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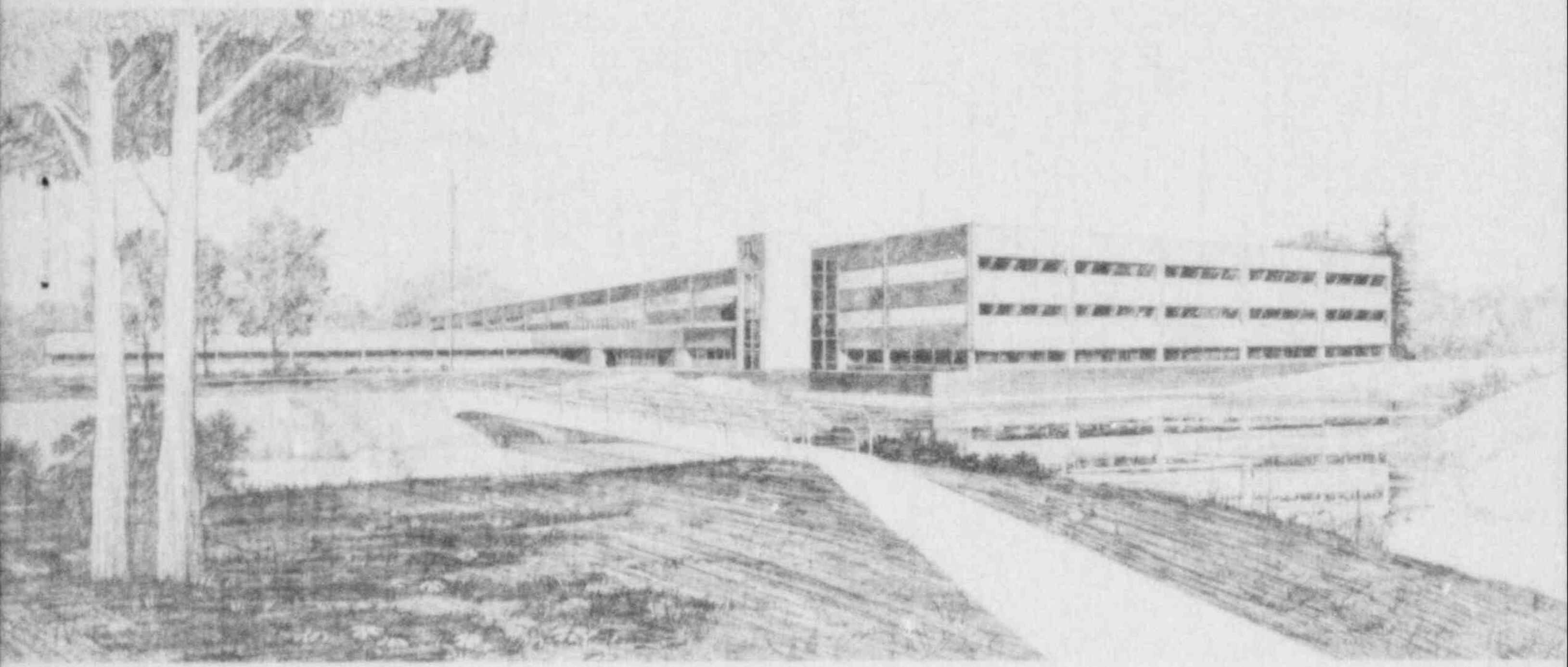

Idaho National Engineering Laboratory
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A Calculational Method for Determining Biological Dose Rates from Irradiated Research Reactor Fuel

B. G. Schnitzler

April 1985

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U.S. Nuclear Regulatory Commission
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ABSTRACT

This report describes a calculational method for the determination of biological dose rate from irradiated research reactor fuels. The calculational method is implemented in a computer program for quick and convenient assessment of multigroup gamma and beta dose rates resulting from an arbitrary (user-supplied) irradiation history. The FUELDR program calculates dose rates at a fixed dose point using built-in fission product impulse source functions and precalculated gamma and beta transport factors. The fixed dose point is located on the axial mid-plane at a distance of 3 ft (91.44 cm) from the fuel element. Transport factors are included for sixteen unique ^{235}U fuel types in use at thirteen nonpower reactor facilities.

NRC FIN No. A6398—A Calculational Method for Determining Dose Rate from Nonpower Reactor Irradiated Fuel

SUMMARY

The determination of biological dose rate from irradiated research reactor fuel may be required for a variety of radiological safety and regulatory purposes. This report describes a method for calculating gamma and beta dose rates from irradiated ^{235}U fuels.

Dose rates are calculated in three steps:

- Determination of the fuel element gamma and beta source strengths
- Determination of the resulting energy flux at the dose point
- Conversion of the dose point energy flux to dose rate.

Time-dependent gamma and beta source strengths are calculated using fission product impulse source functions. The impulse source functions are folded into an arbitrary (user-supplied) irradiation history to generate multigroup photon and electron source strengths at decay times from 1 s to 10^9 s (approximately 30 years). The multigroup structure contains 11 gamma and 11 beta groups with group energies ranging from 100 keV to 7.5 MeV.

Energy fluxes at a fixed dose point (3 ft, or 91.44 cm, from the element) are obtained from the calculated source strengths by using precalculated photon and electron transport factors. The precalculated transport factors account for spatial distribution of the source, for attenuation or energy loss in the source and in any intervening materials, and for the source-dose point geometry. Photon transport factors were calculated using a three-dimensional point-kernel shielding code. Electron transport factors were calculated using a continuous slowing down approximation with corrections for radiative losses. Precalculated factors (11 gamma and 11 beta) are included for sixteen nonpower reactor fuel types.

Calculated dose point energy fluxes are converted to dose rate by using energy-dependent flux-to-dose conversion factors. Photon flux-to-dose conversions were obtained using the methods recommended in ANSI/ANS-6.1; electron flux-to-dose conversions were obtained using the recommendations in International Commission on Radiological Protection (ICRP) Publication 21.

Probable sources of error in the dose rate calculation are discussed. Specifically treated are errors in the power history, errors in the fission product impulse source functions, and errors in the calculated transport factors.

The gamma component of the dose rate can be calculated with high confidence. The dose contribution from electrons is shown to be sensitive to a variety of modeling assumptions, e.g., uranium to aluminum ratio in the fuel matrix. The greatest uncertainty is in the low energy electron groups. Contributions from the low energy groups tend to dominate the calculated total dose rate for practical irradiations and decay times.

The calculation method is implemented in a computer program called FUELDR. FUELDR contains all required fission product impulse source functions, transport factors, and flux-to-dose conversion factors. Minimal user input is required. The facility and element type must be identified (by keywords) and the element power history must be supplied in simple histogram form. Dose rates at the 3 ft (91.44 cm) dose point are calculated at user-specified (or default) decay times. A program listing and a detailed user's manual are provided in the appendices of this report.

It is recognized that the current Code of Federal Regulations (10 CFR 73) self-protection exemption is based on total dose rate and that some nonpower reactor facilities may derive credit from some or all of the beta contribution. It is also recognized the electron dose is of debatable value for fuel self-protection. The electron contribution may be trivially included or excluded from the assessment since total, gamma, and beta dose rates are edited by FUELDR.

The FUELDR program provides a quick and inexpensive method of determining biological dose rates from irradiated ^{235}U fuels. The calculation method should be validated by comparison with dose rate measurements. Relatively straightforward measurements of total gamma and total beta dose rates from one or more fuel types would provide confidence in the overall methodology and implementation. More detailed spectral measurements would probably be required to resolve differences in the predicted and measured dose rates, especially for the low energy beta contributions.

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CONTENTS

A CALCULATIONAL METHOD FOR DETERMINING BIOLOGICAL DOSE RATES FROM IRRADIATED RESEARCH REACTOR FUEL

INTRODUCTION

The determination of biological dose rate from irradiated research reactor fuel may be required for radiological safety and/or regulatory purposes. In particular, the Code of Federal Regulations (10 CFR 73.6) establishes requirements for the physical protection of special nuclear material of moderate and low strategic significance. The level of physical protection at nonpower reactor facilities may be reduced if the special nuclear material has a total external radiation dose rate in excess of 100 rems per hour at a distance of 3 ft (91.44 cm) from any accessible surface without intervening shielding.

The most direct method of determining dose rate is by measurement in air. Although direct, this method can pose practical difficulties. These difficulties include very high dose rates from some fuels, possible conflict with the As-Low-As-Reasonably-Achievable (ALARA) rule, and increased potential for fuel damage. Performing the dose rate measurement in water provides some relief but introduces another difficulty: the measured values must be converted to equivalent dose rates in air. This conversion is not straightforward due to the beta contribution. Although the beta contribution frequently dominates the total dose rate in air, the range of the most energetic fission product beta is only a few centimeters in water. The total dose rate in air can only be inferred from the measured gamma contribution.

A third method is to calculate the dose rate based on the power history of the element. A variety of source and transport approximations have been employed by the research reactor community. Although some methods have provided good agreement with measured data, they are restricted to particular fuel types.

This report describes a consistent calculation method for determining gamma and beta dose rates from any type of fuel element. The required numerical factors are provided for the fuel element types listed in Table 1.

DOSE RATE CALCULATION

The dose rate calculation is accomplished in three steps:

- Determination of the fuel element gamma and beta source strengths resulting from a specified power history
- Determination of the resulting energy flux at the dose point
- Conversion of dose point energy flux to dose rate.

General Methodology

Time-dependent gamma and beta source spectra are calculated using the fission product analytic impulse source functions of LaBauve et al. Gamma and beta energy source functions for a fission pulse are expressed as a summation of exponentials. The source functions are folded into an input power history to generate multigroup gamma and beta energy sources. The multigroup structure contains 11 gamma and 11 beta groups.

Energy fluxes at a fixed dose point (3 ft, or 91.44 cm, from the element) are obtained by multiplying the energy source in each group by a predetermined transport factor for that energy group. The transport factors (11 gamma and 11 beta) are precalculated for each fuel element type. The gamma transport factors were obtained using a three-dimensional point-kernel shielding code. Beta transport factors were calculated using a charged particle energy loss model for the fuel element and air.

Dose rates are obtained by multiplying the energy fluxes at the dose point by an appropriate energy flux-to-dose rate conversion factor.

Source Strength Calculation

The Los Alamos National Laboratory (LANL) developed fission product impulse source functions[1] provide a relatively simple and inexpensive means of calculating fission product decay spectra following an arbitrary irradiation history.

Table 1. Keyword identifiers for research reactor fuel types

Facility	Element	Description
FNR	STANDARD	University of Michigan Ford Nuclear Reactor
FNR	CONTROL	University of Michigan Ford Nuclear Reactor
GATRIGAF	ROD	General Atomic TRIGA Mark F Reactor
GTRR	STANDARD	Georgia Institute of Technology Research Reactor
MITR-II	STANDARD	Massachusetts Institute of Technology Research Reactor
MURR	STANDARD	University of Missouri Research Reactor
NBSR	STANDARD	National Bureau of Standards Reactor
OSTR	ROD	Oregon State University TRIGA Reactor
RINSC	STANDARD	Rhode Island Nuclear Science Center Reactor
TAM-NSCR	STANDARD	Texas A&M University Nuclear Science Center Reactor
TAM-NSCR	CONTROL	Texas A&M University Nuclear Science Center Reactor
UCNR	STANDARD	Union Carbide Nuclear Reactor
UCNR	CONTROL	Union Carbide Nuclear Reactor
UVAR	12-PLATE	University of Virginia Reactor
UVAR	18-PLATE	University of Virginia Reactor
UVAR	PARTIAL	University of Virginia Reactor
UVAR	CONTROL	University of Virginia Reactor
UWNR	STANDARD	University of Wisconsin Nuclear Reactor
WSUR	STANDARD	Washington State University Reactor

The impulse function is expressed as a sum of exponentials

$$f(t) = \sum_{i=1}^{N} \alpha_i e^{-\lambda_i t}$$

and represents the energy release rate at t seconds after a unit fission pulse. The time-dependent energy release rate for a particular fission interval may be obtained by integration. Following the development of Reference 2, the function F(t,T) is defined as the energy release per fission at t seconds after the end of a fission interval with length T seconds and a constant unit fission rate. The energy release rate is then F(t,T) times the fission rate during the interval. The function F(t,T) is given by

$$F(t,T) = \sum_{i=1}^{N} \frac{\alpha_i}{\lambda_i} e^{-\lambda_i t}\left(1 - e^{-\lambda_i T}\right).$$

The coefficients α_i and λ_i are typically obtained by fitting to results of experiments and/or summation calculations. Fits have been obtained for total decay heat[3] and for a number of multigroup gamma and beta energy group structures.[1]

The ^{235}U fission product impulse functions developed in Reference 2 are used in the source calculation. Details of the energy group structure are shown in Table 2. The α_i, λ_i coefficient pairs are included in the Appendix A listing of the FUELDR code.

Energy Transport Calculation

Dose point energy fluxes are obtained from the calculated photon and electron source strengths by using transport factors. The predetermined transport factors account for spatial distribution of the source, for attenuation or energy loss in the source and any intervening materials, and for the source-dose point geometry. A set of factors (11 beta and 11 gamma) is included for each unique fuel type considered.

Table 2. Source function energy group structure

Gamma Group	Energy Range (MeV)	Midpoint (MeV)	Number of α_i - λ_i Pairs	Valid Decay Time Range (s)
1	0.10--0.40	0.250	19	0.1--10^9
2	0.40--0.90	0.650	19	0.4--10^9
3	0.90--1.35	1.125	19	0.1--10^9
4	1.35--1.80	1.575	14	0.1--10^9
5	1.80--2.20	2.000	18	0.1--10^9
6	2.20--2.60	2.400	16	0.1--10^9
7	2.60--3.00	2.800	15	0.1--10^9
8	3.00--4.00	3.500	16	0.1--10^9
9	4.00--5.00	4.500	16	0.1--10^9
10	5.00--6.00	5.500	12	0.1--10^7
11	6.00--7.50	6.750	10	0.1--10^7
Beta Group				
1	0.10--0.40	0.250	16	0.1--10^9
2	0.40--0.90	0.650	17	0.1--10^9
3	0.90--1.35	1.125	17	0.1--10^9
4	1.35--1.80	1.575	17	0.1--10^9
5	1.80--2.20	2.000	16	0.1--10^9
6	2.20--2.60	2.400	16	0.1--10^9
7	2.60--3.00	2.800	15	0.1--10^9
8	3.00--4.00	3.500	15	0.1--10^9
9	4.00--5.00	4.500	14	0.1--10^9
10	5.00--6.00	5.500	13	0.1--10^9
11	6.00--7.50	6.750	11	0.1--10^7

Photon Transport Factors. Photon transport factors were calculated using the three-dimensional point-kernel shielding code QAD.[4] In the QAD series of codes[5] the source volume is represented by an arbitrarily large number of point sources. Geometric attenuation and material attenuation are determined by the distance between the source point and the dose point, and by the material properties of the source and any additional shielding present. Very detailed geometries can be accommodated. QAD solves for the uncollided photon (energy) flux assuming the appropriate geometric attenuation and assuming exponential material attenuation. The flux contribution from scattering events is accounted for by infinite media dose buildup factors.[6]

Solutions were obtained for a unit source (1 MeV/s) in each of the eleven photon groups for each of the sixteen fuel types considered.

Plate Type Fuels. Thirteen plate type fuel elements from ten facilities were considered. The following physical descriptions are divided into three categories: standard, control, and special elements.

Standard Elements—A standard element is rectangular in cross section and contains 9 to 18 flat or curved fuel plates. The element is assembled using aluminum side plates and aluminum end boxes. The fuel plates consist of a fuel core (UAl_x powder dispersed in aluminum alloy powder) with aluminum alloy cladding. The element may also contain dummy (unfueled) plates of solid aluminum.

Geometric models for the seven elements in this category were constructed using data from Reference 7. The fuel (and any dummy) plates were homogenized into a single source region with width and axial length defined by the active fuel width and active fuel length. The remaining source dimension was taken as the overall element depth. The total uranium mass, based on the stated ^{235}U content and enrichment, was uniformly distributed in the homogenized source region. The total aluminum mass was based on the cladding (or dummy plate) thickness and the active fuel width and length. The aluminum occurring in the UAl_x fuel matrix was not included. The unit source was assumed distributed among 1000 source volumes. A chopped cosine distribution was used to represent the axial source variation and flat distributions were assumed across the element width and depth. The dose point was positioned on the axial mid-plane at 3 ft (91.44 cm) from the near edge of the homogenized source. The dose point direction is normal to the fuel plates.

The suitability of the homogenization technique was verified with QAD calculations using a more detailed geometric model. A model of the 18-plate Rhode Island Nuclear Science Center (RINSC) Reactor element[8] was constructed with each fuel plate explicitly modeled. This model was also used to assess the impact of ignoring the aluminum in the UAl_x matrix.

Results for the seven aluminum plate standard fuel elements are listed in Table 3. The tabulated values represent the dose point energy flux (with buildup) from a unit source of 1 MeV/s in that energy group. The aluminum dose buildup factors are quite small (typically 1.1) since the elements are optically thin at the energies of interest. The transport factors for a given energy group differ by only a few percent among the seven elements in this category. Distinction of these types is retained for compatibility with the beta transport treatment.

Control Elements—Control element construction is similar to that used for the standard elements. Control elements are rectangular in cross section but contain fewer (typically 9) fuel plates than the standard elements contain. A central control rod guide slot is formed by two aluminum plates and divides the bundle into a 4-plate region and a 5-plate region. Geometric models were constructed using data from Reference 7.

