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Fast Reactor Design Computations

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of the
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Fast Reactor Design Computations

by

Carroll B. Mills

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FAST REACTOR DESIGN COMPUTATIONS

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ABSTRACT

Neutron physics studies of the existing wide variety of nuclear power systems are best accomplished with fast computing machines because of the large mass of cross-section data to be processed and the complexity of neutron transport in energy and space.

Solution of the neutron transport and absorption problem involves a Master Data Tape from which properly averaged neutron cross sections must be obtained for the integration of the S_n neutron conservation equations. In addition, the neutron cross sections must be modified for different reactors and the system of calculation must be validated by reactivity correlations. Finally, the characteristics of the reactors of interest may be computed.

Codes written to do these things are briefly described, and a study of fast breeder reactors illustrates the applicability of the entire system of computing.

INTRODUCTION

The neutronics of neutron multiplying systems, nuclear reactors, are sensitive functions of neutron cross sections and reactor geometry. The macroscopic cross sections Σ_x and the neutron flux ϕ determine fission density distribution and the critical and kinetic constants of particular reactors. Because of the specific values of neutron absorption integrals over energy E , and space $\vec{r}$, the analysis of a single reactor critical experiment, or a power reactor, involves difficult and tedious procedures. For this reason, reactor-oriented computing codes and data storage methods have been developed to permit the completion of the wide variety

of studies required to thoroughly understand any one reactor. This report correlates simple nuclear physics methods with results of work on fast neutron flux reactor problems and emphasizes the routines of reactor calculations.

As an illustration of the system of computations, we will study the fast breeder reactor. The two basic breeders are $^{232}\text{Th} \rightarrow ^{233}\text{U}$ and $^{238}\text{U} \rightarrow ^{239}\text{Pu}$ reactors. Some of their characteristics are briefly compared. Since isotope resonance integrals have a small effect on a very fast neutron spectrum breeder reactor, we will only show the necessary codes without detailing their effects on the absorption distribution as a function of temperature.

METHODS OF ANALYSIS

1. The essentials of fast reactor analysis are:

a. Neutron cross-section data for a number of isotopes, which have been evaluated and compiled in a depository called the Master Data Tape.^{1,2} The codes required to produce multigroup averaged microscopic neutron cross sections consistent with specific flux spectra (see Appendix A) have been described and written by Hansen, Sandmeier, and Lazarus.²

b. Numerical means of solving the neutron transport equations using multigroup averaged neutron cross sections. These have been developed and detailed by Carlson,³ with further generalizations into flux anisotropies by Lathrop.³ Written for one- and two-dimensional geometries, they enable a wide variety of possible reactor assemblies to be easily and accurately computed.

c. Fission neutron spectra required in a. and b. for proper weighting functions. These have been investigated by Terrell.⁴ Appendix A contains a simple code using prompt fission neutron spectra to produce a weighting function in the form required for the production of group averaged neutron cross sections from the Master Data Tape. This is one of the several places where reiteration of neutron flux spectra is used, since neutron cross sections must be used to make a flux calculation, and a flux weighting is used for a proper averaging of neutron cross sections over the energy variable.

d. Critical reactor experiments such as have been performed and analyzed by Paxton⁵ and his associates, using a combination of experimental and theoretical means to obtain the integral constants required to correlate equations and cross sections.

The names mentioned in this context should certainly include K. Parker at Atomic Weapons Research Establishment, R. Howerton at Livermore Radiation Laboratory, D. Callihan at Oak Ridge National Laboratory, L. Rosen at Los Alamos Scientific Laboratory, and others well known in this field who are quoted in the above references.

2. The first determination in any nuclear reactor physics analysis is the distribution in the

energy variable of the "monoenergetic" neutron cross sections. The need for this determination originates with the fact that different reactor types have different flux spectra emphasizing different energy regions. Since the reactor integrals for absorption, fission, and escape use the differentials (volume element) $\times$ (average cross section $\sigma(E)$) $\times$ (atoms/ml) $\times$ (neutron flux $\phi(E)$ in ΔE), the weighting function, $\phi(E)\Delta E$, will determine ΔE for best results. In this study, because of the extensive and relatively accurate fast reactor assembly work done by Hansen,⁶ we use his group limits for $10^7 < E < 10^4$ eV as bounds to a finer* ($\times 1/4$ in lethargy) group structure. It is then a simple matter to compare these calculations with his previous integral studies. See Appendix B for the group energy bounds used in this study.

3. The second determination is that of static or kinetic performance; we choose static to examine the time-dependent, but very slowly varying, reactor breeding quantities. Breeding gain depends on sharply varying neutron reaction cross sections and flux.

4. Since the reactor type that we wish to analyze determines the kind of critical experiments needed for basic reference studies, and to some degree the methods used, we specify the small, fast reactor critical assemblies⁶ of ^{235}U (Godiva), ^{239}Pu (Jezebel), and ^{235}U reflected by ^{238}U (Topsy). The necessity of a unitary approach is illustrated by the fact that the several components: prompt neutron energy spectrum, neutron flux distribution in a critical assembly, and critical experiment descriptions, are all used together, as is shown in Appendix A. The ^{233}U critical assembly data were placed in a separate Appendix, C, because very little is known of its characteristics. There appear to be no fast reactor experiments using ^{233}U in ^{232}Th to help with the evaluation of this mixture as a fast breeder reactor.

5. The evaluation of a critical assembly is made in terms of integrals over energy and space.

* A previous study showed that Hansen's group structure may contain energy intervals that are too wide, because of the rapid changes in $\sigma(E, E' \rightarrow E, \mu)$ above 0.1 MeV.

The results are given in terms of critical mass and size, temperature coefficients and other kinetic constants, and the reactivity distributions. These last relate position and the effect on critical mass or neutron multiplication constant (as a fraction of the value in terms of Δk_{eff} of the delayed neutron fraction). Since an integral number may include many cancelling errors, a series of measurements of reactivity in the particular critical assembly most like the breeder reactor of interest is helpful in evaluating accuracy of methods of evaluation themselves.

The group averaged neutron capture, fission, and $(n,2n)$ cross sections used in this study of two fast breeder reactors are listed in Appendix B with the "monoenergetic" multigroup energy bounds. These tabulated values are used for (n,γ) capture, (n,f) fission, and $(n,2n)$ distributions along the assembly radius in the code shown in Appendix B. The isotopes formed by (n,γ) and $(n,2n)$ reactions in the several reactor fuel and breeding isotopes may be located on a chart of the nuclides, a section of which is Appendix D.

The comparison of computed with experimental reactivity coefficients in the center of the reference critical assemblies is found in Appendix E. The necessary theory has been discussed by Hansen, in the general summary of LASL fast reactor work published by Paxton.^{5,6} The differences between this calculation and the tabulation are larger than we expected, and may originate in the effects of the uncertainty ($\pm 5\%$) in delayed neutron value as well as in the computed cross section and the experimental reactivity coefficient values. The code used to establish the computed central reactivity value and the spatial distribution of this value for small neutron scattering ($\sum_a \gg \sum_s$) is found in Appendix E. A second reactivity code including cross-section scattering and flux anisotropy effects, but without the detailed component parts, is shown in Appendix F.

The time-dependent composition of a breeding reactor may be quickly computed as a function of $\overline{\phi t}$, the core-averaged exposure to flux over time t , with the use of Code EQUIL, given in Appendix G.

We will show that there are strong neutron flux spectrum shift effects as moderator (and other materials) are inserted into critical systems. In

Appendix H, a code is shown that inserts resonance integrals into neutron cross-section tables. Code MCT 2 computes averaged unresolved resonances of two mixed isotopes as a function of temperature and moderator scattering (σ_p) by Monte Carlo techniques. Code RMITU 72 uses those values for unresolved resonances and performs an integration over the resolved resonance region as a function of temperature and σ_p , or foil thickness, for one or more isotopes for up to 72 groups (total in $10^7 > E > 0.0$ eV). The output of this code is in the correct form to be used directly with the DTF IV codes.³

Appendix I shows the code (FISH) used to integrate the product of the neutron flux, $\phi(r,E)$, over the group-averaged neutron cross sections listed in Appendix B for specific cross-section averaging and total absorption values. A "tape correct" code (COPY 1), shown in Appendix J, permits changing the cross-section tapes provided by Lazarus directly to the form required by the DTF IV code, with corrections where needed.

Code FABELL is used for calculation of thermal intergroup transfer of neutrons using the free atom model. This code is shown in Appendix K.

In résumé, the Master Data Tape was the source of all neutron cross sections. Group-averaged, second Legendre polynomial (three-table) values in $10^7 > E > 10^4$ eV were produced by Lazarus on tape and converted directly to the S_n (DTF code³ by Carlson and Lathrop) format. The weighting function was that of the Godiva (Appendix A) neutron flux spectrum; the group widths were sufficiently narrow that the same weighting function was used for all fast reactor calculations in this study. The DTF program³ written by Carlson and Lathrop and coded by Lathrop and Brinkley (DTF IV) was used to compute all fast assemblies using the S_8 , three-table choice of degree of accuracy in this code. After the relative accuracy of the cross-section sets was established by the calculation of several experiments, reactivity calculations were compared with experimental values in these fast flux spectrum assemblies. Finally, breeding gains were computed for the two basic types of fast breeder reactors, $^{232}\text{Th} \rightarrow ^{233}\text{U}$ and $^{238}\text{U} \rightarrow ^{239}\text{Pu}$, and time-dependent absorption values for several isotopes were computed for both. All the supporting codes (except DTF IV by

Brinkley and MDT by Lazarus) are in the Appendixes.

FAST ^{239}Pu BREEDER FLUX SPECTRUM EFFECTS

The complex function relating neutron cross sections and flux, for the absorption integrals required for breeding gain, is most easily inspected by adding to a pure metal reactor a simple and well-understood moderator such as graphite. This was done for a 50-cm core containing both the daughter and the fertile isotope, surrounded by a 20-cm blanket containing only the fertile isotope. Each mixture was computed with 0, 1/2, and 1 carbon atom per ^{238}U atom, and the results were compared for absorption distribution changes. In addition, other isotopes that will be created by (n,γ) and $(n,2n)$ processes were effectively included by calculation of absorption per prompt neutron assuming the same atomic concentration as the daughter (Pu fuel bred from ^{238}U) isotope. A true value may be established by the simple factor relating atomic density at a specified time with that of the daughter atom (this atomic density will be computed below).

Table I shows the effect of a moderator carbon on the neutron absorption, fuel and fertile atom fission (a part of the total absorption), critical mass, and breeding gain. Dilution of the metallic reactor material by a molecular compound, UC, also increases the fraction of neutrons escaping: $E = 1 - \sum A_i$, although a large breeder reactor will show this effect to a much smaller degree.

Table II compares the components of neutron capture for the three moderator concentration cases, compares the associated (neutron flux weighted) moderator concentration cases, and compares the associated (neutron flux weighted) average cross sections. Note that the absorption figures are absolute (normalized to one neutron) for ^{238}U and ^{239}Pu , but assume the ^{239}Pu concentration for the isotopes ^{240}Pu and ^{241}Pu . To obtain absolute values, multiply those tabulated by the actual ratio* with respect to ^{239}Pu . The sharp reduction in breeding gain as carbon atomic density was increased is seen to be largely due to a decrease in neutron (n,f)

* For example see Table VII.

reactions in ^{238}U . In addition, the parasitic captures in ^{239}Pu and ^{240}Pu are increased.

FAST BREEDER FUEL DISTRIBUTION EFFECTS

A breeder reactor slowly converts ^{238}U to ^{239}Pu by neutron capture followed by decay of the ^{239}U nucleus through several steps. If it is desirable and possible to reduce the chemical processing by breeding the critical ^{239}Pu fuel concentration in the outer regions of the reactor and then moving it radially inward to a neutronically more effective position, the radial density distribution of ^{239}Pu and ^{238}U can be approximated by the distributions given in Figs. 1 and 2. The total capture and fission rate distributions are shown in Figs. 3 and 4 for ^{238}U and in Figs. 5 and 6 for ^{239}Pu , and the integral neutron flux density distribution is shown in Fig. 7. The total fission (power) density distribution is shown in Fig. 8. These figures show that there is no remarkable change from the more familiar distribution found with a radially constant fuel density. However, as we see from Table III, the breeding gain has increased significantly. This change is partly due to a shift in flux spectrum and partly to a change in neutron escape.

Another easily computable effect of fuel redistribution is that resulting in a reasonably flat (constant) fission density (power distribution). Table III shows that while the critical mass and neutron escape are significantly increased, the breeding gain is still quite high. A rough approximation to a flat fission density distribution is shown in Fig. 9. The neutron absorption changes are shown on Table IV as being small. The largest change is that due to having a large amount of the fuel ^{239}Pu in a position of decreased importance. Figure 10 shows the radial fission density of the assembly with constant fuel loading for comparison.

Finally, the effect on neutron absorption distribution of having the fuel ^{239}Pu all in the center of the assembly, surrounded by a thick ^{238}U blanket, may be estimated with the integral results of Topsy (Appendix A). There, $A(^{239}\text{Pu}) = 0.37$; $A(^{238}\text{U}) = 0.46$; and breeding gain ≈ 0.3 , consistent with the relatively small assembly size. These values are changed to $A(^{239}\text{Pu}) = 0.29$, $A(^{238}\text{U}) = 0.70$, and

breeding gain ≈ 1.4 for a very thick blanket.

In summary of this section, a fuel distribution consistent with an equilibrium breeding of ^{239}Pu fuel is also consistent with low critical mass and very high breeding gain. Redistribution to effect a spatially constant fission density does not greatly reduce the breeding gain but complicates the simple disposition of bred fuel.

FAST ^{233}U BREEDER REACTOR

The characteristics of the ^{232}Th - ^{233}U fast breeder reactor can be most easily fixed by a comparison with an otherwise identical ^{238}U - ^{239}Pu breeder reactor. Table V compares these two assemblies for fuel distributed radially in the equilibrium manner, and Table VI shows the ^{233}U absorption distribution and averaged neutron cross sections (compare Table IV). It is interesting that the ^{233}U - ^{232}Th fast breeder reactor may have a breeding gain almost as high as the relatively popular and well-studied ^{239}Pu - ^{238}U breeder assembly (equivalent to U_2O_3). This is at the cost of a higher critical mass, because of the relatively small fission fraction due to ^{232}Th (10% as compared to 30% for ^{238}U), and because of the lower material density of ^{232}Th . A compensating effect is that the daughter product of the fuel, ^{234}U , has a much smaller absorption and larger fission cross section than the daughter product, ^{240}Pu , in the ^{239}Pu breeder assembly. However, as we will show, neither ^{234}U nor ^{240}Pu has a large effect on the criticality of fast reactors. C/U effect is shown by Fig. 11, which gives the net effect of carbon atom moderation on breeding gain. As the ^{239}Pu fast breeder becomes more thermal, the breeding gain approaches that of ^{233}U , which increases with moderation. The ^{233}U reactor must be evaluated for a still more thermal flux spectrum, which begins to include resonance absorption effects on neutron inventory and thus breeding gain. The superiority of the ^{239}Pu breeder reactor as compared with ^{233}U may be reduced by the use of the reactor components required for structure, heat transfer, and control of the kinetic constants. Hansen⁷ has pointed out that Fe isotopes reduce breeding gain in very fast reactors.

NEUTRON ABSORPTION INTEGRALS IN THE BREEDING CYCLE

If the fertile (parent) nuclei are exposed to the average flux in a fast reactor* for a total exposure, ϕt , the primary processes of fission, capture, and $(n,2n)$ reactions produce a wide variety of nuclei (see Chart of the Nuclides, Appendix D). The integration of these processes was computed over an exposure sufficient to breed the critical concentration of distributed daughter isotope. The relative amounts of certain of these nuclei for one initial ^{232}Th or ^{238}U atom are shown in Table VII. These values result from a time integration of

$$\Delta N_{i+1} = \phi \Delta t \left(\sum_i^c - \sum_{i+1}^a \right)$$

where $\sum^x = N_i \bar{\sigma}^x$ for parent (i) and daughter (i+1) isotopes, c is capture (n,γ), and a is total reaction cross section. The time dependence is shown in Figs. 12 to 14 for the ^{239}Pu cycle and the code is shown in Appendix H.

