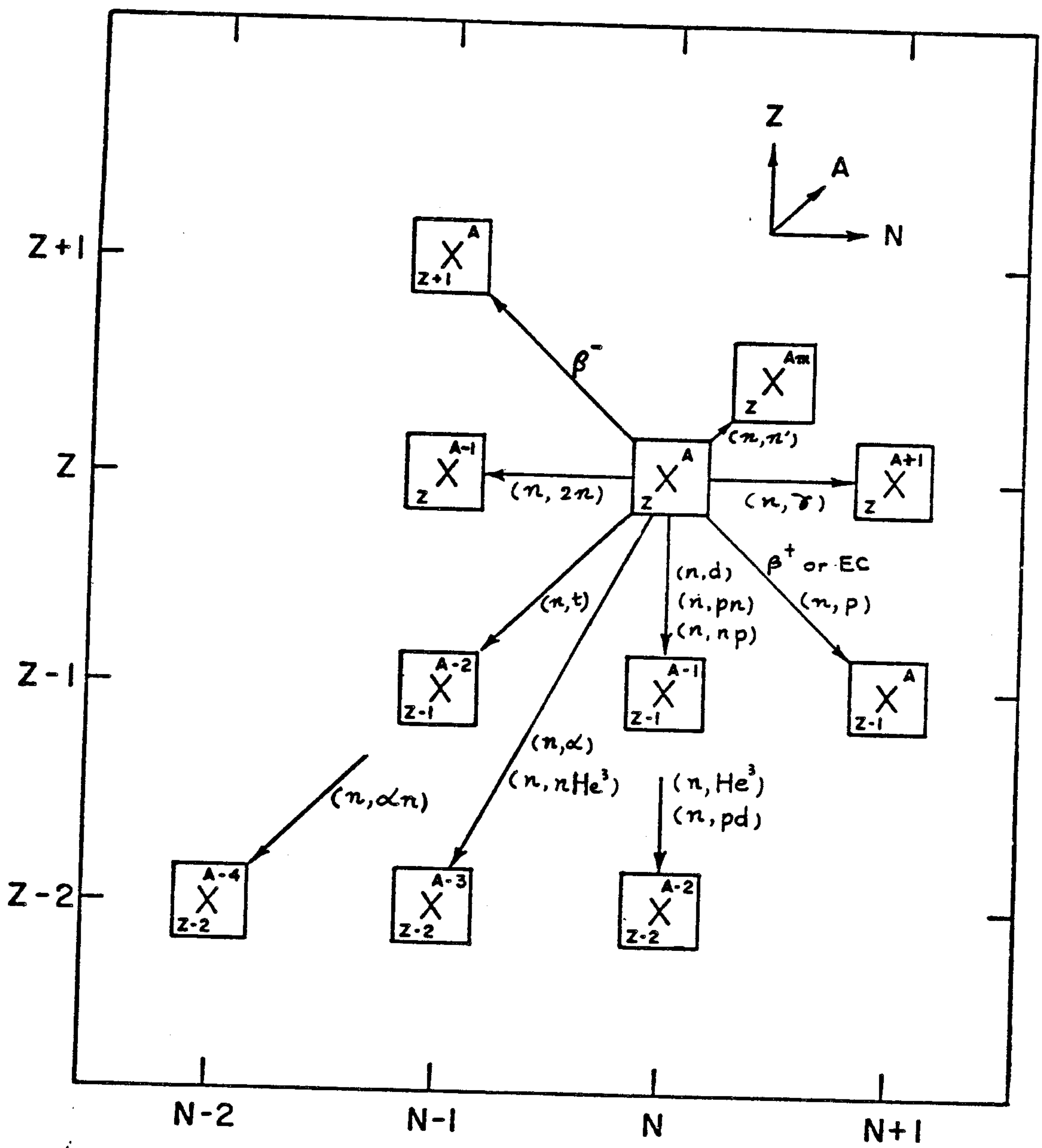
About twenty neutron reactions are possible when nuclei are bombarded by neutrons in the energy range below 20 MeV. [13] Among them, the (n,3n) reaction and (n,n't) reaction are energetically the most unfavorable and generally do not occur below about 15 MeV incident neutron energy, because of their high threshold energy. An important exception is the (n,n't) reaction of ^{7}Li which was treated as an (n,n'α) 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

