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**GAMTEC II: A CODE FOR GENERATING
CONSISTENT MULTIGROUP CONSTANTS UTILIZED
IN DIFFUSION AND TRANSPORT THEORY CALCULATIONS**

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UTILIZED IN DIFFUSION AND TRANSPORT THEORY CALCULATIONS**

Page 36:

The title of case card should have a format type 2 rather than 0.

The end of case card should have a format type 9.

The Fortran II version of the GAMTEC II code has been translated into Fortran IV. The following change should be made in the input instructions when using the Fortran IV version:

Columns 4-5 contain a repetition control constant, K. A nonblank K causes this card to be treated as though it were K cards, all identical except for columns 6-10 which are blank on the last (K-1) of these cards. When the UNIVAC 1107 computer is used, the cross section tape is loaded on tape unit A. Input data is read in from the card reader and output data is to the printer.

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GAMTEC II: A CODE FOR GENERATING
CONSISTENT MULTIGROUP CONSTANTS UTILIZED
IN DIFFUSION AND TRANSPORT THEORY CALCULATIONS

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ABSTRACT

The GAMTEC II code generates multigroup constants in the energy range from 0 to 10 MeV for either homogeneous mixtures or heterogeneous arrays of cylindrical fuel elements. The thermal group constants are average over either (1) Wigner-Wilkins light moderator spectrum, (2) Wilkins heavy moderator spectrum, or (3) Maxwellian distribution. For heterogeneous arrays, the spacial thermal flux is calculated by the P_3 approximation. For epithermal energies, the slowing down distribution is described by either a P_1 or B_1 approximation to the Boltzman equation. Resonance absorption is calculated by the Adler-Nordheim method. Fast fission corrections are calculated by an n-flight collision probability technique. Multigroup constants are punched on cards in both HFN (diffusion theory) and DTF (S_n transport theory) formats.

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GAMTEC II: A CODE FOR GENERATING
CONSISTENT MULTIGROUP CONSTANTS UTILIZED
IN DIFFUSION AND TRANSPORT THEORY CALCULATIONS

INTRODUCTION

Perhaps the best known and most used procedure for calculating multigroup constants used in both transport and diffusion theory analysis of neutron multiplying media is the MUFT-SOFOCATE combination.^(1,2) MUFT provides multigroup constants in the energy range of $10 \text{ MeV} \geq E \geq 0.625 \text{ eV}$ and SOFOCATE the thermal group constants averaged over a Wigner-Wilkins spectrum for neutrons in equilibrium with hydrogen-moderated homogeneous mixtures.⁽³⁾ MUFT solves the multigroup transport equation for the first two Legendre coefficients of the directional neutron flux and the isotropic and anisotropic component of the slowing down densities due to a cosine-shaped neutron source. Hydrogen is treated either exactly or in a Selengut-Goertzel approximation.⁽⁴⁾ For energy degradation by heavy elements, both age and the Grueling-Goertzel⁽⁵⁾ approximation are available. Resonance capture and fission are computed from resonance parameters by the NR-NRIA homogeneous mixture approximation.⁽⁶⁾ Heterogeneous effects and Doppler broadening are incorporated as input to the code.

Joanou, et al., utilizing the P_1 and B_1 approximation developed a 68-group calculation called GAM-I for obtaining multigroup constants in the slowing down spectrum.⁽⁷⁾ Energy degradation is treated exactly for all elements. Resonance absorption by U^{238} and Th^{232} is treated by the Adler-Nordheim method eliminating the necessity of applying heterogeneous self-shielding and Doppler broadening effects. Also, Shudde and Dyer have coded a Fortran II version of SOFOCATE called TEMPEST.⁽⁸⁾

To calculate multigroup constants for neutron multiplying media over the energy range $10 \text{ MeV} \geq E \geq 0$ in one operation, GAM-I and TEMPEST have been combined into a single code GAMTEC II. GAM-I was previously modified by J. L. Carter to include resonance absorbers

other than U^{238} and Th^{232} in the Adler-Nordheim approximation.⁽⁹⁾ This modification was incorporated into GAMTEC II. Also included were

- An improved method for averaging the resonance absorption contribution to the multigroup constants
- An equivalent homogenization of heterogeneous geometries for the thermal group calculation
- Corrections for fuel lumping effects on the fast neutron fission of U^{238} .

GAMTEC II is written in the Fortran II symbolic language. Input is submitted in the form of punched cards and stored by absolute machine locations. After the input data for an initial problem have been prepared, additional calculations may be made by inserting only the input data that differ from the preceding problem. All cross section data and parameters are located on a single magnetic tape. For heterogeneous arrays, the maximum number of concentric regions permissible is 16. A problem requires from 1 to 3 min on an IBM-7090, depending on the complexity and options used. A brief summary of the output available includes

- Energy-dependent neutron flux from $10 \text{ MeV} \geq E \geq 0$
- Macroscopic and microscopic group constants for up to 34 groups including both P_0 and anisotropic P_1 transfers
- A two-group critical buckling calculation
- One-velocity thermal flux distribution within the unit lattice cell
- First-flight collision probabilities for fast neutrons ($E \geq 1.35 \text{ MeV}$)
- Card-punched group constants in HFN (diffusion code) and DTF (S_n transport code) formats.^(10, 11)

In this report, a general description of GAMTEC II is given; detailed discussions of the calculational methods are found in the GAM-I and TEMPEST reports.^(7, 8) The results of several problems of general interest are presented. A description of the library tape is given in Appendix A. The input instructions are given in Appendix B with a description of the output in Appendix C. A sample problem is given in Appendix D.

THERMAL GROUP CONSTANTS

The thermal group constants may be averaged by one of three approximations in TEMPEST:

- The Wigner-Wilkins light moderator equation⁽³⁾
- The Wilkins heavy moderator equation⁽¹²⁾
- A Maxwellian distribution.

For heterogeneous lattice arrays, the unit lattice cell is equivalently homogenized by volume and flux weighting atom densities. A one-velocity P_3 approximation to the transport equation is utilized to calculate the neutron distribution in the lattice cell.⁽¹³⁾ While the TEMPEST calculation has the advantage of requiring little computing time, it ignores spectral hardening within the fuel element, and the P_3 calculation becomes inaccurate for small fuel rods and water-to-uranium ratios. For heterogeneous arrays, one will probably want to obtain the averaged thermal group $\bar{\Sigma}_a$ and $\bar{v}\Sigma_f$ from a THERMOS calculation which computes the scalar thermal neutron spectrum as a function of position in a lattice by numerically solving the integral transport equation with isotropic scattering.^(14,15)

Provisions are made in GAMTEC II input for two temperatures: physical and neutron. The physical temperature is used to compute Doppler effects in the resonance capture calculations and the neutron temperature is primarily used in the thermalization calculation. A temperature value for each region is given and the average temperature used by the TEMPEST calculation is

$$\bar{T} = \frac{\sum_j \text{Regions} V_j T_j \sum_i \text{Isotopes} \xi_{ij} \Sigma_{ij}^S}{\sum_j \text{Regions} V_j \sum_i \text{Isotopes} \xi_{ij} \Sigma_{ij}^S}$$

where

ξ_{ij} = the average increase in lethargy per collision with the
ith material in the jth region

Σ_{ij}^S = the corresponding macroscopic scattering cross section

T_j = neutron temperature of the j th region

V_j = volume of the j th region.

FAST GROUP CONSTANTS

GAM-I calculates the flux and current in each of 68 groups by either a spherical harmonic P_1 or B_1 approximation for a given spatial variation of the flux at all energies, $\cos BZ$. Representation of scattering kernels is exact for all isotopes. For heterogeneous lattice arrays, the unit lattice cell is homogenized by volume weighting atom densities. Fuel geometry is accounted for by the Adler-Nordheim method for calculating resonance absorption. The method is applied to any isotope having resonance parameters on the library tape. Initially the library contains parameters only for U^{238} , Th^{232} , and Pu^{240} ; others may be added by simply generating a new data tape. The method for averaging resonance absorption contributions to the multigroup constants has been improved over that in GAM-I. Also, fuel lumping effects on fast fission in U^{238} has been incorporated into GAMTEC II.

Fast Fission Corrections

Until a fission neutron has undergone at least a few collisions, the homogeneous model is not satisfactory in approximating a lattice array, this being particularly true for fast fissioning U^{238} . Bonalumi has developed a method for calculating first-flight collision probabilities in concentric geometries.⁽¹⁶⁾ This method has been used to correct the atom densities of U^{238} for neutron energies exceeding 1.35 MeV, the approximate U^{238} fission threshold. Bonalumi's method has been extended to include n-flight contributions to the collision probabilities. Interaction effects between neighboring fuel elements are approximated by isotropically reflecting neutrons that escape to the equivalent unit cell boundary with energies exceeding 1.35 MeV.

The following derivation is for neutrons having energies in excess of 1.35 MeV. In a homogeneous mixture, the number of collisions with U^{238} per original fission neutron is

$$C_1 = \frac{f_1 P_u}{1 - \sum_i P_i \left(\frac{\sigma_{il} + v_i f_i \sigma_{if}}{\sigma_{it}} \right)} \quad (1)$$

where the cross sections and processes entering the formulation are

P_i = first flight collision probability for a neutron ($E \geq 1.35$ MeV)
with the i th material*

σ_{if} = fission cross section

σ_{il} = scattering cross section for which the neutron energy
remains above 1.35 MeV

σ_{it} = transport cross section

v_i = mean number of fission neutrons resulting from a fast
fission ($E \geq 1.35$ MeV)

f_1 = the fraction of fission neutrons born with energies
exceeding 1.35 MeV.

The summation in Equation (1) is over all material making up the unit
lattice cell. For a homogeneous mixture

$$P_i = \frac{N_i \sigma_{it}}{\sum_j N_j \sigma_{jt}} \quad (2)$$

therefore

$$C_1 = \frac{f_1 N_u \sigma_{ut}}{\sum_j N_j \sigma_{jr}} \quad (3)$$

where

$$\sigma_{jr} = \sigma_{jt} - \sigma_{jl} - v_j f_1 \sigma_{jf}$$

Equation (3) may be made equivalent to that for a heterogeneous lattice
array by applying correction factors to the atom densities

$$C_1^{\text{Het}} = \frac{w_u N_u f_1 \sigma_{tu}}{w_u N_f \sigma_{fr} + w_o \Sigma_{or}} \quad (4)$$

* The subscript u denotes U^{238}

where $N_f \sigma_{fr}$ is summed over all fast fissile isotopes in U^{238} region*, and

$$\Sigma_{or} = \sum_i N_i \sigma_{ir} - N_f \sigma_{fr} \quad .$$

The correction factor for U^{238} and all other fast fissile isotopes in U^{238} region is w_u ; w_o is the correction factor for all other isotopes. Setting the mean free paths in the weighted and unweighted mixture equal

$$w_u N_f \sigma_{tf} + w_o \Sigma_{ot} = \sum_i N_i \sigma_{it} \quad (5)$$

results in

$$w_u = \frac{C_1^{Het} \Sigma_{or} \left(1 + \frac{N_f \sigma_{ft}}{\Sigma_{ot}} \right)}{\left[f_1 N_u \sigma_{ut} - C_1^{Het} \left(N_f \sigma_{fr} - \Sigma_{or} \frac{N_f \sigma_{ft}}{\Sigma_{ot}} \right) \right]} \quad (6)$$

and

$$w_o = 1 + \frac{(1 - w_u) N_f \sigma_{ft}}{\sum_i N_i \sigma_{it} - N_f \sigma_{ft}} \quad . \quad (7)$$

The number of collisions with U^{238} per original fission neutron in a heterogeneous lattice array is

$$C_1^{Het} = f_1 \sum_{n=1}^{\infty} P_n^u \quad (8)$$

where P_n^u is the nth-flight collision probability with U^{238} . Let $P_n(J)$ be the nth-flight collision probability in the Jth region; therefore, the first flight collision probability is

$$P_1(J) = \sum_{k=1}^N F(k) P(k, J) \quad (9)$$

where N denotes the number of regions in the unit lattice cell and F(k) is the fraction of thermal neutron fission occurring in the kth region;

* An option is provided in the code to either have $N_f \sigma_{fr}$ summed over just U^{238} ($\mathcal{L}[527] = 0$) or all fast fissile materials located within a U^{238} ($\mathcal{L}[527] = 1$) or Th^{232} ($\mathcal{L}[527] = 2$) region.

$$F(k) = \frac{V(k) \bar{\varphi}(k) \sqrt{\Sigma_f(k)}}{\sum_{J=1}^N V(J) \bar{\varphi}(J) \sqrt{\Sigma_f(J)}}$$

with $V(k)$, $\bar{\varphi}(k)$, and $\sqrt{\Sigma_f(k)}$ denoting, respectively, the volume, average thermal flux, and average thermal $\sqrt{\Sigma_f}$ for the k th region. $P(k,J)$ is the probability that a neutron born uniformly in the k th region undergoes its first collision in the J th region. The expression for $P_n(J)$ becomes

$$P_n(J) = \sum_{k=1}^N P_{n-1}(k) \frac{\Sigma_{11}(k)}{\Sigma_t(k)} P(k,J) \quad (10)$$

with

$$\Sigma_{11}(k) = \sum_{i=1}^I N_i (\sigma_{i1} + v_{i1} f_{i1} \sigma_{if}) \quad (I \text{ denotes number of isotopes in } k\text{th region})$$

and

$$\Sigma_t(k) = \sum_{i=1}^I N_i \sigma_{it}$$

The n th-flight collision probability with U^{238} is

$$P_n^u = \sum_{k=1}^N \frac{\Sigma_t^u(k)}{\Sigma_t(k)} P_n(k) \quad (11)$$

where

$$\Sigma_t^u(k) = N_u \sigma_{ut} \text{ for the } k\text{th region.}$$

The interaction effects between neighboring fuel elements are approximated by isotropically reflecting neutrons that escape to the equivalent cell boundary with energies exceeding 1.35 MeV. The probability of a neutron born in the J th region escaping to the equivalent cell boundary is

$$1 - P_c(J) = 1 - \sum_{k=1}^N P(J,k)$$

Let G_{ok} be the probability that a neutron isotropically incident on the equivalent cell boundary undergoes its next collision in the k th region.⁽¹⁶⁾ Now the probability that a neutron uniformly generated in the J th region and escaping to the equivalent cell boundary undergoes its first flight collision in the k th region is

$$P'(J, K) = [1 - P_c(J)] \frac{G_{ok}}{1 - (1 - G_{oT})}$$

where

$$G_{oT} = \sum_{k=1}^N G_{ok} \quad .$$

Upon including the interaction effects, Equations (9) and (10) became

$$P_1(J) = \sum_{k=1}^N F(k) [P(k, J) + P'(k, J)] \quad (12)$$

and

$$P_n(J) = \sum_{k=1}^N P_{n-1}(k) \frac{\Sigma_{11}(k)}{\Sigma_t(k)} [P(k, J) + P'(k, J)] \quad . \quad (13)$$