The two source regions were homogenized separately to allow explicit modeling of the aluminum plates forming the control rod guide slot. The dose point was positioned at 3 ft (91.44 cm) from the near edge of the 5-plate source region.

Results of the three aluminum plate control elements are listed in Table 4.

Special Elements—Plate type elements from three other facilities were evaluated. They included the Massachusetts Institute of Technology, the University of Missouri, and the National Bureau of Standards Reactors.

The Massachusetts Institute of Technology Research Reactor elements are rhombic in cross section and contain 15 flat fueled plates. This element was homogenized like the standard plate element except for the shape of the source region. The

Table 3. Photon transport factors for aluminum plate standard fuel elements

	Dose Point Energy Flux for Unit Energy Source[a] in Indicated Group (MeV/cm^2-s)						
Energy Group	FNR	GTRR	RINSC	UCNR	UVAR 12-Plate	UVAR 18-Plate	UVAR Partial
1	8.4540×10^{-6}	8.5397×10^{-6}	8.4266×10^{-6}	8.4440×10^{-6}	8.5182×10^{-6}	8.4358×10^{-6}	8.4480×10^{-6}
2	8.7885×10^{-6}	8.9230×10^{-6}	8.9058×10^{-6}	8.8282×10^{-6}	8.7572×10^{-6}	8.7991×10^{-6}	8.8118×10^{-6}
3	8.2688×10^{-6}	8.3685×10^{-6}	8.2411×10^{-6}	8.2791×10^{-6}	8.3368×10^{-6}	8.2708×10^{-6}	8.2223×10^{-6}
4	8.2280×10^{-6}	8.3259×10^{-6}	8.1934×10^{-6}	8.2371×10^{-6}	8.3019×10^{-6}	8.2300×10^{-6}	8.1769×10^{-6}
5	8.2243×10^{-6}	8.3223×10^{-6}	8.1920×10^{-6}	8.2335×10^{-6}	8.2980×10^{-6}	8.2262×10^{-6}	8.1743×10^{-6}
6	8.2201×10^{-6}	8.3177×10^{-6}	8.1877×10^{-6}	8.2289×10^{-6}	8.2947×10^{-6}	8.2218×10^{-6}	8.1703×10^{-6}
7	8.2157×10^{-6}	8.3127×10^{-6}	8.1821×10^{-6}	8.2238×10^{-6}	8.2916×10^{-6}	8.2169×10^{-6}	8.1657×10^{-6}
8	8.2081×10^{-6}	8.3038×10^{-6}	8.1717×10^{-6}	8.2150×10^{-6}	8.2865×10^{-6}	8.2085×10^{-6}	8.1578×10^{-6}
9	8.2004×10^{-6}	8.2947×10^{-6}	8.1615×10^{-6}	8.2057×10^{-6}	8.2812×10^{-6}	8.1997×10^{-6}	8.1506×10^{-6}
10	8.1941×10^{-6}	8.2871×10^{-6}	8.1531×10^{-6}	8.1980×10^{-6}	8.2768×10^{-6}	8.1923×10^{-6}	8.1450×10^{-6}
11	8.1875×10^{-6}	8.2790×10^{-6}	8.1443×10^{-6}	8.1898×10^{-6}	8.2719×10^{-6}	8.1845×10^{-6}	8.1391×10^{-6}

a. MeV/s at energy midpoint of group.

Table 4. Photon transport factors for aluminum plate control elements

Energy Group	Dose Point Energy Flux for Unit Energy Source[a] in Indicated Group (MeV/cm^2-s) FNR	UCNR	UVAR
1	8.5477×10^{-6}	8.5814×10^{-6}	8.5432×10^{-6}
2	8.7779×10^{-6}	8.8183×10^{-6}	8.7944×10^{-6}
3	8.3431×10^{-6}	8.3842×10^{-6}	8.3452×10^{-6}
4	8.3064×10^{-6}	8.3478×10^{-6}	8.3081×10^{-6}
5	8.3036×10^{-6}	8.3447×10^{-6}	8.3053×10^{-6}
6	8.3012×10^{-6}	8.3421×10^{-6}	8.3028×10^{-6}
7	8.2989×10^{-6}	8.3396×10^{-6}	8.3002×10^{-6}
8	8.2953×10^{-6}	8.3355×10^{-6}	8.2959×10^{-6}
9	8.2927×10^{-6}	8.3322×10^{-6}	8.2926×10^{-6}
10	8.2908×10^{-6}	8.3296×10^{-6}	8.2899×10^{-6}
11	8.2888×10^{-6}	8.3268×10^{-6}	8.2870×10^{-6}

a. MeV/s at energy midpoint of group.

homogenized source volume was defined as a parallelepiped enclosing the active fuel volume. The dose point was positioned on the axial mid-plane at 3 ft (91.44 cm) from the outermost fuel plate. The dose point direction was chosen parallel to the element side plates rather than normal to the fuel plates to avoid shielding of the dose point by the side plates.

The University of Missouri Research Reactor elements are pie-shaped and contain 24 curved fuel plates of varying radii held in aluminum side plates. Each element is one-eighth of the full core annulus (45° central angle).

The homogenized source volume for this element was defined as a portion (sector) of a right circular annulus with length equal to the active fuel length. The inner and outer radii were defined by the cladding boundary on the inner and outer fuel plates, respectively. The sector central angle was defined so as to include the active fuel but not the element side plates. The dose point was located on the axial mid-plane at 3 ft (91.44 cm) from the outer fuel plate.

National Bureau of Standards Reactor elements consist of curved parallel plates held in aluminum side plates. Each element consists of two identical fueled sections stacked vertically. The fueled regions are separated by a short nonfueled section. Each fueled section contains 17 fueled plates and two dummy exterior plates. These elements were homogenized like the standard plate elements except the source was divided at mid-plane by the nonfueled region. The mid-plane dose point position was retained.

The calculated transport factors for the three aluminum plate special elements are listed in Table 5.

Table 5. Photon transport factors for aluminum plate special elements

Energy Group	Dose Point Energy Flux for Unit Energy Source[a] in Indicated Group (MeV/cm^2-s) MITR-II	MURR	NBS
1	7.8736×10^{-6}	7.8917×10^{-6}	8.0532×10^{-6}
2	8.9897×10^{-6}	8.8212×10^{-6}	8.5996×10^{-6}
3	8.2950×10^{-6}	8.1916×10^{-6}	8.0177×10^{-6}
4	8.2545×10^{-6}	8.1530×10^{-6}	7.9768×10^{-6}
5	8.2516×10^{-6}	8.1499×10^{-6}	7.9733×10^{-6}
6	8.2431×10^{-6}	8.1425×10^{-6}	7.9676×10^{-6}
7	8.2317×10^{-6}	8.1329×10^{-6}	7.9609×10^{-6}
8	8.2099×10^{-6}	8.1143×10^{-6}	7.9485×10^{-6}
9	8.1814×10^{-6}	8.0905×10^{-6}	7.9340×10^{-6}
10	8.1562×10^{-6}	8.0694×10^{-6}	7.9215×10^{-6}
11	8.1288×10^{-6}	8.0461×10^{-6}	7.9081×10^{-6}

a. MeV/s at energy midpoint of group.

TRIGA Fuels. TRIGA fuels[9] have been manufactured in a variety of configurations. All configurations use a uranium-zirconium-hydride ($UZrH_x$) alloy as the fuel material. Various ^{235}U enrichments, hydrogen-to-zirconium ratios, rod diameters, burnable poison and cladding compositions have been employed. Only the high enrichment (70% ^{235}U) fuels with stainless steel cladding are relevant for this evaluation.

The TRIGA fuels from five facilities were considered in three categories for modeling purposes. The categories are single rod, four-rod fuel assemblies, and three-fueled-rod control assemblies. The material composition for the fuel was developed from Reference 10. Geometric models were constructed using data from Reference 7.

Single Rods—The single pin configuration is applicable for two facilities: the General Atomic Mark F TRIGA Reactor and the Oregon State University TRIGA Reactor. Table 6 lists the material compositions and pin dimensions used for the QAD photon transport model. A flat radial, chopped cosine axial source distribution was assumed. The dose point was located 3 ft (91.44 cm) from the rod outer surface on the axial mid-plane.

The validity of the QAD uncollided flux solution was verified using RAFFLE V,[11] a general purpose Monte Carlo transport code. The RAFFLE-calculated total photon fluxes (uncollided plus scattered) also provided a sound basis for the selection of iron dose buildup factors for the QAD calculations.

Table 6. TRIGA single rod material compositions and fuel rod dimensions

Single Rod Material Compositions

$UZrH_x$ Fuel		Type 304 Stainless Steel	
Element	Density (g/cm^3)	Element	Density (g/cm^3)
U	0.5099	Fe	5.4276
Zr	5.3937	Cr	1.5200
H	0.0954	Ni	0.8000
		Mn	0.1600
		Si	0.0800

Fuel Rod Dimensions

	Large Rod (cm)	Small Rod (cm)
Fuel radius	1.8161	1.7399
Active length	38.10	38.10
Clad thickness	0.0508	0.0508

The QAD calculated transport factors for the TRIGA elements are listed in Table 7.

Four Rod Assemblies—The four rod cluster model is applicable for three facilities: the Texas A&M University Reactor, the University of Wisconsin Reactor, and the Washington State University Reactor. The element contains four identical rods in a square array with a rod pitch of 1.53 in. (3.8862 cm). The fuel rods are slightly smaller (1.7399 cm active radius) but are otherwise identical to the fuel rod model used in the single rod configuration.

All four rods were modeled for the QAD transport calculations. A flat radial, chopped cosine axial source distribution was assumed with one-fourth of the total source in each rod. The dose point was positioned on the axial mid-plane at a distance of 3 ft (91.44 cm). The dose point lies on an extension of a line joining the centers of two diagonal rods. Table 7 includes the QAD calculated transport factors for the four-rod assemblies.

Control Assemblies—The three-fueled-rod cluster is applicable to the Texas A&M University Reactor control elements. A three-rod cluster is a four-rod assembly with one rod position vacant. The same QAD model was used with one-third of the total source in each of the three rods nearer the dose point. The transport factors are included in Table 7.

Beta Transport Factors. The basic interaction processes and resulting particle transport are fundamentally different for electrons and for photons. The primary photon interaction processes are photoelectric absorption, Compton scattering, and pair production. Exponential attenuation methods are appropriate since all processes effectively eliminate the photon from consideration in a single event.

In contrast, electrons experience multiple interaction processes that cannot be adequately treated by simple exponential attenuation models. For the energy range of interest, electron energy losses are due almost entirely to excitation and ionization of electrons in the stopping material and to bremsstrahlung.

The energy loss due to excitation and ionization may be expressed[12] as

$$-\frac{dE}{dX} = 2\pi N_a r_o^2 \left(\frac{Z}{A}\right) \frac{mc^2}{\beta^2}$$

$$\times \left[B_o - (2)\ln_e(Z) - (2)\ln_e\left(\frac{I/10}{Z}\right)\right]$$

where N_a is Avogadro's number, r_o is the classical electron radius in cm, mc^2 is the electron rest energy in MeV, $\beta = v/c$, Z and A are the atomic number and weight, respectively, of the stopping material, and I is the mean ionization energy of the stopping material in electron volts. With the indicated units, the energy loss units are $MeV\text{-}cm^2/g$. Reference 12 contains experimentally determined values of the mean ionization energy.

The bracketed term is frequently defined as the stopping number; the energy dependent component represented by B_o is given by

$$B_o = 21.683 + \ln_e[\tau^2(\tau + 2)]$$

$$-\left[1 + \frac{2\tau + 1}{(\tau + 1)^2}\right]\ln_e(2)$$

$$+ \frac{1}{(\tau + 1)^2} + \frac{1}{8}\left[\frac{\tau^2}{(\tau + 1)^2}\right]$$

Table 7. Photon transport factors for TRIGA fuel elements

	Dose Point Energy Flux for Unit Energy Source[a] in Indicated Group (MeV/cm^2-s)		
Energy Group	Single Rod	4-Rod Cluster	3-Rod Cluster
1	5.1411×10^{-6}	3.4675×10^{-6}	4.4957×10^{-6}
2	7.2077×10^{-6}	5.7263×10^{-6}	6.7043×10^{-6}
3	7.5974×10^{-6}	6.1812×10^{-6}	7.0870×10^{-6}
4	7.7277×10^{-6}	6.3901×10^{-6}	7.2332×10^{-6}
5	7.7337×10^{-6}	6.4560×10^{-6}	7.2524×10^{-6}
6	7.6776×10^{-6}	6.4251×10^{-6}	7.2019×10^{-6}
7	7.6261×10^{-6}	6.3903×10^{-6}	7.1537×10^{-6}
8	7.5403×10^{-6}	6.3151×10^{-6}	7.0695×10^{-6}
9	7.4279×10^{-6}	6.1957×10^{-6}	6.9545×10^{-6}
10	7.3181×10^{-6}	6.0671×10^{-6}	6.8397×10^{-6}
11	7.2088×10^{-6}	5.9354×10^{-6}	6.7245×10^{-6}

a. MeV/s at energy midpoint of group.

where

$$\tau = \left[1 - \beta^2\right]^{-1/2} - 1 \ .$$

The rate of energy loss due to bremsstrahlung is proportional to the square of the atomic number. The ratio of radiative loss to collisional loss at a particular energy is approximately given[13] by

$$\frac{(dE/dX)_{rad}}{(dE/dX)_{coll}} = \frac{EZ}{1600\ mc^2}$$

where mc^2 is the electron rest energy in MeV, E is the electron kinetic energy in MeV, and Z is the atomic number of the stopping material. For this study, credit is taken for the electron energy loss due to bremsstrahlung, but the resulting photons are not tracked to the dose point. This approach results in a conservative dose assessment for this application.

The above energy loss formulations include several inherent limitations. In particular, the radiative energy losses are subject to large statistical fluctuations[12] which cannot be treated with an average radiative energy loss model. Fortunately, radiative losses are not dominant over most of the electron energy-atomic number combinations of interest.

Energy and range straggling are not explicitly treated. The impact of energy straggling is minimized by the flux-to-dose conversion. The difference in actual pathlength and source-dose point distance includes both range straggling introduced by the discrete nature of the interaction processes and the effect of multiple elastic scattering.

Ignoring the pathlength differences is conservative for this assessment since greater pathlengths provide lower energy electrons; lower energy electrons result in higher dose rates than calculated at higher energies since flux-to-dose conversions are taken at the peak of the depth-dose curve.

An electron transport program was assembled by combining the collisional and radiative energy loss models described above, the MAGI-developed combinatorial geometry package,[14] and the point source distribution routines from QAD.[4] This program obtains effective beta transport factors using the following methodology. The fuel element is divided into an arbitrary number of source volumes. For each energy group, a unit energy source (1 MeV/s) is distributed among the discrete volume elements. A dose point particle flux for that source volume is determined, based solely on the geometric attenuation provided by the source volume element-detector point separation. Next, the material attenuation is evaluated by tracking a single particle from each source volume element through all intervening materials to the dose point. Energy losses along the track are calculated using the collisional and radiative loss models described above. An initial energy corresponding to the group mid-point energy is assumed and the particle is tracked until either the dose point is reached or the particle energy is reduced to below 100 keV. For those particles reaching the dose point, a particle flux-to-dose conversion is obtained for the final energy and a dose contribution is calculated. The dose contributions from all volume elements are calculated and a total dose rate obtained for the energy group of interest. The effective beta transport factor is obtained by dividing the total group dose rate by the energy flux-to-dose factor for the group mid-point energy.

Plate Type Fuels. Detailed geometric models were constructed for the thirteen plate type fuel elements using data from Reference 7. Individual fuel plates were explicitly modeled for all elements. Dummy plates and control rod guide plates were modeled, if present. The fuel matrix composition was assumed to be UAl_3. The source in each fuel plate was distributed among 200 source volume elements. A chopped cosine distribution was used to represent the axial source variation and flat distributions were assumed across the fuel plate width and thickness. The dose point was positioned on the axial mid-plane at 3 ft (91.44 cm) from the outermost plate.

Results for the seven standard element types, the three control elements, and the three special element types are listed in Tables 8, 9, and 10, respectively.

TRIGA Fuels. All relevant TRIGA fuels were considered in three categories: single rod, four-rod fuel assemblies, and three-fueled-rod control assemblies. The three geometric models developed for the photon transport factor evaluations were also used for the electron transport calculations. The previously described dose point locations and flat radial, chopped cosine source distributions were retained.

The source in each rod was distributed among 2400 source volume elements. The maximum range of the most energetic electrons (6.75 MeV) is less than 0.8 cm in the TRIGA fuel composition. Therefore, the source volume elements could be concentrated in the rod periphery in view of the detector in order to minimize tracking of electrons that could not reach the detector.

Calculated transport factors for the three TRIGA fuel types are listed in Table 11.

Flux-to-Dose Conversions

Flux-to-dose conversion factors were calculated at the gamma group energy mid-points using the analytic expression provided in ANSI/ANS-6.1.1-1977.[15] Flux-to-dose conversion factors for the beta group energy mid-points were obtained from Figure 13 of ICRP Publication 21.[16] The conversion factors are listed in Table 12.

COMPUTER PROGRAM DESCRIPTION

A computer program was written to perform the dose rate calculations using the methods described in the previous sections. The program is coded in the FORTRAN 77 language for easy implementation on any computer system with a full ANSI FORTRAN 77 compiler.

Minimal user input is required. The facility and element type must be identified (by keywords) and the element power history must be supplied in simple histogram form. Gamma and beta dose rates at the 3 ft (91.44 cm) dose point are calculated at user specified (or default) cooling times.

Appendix A contains a listing of the program. Appendix B is a user's manual with a description of the code and a detailed input description.