Note that the breeder reactor poisons, ^{240}Pu and ^{234}U for the two systems, are present in small concentrations -- about 13% of that of the daughter isotope for $\phi t = 0.8$. Their effects on critical mass may be estimated as follows. The central (normalized) reactivity of ^{239}Pu is 4.40, and that of ^{240}Pu is -0.32. With the ^{239}Pu reactivity coefficient $\Delta k/(\Delta N/N) \approx 0.73$, 13% of ^{240}Pu would reduce k_{eff} by $\Delta k \approx 1\%$, a small amount. A smaller effect is expected (see $\bar{\sigma}$, Table VI) for ^{234}U in the ^{233}U breeder reactor, since $\Delta k/(\Delta N/N) \approx 0.6$ and the capture cross section is relatively small. The reactivity distributions of ^{239}Pu and ^{238}U are shown in Fig. 15 to be simple functions, with a reduced integral effect on k_{eff} in the special geometry using inward radial motion of the bred fuel.

Transuranic isotope cross sections that are very small in reactor flux spectra were not included. These are (n,p) , (n,α) , $(n,3n)$, etc. Isotopes formed by capture and $(n,2n)$ reactions but not included in the integral summaries are:

* We choose the flux $\phi(r,u)$ ΔE , in the equilibrium reactor using metallic fuel. The other assemblies have similar neutron flux spectra.

$^{231}\text{Th} \xrightarrow{^{25}\text{h}} ^{231}\text{Pa}, ^{233}\text{Th} \xrightarrow{^{22}\text{m}} ^{233}\text{Pa} \xrightarrow{^{27}\text{d}} ^{233}\text{U}, ^{232}\text{U},$

$^{236}\text{U}, ^{237}\text{U} \xrightarrow{^{7}\text{d}} ^{237}\text{Np}, ^{239}\text{U} \xrightarrow{^{23}\text{m}} ^{239}\text{Np}, ^{238}\text{Pu}, \text{etc.}$

RESUME OF CALCULATIONS

1. Master Data Tape cross sections suffice for parametric fast reactor studies. This has not been true in past studies using data not corrected by reactivity correlations. There should, however, be no implication that intermediate-to-thermal flux spectrum systems may be computed accurately, although the neutronics are somewhat simpler than those of fast reactors, or that no further work must be done on fast reactor cross sections.

2. $^{233}\text{U} - ^{232}\text{Th}$ fast reactors have a breeding gain of 0.2 to 0.3 as compared to 0.2 to 1.3 for the $^{239}\text{Pu} - ^{238}\text{U}$ fast reactors.

3. Breeding gain of the $^{239}\text{Pu} - ^{238}\text{U}$ reactor is sharply reduced when a moderator such as a carbide (or, of course, an oxide) is used to improve the physical properties of the reactor fuel and fertile materials. Though not discussed here, relatively large amounts of the heavier isotopes, such as Na (coolant) and some structural materials, may be added without serious loss in the excellent breeding gain figures if inelastic scattering does not seriously affect the flux spectrum. Also, in connection with reactor materials effects, addition of a void fraction (without the introduction of large

streaming channels), increases critical radius by the simple factor by which neutron mean free path is increased.

4. Critical concentration of the bred fuel isotope is low in large assemblies, particularly the average value for the distribution of fuel that is increased in concentration toward the reactor center. From this, it is conceptually possible to reduce or eliminate associated fuel enrichment and chemical processing steps. The fuel doubling time in the ^{239}Pu breeder may be as low as three to four years if the power level of the assembly is about 1000 MW.

5. The effect on criticality of transuranic isotopes formed by (n, γ) and (n,2n) reactions is small.

6. There must be a large increase in the variety and detail of reactor experiments useful in the detailed analysis of fast breeder reactor neutronics if the highly developed computing schemes are to make a contribution to fast breeder reactor development.

ACKNOWLEDGMENTS

In addition to making many theoretical contributions, G. I. Bell and Gordon Hansen have reviewed this article and have made a number of valuable suggestions and corrections.

Table I. Moderator Effects on Fast ^{239}Pu Breeder Reactors^a

Fertile Material	Absorption ^b		Fission		Fission Neutrons		Critical ^c Mass (kg) ^{239}Pu	Ave. ^{239}Pu in ^{238}U (%)	Breeding Gain
	^{239}Pu	^{238}U	^{239}Pu	^{238}U	^{239}Pu	^{238}U			
^{238}U	0.302	0.675	0.249	0.1005	0.723	0.281	560	2.1	1.24
$^{238}\text{UCl}_2$	0.307	0.647	0.252	0.0960	0.730	0.268	500	2.4	1.11
^{238}UC	0.333	0.608	0.270	0.0879	0.762	0.238	470	3.1	0.82

^aSpherical; 50-cm core, 70-cm reflector boundary radii.^bEscape = $1 - [A(^{239}\text{Pu}) + A(^{238}\text{U})]$.^cThe total ^{238}U mass was 27,000 kg.Table II. ^{239}Pu Fast Spectrum Breeder Reactors

	Absorption								
	^{238}U			$^{238}\text{UCl}_2$			^{238}UC		
	A_c	A_f	$A_{n,2n}$	A_c	A_f	$A_{n,2n}$	A_c	A_f	$A_{n,2n}$
^{238}U	0.579	0.1005	0.00507	0.556	0.0960	0.00463	0.524	0.0879	0.00391
^{239}Pu	0.527	0.2491	0.00011	0.0561	0.252	0.00011	0.0633	0.2697	0.00012
^{240}Pu	0.1763	0.0395	0.00005	0.1898	0.0394	0.00005	0.2170	0.0426	0.00005
^{241}Pu	0.0593	0.362	0.00016	0.0608	0.374	0.00016	0.0660	0.411	0.00017

	Average Cross Sections ^a								
	^{238}U			$^{238}\text{UCl}_2$			^{238}UC		
	σ_c	σ_f	$\sigma_{n,2n}$	σ_c	σ_f	$\sigma_{n,2n}$	σ_c	σ_f	$\sigma_{n,2n}$
^{238}U	0.2097	0.0364	0.00184	0.223	0.0384	0.00185	0.2474	0.0415	0.00184
^{239}Pu	0.385	1.816	0.00082	0.411	1.842	0.00083	0.441	1.877	0.00086
^{240}Pu	1.285	0.2878	0.00036	1.390	0.288	0.00036	1.510	0.2967	0.00036
^{241}Pu	0.432	2.637	0.00118	0.445	2.736	0.00119	0.459	2.859	0.00121

^aThese are equivalent one-velocity cross sections.

Table III. ^{239}Pu Distribution^a Effects in Fast Breeder Reactors

Condition	Absorption ^b		Fission		Fission Neutrons		Critical ^c Mass (kg) ^{239}Pu	Ave. ^{239}Pu in ^{238}U (%)	Breed- ing Gain
	^{239}Pu	^{238}U	^{239}Pu	^{238}U	^{239}Pu	^{238}U			
Equilibrium ^{239}Pu Distribution	0.301	0.684	0.249	0.1006	0.722	0.281	485	1.8	1.27
Flat Power Density (0 to 50 cm; re- flector to 70 cm)	0.301	0.659	0.249	0.1002	0.723	0.280	754	2.8	1.19

^aSpherical, seven regions with radii 10, 20, ... 70 cm.

^bEscape = $1 - \sum_i A_i$.

^cThe total ^{238}U mass is 27000 kg. Atomic density of ^{239}Pu is that resulting from a radial flow of this bred isotope.

Table IV. ^{239}Pu Fast Breeder Reactor

Equilibrium			Flat Power				
	<u>A_c</u>	<u>A_f</u>	<u>A_{n,2n}</u>		<u>A_c</u>	<u>A_f</u>	<u>A_{n,2n}</u>
²³⁸ U	0.589	0.1006	0.00509	²³⁸ U	0.565	0.1002	0.00506
²³⁹ Pu	0.0528	0.2487	0.00011	²³⁹ Pu	0.0524	0.2489	0.00012
²⁴⁰ Pu	0.1766	0.0391	0.00005	²⁴⁰ Pu	0.1752	0.0401	0.00005
²⁴¹ Pu	0.0594	0.362	0.00016	²⁴¹ Pu	0.0590	0.361	0.00016

Neutron Cross Section							
	σ_c	σ_f	$\sigma_{n,2n}$		σ_c	σ_f	$\sigma_{n,2n}$
^{238}U	0.2105	0.0360	0.00182	^{238}U	0.208	0.0370	0.00187
^{239}Pu	0.386	1.817	0.00081	^{239}Pu	0.383	1.816	0.00084
^{240}Pu	1.290	0.2854	0.00036	^{240}Pu	1.278	0.293	0.00037
^{241}Pu	0.434	2.641	0.00116	^{241}Pu	0.430	2.630	0.00120

Table V. Fast Breeder Reactors

²³⁹ Pu Breeder Integrals			
	<u>Captures (total)</u>	<u>Fission Neutrons</u>	<u>Fission Captures</u>
²³⁹ Pu	0.301	0.722	0.249
²³⁸ U	0.684	0.281	0.1006

$$\text{Breeding Gain} = \frac{(n, \gamma) \text{ captures in } ^{238}\text{U}}{\text{Captures in } ^{239}\text{Pu}} - 1 = 1.27.$$

Critical mass $\approx 2\%$ of the total metal weight.

²³³ U Breeder Integrals ^a			
	<u>Captures (total)</u>	<u>Fission Neutrons</u>	<u>Fission Captures</u>
²³³ U	0.404	0.947	0.369
²³² Th	0.582	0.057	0.027

$$\text{Breeding Gain} = \frac{(n, \gamma) \text{ captures in } ^{232}\text{Th}}{\text{Captures in } ^{239}\text{Pu}} - 1 = 0.44.$$

Critical mass $\approx 3\%$ of the total metal weight.

^aNeutron escape E $\approx 1.5\%$.

Table VI. ²³³U Breeder Integrals^a

Absorptions				Average Neutron Cross Sections			
	<u>A_f</u>	<u>A_c</u>	<u>A_{n,2n}</u>		<u>σ_f</u>	<u>σ_c</u>	<u>σ_{n,2n}</u>
²³² Th	0.0241	0.4000	0.00349	²³² Th	0.0121	0.2408	0.00175
²³³ U	0.3967	0.031155	0.0000172	²³³ U	2.152	0.1712	0.00093
²³⁴ U	0.1010	0.03932	0.000061	²³⁴ U	0.548	0.2163	0.000330
²³⁵ U	0.2575	0.05462	0.000261	²³⁵ U	1.397	0.02963	0.00142

^a100-cm radius, 80-cm core, metal system.

Table VII

Relative Isotope Concentration in Breeder Reactors

	N^a	Compare	N^a	
^{238}U	0.822		0.756	^{232}Th
^{239}Pu	0.070		0.095	^{233}U
^{240}Pu	0.0092		0.0133	^{234}U
^{241}Pu	0.0035		0.00063	^{235}U
$^{238}\text{U}(n,2n)$	0.0013		0.00104	$^{232}\text{Th}(n,2n)$
$^{239}\text{Pu}(n,2n)$	0.00003		0.00001	$^{233}\text{U}(n,2n)$
$^{240}\text{Pu}(n,2n)$	0.000001		0.000001	$^{234}\text{U}(n,2n)$
$^{241}\text{Pu}(n,2n)$	0.000001		0.0000002	$^{235}\text{U}(n,2n)$
Total Fissions	0.094		0.1365	Total Fissions
ϕt	0.79		0.79	ϕt

^aAt $\phi t = 0$, $N(^{238}\text{U}) = N(^{232}\text{Th}) = 1$.

Fig. 1. ^{239}Pu distribution in the ^{239}Pu fast breeder.

Fig. 2. ^{238}U distribution in the ^{239}Pu fast breeder.

Fig. 3. Total (n,γ) capture vs radius in ^{238}U breeder.

Fig. 4. Total (n,f) fission vs radius in ^{238}U breeder.

Fig. 5. Total (n,γ) capture vs radius in ^{239}Pu breeder.

Fig. 6. Total (n,f) fission vs radius in ^{239}Pu breeder.

Fig. 7. Average neutron flux vs radius in ^{238}U reactor.

Fig. 8. Fission density vs radius for fuel moved to smaller radii.

Fig. 9. "Flat" fission density vs radius for ^{239}Pu fast breeder.

Fig. 10. Fission density vs radius for constant ^{239}Pu density.

Fig. 11. Breeding gain vs. carbon/parent atomic density ratio.

Fig. 12. Atoms of ^{238}U , ^{239}U and ^{240}Pu vs. ϕ for 1 initial ^{238}U atom.

Fig. 13. Atoms of ^{241}Pu and total fissions vs $\overline{\phi t}$ for 1 initial ^{238}U atom.

Fig. 14. Relative ratio of daughter to fertile atom density vs $\overline{\phi t}$.

Fig. 15. Reactivity distribution of ^{239}Pu and ^{238}U in a fast breeder reactor.

APPENDIX A. FISSION SPECTRUM EFFECTS

The fast experiments Godiva, Jemima (37.5% ²³³U), Jemima (16.25% ²³⁵U), and Jezebel were computed¹⁻³ using the P-2, S₄ (and S₈ transport code with fission spectra in the 24 groups (scattering into the 23 lower energy groups). The spectra were made specific to the critical assembly with the help of a simple multigroup diffusion code which produced a fission spectrum weighted by fission absorption in the specific sources. The data extracted from Terrell's⁴ and Bernard's⁸ papers are:

²³³ U	$\bar{\nu} = 2.47 + 0.149 E$
²³⁵ U	$\bar{\nu} = 2.35 + 0.160 E$
²³⁸ U	$\bar{\nu} = 2.313 + 0.154 E$
²³⁹ Pu	$\bar{\nu} = 2.94 + 0.129 E$
²⁴⁰ Pu*	$\bar{\nu} = 2.19 + 0.1 E$

Nuclear temperatures are given by Bernard et al.⁸

Isotope	Temp. (MeV)	(dT/dE)
²³³ U	1.36	0.02
²³⁵ U	1.30	0.005
²³⁸ U	1.29	0.016
²³⁹ Pu	1.39	0.0135
²⁴⁰ Pu	1.19	0.01

with E in MeV. $\bar{\nu}$ is related to $\bar{E}$ by

$$\bar{E} = 0.75 + 0.65 (\bar{\nu} + 1)^{\frac{1}{2}} \text{ (MeV)}$$

with a postulated⁴ increase in $\bar{E}$ of about 0.025 MeV per MeV of incident neutron energy. Further, according to Terrell, the prompt neutron spectrum can be well represented by

$$N(E) = (4E/\pi T^3)^{\frac{1}{2}} \exp(-E/T)$$

where $T = 2/3 \bar{E}$. The above data are in good agreement with independent work of Bernard⁸ et al. at AERE.

Fission neutron spectrum effects were largest

with Jemima (16.25% ²³⁵U). That determined using $\bar{\nu}$ from fission weighting in ²³⁵U and ²³⁸U was $\Delta k = 0.003$ as compared to a simple spectrum (now used by Hansen) with $T = 1.30$ MeV. The effect of adding the $T = a + bE$ term, with $b = 0.025/\text{MeV}$ was an additional $\Delta k = 0.0015$. Accordingly, prompt neutron spectrum effects are small, but they may be significant in some reactors.

Computer Code SIGNU uses reactor fuel nuclear temperature and prompt neutron production as a function of energy to generate weighting functions for neutron cross-section averaging, an averaged prompt neutron energy distribution, and a multigroup velocity distribution. The formats are consistent with MDT and DTF codes.

A digest of the code below to assist in its use is as follows:

1. Read wide group bounds E_1 (eV) for $10 < E < 0.017$ MeV.
2. Read lethargy limits for $0 < u < 20$.
3. Read isotope data for nuclear temperature (2 isotopes), dT/dE , ν (neutrons/fission), and $d\nu/dE$.
4. Compute wide group bounds as $1/4$ of input data and compute fission source for first nuclear temperature, T .
5. Read description of reactor, and convert neutron cross sections from three-table to one-table (transport approximation).
6. Compute flux, using fission source from T and reactor data, with a simple "bare reactor" code.
7. Use flux output to reiterate a nuclear temperature, using isotope T and dT/dE , ν and $d\nu/dE$, and then punch flux weighting function on cards for MDT code use, and the iterated prompt fission neutron spectrum for DTF-IV use.