In GAMTEC II, the summation in Equation (8) is continued until the contribution to the summation becomes less than 0.01%. The equivalent cell boundary used to calculate the interaction effects is taken to be

$$R_{eff} = \frac{2(\text{volume of lattice cell})}{\text{Circumference of lattice cell}} \quad .$$

Resonance Absorption

The absorption cross section has been divided into two components: (1) a slowly varying cross section, and (2) a resonance absorption cross section

$$\Sigma_{an} = \Sigma_{an}^{smooth} + \Sigma_{an}^{res} \quad (14)$$

where the subscript n denotes the n th fine group. The resonance absorption contribution to the fine group is

$$\Sigma_{an}^{res} = \frac{\sum_i N_i \int_{E_n}^{E_{n+1}} \sigma_i(E) \varphi(E) dE}{\int_{E_n}^{E_{n+1}} \varphi(E) dE} = \frac{\sum_i N_i RI_i}{\Delta u} \quad (15)$$

where RI_i is the resonance integral for i th resonance peaking in the n th fine group, N_i the atom density of the resonance absorber, Δu the lethargy with the fine group ($\Delta u = 0.25$), and $\varphi(E)$ is taken to be $1/E$ in the denominator. The resonance integrals are calculated from resonance parameter utilizing the quantitative methods of Adler-Nordheim.⁽⁶⁾

The absorption cross section averaged over a broad group ℓ consisting of several fine groups is

$$\bar{\Sigma}_{a\ell} = \frac{\sum_{n \text{ in } \ell} \Sigma_{an} \bar{\varphi}_n \Delta u_n}{\sum_{n \text{ in } \ell} \bar{\varphi}_n \Delta u_n}$$

where $\bar{\varphi}_n$ and Δu_n are the average flux and lethargy wide, respectively, of the n th fine group. However, for the broad group averaging the resonance absorption contribution by a fine group, Equation (15) becomes

$$\Sigma_{an}^{res} = \frac{\sum_i N_i RI_i}{\Delta u_n \frac{\bar{\varphi}_n}{\bar{\varphi}_{max}}}$$

where $\bar{\varphi}_n / \bar{\varphi}_{max}$ accounts for any flux depression due to resonance. The unperturbed flux, $\bar{\varphi}_{max}$, is taken to be the largest terms of the sequence

$$\begin{aligned} \bar{\varphi}_{n-2}, \bar{\varphi}_{n-1}, \bar{\varphi}_n & \quad \text{for } E < 101 \text{ eV} \\ \bar{\varphi}_{n-1}, \bar{\varphi}_n & \quad \text{for } E \geq 101 \text{ eV.} \end{aligned}$$

The resonance integral calculations require a value for the mean chord length of the resonance absorber. The mean chord lengths for U^{238} and Th^{232} are computed within GAMTEC II as

$$l = \frac{4V}{S_{eff}} \quad (16)$$

where V is the volume and S_{eff} the effective surface of the resonance absorber. For an internally moderated fuel assembly such as a tube-in-tube element, the effective surface is expressed as ⁽¹⁶⁾

$$S_{eff} = S_m G_m + 4 \frac{\xi_2}{\xi_m} \Sigma_2 V_2 (1 - P'_{22}) + 4 \frac{\xi_1}{\xi_m} \Sigma_1 V_1 (1 - P'_{11}) \quad (17)$$

where the subscripts 1, 2, and m denote the innermost coolant channel, coolant located between concentric fuel tubes, and the exterior moderator, respectively. The other terms in Equation (17) are defined as

S_m = exterior surface of outer fuel tube

ξ_i = average lethargy change per collision in the i th moderator

P'_{ii} = first flight collision probability in the i th region for a neutron uniformly generated in the i th region and not previously crossing an interior region

$G_m = 1 - C$ where C is the usual Dancoff correction for resonance shielding in closely packed lattices. ⁽¹⁷⁾

The ξ_m is an input parameter to GAMTEC II; the purpose of this is as follows. For a graphite moderated reactor having water coolant on the exterior surface of the fuel assembly, ξ_m should be that for water. The majority of the resonance energy neutrons arriving at the exterior fuel surface are scattered into resonance within the exterior water coolant. Stated another way, the probability that a neutron scattered into resonance in the graphite getting through the coolant of any practicable thickness without undergoing a collision is very small.

TWO-GROUP CALCULATION

In addition to obtaining the number of broad group averages requested, GAMTEC II provides two-group constants, one broad fast group and one thermal group. Utilizing these constants, the code calculates the critical buckling from the two-group critical equation

$$\frac{(\eta f)_1 (1 - p_1)}{1 + B_c^2 \tau} + \frac{(\eta f)_2 p_1}{(1 + B_c^2 \tau)(1 + B_c^2 L^2)} = 1 \quad (18)$$

where the subscripts 1 and 2 refer to the fast and thermal groups, respectively. The other quantities in Equation (18) are defined in terms of the two-group constants

$$p_1 = \frac{\Sigma_{R1} - \Sigma_{a1}}{\Sigma_{R1}}, \quad (\eta f)_1 = \frac{\nu_1 \Sigma_{f1}}{\Sigma_{a1}}, \quad (\eta f)_2 = \frac{\nu_2 \Sigma_{f2}}{\Sigma_{a2}}, \quad L^2 = \frac{D_2}{\Sigma_{a2}}$$

with

- Σ_{R1} = removal cross section for group 1
- Σ_{ai} = absorption cross section for i th group
- Σ_{fi} = fission cross section for i th group
- D_i = diffusion coefficient for i th group
- τ = age from the GAM-I moments calculation.

PROBLEMS OF GENERAL INTEREST

Tube-in-Tube Fuel Element in a Graphite Lattice

Measurements of lattice parameters were made by Wood, et al.⁽¹⁹⁾ in the Physical Constants Test Reactor (PCTR) for a concentric tube-natural uranium fuel element in a 10 1/2 in. graphite lattice. The parameters measured included k_∞ , f , p , and ϵ . The radii of the cell components are listed in Table I, and the miscellaneous physical constants of the experimental unit cells are given in Table II.

A comparison is made between these experimental results and the corresponding parameters calculated by GAMTEC II. The calculated value of k_∞ was obtained from the two-group method as given by Equation (18) for zero buckling.

$$k_\infty = (\eta f)_1 (1 - p_1) + (\eta f)_2 p_1 \quad (19)$$

The fast fission factor was calculated from δ the ratio of U^{238} fissions to U^{235} fissions.

$$\epsilon = 1 + 0.697 \delta \quad (20)$$

TABLE I
RADIi OF CELL COMPONENTS
FOR CONCENTRIC TUBE-IN-TUBE FUEL ASSEMBLIES

Material	Inside Radius, cm	Outside Radius, cm
Water	0.000	1.237
Can 1	1.237	1.400
Inner Fuel	1.422	2.108
Can 2	2.133	2.205
Water	2.205	2.439
Can 3	2.439	2.528
Outer Fuel	2.540	3.175
Can 4	3.189	3.278
Water	3.278	3.840
Process Tube	3.840	4.003
Graphite	4.035	15.050 (equivalent)

TABLE II
PHYSICAL CONSTANTS
FOR CONCENTRIC TUBE-IN-TUBE FUEL ASSEMBLIES

Graphite Dih ^(a)	+1.07
Aluminum Type	2-S-99.2% Al
Uranium Density	18.81 g/cm ³
Aluminum Density	2.7 g/cm ³
Water Density	1.0 g/cm ³
Graphite Density	1.615 g/cm ³

(a) Hanford test pile purity index. Corresponds to an absorption cross section of 3.68 millibarns.

As defined in Equation (19), p_1 is the probability that a neutron will escape from the fast group. The value p measured by Wood is the probability that a neutron will escape U^{238} resonance capture. This calculated value may be obtained from the four factor formula as:

$$p = \frac{k_{\infty}}{(\eta f)_2 \epsilon} \quad (21)$$

A summary of the calculated and experimental values is given in Table III. The experimental value for $\eta = k_{\infty}/f p \epsilon$.

Solid Rod Water Lattices

A number of buckling measurements have been made at Brookhaven National Laboratory on slightly enriched uranium fueled reactor lattices

TABLE III
CORRELATION RESULTS
FOR NATURAL URANIUM TUBE-IN-TUBE FUEL ASSEMBLIES

Parameter	Calculated Value	Experimental Value
δ	0.0820	0.0964
ϵ	1.0572	$1.056 \pm 0.007^{(a)}$
p	0.8634	0.867 ± 0.006
f	0.8306	0.831 ± 0.005
η	1.3148	1.295 ± 0.014
k_{∞}	0.9968	0.984 ± 0.002
η_{thermos}	1.312	

(a) Since $\epsilon = 1 + 0.6975 \delta$ it seems that
 $\epsilon = 1 + (0.697)(0.0964) = 1.067$

moderated by light water.⁽²⁰⁾ These measurements were made at various water-to-uranium volume ratios and so provide excellent data for comparison with calculated parameters.

GAMTEC II calculations were made for two different uranium fuel enrichments (1.027% and 1.299% by weight U^{235}) and five water-to-uranium volume ratios (1.0, 1.5, 2.0, 3.0, and 4.0). The fuel rod diameter in each case was 0.600 in.

A summary of the calculated and experimental values is given in Table IV. Since the equivalent homogenization of the unit lattice cell does not accurately account for spectral hardening in the fuel, the values for η_2 and f_2 as calculated by the Thermos code (a multiregion, multienergy

TABLE IV
SLIGHTLY ENRICHED URANIUM-WATER LATTICES
(0.600 in. Rod Diameter)

% U ²³⁵	$\frac{H_2O}{U}$	Calculated									Experimental		
		η_2 GAMTEC II	η_2 Thermos	p_1 GAMTEC II	$\eta_1 f_1$ GAMTEC II	δ^{28} GAMTEC II	$I^2(\text{cm}^2)$ GAMTEC II	$\tau(\text{cm}^2)$ GAMTEC II	k_∞	$B^2 \times 10^4$ (cm^{-2})	$B^2 \times 10^4$ (cm^{-2})	δ^{28}	
1.027	1.0	1.3745	1.3526	0.5449	0.6671	0.1775	2.646	30.97	1.0406	12.33	9.90 ± 0.54	0.193 ± 0.002	
	1.5	1.3396	1.3256	0.6457	0.7083	0.1456	2.562	27.89	1.1069	35.55	29.63 ± 0.42	0.146 ± 0.008	
	2.0	1.3028	1.2844	0.7094	0.7359	0.1246	2.606	26.49	1.1251	43.39	36.07 ± 0.39	0.121 ± 0.003	
	3.0	1.2302	1.2032	0.7847	0.7750	0.0993	2.819	25.31	1.1111	39.73	33.15 ± 0.31	0.098 ± 0.003	
	4.0	1.1609	1.1271	0.8280	0.8040	0.0843	3.083	24.85	1.0715	25.78	20.96 ± 0.26	0.073 ± 0.002	
1.299	1.0	1.4734	1.4622	0.5247	0.7236	0.1772	2.371	30.87	1.1111	34.02	32.11 ± 0.54	0.182 ± 0.010	
	1.5	1.4403	1.4249	0.6291	0.7601	0.1452	2.306	27.84	1.1783	59.66	51.87 ± 0.50	0.138 ± 0.007	
	2.0	1.4049	1.3857	0.6951	0.7853	0.1244	2.355	26.46	1.2026	70.61	61.08 ± 0.32	0.114 ± 0.004	
	3.0	1.3338	1.3044	0.7738	0.8206	0.0990	2.565	25.29	1.1949	69.99	60.99 ± 0.26	0.090 ± 0.004	
	4.0	1.2648	1.2279	0.8192	0.8467	0.0841	2.822	24.84	1.1590	57.56	50.38 ± 0.27	0.076 ± 0.002	

transport code) are also given.⁽²⁰⁾ The material bucklings were calculated using the Thermos values of $n_2 f_2$ with the remainder of the parameters from the GAMTEC II calculation.

Plutonium Nitrate Solutions

Multigroup diffusion theory calculations were made for a number of plutonium nitrate solutions in spherical geometries using multigroup constants generated by GAMTEC II. The solutions, ranging in concentration from 33 to 268 g Pu/liter, were critical in spheres of 11.5, 14.0, and 15.2 in. diam.⁽²¹⁾ Except for two experiments in the bare 15.2 in. sphere, each sphere was surrounded by an effectively infinite water reflector. The calculations were carried out with 17 fast groups, $E > 0.683$ eV, and one thermal group consisting of the Sher normalized thermal cross sections for Pu^{239} (GAMTEC II I.D. No. 36). The results of the calculations are summarized in Table V.

TABLE V
PLUTONIUM NITRATE SOLUTIONS IN SPHERICAL GEOMETRIES
($\text{Pu}^{240} = 4.6\%$)

Sphere Diameter, in.	Reflector ^(a)	Pu, g/liter	NO ₃ , g/liter	H ₂ O, g/liter	H ⁺	Critical Volume, liters	Calculated keff
11.5	Water	73.0	86.0	970.9	0.15	12.94	1.0008
11.5	Water	96.0	203.0	918.3	1.67	12.99	1.0022
11.5	Water	119.0	245.0	904.0	2.00	12.93	1.0059
11.5	Water	132.0	281.0	877.7	2.30	12.97	0.9981
11.5	Water	140.0	284.0	880.8	2.20	12.97	1.0013
11.5	Water	268.0	346.0	868.0	1.06	12.95	1.0005
14.0	Water	33.0	162.0	924.9	2.07	23.22	0.9989
14.0	Water	38.4	292.0	861.5	4.06	23.30	1.0043
14.0	Water	39.2	313.0	841.6	4.20	23.10	0.9975
14.0	Water	47.9	465.0	774.4	6.70	23.13	1.0051
15.2	Water	24.4	58.2	976.9	0.52	30.15	0.9995
15.2	None	39.0	63.8	977.8	0.40	30.27	0.9965
15.2	None	172.3	486.0	765.8	4.90	30.20	0.9932

(a) Stainless Steel Vessel Thickness: 11.5 in. sphere—0.049 in.
14.0 in. sphere—0.044 in.
15.2 in. sphere—0.048 in.