Table 8. Electron transport factors for aluminum plate standard fuel elements

	Equivalent Dose Point Energy Flux for Unit Energy Source[a] in Indicated Group (MeV/cm^2-s)						
Energy Group	FNR	GTRR	RINSC	UCNR	UVAR 12-Plate	UVAR 18-Plate	UVAR Partial
1	0.0	0.0	0.0	0.0	0.0	0.0	0.0
2	3.2085×10^{-6}	0.0	5.2798×10^{-6}	0.0	5.4771×10^{-6}	3.3240×10^{-6}	6.6478×10^{-6}
3	2.6779×10^{-6}	7.8339×10^{-6}	8.9258×10^{-7}	7.9439×10^{-6}	4.6202×10^{-6}	2.7850×10^{-6}	1.5143×10^{-6}
4	2.6732×10^{-6}	5.4755×10^{-6}	1.6874×10^{-6}	5.6275×10^{-6}	4.6666×10^{-6}	2.7899×10^{-6}	8.3056×10^{-6}
5	3.7042×10^{-6}	5.2127×10^{-6}	2.8225×10^{-6}	5.3601×10^{-6}	5.0597×10^{-6}	2.9704×10^{-6}	2.4775×10^{-6}
6	5.8984×10^{-6}	5.8157×10^{-6}	6.7299×10^{-6}	6.0428×10^{-6}	5.7815×10^{-6}	3.2789×10^{-6}	9.9560×10^{-6}
7	6.9775×10^{-6}	6.6306×10^{-6}	2.1674×10^{-6}	7.0282×10^{-6}	6.6331×10^{-6}	3.5910×10^{-6}	3.8939×10^{-6}
8	4.5964×10^{-6}	9.2760×10^{-6}	6.2458×10^{-6}	8.0501×10^{-6}	1.2550×10^{-5}	5.3590×10^{-6}	1.1586×10^{-6}
9	9.1046×10^{-6}	8.5386×10^{-6}	3.5203×10^{-6}	8.7862×10^{-6}	1.0891×10^{-5}	4.5263×10^{-6}	1.1993×10^{-6}
10	7.8851×10^{-6}	7.6974×10^{-6}	7.7172×10^{-6}	8.1463×10^{-6}	1.0310×10^{-5}	8.9985×10^{-6}	2.6510×10^{-6}
11	9.3055×10^{-6}	1.0202×10^{-5}	3.6014×10^{-6}	9.3512×10^{-6}	3.4242×10^{-6}	6.3820×10^{-6}	4.0822×10^{-6}

a. MeV/s at energy midpoint of group.

Table 9. Electron transport factors for aluminum plate control elements

Energy Group	Equivalent Dose Point Energy Flux for Unit Energy Source[a] in Indicated Group (MeV/cm^2-s) FNR	UCNR	UVAR
1	0.0	0.0	0.0
2	6.4170×10^{-6}	6.8305×10^{-6}	6.6478×10^{-6}
3	5.3558×10^{-6}	5.7383×10^{-6}	5.5699×10^{-6}
4	5.3463×10^{-6}	5.7662×10^{-6}	5.5797×10^{-6}
5	7.4083×10^{-6}	6.1737×10^{-6}	5.9407×10^{-6}
6	6.1804×10^{-6}	6.8786×10^{-6}	6.5578×10^{-6}
7	3.9396×10^{-6}	4.1503×10^{-6}	4.0583×10^{-6}
8	2.5325×10^{-6}	2.6119×10^{-6}	2.5780×10^{-6}
9	1.7078×10^{-6}	1.7450×10^{-6}	1.7293×10^{-6}
10	4.9398×10^{-6}	1.1435×10^{-5}	1.0658×10^{-5}
11	1.1574×10^{-5}	7.6254×10^{-6}	9.1412×10^{-6}

a. MeV/s at energy midpoint of group.

ACCURACY OF DOSE RATE CALCULATION

The overall accuracy of a calculated dose rate is a complex function of the accuracy in each of three components of the dose rate assessment. The three controlling components are the power history, the fission product source functions, and the effective transport factors.

Power History

Errors in the specified fuel element power history are reflected as errors in the fission product inventory. Error propagation into the calculated dose rate may be easily assessed only for two limiting cases. At very short decay times, the dose rate is dominated by contributions from short-lived fission products. In this limiting case, the error in the dose rate is less than or equal to the error in the instantaneous power level. At very long decay times, the dose rate is dominated by contributions from long-lived fission products. In this limiting case, the error in the dose rate is less than or equal to the error in the accumulated exposure.

Any specified power history may contain errors in (a) the total accumulated exposure and (b) the instantaneous power levels. The maximum error propagated into the dose rate from errors in the power history is equal to the maximum of (a) and (b) above.

Table 10. Electron transport factors for aluminum plate special elements

	Equivalent Dose Point Energy Flux for Unit Source[a] in Indicated Group (MeV/cm^2-s)		
Energy Group	MITR-II	MURR	NBS
1	0.0	0.0	0.0
2	1.7994×10^{-6}	4.6980×10^{-6}	0.0
3	1.2852×10^{-6}	5.7725×10^{-6}	4.2795×10^{-6}
4	4.3333×10^{-6}	6.4858×10^{-6}	7.1175×10^{-6}
5	1.4968×10^{-6}	8.4220×10^{-6}	7.5393×10^{-6}
6	3.5149×10^{-6}	2.4102×10^{-6}	5.7442×10^{-6}
7	2.5778×10^{-6}	3.0608×10^{-6}	3.8622×10^{-6}
8	2.2849×10^{-6}	7.8754×10^{-6}	6.1694×10^{-6}
9	5.3921×10^{-6}	5.5364×10^{-6}	6.4376×10^{-6}
10	6.9775×10^{-6}	3.8318×10^{-6}	6.9432×10^{-6}
11	4.8036×10^{-6}	3.7395×10^{-6}	5.8047×10^{-6}

a. 1 MeV/s at energy midpoint of group.

Fission Product Impulse Functions

Fission product impulse function coefficients are typically obtained by fitting to results of experiments and/or summation calculations. Very tight fitting tolerances are achievable; Reference 2 reported the ANS-5.1 decay heat standard utilized 23 exponential terms to fit total decay heat to well within 1% over the period from 0.1 to 10^{13} s. The total uncertainty on decay heat is necessarily larger since the fitting tolerance errors are overshadowed by the uncertainties in the experimental measurements and in the fission product data base used in the summation calculations. The one sigma uncertainty on the ANS-5.1 ^{235}U decay heat standard is 40% at 1 s following a fission pulse and 3.3% at 1 s after an infinite (10^{13} s) irradiation. For both pulse and infinite irradiations, the one sigma uncertainty is less than 3% for decay times between 15 and 10^9 s.

Fitting tolerances and uncertainties were not available for the multigroup impulse function coefficients.[2] The overall uncertainty associated with the multigroup data is expected to be higher than that quoted for the total energy release; less experimental data is available and the energy dependent measurements are more demanding than those required for total energy release.

Total energy release calculations, using the multigroup data, provide good agreement with decay heat values obtained using the ANS-5.1 decay heat standard. For a ^{235}U fission pulse, the maximum deviation is 23% during the decay period from 1 to 10^9 s. For the decay period from 30 to 10^9 s, the maximum deviation is less than 7%. Smaller differences are obtained with finite length irradiations.

Table 11. Electron transport factors for TRIGA fuel elements

Energy Group	Dose Point Energy Flux for Unit Energy Source[a] in Indicated Group (MeV/cm²-s) Single Rod	4-Rod Cluster	3-Rod Cluster
1	0.0	0.0	0.0
2	0.0	0.0	0.0
3	2.5222×10^{-7}	2.6296×10^{-7}	3.5061×10^{-7}
4	5.3366×10^{-7}	2.4567×10^{-7}	3.2756×10^{-7}
5	1.0437×10^{-6}	9.6792×10^{-7}	1.2906×10^{-6}
6	1.6448×10^{-6}	4.8966×10^{-7}	6.5289×10^{-7}
7	2.2260×10^{-6}	1.2425×10^{-6}	1.6567×10^{-6}
8	2.4539×10^{-6}	8.0982×10^{-7}	1.0798×10^{-6}
9	3.2345×10^{-6}	1.2304×10^{-6}	1.6405×10^{-6}
10	2.8015×10^{-6}	1.0265×10^{-6}	1.3687×10^{-6}
11	1.4893×10^{-6}	1.5354×10^{-6}	2.0472×10^{-6}

a. 1 MeV/s at energy midpoint of group.

Table 12. Flux-to-dose conversion factors

Group	Energy (MeV)	Flux-to-Dose Conversion Factor (rem/hr)/(MeV/cm²-s) Photons	Electrons
1	0.250	2.52×10^{-6}	1.37×10^{-3}
2	0.650	2.22×10^{-6}	3.62×10^{-4}
3	1.125	1.91×10^{-6}	1.83×10^{-4}
4	1.575	1.73×10^{-6}	1.22×10^{-4}
5	2.000	1.60×10^{-6}	9.21×10^{-5}
6	2.400	1.51×10^{-6}	7.43×10^{-5}
7	2.800	1.43×10^{-6}	6.21×10^{-5}
8	3.500	1.32×10^{-6}	4.82×10^{-5}
9	4.500	1.20×10^{-6}	3.62×10^{-5}
10	5.500	1.13×10^{-6}	2.89×10^{-5}
11	6.750	1.05×10^{-6}	2.30×10^{-5}

Effective Transport Factors

Photons. Exact representations of fuel element geometries and arbitrarily accurate specification of source distributions are afforded by the shielding codes used in this analysis. Photon interaction cross sections are well known for the materials and energies of interest. Uncollided photon fluxes at the dose point can be calculated with high accuracy. The major source of error arises from the use of infinite media buildup factors for the scattered flux contribution. The scattered flux error is usually not more than a few percent and will reach 20 to 30% only in unusual cases.[5]

The plate type fuel elements are optically thin at the energies of interest. The buildup factors and resulting scattered flux contributions are relatively

small. For a typical buildup factor of 1.1, even a worst-case 30% error will introduce only a 3% error on the total flux solution.

Comparable errors can be expected for the TRIGA fuels. The buildup factors are somewhat larger (typically 1.2 to 1.3), but the uncertainty in the total flux solution was reduced by supporting Monte Carlo calculations.

Electrons. Errors in the electron transport calculation arise primarily from failure to explicitly treat energy and range straggling, radiative loss treatment, uncertainties in the fuel element geometry and composition, and uncertainties in the fuel element power distribution.

Errors introduced by the radiative loss treatment are expected to be the larger of the first two types. The ratio of radiative loss to total loss, or radiative loss fraction, is a function of material composition and electron energy. For the TRIGA fuel composition, the radiative loss fractions are about 21% at 5 MeV, 10% at 2 MeV, and 5% at 1 MeV. Radiative loss fractions at the same energies are 36%, 18%, and 11% for UAl_3 and 7%, 3%, and 1% for aluminum.

For practical irradiation and decay times, the electron source distribution is dominated by the low energy groups. For an eight hour constant power irradiation followed by a one hour cooling period, 98% of the electron energy source is below 4 MeV and 82% is at or below 2 MeV. The 0.65 MeV group contains 24% of the electron source.

Errors introduced by both the radiative loss treatment and by energy and range straggling result in a conservative dose assessment from a fuel self-protection standpoint.

Uncertainties in compositions and fuel element dimensions introduce the greatest uncertainty to the electron transport calculations. Typical fuel element specifications provide for very close tolerances on the fuel plate total thickness. However, due to the fuel manufacturing process, much looser tolerances are acceptable (and required) on both the fuel core thickness and the uranium to aluminum ratio in the fuel matrix.

The impact of small composition uncertainties is illustrated by the sensitivity to the assumed air density. The 0.65 MeV electron group dose contribution comes entirely from the outermost plate. The standard atmosphere density difference between sea level and 1400 m (4593 ft) introduces a 35% change in the effective transport factor for the 0.65 MeV group. The change is about 5% for the 1.125 MeV group.

Transport factors for the lower energy electron groups are also sensitive to deviations from the assumed source distributions. Flux depressions or peaking at the element edges alters the source distribution in the outer plates. The error in the effective transport factor for the 0.65 MeV group is equal to the plate power peaking factor. The effect is smaller for the higher energy groups.

CONCLUSIONS AND RECOMMENDATIONS

The FUELDR program provides a quick and inexpensive method of determining biological dose rates from irradiated ^{235}U fuels. Dose rates at a fixed dose point are determined using an input element power history. The code contains effective transport factors for sixteen unique fuel types in use at thirteen facilities. Data for additional element types can be incorporated as required.

The calculation method should be validated by comparison with dose rate measurements. Relatively straightforward measurements of total gamma and total beta dose rates from one or more fuel types would provide confidence in the overall methodology and implementation. More detailed spectral measurements would probably be required to resolve differences in the predicted and measured dose rates, especially for the low energy beta dose contributions.