*Data do not seem to exist for ²⁴⁰Pu $\nu(E)$. Spontaneous⁷ fission $T = 1.19$ and $\bar{\nu} = 2.19$.

k_{eff} from S_8 , P_2 Calculations

(Atomic Density, 10^{24} atoms/ml)

Assembly ^a (% ^{235}U)	^{235}U	^{238}U	^{239}Pu	^{240}Pu	Sphere Radius (cm)	k_{eff}
Jemima (16 $\frac{1}{2}$) ^b	0.00777	0.03971	-	-	20.32	0.9951
Jemima (37 $\frac{1}{2}$)	0.01814	0.02934	-	-	14.57	0.9900
Godiva (93.2)	0.04556	0.00245	-	-	8.71	0.9975
Jezabel (95 $\frac{1}{2}$ ^{239}Pu)	-	-	0.03752	0.001924	6.285	0.9987
Topsy (94.1) ^c	0.04507	0.002825	-	-	6.105	0.9987

^a S_8 , 20-space point because of flux anisotropy (very small radii).

^bReflector tuballoy, radius 27.94 cm.

^c S_8 , 10-space point core, 20-space point reflector (U metal) to 29.0 cm. Note that k_{eff} (total) = $k(\text{core}) + k(\text{reflector}) = 0.81 + 0.19$ in Topsy.

```

PROGRAM SIGNU (INPUT,TAPE 10=INPUT, OUTPUT,TAPE 9=OUTPUT,FILM,
1PUNCH)
C
C   DESIGNED TO COMPUTE AND PUNCH WEIGHTING FUNCTION FOR M.D.T.
C
  DIMENSION U(500),E(500),F0(500),SF0(500),Z(500),E2(210),SZ(210),
1S( 30,28,10),ANEL(10),ATWT(10),CAP(210,28),COL1(210),
2COL2(210),FLUX(210),OFLUX(210),COL7(210),COL10(210),COL12(210),
3COL17(210),ANU1(210),T1(210),ANU2(210),T2(210),F1(210),F2(210),
4,TFLUX(210),COL3(210)
  DIMENSION E1(50),U1(50),RTE(500),WTE(500),S1(30,28),S2(30,28)
1.ELEM(12),U2(210),A(12)
  DIMENSION SF1(500),Z1(500),SZ1(500),F(210),ZIN(210)
  DIMENSION AVT5(210),AVT8(210),F5(210),F8(210),Z58(210),V(100)
  DIMENSION W(100)
100 FORMAT(6F12.6)
101 FORMAT(1X,9F14.4)
102 FORMAT(3I5,10F11.4 )
103 FORMAT(6F12.6)
104 FORMAT(1X,10(I4,F12.6)
105 FORMAT(15,6(1X,F10.1))
106 FORMAT(10F12.4)
107 FORMAT(12A6)
108 FORMAT(1H0)
109 FORMAT(1X,8F16.6)
110 FORMAT(1H1)
111 FORMAT(1X,9F10.7)
112 FORMAT(1X,9F14.6)
113 FORMAT(6F12.4)
114 FORMAT(6(3X,F9.5))
115 FORMAT(1X,15,1P6E12.5)
  WRITE(9,110)
C   INITIALIZE   LOOPS
  WTE(1)=1.0
  SF0(1)=0.0
  E(1)=1.0E+07
  E2(1)=F(1)
  DU=0.02
  U(1)=0.0
  AVNU1=0.0
  AVNU2=0.0
  U2(1)=0.0
  WRITE(9,108)
C   COMPUTE U,E,FISSIONS
C
C   FISSION SPECTRUM,T=
  READ(10,100) (E1(I),I=2,7)
  WRITE(9,101) (E1(I),I=2,7)
  WRITE(9,108)
  READ(10,100) (U2(I),I=2,19)
  WRITE(9,101) (U2(I),I=2,19)
  WRITE(9,108)
  READ(10,100) T,T0,T8,DTDE0,DTDE8,ANU0,ANU8,DNU0,DNU8
  WRITE(9,101) T,T0,T8,DTDE0,DTDE8,ANU0,ANU8,DNU0,DNU8
  F0(1)=0.0
  F1(1)=0.0
  U1(1)=0.0
  SF0(1)=0.0
  SF1(1)=0.0
  SZ(1)=0.0
  SZ1(1)=0.0

```

```

U(1)=0.0
E(1)=1.0E+7
M=2
K=2
N1=2
DO 1 I=2.7
  U1(I)=ALOG(10.0**7/E1(I))
  DU=(U1(I)-U1(I-1))/4.0
  DO 1 J=1.4
    DSV=0.0
    SV=0.0
    DO 2 L=1.10
      U(K)=U(K-1)+DU/10.0
      E(K)=10.0**7*EXP(-U(K))
      RTE(K)=SQRT(E(K))
      AVEE=0.5*(E(K)+E(K-1))
      F0(K)=SQRT(AVEE)*EXP(-AVEE/T)
      F1(K)=SQRT(AVEE)*EXP(-AVEE/T8)
      SF0(K)=SF0(K-1)+F0(K)*(E(K-1)-E(K))
      SF1(K)=SF1(K-1)+F1(K)*(E(K-1)-E(K))
      VE=SQRT(1.9132*AVEE)*10.0**-2
      SV=SV+VE*SF0(K)
      DSV=DSV+SF0(K)
    2 K=K+1
    V(M)=SV/DSV
    E2(M)=E(K-1)
    Z(M)=SF0(K-1)-SZ(M-1)
    Z1(M)=SF1(K-1)-SZ1(M-1)
    SZ(M)=Z(M)+SZ(M-1)
    SZ1(M)=Z1(M)+SZ1(M-1)
    COL3(N1)=(U(K-1)-U(K-11))/2.0
    N1=N1+1
  1 M=M+1
  M=K-1
  WRITE(9,108)
  PRINT101,(E2(I),I=2,25)
  WRITE(9,108)
  PRINT 101,(COL3(I),I=2,25)
  WRITE(9,108)
  PRINT 106,( E(I),I=1,M,10)
  WRITE(9,108)
  PRINT 106,(U(I),I=11,M,10)
  WRITE(9,108)
  K=2
  DO 3 I=2.25
    Z(I)=Z(I)/SF0(M)
    Z1(I)=Z1(I)/SF1(M)
    SZ(I)=SZ(I)/SF0(M)
    SZ1(I)=SZ1(I)/SF1(M)
    DO 3 J=1.10
      WTE(K)=SF0(K)/SF0(M)
    3 K=K+1
  WRITE(9,108)
  DO 43 I=1.25
43 E2(I)=E2(I)*10.0**(-6)
C      SPECIFIC ISOTOPE FISSION DATA:
DO 11 I=2.25
  EAVE=(E2(I)+E2(I-1))*0.5
  ANU1(I)=ANU0+DNU0*EAVE
  ANU2(I)=ANU8+DNU8*EAVE
  T1(I)=T0*10.0**(-6)+DTDE0*EAVE
  T2(I)=T8*10.0**(-6)+DTDE8*EAVE

```

```

11 CONTINUE
C
PRINT T(E) FROM T+0.025 E
PRINT 101. (T1(I), I=2,25)
WRITE(9,108)
PRINT 101. (ANU1(I), I=2,25)
WRITE(9,108)
PRINT 101. (T2(I), I=2,25)
WRITE(9,108)
PRINT 101. (ANU2(I), I=2,25)
WRITE(9,108)
PRINT 101. (Z(I), I=2,25)
WRITE(9,108)
C
READ(10,102) NOEL, IGEOM, IGROUP, RADIUS
WRITE(9,102) NOEL, IGEOM, IGROUP, RADIUS
WRITE(9,108)
READ(10,100) (ANEL(I), I=1, NOEL)
WRITE(9,109) (ANEL(I), I=1, NOEL)
WRITE(9,108)
READ(10,100) (ATWT(I), I=1, NOEL)
WRITE(9,109) (ATWT(I), I=1, NOEL)
C
MICROSCOPIC CROSS SECTIONS.
DO 42 J=1,27
DO 42 I=2,30
DO 41 K=1, NOEL
S(I,J,K)=0.0
41 CONTINUE
S1(I,J)=0.0
S2(I,J)=0.0
42 CONTINUE
DO 7 K=1, NOEL
READ(10,107) (ELEM(I), I=1,12)
WRITE(9,107) (ELEM(I), I=1,12)
READ(10,100) ((S(I,J,K), J=1,27), I=2,25)
READ(10,107) (ELEM(I), I=1,12)
READ(10,100) ((S1(I,J), J=1,27), I=2,25)
READ(10,107) (ELEM(I), I=1,12)
READ(10,100) ((S2(I,J), J=1,27), I=2,25)
C
CONVERT TO TRANSPORT APPROX.
DO 40 I=2,25
SDEL=S1(I,4)+S1(I+1,5)+S1(I+2,6)
S(I,3,K)=S(I,3,K)-SDEL
S(I,4,K)=S(I,4,K)-SDEL
40 CONTINUE
7 CONTINUE
WRITE(9,108)
C
MACROSCOPIC
DO 8 I=2,25
DO 9 L=1,27
9 CAP(I,L)=0.0
DO 8 K=1, NOEL
DO 8 J=1,27
8 CAP(I,J)=CAP(I,J)+S(I,J,K)*ANEL(K)
C
SPHERICAL SHAPE ASSUMED.
RSQ=(3.1416/RADIUS)**2
TFIN=T
C
INITIALIZE
L=1
15 ESC=0.0
PRINT 101. TFIN

```

26

```

      ST1=WT1*T1(I) +ST1
      ST2=WT2*T2(I) +ST2
      SWT1=SWT1+WT1
      SWT2=SWT2+WT2
      SWT=SWT+WT1+WT2
210  CONTINUE
      T0=ST1/SWT1
      T8=ST2/SWT2
      T=(ST1+ST2)/(SWT1+SWT2)
      PRINT 101, T0,T8,T
C
C                                     REAVERAGE Z(I)
      FLUX(1)=0.0
      E2(1)=10.
      SF12=0.0
      DO 12 I=2,25
      F1(I)=0.0
      DE=(E2(I-1)-E2(I))*0.1
      DO 212 J=1,10
      AJ=J
      EAVE=E2(I-1)-AJ*DE
      F1(I)=F1(I)+DE
      1 *SQRT(EAVE/(T **3))*EXP(-EAVE/T)
212  CONTINUE
      SF12=SF12+F1(I)
      12 CONTINUE
C
C                                     FLUX WEIGHTED Z(I)
      ZTOT=0.0
      DO 13 I=2,25
      Z(I)=F1(I)/SF12
      ZTOT=ZTOT+Z(I)
      13 CONTINUE
      PRINT 103,ZTOT
C
      TFIN=T+T8
C
      IF(L-5) 63,63.64
63  L=L+1
      IF(L-4) 15,15.60
C
C      FLUX IN FINE LOOP
C
C      DELAYED NEUTRONS AT 0.4 MEV. AND SUM IS 0.01.
C
60  IF(L-5) 15,61.62
61  Z(16)=Z(13)-0.01
      GO TO 15
62  Z(16)=Z(16)
      GO TO 15
64  FLUX(1)=0.0
      I=2
      N=2
      DO 18 N0=2,25
      DO 19 N1=1,10
      FLUX(N)=FLUX(N-1)+DFLUX(I)
      DE=E(N-1)-E(N)
      EAVE=(E(N)+E(N-1))/2.0/(10.**6)
      WTE(N)=FLUX(N)*10.**3
      19 N=N+1
      18 I=I+1
      WRITE(9,108)

```

```

AIG=1.0
NAIG=1
IP=25
49 L=1
WRITE(9,108)
READ(10,107) (A(I),I=1,12)
WRITE(9,107) (A(I),I=1,12)
ILO=24/NAIG
INN=NAIG
DO 4 K=1,ILO
M=K
DO 5 I=1,INN
PRINT 105,M,E(L),WTE(L),E(L+1),WTE(L+1),E(L+2),WTE(L+2)
PUNCH 105,M,E(L),WTE(L),E(L+1),WTE(L+1),E(L+2),WTE(L+2)
PRINT 105,M,E(L+3),WTE(L+3),E(L+4),WTE(L+4),E(L+5),WTE(L+5)
PUNCH 105,M,E(L+3),WTE(L+3),E(L+4),WTE(L+4),E(L+5),WTE(L+5)
PRINT 105,M,E(L+6),WTE(L+6),E(L+7),WTE(L+7),E(L+8),WTE(L+8)
PUNCH 105,M,E(L+6),WTE(L+6),E(L+7),WTE(L+7),E(L+8),WTE(L+8)
PRINT 105,M,E(L+9),WTE(L+9),E(L+10),WTE(L+10)
PUNCH 105,M,E(L+9),WTE(L+9),E(L+10),WTE(L+10)
5 L=L+10
4 CONTINUE
K=2
U(1)=U2(5)
V(1)=10.**7*EXP(-U(1))
READ(10,107) (A(I),I=1,12)
WRITE(9,107) (A(I),I=1,12)
DO 46 I=6,12
DU=(U2(I-1)-U2(I))/4.0*AIG*(-1.0)
IG=4/NAIG
DO 46 J=1,IG
U(K)=U(K-1)+DU
V(K)=10.**7*EXP(-U(K))
46 K=K+1
M=1
ILP=16/NAIG+1
ILO=40/NAIG
DO 90 K=ILP,ILO
L=K-16/NAIG
PRINT 115,M,V(L)
PRINT 115,M,V(L+1)
PUNCH 115,M,V(L)
PUNCH 115,M,V(L+1)
M=M+1
90 CONTINUE
K=2
U(1)=U2(9)
V(1)=10.**7*EXP(-U(1))
READ(10,107) (A(I),I=1,12)
DO 47 I=10,15
DU=(U2(I-1)-U2(I))/4.0*AIG*(-1.0)
IG=4/NAIG
DO 47 J=1,IG
U(K)=U(K-1)+DU
V(K)=10.**7*EXP(-U(K))
47 K=K+1
M=1
ILP=32/NAIG+1
ILO=56/NAIG
DO 91 K=ILP,ILO
L=K-32/NAIG
PRINT 115,M,V(L)

```



```

PRINT 115.M.V(L+1)
PUNCH 115.M.V(L)
PUNCH 115.M.V(L+1)
M=M+1
91 CONTINUE
K=2
U(1)=U2(13)
V(1)=10.**7*EXP(-U(1))
READ(10,107) (A(I),I=1,12)
WRITE(9,107) (A(I),I=1,12)
DO 48 I=14,19
DU=(U2(I-1)-U2(I))/4.0*AI*G*(-1.0)
IG=4/NAIG
DO 48 J=1,IG
U(K)=U(K-1)+DU
V(K)=10.**7*EXP(-U(K))
48 K=K+1
M=1
IT=72/NAIG+1
V(IT)=0.0
ILP=48/NAIG+1
ILQ=72/NAIG
DO 92 K=ILP,ILQ
L=K-48/NAIG
PRINT 115.M.V(L)
PRINT 115.M.V(L+1)
PUNCH 115.M.V(L)
PUNCH 115.M.V(L+1)
M=M+1
92 CONTINUE
DO 80 I=2,19
DU=(U2(I-1)-U2(I))/4.0*AI*G*(-1.0)
IG=4/NAIG
DO 81 J=1,IG
W(M)=W(M-1)+DU
AVEE=10.**7*EXP((W(M)+W(M-1))*0.5*(-1.0))
V(M)=SQRT(1.9132*AVEE)*10.**(-2)
81 M=M+1
80 CONTINUE
PRINT101.(V(I),I=2,IT)
PUNCH 100.(V(I),I=2,IT)
AI*G=2*NAIG
NAIG=2*NAIG
IG=1+24/NAIG
IF(NAIG-4) 200,200,201
200 GO TO 49
201 STOP
STOP
END

```

APPENDIX B. TWENTY-FOUR GROUP-AVERAGED
NEUTRON CROSS SECTIONS

Sources:

1. Master Data Tape (Roger Lazarus, LASL T-7).

2. The flux spectrum in Godiva as shown by the code in Appendix A was used for weighting.

3. Energy boundary groups (MeV).

* The (n,c) reaction in the cross-section table is that for the (n, γ) reaction.

i	10.0	3.0	1.4	0.9	0.4	0.1
i+1	7.401	2.480	1.254	0.7348	0.2828	0.04621
i+2	5.477	2.049	1.122	0.6000	0.2000	0.04123
i+3	4.054	1.694	1.005	0.4899	0.1414	0.02648
						0.0170

i = 1, 5, 9, 11, 17, 21 in row 1.