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APPENDIX A
LIBRARY TAPE

The cross section library consists of three general sections. The first section contains the thermal cross sections for the P_3 calculation and the fast group cross sections for the n-flight collision probability calculation. The second section contains the cross sections for the TEMPEST (thermal) calculation. The third section contains the cross sections and resonance parameters for the GAM (neutron slowing down) calculation.

The library tape is updated with the GAMTEC II update program. The new tape is placed on Tape Unit B6 and the GAMTEC II library tape used to update the new tape is placed on Tape Unit A5. An additional GAMTEC II tape may be used to furnish cross sections for certain isotopes. This tape (if used) is placed on Tape Unit B8.

The following input is required:

<u>Card Number</u>	<u>Format</u>	<u>Columns</u>	<u>Description</u>
1	8I5	1-5	P_3 and Collision Probability cross section option.
		=0	Duplicate old tape.
		=N	Cross section changes will be made for N isotopes.
		=-N	N isotopes will be added to the library.
		6-10	Thermal TEMPEST cross section option.
		=0	Duplicate old tape.
		=N	Cross section changes will be made for N isotopes.
		=-N	N isotopes will be added to the library.
		11-15	GAM cross section option
		=0	Duplicate old tape.
		=N	Cross section changes will be made for N isotopes.
		=-N	N isotopes will be added to the library.

<u>Card Number</u>	<u>Format</u>	<u>Columns</u>	<u>Description</u>
		16-20	Number of isotopes on old tape (Tape Unit A5).
		21-25	=0 No changes will be made in the fission source spectrum tables.
			=N The Nth source spectrum will be changed.
			=-N N additional source spectrums will be added to the library.
		26-30	Resonance capture table option. Normally zero.
			=0 No change in the table.
			=1 Changes will be made in the table.
		31-35	Zeta Table option. Normally zero.
			=0 No change in the table.
			=1 Changes will be made.
		36-40	Resonance geometry table option. Normally zero.
			=0 No change will be made in the tables.
			=1 Changes will be made in the table for a slab.
			=2 Changes will be made in the table for a cylinder.
			=3 Changes will be made in the table for a sphere.
2	12A6	1-72	Title card for the P_3 and collision probability cross sections.
3	12A6	1-72	Title card for the TEMPEST cross sections.
4	12A6	1-72	Title card for the GAM cross sections.
If the old tape is being duplicated with no changes or additions (card number one contains all zeros except columns 16-20), then this is all the input required.			
Omit cards 5, 6, and 7 if columns 1-5 of card number one are zero or negative.			

<u>Card Number</u>	<u>Format</u>	<u>Columns</u>	<u>Description</u>
5	14I5	1-70	The identification numbers* of the isotopes for which cross section changes are being made in the P_3 are input on this card. These ID numbers must go from low to high.
6	I10, 6E10.3	1-10 11-20 21-30 31-40 41-50 51-60 61-70	Isotope identification number. Atomic weight. Logarithmic energy decrement. Average (>1.35 MeV) fast transport cross section. Absorption cross section (2200 meter). Thermal scattering cross section (2200 meter). Average (>1.35 MeV) fast $(\sigma_r - v_{11} f_{11})^{**}$ cross section.
7	6E12.8	1-12 13-24 25-72	Resonance scattering cross section. v times fission cross section (2200 meters). and the following 21 cards: 42 absorption G factors ⁽²²⁾ 42 fission G factors ⁽²²⁾ 42 corresponding temperatures in °K, low to high.
Repeat cards 6 and 7 for each isotope for which corrections are being made. Omit cards 8 and 9 if columns 1-5 of card number one are zero or positive.			
8	I10, 6E10.3	1-70	This card is required for each isotope that is being added to the library. The same data are required as for card number 6.
9	6E12.8	1-72	Same data as card number 7.
Repeat cards 8 and 9 for each isotope that is being added to the library.			
If columns 6-10 of card number one are zero or negative, omit cards 10, 11, 12, and 13.			

* All descriptive footnotes for Appendix A begin on Page 26.

<u>Card Number</u>	<u>Format</u>	<u>Columns</u>	<u>Description</u>
10	14I5	1-72	The identification numbers*** of the isotopes for which cross section changes are being made in the TEMPEST section are input on this card. These ID numbers must go from low to high.
11	(A6, 3X, I3, 57X, I3)	1-6 10-12 70-72	Element name. Element identification.**** Number of tabulated cross section entries.
12	(5E10.4/5E 10.4)	1-50	Cross section power series**** {R _j }, resonance parameters {n, r, K, E ₀ }, and v.
13	7E10.4	1-72	N tabulated cross sections where N is the quantity in columns 70-72 of card number 11*****. If N is zero, omit this card.
Repeat cards 11, 12, and 13 for the isotopes for which corrections are being made to the TEMPEST section of the library.			
Omit cards 14, 15, 16, and 17 if columns 6-10 of card number one are zero or positive.			
14	(A6, 3X, I3, 57X, I3)	1-72	Same as card number 11 for the isotopes being added to the TEMPEST section of the library.
15	(5E10.4/5E 10.4)		Same as card number 12 for the isotopes being added to the TEMPEST section of the library.
16	7E10.4		Same as card number 13 for the isotopes being added to the TEMPEST section of the library.
Repeat cards 14, 15, and 16 for the isotopes that are being added to the library.			
17	A6	1-6	The word X-LAST.
Omit cards 18 and 19 unless columns 21-25 of card number one are not zero.			

<u>Card Number</u>	<u>Format</u>	<u>Columns</u>	<u>Description</u>
18	12A6	1-72	Description of the source being added or changed.
19	6E12.5	1-72	Source probability spectrum for the 68 groups.
Repeat cards 18 and 19 for each source that is being added to the library.			
Omit cards 20, 21, and 22 unless columns 26-30 of card number one are not zero. These cards are used to make corrections to that resonance capture table, FJ(I, J).			
20	4I5	1-5	IS—smallest value of I for corrections.
		6-10	IL—largest value of I for corrections.
		11-15	JS—smallest value of J for corrections.
		16-20	JL—largest value of J for corrections.
21	6E12.5	1-72	Resonance capture table corrections. ((FJ(I, J), I = IS, IL), J = JS, JL)
Repeat cards 20 and 21 for each set of corrections.			
22			Blank card to indicate there are no more corrections.
Omit cards 23, 24, and 25 unless columns 31-35 of card number one are not zero. These cards are used to make corrections to the Zeta table, CJ(I, J).			
23	4I5	1-5	IS—smallest value of I for corrections.
		6-10	IL—largest value of I for corrections.
		11-15	JS—smallest value of J for corrections.
		16-20	JL—largest value of J for corrections.
24	6E12.5	1-72	Zeta table corrections. ((CJ(I, J), I = IS, IL), J = JS, JL)
Repeat cards 24 and 25 for each set of corrections.			
25			Blank card to indicate there are no more corrections.

<u>Card Number</u>	<u>Format</u>	<u>Columns</u>	<u>Description</u>
Omit cards 26, 27, and 28 unless columns 36-40 of card number one are not zero. These cards are used to make corrections to the resonance geometry table, F(I, J, K).			
26	6I5	1-5	IS—smallest value of I for corrections.
		6-10	IL—largest value of I for corrections.
		11-15	JS—smallest value of J for corrections.
		16-20	JL—largest value of J for corrections.
		21-25	KS—smallest value of K for corrections.
		26-30	KL—largest value of K for corrections.
27	6E12.5	1-72	Resonance geometry table corrections. (((F(I, J, K), I=IS, IT), J=JS, JT), K=KS, KT).
Repeat cards 26 and 27 for each set of corrections.			
28			Blank card to indicate there are no more corrections
Omit cards 29-41 unless columns 11-15 of card number one are not zero.			
29	14I5	1-72	These entries identify the isotopes being corrected or added to the library in the following manner.
			=ID(J) The identification number of the Jth isotope for which corrections are being made or that is being added to the library. The new cross sections will be read from a GAMTEC II tape on unit B8. These identification numbers must be in sequence.
			-ID(J) The negative of the identification number of the Jth isotope for which corrections are being made or that is being added to the library. The new cross sections will be input by cards.

<u>Card Number</u>	<u>Format</u>	<u>Columns</u>	<u>Description</u>
30	14I5	1-72	The identification number of the isotopes ID(J) on tape unit B8. This is set to zero if the entry on card number 29 is negative.
Omit cards 31-41 unless the new cross section corrections or additions are being read in by cards.			
31	4A6	1-24	Description of the nuclide.
32	14I5	1-5	Identification number
		6-10	$LTOT = I_a + I_f + I_v + I_{n,n} + I_{n,2n} + I_{P(0)} + I_{P(1)}$ where: I_a = 68 if tabulated absorption, = 0 otherwise. I_f = 68 if tabulated fission, = 0 otherwise. I_v = 68 if tabulated fission, = 0 otherwise. $I_{n,n}$ = length of (n,n) matrix. $I_{n,2n}$ = length of (n,2n) matrix. $I_{P(0)}$ = length of P(0) matrix. $I_{P(1)}$ = length of P(1) matrix.
		11-15	=0 No absorption tabulation.
			=1 Absorption tabulation.
		16-20	=0 No fission tabulation.
			=1 Fission tabulation.
		21-25	=0 No resonance parameters.
			=1 Resonance parameters.
		26-30	$I_{n,n}$
		31-35	The number of groups in which inelastic processes occur.
		36-40	The number of groups to which downscatter may occur (not counting ingroup scatter).
		41-55	The data regarding the n, 2n array-- similar to columns 26-40 for the n, n array.
		56-70	The data regarding the P(0) array-- similar to columns 26-40 for the n, n array.

<u>Card Number</u>	<u>Format</u>	<u>Columns</u>	<u>Description</u>
33		1-15	The data regarding the P(1) array similar to columns 26-40 of card number 32 for the n,n array. If there is no P(1) array, this is a blank card.
34	3I5	1-5	IST = starting number of the corrections or additions. See card number 35.
		6-10	IEND = last number of the corrections or additions.
35	E11.4, 5E12.4	1-71	ADUM(I), I=IST, IEND where: ADUM(1 to I _a) = absorption cross sections if they are used. High energy to low. ADUM[I _a + 1 to (I _a + I _f)] = fission cross sections if they are used. ADUM[I _a + I _f + 1 to (I _a + I _f + I _v)] = v values if they are used. ADUM[I _a + I _f + I _v + 1 to (I _a + I _f + I _v + I _{n,n})] = inelastic scattering array in the following sequence; (1 to 1), (1 to 2), (1 to 3), ..., (1 to J), (2 to 2), (2 to 3), ..., (2 to J), etc. ADUM[I _a + I _f + I _v + I _{n,n} + 1 to LTOT] = the (n, 2n), P(0), and P(1) arrays stored in the same sequence manner as the inelastic scattering array.
Cards 34 and 35 are repeated for each set of data. If corrections are being made only the ADUM(I) that are being changed need to be input.			
36	I15	10-15	=0 No resonance parameters are being changed. =1 Resonance parameters are being input on cards 37, 38, 39, 40, and 41.

Omit cards 37, 38, 39, 40, and 41 if card number 36 contains only zeros.

<u>Card Number</u>	<u>Format</u>	<u>Columns</u>	<u>Description</u>
37	I10, 6E10.5	1-10	Number of resolved resonances.
		11-20	Atomic mass.
		21-30	Upper energy boundary (eV) for resolved resonances.
		31-40	Upper energy boundary (eV) for unresolved resonances.
		41-50	Unresolved level spacing D in eV.
		51-60	Average unresolved $\langle \Gamma_n^o \rangle$.
			$\langle \Gamma_n^o \rangle = \langle \Gamma_n \rangle / \sqrt{E}$.
38	2E10.5	61-70	Average unresolved Γ_γ .
		1-10	Average unresolved Γ_f .
		11-20	Unresolved statistical weight factor.
39	2I5	1-5	IA = first resonance number being corrected. This will be unity when the isotope is being added to the library. Number the resonances from low energies to high.
		6-10	IB = last resonance being corrected in this set. This will be the number of resolved resonances when the isotope is being added to the library.
40	(5E10.4, I5)	1-10	Resonance energy E_o .
		11-20	Γ_n
		21-30	Γ_r
		31-40	Γ_f
		41-50	G
		51-55	=1 Use IM approximation.
			=2 Use NR approximation.

Repeat card number 40 IB - IA + 1 times where IB and IA are input on card number 39. Then repeat cards 39 and 40 for each set of resonances for the isotope. Only the resonances that are being changed need to be input.

<u>Card Number</u>	<u>Format</u>	<u>Columns</u>	<u>Description</u>
41			Blank card to indicate there are no more resonance parameter corrections.
Repeat cards 31-41 for each nuclide for which the new cross section corrections read in by cards.			

- * The identification numbers on the cross section tape start with one and increase consecutively to the total number of isotopes. The following isotopes must have the corresponding identification numbers.

H ¹	1	U ²³⁸	12
D ₂ O	2	Pu ²⁴⁰	15
Th ²³²	6	H ₂ O	21
		O ¹⁶	23

- ** σ_{rl} = removal cross section
 σ_{fl} = fission cross section
fl = fraction of fissions above 1.35 MeV.

- *** Tempest actually has four ID numbers per isotope.
ID numbers 1-100 are absorption ID's.
ID numbers 101-200 are fission ID's.
ID numbers 201-300 are $(1-\mu)\sigma_s$ ID's.
ID numbers 301-400 are $\xi\sigma_s$ ID's.

Example:

The ID numbers for U²³⁸ would be:
12 Thermal absorption cross sections.
112 Thermal fission cross sections.
212 Thermal $(1-\mu)\sigma_s$ cross sections.
313 Thermal $\xi\sigma_s$ cross sections.

See Reference 8 for additional details.

- **** If the cross sections are not tabulated, they are determined from the power series:

$$\sqrt{E}\sigma(E) = \sum_{j=0}^4 R_j (\sqrt{E})^j$$

The entries n, γ , K, and E are required (they may be zero) for the resonance term.

$$\frac{K \gamma}{(\gamma + n \sqrt{E})^2 + 4(E - E_0)}$$

ν is used in the following manner:

ID Number	Value
1-99	Not used
101-199	ν_{thermal}
201-299	$(1 - \mu)$
301-399	ξ

***** The $\sqrt{E}\sigma(E)$ is tabulated at the following mesh intervals in eV:
 0.0, 0.001, 0.002, 0.003, ..., 0.050, 0.060, 0.070, ...,
 E_{max} .

The total tabulated TEMPEST library must be less than 5000 words in length.