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APPENDIX A
FUELDR LISTING

```
      PROGRAM FUELDR
C
C    FUELDR - A PROGRAM TO CALCULATE BIOLOGICAL DOSE RATES FROM
C             IRRADIATED HIGH ENRICHMENT RESEARCH REACTOR FUELS
C
      COMMON /CHDAT/ CHEAD, CTITL, CNAM1, CNAM2, CNAME(19),
     A               CTIMP(100), CTIMD(100), CUPWR(100)
      CHARACTER CHEAD*120, CTITL*80, CNAM1*8, CNAM2*8, CNAME*98
      CHARACTER CTIMP*3, CTIMD*3, CUPWR*3
      COMMON /COEFF/ BALF(23,11), BLAM(23,11), GALF(23,11), GLAM(23,11)
      COMMON /DRATE/ BDRAT(11,100), GDRAT(11,100), BDTOT(100),
     A               GDTOT(100), DRTOT(100)
      COMMON /FACTR/ BTRAN(19,11), GTRAN(19,11), BFTOD(11), GFTOD(11)
      COMMON /FLAGS/ NAMID, MPTIM, MDTIM, NPTIM, NDTIM, NCASE, MEDIT
      COMMON /IRRAD/ FRATE(100), PWRIN(100), ENGPF(100)
      COMMON /LIMIT/ NBG, NGG, NPMAX, NDMAX, LB(11), LG(11),
     A               TMINF, TMAXF, EXPLL, ERR
      COMMON /SORCE/ BESRC(11,100), GESRC(11,100), BETOT(100),
     A               GETOT(100), ENTOT(100)
      COMMON /TIMES/ TP(100), TD(100), TX(100), TPINP(100), TDINP(100)
      COMMON /UFRYD/ BEPFR(11), GEPFR(11)
      OPEN(5,FILE='INPUT')
      OPEN(6,FILE='OUTPUT')
   10 CALL RDINP
      IF(NCASE.EQ.0)  GO TO 20
      CALL EDINP
      CALL DRCAL
      CALL EDOUT
      GO TO 10
   20 CONTINUE
      CLOSE(5)
      CLOSE(6)
      STOP
      END
```

```
      SUBROUTINE CKINP(NCK)
C
C   SUBROUTINE TO CHECK INPUT DATA AND PERFORM UNITS CONVERSIONS
C
      COMMON /CHDAT/ CHEAD, CTITL, CNAM1, CNAM2, CNAME(19),
     A               CTIMP(100), CTIMD(100), CUPWR(100)
      CHARACTER CHEAD*120, CTITL*80, CNAM1*8, CNAM2*8, CNAME*98
      CHARACTER CTIMP*3, CTIMD*3, CUPWR*3
      COMMON /FLAGS/ NAMID, MPTIM, MDTIM, NPTIM, NDTIM, NCASE, MEDIT
      COMMON /IRRAD/ FRATE(100), PWRIN(100), ENGPF(100)
      COMMON /LIMIT/ NBG, NGG, NPMAX, NDMAX, LB(11), LG(11),
     A               TMINF, TMAXF, EXPLL, ERR
      COMMON /TIMES/ TP(100), TD(100), TX(100), TPINP(100), TDINP(100)
      IF(NCK.EQ.2)  GO TO 20
C
C   POWER STEP DATA
C
      DO 10 IL = 1, NPTIM
      LL = IL
      IF(TPINP(IL).LE.0.0)  CALL ERRS(8,LL,0)
      IF(PWRIN(IL).LT.0.0)  CALL ERRS(9,LL,0)
      IF(ENGPF(IL).LT.0.0)  CALL ERRS(10,LL,0)
      TCONV = 0.0
      IF(CTIMP(IL).EQ.'   ')  CTIMP(IL) = 'SEC'
      IF(CTIMP(IL).EQ.'SEC')  TCONV = 1.0
      IF(CTIMP(IL).EQ.'MIN')  TCONV = 60.0
      IF(CTIMP(IL).EQ.'HRS')  TCONV = 3600.0
      IF(CTIMP(IL).EQ.'DAY')  TCONV = 86400.0
      IF(CTIMP(IL).EQ.'YRS')  TCONV = 3.15576E+7
      IF(TCONV.LT.1.0)  CALL ERRS(11,LL,0)
      TP(IL) = TPINP(IL)*TCONV
      PCONV = 0.0
      IF(CUPWR(IL).EQ.'   ')  CUPWR(IL) = 'KW '
      IF(CUPWR(IL).EQ.'W  ')  PCONV = 1.0
      IF(CUPWR(IL).EQ.' W ')  PCONV = 1.0
      IF(CUPWR(IL).EQ.'  W')  PCONV = 1.0
      IF(CUPWR(IL).EQ.'KW ')  PCONV = 1000.0
      IF(CUPWR(IL).EQ.' KW')  PCONV = 1000.0
      IF(CUPWR(IL).EQ.'MW ')  PCONV = 1.0E+6
      IF(CUPWR(IL).EQ.' MW')  PCONV = 1.0E+6
      IF(PCONV.LT.1.0)  CALL ERRS(12,LL,0)
C
C   CONVERT SPECIFIED THERMAL POWER TO FISSION RATE USING EITHER
C   200 MEV/FISSION OR THE ENERGY/FISSION VALUE INPUT FOR THIS STEP
C       AT 200 MEV/FISSION,  1 WATT = 3.12098E+10 FISSIONS/SEC
C
      PCONV = PCONV*3.12098E+10
      IF(ENGPF(IL).NE.200.0)  PCONV = PCONV*200.0/ENGPF(IL)
      FRATE(IL) = PWRIN(IL)*PCONV
   10 CONTINUE
      RETURN
C
C   DECAY TIME DATA
C
```

```
20 CONTINUE
   DO 30 IL = 1, NDTIM
   LL = IL
   IF(TDINP(IL).LE.C.0)  CALL ERRS(13,LL,0)
   TCONV = 0.0
   IF(CTIMD(IL).EQ.'   ')  CTIMD(IL) = 'SEC'
   IF(CTIMD(IL).EQ.'SEC')  TCONV = 1.0
   IF(CTIMD(IL).EQ.'MIN')  TCONV = 60.0
   IF(CTIMD(IL).EQ.'HRS')  TCONV = 3600.0
   IF(CTIMD(IL).EQ.'DAY')  TCONV = 86400.0
   IF(CTIMD(IL).EQ.'YRS')  TCONV = 3.15576E+7
   IF(TCONV.LT.1.0)  CALL ERRS(14,LL,0)
   TD(IL) = TDINP(IL)*TCONV
   IF(TD(IL).LT.TMINF)  CALL ERRS(15,LL,0)
   IF(TD(IL).GT.TMAXF)  CALL ERRS(16,LL,0)
30 CONTINUE
   RETURN
   END
```

```
      SUBROUTINE DRCAL
C
C   SUBROUTINE TO MANAGE DOSE RATE CALCULATION
C
      COMMON /DRATE/ BDRAT(11,100), GDRAT(11,100), BDTOT(100),
     A               GDTOT(100), DRTOT(100)
      COMMON /FACTR/ BTRAN(19,11), GTRAN(19,11), BFTOD(11), GFTOD(11)
      COMMON /FLAGS/ NAMID, MPTIM, MDTIM, NPTIM, NDTIM, NCASE, MEDIT
      COMMON /IRRAD/ FRATE(100), PWRIN(100), ENGPF(100)
      COMMON /LIMIT/ NBG, NGG, NPMAX, NDMAX, LB(11), LG(11),
     A               TMINF, TMAXF, EXPLL, ERR
      COMMON /SORCE/ BESRC(11,100), GESRC(11,100), BETOT(100),
     A               GETOT(100), ENTOT(100)
      COMMON /TIMES/ TP(100), TD(100), TX(100), TPINP(100), TDINP(100)
      COMMON /UFRYD/ BEPFR(11), GEPFR(11)
      TX(NPTIM) = 0.0
      DO 10 NP = NPTIM, 2, -1
   10 TX(NP-1) = TX(NP) + TP(NP)
      CALL ZARAY
      DO 40 ND = 1, NDTIM
      DO 30 NP = 1, NPTIM
      TTC = TD(ND) + TX(NP)
      IF(TTC.GT.TMAXF)  CALL ERRS(17,NP,ND)
      TAP = TP(NP)
      CALL UFYLD(TAP,TTC)
      DO 24 NG = 1, NBG
      BESRC(NG,ND) = BESRC(NG,ND) + BEPFR(NG)*FRATE(NP)
      BDRAT(NG,ND) = BESRC(NG,ND)*BTRAN(NAMID,NG)*BFTOD(NG)
   24 CONTINUE
      DO 26 NG = 1, NGG
      GESRC(NG,ND) = GESRC(NG,ND) + GEPFR(NG)*FRATE(NP)
      GDRAT(NG,ND) = GESRC(NG,ND)*GTRAN(NAMID,NG)*GFTOD(NG)
   26 CONTINUE
   30 CONTINUE
   40 CONTINUE
      DO 70 ND = 1, NDTIM
      DO 50 NG = 1, NBG
      BETOT(ND) = BETOT(ND) + BESRC(NG,ND)
      BDTOT(ND) = BDTOT(ND) + BDRAT(NG,ND)
   50 CONTINUE
      DO 60 NG = 1, NGG
      GETOT(ND) = GETOT(ND) + GESRC(NG,ND)
      GDTOT(ND) = GDTOT(ND) + GDRAT(NG,ND)
   60 CONTINUE
      ENTOT(ND) = BETOT(ND) + GETOT(ND)
      DRTOT(ND) = BDTOT(ND) + GDTOT(ND)
   70 CONTINUE
      RETURN
      END
```

```
      SUBROUTINE EDINP
C
C   INPUT DATA EDITS
C
      COMMON /CHDAT/ CHEAD, CTITL, CNAM1, CNAM2, CNAME(19),
     A               CTIMP(100), CTIMD(100), CUPWR(100)
      CHARACTER CHEAD*120, CTITL*80, CNAM1*8, CNAM2*8, CNAME*98
      CHARACTER CTIMP*3, CTIMD*3, CUPWR*3
      COMMON /FLAGS/ NAMID, MPTIM, MDTIM, NPTIM, NDTIM, NCASE, MEDIT
      COMMON /IRRAD/ FRATE(100), PWRIN(100), ENGPF(100)
      COMMON /TIMES/ TP(100), TD(100), TX(100), TPINP(100), TDINP(100)
      DIMENSION CT(5)
      CHARACTER CT*3
      DATA CT / 'SEC', 'MIN', 'HRS', 'DAY', 'YRS' /
      WRITE(6,600)
      WRITE(6,610)  CHEAD
      WRITE(6,610)  CNAME(NAMID)(17:98)
      WRITE(6,620)  CTITL
      WRITE(6,601)
      WRITE(6,630)
      WRITE(6,640)
      WRITE(6,641)
      WRITE(6,642)
      WRITE(6,602)
      ACEXP = 0.0
      ACTIM = 0.0
      DO 100 IL = 1, NPTIM
      STEXP = FRATE(IL)*ENGPF(IL)*TP(IL)/6.24196E+15
      ACEXP = ACEXP + STEXP
      ACTIM = ACTIM + TP(IL)
      WRITE(6,645) IL, TPINP(IL), CTIMP(IL), TP(IL), CT(1), PWRIN(IL),
     A             CUPWR(IL), ENGPF(IL), FRATE(IL), STEXP, ACEXP, ACTIM
  100 CONTINUE
      WRITE(6,600)
      WRITE(6,610)  CHEAD
      WRITE(6,610)  CNAME(NAMID)(17:98)
      WRITE(6,620)  CTITL
      WRITE(6,601)
      WRITE(6,660)
      WRITE(6,670)
      WRITE(6,671)
      WRITE(6,672)
      WRITE(6,673)
      WRITE(6,602)
      DO 200 IL = 1, NDTIM
      T1 = TD(IL)
      T2 = TD(IL)/60.0
      T3 = TD(IL)/3600.0
      T4 = TD(IL)/86400.0
      T5 = TD(IL)/3.15576E+7
      WRITE(6,675) IL, TDINP(IL), CTIMD(IL), T1, CT(1), T2, CT(2),
     A             T3, CT(3), T4, CT(4), T5, CT(5)
  200 CONTINUE
      RETURN
```

```
600 FORMAT('1')
601 FORMAT('0')
602 FORMAT(' ')
610 FORMAT('0',(A))
620 FORMAT('0',' CASE:   ',(A))
630 FORMAT(1X,' POWER HISTORY EDIT ',/,'+',20('_')/)
640 FORMAT(1X,1X,'STEP',7X,'STEP LENGTH',8X,'STEP LENGTH',8X,
   A        'STEP POWER',6X,'MEV PER',3X,'FISSION RATE',6X,'STEP',6X,
   B        'CUMULATIVE',3X,'CUMULATIVE')
641 FORMAT(1X,'NUMBER',5X,'(INPUT UNITS)',8X,'(SECONDS)',8X,
   A        '(INPUT UNITS)',4X,'FISSION',5X,'(FIS/SEC)',5X,' KW-SEC ',
   B        5X,' KW-SEC ',5X,' SECONDS ')
642 FORMAT('+',6('_'),3X,16('_'),3X,16('_'),3X,16('_'),3X,7('_'),
   A        3X,12('_'),3X,10('_'),3X,10('_'),3X,10('_'))
645 FORMAT(1X,1X,I3,5X,1PE12.6,1X,A3,3X,1PE12.6,1X,A3,3X,1PE12.6,1X,
   A        A3,3X,0PF7.3,1X,1PE13.4,1X,1P3E13.4)
660 FORMAT(1X,' DECAY TIME EDIT ',/,'+',17('_')/)
670 FORMAT(1X,40X,'DECAY TIME (TIME AFTER END OF LAST POWER STEP)')
671 FORMAT('+',8X,111('_'))
672 FORMAT(1X,'DECAY',/,1X,'STEP',7X,'INPUT UNITS',10X,'SECONDS',
   A        12X,'MINUTES',13X,'HOURS',14X,'DAYS',13X,'YEARS')
673 FORMAT('+',5('_'),6(3X,16('_'))/)
675 FORMAT(1X,I4,1X,6(2X,1PE13.6,1X,A3))
    END
```

```
      SUBROUTINE EDOUT
C
C   OUTPUT EDITS
C
      COMMON /CHDAT/ CHEAD, CTITL, CNAM1, CNAM2, CNAME(19),
     A               CTIMP(100), CTIMD(100), CUPWR(100)
      CHARACTER CHEAD*120, CTITL*80, CNAM1*8, CNAM2*8, CNAME*98
      CHARACTER CTIMP*3, CTIMD*3, CUPWR*3
      COMMON /DRATE/ BDRAT(11,100), GDRAT(11,100), BDTOT(100),
     A               GDTOT(100), DRTOT(100)
      COMMON /FLAGS/ NAMID, MPTIM, MDTIM, NPTIM, NDTIM, NCASE, MEDIT
      COMMON /LIMIT/ NBG, NGG, NPMAX, NDMAX, LB(11), LG(11),
     A               TMINF, TMAXF, EXPLL, ERR
      COMMON /SORCE/ BESRC(11,100), GESRC(11,100), BETOT(100),
     A               GETOT(100), ENTOT(100)
      COMMON /TIMES/ TP(100), TD(100), TX(100), TPINP(100), TDINP(100)
      LPERP = 50
C --- DOSE RATE SUMMARY
      NP = 1
   10 WRITE(6,600)
      WRITE(6,610)  CHEAD
      WRITE(6,610)  CNAME(NAMID)(17:98)
      WRITE(6,620)  CTITL
      WRITE(6,601)
      WRITE(6,630)
      WRITE(6,601)
      WRITE(6,640)
      WRITE(6,601)
      DO 20 NL = 1, LPERP
      IF(NP.GT.NDTIM)  GO TO 30
      WRITE(6,650) NP,TDINP(NP),CTIMD(NP),DRTOT(NP),GDTOT(NP),BDTOT(NP)
      NP = NP + 1
   20 CONTINUE
      IF(NP.LT.NDTIM)  GO TO 10
C --- ENERGY RELEASE RATE SUMMARY
   30 IF(MEDIT.LT.1)  GO TO 99
      NP = 1
      IF(NDTIM.LT.20)  GO TO 45
   40 WRITE(6,600)
      WRITE(6,610)  CHEAD
      WRITE(6,610)  CNAME(NAMID)(17:98)
      WRITE(6,620)  CTITL
   45 WRITE(6,601)
      WRITE(6,660)
      WRITE(6,601)
      WRITE(6,670)
      WRITE(6,601)
      DO 50 NL = 1, LPERP
      IF(NP.GT.NDTIM)  GO TO 60
      WRITE(6,650) NP,TDINP(NP),CTIMD(NP),ENTOT(NP),GETOT(NP),BETOT(NP)
      NP = NP + 1
   50 CONTINUE
      IF(NP.LT.NDTIM)  GO TO 40
   60 IF(MEDIT.LT.2)  GO TO 99
```

```
C --- MULTIGROUP DOSE RATE EDITS
      NB = 1
      NE = 8
   70 IF(NB.GT.NDTIM)  GO TO 80
      IF(NE.GT.NDTIM)  NE = NDTIM
      WRITE(6,600)
      WRITE(6,610)  CHEAD
      WRITE(6,610)  CNAME(NAMID)(17:98)
      WRITE(6,620)  CTITL
      WRITE(6,601)
      WRITE(6,680)
      WRITE(6,681)  (TDINP(N),CTIMD(N),N=NB,NE)
      WRITE(6,682)
      WRITE(6,602)
      DO 72 NG = 1, NGG
   72 WRITE(6,683)  NG, (GDRAT(NG,N),N=NB,NE)
      WRITE(6,684)  (GDTOT(N),N=NB,NE)
      WRITE(6,601)
      WRITE(6,685)  (TDINP(N),CTIMD(N),N=NB,NE)
      WRITE(6,682)
      WRITE(6,602)
      DO 74 NG = 1, NBG
   74 WRITE(6,683)  NG, (BDRAT(NG,N),N=NB,NE)
      WRITE(6,686)  (BDTOT(N),N=NB,NE)
      WRITE(6,601)
      WRITE(6,687)  (DRTOT(N),N=NB,NE)
      NB = NB + 8
      NE = NE + 8
      GO TO 70
C --- MULTIGROUP ENERGY RELEASE RATE EDITS
   80 NB = 1
      NE = 8
   81 IF(NB.GT.NDTIM)  GO TO 99
      IF(NE.GT.NDTIM)  NE = NDTIM
      WRITE(6,600)
      WRITE(6,610)  CHEAD
      WRITE(6,610)  CNAME(NAMID)(17:98)
      WRITE(6,620)  CTITL
      WRITE(6,601)
      WRITE(6,690)
      WRITE(6,691)  (TDINP(N),CTIMD(N),N=NB,NE)
      WRITE(6,682)
      WRITE(6,602)
      DO 82 NG = 1, NGG
   82 WRITE(6,683)  NG, (GESRC(NG,N),N=NB,NE)
      WRITE(6,684)  (GETOT(N),N=NB,NE)
      WRITE(6,601)
      WRITE(6,692)  (TDINP(N),CTIMD(N),N=NB,NE)
      WRITE(6,682)
      WRITE(6,602)
      DO 84 NG = 1, NBG
   84 WRITE(6,683)  NG, (BESRC(NG,N),N=NB,NE)
      WRITE(6,686)  (BETOT(N),N=NB,NE)
      WRITE(6,601)
```

```
      WRITE(6,687)   (ENTOT(N),N=NB,NE)
      NB = NB + 8
      NE = NE + 8
      GO TO 61
   99 CONTINUE
      RETURN
  600 FORMAT('1')
  601 FORMAT('0')
  602 FORMAT(' ')
  610 FORMAT('0',(A))
  620 FORMAT('0',' CASE:   ',(A))
  630 FORMAT('0',' DOSE RATE SUMMARY ',/,'+',19('_'))
  640 FORMAT('0',28X,'  DOSE RATE AT THREE FEET (REM/HR)   ',/,2X,
     A         'STEP',23X,36('-'),/,1X,'NUMBER',6X,'DECAY TIME',8X,
     B         'TOTAL',8X,'GAMMA',9X,'BETA',/,'+',6('_'),3X,16('_'),
     C         3(3X,10('_')))
  650 FORMAT(1X,I4,4X,1PE13.6,1X,A3,3(1PE13.4))
  660 FORMAT('0',' SOURCE ENERGY RELEASE SUMMARY ',/,'+',31('_'))
  670 FORMAT('0',31X,'SOURCE ENERGY RELEASE (MEV/SEC)',/,2X,'STEP',
     A         23X,36('-'),/,1X,'NUMBER',6X,'DECAY TIME',8X,'TOTAL',8X,
     B         'GAMMA',9X,'BETA',/,'+',6('_'),3X,16('_'),3(3X,10('_')))
  680 FORMAT('0',' MULTIGROUP DOSE RATE EDITS ',/,'+',28('_')/)
  681 FORMAT(1X,41X,'GAMMA DOSE RATE AT THREE FEET (REM/HR)',
     A         ' AT SELECTED DECAY TIMES',/,3X,'GAMMA',5X,117('-'),/,
     B         3X,'GROUP',2X,8(1PE13.4,1X,A1))
  682 FORMAT('+',9('_'),8(3X,12('_')))
  683 FORMAT(1X,I5,3X,1P8E15.4)
  684 FORMAT('0','GAMMA SUM',1PE14.4,1P7E15.4)
  685 FORMAT(1X,41X,' BETA DOSE RATE AT THREE FEET (REM/HR)',
     A         ' AT SELECTED DECAY TIMES',/,3X,' BETA',5X,117('-'),/,
     B         3X,'GROUP',2X,8(1PE13.4,1X,A1))
  686 FORMAT('0',' BETA SUM',1PE14.4,1P7E15.4)
  687 FORMAT('0','   TOTAL   ',1PE14.4,1P7E15.4)
  690 FORMAT('0',' MULTIGROUP SOURCE ENERGY RELEASE RATE EDITS ',/,
     A         '+',45('_')/)
  691 FORMAT(1X,39X,'SOURCE GAMMA ENERGY RELEASE RATE (MEV/SEC)',
     A         ' AT SELECTED DECAY TIMES',/,3X,'GAMMA',5X,120('-'),/,
     B         3X,'GROUP',2X,8(1PE13.4,1X,A1))
  692 FORMAT(1X,39X,' SOURCE BETA ENERGY RELEASE RATE (MEV/SEC)',
     A         ' AT SELECTED DECAY TIMES',/,3X,' BETA',5X,120('-'),/,
     B         3X,'GROUP',2X,8(1PE13.4,1X,A1))
      END
```

```
      SUBROUTINE ERRS(IP,I1,I2)
C
C     DIAGNOSTIC MESSAGES FOLLOWING ERROR DETECTION
C
      COMMON /CHDAT/ CHEAD, CTITL, CNAM1, CNAM2, CNAME(19),
     A               CTIMP(100), CTIMD(100), CUPWR(100)
      CHARACTER CHEAD*120, CTITL*60, CNAM1*8, CNAM2*8, CNAME*98
      CHARACTER CTIMP*3, CTIMD*3, CUPWR*3
      COMMON /IRRAD/ FRATE(100), PWRIN(100), ENGPF(100)
      COMMON /LIMIT/ NBG, NGG, NPMAX, NDMAX, LB(11), LG(11),
     A               TMINF, TMAXF, EXPLL, ERR
      COMMON /TIMES/ TP(100), TD(100), TX(100), TPINP(100), TDINP(100)
      WRITE(6,600)
      GO TO (1,2,3,4,5,6,7,8,9,10,11,12,13,14,15,16,17), IP
    1 WRITE(6,601)  CNAM1, CNAM2
      GO TO 99
    2 WRITE(6,602)  NPMAX, I1
      GO TO 99
    3 WRITE(6,603)
      GO TO 99
    4 WRITE(6,604)  I1, I2
      GO TO 99
    5 WRITE(6,605)
      GO TO 99
    6 WRITE(6,606)  NDMAX, I1
      GO TO 99
    7 WRITE(6,607)
      GO TO 99
    8 WRITE(6,608)  I1, TPINP(I1), CTIMP(I1)
      GO TO 99
    9 WRITE(6,609)  I1, PWRIN(I1), CUPWR(I1)
      GO TO 99
   10 WRITE(6,610)  I1, ENGPF(I1)
      GO TO 99
   11 WRITE(6,611)  I1, CTIMP(I1)
      GO TO 99
   12 WRITE(6,612)  I1, CUPWR(I1)
      GO TO 99
   13 WRITE(6,613)  I1, TDINP(I1), CTIMD(I1)
      GO TO 99
   14 WRITE(6,614)  I1, CTIMD(I1)
      GO TO 99
   15 WRITE(6,615)  I1, TMINF, TDINP(I1), CTIMD(I1)
      GO TO 99
   16 WRITE(6,616)  I1, TMAXF, TDINP(I1), CTIMD(I1)
      GO TO 99
   17 WRITE(6,617)  I1, TMAXF, I2
   99 CONTINUE
      WRITE(6,699)
      STOP
  600 FORMAT('1')
  601 FORMAT('ONO MATCH FOUND FOR THE INPUT FACILITY AND ELEMENT',/,
     1       'OINPUT VALUES = ',(A),(A))
  602 FORMAT('OTHE NUMBER OF POWER STEPS SHOULD BE POSITIVE AND',
```

```
     1         ' SHOULD NOT EXCEED',I4,/,
     2         '0INPUT VALUE =',I4)
 603 FORMAT('0NO POWER STEP DATA INPUT;  ALLOWED ONLY AS A',
     1         ' CHANGE CASE')
 604 FORMAT('0POWER STEP DATA OUT OF SEQUENCE',/,
     1         '0STEP NUMBER',I4,' INPUT AS',I4)
 605 FORMAT('0NO DECAY STEP DATA INPUT;  ALLOWED ONLY AS A',
     1         ' CHANGE CASE')
 606 FORMAT('0ERROR IN THE NUMBER OF DECAY STEPS REQUESTED',/,
     1         '0ALLOWABLE VALUES ARE:',
     2         2X,'1 TO',I4,' TO READ THAT MANY STEPS FROM INPUT',/,
     3         24X,'0 TO USE DECAY STEP DATA FROM THE PREVIOUS CASE',/,
     4         24X,'-1 TO -7 TO SELECT ONE OF THE DEFAULT TIME SETS',/,
     5         '0INPUT VALUE =',I4)
 607 FORMAT('0PREMATURE END OF INPUT DATA ENCOUNTERED')
 608 FORMAT('0ERROR IN POWER STEP NUMBER',I4,/,
     1         '0POWER STEP LENGTH MUST BE GREATER THAN ZERO',/,
     2         '0INPUT VALUE =',1PE12.5,1X,(A))
 609 FORMAT('0ERROR IN POWER STEP NUMBER',I4,/,
     1         '0NEGATIVE POWER LEVEL NOT ALLOWED',/,
     2         '0INPUT VALUE =',1PE12.5,1X,(A))
 610 FORMAT('0ERROR IN POWER STEP NUMBER',I4,/,
     1         '0NEGATIVE ENERGY PER FISSION (MEV/FIS) NOT ALLOWED',/,
     2         '0INPUT VALUE =',1PE12.5)
 611 FORMAT('0ERROR IN POWER STEP NUMBER',I4,/,
     1         '0NO MATCH FOUND FOR INPUT UNITS ON POWER STEP LENGTH',/,
     2         '0INPUT VALUE =',1X,(A))
 612 FORMAT('0ERROR IN POWER STEP NUMBER',I4,/,
     1         '0NO MATCH FOUND FOR UNITS ON INPUT POWER LEVEL',/,
     2         '0INPUT VALUE =',1X,(A))
 613 FORMAT('0ERROR IN DECAY STEP NUMBER',I4,/,
     1         '0NEGATIVE DECAY TIME NOT ALLOWED',/,
     2         '0INPUT VALUE =',1PE12.5,1X,(A))
 614 FORMAT('0ERROR IN DECAY STEP NUMBER',I4,/,
     1         '0NO MATCH FOUND FOR INPUT UNITS ON DECAY STEP LENGTH',/,
     2         '0INPUT VALUE =',1X,(A))
 615 FORMAT('0ERROR IN DECAY STEP NUMBER',I4,/,
     1         '0DECAY TIME MUST BE GREATER THAN',1PE12.5,/,
     2         '0INPUT VALUE =',1PE12.5,1X,(A))
 616 FORMAT('0ERROR IN DECAY STEP NUMBER',I4,/,
     1         '0DECAY TIME MUST BE LESS THAN',1PE12.5,/,
     2         '0INPUT VALUE =',1PE12.5,1X,(A))
 617 FORMAT('0THE TOTAL TIME FROM THE END OF POWER STEP',I4,/,
     1         '0EXCEEDS',1PE12.5,' SECONDS AT THE END OF',
     2         ' DECAY STEP',I4)
 699 FORMAT('0INPUT DATA CHECKS TERMINATED')
     END
```

```
      SUBROUTINE MATCH
C
C     SUBROUTINE TO CHECK INPUT FACILITY NAME AND ELEMENT TYPE AND TO
C     IDENTIFY APPROPRIATE GAMMA-BETA TRANSPORT AND DOSE RATE MODEL
C
      COMMON /CHDAT/ CHEAD, CTITL, CNAM1, CNAM2, CNAME(19),
     A               CTIMP(100), CTIMD(100), CUPWR(100)
      CHARACTER CHEAD*120, CTITL*80, CNAM1*8, CNAM2*8, CNAME*98
      CHARACTER CTIMP*3, CTIMD*3, CUPWP*3
      COMMON /FLAGS/ NAMID, MPTIM, MDTIM, NPTIM, NDTIM, NCASE, MEDIT
      NAMID = 0
      DO 10 N = 1, 19
      IF(CNAM1.EQ.CNAME(N)(1:8).AND.CNAM2.EQ.CNAME(N)(9:16)) NAMID = N
   10 CONTINUE
      IF(NAMID.EQ.0)  CALL ERRS(1,0,0)
      RETURN
      END
```

```
      SUBROUTINE RDINP
C
C   SUBROUTINE TO READ INPUT
C
      COMMON /CHDAT/ CHEAD, CTITL, CNAM1, CNAM2, CNAME(19),
     A               CTIMP(100), CTIMD(100), CUPWR(100)
      CHARACTER CHEAD*120, CTITL*80, CNAM1*8, CNAM2*8, CNAME*98
      CHARACTER CTIMP*3, CTIMD*3, CUPWR*3
      COMMON /FLAGS/ NAMID, MPTIM, MDTIM, NPTIM, NDTIM, NCASE, MEDIT
      COMMON /IRRAD/ FRATE(100), PWRIN(100), ENGPF(100)
      COMMON /LIMIT/ NBG, NGG, NPMAX, NDMAX, LB(11), LG(11),
     A               TMINF, TMAXF, EXPLL, ERR
      COMMON /TIMES/ TP(100), TD(100), TX(100), TPINP(100), TDINP(100)
      READ(5,500,END=50)  CTITL
      READ(5,510,END=90)  CNAM1, CNAM2, MPTIM, MDTIM, MEDIT
      CALL MATCH
      IF(MPTIM.LT.0.OR.MPTIM.GT.NPMAX)  CALL ERRS(2,MPTIM,0)
      IF(MPTIM.EQ.0.AND.NPTIM.NE.0)  GO TO 15
      IF(MPTIM.EQ.0)  CALL ERRS(3,0,0)
      NPTIM = MPTIM
      DO 10 I = 1, NPTIM
      LL = I
      READ(5,520,END=90) NS,TPINP(I),CTIMP(I),PWRIN(I),CUPWR(I),FISEN
      IF(NS.NE.I)  CALL ERRS(4,LL,NS)
      IF(FISEN.GT.ERR)  ENGPF(I) = FISEN
      IF(FISEN.LE.ERR)  ENGPF(I) = 200.0
   10 CONTINUE
      CALL CKINP(1)
   15 CONTINUE
      IF(MDTIM.EQ.0.AND.NDTIM.NE.0)  GO TO 25
      IF(MDTIM.EQ.0.AND.NDTIM.EQ.0)  CALL ERRS(5,0,0)
      IF(MDTIM.LT.-7.OR.MDTIM.GT.NDMAX)  CALL ERRS(6,MDTIM,0)
      IF(MDTIM.LT.0)  CALL TLOAD
      IF(MDTIM.LT.0)  GO TO 20
      NDTIM = MDTIM
      READ(5,530,END=90)  (TDINP(I), CTIMD(I), I = 1, NDTIM)
   20 CALL CKINP(2)
   25 CONTINUE
      RETURN
   50 NCASE = 0
      RETURN
   90 CALL ERRS(7,0,0)
      RETURN
  500 FORMAT(A)
  510 FORMAT(A8,A8,3I4)
  520 FORMAT(I4,E12.5,1X,A3,E12.5,1X,A3,E12.5)
  530 FORMAT(5(E12.5,1X,A3))
      END
```

```
      SUBROUTINE TLOAD
C
C     INITIALIZE DEFAULT DECAY TIMES
C
      COMMON /CHDAT/ CHEAD, CTITL, CNAM1, CNAM2, CNAME(19),
     A               CTIMP(100), CTIMD(100), CUPWR(100)
      CHARACTER CHEAD*120, CTITL*80, CNAM1*8, CNAM2*8, CNAME*98
      CHARACTER CTIMP*3, CTIMD*3, CUPWR*3
      COMMON /FLAGS/ NAMID, MPTIM, MDTIM, NPTIM, NDTIM, NCASE, MEDIT
      COMMON /TIMES/ TP(100), TD(100), TX(100), TPINP(100), TDINP(100)
      DIMENSION TSET(5,50)
      DATA (TSET(1,I), I = 1, 24)
     A / 0.1,  0.2,  0.3,  0.5,  1.0,  2.0,  3.0,  5.0, 10.0, 20.0,
     B    30.,  50.,  60., 100., 120., 180., 200., 240., 300., 360.,
     C   420., 480., 500., 600.                                      /
      DATA (TSET(2,I), I = 1, 24)
     A / 0.1,  0.2,  0.3,  0.5,  1.0,  2.0,  3.0,  5.0,  8.0, 10.0,
     B    12.,  16.,  20.,  24.,  30.,  48.,  50.,  72.,  96., 100.,
     C   120., 144., 168., 240.                                      /
      DATA (TSET(3,I), I = 1, 32)
     A / 0.1,  0.2,  0.3,  0.5,  1.0,  2.0,  3.0,  5.0,  7.0, 10.0,
     B    14.,  20.,  21.,  28.,  30.,  50.,  60.,  90., 100., 120.,
     C   150., 180., 200., 210., 240., 270., 300., 330., 365., 730.,
     C  1095.,1825.                                                  /
      DATA (TSET(4,I), I = 1, 24)
     A / 0.1,  0.2,  0.3,  0.4,  0.5,  0.6,  0.7,  0.8,  0.9,  1.0,
     B    1.2,  1.4,  1.6,  1.8,  2.0,  2.5,  3.0,  4.0,  5.0,  6.0,
     C    7.0,  8.0,  9.0, 10.0                                      /
      DATA (TSET(5,I), I = 1, 48)
     A / 1.0,  2.0,  3.0,  4.0,  5.0, 10.0, 30.0,100.0,  1.0,  2.0,
     B    3.0,  4.0,  5.0, 10.0, 15.0, 30.0, 60.0,  1.0,  2.0,  3.0,
     C    4.0,  5.0,  8.0, 16.0, 24.0, 48.0, 72.0, 84.0,  1.0,  2.0,
     D    3.0,  4.0,  5.0,  6.0,  7.0, 10.0, 14.0, 21.0, 28.0, 30.0,
     E   60.0, 90.0,120.0,150.0,180.0,  1.0,  2.0,  3.0              /
      NSET = -MDTIM
      GO TO (10, 20, 30, 40, 50, 60, 70), NSET
   10 NDTIM = 32
      TDINP(1) = 1.0
      TDINP(2) = 2.0
      TDINP(3) = 3.0
      TDINP(4) = 5.0
      DO 12 ND = 5, 32
   12 TDINP(ND) = TDINP(ND-4)*10.0
      DO 14 ND = 1, 32
   14 CTIMD(ND) = 'SEC'
      GO TO 99
   20 NDTIM = 32
      XP = 1.0
      DO 22 ND = 1, 32
      TDINP(ND) = 10.0**XP
      XP = XP + 0.25
   22 CTIMD(ND) = 'SEC'
      GO TO 99
   30 NDTIM = 24
```

```
      DO 32 ND = 1, NDTIM
      TDINP(ND) = TSET(1,ND)
   32 CTIMD(ND) = 'MIN'
      GO TO 99
   40 NDTIM = 24
      DO 42 ND = 1, NDTIM
      TDINP(ND) = TSET(2,ND)
   42 CTIMD(ND) = 'HRS'
      GO TO 99
   50 NDTIM = 32
      DO 52 ND = 1, NDTIM
      TDINP(ND) = TSET(3,ND)
   52 CTIMD(ND) = 'DAY'
      GO TO 99
   60 NDTIM = 24
      DO 62 ND = 1, NDTIM
      TDINP(ND) = TSET(4,ND)
   62 CTIMD(ND) = 'YRS'
      GO TO 99
   70 NDTIM = 48
      DO 72 ND = 1, NDTIM
   72 TDINP(ND) = TSET(5,ND)
      DO 74 ND = 1, 8
   74 CTIMD(ND) = 'SEC'
      DO 75 ND = 9, 17
   75 CTIMD(ND) = 'MIN'
      DO 76 ND = 18, 28
   76 CTIMD(ND) = 'HRS'
      DO 77 ND = 29, 45
   77 CTIMD(ND) = 'DAY'
      DO 78 ND = 46, 48
   78 CTIMD(ND) = 'YRS'
   99 CONTINUE
      RETURN
      END
```

```
      SUBROUTINE UFYLD (TPR,TDK)
C
C   SUBROUTINE UFYLD FINDS BETA AND GAMMA ENERGY YIELDS RESULTING FROM
C   A CONSTANT UNIT FISSION RATE OVER AN INTERVAL OF 'TPR' SECONDS.
C   ENERGY YIELDS ARE IN MEV/SEC AT 'TDK' SECONDS AFTER THE END OF THE
C   FISSION INTERVAL.
C
      COMMON /COEFF/ BALF(23,11), BLAM(23,11), GALF(23,11), GLAM(23,11)
      COMMON /LIMIT/ NBG, NGG, NPMAX, NDMAX, LB(11), LG(11),
     A               TMINF, TMAXF, EXPLL, ERR
      COMMON /UFRYD/ BEPFR(11), GEPFR(11)
C
C   BETA GROUPS
C
      DO 200 NB = 1, NBG
      L = LB(NB)
      BEPFR(NB) = 0.0
      DO 100 NC = 1, L
      E2 = -BLAM(NC,NB)*TDK
      E3 = -BLAM(NC,NB)*TPR
      IF(E2.LE.EXPLL)  T2 = 0.0
      IF(E2.GT.EXPLL)  T2 = EXP(E2)
      IF(E3.LE.EXPLL)  T3 = 1.0
      IF(E3.GT.EXPLL)  T3 = 1.0 - EXP(E3)
      T1 = BALF(NC,NB)/BLAM(NC,NB)
      BEPFR(NB) = BEPFR(NB) + T1*T2*T3
  100 CONTINUE
  200 CONTINUE
C
C   GAMMA GROUPS
C
      DO 400 NG = 1, NGG
      L = LG(NG)
      GEPFR(NG) = 0.0
      DO 300 NC = 1, L
      E2 = -GLAM(NC,NG)*TDK
      E3 = -GLAM(NC,NG)*TPR
      IF(E2.LE.EXPLL)  T2 = 0.0
      IF(E2.GT.EXPLL)  T2 = EXP(E2)
      IF(E3.LE.EXPLL)  T3 = 1.0
      IF(E3.GT.EXPLL)  T3 = 1.0 - EXP(E3)
      T1 = GALF(NC,NG)/GLAM(NC,NG)
      GEPFR(NG) = GEPFR(NG) + T1*T2*T3
  300 CONTINUE
  400 CONTINUE
      RETURN
      END
```

```
      SUBROUTINE ZARAY
C
C   ZERO ENERGY SOURCE AND DOSE RATE ARRAYS
C
      COMMON /DRATE/ BDRAT(11,100), GDRAT(11,100), BDTOT(100),
     A               GDTOT(100), DRTOT(100)
      COMMON /LIMIT/ NBG, NGG, NPMAX, NDMAX, LB(11), LG(11),
     A               TMINF, TMAXF, EXPLL, ERR
      COMMON /SORCE/ BESRC(11,100), GESRC(11,100), BETOT(100),
     A               GETOT(100), ENTOT(100)
      DO 30 ND = 1, NDMAX
      BETOT(ND) = 0.0
      GETOT(ND) = 0.0
      BDTOT(ND) = 0.0
      GDTOT(ND) = 0.0
      DO 10 NG = 1, NBG
      BESRC(NG,ND) = 0.0
      BDRAT(NG,ND) = 0.0
   10 CONTINUE
      DO 20 NG = 1, NGG
      GESRC(NG,ND) = 0.0
      GDRAT(NG,ND) = 0.0
   20 CONTINUE
   30 CONTINUE
      RETURN
      END
```

```
      BLOCK DATA COEFS
C
C   INITIALIZE IMPULSE SOURCE FUNCTION COEFFICIENTS
C
      COMMON /COEFF/ BALF(23,11), BLAM(23,11), GALF(23,11), GLAM(23,11)
C
C
C -------- BETA RELEASE COEFFICIENTS IN 11 ENERGY GROUPS -------
C
      DATA (BALF(I, 1),I=1,16)
     $/ 2.76879E-03, 1.28097E-03, 4.76613E-04, 5.01187E-05, 2.57695E-05,
     $  9.36897E-06, 2.80721E-06, 6.97162E-07, 5.25675E-07, 6.42172E-08,
     $  3.27577E-08, 7.27157E-09, 4.04634E-09, 7.20339E-10, 4.66955E-11,
     $  1.50326E-11/
      DATA (BALF(I, 2),I=1,17)
     $/ 1.82626E-02, 5.81826E-03, 5.02635E-03, 2.41635E-03, 2.47292E-04,
     $  1.10704E-04, 3.75435E-05, 1.28235E-05, 1.86468E-06, 1.59380E-06,
     $  2.33907E-07, 5.02395E-08, 1.07640E-08, 6.05622E-09, 1.32156E-09,
     $  1.24375E-10, 1.25056E-11/
      DATA (BALF(I, 3),I=1,17)
     $/ 3.83669E-02, 1.20318E-02, 9.60829E-03, 3.91077E-03, 3.89244E-04,
     $  1.30117E-04, 4.08136E-05, 1.38431E-05, 2.41778E-06, 1.51009E-06,
     $  1.32817E-07, 1.22656E-08, 2.75929E-09, 2.16993E-09, 7.13751E-10,
     $  2.26836E-10, 1.55507E-11/
      DATA (BALF(I, 4),I=1,17)
     $/ 5.65416E-02, 2.38913E-02, 1.53936E-02, 4.59877E-03, 4.07969E-04,
     $  1.44031E-04, 3.93732E-05, 1.00242E-05, 1.56505E-06, 1.63683E-06,
     $  7.92657E-08, 4.39155E-09, 3.14023E-10, 8.58378E-11, 4.47296E-10,
     $  3.65979E-11, 1.30160E-11/
      DATA (BALF(I, 5),I=1,16)
     $/ 6.18952E-02, 2.63810E-02, 1.70417E-02, 4.46878E-03, 4.26161E-04,
     $  1.19508E-04, 3.70577E-05, 4.31425E-06, 1.08707E-06, 1.17447E-06,
     $  7.87866E-08, 5.38424E-10, 3.25314E-11, 4.33761E-10, 3.78592E-13,
     $  3.56942E-12/
      DATA (BALF(I, 6),I=1,16)
     $/ 5.68253E-02, 4.28895E-02, 1.84235E-02, 4.08763E-03, 4.68408E-04,
     $  1.09553E-04, 2.91333E-05, 3.01788E-06, 2.32268E-07, 1.15956E-06,
     $  1.10940E-08, 2.66705E-11, 1.76405E-10, 3.70529E-11, 2.96361E-11,
     $  2.41157E-14/
      DATA (BALF(I, 7),I=1,15)
     $/ 6.43156E-02, 2.65202E-02, 1.99957E-02, 3.55701E-03, 4.71405E-04,
     $  6.61565E-05, 2.60429E-05, 4.25276E-07, 4.13834E-08, 1.45412E-06,
     $  1.35054E-08, 3.93246E-11, 2.63822E-11, 3.17023E-11, 1.83652E-17/
      DATA (BALF(I, 8),I=1,15)
     $/ 1.23767E-01, 4.34200E-02, 2.94350E-02, 4.83717E-03, 7.18957E-04,
     $  1.35340E-04, 2.33256E-05, 1.55786E-06, 3.54762E-07, 1.67968E-06,
     $  6.39867E-09, 5.07757E-11, 3.62214E-13, 3.94038E-12, 1.43663E-17/
      DATA (BALF(I, 9),I=1,14)
     $/ 5.21141E-02, 1.40632E-02, 1.72253E-02, 1.44734E-03, 3.68088E-04,
     $  2.81370E-05, 7.55437E-07, 4.60218E-08, 7.22653E-07,-2.30797E-08,
     $  6.45734E-12, 1.57561E-16, 5.34692E-19, 7.58336E-19/
      DATA (BALF(I,10),I=1,13)
     $/ 2.47958E-02, 7.73591E-03, 7.31443E-03, 4.07547E-04, 1.17944E-04,
     $  1.24718E-05, 1.25100E-07, 1.32243E-08, 2.93470E-14, 8.93566E-13,
```