The following data are group-averaged: (n, γ)*, (n,f) and (n,2n) neutron cross sections for ²³²Th, ²³³U, ²³⁴U, ²³⁵U, ²³⁸U, ²³⁹Pu, ²⁴⁰Pu, and ²⁴¹Pu.

TH 232	(N,C)	(N,F)	(N,2N)	U 233	(N,C)	(N,F)	(N,2N)
1	.00638	.40400	1.12420	0.	2.27500	.55980	
2	.00944	.24740	.06319	0.	1.87400	.02797	
3	.01578	.15280	0.	.00221	1.65500	0.	
4	.02658	.14860	0.	.07170	1.82400	0.	
5	.04019	.12410	0.	.03748	1.91300	0.	
6	.05713	.11750	0.	.01398	1.93700	0.	
7	.08003	.09520	0.	.02536	1.93700	0.	
8	.11272	.08460	0.	.04530	1.90400	0.	
9	.14684	.02800	0.	.06007	1.87500	0.	
10	.18740	.00376	0.	.06774	1.86300	0.	
11	.22840	.00118	0.	.07541	1.85500	0.	
12	.21010	.00007	0.	.08465	1.86500	0.	
13	.22470	0.	0.	.10099	1.91300	0.	
14	.23040	0.	0.	.12610	1.97400	0.	
15	.18750	0.	0.	.15040	2.02600	0.	
16	.18200	0.	0.	.16776	2.07600	0.	
17	.18650	0.	0.	.18334	2.15400	0.	
18	.20400	0.	0.	.20830	2.22700	0.	
19	.23850	0.	0.	.23260	2.26100	0.	
20	.28900	0.	0.	.25010	2.28600	0.	
21	.37080	0.	0.	.27710	2.43500	0.	
22	.49540	0.	0.	.31650	2.72500	0.	
23	.67250	0.	0.	.35940	3.02000	0.	
24	.86750	0.	0.	.39680	3.43900	0.	

U 234	(N,C)	(N,F)	(N,2N)	U 235	(N,C)	(N,F)	(N,2N)
1	0.	1.98700	.23060	0.	1.77740	.61930	
2	0.	1.54700	.00080	0.	1.31810	.10880	
3	0.	1.48600	0.	0.	1.23370	0.	
4	.00082	1.51000	0.	.00083	1.27980	0.	
5	.04147	1.52800	0.	.04145	1.31050	0.	
6	.05724	1.54200	0.	.05715	1.29670	0.	
7	.07093	1.49300	0.	.07095	1.25750	0.	
8	.08163	1.35800	0.	.08292	1.22480	0.	
9	.08892	1.26700	0.	.09055	1.20340	0.	
10	.09360	1.20900	0.	.09516	1.19020	0.	
11	.09770	1.15700	0.	.10000	1.17720	0.	
12	.10905	1.18500	0.	.10433	1.16560	0.	
13	.13600	1.21400	0.	.11137	1.15360	0.	
14	.16601	.91100	0.	.14181	1.15390	0.	
15	.19047	.65200	0.	.18724	1.15830	0.	
16	.21034	.44000	0.	.22920	1.18260	0.	
17	.23090	.22900	0.	.27920	1.23550	0.	
18	.25060	.10820	0.	.34780	1.31690	0.	
19	.26450	.00457	0.	.37770	1.44360	0.	
20	.27440	0.	0.	.40470	1.58700	0.	
21	.31310	0.	0.	.54720	1.89540	0.	
22	.39990	0.	0.	.84710	2.36480	0.	
23	.51090	0.	0.	1.02930	2.62880	0.	
24	.62560	0.	0.	1.28730	2.96330	0.	

U 238	(N,C)	(N,F)	(N,2N)	PU 239	(N,C)	(N,F)	(N,2N)
1	.00706	1.00200	1.24400	0.	2.37800	.30470	
2	.00884	.70890	.11780	0.	1.94900	.10330	
3	.01184	.54610	0.	0.	1.88800	0.	
4	.02006	.53990	0.	.00881	1.95100	0.	
5	.03031	.53560	0.	.03940	2.02300	0.	
6	.04381	.53280	0.	.05440	2.07000	0.	
7	.06207	.47980	0.	.07010	2.06200	0.	
8	.08738	.28900	0.	.08130	1.96500	0.	
9	.11660	.08837	0.	.08880	1.89900	0.	
10	.13140	.04640	0.	.09350	1.85700	0.	
11	.14620	.03904	0.	.09760	1.82000	0.	
12	.15410	.03238	0.	.10640	1.78500	0.	
13	.15140	.02435	0.	.12820	1.74000	0.	
14	.14700	.01460	0.	.15420	1.70000	0.	
15	.14210	.00766	0.	.18200	1.69200	0.	
16	.14000	.00037	0.	.20610	1.67600	0.	
17	.14000	0.	0.	.22940	1.65100	0.	
18	.14000	0.	0.	.27220	1.63000	0.	
19	.15210	0.	0.	.36810	1.60000	0.	
20	.18180	0.	0.	.45630	1.63000	0.	
21	.22510	0.	0.	.51940	1.77200	0.	
22	.29110	0.	0.	.59150	1.96400	0.	
23	.36920	0.	0.	.69170	2.13400	0.	
24	.46620	0.	0.	.84610	2.23100	0.	

PU 240	(N,C)	(N,F)	(N,2N)	PU 241	(N,C)	(N,F)	(N,2N)
1	0.	2.11000	.28470	0.	2.10700	.54330	
2	0.	1.71900	.00113	0.	1.70400	.11570	
3	0.	1.51900	0.	0.	1.69200	0.	
4	.01713	1.55600	0.	.00083	1.77000	0.	
5	.05376	1.63600	0.	.03940	1.80000	0.	
6	.07383	1.62300	0.	.05460	1.80000	0.	
7	.08991	1.55100	0.	.07060	1.78700	0.	
8	.10340	1.49800	0.	.08040	1.75400	0.	
9	.11260	1.46300	0.	.08770	1.72200	0.	
10	.11850	1.42400	0.	.09300	1.68900	0.	
11	.12390	1.36900	0.	.09750	1.66300	0.	
12	.12960	1.28200	0.	.10800	1.65200	0.	
13	.13810	1.11300	0.	.13730	1.65700	0.	
14	.14300	.81700	0.	.16780	1.65700	0.	
15	.14380	.53200	0.	.19000	1.65200	0.	
16	.31470	.33100	0.	.22670	1.68700	0.	
17	.75210	.15440	0.	.30090	1.77400	0.	
18	1.21350	.05220	0.	.41630	1.95400	0.	
19	1.34520	.02653	0.	.47940	2.31600	0.	
20	1.43750	.01417	0.	.52420	2.62000	0.	
21	1.64400	.00993	0.	.57710	3.05900	0.	
22	2.04100	.00994	0.	.63960	3.68700	0.	
23	2.53400	.00995	0.	.68360	4.11100	0.	
24	3.14700	.00996	0.	.72380	4.49300	0.	

APPENDIX C. THE ^{233}U CRITICAL ASSEMBLYAtomic Density ($\times 10^{24}$ atoms/ml)

^{233}U	^{234}U	^{235}U	Sphere Radius (cm)	k_{eff}^*
0.04678	0.00059	0.00029	5.9647	1.0109

20-space point S_8 (P-2; three-table) calculation.

APPENDIX D. CHART OF THE NUCLIDES

[illegible]

APPENDIX E. REACTIVITY IN FAST ASSEMBLIES

Computed Central Reactivity (Cents/Mole)*

Assembly	²³³ U	Godiva	Jezebel	Topsy
Isotope				
²³² th	- 6.9	- 9.12	-70.1	-16.8
²³³ U	850.0	257.9	1189.0	355.0
²³⁵ U	518.0	148.2	700.0	202.9
²³⁸ U	126.4	25.89	92.9	29.56
²³⁹ Pu	990.0	289.2	1391.0	394.8
²⁴⁰ Pu	537.0	135.0	748.0	168.2
D	71.6	26.3	- 3.99	30.7
C	6.73	2.13	- 6.21	2.47
Delayed Neutron				
Value (Cents)	0.0029	0.0059	0.00187	0.0062
Fraction	0.0029	0.0065	0.0021	0.0065
ΔR (Sample;cm)	0.298	0.436	0.314	0.611

Experimental⁶ Central Reactivity (Cents/Mole)

Assembly	²³⁸ U ⁷	Godiva	Jezebel	Topsy
Isotope				
²³² th	-	- 1.4	-64.7	- 7.6
²³³ U	955.0	-	1359.0	358.0
²³⁵ U	-	149.3	804.0	208.0
²³⁸ U	135.0	24.3	114.0	26.7
²³⁹ Pu	-	285.2	1592.0	402.6
²⁴⁰ Pu	-	170.0	1038.0	286.0
D	70.0	17.8	- 5.3	24.0
C	3.0	2.4	- 6.9	2.4
Delayed Neutron				
Value (Cents)	0.0029	0.0066	0.00194	

The code (ACT 24) used for the perturbation calculations first makes the transport correction on the 24-group cross sections to provide one table from the three-table set. Then the reactivity is computed⁶ in the usual fashion as

$$\frac{\Delta k}{k} = \frac{\sum_i \phi_i^+ \left[\sum_j (\nu \sigma_{fj} X_i + \sigma_{j \rightarrow i}) \phi_j - (\sigma_a + \sum_j \sigma_{i \rightarrow j}) \phi_i \right] \Delta N}{\sum_{\ell} \sum_i \sum_j \phi_i^+ \nu \sigma_{fj} X_i \phi_j \Delta V_{\ell} N(\text{fuel}) \Delta k_d}$$

where

$$\Delta V_{\ell} = \frac{4}{3} \pi (r_{\ell}^3 - r_{\ell-1}^3), \quad \Delta N = 0.6023 \times 10^{24},$$

and Δk_d = the value in k_{eff} of the delayed neutrons.

*Central void radius 0.709 for all calculations.

The radial distribution of the reactivity is illustrated in Fig. E1 for a simple bare homogeneous sphere (^{235}U in Godiva) and in Figs. E2 - E4 for a more complex case of a small core and a thick reflector (Topsy) (^{233}U , ^{235}U , and ^{239}Pu).

Disagreements shown on table may be either experimental* -- not a true perturbation -- or theoretical in the sense of inadequately studied cross sections. A more proper comparison would involve a complete reactor integral k_{eff} , which includes the effect of sample size or an extrapolation of the reactivity measurements to zero sample size. Both methods must result in agreement, given sufficient attention to detail in experiment and calculation, if the values of neutron cross sections, $\sigma_x(E, \theta, E' \rightarrow E)$, are correct. See Reference 6 for more detail.

*The ^{239}Pu measurement reflects a full 0.789 cm cavity, so (see Fig. F3) the central error is too large.

Fig. E1. Radial reactivity distribution of ^{235}U in Godiva.

Fig. E3. Radial reactivity distribution of ^{235}U in Topsy.

Trial calculations for Godiva gave good agreement between Δk from a complete reactor calculation and reactivity values.

This code is not adequate for scattering effects on reactivity except at the reactor center where neutron current values are zero. For a comparison of the radial distribution of reactivity with experiment for small scattering see Figs. E1 - E4.

The ACT 24 code computes $\Delta k/k$ in terms of cents/mole as follows:

1. Read radii, prompt fission neutron spectrum, reactor description, and forward and adjoint flux punch for the DTF-IV code.
2. Change cross sections from three-table to one-table, and reverse adjoint flux energy groups.
3. Compute details of above formula for Δk for core and reflector and add factors for atoms/mole and value of delayed neutrons (cents/ Δk).

Fig. E2. Radial reactivity distribution of ^{233}U in Topsy.

Fig. E4. Radial reactivity distribution of ^{239}Pu in Topsy.

```

PROGRAM ACT24 (INPUT,TAPE 10=INPUT, OUTPUT,TAPE 9=OUTPUT,FILM,
1PUNCH)
000003 DIMENSION PHI(30,40),PHIA(30,40),IB(20,20),ANU(30)
1.V(30),S(24,27,15),A(30,30),APU(30),AP(30,30),U(30),IC(20,20)
000003 DIMENSION C(12,20),ANE(24,40,4),ANEA(24,40,4),IIG(9),IGEO(9)
000003 DIMENSION FUEL(4), R(40),RANL(40,10),ICORE(9),IRTOT(9)
000003 DIMENSION PAPA(30,30),PAPS(30,30),S1(24,27),S2(24,27)
000003 DIMENSION VEC(325),DELK(4),CCT(40)
000003 DIMENSION SANUD(15),BCT(40,15),CORNUD(15)
000003 DIMENSION ISAMPL(9)
000003 100 FORMAT(3I5,2F10,4)
000003 101 FORMAT(12A6)
000003 102 FORMAT(6F12,6)
000003 103 FORMAT(9F13,3)
000003 104 FORMAT(3I5,2F12,6)
000003 105 FORMAT(1I,F8,3,7F9,3)
000003 106 FORMAT(1X,F8,3,7F9,3)
000003 107 FORMAT(1X,10E12,4)
000003 108 FORMAT(5I5)
000003 109 FORMAT(6A6,A1,I5,5A6)
000003 110 FORMAT(6(3X,E9,4))
000003 111 FORMAT(6A6,A1,5I1,5A6)
000003 112 FORMAT(1H1)
000003 113 FORMAT(1X,9E12,4)
000003 114 FORMAT(1H0)
000003 115 FORMAT(2I5,4E12,6)
000003 R(1)=0.0
000004 DO 444 I=1,10
000006 444 R(I+1)=R(I)+.6105
000012 DO 445 I=11,30
000014 445 R(I+1)=R(I)+1.14475
000020 PUNCH 102,(R(I),I=1,31)
000032 PRINT 103,(R(I),I=1,31)
000044 WRITE(9,112)
C 20 ISOTOPES MAX
C 10 FLUX SETS
C IEL IS NO. OF ISOTOPE DECKS
000050 READ(10,100) IEL,IG
000060 WRITE(9,100) IEL,IG
C ANU(L)=FRACTION OF FISSION SPECTRUM
000070 READ(10,102) (ANU(L),L=1,IG)
000103 WRITE(9,107) (ANU(L),L=1,IG)
C JANL=NO. OF FLUX SETS
000116 READ(10,100) JANL
000124 WRITE(9,100) JANL
000132 DO 5 IANL=1,JANL
000134 READ(10,101) (C(I,IANL),I=1,12)
000147 WRITE(9,101) (C(I,IANL),I=1,12)
C ISAMPL IS NO. OF SPACE POINTS IN SAMPLE TEST
C ICORE IS NO. OF SPACE POINTS IN REACTOR CORE
C IRTOT IS NO. OF SPACE POINTS IN REACTOR
C IIG IS NO. OF GROUPS IN CROSS SECTIONS
C IGEO IS 1 FOR SLAB 2 FOR CYL. 3 FOR SPHERE
C 4 SETS OF FLUXES.
C 15 ISOTOPES.
000163 READ(10,108) ISAMPL(IANL),ICORE(IANL),IRTOT(IANL),IIG(IANL),IGEO
1 (IANL)
000201 WRITE(9,108) ISAMPL(IANL),ICORE(IANL),IRTOT(IANL),IIG(IANL),IGEO
1 (IANL)
000217 READ(10,115) NFUEL1,NFUEL2,FUEL1,FUEL2,RTOT,DELK(IANL)
000237 WRITE(9,115) NFUEL1,NFUEL2,FUEL1,FUEL2,RTOT,DELK(IANL)