APPENDIX B
GAMTEC II INPUT INSTRUCTIONS

The input data is stored in absolute machine locations by a generalized Fortran input routine.⁽¹⁸⁾ Only the data which are nonzero initially or changes from case to case need to be punched on cards. This minimizes input volume while using only a few simple standard formats. The formats are interpreted as follows:

<u>Column 1</u>	<u>Maximum Number Data Words per Card</u>	<u>Type of Data</u>	<u>Columns per Word</u>	<u>Words Start in Columns</u>
0	6	Floating Point	10	11, 21, 31...61
1	12	Integer	5	11, 16, 21...66
2	10	Hollerith	6	11, 17, 23...65
9	--	End of Case		

Columns 2, 3

These columns optionally contain the number of words to be read from the data card. If blank, the whole card will be read.

Column 4

A nonzero entry in column four causes the first data word on the card to be stored in N successive locations where N is the entry in columns 2 and 3.

Columns 5-10

These columns contain the (decimal) location of the first data word on the card. Succeeding words go to consecutive, higher locations. A blank causes the continuation of sequencing from the last card.

Columns 11-70

These columns contain the input data in the format described in column 1.

INPUT

<u>Column 1</u> <u>Format Type</u>	<u>Column 6-10</u> <u>Location</u>	<u>Description</u>
1	1	Case identification number.
1	2	Number of cylindrical regions.
1	3	Normally zero. Nuclei densities will be in atoms/barn-cm.
		=2 A special routine for spherical solution experiments at Critical Mass Laboratory. If 2 is entered, input is required in locations 846-850 and may be omitted from locations 63-232 and 676-845.
1	4	=1 Average thermal cross sections over the flux from the Wilkins heavy moderator equation.
		=2 Average thermal cross sections over the flux from the Wigner-Wilkins light moderator equation.
		=3 Average thermal cross sections over a Maxwellian distribution.
1	5	Indicates the mode of hydrogen bonding (i.e., water, free gas, etc.). This must be an element in the 200 series.
1	8	=0 Use one over E flux spectrum built into code for neutron slowing down.
		=1 Use P_1 approximation for neutron slowing down.
		=2 Use B_1 approximation for neutron slowing down.
		=3 Read in fluxes and currents (locations 884 and 953).
1	9	Fission source:
		=0 Unit source in one group (location 10).
		=1 U^{233} fission spectrum.
		=2 U^{235} fission spectrum.
		=3 Pu^{239} fission spectrum.
		=4 Pu^{241} fission spectrum.
		=5 Cf^{252} fission spectrum.

<u>Column 1</u> <u>Format Type</u>	<u>Column 6-10</u> <u>Location</u>	<u>Description</u>	
		=6	P ₀ -Be (Whitemore-Baker).
		=7	P ₀ -Be (Cochrane-Henry).
		=8	Ra-Be (Hill).
1	10		If location 9 is zero, then put group number* for unit source here.
1	11		Number (≤ 33) of epithermal** broad energy groups for which cross section averages are desired.
1	12	=0	Do not calculate the age.
		=1	Calculate the age.
1	13	=0	Do not calculate microscopic epithermal cross sections.
		=1	Calculate microscopic epithermal cross sections.
1	14		Geometry type for resonance calculation.
		=1	Slab
		=2	Cylinder
		=3	Sphere
1	15	=0	One set of cross sections for the HFN diffusion code will be punched on cards.
		=n	n + 1 sets of cross sections for HFN will be punched on cards. $n \leq 5$
		=6+n	n sets of cross sections will be punched for HFN and one set for the DTF transport code.
		=-1	No cross sections will be punched.
1	16		Ignore if location 15 is negative. Otherwise, this is the desired storage location for the first HFN cross section (13881 for region 1 and group 1).

* See Appendix C, List A.

** Epithermal includes the entire energy range above thermal.

<u>Column 1</u> <u>Format Type</u>	<u>Column 6-10</u> <u>Location</u>	<u>Description</u>
1	17	Ignore if location 15 is negative. Otherwise, this is the type of geometry for the perpendicular buckling*** for the punched HFN cards.
		=1 Slab
		=2 Cylinder
		=3 Constant $B_{\perp}^2$
1	18-27	Ignore if location 15 is equal to or less than zero. Otherwise, repeat the instructions for locations 16 and 17 for each set of HFN cross sections.
1	28	Ignore unless location 8 is equal to one.
		=0 Leakage terms for the P_1 calculation will be by broad energy groups (location 1022). If the leakage terms are carried down from a previous case, this entry must be set to one.
		=1 Leakage terms for the P_1 calculation will be by fine groups (location 1022).

$$B_{\perp}^2 = \frac{C_1^2}{(A + 2\lambda)^2} + \frac{C_2^2}{(B + 2\lambda)}$$

where: $\lambda = C_3 \lambda_{tr}$

C_3 = constant input in location 609.

Constant $B_{\perp}^2$:

$$C_2 = 0$$

$B_{\perp}^2$ = constant input in location 607; for spherical geometry $B_{\perp}^2 = 0$.

Cylinder:

$$C_2 = 0$$

A = length of cylinder (cm) input in location 607.

Slab:

A = length of one dimension of slab (cm) input in location 607.

B = length of second dimension of slab (cm) input in location 608.

<u>Column 1</u> <u>Format Type</u>	<u>Column 6-10</u> <u>Location</u>		<u>Description</u>																																								
1	29		Number of isotopes in inner region. H ₂ O and D ₂ O are considered to be single isotopes.																																								
1	30-45		Number of isotopes in successive regions.																																								
1	46		Number of flux points to be printed out across the inner region from the P ₃ calculation. The calculation does not depend upon number of points chosen.																																								
1	47-62		Number of flux points for successive regions.																																								
1	63-72		Identification numbers of nuclides in region one (See Table VI, Page 36).																																								
1	73-232		Identification numbers of nuclides in successive regions.																																								
			<table> <tr> <th><u>Region</u></th><th><u>Location</u></th><th><u>Region</u></th><th><u>Location</u></th></tr> <tr><td>1</td><td>63-72</td><td>9</td><td>143-152</td></tr> <tr><td>2</td><td>73-82</td><td>10</td><td>153-162</td></tr> <tr><td>3</td><td>83-92</td><td>11</td><td>163-172</td></tr> <tr><td>4</td><td>93-102</td><td>12</td><td>173-182</td></tr> <tr><td>5</td><td>103-112</td><td>13</td><td>183-192</td></tr> <tr><td>6</td><td>113-122</td><td>14</td><td>193-202</td></tr> <tr><td>7</td><td>123-132</td><td>15</td><td>203-212</td></tr> <tr><td>8</td><td>133-142</td><td>16</td><td>213-222</td></tr> <tr><td></td><td></td><td>17</td><td>223-232</td></tr> </table>	<u>Region</u>	<u>Location</u>	<u>Region</u>	<u>Location</u>	1	63-72	9	143-152	2	73-82	10	153-162	3	83-92	11	163-172	4	93-102	12	173-182	5	103-112	13	183-192	6	113-122	14	193-202	7	123-132	15	203-212	8	133-142	16	213-222			17	223-232
<u>Region</u>	<u>Location</u>	<u>Region</u>	<u>Location</u>																																								
1	63-72	9	143-152																																								
2	73-82	10	153-162																																								
3	83-92	11	163-172																																								
4	93-102	12	173-182																																								
5	103-112	13	183-192																																								
6	113-122	14	193-202																																								
7	123-132	15	203-212																																								
8	133-142	16	213-222																																								
		17	223-232																																								
1	503-522		Identification numbers of the isotopes for which thermal microscopic cross sections are desired that are not present in the lattice cell.																																								
1	525	=n	n neutron flights will be traced in the fast effect collision probability calculation. If zero, the flights will be traced until the contribution from the last flight is less than 0.001.																																								
1	527	=0	All isotopes other than U ²³⁸ or Th ²³² will have the same weighing factors for the fast effect.																																								
		=1	All fast fissile isotopes admixed with U ²³⁸ will have the same weight as U ²³⁸ .																																								
		=2	All fast fissile isotopes admixed with Th ²³² will have the same weight as Th ²³² .																																								

<u>Column 1 Format Type</u>	<u>Column 6-10 Location</u>	<u>Description</u>
0	601	Geometric buckling of the system.
0	602	Dancoff shielding factor.
0	603	Logarithmic energy decrement for outermost moderator.
0	604	$S_{\text{eff}} = \frac{2 \times \text{volume}}{R_{\text{eff}}}$ where R_{eff} is the radius for the outer boundary used in computing the fast neutron interaction effect. R_{eff} is normally the radius of the equivalent unit cell.
0	605	Minimum energy for the thermal tempest calculation.
0	606	Maximum energy for the thermal tempest calculation.
0	607	Ignore if location 15 is negative. Otherwise, this quantity is used for the perpendicular buckling calculation for HFN.***
0	608	Ignore if location 15 is negative. Otherwise, this quantity is used for the perpendicular buckling calculation for HFN.***
0	609	Ignore if location 15 is negative. Otherwise, this is the extrapolation constant (normally 0.71).***
0	610-624	Ignore if location 15 is equal to or less than zero. Otherwise, repeat the instructions for locations 607, 608, and 609 n times.
0	625	Radius of inner region in centimeters.
0	626-641	Radii of successive regions.
0	642	Physical temperature of inner region.
0	643-658	Physical temperatures of successive regions.
0	659	Neutron temperature of inner region.
0	660-675	Neutron temperatures of successive regions.
0	676-685	Densities of the nuclides in atoms per barn centimeter for region one. These densities must be input in the same order as the identification numbers were. That is, the density in location 676 must be for the isotope whose identification number is stored in location 63.

<u>Column 1</u> <u>Format Type</u>	<u>Column 6-10</u> <u>Location</u>	<u>Description</u>																																								
0	686-845	Densities of nuclides in successive regions. <table><tr><th><u>Region</u></th><th><u>Location</u></th><th><u>Region</u></th><th><u>Location</u></th></tr><tr><td>1</td><td>676-685</td><td>9</td><td>756-765</td></tr><tr><td>2</td><td>686-695</td><td>10</td><td>766-775</td></tr><tr><td>3</td><td>696-705</td><td>11</td><td>776-785</td></tr><tr><td>4</td><td>706-715</td><td>12</td><td>786-795</td></tr><tr><td>5</td><td>716-725</td><td>13</td><td>796-805</td></tr><tr><td>6</td><td>726-735</td><td>14</td><td>806-815</td></tr><tr><td>7</td><td>736-745</td><td>15</td><td>816-825</td></tr><tr><td>8</td><td>746-755</td><td>16</td><td>826-835</td></tr><tr><td></td><td></td><td>17</td><td>836-845</td></tr></table>	<u>Region</u>	<u>Location</u>	<u>Region</u>	<u>Location</u>	1	676-685	9	756-765	2	686-695	10	766-775	3	696-705	11	776-785	4	706-715	12	786-795	5	716-725	13	796-805	6	726-735	14	806-815	7	736-745	15	816-825	8	746-755	16	826-835			17	836-845
<u>Region</u>	<u>Location</u>	<u>Region</u>	<u>Location</u>																																							
1	676-685	9	756-765																																							
2	686-695	10	766-775																																							
3	696-705	11	776-785																																							
4	706-715	12	786-795																																							
5	716-725	13	796-805																																							
6	726-735	14	806-815																																							
7	736-745	15	816-825																																							
8	746-755	16	826-835																																							
		17	836-845																																							
0	846	Ignore if location 3 is not equal to 2. Equal to the density of plutonium (g/liter).																																								
0	847	Ignore if location 3 is not equal to 2. Equal to the NO ₃ density (g/liter).																																								
0	848	Ignore if location 3 is not equal to 2. Equal to the water density (g/liter).																																								
0	849	Ignore if location 3 is not equal to 2. Equal to the ionized hydrogen density (g/liter).																																								
0	850	Ignore if location 3 is not equal to 2. Equal to the weight percent of the plutonium that is Pu ²⁴⁰ .																																								
0	852	Lower energy boundary of highest energy group for cross section averaging.																																								
0	853-883	Successive energy boundaries of the n groups where n is the parameter in location 11.																																								
0	884-952	Ignore if location 8 is not equal to 3. Otherwise, enter the fluxes for each of the 68 fine energy groups (from high energy to low).																																								
0	953-1021	Ignore if location 8 is not equal to 3. Otherwise, enter the currents for each fine energy group.																																								
0	1022-1090	Ignore if location 8 is not equal to 1. Enter N leakage terms where N depends upon the quantity entered in location 28.																																								

<u>Column 1</u> <u>Format Type</u>	<u>Column 6-10</u> <u>Location</u>	<u>Description</u>
0	1761	Normally zero. Mean cord length in the absorber for isotope whose ID number is 1.
0	1762-1811	Normally zero. Mean cord length in the absorber for isotope whose ID number is location 1760. No input required for U ²³⁸ or Th ²³² .
2	1091-1106	Title of the case.
9		End of case.

All input quantities for this case will be carried over for the next case so only those quantities which change will need to be input in successive cases. One magnetic cross section tape is required as described in Appendix A. This tape is loaded on tape unit B9. Input data is read in from tape unit A2 and output is to A3.

TABLE VI

ID NUMBERS

ON THE MASTER GAMTEC II CROSS SECTION TAPE

<u>Nuclide</u>	<u>Number</u>	<u>Nuclide</u>	<u>Number</u>
Aluminum	25	Manganese	28
Beryllium	3	Nickel	31
Boron	4	Nitrogen	22
Cadmium	18	Oxygen	23
Carbon	5	Pu ²³⁹	36
Chromium	27	Pu ²⁴⁰	15
Cobalt	30	Pu ²⁴¹	16
Copper	32	Pu ²⁴²	17
Deuterium	2	Silicon	26
Eu ¹⁵¹	33	Th ²³²	6
Gold	19	U ²³³	8
Hydrogen	1	U ²³⁴	9
Indium	13	U ²³⁵	10
Iron	29	U ²³⁶	11
Lead	20	U ²³⁸	12
Lu ¹⁷⁵	34	Water	21
Lu ¹⁷⁶	35	Zirconium	7
Magnesium	24		

APPENDIX C

SUMMARY OF THE OUTPUT AVAILABLE AND THE AVERAGING PROCESSES USED

The input data provides a number of options which control the printed output. The majority of these options take very little additional computer time. However, the option for epithermal microscopic cross sections generally requires considerable time (up to 30% of the total time for a case) so this option should not be used if microscopic cross sections are not needed.