Fig. 2-1. Transmutation Products by Neutron Reaction and by Radioactive Decay



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

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 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 k^{th} nuclide, γ_{k-1}^{k} is 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

$$\frac{d}{dt} N_k = Q_k - \beta_k N_k \, , \qquad (2.3)$$

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 equation (2.3) is

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

The first term in equation (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 equations (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 equation (2.5) with no source related terms, then

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

Rearranging equation (2.7) yields

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

$$+ \gamma_{k-1}^k \sum_{j=1}^{k-1} \frac{a_j^{k-1}}{\beta_k-\beta_j} \, e^{-\beta_j(t-t_o)} .$$

And

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

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

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

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 $\beta_k \simeq \beta_j$ 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 equation (2.9) or (2.10) cannot be used in this case because of the singularity. Going back to the equation (2.7) and considering the case where β_k is very close to $\beta_j (j \neq k)$,

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

the quantity in the brackets becomes

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

$$= (t-t_o)\cdot e^{-\beta_j(t-t_o)} \sum_{n=1}^{\infty} \frac{[\beta_j(t-t_o)-\beta_k(t-t_o)]^{n-1}}{n!} \quad . \quad (2.12)$$

Substituting equation (2.12) in equation (2.11) results

$$N_k(t) = N_k(t_o)\, e^{-\beta_k(t-t_o)}$$

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

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

The modified recursion coefficients are

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

$$a_j^k = \gamma_{k-1}^{k} \frac{a_j^{k-1}}{\beta_k-\beta_j} \quad , \; \beta_k \neq \beta_j \qquad (2.14b)$$

$$a_j^k = \gamma_{k-1}^{k}\, a_j^{k-1}\cdot(t-t_o) \sum_{n=1}^{\infty} \frac{[\beta_j(t-t_o)-\beta_k(t-t_o)]^{n-1}}{n!} \quad ,$$

$$\beta_k \simeq \beta_j \quad . \qquad (2.14c)$$

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

$$b^j_{j-1} = S_j \ ,$$

we get

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

where the coefficients are

$$b^k_k = -\gamma^k_{k-1} \sum_{\substack{i=j-1 \\ \beta_k \neq \beta_i}}^{k-1} \frac{b^{k-1}_i}{\beta_k - \beta_i} \tag{2.16a}$$

$$b^k_i = \gamma^k_{k-1} \frac{b^{k-1}_i}{\beta_k - \beta_i} \ , \ \beta_k \neq \beta_i \tag{2.16b}$$

$$b^k_i = \gamma^k_{k-1} \, b^{k-1}_i \, (t-t_o) \sum_{n=1}^{\infty} \frac{[\beta_i(t-t_o) - \beta_k(t-t_o)]^{n-1}}{n!} \ , \ \beta_k \simeq \beta_i \ . \tag{2.16c}$$

If β_k is very close to β_i, b^k_i is computed in the same way as a^k_j in equation (2.14). When solving the chains, it is not always easy to choose an appropriate form from equations (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 balance equation (2.3) is

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

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

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 equation (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 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 an (n,p) reaction followed by a β^- decay or an (n,2n) reaction followed by an (n,γ) reaction.

2.3 Radioactivity

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

$$N_k(\underline{r},t) = \sum_{j=1}^{n} a_j^n e^{-\beta_j t} \qquad (2.18)$$

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

$$R(t) = \int_{\underline{r}} \sum_{\substack{k:\text{all} \\ \text{radioisotopes}}} \lambda_k N_k(\underline{r},t)\, d\underline{r} \qquad (2.19)$$

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 are also important. Among the many quantities which have been used to try and estimate the radiological hazard more accurately, BHP has widely been 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 human body.

The BHP of a given system is

$$B(t) = \int_{\underline{r}} \sum_{\substack{k:\text{all} \\ \text{radioisotopes}}} \xi_k \lambda_k N_k(\underline{r},t)\,d\underline{r} \qquad (2.20)$$

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 birth place; and secondly, to apply 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 there is small γ-ray leakage at the boundary. 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

$$H(t) = H_\gamma(t) + \int_{\underline{r}} \sum_{\substack{k:\text{all} \\ \text{radioisotopes}}} \bar{E}_k \lambda_k N_k(\underline{r},t)d\underline{r} \qquad (2.21)$$

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

$$L\ \phi_\gamma = \Omega\ . \qquad (2.22)$$

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

$$\Omega_g(\underline{r},t) = \sum_{\substack{k:\text{all} \\ \text{radioisotope}}} y_g^k \lambda_k N_k(\underline{r},t) \qquad (2.23)$$

where y_g^k is the gamma yield in the g^{th} group by the decay of the nuclide k.