```



```

000257      IF (IANL-4) 341,342,342
000262      342 READ(10,102) (RANL(I,4),I=1,31)
000274      IR=IRTOT(IANL)+1
000277      GO TO 343
000277      341 AIR=IRTOT(IANL)
000301      DR=RTOT/AIR
000303      RANL(1,IANL)=0.0
000305      IR=IRTOT(IANL)+1
000307      DO 333 I=2,IR
000311      AIR=I-1
000313      333 RANL(I,IANL)=DR*AIR
000321      343 WRITE(9,107) (RANL(I,IANL),I=1,IR)
000336      PUNCH 102, (RANL(I,IANL),I=2,IR)
000353      IR=IR-1
000355      IG=1IG(IANL)
C          ANE ANEA ARE FLUX AND ADJOINT
000357      READ(10,102) ((ANE(I,J,IANL),J=1,IR),I=1,IG)
000400      READ(10,102) ((ANEA(I,J,IANL),J=1,IR),I=1,IG)
000422      5 CONTINUE
000425      DO 8 K=1,IEL
000426      DO 8 I=1,24
000427      DO 8 J=1,27
000430      8 S(I,J,K)=0.0
C          IZERO IS LAST ELEMENT NO ON TAPE
000445      READ(10,100) IZERO
000452      WRITE(9,100) IZERO
000460      DO 3 K=1,IEL
000462      READ(10,111) (IB(I,K),I=1,17)
000475      WRITE(9,111) (IB(I,K),I=1,17)
C
C      CALCULATE OCTAL NUMBER OF ELEMENT.
000511      N=IZERO*K
000513      IB(8,K)=N/4096
000520      N=N-4096*IB(8,K)
000525      IB(9,K)=N/512
000532      IB(10,K)=(N-IB(9,K)*512)/64
000542      IB(11,K)=(N-IB(9,K)*512-IB(10,K)*64)/8
000553      IB(12,K)=N-IB(9,K)*512-IB(10,K)*64-IB(11,K)*8
000565      WRITE(9,111) (IB(I,K),I=1,17)
C          NOTE 6E12,6FORMAT.
000600      READ(10,102) ((S(I,J,K),J=1,27),I=1,IG)
000622      READ(10,111) (IC(I,K),I=1,17)
000636      READ(10,102) ((S1(I,J),J=1,27),I=1,IG)
000656      READ(10,111) (IC(I,K),I=1,17)
000672      READ(10,102) ((S2(I,J),J=1,27),I=1,IG)
000712      DO 300 I=1,IG
000714      STR=0.0
000715      DO 301 J=4,27
000717      L=I+J-4
000722      IF (L-24) 302,302,301
000724      302 STR=STR+S1(L,J)
000731      301 CONTINUE
000733      S(I,3,K)=S(I,3,K)-STR
000740      S(I,4,K)=S(I,4,K)-STR
000743      300 CONTINUE
000745      WRITE(9,113) ((S(I,J,K),J=1,9),I=1,IG)
000766      3 CONTINUE
C          24 GROUP 27 WIDE.
C          CALCULATE ALL ELEMENTS FOR EACH FLUX.
C
C          FLUX LOOP ON IANL TO JANL

```



```

000771      DO 6 IANL=1,JANL
000772      WRITE(9,112)
000775      ISMPL=ISMPL(IANL) +1
001000      IG=IIG(IANL)
001001      JEO=JGEO(IANL)
001003      WRITE(9,101) (C(I,IANL),I=1,12)
001016      IR=IRTOT(IANL)
001020      IR=IR+1
001022      DO 7 I=1,IR
001023      7 R(I)=RANL(I,IANL)

C
C      TURN OVER ADJOINT.
001033      IR=IR-1
001034      DO 66 I=1,IG
001036      DO 66 J=1,IR
001037      PHI(I,J)=ANE(I,J,IANL)
001047      L=IG-I+1
001051      66 PHIA(L,J)=ANEA(I,J,IANL)

C
C      CALCULATE EACH ELEMENT.
001066      CALL ADV(1)
001067      DO 1 K=1,IEL
001071      WRITE(9,112)
001074      SANUD(K)=0.0
001076      WRITE(9,111) (IB(I,K),I=1,17)

C
C      CHECK SIGMS GG
001111      DO 44 I=1,IG
001113      S(I,4,K)=0.0
001116      W=0.0
001117      DO 4 J=5,27
001121      M=1,J-4
001124      4 W=W+S(M,J,K)
001134      44 S(I,4,K)=S(I,3,K)-S(I,1,K)-W
001145      SRACTN=0.0
001146      VOL=0.0
001147      SDENFL=0.0
001150      SDNU=0.0
001151      SACT=0.0
001152      SANUD=0.0
001153      ACTSMPL=0.0
001154      WT=0.0
001155      IC=IRTOT(IANL)

C
C
C      VOLUME INTEGRAL.
C
C      DO 60 JR=1,IC
C
C      DO 60 JR=1,IC
C      DENFA=0.0
C      ANUD=0.0
C      ANUM=0.0
C      DENFS=0.0
C      ACTS=0.0
C      ACTA=0.0
C      DNU=0.0
C
C      ANUD=NU SIGMA F *PHI PHIA INTEG.
C      ACTA=ABSORBER ACT INTEG.
C      ACTS=SCATTER ACT INTEG.
C      PAPS=FISSION NEUTRON SOURCE *PHI*PHIA
C      DENFS=INTEGRAL PHI*PHIA
C      PAPA=DIAGONAL SOURCE
C      JEO=GEOM. IGEO

```

```

C          ACT=ACTS+ACTA
C          END OF JR=1 AND K=1 LOOP
C          RELATIVE VALUES OF ACT

001167      DO 2 I=1,IG
001170      ANU(I)=0.0
001171      DO 2 J=1,IG
001173      PAPS(I,J)=0.0
001176      2 A(I,J)=0.0
001204      A(1,2)=ANU(1) *S(2,2,K)
001210      ANUD=PHIA(1,JR)*PHI(2,JR)*(ANU(1)*S(2,2,K))
001216      ACTS=ACTS+A(1,2)*PHIA(1,JR)*PHI(2,JR)
001223      PAPS(1,2)=A(1,2)*PHIA(1,JR)*PHI(2,JR)
001226      DENFS=DENFS+PHIA(1,JR)*PHI(2,JR)
001231      DO 10 J=3,IG
001233      L=J-1
001235      DO 10 I=1,L
001236      M=1+I
001240      N=IG+1-(L-I+1)
001243      A(I,J)=ANU(I) *S(J,2,K)
001251      DENFS=DENFS+PHIA(I,JR)*PHI(J,JR)
001260      ANUD=ANUD+PHIA(I,JR)*PHI(J,JR)*(ANU(I)*S(J,2,K))
001273      PAPS(I,J)=A(I,J)*PHI(J,JR) *PHIA(I,JR)
001305      10 ACTS=ACTS+A(I,J)*PHI(J,JR) *PHIA(I,JR)
001321      A(2,1)=ANU(2) *S(1,2,K)+S(2,5,K)
001327      PAPS(2,1)=A(2,1)*PHIA(2,JR)*PHI(1,JR)
001333      ACTS=ACTS+A(2,1)*PHIA(2,JR)*PHI(1,JR)
001337      ANUD=ANUD+PHIA(2,JR)*PHI(1,JR)*(ANU(2)*S(1,2,K))
001345      DENFS=DENFS+PHIA(2,JR)*PHI(1,JR)
001351      DO 20 I=3,IG
001353      L=I-1
001355      DO 22 J=1,L
001356      M=4+I-J
001360      IF (M=27) 21,21,22
001362      21 A(I,J)=ANU(I) *S(J,2,K)+S(I,M,K)
001376      ACTS=ACTS+A(I,J)*PHIA(I,JR)*PHI(J,JR)
001410      PAPS(I,J)=A(I,J)*PHIA(I,JR)*PHI(J,JR)
001420      ANUD=ANUD+PHIA(I,JR)*PHI(J,JR)*(ANU(I)*S(J,2,K))
001431      DENFS=DENFS+PHIA(I,JR)*PHI(J,JR)
001436      22 CONTINUE
001441      20 CONTINUE
001443      DO 30 I=1,IG
001445      A(I,1)=ANU(I) *S(1,2,K)-S(I,3,K)+S(I,4,K)
001457      ANUD=ANUD+PHIA(I,JR)*PHI(1,JR)*(ANU(I)*S(1,2,K))
001471      PAPA(I,1)=A(I,1)*PHIA(I,JR)*PHI(1,JR)
001500      PAPS(I,1)=PAPA(I,1)
001503      DENFA = DENFA + PHIA(I,JR)*PHI(1,JR)
001507      ACTA1=ACTA
001511      ACTA=ACTA+A(I,1)*PHIA(I,JR)*PHI(1,JR)
001517      IF (ACTA=10.E+10) 30,35,35
001522      35 PRINT 108, I,JR,K
001534      PRINT 107,ACTA,A(I,1),PHI(1,JR),PHIA(I,JR),ANU(I),S(I,2,K),
1 S(I,3,K),S(I,4,K)
001573      ACTA=ACTA1
001575      30 CONTINUE
001600      34 ACT=ACTA+ACTS
001602      IF (JR=1) 33,33,32
001605      33 DENFAC=DENFA
001607      IF (K=1) 31,31,32
001611      31 RACTZ=ACT/DENFA
001613      ACTZ=ACT
001614      ACTAZ=ACTA
001615      ACTSZ=ACTS

```

```

001617      ACTC=R(JR+1)**JEO*ACT*4.189
001624      WRITE(9,107) RACTZ,ACTZ,ACTAZ,ACTSZ,ACTC
001641      32 RACTN=ACT/ACTZ*RACTZ
001644      ACTOD=ACT/DENFAC
001646      RACTAN=ACTA/ACTAZ*RACTZ
001650      RACTSN=ACTS/ACTSZ*RACTZ
001652      DVOL=4.189*(R(JR+1)**JEO-R(JR)**JEO)
001663      SHACTN=SPACTN+RACTN* DVOL
001665      VOL=VOL+DVOL
001667      DENFL=DENFA+DENFS
001671      SANUD(K)=SANUD(K)+ANUD*DVOL
001675      IF (JR-ICORE(IANL)) 351,352,351
001677      352 CORNUD(K)=SANUD(K)
001701      351 RCT(JR,K)=ACT*0.6023
C
C      BCT(JR) IS NUMERATOR OF DK/K.
001706      WRITE(9,107) ACTOD,RACTAN,RACTSN,DENFA,DENFL,RACTN,ANUD
1,ACTA,VOL,RCT(JR,K)
001741      IF (JR -ISMPL) 70,71,60
001744      70 ACTSMPL=ACT*DVOL+ACTSMPL
001747      DWT=DVOL*DENFA
001751      WT=WT+DWT
001752      GO TO 60
001753      71 SMPL=ACTSMPL/WT
001755      WRITE(9,107) SMPL ,ACTSMPL,WT
001767      60 CONTINUE
001772      AVRACT=SPACTN/VOL
001774      WRITE(9,107) SANUD(K),AVRACT
002003      DENOM=CORNUD(NFUEL1)*FUEL1+CORNUD(NFUEL2)*FUEL2
C
C      CCT(JR) IS DK/K.
002010      IF (IANL=4) 1,361,1
002012      361 DENOM=CORNUD(1)*FUEL1+(SANUD(1)-CORNUD(1))*0.00346
1+
CORNUD(4)*FUEL2+(SANUD(4)-CORNUD(4))* 0.048
002025      1 CONTINUE
002030      DO 335 K=1,IEL
002031      WRITE(9,114)
002034      WRITE(9,111) (IB(I,K),I=1,17)
002050      WRITE(9,114)
002054      DO 334 I=1,IC
002056      334 CCT(I)=RCT(I,K)/DENOM/DELK(IANL)
002067      WRITE(9,114)
002073      WRITE(9,101) (C(I,IANL),I=1,12)
002107      PRINT 113, (CCT(I),I=1,IC)
002122      CALL SPLOT(1,IC,R,CCT,42,1)
002126      WRITE (9,102) DENOM
002134      335 CONTINUE
002137      6 CONTINUE
002141      RETURN
002143      END

```

APPENDIX F. CODE ACT-I2

The Code (ACT-I2) used to include neutron flux and scattering isotropies requires flux and current forward and adjoint values (available from a DTF code modified by F. Brinkley). The reactivity is computed* from the formula

$$\frac{\Delta k}{k} = \frac{\int d\vec{r} \left(\sum_{g'} \sum_g \phi_{og'} \Delta v \sigma_{fg} X_{og} \phi_{og}^+ - \text{Tr} \right)}{\int d\vec{r} \sum_{g'} \sum_g \phi_{og'} v \sigma_{fg} X_{og} \phi_{og}^+}$$

$$\text{Tr} = \sum_n \sum_g (2n+1) \left(\phi_{n,g} \Delta \sigma_{t,g} \phi_{n,g}^+ - \sum_{g'} \Delta \sigma_{n,g \rightarrow g'} \phi_{n,g'}^+ \right)$$

where

- X_{og} = fraction of neutrons in group g ,
- n = the n^{th} Legendre polynomial,
- ϕ, ϕ^+ = flux and flux adjoints in group g ,
- σ_t, σ_f = total and fission macroscopic cross sections, and
- ϕ_0 = flux and ϕ_1 = current, as printed in DTF codes.

Figures F1-F5 illustrate reactivity coefficient (cents/mole) in Godiva and Jezebel as compared to experiment. Note that in this case the actual assumption of constant flux N_{ij} in space interval Δr_j for each energy group i is portrayed as computed.

Reactivity distribution $\frac{\Delta k}{k}$ (cents/mole) as a function of position, including flux and cross section anisotropies, is computed by:

1. Reading reactor description, fission neutron spectrum, radial points, and forward and adjoint flux values N_0, I_1 , and I_2 from the DTF-IV code as modified by Brinkley.

2. Reversing the adjoint fluxes in group indices.

3. Computing the three components of the above formula as

$$\sum_i \sum_l (2l+1) N_{il}^+ \sigma_{il}, \sum_i \sum_{l=1}^i N_{il}^+ \sigma(k \leftarrow) N_{kl}$$

and

$$\sum_i \sum_j N_{jo}^+ (v \sigma_f) X_i N_{io}$$

for each volume increment in Δr , where $1 < i < G$, G = number of groups, X_i is the fraction of fission

neutrons in group i , N_i is the neutron flux in group i , N^+ is the adjoint flux and l is 0, 1, or 2. The third term summed over the reactor volume corresponds to k in $\Delta k/k$, and the reactivity distribution in cents/mole requires the added factors (atoms/mole) and (cents/ Δk).

Fig. F1. Radial reactivity distribution of ^{235}U in Godiva.

Fig. F2. Radial reactivity distribution of ^{232}Th in Godiva.

*Communicated by G. Hansen, LASL

Fig. F3. Radial reactivity distribution of ^{239}Pu in Jezebel.

Fig. F4. Radial reactivity distribution of ^{238}U in Jezebel.

Fig. F5. Radial reactivity distribution of ^{12}C in Jezebel.