The output may be summarized in the following general categories:

A. Input data

1. The information on the input cards
2. A summary of the basic parameters used in each region

B. P₃ calculation

1. The average flux in each region i

$$\bar{\phi}_i = \frac{\int_{r_{i-1}}^r \phi 2\pi r dr}{\int_{r_{i-1}}^{r_i} 2\pi r dr} = \frac{\int_{r_{i-1}}^{r_i} \phi 2\pi r dr}{V_i}$$

where:

r_i = radius of region i

2. Thermal utilization of each region i

$$f_i = \frac{\bar{\phi}_i \Sigma_a^i V_i}{\sum_{j=1}^{\text{Number Regions}} \bar{\phi}_j \Sigma_a^j V_j}$$

3. Flux traverse across the cell

4. Homogenized isotope cell densities for the thermal calculation

$$N_H^i = \frac{\sum_{j=1}^{\text{Number Regions}} N^{ij} \bar{\varphi}_j V_j}{\sum_{j=1}^{\text{Number Regions}} \bar{\varphi}_j V_j}$$

where:

N^{ij} = density of isotope i in region j .

C. Homogenized thermal calculation using the light or heavy moderator equations or a Maxwellian distribution to obtain the thermal flux as a function of energy

1. Microscopic cross section averages

$$\bar{\sigma}_k = \frac{1}{\bar{\varphi}} \int_{E_{\min}}^{E_{\max}} \varphi(E) \sigma_k(E) dE$$

where:

k = element identification number

$\sigma_i(E) = \sigma_a(E)$ for isotope whose ID is i

$\sigma_{i+100}(E) = \sigma_f(E)$ for isotope whose ID is i

$\sigma_{i+200}(E) = (1 - \mu) \sigma_s(E)$ for isotope whose ID is i

$\sigma_{i+300}(E) = \xi \sigma_s(E)$ for isotope whose ID is i

$E_{\max}$ = maximum thermal energy for cross section averages

$E_{\min}$ = minimum thermal energy for cross section averages

$$\bar{\varphi} = \int_{E_{\min}}^{E_{\max}} \varphi(E) dE$$

2. Macroscopic cross section averages (Transport, Diffusion Constant, XI Sigma S, Absorption, ν Fission)

$$\overline{\Sigma}_{tr} = \frac{1}{\phi} \int_{E_{min}}^{E_{max}} \phi(E) \Sigma_{tr}(E) dE$$

$$\overline{D} = \frac{1}{\phi} \int_{E_{min}}^{E_{max}} \phi(E) D(E) dE$$

$$\overline{\xi \Sigma}_S = \frac{1}{\phi} \int_{E_{min}}^{E_{max}} \phi(E) \left[\sum_i N_H^i \sigma_{Si}(E) \xi_i \right] dE$$

$$\overline{\Sigma}_a = \frac{1}{\phi} \int_{E_{min}}^{E_{max}} \phi(E) \left[\sum_i N_H^i \sigma_{ai}(E) \right] dE$$

$$\nu \Sigma_f = \frac{1}{\phi} \int_{E_{min}}^{E_{max}} \phi(E) \left[\sum_i N_H^i \nu_i \sigma_{fi}(E) \right] dE$$

where:

$$\Sigma_{tr}(E) = \sum_{\substack{\text{Isotopes} \\ \text{in cell}}} N_i [\sigma_{ai}(E) + (1 - \mu)_i \sigma_{Si}(E)]$$

$$D(E) = \frac{1}{3\overline{\Sigma}_{tr}}(E)$$

The $(1 - \mu)_i \sigma_{Si}(E)$ for the transport cross section is taken from the 200 series TEMPEST cross section library. The effect on the transport cross section due to chemical binding of hydrogen may be included by designating in the input which element in the 200 series is tabulated to include this effect.

3. Average thermal neutron velocity

$$\bar{V}_{th} = \frac{\int_{E_{min}}^{E_{max}} \varphi(E) dE}{\int_{E_{min}}^{E_{max}} N(E) dE}$$

4. A table of E, N(E), $\varphi(E)$, and $E\varphi$

D. Collision probability calculation

1. First collision probabilities, $P(I, J)$

where:

$P(I, J)$ = the probability that a neutron born uniformly in region I will suffer its first collision in region J.

Number
Regions
1 - $\sum_{J=1}^{Number\ Regions} P(I, J)$ = the probability that a neutron born uniformly in region I will reach the cell boundary before suffering its first collision.

2. The number of collisions with U^{238} or Th^{232} per original fission neutron for the first n- flights as given by Equation (8).
3. The mean cord length and effective surface of the fuel as given by Equations (16) and (17).
4. The weighing factors above 1.35 MeV as given by Equations (6) and (7).

E. Information on the parameters used in the epithermal calculation

1. The desired energies (which were input) with their corresponding lethargies and the actual energies and lethargies used for broad group cross section averaging are printed out. Since the energy range from 0.414 eV to 10 MeV is divided into 68 fine groups of one quarter lethargy each, the broad group averages are over whole fine groups. See Table VII (Page 46) for the fine group listing.

2. The nuclei densities N^i for the epithermal calculation

$$N^i = \frac{\sum_{j=1}^{\text{Number Regions}} N^{ij} V_j}{\sum_{j=1}^{\text{Number Regions}} V_j}$$

3. σ_m , σ_p , $\sigma_{m(\text{eff})}$, ℓ , and T are printed out for each isotope treated by resonance theory. (7)

$$\sigma_m = \frac{\sum_{j=1}^{\text{Number Regions}} \sum_{i=1}^{\text{Moderating Isotopes}} N^{ij} \sigma_s^{ij} V_j \delta_j}{N_a \sum_{j=1}^{\text{Number Regions}} V_j \delta_j}$$

$$\sigma_p = \frac{\sum_{j=1}^{\text{Number Regions}} \sum_{i=1}^{\text{Number Isotopes}} N^{ij} \sigma_s^{ij} V_j \delta_j}{N_a \sum_{j=1}^{\text{Number Regions}} V_j \delta_j}$$

$$\sigma_{m(\text{eff})} = \sigma_p + \frac{1}{\ell N_a}$$

ℓ is given by Equation (16) if the isotope is U^{238} or Th^{232} . Otherwise it is zero unless changed in the input.

$$T = \frac{\sum_{j=1}^{\text{Number Regions}} N^{i aj} T_j V_j}{\sum_{j=1}^{\text{Number Regions}} N^{i aj} V_j}$$

where:

δ_j = zero if the resonance isotope is not present in the region.

δ_j = one if the resonance isotope is present in the region.

N^{ij} = density of isotope i in region j

$N^{i aj}$ = density of the resonance isotope in region j

σ_s^{ij} = nonresonance scattering cross section

T_j = physical temperature in region j . The isotope is considered to be a moderating isotope if its mass is

Capture cross section

$$\Sigma_{c\ell} = \Sigma_{a\ell} - \Sigma_{f\ell}$$

Fission cross section

$$\Sigma_{f\ell} = \sum_{n \text{ in } \ell} \sum_{\text{Number Isotopes}} N^i \sigma_{fn}^i \varphi_n \Delta_n / \alpha_\ell$$

Total transfer cross section from broad group ℓ to broad group k

$$\Sigma_{\ell k} = \sum_{n \text{ in } \ell} \sum_{i \text{ in } k} \sum_{\text{Number Isotopes}} N^i \left(\sigma_{nj}^{i,o} + \sigma_{nj}^{i,n} + 2\sigma_{nj}^{i,2n} \right) \varphi_n \Delta_n / \alpha_\ell$$

G. Epithermal cross section averages over broad energy groups

1. The broad group constants D , Σ_{tr} , Σ_a , Σ_c , Σ_f , $\nu\Sigma_f$, ν , $P(0)$, $P(1)$, $P_{n,n'}$, $P_{n,2n}$, Σ_R , and Σ_o are printed out for one fast group.
2. The broad group constants D , Σ_{tr} , f , Σ_R , Σ_o , Σ_a , Σ_c , Σ_f , $\nu\Sigma_f$, ν , and the transfer matrix are printed out for each broad group designated in section E. The averaging process used to obtain these constants is:

Diffusion coefficient for broad group ℓ

$$D_\ell = \frac{\beta_\ell}{\gamma_\ell}$$

Transport cross section

$$\Sigma_{tr \ell} = \frac{1}{3D_\ell}$$

Fission source spectrum

$$f_\ell = \sum_{n \text{ in } \ell} f_n \Delta n$$

Removal cross section

$$\Sigma_{R\ell} = \Sigma_{a\ell} \sum_{n \text{ in } \ell} \sum_{j > \ell} \sum_{i=1}^{\text{Number Isotopes}} N^i \left(\sigma_{nj}^{i,o} + \sigma_{nj}^{i,n} + \sigma_{nj}^{i,2n} \right) \varphi_n \Delta n / \alpha_\ell$$

Scattering out cross section

$$\Sigma_{o\ell} = \Sigma_{R\ell} - \Sigma_{a\ell}$$

Absorption cross section

$$\Sigma_{a\ell} = \sum_{n \text{ in } \ell} \sum_{i=1}^{\text{Number Isotopes}} N^i \sigma_{an}^i \varphi_n \Delta n / \alpha_\ell$$

Capture cross section

$$\Sigma_{c\ell} = \Sigma_{a\ell} - \Sigma_{f\ell}$$

Fission cross section

$$\Sigma_{f\ell} = \sum_{n \text{ in } \ell} \sum_{\substack{\text{Number} \\ \text{Isotopes}}} N^i \sigma_{fn}^i \varphi_n \Delta_n / \alpha_\ell$$

Total transfer cross section from broad group ℓ to broad group k

$$\Sigma_{\ell k} = \sum_{n \text{ in } \ell} \sum_{j \text{ in } k} \sum_{\substack{\text{Number} \\ \text{Isotopes}}} N^i \left(\sigma_{nj}^{i,o} + \sigma_{nj}^{i,n} + 2\sigma_{nj}^{i,2n} \right) \varphi_n \Delta_n / \alpha_\ell$$

where:

φ_n = lethargy flux in fine group n .

Δ_n = lethargy width of fine group n ; = 0.25.

J_n = current for fine group n .

$\sigma_{nj}^{i,o}$, $\sigma_{nj}^{i,n}$, $\sigma_{nj}^{i,2n}$ are the $P(o)$, (n,n) , and $(n,2n)$ cross sections for transfer from fine group n to fine group j for isotope i .

$$\alpha_\ell = \sum_{n \text{ in } \ell} \varphi_n \Delta_n$$

$$\beta_\ell = \sum_{n \text{ in } \ell} J_n \Delta_n$$

$$\gamma_\ell = \sum_{n \text{ in } \ell} \mathfrak{L}_n \varphi_n \Delta_n$$

$\mathfrak{L}_n$ = leakage factor for group n in the P_1 approximation.

$\mathfrak{L}_n$ = the negative of the square root of the buckling for the B_1 approximation.

3. The broad group microscopic constants (if needed) are averaged and printed out in the same manner as the broad group macroscopic constants.
4. Average neutron velocity for each broad group

$$\bar{v}_\ell = \frac{\sum_{n \text{ in } \ell} \varphi_n \Delta_n}{\sum_{n \text{ in } \ell} \frac{\varphi_n \Delta_n}{\bar{v}_n}}$$

where $\bar{v}_n$ is obtained above 50 keV for each fine group by assuming a flat flux and below 50 keV by assuming a $1/E$ flux.

H. Two group constants and buckling calculation

1. The quantities $(nf)_1$, P_1 , τ , $(nf)_2$, L^2 , k_∞ , and B^2 as defined in Equation (18) are printed out. The total resonance absorption and fission integrals and δ (ratio of fast to thermal fissions) are also given:

$$(RI)_{\text{abs}} = \sum_i (RI)_{\text{abs}}^i$$

$$(RI)_{\text{fis}} = \sum_i (RI)_{\text{fis}}^i$$

$$\delta = \nu_s \frac{\sigma_{\text{fl}}}{\sigma_{\text{tl}}} C_1$$

where:

The summation is over all resonances.

C_1 is the number of collisions with U^{238} per original fission neutron (Equation 8).

$\sigma_{\text{fl}}/\sigma_{\text{tl}}$ = ratio of fast fission cross section to fast transport cross section.

ν_s = average number of neutrons produced per thermal fission.

- I. Punched card output for HFN diffusion code⁽¹⁰⁾ and DTF transport code⁽¹¹⁾

TABLE VII
EPITHERMAL GROUP STRUCTURE

<u>Group</u>	<u>U(lower)</u>	<u>E(lower)</u>	<u>Group</u>	<u>U(lower)</u>	<u>E(lower)</u>
1	0.25	7.79×10^6	35	8.75	1.59×10^3
2	0.50	6.07×10^6	36	9.00	1.23×10^3
3	0.75	4.72×10^6	37	9.25	961
4	1.00	3.68×10^6	38	9.50	748
5	1.25	2.87×10^6	39	9.75	583
6	1.50	2.23×10^6	40	10.00	454
7	1.75	1.74×10^6	41	10.25	354
8	2.00	1.35×10^6	42	10.50	275
9	2.25	1.05×10^6	43	10.75	215
10	2.50	8.21×10^5	44	11.00	167
11	2.75	6.39×10^5	45	11.25	130
12	3.00	4.98×10^5	46	11.50	101
13	3.25	3.88×10^5	47	11.75	78.9
14	3.50	3.02×10^5	48	12.00	61.4
15	3.75	2.35×10^5	49	12.25	47.9
16	4.00	1.83×10^5	50	12.50	37.3
17	4.25	1.43×10^5	51	12.75	29.0
18	4.50	1.11×10^5	52	13.00	22.6
19	4.75	8.65×10^4	53	13.25	17.6
20	5.00	6.74×10^4	54	13.50	13.7
21	5.25	5.25×10^4	55	13.75	10.68
22	5.50	4.09×10^4	56	14.00	8.32
23	5.75	3.18×10^4	57	14.25	6.48
24	6.00	2.48×10^4	58	14.50	5.04
25	6.25	1.93×10^4	59	14.75	3.93
26	6.50	1.50×10^4	60	15.00	3.06
27	6.75	1.17×10^4	61	15.25	2.38
28	7.00	9.12×10^3	62	15.50	1.86
29	7.25	7.10×10^3	63	15.75	1.44
30	7.50	5.53×10^3	64	16.00	1.125
31	7.75	4.31×10^3	65	16.25	0.876
32	8.00	3.36×10^3	66	16.50	0.683
33	8.25	2.61×10^3	67	16.75	0.532
34	8.50	2.04×10^3	68	17.00	0.414

APPENDIX D
SAMPLE PROBLEM

The 1.299% U^{235} enriched uranium-water lattice having a water-to-uranium volume ratio of 1 is used as a sample problem. The dimensions of the unit cell and the isotope densities are listed in Table VIII. Note that the results for the sample problem differ from those given in Table IV (Page 13) for the same problem. The two problems differ in the S_{eff} , the circumference of the unit lattice cell, used. This was done purposely to illustrate the control one has on the fast neutron interaction effect through the use of Location 604 without affecting the calculations for other parameters.