Decay gamma heating is given by

$$H_\gamma(t) = \int_{\underline{r}} \sum_\ell N_\ell(\underline{r}) \int_0^\infty K_\ell(\underline{r},E)\ \phi_\gamma(\underline{r},E,t)dE\ d\underline{r} \qquad (2.24)$$

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

$$D(t) = \frac{\int_{\underline{r}} \Gamma\, \phi_\gamma(\underline{r},t) d\underline{r}}{\int_{\underline{r}} d\underline{r}} \tag{2.25}$$

where Γ is the flux-to-dose 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 dose rate functional I_d is

$$I_D = (\Gamma, \phi_\gamma) - (\phi_\gamma^*, \Delta L_{\phi_\gamma}) \tag{2.26}$$

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

$$L^* \phi_\gamma = \Gamma \tag{2.27}$$

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 constitutent 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

Fig. 3-1. Flowchart for Calculating Chains

(a)

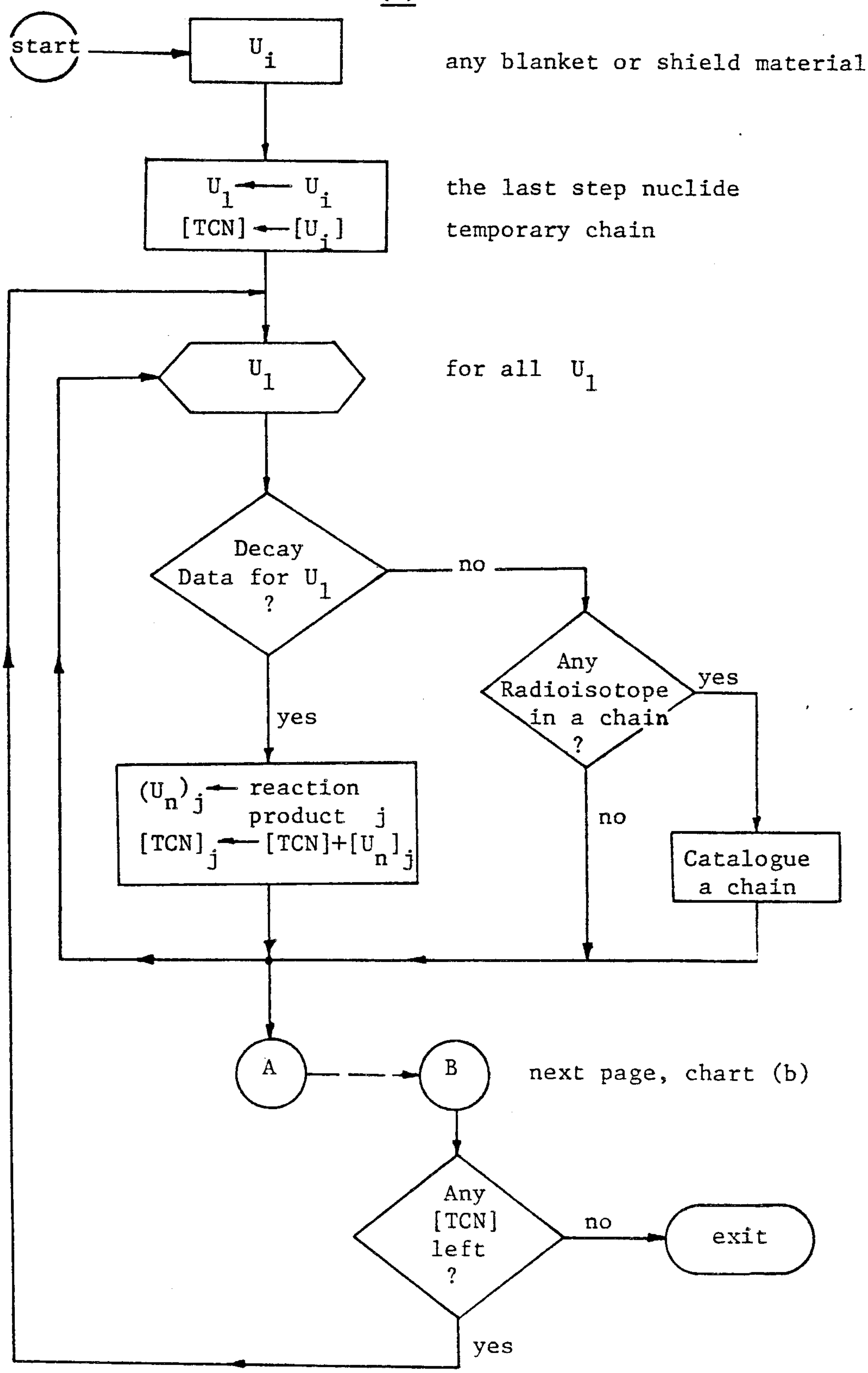
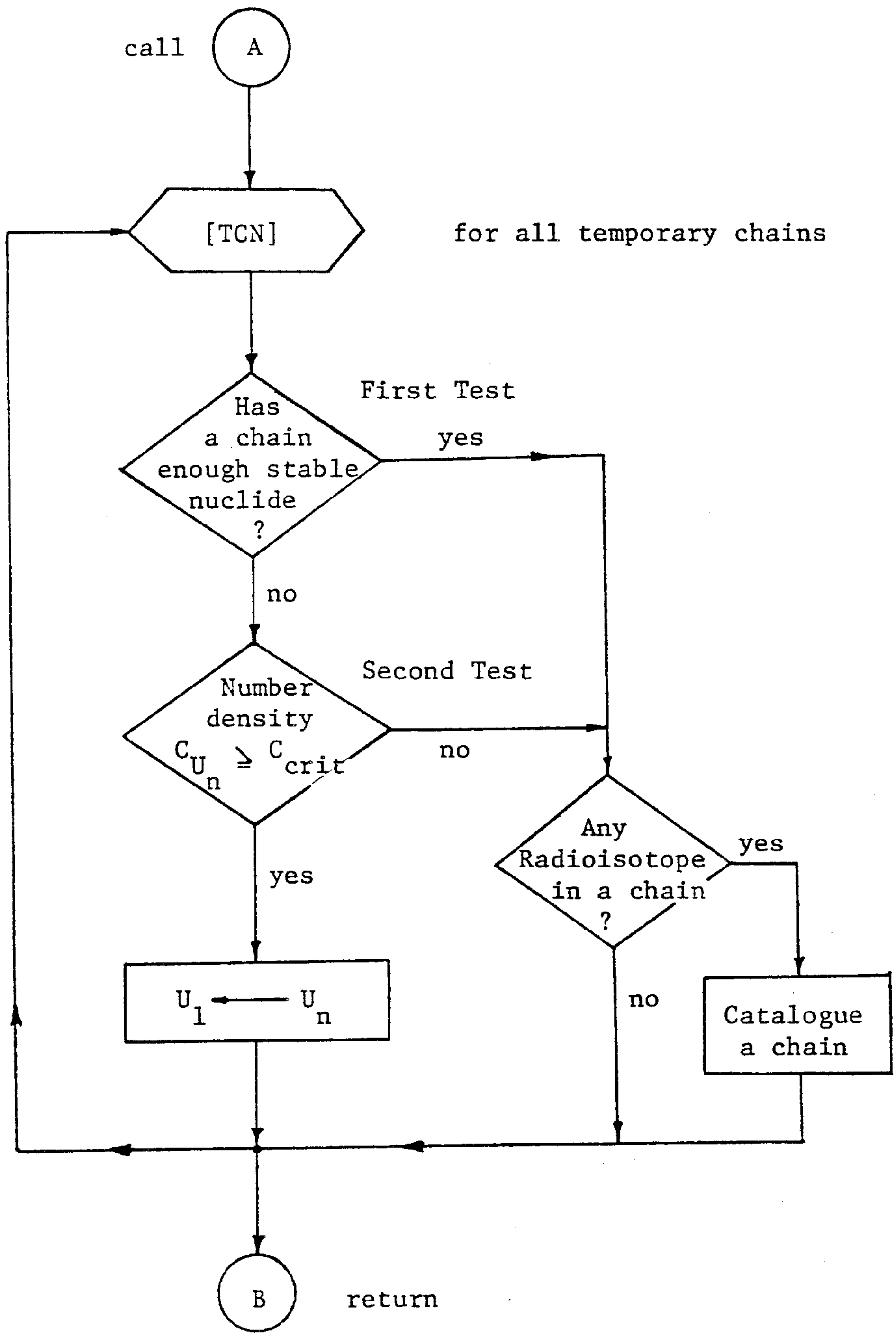


Fig. 3-1. (b)


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

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.