PROGRAM ACTL2 (INPUT,TAPE 10=INPUT,OUTPUT,
ITAPE 9=OUTPUT,FILM,PUNCH)

```

000003      C      DIMENSION S(24,27,3), RT(40,9),AN(24,40,3),
      1ANA(24,40,3), PHI(24,40,3),PHIA(24,40,3),C(12,9),
      2ISAMPL(9),ICORE(9),IRT(9),IIG(9),DELK(9),D(12,9),
      3F(24,9),R(41),ANUM(40),BCT(42,9),CORNUD(9),DEN(40,9),
      4SANUD(9),RAD(41),CCT(40),IGEO(9),ACT(41)

000003      100 FORMAT(6E12.5)
000003      101 FORMAT(1X,1P10E12.4)
000003      102 FORMAT(2I5,4E12.6)
000003      103 FORMAT(12A6)
000003      104 FORMAT(5I5)
000003      105 FORMAT(1H0)
000003      106 FORMAT(1H1)
000003      WRITE(9,106)
000007      READ(10,102) IEL,IG
000017      WRITE(9,102) IEL,IG
000027      WRITE(9,105)

      C
      C      ACTIVITY LOOP
000033      IN=1
000034      READ(10,103) (C(I,IN),I=1,12)
000050      WRITE(9,105)
000054      WRITE(9,103) (C(I,IN),I=1,12)
000070      WRITE(9,105)

      C
      C      IANL=4 FOR 2 REGION EXPT. NOTE AT. DENS. IN DEN.
      C      CORE SPECIFICATION.
000074      READ(10,104) ISAMPL(IN),ICORE(IN),IRT(IN),IGEO(IN),IANL
000112      WRITE(9,104) ISAMPL(IN),ICORE(IN),IRT(IN),IGEO(IN),IANL
000130      JEO=IGEO(IN)
000132      ICORE(IN)=ICORE(IN)+1
000134      ISAMPL(IN)=ISAMPL(IN)+1
000136      READ(10,102) NFUEL1,NFUEL2,FUEL1,FUEL2,RTOT,DELK(IN)
000155      WRITE(9,102) NFUEL1,NFUEL2,FUEL1,FUEL2,RTOT,DELK(IN)
000175      WRITE(9,105)

      C
      C      FISSION SPECTRUM.
000201      READ(10,100) (F(I,IN),I=1,IG)
000216      WRITE(9,100) (F(I,IN),I=1,IG)
000233      IR=IRT(IN)+1
000236      READ(10,100) (R(I),I=1,IR)
000250      L=1

      C
      C      REACTOR FLUX SPECTRUM (FORWARD).
000251      READ(10,100) ((AN(I,JR,L),JR=2,IR),I=1,IG)
000273      READ(10,100) ((AN(I,JR,L),I=1,IG),JR=2,IR),L=2,3)
000320      READ(10,100) (R(I),I=1,IR)
000333      L=1

      C
      C      REACTOR FLUX SPECTRUM (ADJOINT).
000334      READ(10,100) ((ANA(I,JR,L),JR=2,IR),I=1,IG)
000356      READ(10,100) ((ANA(I,JR,L),I=1,IG),JR=2,IR),L=2,3)
000403      EA=0
000404      E=0.0
000405      PRINT 101,(R(I),I=1,IR)
000417      PRINT 101,(AN(I,2,1),I=1,IG)
000432      PRINT 101,(ANA(I,2,1),I=1,IG)
000445      WRITE(9,105)

      C
      C      INTEGRAL OF MU * FLUX.
000451      DO 18 I=1,IG
000453      E=F*AN(I,IR,2)
000457      18 EA=EA+ANA(I,IR,2)
000464      EA=EA*.03,1416*R(I)*.02*.5*(-1.0)
000472      E=F*.03,1416*R(IR)*.02*.5
000477      WRITE(9,105)
000502      PRINT 101,E,EA
000512      WRITE(9,105)

      C
      C      ADJOINT REVERSE
      C      COPY FORWARD AND REVERSE ADJOINT.
000516      L=1
000517      DO 6 J=2,IR
000521      DO 6 I=1,IG
000522      PHI(I,J,L)=AN(I,J,L)
000532      M=IG-I+1
000534      6 PHIA(M,J,L)=ANA(I,J,L)

```

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C      FLUX TWICE ACTUAL FOR L=2 AND 3.
000546      DO 5 L=2,3
000547      DO 1 J=2,IR
000550      DO 1 I=1,IG
000551      PHI(I,J,L)=AN(I,J,L)*0.5
000562      M=IG-I+1
000564      1 PHIA(M,J,L)=ANA(I,J,L)*0.5
000577      5 CONTINUE

C      ELEMENT REACT.
K=LUO,
000601      DO 7K=1,IFL
000602      DO 2 L=1,3
000603      READ(10,103) (D(I,K),I=1,12)
000616      2 READ(10,100) ((S(I,J,L),J=1,27),I=1,24)
000641      DO 4 L=2,3
000643      DO 4 J=1,3
000644      DO 4 I=1,IG
000645      4 S(I,J,L)=0.0
000661      DO 19 L=1,3
000662      DO 19 I=1,24
000663      X=0.0
000664      DO 20 J=4, 27
000666      M=I+J-4
000671      IF (M=24) 21,21,22
000673      21 X=X+S(M,J,L)
000702      20 CONTINUE
000704      22 S(I,3,L)=X+S(I,1,L)
000713      19 CONTINUE
000716      DENOM=0.0
000717      VOL=0.0
000720      WRITE(9,103) (D(I,K),I=1,12)
000733      WRITE(9,105)
000737      DO 8 JR=2,IR
000741      DV= 4.189 *(R(JR)**JEO-R(JR-1)**JEO)
000752      SDD=0.0
000753      TSS=0.0
000754      SS=0.0

C      INTEGRATION OVER ADJOINT.
000755      DO 9 I=1,IG

C      TSS IS TOTAL CROSS SECTION * FLUX INTEGRAL.
000756      L=1
000757      AL=L
000761      TSS1=((AL-1.0)*2.0+1.0)*PHIA(I,JR,L)*(S(I,3,L) )*PHI(I,JR,L)
001002      TSS=TSS+TSS1

C      SS IS INTER-GROUP SCATTERING INTEGRAL.
001004      DO 11 JS=1,I
001005      NSC=I-JS+4
001010      SS1= S( I-NSC,L)* PHI (JS,JR,L)*((AL-1.0)*2.0+1.0)*PHIA(I,JR,L)
001033      SS=SS+SS1
001034      11 CONTINUE
001037      L=2
001040      AL=L
001041      TSS1=((AL-1.0)*2.0+1.0)*PHIA(I,JR,L)*(S(I,3,L) )*PHI(I,JR,L)
001057      TSS=TSS+TSS1*(-1.0)
001061      DO 23 JS=1,I
001063      NSC=I-JS+4
001066      SS1= S( I-NSC,L)* PHI (JS,JR,L)*((AL-1.0)*2.0+1.0)*PHIA(I,JR,L)
001111      SS=SS+SS1*(-1.0)
001113      23 CONTINUE
001116      L=3
001117      AL=L
001120      TSS1=((AL-1.0)*2.0+1.0)*PHIA(I,JR,L)*(S(I,3,L) )*PHI(I,JR,L)
001136      TSS=TSS+TSS1
001137      DO 24 JS=1,I
001141      NSC=I-JS+4
001144      SS1= S( I-NSC,L)* PHI (JS,JR,L)*((AL-1.0)*2.0+1.0)*PHIA(I,JR,L)
001167      SS=SS+SS1
001170      24 CONTINUE

C      SDD IS FISSION INTEGRAL.
001173      DD1=0.0
001174      DD2=0.0
001175      DD3=0.0
001176      DO 12 J=1,IG

```

```

001177      DD1=DD1+PHI(J,JR,1)*S(J,2,1)
001204      DD2=DD2+PHI(J,JR,2)*S(J,2,1)
001210      DD3=DD3+PHI(J,JR,3)*S(J,2,1)
001214      12 CONTINUE
001216      SDD=SDD+PHI4(I,JR,1)*F(I)*DD1
001225      9 CONTINUE
001230      ANUM(JR)= SDD-TSS +SS
001233      PRINT 101, SDD,TSS,SS
001245      BCT(JR,K)=ANUM(JR)*0.6023
001252      DENOM=DENOM+SDD*DV
001255      VOL=VOL+DV
001257      SANUD(K)=DENOM
001261      DEN(JR,K)=DENOM
001264      IF (JR-ICORE(IN)) 8,15,8
001266      15 CORNUD(K)=DEN(IR,K)
001272      8 CONTINUE
001275      WRITE(9,101) (BCT(I,K),I=2,IR)
001311      WRITE(9,101) DENOM
001317      7 CONTINUE
001322      DENOM=CORNUD(NFUEL1)*FUEL1+CORNUD(NFUEL2)*FUEL2
001326      IF (IANL-4) 13,14,13
C      FOR TOPSY IANL=4. NOTE ORDER OF K VALUES 235-U,238-U ...
001339      14 DENOM=CORNUD(1)*0.0025 + (SANUD(1)-CORNUD(1))*0.0
      +CORNUD(4)*0.045525 + (SANUD(4)-CORNUD(4))*0.048
001343      13 CCT(1)=0.0
001344      DO 16 K=1,IEL
001346      WRITE(9,105)
001351      RAD(1)=0.0
001352      M=2
001353      ACT(1)=BCT(2,K)/DENOM/DELK(IN)
001360      DO 17 I=2,IR
001362      DR=R(I)-R(I-1)
001364      RAD(M)=RAD(M-1)+DR-0.0001
001370      RAD(M+1)=RAD(M)+0.0001
001373      ACT(M)=ACT(M-1)
001374      ACT(M+1)=BCT(I+1,K)/DENOM/DELK(IN)
001402      17 M=M+2
001406      KR=2*(IR-1)
001411      WRITE(9,105)
001415      WRITE(9,103) (D(I,K),I=1,12)
001439      WRITE(9,105)
001435      PRINT 101, (ACT(I),I=1,KR)
001450      CALL SPLOT(1,KR,RAD,ACT,42,1)
001454      16 CONTINUE
001457      WRITE(9,105)
001462      PRINT 101, (ANUM(JR),JR=2,IR)
001475      WRITE(9,105)
001501      PRINT 101, (CORNUD(K),K=1,IEL)
001514      WRITE(9,105)
001520      PRINT 101, (SANUD(K),K=1,IEL)
001533      RETURN
001535      END

```

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TEST

APPENDIX G. CODE EQUIL

This code (EQUIL) sums the changes in atomic density of an isotope, i , with a microscopic capture (n,γ) cross section, σ_i^c , and an absorption $(n,\gamma) + (n,f)$ cross section, σ_i^a .

$$\Delta N_{i+1} = \overline{\phi \Delta t} N_i \sigma_i^c - N_{i+1} \sigma_{i+1}^a,$$

where N_i is the atomic density of parent isotope i , and $i+1$ indicates the daughter isotope.

The code EQUIL simply integrates the differential form above, using mean reactor group-averaged neutron cross sections obtained by flux weighting the tables in Appendix B using Code FISH in Appendix I.

```

PROGRAM EQUIL (INPUT,TAPE 10=INPUT, OUTPUT,TAPE 9=OUTPUT,FILM,
1PUNCH)
C   TIME DEPENDENT ABSORPTION OF TRANS URANIUM ELEMENTS.
    DIMENSION SF(10),SC(10),SN2N(10),AN(10),DN(10),FISSP(1000),
1AN2N(10),T(1001),AN1(1000),AN2(1000),AN3(1000),AN4(1000),
2AN2N1(1000),AN2N2(1000),AN2N3(1000),AN2N4(1000),DN2N(10)
3,R2928(1000)
100 FORMAT(6F12.4)
101 FORMAT(1X,10E12.4)
102 FORMAT(1X,16,10E12.4)
103 FORMAT(1H0)
104 FORMAT(1H1)
    K=1
    T(1)=0.0
    DT=.0008
    WRITE(9,104)
    7 READ(10,102) NISO
    READ(10,100) (SF(I),I=1,NISO)
    READ(10,100) (SC(I),I=1,NISO)
    READ(10,100) (SN2N(I),I=1,NISO)
    4 AN(1)=1.0
    AN(2)=0.0
    AN(3)=0.0
    AN(4)=0.0
    FISSP=0.0
    SFP=0.0
    AN1(1)=0.95
    AN2(1)=0.05
    AN3(1)=0.0
    AN4(1)=0.0
    AN2N(1)=0.0
    AN2N(2)=0.0
    AN2N(3)=0.0
    AN2N(4)=0.0
    AN2N1(1)=0.0
    AN2N2(1)=0.0
    AN2N3(1)=0.0
    AN2N4(1)=0.0
    R2928(1)=0.0
    DO 11=2,1000
    DO 20 LP=1,10
    DN(1)= - AN(1)*(SF(1)+SC(1)+SN2N(1))*DT
    DN(2)=DT*(SC(1)+AN(1)-(SF(2)+SC(2)+SN2N(2))*AN(2))
    DN(3)=DT*(SC(2)+AN(2)-(SF(3)+SC(3)+SN2N(3))*AN(3))
    DN(4)=DT*(SC(3)+AN(3)-(SF(4)+SC(4)+SN2N(4))*AN(4))
    DN2N(1)=SN2N(1)*AN(1)*DT

```

```

DN2N(2)=SN2N(2)*AN(2)*DT
DN2N(3)=SN2N(3)*AN(3)*DT
DN2N(4)=SN2N(4)*AN(4)*DT
AN(1)=AN(1)+DN(1)
AN(2)=AN(2)+DN(2)
AN(3)=AN(3)+DN(3)
AN(4)=AN(4)+DN(4)
AN2N(1)=AN2N(1)+DN2N(1)
AN2N(2)=AN2N(2)+DN2N(2)
AN2N(3)=AN2N(3)+DN2N(3)
AN2N(4)=AN2N(4)+DN2N(4)
20 CONTINUE
AN1(I)=AN(1)
AN2(I)=AN(2)
AN3(I)=AN(3)
AN4(I)=AN(4)
AN2N1(I)=AN2N(1)
AN2N2(I)=AN2N(2)
AN2N3(I)=AN2N(3)
AN2N4(I)=AN2N(4)
R2928(I)=AN2(I)/AN1(I)
SFP=SFP+DT*(SF(1)*AN(1)+SF(2)*AN(2)+SF(3)*AN(3)+AN(4)*SF(4))
FISSP(I)=SFP
T(I)=T(I-1)+DT*10.
1 CONTINUE
C RADIO-DECAY ASSUMED SMALL
WRITE(9,103)
PRINT 102,(I,T(I),AN1(I),AN2(I),AN3(I),AN4(I),
1AN2N1(I),AN2N2(I),AN2N3(I),AN2N4(I),FISSP(I),I=1,1000,10)
PRINT 101,SFP
PRINT 101,(R2928(I),I=1,1000,10)
9 CALL ADV(1)
CALL SPLOT(1,1000,T,AN1,42,1)
CALL SPLOT(1,1000,T,AN2,42,1)
CALL SPLOT(1,1000,T,AN3,42,1)
CALL SPLOT(1,1000,T,AN4,42,1)
CALL SPLOT(1,1000,T,AN2N1,42,1)
CALL SPLOT(1,1000,T,AN2N2,42,1)
CALL SPLOT(1,1000,T,AN2N3,42,1)
CALL SPLOT(1,1000,T,AN2N4,42,1)
CALL SPLOT(1,1000,T,FISSP,42,1)
CALL SPLOT(1,1000,T,R2928,42,1)
K=K+1
IF(K-2) 5,5,6
5 GO TO 7
6 RETURN
END

```

APPENDIX H. CODES MCT-2, RMITU AND ADLER

The MCT-2 code computes unresolved resonance, Monte Carlo averaged neutron cross sections for two mixed isotopes as a function of energy (E), temperature (T), and moderator scattering $\left[\sigma_p \equiv \sum_s / N(\text{absorber})\right]$ where $\sum_s$ is the macroscopic moderator scattering and N is the atomic density of the absorber atom. Foil effects are included as an option.

The RMITU code uses Bell's integral formulae* for group averaging of resolved resonances as a function of group (energy bounds ΔE), temperature (T), and moderator scattering or absorber foil thickness, including the temperature effects below the resolved resonance region. The output is on DTF-IV consistent tape or cards.

Code RMITU averages resolved resonances by

1. Computation of $\sigma_a(E)$, $\sigma_p(E)$, and $\sigma_t(E)$ over the entire resolved resonance region for each temperature using the Voigt profile tables for Doppler broadening effects.
2. The effects of the NR or IM approximations, or foil self shielding, added as a factor and group averaged cross sections computed using a bare reactor flux weighting factor.
3. Unresolved resonances values added from T and σ_p dependent tables and the entire set of cross section decks punched or listed on tape for further use in DTF-IV.

Code MCT-2 averages unresolved resonances by setting up a display of resonances parameters repeatedly by Monte Carlo sampling techniques. The simple integrals described above are used for tabulation of T and σ_p dependence.

The resolved resonance region has been shown by Adler and Adler** to be better represented by

the multilevel-multichannel formalism when fissile isotopes are studied. The code ADLER below simply sums the contribution $\sigma_i(E)$ of isotope resonance parameter μ_i , ν_i , G_i^t , H_i^t , etc. for all i, using the Adler parameter fit to experimental values of $\sigma_f(E)$ and $\sigma_t(E)$ for ^{235}U .

* Communicated by G. I. Bell, LASL. See also LA-2322 (1959), "Theory of Effective Cross Sections" by G. I. Bell.

** See F. T. Adler and D. B. Adler, "Calculations of Resonance Integrals for Fissile Materials," presented at the National Topical Meeting on Reactor Physics in the Resonance and Thermal Regions, February 1966, San Diego, California

PROGRAM MCT2 (INPUT,TAPE 10=INPUT,OUTPUT,
1TAPE 9=OUTPUT,FILM)

C
C MONTE CARLO CALCULATION OF UNRESOLVED RESONANCE NEUTRON CROSS SECTION
C