TABLE VIII
CONSTANTS FOR THE SAMPLE PROBLEM

Cell dimensions (cm)

Radius of uranium rod	0.7620
Outside radius of aluminum	0.8458
Effective cell radius	1.1384
Lattice pitch	2.1684

Isotope densities (atoms/barn-cm)

U^{235} density	6.288×10^{-4}
U^{238} density	4.719×10^{-2}
Effective aluminum density	5.157×10^{-2}
Water density	3.344×10^{-2}

A computer printout of this sample problem follows.

```

      *   *   *   *   *
      *   PRINTOUT OF   *
      *   CARD INPUT DATA   *
      *   *   *   *   *
1      1      1      3      2      221      2      2      3      1
1      13      0      2      713881      3
1 3      29      2      1      1
1 3      46      5      5      5
1 2      63      10      12
1 1      73      25
1 1      83      21
0      60132.11      -04      .165      .948      7.5115      0.0      0.683
0      6090.71
0      625.762      .84582      1.1384
0 31      642300.0
0 31      659300.0
0 2      676 .6288 -03 .4719 -01
0 1      686 .5157 -01
0 1      696 .3344 -01
0      852 1.35 +06 454.0      .683
2      1091      1.3 ( ENRICHED URANIUM-WATER LATTICE      W/U = 1.0
9      FUEL ROD DIAMETER = .600 INCHES

```

INPUT DATA

CASE 1	NO. REGIONS 3	BUCKLING 0.32110E-02	SHIELDING FACTOR 0.16500	SQUIGGLY 0.94800E 00	SURFACE AREA OF U.L.C. 0.75115E 01
-----------	------------------	-------------------------	-----------------------------	-------------------------	---------------------------------------

REGION NUMBER 1

NO. NUCLIDES 2	RADIUS OF REGION 0.76200	PHYSICAL TEMP. 300.00000	NEUTRON TEMP. 300.00000	FLUX POINTS 5
-------------------	-----------------------------	-----------------------------	----------------------------	------------------

NUCLIDE ID NO. 10	NUCLEI DENSITY 0.62880E-03
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NUCLIDE ID NO. 12	NUCLEI DENSITY 0.47190E-01
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REGION NUMBER 2

NO. NUCLIDES 1	RADIUS OF REGION 0.84582	PHYSICAL TEMP. 300.00000	NEUTRON TEMP. 300.00000	FLUX POINTS 5
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NUCLIDE ID NO. 25	NUCLEI DENSITY 0.51570E-01
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REGION NUMBER 3

NO. NUCLIDES 1	RADIUS OF REGION 1.13840	PHYSICAL TEMP. 300.00000	NEUTRON TEMP. 300.00000	FLUX POINTS 5
-------------------	-----------------------------	-----------------------------	----------------------------	------------------

NUCLIDE ID NO. 21	NUCLEI DENSITY 0.33440E-01
----------------------	-------------------------------

P3 OUTPUT FOR CASE 1

REG	RADIUS	AVERAGE FLUX	THERMAL UTIL.
1	0,7620	2,7929	0,94187252
2	0,8458	3,3102	0,00548697
3	1,1384	3,8354	0,05264051

TOTAL NEUTRON PRODUCTION 2,4944

1,3 (ENRICHED URANIUM-WATER LATTICE W/U = 1,0 0000000000000000

SIGMA-A	SIGMA-S	MASS NUMBER	SOURCE TERM
0,46116E-00	0,39796E-00	238,011	0,007
0,97659E-02	0,72198E-01	26,980	0,535
0,18772E-01	0,31422E 01	1,923	72,618

SHOULD EQUAL TOTAL NEUTRON CAPTURE 2,4944

FLUX TRAVERSE IN REGION 1 CASE 1

RADIUS	FLUX
0,1524	2,395
0,3048	2,492
0,4572	2,660
0,6096	2,911
0,7620	3,262

FLUX TRAVERSE IN REGION 2 CASE 1

RADIUS	FLUX
0,7788	3,281
0,7955	3,300
0,8123	3,319
0,8291	3,338
0,8458	3,358

FLUX TRAVERSE IN REGION 3 CASE 1

RADIUS	FLUX
0,9043	3,605
0,9629	3,789
1,0214	3,925
1,0799	4,024
1,1384	4,095

VOL(1) = 0,18241453E 01
 VOL(2) = 0,42338414E-00
 VOL(3) = 0,18238282E 01
 TVOL = 0,40713577E 01

1.3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 0000000000000000
LIGHT MODERATOR

OUTPUT FROM TEMPEST

CASE 1

TEMPERATURE
0.3000E 03

E MIN
0.

E MAX
0.6830E 00

H1 SCATTERER
221

BUCKLING
0.3211E-02

ISOTOPE	NUMBER	DENSITY
U235 A	10	0.2375E-03
U238 A	12	0.1782E-01
AL A	25	0.5357E-02
H1 A	1	0.3468E-01
O16 A	23	0.1734E-01

MODERATOR	NUMBER	DENSITY
H GAS	301	0.3468E-01
OXYGEN	323	0.1734E-01

ELEMENT NUMBER 310 IS NOT IN THE LIBRARY, ITS CROSS SECTION HAS BEEN SET EQUAL TO ZERO.
 ELEMENT NUMBER 312 IS NOT IN THE LIBRARY, ITS CROSS SECTION HAS BEEN SET EQUAL TO ZERO.
 ELEMENT NUMBER 325 IS NOT IN THE LIBRARY, ITS CROSS SECTION HAS BEEN SET EQUAL TO ZERO.
 ELEMENT NUMBER 112 IS NOT IN THE LIBRARY, ITS CROSS SECTION HAS BEEN SET EQUAL TO ZERO.
 ELEMENT NUMBER 125 IS NOT IN THE LIBRARY, ITS CROSS SECTION HAS BEEN SET EQUAL TO ZERO.
 ELEMENT NUMBER 101 IS NOT IN THE LIBRARY, ITS CROSS SECTION HAS BEEN SET EQUAL TO ZERO.
 ELEMENT NUMBER 123 IS NOT IN THE LIBRARY, ITS CROSS SECTION HAS BEEN SET EQUAL TO ZERO.

AVERAGE TRANSPORT
CROSS SECTION
0.1245E 01
TOTAL FLUX
0.1655E 01

DIFFUSION CONSTANT
D AVERAGED
0.3296E-00
XI SIGMA S
0.7256E 00

MACRO ABSORPTION
CROSS SECTION
0.1390E-00
NU SIGMA F
0.2048E-00

	ELEMENT	ID NUMBER	MICRO CROSS SECTION
	U235 A	10	0.4169E 03
	U235 F	110	0.3550E 03
	U235 X	210	0.9972E 01
ELEMENT NUMBER 310	IS NOT IN THE LIBRARY,		ITS CROSS SECTION HAS BEEN SET EQUAL TO ZERO,
	U238 A	12	0.1782E 01
ELEMENT NUMBER 112	IS NOT IN THE LIBRARY,		ITS CROSS SECTION HAS BEEN SET EQUAL TO ZERO,
	U238 X	212	0.8067E 01
ELEMENT NUMBER 312	IS NOT IN THE LIBRARY,		ITS CROSS SECTION HAS BEEN SET EQUAL TO ZERO,
	AL A	25	0.1554E-00
ELEMENT NUMBER 125	IS NOT IN THE LIBRARY,		ITS CROSS SECTION HAS BEEN SET EQUAL TO ZERO,
	AL X	225	0.1365E 01
ELEMENT NUMBER 325	IS NOT IN THE LIBRARY,		ITS CROSS SECTION HAS BEEN SET EQUAL TO ZERO,
	H1 A	1	0.2142E-00
ELEMENT NUMBER 101	IS NOT IN THE LIBRARY,		ITS CROSS SECTION HAS BEEN SET EQUAL TO ZERO,
	H WA X	221	0.2547E 02
	H GAS	301	0.2070E 02
	O16 A	23	0.1298E-03
ELEMENT NUMBER 123	IS NOT IN THE LIBRARY,		ITS CROSS SECTION HAS BEEN SET EQUAL TO ZERO,
	O16 X	223	0.4025E 01
	OXYGEN	323	0.4560E-00

ENERGY	N(E)	PHI(E)	E PHI(E)
0.	0.	0.	0.
0.0001	0.270420E 01	0.149051E-00	0.149051E-04
0.0002	0.380967E 01	0.206960E-00	0.593920E-04
0.0003	0.464804E 01	0.443735E-00	0.133121E-03
0.0004	0.534652E 01	0.589383E 00	0.235753E-03
0.0005	0.595472E 01	0.733910E 00	0.366955E-03
0.0006	0.649810E 01	0.877320E 00	0.526392E-03
0.0007	0.699188E 01	0.101962E 01	0.713736E-03
0.0008	0.744603E 01	0.116082E 01	0.928658E-03
0.0009	0.786748E 01	0.130093E 01	0.117083E-02
0.0010	0.824310E 01	0.114046E 01	0.114046E-02
0.0011	0.683349E 01	0.124921E 01	0.137413E-02
0.0012	0.710654E 01	0.135689E 01	0.162827E-02
0.0013	0.736406E 01	0.146348E 01	0.190252E-02
0.0014	0.760753E 01	0.156893E 01	0.219650E-02
0.0015	0.783814E 01	0.167323E 01	0.250984E-02
0.0016	0.805692E 01	0.177634E 01	0.284214E-02
0.0017	0.826472E 01	0.187823E 01	0.319299E-02
0.0018	0.846228E 01	0.197888E 01	0.356199E-02
0.0019	0.865024E 01	0.207827E 01	0.394870E-02
0.0020	0.882915E 01	0.217636E 01	0.435271E-02
0.0021	0.900754E 01	0.227516E 01	0.477784E-02
0.0022	0.917957E 01	0.237317E 01	0.522098E-02
0.0023	0.934566E 01	0.247041E 01	0.568195E-02
0.0024	0.950620E 01	0.256690E 01	0.616056E-02
0.0025	0.966155E 01	0.266264E 01	0.665661E-02
0.0026	0.981203E 01	0.275767E 01	0.716993E-02
0.0027	0.995792E 01	0.285198E 01	0.770035E-02
0.0028	0.100995E 02	0.294561E 01	0.824771E-02
0.0029	0.102370E 02	0.303857E 01	0.881185E-02
0.0030	0.103707E 02	0.313087E 01	0.939260E-02
0.0031	0.105070E 02	0.322444E 01	0.999577E-02
0.0032	0.106411E 02	0.331786E 01	0.106171E-01
0.0033	0.107733E 02	0.341115E 01	0.112568E-01
0.0034	0.109037E 02	0.350437E 01	0.119148E-01
0.0035	0.110326E 02	0.359755E 01	0.125914E-01
0.0036	0.111600E 02	0.369074E 01	0.132867E-01
0.0037	0.112863E 02	0.378398E 01	0.140007E-01
0.0038	0.114115E 02	0.387731E 01	0.147338E-01
0.0039	0.115358E 02	0.397078E 01	0.154861E-01
0.0040	0.116593E 02	0.406443E 01	0.162577E-01
0.0041	0.117797E 02	0.415740E 01	0.170454E-01
0.0042	0.118991E 02	0.425043E 01	0.178518E-01
0.0043	0.120175E 02	0.434355E 01	0.186773E-01
0.0044	0.121352E 02	0.443679E 01	0.195219E-01
0.0045	0.122522E 02	0.453018E 01	0.203858E-01
0.0046	0.123686E 02	0.462375E 01	0.212692E-01
0.0047	0.124845E 02	0.471753E 01	0.221724E-01
0.0048	0.126000E 02	0.481156E 01	0.230955E-01

0.0049	0.127152E 02	0.490587E 01	0.240388E-01
0.0050	0.128301E 02	0.500049E 01	0.250024E-01
0.0060	0.135415E 02	0.578149E 01	0.346890E-01
0.0070	0.141425E 02	0.652187E 01	0.456531E-01
0.0080	0.146353E 02	0.721509E 01	0.577207E-01
0.0090	0.150616E 02	0.787569E 01	0.708812E-01
0.0100	0.154234E 02	0.850112E 01	0.850112E-01
0.0110	0.156032E 02	0.901997E 01	0.992197E-01
0.0120	0.157381E 02	0.950250E 01	0.114030E-00
0.0130	0.158328E 02	0.995005E 01	0.129351E-00
0.0140	0.158923E 02	0.103644E 02	0.145102E-00
0.0150	0.159241E 02	0.107497E 02	0.161246E-00
0.0160	0.159311E 02	0.111071E 02	0.177714E-00
0.0170	0.159172E 02	0.114390E 02	0.194463E-00
0.0180	0.158855E 02	0.117471E 02	0.211449E-00
0.0190	0.158304E 02	0.120272E 02	0.228516E-00
0.0200	0.157526E 02	0.122790E 02	0.245581E-00
0.0210	0.156177E 02	0.124745E 02	0.261964E-00
0.0220	0.154715E 02	0.126486E 02	0.278268E-00
0.0230	0.153157E 02	0.128026E 02	0.294459E-00
0.0240	0.151503E 02	0.129366E 02	0.310479E-00
0.0250	0.149784E 02	0.130536E 02	0.326341E-00
0.0260	0.147892E 02	0.131440E 02	0.341745E-00
0.0270	0.145900E 02	0.132140E 02	0.356777E-00
0.0280	0.143864E 02	0.132687E 02	0.371923E-00
0.0290	0.141806E 02	0.133104E 02	0.386000E-00
0.0300	0.139705E 02	0.133374E 02	0.400121E-00
0.0310	0.137461E 02	0.133400E 02	0.413541E-00
0.0320	0.135194E 02	0.133300E 02	0.426559E-00
0.0330	0.132924E 02	0.133093E 02	0.439207E-00
0.0340	0.130653E 02	0.132787E 02	0.451476E-00
0.0350	0.128377E 02	0.132379E 02	0.463327E-00
0.0360	0.126107E 02	0.131882E 02	0.474777E-00
0.0370	0.123855E 02	0.131314E 02	0.485861E-00
0.0380	0.121605E 02	0.130659E 02	0.496504E-00
0.0390	0.119377E 02	0.129942E 02	0.506775E 00
0.0400	0.117171E 02	0.129165E 02	0.516662E 00
0.0410	0.114905E 02	0.128241E 02	0.525789E 00
0.0420	0.112655E 02	0.127254E 02	0.534467E 00
0.0430	0.110422E 02	0.126208E 02	0.542695E 00
0.0440	0.108224E 02	0.125125E 02	0.550552E 00
0.0450	0.106050E 02	0.123998E 02	0.557991E 00
0.0460	0.103913E 02	0.122841E 02	0.565070E 00
0.0470	0.101800E 02	0.121644E 02	0.571728E 00
0.0480	0.997265E 01	0.120428E 02	0.578054E 00
0.0490	0.976820E 01	0.119181E 02	0.583989E 00
0.0500	0.956668E 01	0.117908E 02	0.589539E 00
0.0600	0.770494E 01	0.104026E 02	0.624155E 00
0.0700	0.616740E 01	0.899388E 01	0.629572E 00
0.0800	0.493509E 01	0.769372E 01	0.615497E 00
0.0900	0.396034E 01	0.654863E 01	0.589376E 00