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 a 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 on material. Consequently a reference operating time is set to 10^8 seconds, about 3.2 years. 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 criteria, the chain is terminated. 10 appm of the product nuclide is set as the number density criteria, 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 liner 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 the equations (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 the Taylor's formula with remainder.[19] The expression for an exponential is

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

for some ξ between 0 and x,

where $\frac{x^n e^{-\xi}}{(n+1)!}$ term represents the error bound. If only the first three terms

Table 3-1. Error Range due to approximation

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

$$= \sum_{n=1}^{\infty} \frac{(-x)^{n-1}}{n!}$$

x	$A = \frac{1-e^{-x}}{x}$	$B = 1- \frac{x}{2!} + \frac{x^2}{3!}$	$\frac{B-A}{A} x100(\%)$
0.1	0.95162582	0.95166667	0.43×10^{-2}
0.05	0.97541151	0.97541667	0.53×10^{-3}
0.01	0.99501663	0.99501667	0.40×10^{-5}
0.005	0.99750416	0.99750417	0.10×10^{-5}
0.001	0.99950020	0.99950017	-0.30×10^{-5}
0.0005	0.99975020	0.99975004	-0.16×10^{-5}
0.0001	0.99995000	0.99995000	–

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 equations (2.14c) and (2.16c) are applied to the computer algorithm instead those of equations (2.14b) and (2.16b) in section 2. Table 3-1 shows the error range due to this approximation.

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.

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 46 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[24,25,26] 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 coupled 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; if NOP=0, nine built-in times are used.
NAS	43-48	number of after shutdown time; if NAS=0, twelve built-in times are used.
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(6I6)

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
LPRT4	19-24	print option for chain results
LFLX	25-30	reference flux option for chain calculation: if LFLX=0, uniform flux of 10^{14} n/cm^2-sec is used as reference flux; if LFLX=1, the first wall flux of UWMAK-I design is used; if LFLX=2, Card No. 11 and 12 are supplied as a reference flux set.
LFCF	31-36	flag for FCF: if LFCF=0, FCF is calculated in the program; if LFCF=1, FCF given in Card No. 4 is used.

Card No. 4(5F12.5)

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 converstion factor; if LFCF=1, FCF value other than zero must be given.

Card No. 5(3F12.5)

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

Card No. 6(3I6, 6X, 2F12.3)

As many cards as IZM are required

IZ	1-6	zone number
NZI	7-12	number of intervals in a zone
LCØ	13-18	flag for zone radioactivity calculation
RRI	25-36	inner radius of zone in cm
RRO	37-48	outer radius of zone in cm

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

4.2 Detailed Data Notes

More detailed information for some parameters, variables, and arrays is described below. The parameters 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 a 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 is 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 this option, the segment PICKUP in the program is bypassed.

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.

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.

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

Card No. 10(A6, E12.3)

As many cards as NAS are required,

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

Table 4-1. Dimension Parameters

MZN	Number of zones ($\leq$ 17)
MRG	Number of intervals ($\leq$ 81)
MRZ	Maximum number of intervals in a zone ($\leq$ 20)
MOP	Number of operating times ($\leq$ 9)
MAS	Number of after shutdown times ($\leq$ 12)
MKT	Number of transmutation types (= 29)
MXN	Number of neutron reaction types (= 19)
MCP	Number of composition tables ($\leq$ 11)
MNN	Number of nuclides in the system ($\leq$ 39)
MRD	Number of radioactive reaction products ($\leq$ 61)
MPX	Number of radioisotopes for which data is given in BLOCK DATA ($\leq$ 146)
MGX	Number of radioisotopes for which decay γ-ray data is given in BLOCK DATA ($\leq$ 95)
MNG	Number of neutron energy groups (= 46)
MGG	Number of γ-ray energy groups (= 43)
MND	Number of nuclides for which data is given in DCDLIB ($\leq$ 222)
MC	Number of chains from one nuclide ($\leq$ 33)
MK	Number of steps in a chain ($\leq$ 9)

If LNK=4, same as LNK=3 case, except it uses the decay chains and destruction data table from the preceding run.

LFX

Neutron flux is provided in either a DKR format or a 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 46 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/sec, and the real fluxes are calculated by multiplying 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/sec-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/sec 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.