```
000003 DIMENSION AID(1100),AIN(1100),AIE(1100)
000003 DIMENSION SCZ(40),SSZ(40),GN(40),GG(40),GE(40),GT(40),
1E(1),ALZ(1),A(1),BI(1),HI(100),EZ(10),EZ(40),EZR(40),
2WI(100),RIJ( 50,46),AKJ(60),AI(60),DELT(1)
000003 DIMENSION SPGN(20),SPD(20),G(40)
000003 DIMENSION SA(20),SF(20),ST(20),W(40),EZW(10),F1(20)
000003 DIMENSION T(10),U(10),SP2(4),GZN(21),GZE(21)
000003 DIMENSION GIM1(9,3,3),GIN1(9,3,3),GFN1(9,3,3),GFN1(9,3,3),
1GIM1(9,3,3),GIN1(9,3,3),SSA1(9,3,3),SSF1(9,3,3),SST1(9,3,3),
2FSA1(9,3,3),FST1(9,3,3),XA(9,3,3),XE(9,3,3),XAT(9,3,3),
3XFT(9,3,3),FSF1(9,3,3)
000003 100 FORMAT (6(1X,F8.0))
000003 101 FORMAT (10(1X,E10.5))
000003 102 FORMAT(7F10,6/3F10.6)
000003 103 FORMAT (1X,3I5,5F8.4,3F10.1)
000003 104 FORMAT (3I5,2F10.5)
000003 105 FORMAT (I5,2F12.6)
000003 106 FORMAT (I5,10E10.3)
000003 107 FORMAT (3I5,7E14.4)
000003 108 FORMAT(4I5,9F12.5)
000003 109 FORMAT(9F8.4)
000003 110 FORMAT(8F9.4)
000003 111 FORMAT (1X,3F10.4)
000003 112 FORMAT (8(F8.4,1X))
000003 113 FORMAT (1X,I5,7E14.6)
000003 114 FORMAT (3I5,2F14.6)
000003 115 FORMAT(2X,10E12.4)
000003 116 FORMAT(1H1)
000003 117 FORMAT(1X,10F12.5)
000003 118 FORMAT(1H0)
000003 119 FORMAT(1X,9F12.2)
000003 120 FORMAT(4I5,8E14.4)
000003 121 FORMAT(1X,E12.4,8F10.6,3E12.4)
000003 3006 FORMAT(3I5,6E12.4)
000003 3007 FORMAT(6E12.6)
000003 3008 FORMAT(2(I6,3F12.6))
000003 WRITE(9,116)
000007 READ (10,102) (AI(I),I=1,50)
000021 READ (10,102) (HI(I),I=1,50)
000033 READ (10,102) (WI(I),I=1,50)
000045 READ(10,102) (AKJ(J),J=1,46)
000057 READ(10,102) ((RIJ(I,J),I=1,50),J=1,46)
000076 READ(10,3007) (AIN(I),I=1,1000)
000110 READ(10,3007) (AID(I),I=1,1000)
000122 READ(10,3007) (AIE(I),I=1,1000)
000134 WRITE(9,119)
000140 READ(10,109) (U(I),I=1,9)
```

C TWO ISOTOPES. W1+W2=1
C AVD1 LESS THAN AVD2
000152 READ(10,112) AMAS1,RAU1,G1,GC1,AVD1,AVGNZ1,AVGEZ1
000174 READ(10,112) AMAS2,RAU2,G2,GC2,AVD2,AVGNZ2,AVGEZ2

C MILLIVOLTS TO VOLTS.

```
000216 AVGNZ1=AVGNZ1*.001
000220 AVGNZ2=AVGNZ2*.001
000221 AVGEZ1=AVGEZ1*.001
000222 AVGEZ2=AVGEZ2*.001
000223 GC1=GC1*.001
000224 GC2=GC2*.001
000225 SA5=0.0
000226 SF5=0.0
000227 ST5=0.0
000230 AMAS=AMAS1
000231 AVD=AVD1
```

INIT1
INIT1
INIT1

C INTEGRATION INTERVAL = D1/10

```

C          AVERAGE RADIUS USED FOR SIG S
C          SIGMA P OR LUMPS FOR WHICH THE MEAN FREE PATH IS LARGE WITH RESPECT
C          TO ABSORPTION PROBABILITY.
000233      RAD=(RAD1+RAD2)/2.0
000236      AHEA=12.566*RAD**2
000240      AVGNZ=AVGNZ1
000241      AVGEZ=AVGEZ1
C          READ GEOM, AT DENSITY, AND RADIUS. WHEN GEOT=0 USE SIGMA-P
C          MACROSCOPIC AVERAGED CROSS SECTIONS WHEN ATOMIC DENSITIES OF TWO ELEMENTS
C          ARE USED FOR LUMP CALCULATIONS.
000243      READ(10,110) (SP2(I),I=1,3),GEOM,ATDEN1,ATDEN2,RADIUS
000244      READ(10,111) T(1),T(2),T(3)
000276      WRITE(9,119)
C          DISPLAY 5 RESONANCES FOR EACH ISOTOPE.
000302      DO 82 N=1,30
000304          G(N)=G1
000306      A2 W(N)=ATDEN1
000311      DO 83 N=31,40
000312          G(N)=G2
000314      A3 W(N)=ATDEN2
000317          SK=0.0
000320          YT=0.0
000321          PI=3.14159265
000322      DO 150 IV=1,9
000324          EZZ(IV)= 10.0**7.0/EXP(U(IV))
000336      WRITE(9,117) (EZZ(I),I=1,9)
C          INITIALIZE INTEGRALS.
000347      DO 1003 I=1,9
000351          DO 1002 J=1,3
000352              DO 1001 K=1,3
000353                  XE(I,J,K)=0.0
000361                  XFT(I,J,K)=0.0
000365                  GFN1(I,J,K)=0.0
000372                  GFM1(I,J,K)=0.0
000376                  GIM1(I,J,K)=0.0
000402                  GIN1(I,J,K)=0.0
000406                  DIM1(I,J,K)=0.0
000412                  DIN1(I,J,K)=0.0
000416                  SSA1(I,J,K)=0.0
000422                  SSF1(I,J,K)=0.0
000426                  SST1(I,J,K)=0.0
000432      1001 CONTINUE
000433      1002 CONTINUE
000435      1003 CONTINUE
000437          EAVG=0.0
000440          ANVG=0.0
000441          OAVG=0.0
000442          ANVG2=0.0
000443          OAVG2=0.0
000444          GAVG9=0.0
C          OUTER LOOPS FOR CONVERGENCE. INNER LOOPS FOR ACCURACY.
C          MONTE CARLO SELECTION OF PARAMETERS ON
C          TABLES OF 1000 POINTS RELATING Y=RANDOM(K) AND XI, THE
C          PROBABILITY FUNCTION FOR GN(ZERO) AND G FISS.
C          LINEAR INTERPOLATION BETWEEN ORDINATE POINTS.
000445      WRITE(9,119) (U(I),I=1,9)
000456      WRITE(9,117) AMAS1,RAD1,G1,GC1,AVD1,AVGNZ1,AVGEZ1
000500      WRITE(9,117) AMAS2,RAD2,G2,GC2,AVD2,AVGNZ2,AVGEZ2
000522      WRITE(9,117) (SP2(I),I=1,3),GEOM,ATDEN1,ATDEN2,RADIUS
000544      WRITE(9,111) T(1),T(2),T(3)
000556      WRITE(9,117) AVGNZ1,AVGEZ1,AVD1,AVGNZ2,AVD2,GC1,GC2
000600      SK=1
000602      DO 2 KC=1,4
000603          Y=RANDOM(K)
000605      DO 1 K1=1,100
000606          Y=RANDOM(K)
000610          Y=RANDOM(K)
000612          YT=YT+Y
000613          SK=SK+1.0
000615      DO 5 NY=1,10
000617          Y=RANDOM(K)

```

```

000621      TRN=1000.*Y+1.
000624      ARN=TRN-1
000626      RFN= 1000.*Y-ARN
000630      IF (TRN-1)1010,1010,1020
000631 1010  GZN1=RFN*AIN(1)*AVGN71
000634      GZE1=RFN*AIE(1)*AVGE71
000636      GO TO 1030
000637 1020  XIN=(AIN(TRN-1)+RFN*(AIN(TRN)-AIN(TRN-1)))
000644      GZN1=XIN*AVGN71
000646      IRE=1000.*Y+1.
000651      ARE=IRE-1
000653      RFE= 1000.*Y-ARE
000655      XIE=(AIE(IRE-1)+RFE*(AIE(IRE)-AIE(IRE-1)))
000661      GZE1=XIE*AVGE71
000663 1030  Y=RANDOM(K)
000666      GZN(NY)=GZN1
000670      GZE(NY)=GZE1
000671      5 CONTINUE
000673      DO 6 NY=11,20
000675      TRN=1000.*Y+1.
000678      ARN=TRN-1
000683      RFN= 1000.*Y-ARN
000685      IF (TRN-1)1011,1011,1021
000686 1011  GZN2=RFN*AIN(1)*AVGN72
000688      GZE2=RFN*AIE(1)*AVGE72
000691      GO TO 1031
000694 1021  XIN=(AIN(TRN-1)+RFN*(AIN(TRN)-AIN(TRN-1)))
000697      GZN2=XIN*AVGN72
000700      IRE=1000.*Y+1.
000703      ARE=IRE-1
000706      RFE= 1000.*Y-ARE
000708      XIE=(AIE(IRE-1)+RFE*(AIE(IRE)-AIE(IRE-1)))
000711      GZE2=XIE*AVGE72
000714 1031  Y=RANDOM(K)
000717      GZN(NY)=GZN2
000720      GZE(NY)=GZE2
000723      6 CONTINUE
000726
000730      C
000732      1036  EAVG=EAVG+GZE(4)
000734      ANVG=ANVG+GZN(4)
000736      ANVG2=ANVG2+GZN(15)
000738      C
000740      C
000742      DO 301 IV=1,9
000744      SQT=SQRT(EZZ(IV))
000746      EZ(1)=EZZ(IV)-5.0*AVD1
000748      DO 7 NY=2,10
000750      Y=RANDOM(K)
000752      IRD=1000.*Y+1.
000754      ARD=IRD-1
000756      RFD= 1000.*Y-ARD
000758      IF (IRD-1)1007,1007,1017
000760 1007  D1=RFD*AID(1)*AVD1
000762      GO TO 1033
000764 1017  XID=(AID(IRD-1)+RFD*(AID(IRD)-AID(IRD-1)))
000766      D1=XID*AVD1
000768 1033  EZ(NY)=EZ(NY-1)+D1
000770      GE(NY)=SQT*GZE(NY)
000772      GN(NY)=SQT*GZN(NY)
000774      GG(NY)= GC1 +GE(NY)
000776      7 CONTINUE
000778      DAVG=DAVG+D1
000780      EZ(11)=EZZ(IV)-5.0*AVD2
000782      DO 8 NY=12,20
000784      Y=RANDOM(K)
000786      IRD=1000.*Y+1.
000788      ARD=IRD-1
000790      RFD= 1000.*Y-ARD
000792      IF (IRD-1)1008,1008,1018
000794 1008  D1=RFD*AID(1)*AVD2
000796      GO TO 1035
000798 1018  XID=(AID(IRD-1)+RFD*(AID(IRD)-AID(IRD-1)))

```

AVD1- LESS THAN AVD2

```

001060      D1=X10*AVD2
001062      1075 EZ(NY)=EZ(NY-1)+D1
001065      GN(NY)=SQT*GZN(NY)
001067      GE(NY)=SQT*GZF(NY)
001071      GG(NY)=GC2      +GE(NY)
001073      9 CONTINUE
001075      GN(1)=SQT*GZN(1)
001077      GN(11)=SQT*GZN(11)
001101      GE(1)=SQT*GZF(1)
001103      GE(11)=SQT*GZF(11)
001105      GG(1)=GC1+GE(1)
001107      GG(11)=GC2+GE(11)
001111      DAVG2=DAVG2+D1
001113      GAVG9=GAVG9+GC1      +SQT(EZZ(9))*GZE(4)
C          INTEGRATION A FUNCTION OF T, GROUP BOUNDS, AND
C          MODERATOR SCATTERING. G I BELL FORMULA USED.
C          AREA NOT IN TOTAL CROSS SECTION.
C
C          30 VALUES FOR ISOTOPE 1 (SMALL AVE D).
C          10 VALUES FOR ISOTOPE 2.
001120      DO 93 I=31,40
001121      EZR(I)=EZ(I-20)
001123      GN(I)=GN(I-20)
001124      GE(I)=GE(I-20)
001125      93 GG(I)=GG(I-20)
001131      J=1
001132      DO 91 I=11,20
001133      EZR(I)=EZ(J)
001135      GN(I)=GN(J)
001137      GE(I)=GE(J)
001140      GG(I)=GG(J)
001142      91 J=J+1
001145      EZR(1)=EZZ(IV)-15.*AVD1
001150      DO 90 I=2,10
001152      EZR(I)=EZR(I-1)+AVD1
001155      GN(I)=SQT*AVGNZ1
001157      GE(I)=SQT*AVGF71
001160      90 GG(I)=GC1+GE(I)
001164      DO 92 I=21,30
001166      EZR(I)=EZR(I-1)+AVD1
001171      GN(I)=SQT*AVGN71
001173      GE(I)=SQT*AVGF71
001174      92 GG(I)=GC1+GF(I)
001200      GN(1)=SQT*AVGNZ1
001202      GE(1)=SQT*AVGF71
001204      GG(1)=GC1+GF(1)
001206      DO 300 IT=1,3
001207      ET=T(IT)
001211      DELT(1)=2.0*SQT(ET*EZZ(IV)/AMAS)
001217      DE=DELT(1)/10.
001221      DO 84 M=1,10
001222      EVBL=M
001223      E1(M)=EZZ(IV)-DELT(1)*0.5+EVBL*DE
001231      E(1)=E1(M)
001233      SAI=0.0
001234      SFI=0.0
001235      ST1=0.0
001236      DO 81 N=1,40
001237      EZ(N)=EZR(N)
001241      IF(EZ(N)-0.0) 81,81,85
001243      85 GT(N)=GG(N)+GN(N)
001246      SCZ(N)=2.601*10.**6 *GN(N)*GG(N)/GT(N)**2/EZ(N)*G(N)
001256      SSZ(N)=SCZ(N)/GG(N)*GN(N)
001262      ALZ(1)=4.56*10.**2/SQT
001264      91(1)=16.*3.1416*ALZ(1)*GN(N)*RAD*G(N)
001272      A(1)=GT(N)/(2.0*DELT(1))
001275      IF(AI(49)-A(1))213,212,212
001300      212 CALL SEARCH (A(1),AI(1),50,1,MFLAG)
001304      AR=(A(1)-AI(1))/(AI(1+1)-AI(1))
001311      H=HI(1)-AR*(HI(1)-HI(1+1))
001315      YRES=WI(1)+AR*(WI(1+1)-WI(1))
001322      V=(E(1)-EZ(N))/DELT(1)

```

LOOP

```

001325      AK=ABS(V/YRES)
001327      IF(20.-AK) 218,220,220
001332 218 R=RIJ(I,46)*EXP(-0.14*(AK-20.))
001342      GO TO 219
001342 220 CALL SEARCH (AK ,AKJ(1),46,J,MFLAG)
001346      AKR=(AKJ(J)-AK)/(AKJ(J)-AKJ(J+1))
001353      RKM=RIJ(I,J)-AKR*(RIJ(I,J)-RIJ(I+1,J))
001362      RKP=RIJ(I,J+1)-AKR*(RIJ(I,J+1)-RIJ(I+1,J+1))
001370      R=RKM-(RKM-RKP)*AKR
001374 219 HFN=R*H
001376      PSI=1.772*A(1)*HFN
001401      GO TO 214
001401 213 PSI=0.0
001402 214 SIGS=SSZ(N)*PSI
001405      SIGA=SCZ(N)*PSI
001406      SA1=SA1+SIGA*W(N)
001411      SF1=SF1+SIGA*GF(N)/GG(N)*W(N)
001415      ST1=ST1+(SIGA+SIGS)*W(N)
001421      R1 CONTINUE
C      MACROSCOPIC IF W(N)=ATOMIC DENSITY.
001423      AE=SA1
001425      FE=SF1
001426      TE=ST1+AREA*(W(1)+W(31))

C
C      NOTE 1 SIGMA P.