0,1000	0,320000E 01	0,557758E 01	0,557758E 00
0,1100	0,260671E 01	0,476525E 01	0,524177E 00
0,1200	0,214596E 01	0,409741E 01	0,491689E-00
0,1300	0,178780E 01	0,355293E 01	0,461881E-00
0,1400	0,150809E 01	0,311019E 01	0,435426E-00
0,1500	0,128833E 01	0,275022E 01	0,412533E-00
0,1600	0,111405E 01	0,245619E 01	0,392991E-00
0,1700	0,974910E 00	0,221557E 01	0,376647E-00
0,1800	0,863055E 00	0,201823E 01	0,363282E-00
0,1900	0,771773E 00	0,185423E 01	0,352303E-00
0,2000	0,695786E 00	0,171509E 01	0,343018E-00
0,2100	0,632633E 00	0,159793E 01	0,335565E-00
0,2200	0,579457E 00	0,149806E 01	0,329573E-00
0,2300	0,533801E 00	0,141104E 01	0,324540E-00
0,2400	0,494086E-00	0,133415E 01	0,320196E-00
0,2500	0,459602E-00	0,126662E 01	0,316656E-00
0,2600	0,429467E-00	0,120701E 01	0,313824E-00
0,2700	0,402874E-00	0,115385E 01	0,311538E-00
0,2800	0,379671E-00	0,110735E 01	0,310057E-00
0,2900	0,359061E-00	0,106577E 01	0,309073E-00
0,3000	0,340465E-00	0,102785E 01	0,308354E-00
0,3100	0,324475E-00	0,995768E 00	0,308688E-00
0,3200	0,309877E-00	0,966185E 00	0,309179E-00
0,3300	0,296492E-00	0,938785E 00	0,309799E-00
0,3400	0,284204E-00	0,913411E 00	0,310560E-00
0,3500	0,272802E-00	0,889565E 00	0,311348E-00
0,3600	0,261920E-00	0,866195E 00	0,311830E-00
0,3700	0,251806E-00	0,844235E 00	0,312367E-00
0,3800	0,242367E-00	0,823497E 00	0,312929E-00
0,3900	0,233553E-00	0,803921E 00	0,313529E-00
0,4000	0,225293E-00	0,785371E 00	0,314148E-00
0,4100	0,217343E-00	0,767067E 00	0,314497E-00
0,4200	0,209862E-00	0,749645E 00	0,314851E-00
0,4300	0,202822E-00	0,733070E 00	0,315220E-00
0,4400	0,196185E-00	0,717279E 00	0,315603E-00
0,4500	0,189917E-00	0,702210E 00	0,315995E-00
0,4600	0,183885E-00	0,687420E 00	0,316213E-00
0,4700	0,178176E-00	0,673278E 00	0,316441E-00
0,4800	0,172764E-00	0,659736E 00	0,316673E-00
0,4900	0,167631E-00	0,646767E 00	0,316916E-00
0,5000	0,162753E-00	0,634322E 00	0,317161E-00
0,5100	0,158069E-00	0,622197E 00	0,317320E-00
0,5200	0,153611E-00	0,610548E 00	0,317485E-00
0,5300	0,149364E-00	0,599351E 00	0,317656E-00
0,5400	0,145315E-00	0,588576E 00	0,317831E-00
0,5500	0,141451E-00	0,578206E 00	0,318013E-00
0,5600	0,137734E-00	0,568109E 00	0,318141E-00
0,5700	0,134182E-00	0,558378E 00	0,318276E-00
0,5800	0,130784E-00	0,548992E 00	0,318415E-00
0,5900	0,127533E-00	0,539941E 00	0,318565E-00

0.6000
0.6100
0.6200
0.6300
0.6400
0.6500
0.6600
0.6700
0.6800
0.6900

0.124414E-00
0.121407E-00
0.118519E-00
0.115744E-00
0.113078E-00
0.110515E-00
0.108037E-00
0.105652E-00
0.103355E-00
0.101142E-00

0.531181E 00
0.522641E 00
0.514374E 00
0.506368E 00
0.498613E-00
0.491106E-00
0.483772E-00
0.476663E-00
0.469766E-00
0.463074E-00

0.318709E-00
0.318811E-00
0.318912E-00
0.319012E-00
0.319113E-00
0.319219E-00
0.319290E-00
0.319364E-00
0.319441E-00
0.319521E-00

FIRST COLLISION PROBABILITIES /
P(1,J) J=1 TO 3
0.17652E-00 0.10967E-01 0.46373E-01
P(2,J) J=1 TO 3
0.10822E-00 0.17979E-01 0.69353E-01
P(3,J) J=1 TO 3
0.75269E-01 0.11407E-01 0.79611E-01

URANIUM COLLISION PROBABILITY = 0.59967E 00 FOR 1-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.
THORIUM COLLISION PROBABILITY = 0. FOR 1-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.
URANIUM COLLISION PROBABILITY = 0.89200E 00 FOR 2-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.32772E-00
THORIUM COLLISION PROBABILITY = 0. FOR 2-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.
URANIUM COLLISION PROBABILITY = 0.10381E 01 FOR 3-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.14077E-00
THORIUM COLLISION PROBABILITY = 0. FOR 3-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.
URANIUM COLLISION PROBABILITY = 0.11113E 01 FOR 4-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.65837E-01
THORIUM COLLISION PROBABILITY = 0. FOR 4-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.
URANIUM COLLISION PROBABILITY = 0.11479E 01 FOR 5-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.31911E-01
THORIUM COLLISION PROBABILITY = 0. FOR 5-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.
URANIUM COLLISION PROBABILITY = 0.11663E 01 FOR 6-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.15726E-01
THORIUM COLLISION PROBABILITY = 0. FOR 6-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.
URANIUM COLLISION PROBABILITY = 0.11755E 01 FOR 7-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.78121E-02
THORIUM COLLISION PROBABILITY = 0. FOR 7-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.
URANIUM COLLISION PROBABILITY = 0.11801E 01 FOR 8-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.38961E-02
THORIUM COLLISION PROBABILITY = 0. FOR 8-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.
URANIUM COLLISION PROBABILITY = 0.11824E 01 FOR 9-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.19469E-02
THORIUM COLLISION PROBABILITY = 0. FOR 9-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.
URANIUM COLLISION PROBABILITY = 0.11835E 01 FOR 10-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.97382E-03
THORIUM COLLISION PROBABILITY = 0. FOR 10-FLIGHT COLLISIONS, CONTRIBUTION OF LAST FLIGHT = 0.

DANCOFF SHIELDING FACTOR	EFFECTIVE CORE LENGTH OF FUEL	EFFECTIVE SURFACE OF FUEL	VOLUME OF FUEL
0.16500E-00	0.18251E 01	0.39978E 01	0.18241E 01

OUTPUT FROM GAM CASE 1 1.3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 000000000000000000 PAGE

THE NUMBER OF NUCLIDES FROM CHAIN 2 IS 5

NUMBER OF NUCLIDES PRESENT IS 5

NUMBER OF BROAD GROUPS IS 3

IF YES(R=1),IF NO(R=0)

SHOULD AGE BE CALCULATED • R=1

SHOULD MICROSCOPIC CROSS-SECTIONS BE AVERAGED • R=0

OUTPUT FROM GAM CASE 1 1.3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 000000000000000000 PAGE 2

REVISED BROAD GROUP BOUNDARIES BY ENERGY AND LETHARGY

DESIRED ENERGY	ACTUAL ENERGY USED	DESIRED LETHARGY	ACTUAL LETHARGY USED
0.09999999E 08	0.09999999E 08	0.	0.
0.13500000E 07	0.13533528E 07	0.20024804E 01	0.20000000E 01
0.45400000E 03	0.45399930E 03	0.99999981E 01	0.09999999E 02
0.68300000E 00	0.68256035E 00	0.16499355E 02	0.16500000E 02

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OUTPUT FROM GAM CASE 1 . 1.3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 000000000000000000 PAGE 3

GROUP	DELTA-U
1	0.20000000E 01
2	0.80000000E 01
3	0.64999999E 01

THE SOURCE SPECTRUM USED FOR THIS PROBLEM IS
THE U-235 SPECTRUM (CRANBERG)

OUTPUT FROM GAM CASE 1 1.3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 0000000000000000 PAGE 4

RESONANCE GEOMETRY

(G=1) FOR SLAB

(G=2) FOR CYLINDER

(G=3) FOR SPHERE

G=2

SELF SHIELDING FACTORS ARE SET EQUAL TO UNITY BELOW 1.35 MEV.

ABOVE 1.35 MEV.

SELF SHIELDING FACTORS FOR U-238 = 0.10266E 01

SELF SHIELDING FACTORS FOR TH-232 = 0.96406E 00

SELF SHIELDING FACTORS FOR ALL OTHER ISOTOPES = 0.96406E 00

TAPE I.D. NUMBER IS 1

HYDROGEN

NUCLEI DENSITY= 0.29959940E-01 ATOMS*10E24/C.C.

TAPE I.D. NUMBER IS 10

URANIUM-235

NUCLEI DENSITY= 0.28172974E-03 ATOMS*10E24/C.C.

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OUTPUT FROM GAM CASE 1 1.3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 0000000000000000 PAGE 5

TAPE I.D. NUMBER IS 12

URANIUM-238

NUCLEI DENSITY= 0.21143172E-01 ATOMS*10E24/C.C.

THIS NUCLIDE HAS RESONANCE ABSORPTION

RESONANCE DATA

SIGMA-M= 0.

SIGMA-P= 0.84332483E 01

SIGMA-M(EFF)= 0.20043765E 02

L(BAR)= 0.18251496E 01

TEMPERATURE= 0.30000000E 03

LUMPED NUCLEI DENSITY= 0.47189999E-01 ATOMS*10E24/C.C.

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OUTPUT FROM GAM CASE 1 1.3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 000000000000000000 PAGE 6

TAPE I,D, NUMBER IS 23

OXYGEN

NUCLEI DENSITY= 0.14979970E+01 ATOMS*10E24/C.C.

TAPE I,D, NUMBER IS 25

ALUMINUM

NUCLEI DENSITY= 0.53628104E+02 ATOMS*10E24/C.C.

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OUTPUT FROM GAM CASE 1 1,3 (ENRICHED URANIUM-WATER LATTICE W/U = 1,0 0000000000000000 PAGE 7

THE FLUXES AND CURRENT TERMS ARE CALCULATED USING THE B-1 APPROXIMATION

K= 0,56665686E-01

GROUP	FLUX	CURRENT TERM
1	0,65683994E-02	0,67331374E-03
2	0,18847663E-01	0,19733748E-02
3	0,44664917E-01	0,46348594E-02
4	0,73214389E-01	0,68893247E-02
5	0,10623270E-00	0,84384047E-02
6	0,13503180E-00	0,11310947E-01
7	0,13372881E-00	0,98947929E-02
8	0,14603198E-00	0,10490803E-01
9	0,14776142E-00	0,10187743E-01
10	0,15923359E-00	0,10958380E-01
11	0,16835254E-00	0,11443865E-01
12	0,15280474E-00	0,10288697E-01
13	0,12325443E-00	0,73471412E-02
14	0,13548519E-00	0,73065817E-02
15	0,11974043E-00	0,64306737E-02
16	0,10476404E-00	0,53403040E-02
17	0,92042931E-01	0,43481380E-02
18	0,85507775E-01	0,39660402E-02
19	0,74224429E-01	0,33078200E-02
20	0,73327382E-01	0,31284764E-02
21	0,65982883E-01	0,28237824E-02
22	0,62147521E-01	0,25726566E-02
23	0,55948891E-01	0,21070601E-02
24	0,57600228E-01	0,22055913E-02
25	0,54369087E-01	0,20852285E-02
26	0,51704627E-01	0,19364733E-02
27	0,50913621E-01	0,18728738E-02
28	0,49458121E-01	0,17814285E-02
29	0,47419519E-01	0,16953082E-02
30	0,47429547E-01	0,16842561E-02
31	0,46959956E-01	0,16587155E-02
32	0,45157017E-01	0,15510660E-02
33	0,45854697E-01	0,15441364E-02
34	0,45694914E-01	0,15252475E-02
35	0,43591964E-01	0,14307757E-02
36	0,44755886E-01	0,14134583E-02
37	0,42022239E-01	0,12670663E-02
38	0,42583961E-01	0,12537092E-02
39	0,42172938E-01	0,12781302E-02
40	0,41670077E-01	0,12587603E-02

OUTPUT FROM GAM CASE 1 1,3 (ENRICHED URANIUM-WATER LATTICE W/U = 1,0 0000000000000000 PAGE 8

GROUP	FLUX	CURRENT TERM
41	0,41483557E-01	0,13055227E-02
42	0,40923141E-01	0,11871040E-02
43	0,40556053E-01	0,11649731E-02
44	0,36381628E-01	0,47095283E-03
45	0,41761235E-01	0,13889579E-02
46	0,33752865E-01	0,38629930E-03
47	0,40410569E-01	0,13094723E-02
48	0,34766895E-01	0,61196133E-03
49	0,39207332E-01	0,13091769E-02
50	0,38077614E-01	0,13336018E-02
51	0,26970159E-01	0,17757822E-03
52	0,41257043E-01	0,13386078E-02
53	0,27242970E-01	0,28818417E-03
54	0,36558174E-01	0,12316424E-02
55	0,32869285E-01	0,11192982E-02
56	0,31987137E-01	0,10793514E-02
57	0,22333645E-01	0,47302026E-03
58	0,29773238E-01	0,93651916E-03
59	0,30246707E-01	0,11040086E-02
60	0,29551231E-01	0,11029698E-02
61	0,29608242E-01	0,11322518E-02
62	0,29196671E-01	0,11319984E-02
63	0,28984617E-01	0,11351658E-02
64	0,27946462E-01	0,10970405E-02
65	0,27238674E-01	0,10719965E-02
66	0,26597971E-01	0,10487947E-02
67	0,25647013E-01	0,10071958E-02
68	0,24413876E-01	0,94970725E-03