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.

Table 4-2. Built-in Times

9 Operating Times		12 After Shutdown Times	
1.	1 day = 8.640×10^4 sec	1.	0
2.	2 wk = 1.315×10^6 sec	2.	1 m = 6.000×10 sec
3.	1 mo = 2.630×10^6 sec	3.	10 m = 6.000×10^2 sec
4.	6 mo = 1.578×10^7 sec	4.	1 h = 3.600×10^3 sec
5.	1 yr = 3.156×10^7 sec	5.	6 h = 2.160×10^4 sec
6.	2 yr = 6.312×10^7 sec	6.	1 d = 8.640×10^4 sec
7.	4 yr = 1.262×10^8 sec	7.	1 wk = 6.048×10^5 sec
8.	8 yr = 2.525×10^8 sec	8.	1 mo = 2.630×10^6 sec
9.	16 yr = 5.050×10^8 sec	9.	1 yr = 3.156×10^7 sec
		10.	10 yr = 3.156×10^8 sec
		11.	100 yr = 3.156×10^9 sec
		12.	1000 yr = 3.156×10^{10} sec

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. Reference flux is used to produce this table, which can be used as the table for reaction rates or transmutation rates.

Next section shows the procedures of 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 radionuclides 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 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

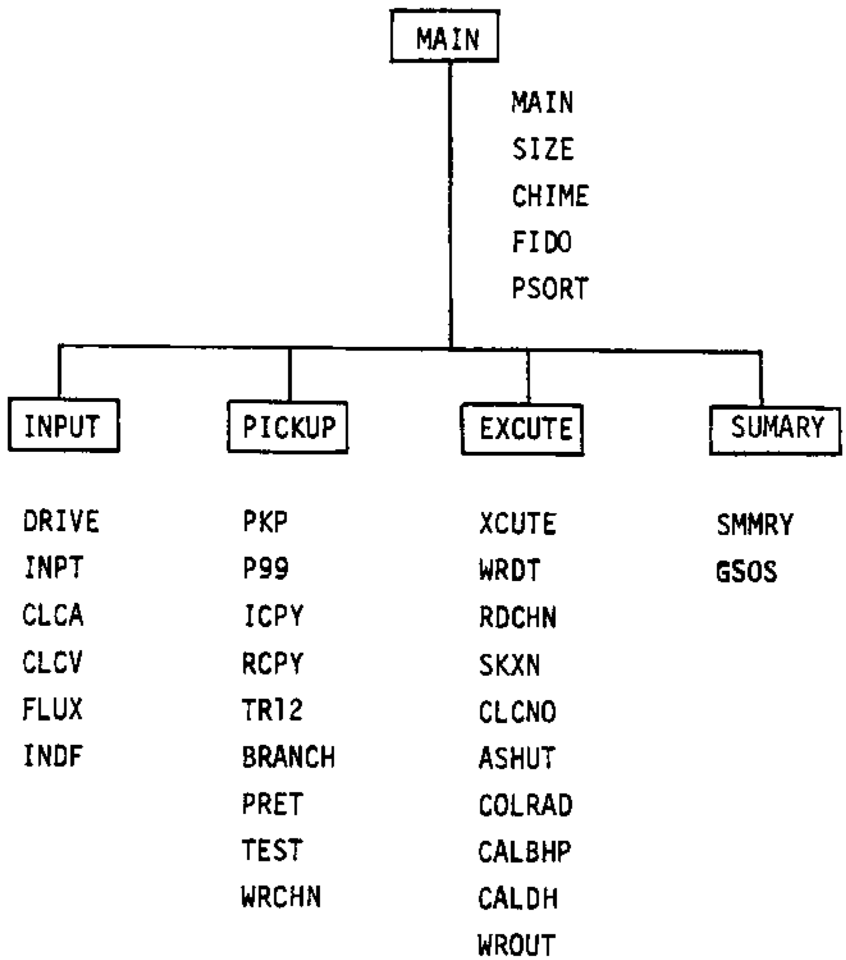
5.1 Program Features

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 65 K 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 are shown in Fig. 5-1.

Figure 5-1

Overlay Structure



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):

Table 5-1 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 parts of the program.

CHIME (1) is for the clock setting, and

CHIME (2) is for the collapsed time after CHIME (1).

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 5-2.

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.