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$  4.96936E-19, 4.16160E-21, 1.59019E-20/
 DATA (BALF(I,11),I=1,11)
$/ 1.06640E-02, 2.37762E-03, 3.76743E-03, 2.80087E-04, 2.57530E-05,
$  4.51912E-07, 4.97950E-09, 3.33028E-10, 4.79124E-13, 2.02480E-14,
$  2.00667E-21/
 DATA (BLAM(I, 1),I=1,16)
$/ 1.45381E-00, 1.38592E-01, 1.70696E-02, 2.40758E-03, 9.23360E-04,
$  3.78094E-04, 1.47429E-04, 4.22706E-05, 1.47919E-05, 4.56836E-06,
$  1.29054E-06, 5.08353E-07, 2.30415E-07, 7.66225E-08, 1.58689E-08,
$  8.01828E-10/
 DATA (BLAM(I, 2),I=1,17)
$/ 1.66124E-00, 3.21188E-01, 1.02291E-01, 1.84530E-02, 3.88907E-03,
$  1.25923E-03, 4.35588E-04, 1.66976E-04, 4.67802E-05, 1.70337E-05,
$  5.93531E-06, 1.48776E-06, 5.02256E-07, 2.20085E-07, 8.12784E-08,
$  2.09389E-08, 7.91748E-10/
 DATA (BLAM(I, 3),I=1,17)
$/ 1.64979E-00, 3.21283E-01, 1.05840E-01, 1.99022E-02, 4.50680E-03,
$  1.39059E-03, 4.64025E-04, 1.77849E-04, 5.86555E-05, 2.31590E-05,
$  7.22542E-06, 1.40600E-06, 3.66400E-07, 1.58688E-07, 5.92765E-08,
$  2.16017E-08, 7.80343E-10/
 DATA (BLAM(I, 4),I=1,17)
$/ 1.89688E-00, 4.14764E-01, 1.03639E-01, 2.01082E-02, 4.42658E-03,
$  1.42549E-03, 4.67963E-04, 1.77740E-04, 5.77599E-05, 2.79235E-05,
$  9.64535E-06, 1.50572E-06, 2.38752E-07, 3.40681E-08, 2.76327E-08,
$  2.22880E-08, 7.92748E-10/
 DATA (BLAM(I, 5),I=1,16)
$/ 1.85587E-00, 4.44567E-01, 1.13852E-01, 2.19628E-02, 5.16312E-03,
$  1.53095E-03, 5.23311E-04, 1.90690E-04, 5.10860E-05, 3.24598E-05,
$  1.22247E-05, 1.15084E-06, 3.88620E-07, 2.76019E-08, 2.96448E-08,
$  7.82691E-10/
 DATA (BLAM(I, 6),I=1,16)
$/ 2.60230E-00, 5.53390E-01, 1.17430E-01, 2.29680E-02, 5.76010E-03,
$  1.59460E-03, 5.72220E-04, 1.97920E-04, 4.03530E-05, 3.61940E-05,
$  1.16860E-05, 1.76488E-07, 2.68330E-08, 2.75715E-08, 2.78522E-08,
$  8.07920E-10/
 DATA (BLAM(I, 7),I=1,15)
$/ 1.93066E-00, 5.46034E-01, 1.38453E-01, 2.43432E-02, 6.14342E-03,
$  1.66642E-03, 6.04011E-04, 2.57148E-04, 7.12453E-05, 5.05745E-05,
$  1.66286E-05, 1.06953E-06, 3.17196E-08, 2.34170E-08, 1.18534E-09/
 DATA (BLAM(I, 8),I=1,15)
$/ 1.81367E-00, 4.59880E-01, 1.24126E-01, 2.33524E-02, 6.69854E-03,
$  1.76473E-03, 6.43176E-04, 2.38478E-04, 7.64934E-05, 6.02412E-05,
$  1.78196E-05, 1.07271E-06, 9.82652E-07, 2.18503E-08, 8.20427E-10/
 DATA (BLAM(I, 9),I=1,14)
$/ 1.77610E-00, 4.09978E-01, 1.33428E-01, 2.70348E-02, 8.67964E-03,
$  2.69946E-03, 4.76599E-04, 1.95245E-04, 6.91984E-05, 8.19106E-05,
$  1.03762E-06, 2.48808E-07, 3.24306E-11, 2.41889E-12/
 DATA (BLAM(I,10),I=1,13)
$/ 1.76149E-00, 2.83592E-01, 1.22365E-01, 3.36318E-02, 1.12570E-02,
$  4.31053E-03, 1.48404E-03, 6.87533E-05, 1.22893E-06, 1.03353E-06,
$  1.53456E-07, 3.63928E-11, 1.60657E-12/
 DATA (BLAM(I,11),I=1,11)
$/ 1.84601E-00, 3.80248E-01, 1.47464E-01, 6.18521E-02, 1.56054E-02,
$  4.99222E-03, 1.78649E-03, 8.25413E-04, 8.58480E-04, 1.03618E-06,
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     $  2.06826E-07/
C
C
C ------- GAMMA RELEASE COEFFICIENTS IN 11 ENERGY GROUPS -------
C
      DATA (GALF(I, 1),I=1,19)
     $/ 2.81727E-02, 1.61475E-02, 1.24735E-02, 6.96348E-03, 1.87744E-03,
     $  9.71714E-04, 1.09023E-04, 4.14073E-05, 2.04980E-05, 5.83658E-06,
     $  3.46526E-08, 3.74680E-07, 7.46642E-08, 1.31237E-08, 1.29306E-09,
     $  2.28722E-11, 1.35579E-12, 7.68209E-14, 1.09613E-17/
      DATA (GALF(I, 2),I=1,19)
     $/ 1.84161E-01, 1.01402E-01, 8.91371E-02, 2.68436E-02, 6.38638E-03,
     $  2.05240E-03, 5.84780E-04, 1.37557E-04, 1.80822E-05, 4.74585E-05,
     $  1.53297E-06, 1.70129E-06, 2.14785E-07, 2.85053E-08, 6.34009E-08,
     $  1.47132E-08, 1.86140E-11, 3.41440E-13, 2.78360E-11/
      DATA (GALF(I, 3),I=1,19)
     $/ 3.79060E-02, 2.97680E-02, 1.38500E-02, 5.70730E-03, 2.68260E-03,
     $  5.68940E-04, 1.53400E-04, 9.49510E-05, 2.19780E-05, 2.46670E-06,
     $  6.55250E-07, 4.19120E-08, 3.25060E-09, 1.07120E-09, 2.92050E-11,
     $  1.92510E-12, 2.44020E-13, 2.17720E-17, 5.41800E-18/
      DATA (GALF(I, 4),I=1,14)
     $/ 1.54224E-02, 1.50988E-02, 8.18830E-03, 5.02757E-03, 2.94196E-03,
     $  7.46236E-04, 9.33988E-05, 6.77752E-05, 9.79902E-06, 4.06647E-06,
     $  7.51465E-08, 7.30439E-12, 2.30393E-16, 4.36459E-17/
      DATA (GALF(I, 5),I=1,18)
     $/ 4.08810E-03, 3.77566E-03, 1.83077E-03, 1.41679E-03, 1.00882E-03,
     $  1.63591E-04, 6.12283E-05, 1.86629E-05, 6.57733E-06, 4.07153E-06,
     $  3.64684E-07, 2.07467E-08, 9.06640E-09, 5.35514E-11,-4.17564E-12,
     $  2.82216E-11, 2.93299E-15, 1.27293E-17/
      DATA (GALF(I, 6),I=1,16)
     $/ 7.22591E-03, 6.79147E-03, 3.49954E-03, 2.04670E-03, 1.13840E-03,
     $  3.94452E-05, 2.42358E-05, 7.90396E-06, 2.39364E-06,-4.22440E-09,
     $  2.09726E-10, 3.16494E-09, 1.53665E-09, 1.23943E-13, 4.77990E-20,
     $  1.46640E-17/
      DATA (GALF(I, 7),I=1,15)
     $/ 5.89171E-03, 5.67017E-03, 3.04604E-03, 1.68190E-03, 7.27511E-04,
     $  2.65719E-04, 6.97154E-05, 2.46474E-06, 9.38050E-06, 4.27331E-07,
     $  7.20586E-11, 4.09786E-11, 1.50575E-14, 1.78360E-22, 4.33630E-18/
      DATA (GALF(I, 8),I=1,16)
     $/ 3.99571E-03, 3.92439E-03, 2.13233E-03, 2.05007E-03, 1.32325E-03,
     $  3.18426E-04, 2.66773E-05, 5.77121E-06, 1.84319E-06, 3.93096E-07,
     $  1.56163E-07, 6.10931E-11, 2.05073E-11, 7.37639E-17, 1.32823E-18,
     $  1.46853E-18/
      DATA (GALF(I, 9),I=1,16)
     $/ 3.18297E-03, 2.87557E-03, 1.71152E-03, 1.07626E-03, 5.61154E-04,
     $  1.39928E-04, 1.98783E-04, 1.19133E-06, 8.09801E-08, 1.98030E-08,
     $  6.39300E-09, 4.66038E-11, 2.47843E-14, 4.98744E-17, 1.59311E-21,
     $  5.61784E-20/
      DATA (GALF(I,10),I=1,12)
     $/ 4.34812E-03, 2.76970E-03, 1.28471E-03, 1.16145E-04, 3.57022E-05,
     $  5.59007E-06, 7.51496E-08, 2.92101E-09, 4.09410E-13, 5.39912E-12,
     $  1.15456E-13, 9.99597E-24/
      DATA (GALF(I,11),I=1,10)
     $/ 8.05801E-04, 4.59189E-04, 1.28026E-04, 1.36264E-06, 4.33460E-07,
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$  4.58608E-10, 5.41038E-12, 2.32383E-14, 1.94953E-14, 8.30431E-23/
 DATA (GLAM(I, 1),I=1,17)
$/ 2.82757E-00, 8.89029E-01, 4.22587E-01, 1.08402E-01, 4.33672E-02,
$  1.60004E-02, 4.18209E-03, 7.22565E-04, 5.44143E-04, 2.11393E-04,
$  1.37534E-05, 1.37567E-05, 2.97704E-06, 8.40208E-07, 2.69565E-07,
$  2.60274E-08, 2.80247E-08, 8.09492E-09, 1.19086E-10/
 DATA (GLAM(I, 2),I=1,19)
$/ 2.93360E-00, 1.03920E-00, 4.25680E-01, 1.36710E-01, 4.83600E-02,
$  1.49010E-02, 3.81150E-03, 1.02160E-03, 2.18250E-04, 1.73490E-04,
$  3.49170E-05, 1.19200E-05, 3.29030E-06, 3.06010E-06, 1.14770E-06,
$  1.14540E-07, 1.58230E-08, 1.58230E-08, 7.20110E-10/
 DATA (GLAM(I, 3),I=1,19)
$/ 2.56030E-00, 5.12960E-01, 1.30490E-01, 4.28240E-02, 1.44110E-02,
$  4.92420E-03, 1.04920E-03, 5.90630E-04, 1.97380E-04, 3.60340E-05,
$  1.68960E-05, 2.64920E-06, 6.91700E-07, 5.69430E-07, 1.56540E-07,
$  2.17630E-08, 2.17630E-08, 6.77520E-11, 5.55460E-11/
 DATA (GLAM(I, 4),I=1,14)
$/ 2.60302E-00, 5.01187E-01, 1.36406E-01, 4.35751E-02, 1.55072E-02,
$  6.91643E-03, 2.51559E-03, 3.78648E-04, 1.29181E-04, 4.45812E-05,
$  6.31631E-07, 2.70265E-08, 9.05062E-09, 6.27813E-11/
 DATA (GLAM(I, 5),I=1,18)
$/ 2.55790E-00, 4.94000E-01, 1.34410E-01, 3.97860E-02, 1.43250E-02,
$  5.49880E-03, 9.39940E-04, 5.98660E-04, 1.73200E-04, 8.35760E-05,
$  3.84700E-05, 1.15070E-05, 2.45130E-06, 5.31110E-07, 5.83268E-08,
$  2.81100E-08, 1.64668E-08, 1.48020E-10/
 DATA (GLAM(I, 6),I=1,16)
$/ 2.48000E-00, 4.53430E-01, 1.03040E-01, 2.78320E-02, 9.86430E-03,
$  2.37020E-03, 3.97880E-04, 1.27710E-04, 6.64270E-05, 8.47111E-06,
$  5.98080E-07, 6.67160E-07, 5.97900E-07, 2.15960E-08, 9.38200E-11,
$  1.04240E-11/
 DATA (GLAM(I, 7),I=1,15)
$/ 2.66356E-00, 5.13407E-01, 1.38354E-01, 4.75150E-02, 1.68302E-02,
$  7.90924E-03, 3.05388E-03, 1.46808E-04, 3.79582E-04, 7.59741E-05,
$  7.66677E-07, 5.82384E-07, 2.12375E-08, 2.45925E-11, 2.45775E-12/
 DATA (GLAM(I, 8),I=1,16)
$/ 2.63674E-00, 5.08947E-01, 1.47318E-01, 3.95919E-02, 1.27392E-02,
$  4.17817E-03, 1.46059E-03, 4.67151E-04, 1.31245E-04, 1.31198E-04,
$  7.49333E-05, 9.56698E-07, 5.97448E-07, 1.01114E-07, 3.42943E-11,
$  3.16140E-12/
 DATA (GLAM(I, 9),I=1,16)
$/ 2.56850E-00, 4.95990E-01, 1.35710E-01, 3.95590E-02, 1.30330E-02,
$  5.95610E-03, 3.91180E-03, 9.26650E-04, 3.09690E-04, 8.40180E-05,
$  6.09890E-05, 1.03750E-06, 5.06180E-07, 1.72020E-07, 8.82550E-09,
$  3.35125E-13/
 DATA (GLAM(I,10),I=1,12)
$/ 2.10807E-00, 3.51749E-01, 1.07055E-01, 1.65795E-02, 4.87316E-03,
$  3.18480E-03, 8.68416E-04, 6.78993E-04, 1.67665E-04, 1.04071E-06,
$  6.16916E-07, 4.87660E-08/
 DATA (GLAM(I,11),I=1,10)
$/ 1.85017E-00, 7.69562E-01, 8.55785E-02, 1.36426E-02, 8.13130E-03,
$  1.54721E-03, 7.82246E-04, 1.06090E-06, 6.17850E-07, 6.23540E-09/
 END
```