GROUP	PHI(0,0) TERM	PHI(1,1) TERM	PHI(2,0) TERM	AGE
1	0.19908703E-00	0.11040902E 01	0.30712425E 01	0.15426633E 02
2	0.57193717E 00	0.32381451E 01	0.90416774E 01	0.15808865E 02
3	0.13563669E 01	0.76065207E 01	0.21764499E 02	0.16046172E 02
4	0.22084389E 01	0.11184108E 02	0.30397529E 02	0.13764260E 02
5	0.31890699E 01	0.13583693E 02	0.38751385E 02	0.12151313E 02
6	0.40672492E 01	0.18282313E 02	0.53858058E 02	0.13241887E 02
7	0.40096048E 01	0.15882917E 02	0.47075251E 02	0.11740620E 02
8	0.43892729E 01	0.16867069E 02	0.54978748E 02	0.12525707E 02
9	0.44518074E 01	0.16398532E 02	0.59131376E 02	0.13282554E 02
10	0.48217998E 01	0.17713852E 02	0.71849208E 02	0.14900910E 02
11	0.51251421E 01	0.18584343E 02	0.85084919E 02	0.16601475E 02
12	0.46653984E 01	0.16736633E 02	0.81769966E 02	0.17526898E 02
13	0.37593789E 01	0.11922003E 02	0.64659740E 02	0.17199580E 02
14	0.41435406E 01	0.11883567E 02	0.74771852E 02	0.18049401E 02
15	0.36696664E 01	0.10472054E 02	0.68629154E 02	0.18701741E 02
16	0.32147762E 01	0.87013307E 01	0.61405196E 02	0.19100923E 02
17	0.28268567E 01	0.70868023E 01	0.54758444E 02	0.19370788E 02
18	0.26299229E 01	0.64704616E 01	0.52127737E 02	0.19821013E 02
19	0.22849912E 01	0.53993101E 01	0.45949814E 02	0.20109404E 02
20	0.22604462E 01	0.51119569E 01	0.46418044E 02	0.20534903E 02
21	0.20362519E 01	0.46178517E 01	0.42507058E 02	0.20875145E 02
22	0.19197749E 01	0.42103671E 01	0.40664860E 02	0.21182097E 02
23	0.17288986E 01	0.34489358E 01	0.36809921E 02	0.21290965E 02
24	0.17821793E 01	0.36145068E 01	0.38648879E 02	0.21686301E 02
25	0.16837140E 01	0.34199081E 01	0.36985072E 02	0.21966362E 02
26	0.16023627E 01	0.31779949E 01	0.35561448E 02	0.22193132E 02
27	0.15792395E 01	0.30761409E 01	0.35483049E 02	0.22468439E 02
28	0.15353230E 01	0.29280734E 01	0.34880494E 02	0.22718668E 02
29	0.14730494E 01	0.27883729E 01	0.33781565E 02	0.22933082E 02
30	0.14745761E 01	0.27724406E 01	0.34196439E 02	0.23190691E 02
31	0.14611378E 01	0.27325367E 01	0.34247792E 02	0.23439123E 02
32	0.14058862E 01	0.25566819E 01	0.33217129E 02	0.23627181E 02
33	0.14286332E 01	0.25470798E 01	0.34075184E 02	0.23851597E 02
34	0.14248040E 01	0.25179317E 01	0.34342930E 02	0.24103615E 02
35	0.13599621E 01	0.23632350E 01	0.33008005E 02	0.24271268E 02
36	0.13971224E 01	0.23360488E 01	0.34175120E 02	0.24461078E 02
37	0.13124417E 01	0.20951021E 01	0.32308062E 02	0.24616760E 02
38	0.13307670E 01	0.20742494E 01	0.33003303E 02	0.24800210E 02
39	0.13188657E 01	0.21162536E 01	0.33006755E 02	0.25026622E 02
40	0.13039771E 01	0.20855268E 01	0.32891824E 02	0.25224234E 02

OUTPUT FROM GAM CASE 1 1.3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 0000000000000000 PAGE 10

GROUP	PHI(0,0) TERM	PHI(1,1) TERM	PHI(2,0) TERM	AGE
41	0.12991763E 01	0.21648222E 01	0.33094447E 02	0.25473406E 02
42	0.12821428E 01	0.19691811E 01	0.32822188E 02	0.25599478E 02
43	0.12714634E 01	0.19337322E 01	0.32805448E 02	0.25801329E 02
44	0.11394012E 01	0.78086427E 00	0.29026778E 02	0.25475466E 02
45	0.15108886E 01	0.23092888E 01	0.34336013E 02	0.26192929E 02
46	0.10579528E 01	0.64110907E 00	0.27226750E 02	0.25735314E 02
47	0.12696142E 01	0.21793519E 01	0.33605993E 02	0.26469452E 02
48	0.10914160E 01	0.10172515E 01	0.28612547E 02	0.26215985E 02
49	0.12333746E 01	0.21816910E 01	0.33135505E 02	0.26865727E 02
50	0.11989779E 01	0.22244574E 01	0.32568485E 02	0.27163541E 02
51	0.84736277E 00	0.29551622E 00	0.22435500E 02	0.26476853E 02
52	0.12998601E 01	0.22346612E 01	0.35549488E 02	0.27348702E 02
53	0.85698931E 00	0.48015043E 00	0.23020119E 02	0.26861618E 02
54	0.11530143E 01	0.20583230E 01	0.31908207E 02	0.27673730E 02
55	0.10374491E 01	0.18716868E 01	0.28953677E 02	0.27908527E 02
56	0.10102435E 01	0.18058796E 01	0.28393675E 02	0.28105772E 02
57	0.70461585E 00	0.79047507E 00	0.19571777E 02	0.27776521E 02
58	0.94095761E 00	0.15678404E 01	0.26644709E 02	0.28316588E 02
59	0.95718300E 00	0.18508197E 01	0.27498849E 02	0.28728936E 02
60	0.93595525E 00	0.18505361E 01	0.27133307E 02	0.28989962E 02
61	0.93862165E 00	0.19013393E 01	0.27479789E 02	0.29276748E 02
62	0.92643109E 00	0.19025552E 01	0.27390856E 02	0.29565993E 02
63	0.92040128E 00	0.19092607E 01	0.27431108E 02	0.29803422E 02
64	0.88819804E 00	0.18465862E 01	0.26709954E 02	0.30072071E 02
65	0.86633515E 00	0.18056678E 01	0.26250118E 02	0.30300188E 02
66	0.84661770E 00	0.17678795E 01	0.25859118E 02	0.30544031E 02
67	0.81690455E 00	0.16988377E 01	0.25125352E 02	0.30756778E 02
68	0.77809294E 00	0.16027626E 01	0.24077273E 02	0.30943955E 02

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OUTPUT FROM GAM CASE 1 1.3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 000000000000000000 PAGE 11

THE FOLLOWING DATA HAS BEEN AVERAGED TO PROVIDE DATA FOR 1 FAST GROUP
OVER THE ENERGY RANGE FROM 10 MEV. TO 0.68256035E 00E.V.

DIFFUSION COEFFICIENT = 0.89016382E 00
SIGMA TRANSPORT = 0.37446290E-00
SIGMA ABSORPTION = 0.16359875E-01
SIGMA CAPTURE = 0.11864873E-01
SIGMA FISSION = 0.44950016E-02
NU-SIGMA FISSION = 0.11730406E-01
AVERAGE NU = 0.26096556E 01
P(0)-TRANSFER = 0.17966001E-01
P(1)-TRANSFER = 0.23122391E-01
INELASTIC TRANSFER = 0.
N-2N TRANSFER = 0.
TOTAL SIGMA REMOVAL = 0.34325875E-01
TOTAL SIGMA OUT = 0.17966001E-01

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OUTPUT FROM GAM CASE 1 1,3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 0000000000000000 PAGE 11

THE FOLLOWING DATA HAS BEEN AVERAGED TO PROVIDE DATA FOR 1 FAST GROUP
OVER THE ENERGY RANGE FROM 10 MEV, TO 0.68256035E 00E.V.

DIFFUSION COEFFICIENT = 0.89016382E 00
SIGMA TRANSPORT = 0.37446290E-00
SIGMA ABSORPTION = 0.16359875E-01
SIGMA CAPTURE = 0.11864873E-01
SIGMA FISSION = 0.44950016E-02
NU-SIGMA FISSION = 0.11730406E-01
AVERAGE NU = 0.26096556E 01
P(0)-TRANSFER = 0.17966001E-01
P(1)-TRANSFER = 0.23122391E-01
INELASTIC TRANSFER = 0.
N-2N TRANSFER = 0.
TOTAL SIGMA REMOVAL = 0.34325875E-01
TOTAL SIGMA OUT = 0.17966001E-01

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OUTPUT FROM GAM CASE 1 1.3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 00000000000000000000 PAGE 12
 BROAD GROUP MACROSCOPIC AVERAGED CROSS-SECTIONS

GROUP	DIFF. COEF.	SIGMA TRANSPORT	FISSION SOURCE	SIGMA REMOVAL	SIGMA OUT
1	0.14426086E 01	0.23106290E-00	0.57308519E 00	0.11490142E-00	0.10214314E-00
2	0.86780348E 00	0.38411154E-00	0.42690913E-00	0.49796230E-01	0.43559491E-01
3	0.52872694E 00	0.63044515E 00	0.	0.12921728E-00	0.81795291E-01

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OUTPUT FROM GAM CASE 1 1,3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 00000000000000000000 PAGE 13
 BROAD GROUP MACROSCOPIC AVERAGED CROSS-SECTIONS

GROUP	SIGMA ABSORPTION	SIGMA CAPTURE	SIGMA FISSION	NU-SIGMA FISSION	AVERAGE NU
1	0.12758274E-01	0.10632776E-02	0.11694995E-01	0.33153049E-01	0.28348063E 01
2	0.62367395E-02	0.50540258E-02	0.11827137E-02	0.28924870E-02	0.24456358E 01
3	0.47421996E-01	0.39193039E-01	0.82289562E-02	0.19996363E-01	0.24299999E 01

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OUTPUT FROM GAM CASE 1 1.3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 000000000000000000 PAGE 14
BUCKLING AND K INFINITE

FAST GROUP PARAMETERS

ETA TIMES F	RES ABS INT	RES FIS INT	RES ESC PROB	DEL U238	DEL TH232	AGE
0.71702E 00	0.97001E 01	0.	0.52340E 00	0.19603E-00	0.	0.30944E 02

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THERMAL GROUP PARAMETERS , MATERIAL BUCKLING , AND K-INFINITE

ETA TIMES F	DIF LENGTH**2	K-INFINITE	BUCKLING
0.14734E 01	0.23709E 01	0.11129E 01	0.34465E-02

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OUTPUT FROM GAM CASE 1 1.3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 000000000000000000 PAGE 15
BROAD GROUP MACROSCOPIC AVERAGED CROSS-SECTIONS

GROUP	P(0) TRANSFER	P(1) TRANSFER	INELASTIC TRANSFER	N-2N TRANSFER	TOTAL TRANSFER
FROM 1 TO 10,16712153E-00	0.16898049E-00	0.65000474E-02	0.16905162E-03	0.17395968E-00	
FROM 1 TO 20,49902089E-01	0.67617302E-01	0.51880261E-01	0.34554289E-03	0.10247343E-00	
FROM 1 TO 30,15176442E-04	0.45973373E-06	0.53178595E-07	0.13072468E-09	0.15229882E-04	
FROM 1 TO 40,22832461E-07	0.26825960E-10	0.	0.	0.22832461E-07	
FROM 2 TO 20,60988040E 00	0.70032279E 00	0.10858301E-01	0.	0.62073870E 00	
FROM 2 TO 30,43494936E-01	0.55871752E-01	0.17061248E-06	0.	0.43495107E-01	
FROM 2 TO 40,64384284E-04	0.32907798E-05	0.	0.	0.64384284E-04	
FROM 3 TO 30,11665454E 01	0.11117377E 01	0.	0.	0.11665454E 01	
FROM 3 TO 40,81795291E-01	0.10549366E-00	0.	0.	0.81795291E-01	

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CROSS SECTION DATA FOR MFN CODE

GEOMETRY = 3 A = 0. B = 0. EXTRAPOLATION CONST. = 0.71000

0 5	13881.14426E 01.	.11490E=00.33153E-01.57309E 00	1
0 5	13888.86780E 00.	.49796E=01.28925E-02.42691E=00	2
0 5	13895.52873E 00.	.12922E=00.19996E=01.	3
0 5	13902.32956E=00.	.13901E=00.20481E=00	4
0 6	13909.	.10247E=00. .15230E=04.43495E=01.	3- 3
0 4	13915.22832E=07.64384E=04.81795E=01.	.	3- 4

CROSS SECTION DATA FOR DTF CODE

G= 04 L=11 NUCLIDE NO.00001		ISO	GAMTEC	
0.12758E=01	0.33153E=01	0.28886E=00	0.17396E=00 0.	0- 1
0.	0.	0.	0.62367E=02 0.28925E=02 0.67053E 00	2- 2
0.62074E 00	0.10247E=00 0.	0.	0.	0- 2
0.47422E=01	0.19996E=01	0.12958E 01	0.11665E 01 0.43495E=01 0.15230E=04	1- 3
0.	0.	0.	0.13901E=00 0.20481E=00 0.12454E 01	4- 4
0.11064E 01	0.81795E=01	0.64384E=04	0.22832E=07 0.	0- 4

G= 04 L=11 NUCLIDE NO.00002		ANISO	GAMTEC	
-0.78866E=01	0.	0.	0.56327E=01 0.	0- 1
0.	0.	0.	-0.25207E=00 0.	2- 2
0.23344E=00	0.22539E=01	0.	0.	0- 2
-0.40574E=00	0.	0.	0.37058E=00 0.18624E=01 0.15324E=06	1- 3
0.	0.	0.	0.	4- 4
0.	0.35165E=01	0.10969E=05	0.89420E=11 0.	0- 4

OUTPUT FROM GAM CASE 1 1.3 (ENRICHED URANIUM-WATER LATTICE W/U = 1.0 0000000000000000 PAGE 17

INTERMEDIATE INTEGRALS

WEIGHTING FACTORS USED FOR EACH BROAD GROUP

GROUP	ALPHA(FLUX)	BETA(CURRENT TERM)	GAMMA(LEAKAGE*FLUX)
1	0.66432057E 00	0.54305820E-01	=0.37644180E-01
2	0.24199359E 01	0.11899957E-00	=0.13712732E-00
3	0.86568307E 00	0.25936449E-01	=0.49054524E-01

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