Table 5-2

Geometry	First Wall Area	Zone Volume
slab	1	$(R_o - R_i)\cdot 1$
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

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.

<u>BRANCH</u>

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.

<u>ICPY</u>, <u>RCPY</u>, and <u>TR12</u>

During the chain construction, these subroutines are used to transfer information for each chain.

<u>PRET</u> and <u>TEST</u>

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.

<u>WRCHN</u>

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

<u>XCUTE</u>, <u>RDCHN</u>, and <u>SKXN</u>

Subroutine XCUTE is the administration subroutine for calculating the radioactivity, biological hazard potential (BHP), and afterheat. RDCHN retrieves the chains and SKXN retrives 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.

<u>CLCNO</u> and <u>ASHUT</u>

Each decay chain is solved in the subroutine CLCNO to get the number density of nuclides at designed 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 printout 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 system.

GSOS

Subroutine GSOS edits the γ-ray source into a format which can be accepted as an ANISN input for a γ-ray transport calculations.

5.3 Error Messages

This section contains error messages due to inconsistent input data.

Table 5-3. 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
321	RDCHN	Incorrect transfer of chain information
331	SKXN	Incorrect number of intervals
332	SKXN	Error in the transmutation rate table

References

1. B. Badger et al., "A Wisconsin Tokamak Reactor Design, UWMAK-I," UWFDM-68, Vol. 1 (revised March 1974), Vol. 2 (May 1975), Nuclear Engineering Dept., University of Wisconsin.

2. A. P. Fraas, "Conceptual Design of the Blanket and Shield Region and Related Systems for a Full Scale Toroidal Fusion Reactor," ORNL-TM-3096 (May 1973).

3. J. D. Lee, "Geometry and Heterogeneous Effects on the Neutronic Performance of a Yin Yang Mirror-Reactor Blanket," UCRL-75141 (October 1973).

4. R. G. Mills, Editor, "A Fusion Power Plant," MATT-1050 (August 1974).

5. J. R. Powell, F. T. Miles, A. Aronson and W. E. Winsche, "Studies of Fusion Reactor Blanket with Minimum Radioactive Inventory and with Tritium Breeding in Solid Lithium Compounds," BNL-18236 (June 1973).

6. T. Y. Sung, "Radioactivity Calculations in Fusion Reactors," Ph.D. Thesis, University of Wisconsin (1976).

7. M. K. Drake, Editor, "Data Formats and Procedures for the ENDF Neutron Cross Section Library (ENDF-102, Vol.-I)," BNL-50274 (October 1970), revised (April 1974).

8. D. J. Dudziak, Editor, "ENDF Formats and Procedures for Photon Production and Interaction Data (ENDF-102, Vol.-II)," LA-4549 (July 1971).

9. T. Y. Sung and W. F. Vogelsang, "Decay Chain Data Library for Radioactivity Calculations," UWFDM-171 (September 1976), University of Wisconsin.

10. N. M. Green, J. L. Lucins, L. M. Petrie, W. E. Ford, III, J. E. White, and R. Q. Wright, "AMPX: A Modular Code System for Generating Coupled Multi-group Neutron-Gamma Libraries from ENDF/B," ORNL-TM-3706 (March 1976).

11. M. A. Abdou, C. W. Maynard, and R. Q. Wright, "MACK: A Program to Calculate Neutron Energy Release Parameters (Fluence-to-Kerma Factors) and Multi-group Neutron Reaction Cross Section from Nuclear Data in ENDF Format," ORNL-TM-3994 (January 1973).

12. W. W. Engle, Jr., "A User's Manual for ANISN," K-1693 (March 1967).

13. S. Pearlstein, "Neutron Induced Reactions in Medium Mass Nuclei," J. of Nucl. Energy 27, 81 (1973).

14. S. J. Ball and R. K. Adams, "MATEXP: A General Purpose Program for Solving Ordinary Differential Equation by Matrix Exponential Method," ORNL-TM-1933 (August 1967).

15. M. J. Bell, "ORIGEN-The ORNL Isotope Generation and Depletion Code," ORNL-4628 (May 1973).

16. D. Steiner and A. P. Fraas, "Preliminary Observations on the Radiological Implications of Fusion Power," Nuclear Safety 13, 5 (1972).

17. ICRU, "Radiation Quantities and Units," ICRU-19 (July 1971).

18. W. F. Vogelsang, G. L. Kulcinski, R. G. Lott, and T. Y. Sung, "Transmutations, Radioactivity and Afterheat in a D-T Tokamak Fusion Reactor," Nucl. Tech. 22, 379 (June 1974).