```
      BLOCK DATA NAMES
C
C   INITIALIZE FACILITY IDENTIFIERS
C
      COMMON /CHDAT/ CHEAD, CTITL, CNAM1, CNAM2, CNAME(19),
     A               CTIMP(100), CTIMD(100), CUPWR(100)
      CHARACTER CHEAD*120, CTITL*80, CNAM1*8, CNAM2*8, CNAME*98
      CHARACTER CTIMP*3, CTIMD*3, CUPWR*3
      DATA CHEAD( 1: 40) / ' IRRADIATED FUEL DOSE RATE CALCULATION A' /
      DATA CHEAD(41: 80) / 'T THREE FOOT DOSE POINT FOR            ' /
      DATA CHEAD(81:120) / '                                        ' /
      DATA CNAME( 1)(1:16) / 'FNR     STANDARD' /
      DATA CNAME( 1)(17:57)/' UNIVERSITY OF MICHIGAN FORD NUCLEAR REAC'/
      DATA CNAME( 1)(58:98)/'TOR (FNR) STANDARD ELEMENT               '/
      DATA CNAME( 2)(1:16) / 'FNR     CONTROL ' /
      DATA CNAME( 2)(17:57)/' UNIVERSITY OF MICHIGAN FORD NUCLEAR REAC'/
      DATA CNAME( 2)(58:98)/'TOR (FNR) CONTROL ELEMENT                '/
      DATA CNAME( 3)(1:16) / 'GATRIGAFROD     ' /
      DATA CNAME( 3)(17:57)/' GENERAL ATOMIC TRIGA MARK F (GA-TRIGA F)'/
      DATA CNAME( 3)(58:98)/' SINGLE PIN ELEMENT                      '/
      DATA CNAME( 4)(1:16) / 'GTRR    STANDARD' /
      DATA CNAME( 4)(17:57)/' GEORGIA TECH RESEARCH REACTOR (GTRR) STA'/
      DATA CNAME( 4)(58:98)/'NDARD ELEMENT                            '/
      DATA CNAME( 5)(1:16) / 'MITR-II STANDARD' /
      DATA CNAME( 5)(17:57)/' MASSACHUSETTS INSTITUTE OF TECHNOLOGY RE'/
      DATA CNAME( 5)(58:98)/'SEARCH REACTOR (MITR-II) STANDARD ELEMENT'/
      DATA CNAME( 6)(1:16) / 'MURR    STANDARD' /
      DATA CNAME( 6)(17:57)/' UNIVERSITY OF MISSOURI RESEARCH REACTOR '/
      DATA CNAME( 6)(58:98)/'(MURR) STANDARD ELEMENT                  '/
      DATA CNAME( 7)(1:16) / 'NBSR    STANDARD' /
      DATA CNAME( 7)(17:57)/' NATIONAL BUREAU OF STANDARDS REACTOR (NB'/
      DATA CNAME( 7)(58:98)/'SR) STANDARD ELEMENT                     '/
      DATA CNAME( 8)(1:16) / 'OSTR    ROD     ' /
      DATA CNAME( 8)(17:57)/' OREGON STATE UNIVERSITY TRIGA REACTOR (O'/
      DATA CNAME( 8)(58:98)/'STR) SINGLE PIN ELEMENT                  '/
      DATA CNAME( 9)(1:16) / 'RINSC   STANDARD' /
      DATA CNAME( 9)(17:57)/' RHODE ISLAND NUCLEAR SCIENCE CENTER REAC'/
      DATA CNAME( 9)(58:98)/'TOR (RINSC) STANDARD ELEMENT             '/
      DATA CNAME(10)(1:16) / 'TAM-NSCRSTANDARD' /
      DATA CNAME(10)(17:57)/' TEXAS A&M NUCLEAR SCIENCE CENTER REACTOR'/
      DATA CNAME(10)(58:98)/' (TAM-NSCR) 4-ROD STANDARD ELEMENT       '/
      DATA CNAME(11)(1:16) / 'TAM-NSCRCONTROL ' /
      DATA CNAME(11)(17:57)/' TEXAS A&M NUCLEAR SCIENCE CENTER REACTOR'/
      DATA CNAME(11)(58:98)/' (TAM-NSCR) 3-ROD CONTROL ELEMENT        '/
      DATA CNAME(12)(1:16) / 'UCNR    STANDARD' /
      DATA CNAME(12)(17:57)/' UNION CARBIDE NUCLEAR REACTOR (UCNR) STA'/
      DATA CNAME(12)(58:98)/'NDARD ELEMENT                            '/
      DATA CNAME(13)(1:16) / 'UCNR    CONTROL ' /
      DATA CNAME(13)(17:57)/' UNION CARBIDE NUCLEAR REACTOR (UCNR) CON'/
      DATA CNAME(13)(58:98)/'TROL ELEMENT                             '/
      DATA CNAME(14)(1:16) / 'UVAR    12-PLATE' /
      DATA CNAME(14)(17:57)/' UNIVERSITY OF VIRGINIA REACTOR (UVAR) 12'/
      DATA CNAME(14)(58:98)/'-PLATE ELEMENT                           '/
      DATA CNAME(15)(1:16) / 'UVAR    18-PLATE' /
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DATA CNAME(15)(17:57)/' UNIVERSITY OF VIRGINIA REACTOR (UVAR) 18'/
DATA CNAME(15)(58:98)/'-PLATE ELEMENT                          '/
DATA CNAME(16)(1:16) / 'UVAR    PARTIAL ' /
DATA CNAME(16)(17:57)/' UNIVERSITY OF VIRGINIA REACTOR (UVAR) PA'/
DATA CNAME(16)(58:98)/'RTIAL ELEMENT                           '/
DATA CNAME(17)(1:16) / 'UVAR    CONTROL ' /
DATA CNAME(17)(17:57)/' UNIVERSITY OF VIRGINIA REACTOR (UVAR) CO'/
DATA CNAME(17)(58:98)/'NTROL ELEMENT                           '/
DATA CNAME(18)(1:16) / 'UWNR    STANDARD' /
DATA CNAME(18)(17:57)/' UNIVERSITY OF WISCONSIN NUCLEAR REACTOR '/
DATA CNAME(18)(58:98)/'(UWNR) 4-ROD STANDARD ELEMENT           '/
DATA CNAME(19)(1:16) / 'WSUR    STANDARD' /
DATA CNAME(19)(17:57)/' WASHINGTON STATE UNIVERSITY REACTOR (WSU'/
DATA CNAME(19)(58:98)/'R) 4-ROD STANDARD ELEMENT               '/
END
```

```
      BLOCK DATA NUMBS
C
C   INITIALIZE CONTROL VARIABLES AND FLUX-TO-DOSE CONVERSIONS
C
      COMMON /FACTR/ BTRAN(19,11), GTRAN(19,11), BFTOD(11), GFTOD(11)
      COMMON /FLAGS/ NAMID, MPTIM, MDTIM, NPTIM, NDTIM, NCASE, MEDIT
      COMMON /LIMIT/ NBG, NGG, NPMAX, NDMAX, LB(11), LG(11),
     A               TMINF, TMAXF, EXPLL, ERR
      DATA NPTIM, NDTIM, NCASE /0,0,1/
      DATA NBG, NGG, NPMAX, NDMAX /11,11,100,100/
      DATA LB /16,17,17,17,16,16,15,15,14,13,11/
      DATA LG /19,19,19,14,18,16,15,16,16,12,10/
      DATA TMINF, TMAXF, ERR, EXPLL / 0.4, 1.0E+9, 1.0E-25, -675.81 /
C
C
C -------- BETA FLUX-TO-DOSE CONVERSION FACTORS ----------------------
C --------       (REM/HR) PER (MEV/CM**2-SEC)     ----------------------
C
      DATA (BFTOD(I),I=1,11)
     A/ 1.3700E-03, 3.6200E-04, 1.8300E-04, 1.2200E-04, 9.2100E-05,
     B  7.4300E-05, 6.2100E-05, 4.8200E-05, 3.6200E-05, 2.8900E-05,
     C  2.3000E-05 /
C
C
C ------- GAMMA FLUX-TO-DOSE CONVERSION FACTORS ----------------------
C -------        (REM/HR) PER (MEV/CM**2-SEC)     ----------------------
C
      DATA (GFTOD(I),I=1,11)
     A/ 2.5200E-06, 2.2200E-06, 1.9100E-06, 1.7300E-06, 1.6000E-06,
     B  1.5100E-06, 1.4300E-06, 1.3200E-06, 1.2000E-06, 1.1300E-06,
     C  1.0500E-06 /
      END
```

```
      BLOCK DATA TRANS
C
C   INITIALIZE ENERGY TRANSPORT FACTORS
C
      COMMON /FACTR/ BTRAN(19,11), GTRAN(19,11), BFTOD(11), GFTOD(11)
C
C
C -------- BETA ENERGY SOURCE-TO-FLUX AT THREE FEET CONVERSIONS -------
C --------             (MEV/CM**2-SEC) PER (MEV/SEC)              -------
C
      DATA (BTRAN( 1,I),I=1,11)
     A / 0.0000E+00, 3.2085E-06, 2.6779E-06, 2.6732E-06, 3.7042E-06,
     B   5.8984E-06, 6.9775E-06, 4.5964E-06, 9.1046E-06, 7.8851E-06,
     C   9.3055E-06 /
      DATA (BTRAN( 2,I),I=1,11)
     A / 0.0000E+00, 6.4170E-06, 5.3558E-06, 5.3463E-06, 7.4083E-06,
     B   6.1804E-06, 3.9396E-06, 2.5325E-06, 1.7078E-06, 4.9398E-06,
     C   1.1574E-05 /
      DATA (BTRAN( 3,I),I=1,11)
     A / 0.0000E+00, 0.0000E+00, 2.5222E-07, 5.3366E-07, 1.0437E-06,
     B   1.6448E-06, 2.2260E-06, 2.4539E-06, 3.2345E-06, 2.8015E-06,
     C   1.4893E-06 /
      DATA (BTRAN( 4,I),I=1,11)
     A / 0.0000E+00, 0.0000E+00, 7.8339E-06, 5.4755E-06, 5.2127E-06,
     B   5.8157E-06, 6.6306E-06, 9.2760E-06, 8.5386E-06, 7.6974E-06,
     C   1.0202E-05 /
      DATA (BTRAN( 5,I),I=1,11)
     A / 0.0000E+00, 1.7994E-06, 1.2852E-06, 4.3333E-06, 1.4968E-06,
     B   3.5194E-06, 2.5778E-06, 2.2849E-06, 5.3921E-06, 6.9775E-06,
     C   4.8036E-06 /
      DATA (BTRAN( 6,I),I=1,11)
     A / 0.0000E+00, 4.6980E-06, 5.7725E-06, 6.4858E-06, 8.4220E-06,
     B   2.4102E-06, 3.0608E-06, 7.8754E-06, 5.5364E-06, 3.8318E-06,
     C   3.7395E-06 /
      DATA (BTRAN( 7,I),I=1,11)
     A / 0.0000E+00, 0.0000E+00, 4.2795E-06, 7.1175E-06, 7.5393E-06,
     B   5.7442E-06, 3.8622E-06, 6.1694E-06, 6.4376E-06, 6.9432E-06,
     C   5.8047E-06 /
      DATA (BTRAN( 8,I),I=1,11)
     A / 0.0000E+00, 0.0000E+00, 2.5222E-07, 5.3366E-07, 1.0437E-06,
     B   1.6448E-06, 2.2260E-06, 2.4539E-06, 3.2345E-06, 2.8015E-06,
     C   1.4893E-06 /
      DATA (BTRAN( 9,I),I=1,11)
     A / 0.0000E+00, 5.2798E-06, 8.9258E-07, 1.6874E-06, 2.8225E-06,
     B   6.7299E-06, 2.1674E-06, 6.2458E-06, 3.5203E-06, 7.7172E-06,
     C   3.6014E-06 /
      DATA (BTRAN(10,I),I=1,11)
     A / 0.0000E+00, 0.0000E+00, 2.6296E-07, 2.4567E-07, 9.6792E-07,
     B   4.8966E-07, 1.2425E-06, 8.0982E-07, 1.2304E-06, 1.0265E-06,
     C   1.5354E-06 /
      DATA (BTRAN(11,I),I=1,11)
     A / 0.0000E+00, 0.0000E+00, 3.5061E-07, 3.2756E-07, 1.2906E-06,
     B   6.5289E-07, 1.6557E-06, 1.0798E-06, 1.6405E-06, 1.3687E-06,
     C   2.0472E-06 /
```