19. B. Carnahan, H. A. Luther, and J. O. Wilkes, "Applied Numerical Methods," John-Wiley (1969).

20. M. R. Bhat, B. A. Magruno, S. Pearlstein, and F. M. Scheffel, "Nuclear Data for CTR Related Projects," BNL-19344 (October 1974).

21. D. J. Hughes et al., "Neutron Cross Sections, 2nd Ed.," BNL-325 (1958) and supplement.

22. C. M. Lederer, J. M. Hollander, and I. Perlman, "Table of Isotopes," John-Wiley (1967).

23. USNRC, "Standards for Protection Against Radiation," NRC Rules and Regulations, Title 10, Part 20 (1975).

24. D. J. Dudziak and R. A. Krakowski, "Radioactivity Induced in Theta-Pinch Reactor," Nucl. Tech. 25, 32 (January 1975).

25. R. W. Conn, T. Y. Sung, and M. A. Abdou, "Comparative Study of Radioactivity and Afterheat in Several Fusion Reactor Blanket Designs," Nucl. Tech. 26, 391 (August 1975).

26. D. Steiner, "The Neutron Induced Activity and Decay Power of Niobium Structure of a D-T Fusion Blanket," ORNL-TM-3094 (August 1970).

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 an 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 65 K words. DKR accommodates nuclear data in DCDLIB,which are in 29 transmutation types, with 46 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 65 K words of core memory is 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," UWFDM-170 (this report).

(2) T. Y. Sung and W. F. Vogelsang, "Decay Chain Data Library for Radioactivity Calculations," UWFDM-171 (September 1976).

(3) T. Y. Sung, "Radioactivity Calculations in Fusion Reactors," Ph.D. Thesis, University of Wisconsin (1976).

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. Sample Input Data

```
                                                                                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. Sample Output

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	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																	

*** 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					
	↗ stable nuclide		30060	3.626-10	1	TOTX	
	(11 ← radioisotope which has cross sections)		30070	1.093-14	2	G	← (n,γ)
			20060	1.487-12	3	P	
			10030	3.526-10	8	A	← (n,α)
			10010	8.289-12	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	8.222-01	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	8.643-15	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     λ_{i-1}    λ_T
       30060   103          .8664+00     .3626-09     .8664+00   .0000        101  1
       20060   1022         .1487-11     .8664+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	4.386-06	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	6.239+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	1.788+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	1.648+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	3.802+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	2.069+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 I 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 TIME 6.312+07 SEC

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	3.266-04	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	1.022-03	1.662-05	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 C.

The Dose Program

DOSE is an auxiliary program to DKR, and calculates the dose rate due to decay γ-rays. The photon transport calculation is performed with decay γ-ray sources at times after shutdown or with adjoint sources (kerma of tissue) at the specified position, with ANISN. Dose reads the input data for dimensions of the system to compute spatially dependent or time dependent dose rates with either forward or adjoint γ-ray flux, respectively. The logical units used in DOSE is listed below.

Table C-1. 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.
N8(18)	Response function (γ-ray source) file unit. Kerma-to-fluence of tissue is stored in the program.
N9 (9)	Photon flux file unit.

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

Card No. 1 (18A4)

Title Card

Card No. 2 (1216)

LID 1-6 Identification number

LTH 7-12 Program execution option.

1: Calculate spatial dependent dose rate with forward flux.

2: Calculate time dependent dose rates with adjoint flux.

LGE	13-18	Geometry
		1: Slab.
		2: Cylinder.
		3: Sphere.
NGRP	19-24	Number of gamma energy groups.
NAS	25-30	Number of after shutdown time if NAS=0, twelve built-in times are used (Table 4-2).
IZN	31-36	Number of zones.
INTVAL	37-42	Number of intervals.
IDZN	43-48	Zone number of tissue.
IPOS	49-54	Interval number of tissue.

Card No. 3 (12F6.2)

NINT(1)	1-6	Number of intervals in the first zone.
NINT(2)	7-12	Number of intervals in the second zone.
NINT(IZM)		Number of intervals in the last zone.

Card No. 4 (6E12.3)

RAD(1)	1-12	Outer radius of first zone.
RAD(2)	13-24	Outer radius of second zone.
RAD(IZN)		Outer radius of last zone.