```
      DATA (BTRAN(12,I),I=1,11)
     A / 0.0000E+00, 0.0000E+00, 7.9439E-06, 5.6275E-06, 5.3601E-06,
     B    6.0428E-06, 7.0282E-06, 8.0501E-06, 8.7862E-06, 8.1463E-06,
     C    9.3512E-06 /
      DATA (BTRAN(13,I),I=1,11)
     A / 0.0000E+00, 6.8305E-06, 5.7383E-06, 5.7662E-06, 6.1737E-06,
     B    6.8786E-06, 4.1503E-06, 2.6119E-06, 1.7450E-06, 1.1435E-05,
     C    7.6254E-06 /
      DATA (BTRAN(14,I),I=1,11)
     A / 0.0000E+00, 5.4771E-06, 4.6202E-06, 4.6666E-06, 5.0597E-06,
     B    5.7815E-06, 6.6331E-06, 1.2550E-05, 1.0891E-05, 1.0310E-05,
     C    3.4242E-06 /
      DATA (BTRAN(15,I),I=1,11)
     A / 0.0000E+00, 3.3240E-06, 2.7850E-06, 2.7899E-06, 2.9704E-06,
     B    3.2789E-06, 3.5910E-06, 5.3590E-06, 4.5263E-06, 8.9985E-06,
     C    6.3820E-06 /
      DATA (BTRAN(16,I),I=1,11)
     A / 0.0000E+00, 6.6478E-06, 1.5143E-06, 8.3056E-06, 2.4775E-06,
     B    9.9560E-06, 3.8939E-06, 1.1586E-05, 1.1993E-05, 2.6510E-06,
     C    4.0822E-06 /
      DATA (BTRAN(17,I),I=1,11)
     A / 0.0000E+00, 6.6478E-06, 5.5699E-06, 5.5797E-06, 5.9407E-06,
     B    6.5578E-06, 4.0583E-06, 2.5780E-06, 1.7293E-06, 1.0658E-05,
     C    9.1412E-06 /
      DATA (BTRAN(18,I),I=1,11)
     A / 0.0000E+00, 0.0000E+00, 2.6296E-07, 2.4567E-07, 9.6792E-07,
     B    4.8966E-07, 1.2425E-06, 8.0982E-07, 1.2304E-06, 1.0265E-06,
     C    1.5354E-06 /
      DATA (BTRAN(19,I),I=1,11)
     A / 0.0000E+00, 0.0000E+00, 2.6296E-07, 2.4567E-07, 9.6792E-07,
     B    4.8966E-07, 1.2425E-06, 8.0982E-07, 1.2304E-06, 1.0265E-06,
     C    1.5354E-06 /
C
C
C ------- GAMMA ENERGY SOURCE-TO-FLUX AT THREE FEET CONVERSIONS -------
C -------               (MEV/CM**2-SEC) PER (MEV/SEC)             -------
C
      DATA (GTRAN( 1,I),I=1,11)
     A/ 8.4540E-6, 8.7885E-6, 8.2688E-6, 8.2280E-6, 8.2243E-6,
     B  8.2201E-6, 8.2157E-6, 8.2081E-6, 8.2004E-6, 8.1941E-6,
     C  8.1875E-6 /
      DATA (GTRAN( 2,I),I=1,11)
     A/ 8.5477E-6, 8.7779E-6, 8.3431E-6, 8.3064E-6, 8.3036E-6,
     B  8.3012E-6, 8.2989E-6, 8.2953E-6, 8.2927E-6, 8.2908E-6,
     C  8.2888E-6 /
      DATA (GTRAN( 3,I),I=1,11)
     A/ 5.1411E-6, 7.2077E-6, 7.5974E-6, 7.7277E-6, 7.7337E-6,
     B  7.6776E-6, 7.6261E-6, 7.5403E-6, 7.4279E-6, 7.3181E-6,
     C  7.2088E-6 /
      DATA (GTRAN( 4,I),I=1,11)
     A/ 8.5397E-6, 8.9230E-6, 8.3685E-6, 8.3259E-6, 8.3223E-6,
     B  8.3177E-6, 8.3127E-6, 8.3038E-6, 8.2947E-6, 8.2871E-6,
     C  8.2790E-6 /
      DATA (GTRAN( 5,I),I=1,11)
```

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A/ 7.8736E-6, 8.9897E-6, 8.2950E-6, 8.2545E-6, 8.2516E-6,
B  8.2431E-6, 8.2317E-6, 8.2099E-6, 8.1814E-6, 8.1562E-6,
C  8.1288E-6 /
 DATA (GTRAN( 6,I),I=1,11)
A/ 7.8917E-6, 8.8212E-6, 8.1916E-6, 8.1530E-6, 8.1499E-6,
B  8.1425E-6, 8.1329E-6, 8.1143E-6, 8.0905E-6, 8.0694E-6,
C  8.0461E-6 /
 DATA (GTRAN( 7,I),I=1,11)
A/ 8.0532E-6, 8.5996E-6, 8.0177E-6, 7.9768E-6, 7.9733E-6,
B  7.9676E-6, 7.9609E-6, 7.9485E-6, 7.9340E-6, 7.9215E-6,
C  7.9081E-6 /
 DATA (GTRAN( 8,I),I=1,11)
A/ 5.1411E-6, 7.2077E-6, 7.5974E-6, 7.7277E-6, 7.7337E-6,
B  7.6776E-6, 7.6261E-6, 7.5403E-6, 7.4279E-6, 7.3181E-6,
C  7.2088E-6 /
 DATA (GTRAN( 9,I),I=1,11)
A/ 8.4266E-6, 8.9058E-6, 8.2411E-6, 8.1934E-6, 8.1920E-6,
B  8.1877E-6, 8.1821E-6, 8.1717E-6, 8.1615E-6, 8.1531E-6,
C  8.1443E-6 /
 DATA (GTRAN(10,I),I=1,11)
A/ 3.4675E-6, 5.7263E-6, 6.1812E-6, 6.3901E-6, 6.4560E-6,
B  6.4251E-6, 6.3903E-6, 6.3151E-6, 6.1957E-6, 6.0671E-6,
C  5.9354E-6 /
 DATA (GTRAN(11,I),I=1,11)
A/ 4.4957E-6, 6.7043E-6, 7.0670E-6, 7.2332E-6, 7.2524E-6,
B  7.2019E-6, 7.1537E-6, 7.0695E-6, 6.9545E-6, 6.8397E-6,
C  6.7245E-6 /
 DATA (GTRAN(12,I),I=1,11)
A/ 8.4440E-6, 8.8282E-6, 8.2791E-6, 8.2371E-6, 8.2335E-6,
B  8.2289E-6, 8.2238E-6, 8.2150E-6, 8.2057E-6, 8.1980E-6,
C  8.1898E-6 /
 DATA (GTRAN(13,I),I=1,11)
A/ 8.5814E-6, 8.8183E-6, 8.3842E-6, 8.3478E-6, 8.3447E-6,
B  8.3421E-6, 8.3396E-6, 8.3355E-6, 8.3322E-6, 8.3296E-6,
C  8.3268E-6 /
 DATA (GTRAN(14,I),I=1,11)
A/ 8.5182E-6, 8.7572E-6, 8.3368E-6, 8.3019E-6, 8.2980E-6,
B  8.2947E-6, 8.2916E-6, 8.2865E-6, 8.2812E-6, 8.2768E-6,
C  8.2719E-6 /
 DATA (GTRAN(15,I),I=1,11)
A/ 8.4358E-6, 8.7991E-6, 8.2708E-6, 8.2300E-6, 8.2262E-6,
B  8.2218E-6, 8.2169E-6, 8.2085E-6, 8.1997E-6, 8.1923E-6,
C  8.1845E-6 /
 DATA (GTRAN(16,I),I=1,11)
A/ 8.4480E-6, 8.8118E-6, 8.2223E-6, 8.1769E-6, 8.1743E-6,
B  8.1703E-6, 8.1657E-6, 8.1578E-6, 8.1506E-6, 8.1450E-6,
C  8.1391E-6 /
 DATA (GTRAN(17,I),I=1,11)
A/ 8.5432E-6, 8.7944E-6, 8.3452E-6, 8.3081E-6, 8.3053E-6,
B  8.3028E-6, 8.3002E-6, 8.2959E-6, 8.2926E-6, 8.2899E-6,
C  8.2870E-6 /
 DATA (GTRAN(18,I),I=1,11)
A/ 3.4675E-6, 5.7263E-6, 6.1812E-6, 6.3901E-6, 6.4560E-6,
B  6.4251E-6, 6.3903E-6, 6.3151E-6, 6.1957E-6, 6.0671E-6,
```

```
C  5.9354E-6 /
 DATA (GTRAN(19,I),I=1,11)
A/ 3.4675E-6, 5.7263E-6, 6.1812E-6, 6.3901E-6, 6.4560E-6,
B  6.4251E-6, 6.3903E-6, 6.3151E-6, 6.1957E-6, 6.0671E-6,
C  5.9354E-6 /
 END
```

APPENDIX B
FUELDR USER'S GUIDE

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FUELDR USER'S GUIDE

Program flow for FUELDR is illustrated schematically in Figure B-1. User supplied input data is processed by subroutine RDINP. Subroutine MATCH checks the input facility name and element type and flags the appropriate set of effective transport factors. Some initial input checking is done in RDINP and in MATCH. More extensive input checking is done in CKINP and units conversions are performed. If errors are detected, control is passed to subroutine ERRS for issuance of a diagnostic message and the problem is terminated. Subroutine TLOAD is used to initialize decay times if one of the default sets has been requested.

Subroutine EDINP produces an edit of the power history model constructed from user input. The step length, power level, fission rate, and step exposure are listed for each power step. Cumulative exposures and irradiation times are also listed. EDINP also produces an edit of the user input or default decay times. All decay times are referenced to the end of the last power step.

Subroutine DRCAL manages the dose rate calculation by reconciling the different effective decay times for each power step and performing successive calls to UFYLD for each power step-decay time combination. Subroutine UFYLD determines beta and gamma energy yields resulting from a unit fission rate over the specified power step length. The multigroup energy yields are then normalized by the total fission rate in the power step.

Subroutine EDOUT edits the calculated dose rates at the 3 ft (91.44 cm) dose point. The total gamma, total beta, and the sum of total beta and total gamma dose rates are listed for each decay time. Depending on the edit option selected, an energy release rate edit may also be produced. Multigroup edits of the beta and gamma dose rates and energy release rates may also be requested.

The fission product impulse source function coefficients are initialized in BLOCK DATA COEFS. Effective gamma and beta energy transport factors for each of the element types are initialized in BLOCK DATA TRANS. Facility names and element types are initialized in BLOCK DATA NAMES. Beta and gamma flux-to-dose conversion factors and miscellaneous control variables are initialized in BLOCK DATA NUMBS.

FUELDR input consists of a title card, a control card, one or more power step cards, and one or more optional decay time cards. Table B-1 contains detailed variable and input format descriptions. Change cases require only a title card and control card if the previously specified power history and decay time data are to be used.

Any power history may be represented by a series of constant power steps. The only restrictions are that the total number of steps not exceed 100 and that the total decay time from the end of any power step not exceed 10^9 s. The minimum decay time should not be less than 0.4 s.

At short decay times, dose rates are sensitive to both the total exposure on the element and to the details of the irradiation history that generated that total exposure. With increasing decay time, the dose rate remains sensitive to the total exposure but the manner in which that exposure was accumulated becomes less significant.

In many cases, modeling only the last few times at power is adequate to verify the element is self-protecting for some specified period. Very simple constant power irradiation history models may be used for exposures accumulated in the distant past.

The entire exposure for some past time interval may be modeled as a burst either at the beginning or at the end of that time interval. The calculated dose rates from these two cases will envelope the actual dose rate. In general, these simple power history models may be used with little error if the decay time is long compared to the length of the irradiation interval.

Table B-3 contains input listings for several sample problems. Sample problem A models a Rhode Island Nuclear Science Center element irradiation. One of the default decay time sets is used and full output editing is requested. The irradiation is modeled as a single eight hour irradiation step at 65 kilowatts. The default energy per fission value (200 MeV/fission) is used.

Sample problem B1 models a one hour irradiation at one kilowatt for the University of Virginia Reactor 12-plate element. A default decay time set is used and only the dose rate summary edits are

```
FUELDR
:
:....RDINP
:    :
:    :....MATCH
:    :    :
:    :    :....ERRS
:    :
:    :....CKINP
:    :    :
:    :    :....ERRS
:    :
:    :....TLOAD
:    :
:    :....ERRS
:
:....EDINP
:
:... DRCAL
:    :
:    :....ZARAY
:    :
:    :....ERRS
:    :
:    :....UFYLD
:
:... EDOUT
```

Figure B-1. FUELDR program flow.

Table B-1. FUELDR input description

CARD A: TITLE CARD - FORMAT(80A1)

Columns	Variable	Description
1 - 80	CTITL	Case title of up to 80 characters. May be blank.

CARD B: CONTROL CARD - FORMAT(A8,A8,3I4)

Columns	Variable	Description
1 - 8	CNAM1	Left justified facility identifier from Table B-2
9 - 16	CNAM2	Left justified element identifier from Table B-2
17 - 20	MPTIM	Number of power steps (1 to 100 allowed). = 0 to use power history from previous case.
21 - 24	MDTIM	Number of decay times (1 to 100 allowed).
		= 0 to use decay times from previous case.
		< 0 to select a default decay time set.
		- 1 for 32 times (1, 2, 3, 5, ... 5 x 10^7 s)
		- 2 for 32 times (4 per decade beginning at 1 s)
		- 3 for 24 times (0.1 to 600 minutes)
		- 4 for 24 times (0.1 to 240 hours)
		- 5 for 32 times (0.1 to 1825 days)
		- 6 for 24 times (0.1 to 10 years)
		- 7 for 48 times (mixed units)
25 - 28	MEDIT	Output edit control flag
		= 0 for dose rate summary edits
		= 1 to also edit energy release rates
		= 2 to also edit multigroup dose rates and multigroup energy release rates

Table B-1. (continued)

CARDS C: POWER STEP CARDS - FORMAT (I4,E12.5,1X,A3,E12.5,1X,A3, E12.5)
(MPTIM cards; no cards if MPTIM = 0)

Columns	Variable	Description
1 - 4	NS	Number of power step described on this card; 1 to MPTIM. Must be in ascending order; MPTIM is the most recent power step.
5 - 16	TPINP(NS)	Length of power step NS.
13 - 20	CTIMP(NS)	Units of power step length; may be "SEC", "MIN", "HRS", "DAY", "YRS", or " ". The default units are seconds if no entry is made.
21 - 32	PWRIN(NS)	Power level of step NS.
34 - 36	CUPWR(NS)	Units of power level; may be " W", " W ", "W ", " KW", "KW ", " MW", or "MW ". The default units are kilowatts if no entry is made.
37 - 48	FISEN(NS)	Energy per fission during this power step. If no entry is made, 200 MeV per fission is assumed.

CARDS D: DECAY TIMES - FORMAT(5(E12.5,1X,A3))
(MDTIM entries; none if MDTIM ≤ 0)

Columns	Variable	Description
1 - 12	TDINP(i)	Decay time for first decay step on this card.
14 - 16	CTIMD(i)	Units of decay time. Allowable values are the same as for the power step length.
17 - 28	TDINP(j)	Decay time for second decay step on this card.
30 - 32	CTIMD(j)	Units for this decay step length.
.	.	.
.	.	.
.	.	.
65 - 76	TDINP(m)	Decay time for fifth decay step on this card.
78 - 80	CTIMD(m)	Units for this decay step length.

Table B-2. Keyword identifiers for research reactor fuel types

Facility	Element	Description
FNR	STANDARD	University of Michigan Ford Nuclear Reactor
FNR	CONTROL	University of Michigan Ford Nuclear Reactor
GATRIGAF	ROD	General Atomic TRIGA Mark F Reactor
GTRR	STANDARD	Georgia Institute of Technology Research Reactor
MITR-II	STANDARD	Massachusetts Institute of Technology Research Reactor
MURR	STANDARD	University of Missouri Research Reactor
NBSR	STANDARD	National Bureau of Standards Reactor
OSTR	ROD	Oregon State University TRIGA Reactor
RINSC	STANDARD	Rhode Island Nuclear Science Center Reactor
TAM-NSCR	STANDARD	Texas A&M University Nuclear Science Center Reactor
TAM-NSCR	CONTROL	Texas A&M University Nuclear Science Center Reactor
UCNR	STANDARD	Union Carbide Nuclear Reactor
UCNR	CONTROL	Union Carbide Nuclear Reactor
UVAR	12-PLATE	University of Virginia Reactor
UVAR	18-PLATE	University of Virginia Reactor
UVAR	PARTIAL	University of Virginia Reactor
UVAR	CONTROL	University of Virginia Reactor
UWNR	STANDARD	University of Wisconsin Nuclear Reactor
WSUR	STANDARD	Washington State University Reactor

requested. Cases B2, B3, and B4 model the same irradiation for three other element types in use at that facility using the change case feature.

Sample problem C illustrates a one megawatt-second pulse modeled as a one second step at one megawatt. The energy per fission is specified and three decay times are specified.

Sample problem D illustrates modeling for a power history consisting of multiple irradiation steps. The element is assumed to operate for eight hours per day at 1.25 megawatts. Six cycles are modeled and several methods of specifying the power level are shown.

Output from execution of the sample problems shown in Table B-3 is included in the microfiche envelope attached to the back cover.

Table B-3. FUELDR sample problems

```
         1         2         3         4         5         6         7         8
12345678901234567890123456789012345678901234567890123456789012345678901234567890

 SAMPLE PROBLEM A  -  EIGHT HOUR CONSTANT POWER IRRADIATION
RINSC   STANDARD   1  -1   3
   1    8.00      HRS    65.0      KW
 SAMPLE PROBLEM B1  -  UVAR 12-PLATE ELEMENT AT ONE KW-HOUR
UVAR    12-PLATE   1  -1   0
   1    1.00      HRS    1.00      KW
 SAMPLE PROBLEM B2  -  UVAR 18-PLATE ELEMENT AT ONE KW-HOUR
UVAR    18-PLATE   0   0   0
 SAMPLE PROBLEM B3  -  UVAR PARTIAL ELEMENT AT ONE KW-HOUR
UVAR    PARTIAL    0   0   0
 SAMPLE PROBLEM B4  -  UVAR CONTROL ELEMENT AT ONE KW-HOUR
UVAR    CONTROL    0   0   0
 SAMPLE PROBLEM C  -  ONE MEGAWATT-SECOND PULSE
OSTR    ROD        1   3   0
   1    1.00      SEC    1.00      MW     180.
    1.00      HRS    1.00    DAY     1.00      YRS
 SAMPLE PROBLEM D  -  MULTIPLE EIGHT-HOUR IRRADIATIONS
MURR    STANDARD  11  -7   0
   1    8.0       HRS    1.25      MW
   2   16.0       HRS    0.00
   3    8.0       HRS    1.25       MW
   4   16.0       HRS
   5    0.0       HRS    1.25E+06 W
   6   16.0       HRS
   7    8.0       HRS    1.25E+06  W
   8   16.0       HRS
   9    8.0       HRS    1.25E+06   W
  10   16.0       HRS
  11    8.0       HRS    1.25E+03
```

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This report describes a calculational method for the determination of biological dose rate from irradiated research reactor fuels. The calculational method is implemented in a computer program for quick and convenient assessment of multigroup gamma and beta dose rates resulting from an arbitrary (user-supplied) irradiation history. The FUELDR program calculates dose rates at a fixed dose point using built-in fission product impulse source functions and precalculated gamma and beta transport factors. The fixed dose point is located on the axial mid-plane at a distance of 91.44 cm (3 ft) from the fuel element. Transport factors are included for sixteen unique ^{235}U fuel types in use at thirteen nonpower reactor facilities.

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