001432      DO 302 IP=1,3
001434      SP=SP2(IP)
C      CALCULATE LUMPS OR SIGMA P BY CHOICE OF DATA. SLAB,CYL.,OR SPHERE 1,2,3
001436      IF(GEOM=1.0) 320,321,322
001440 320 SWT1=1.0/(TE/SP+1.0)
001444      SWT2=1.0/(AF/SP+1.0)
001446      GO TO 333
001447 321 AOL= RADIUS*AE
001451      PZRO=1.0-0.5*AOL*ALOG(1.0/AOL)-0.5*AOL*(1.5-0.577216)-0.1666*(AOL)
001451      **2
001457      SWT1=(1.0-PZRO*(1.0-AEA/TE))/(1-PZRO*(TE-AE)/TE)
001502      SWT2=SWT1
001503      GO TO 333
001503 322 IF(GEOM=2.0) 324,324,325
001506 325 AOL= RADIUS*AE
001510      PZRO=1.0-0.75*AOL
001513      SWT1=(1.0-PZRO*(1.0-AEA/TE))/(1-PZRO*(TE-AE)/TE)
001525      SWT2=SWT1
001526      GO TO 333
001526 324 AOL= RADIUS*AE
001530      PZRO=1.0-1.333*AOL+0.5*(AOL)**2*ALOG(2.0*AOL)+0.5*AOL**2*(1.25-
10.577216)
001547      SWT1=(1.0-PZRO*(1.0-AEA/TE))/(1-PZRO*(TE-AE)/TE)
001552      SWT2=SWT1
001553 333 IP=IP
001554      GIM1(IV,IP,IT)=GIM1(IV,IP,IT)+ AE *DE/E(1)*SWT1
001576      GIN1(IV,IP,IT)=GIN1(IV,IP,IT)+ AE*DE/E(1)*SWT2
001510      GFM1(IV,IP,IT)=GFM1(IV,IP,IT)+ FE*DE/E(1)*SWT1
001622      GFN1(IV,IP,IT)=GFN1(IV,IP,IT)+ FE*DE/E(1)*SWT2
001634      DIM1(IV,IP,IT)=DIM1(IV,IP,IT)+1.0*DE/E(1)*SWT1
001646      DIN1(IV,IP,IT)=DIN1(IV,IP,IT)+1.0*DE/E(1)*SWT2
001660      SSA1(IV,IP,IT)=SSA1(IV,IP,IT)+AE
001667      SSF1(IV,IP,IT)=SSF1(IV,IP,IT)+FE
001675      SST1(IV,IP,IT)=SST1(IV,IP,IT)+TE
001702 302 CONTINUE
001703      R4 CONTINUE
001705 300 CONTINUE
001707 301 CONTINUE
001711      WRITE(9,118)
C      AVERAGE GAMMA ZERO SUB N. SEE LINES 1055 AND 636.
001715      ANVG1=ANVG/SK
C      AVERAGE GAMMA ZERO SUB F 1. SEE LINES 1053 AND 640.
001717      EAVG1=EAVG/SK
C      AVERAGE RESONANCE SPACING 1. SEE LINE 771.
001721      DAVG1=DAVG/SK/9.0
C      AVERAGE GAMMA GAMMA 1
001723      GAVG1=GAVG/SK/9.0

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C      AVERAGE GAMMA ZERO SUB N 2
001725      GNZ2=ANVG2/SK
001727      AVD22=UAVG2/SK/9.0
001731      YTI=YI/SK
001732      K=SK
001733      WRITE(9,13) K,ANVG1,EAVG1,DAVG1,GNZ2,AVD22,GAVG1,YTI
001757      1 CONTINUE
C
NOTE TEST CASE 1 TEMP. AND 1 SIGMA-P.
001761      DO 201 IV=1,9
001763      DO 202 IP=1,3
001764      DO 200 IT=1,3
001765      FSA1N(IV,IP,IT)=SSA1(IV,IP,IT)/10. /SK
001777      FST1N(IV,IP,IT)=SST1(IV,IP,IT)/10. /SK
002005      FSF1N(IV,IP,IT)=SSF1(IV,IP,IT)/10. /SK
002013      XA(IV,IP,IT)= GIM1(IV,IP,IT)/DIM1(IV,IP,IT)
002021      XE(IV,IP,IT)= GFM1(IV,IP,IT)/DIM1(IV,IP,IT)
002024      XAT(IV,IP,IT)= GIN1(IV,IP,IT)/DIN1(IV,IP,IT)
002030      XFT(IV,IP,IT)= GFN1(IV,IP,IT)/DIN1(IV,IP,IT)
002033      200 CONTINUE
002035      202 CONTINUE
002037      201 CONTINUE
002041      WRITE( 9,107) (((IV,IP,IT,XA(IV,IP,IT),XAT(IV,IP,IT), XE(IV,IP,IT
1),XFT(IV,IP,IT),FSA1N(IV,IP,IT),FST1N(IV,IP,IT),FSF1N(IV,IP,IT),
2 IT=1,3),IP=1,3),IV=1,9)
002124      2 CONTINUE
002126      STOP
002130      END

```

RMITU

'PE,FORTTRAN,LMAP

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C
C      FOIL SELF SHIELDING FACTOR FROM BELL (LA 1011 P 25 EQ 21 (IM 0)
C      AND CASE P 23 AND P 155 TABLE 284)
C      ENERGY VALUES E(M) VOLTS (PRACT) SET AND SIGMA A SUMMED OVER
C      ADJACENT E(N) FOR SIGMA A GREATER THAN 10E-4.
C      DIMENSION AB(12),AG(1000),AI(50),AKJ(50),DELT(1000),
1E(1000),SGA(1000),SGE(1000),SGT(1000),EB(73),E3(176),
2GG(1000),GE(1000),GN(1000),GT(1000),WI(1000)
C      DIMENSION SA(72),SE(72),SAT(72),SFT(72), ASA(72),ASE(72),
1EST(72),DST(72),DSA(72),QSA(72),QSE(72),QSAT(72),QSFT(72),AST(72),
2QDSA(72),QDST(72),QASA(72),QASE(72),QAST(72),QEST(72)
C      DIMENSION SG3(72),SCZ(1000),SSZ(1000),ALZ(1000),BI(1000),
1SZ(1000),SREZ(1000),SZGN1(1000),SZGN2(1000),SAD(9,3,3),
2SFD(9,3,3), SP1D(3),SP1(20),T(4),TD(3)
C      DIMENSION RIJ(60,46),WI(60),SG(72,27), SAMC(9,3,3),SFGC(9,3,3)
1,EZ(1000),A(1),HI(60)
C      DIMENSION X(1000), Y(1000),U(50),SB(72,27),SC(72,27)
C      DIMENSION AI(17),I2(3),A3(4),I4(2),A5(13),IB(5,8), A7(30,8)
C      DIMENSION VEC(1944),ENDT(72)
C      DIMENSION SPLT(300)
C      DIMENSION CHI(72),V(72),A8(30),SD(72,27)
C      COMMON RAD,G,AI,AKJ,DELT,E,GG,GN,GT,BI,RIJ,HI,EZ,HI,SIGA,SIGS,
1SIGT,SSZ,SCZ
C      COMMON I,J,K,L,M,N
100 FORMAT (5(1X,F8.0))
101 FORMAT (10(1X,F10.5) )
102 FORMAT(7F10.6/3F10.6)
103 FORMAT (1X,3I5,5F8.4,3F10.1)
104 FORMAT(3I5,8F12.5)
105 FORMAT(5F9.5)
106 FORMAT(9A6,F14.5,F8.4)
107 FORMAT(11,F8.3,7F9.3)
108 FORMAT(4F12.4)
109 FORMAT(1X,4E12.4)

```

```

110 FORMAT( 15HOT NOT IN TABLE)
111 FORMAT( 16HOSP NOT IN TABLE)
112 FORMAT(016)
112 FORMAT(2X, 3F12.1)
114 FORMAT(8F9.4)
115 FORMAT(1X,15(1X,F7.2))
116 FORMAT(17HQAOL NOT IN TABLE)
117 FORMAT(1X,10E12.4)
118 FORMAT(15,E12.4)
119 FORMAT(15,10E12.4)
120 FORMAT(415,8F13.5)
121 FORMAT (4(1X,F8.4),4(1X,F12.4))
122 FORMAT(1P6E12.5)
123 FORMAT(1X,13F10.4)
124 FORMAT(1X, 9F10.4)
125 FORMAT(6E12.6)
126 FORMAT(9A6, F12.5,E5.4)
127 FORMAT(1X,4(15,2F13.4))
128 FORMAT(54HO NO. OF SIGMA P VALUES. )
129 FORMAT(54HO SIGMA P OR FOIL THICKNESS VALJES )
130 FORMAT(12A6)

131 FORMAT(15A1,3I1,4A1,2I1,13A1,5I1,30A1)
132 FORMAT(315,16F6.2)
200 FORMAT(15,5F15.3)
201 FORMAT (4(1X,F8.4),4E9.4)
202 FORMAT (1X,15,F10.4,4F16.4)
203 FORMAT(215,6F9.4)
305 FORMAT(11,4F9.3,F9.5,F8.4)
306 FORMAT (15,2X,5E12.6)
307 FORMAT (1X,15,6F12.4)
308 FORMAT (15,2X,3E12.6)
309 FORMAT (15,4F15.4)
310 FORMAT(1X,9F14.3)
311 FORMAT(120HO AT.WT. AT.NO. RAD. NFOIL NU
1
312 FORMAT( 65HONO.RES.IO ABS.SIG. FISS.SIG. T(1) T(2) T(3)
1 T(4))
313 FORMAT( 54HO XI N(ABS) 3(1-MU) B SQ Q )
314 FORMAT( 83HO E(EV) GG(MV) GF(MV) GNO(MV)
1 CONC. G )
315 FORMAT( 84HO E(EV) SIG 0 SCAP 0
1 SIGO*GT SIGO*GT SQ)
316 FORMAT( 75HO IG-12 LOWER E A(IM) A(NR) F(IM)
1 F(NR) )
317 FORMAT( 75HOQ WTIG-12 LOWER E A(IM) A(NR) F(IM)
1 F(NR) )
318 FORMAT(1H1)
319 FORMAT(1HO)
321 FORMAT(55H IG USED NOT 18 OR 25 GROUP.CHANGE CONST IN IF NEAR 507)
C
READ(10,102) (AI(I),I=1,60)
READ (10,102) (HI(I),I=1,60)
READ (10,102) (WI(I),I=1,60)
322 FORMAT(30A1)
READ(10,102) (AKJ(J),J=1,46)
READ(10,102) ((RIJ(I,J),I=1,60),J=1,46)
READ(10,114) (E3(J),J=1,176)
C
NOTE 2E9.4 FORMAT FOR SP1(3) OR D(FOIL) AND SP1(4) OR N(FOIL)
READ(10,104) NOEL
READ(10,125) AMAS,ZA,RAD,ANFOIL,ANU
WRITE(9,319)
WRITE(9,319)
READ(10,104) NSP1
WRITE(9,319)
C
C
SP1(1) IS ALWAYS SP. SP1(2) ET SEQ. MAY BE SP OR FOILS.
READ(10,125) (SP1(I),I=1,NSP1)
WRITE(9,318)
WRITE(9,311)
C
WRITE(9,121) AMAS,ZA,RAD,ANFOIL ,ANU

```

```

C     NOTE THAT FORMAT FOR SP1 IS 6E12.4.SP1(1) IS ALWAYS SP (SIGMA SUB
C     IF FOILS ARE USED, SP1(2) AND (3) ARE FOIL THICKNESS IN CM AND
C     ANFOIL IS FOIL ATOMIC DENSITY.
C     NOTE THAT SP1(1) MUST HAVE A VALUE OVE 2000 FOR FOIL CALC. (+ C1)
C
WRITE(9,128)
WRITE(9,104) NSP1
WRITE(9,310)
WRITE(9,120)
WRITE(9,319)
WRITE(9,101) (SP1(I),I=1,NSP1)
READ(10,203) IR,ITOP,STHA,STHE,T(1),T(2),T(3),T(4)
WRITE(9,312)
WRITE(9,203) IR,ITOP,STHA,STHE,T(1),T(2),T(3),T(4)
READ(10,305) (I,EZ(N),GG(N),GE(N),GN(N),W(N),AG(N),N=1,IR)
WRITE(9,314)
WRITE(9,307) (N,EZ(N),GG(N),GE(N),GN(N),W(N),AG(N),N=1,IR)
WRITE(9,318)

C
READ(10,105) XI, RHOA, COS1,BSQ,Q
WRITE(9,313)
WRITE(9,105) XI,RHOA, COS1,BSQ,Q
WRITE(9,319)
READ(10,122) (((SAMC(I,J,K),I=1,9),J=1,3),K=1,3)
WRITE(9,123) (((SAMC(I,J,K),I=1,9),J=1,3),K=1,3)
READ(10,122) (((SFMCI(I,J,K),I=1,9),J=1,3),K=1,3)
WRITE(9,123) (((SFMCI(I,J,K),I=1,9),J=1,3),K=1,3)
C
DO 715 K=1,3
DO 715 J=1,3
L=1
DO 715 I=2,9
DO 716 M=1,2
AM=M*2
SFD(L,J,K)=(SFMCI(I,J,K)-SFMCI(I-1,J,K))*(AM-1.)/4.+SFMCI(I-1,J,K)
SAD(L,J,K)=(SAMC(I,J,K)-SAMC(I-1,J,K))*(AM-1.)/4.+SAMC(I-1,J,K)
716 L=L+1
715 CONTINUE
WRITE(9,310)
PRINT 132,((K,J,I),(SAD(I,J,K),I=1,16),J=1,3),(<=1,3)
PRINT 132,((K,J,I),(SFD(I,J,K),I=1,16),J=1,3),(<=1,3)
READ(10,108) (SP1D(I),I=1,3)
ISP1D=SP1D(1)*10.
SP1D(1)=ISP1D/10
ISP1D=SP1D(3)*10.
SP1D(3)=ISP1D/10
WRITE(9,113) (SP1D(I),I=1,3)
READ(10,108) (TD(I),I=1,3)
WRITE(9,109) (TD(I),I=1,3)
READ(10,122) (U(I),I=1,12)
WRITE(9,123) (U(I),I=1,12)
K=1
EB(1)=EXP(-U(1))*(1.0E+7)
DO 30 I = 2,12
DU = (U(I)- U(I-1))*0.25
DO 31 J = 1, 4
AJ = J
AU=U(I-1)+AJ*DU
EB(K)=EXP(-AU)*(1.0E+7)
31 K =K+1
30 CONTINUE
EB(45)=0.0189
EB(46)=0.0126
EB(47)=0.0063
EB(48)= 0.0
PRINT 117,(EB(I),I=1,48)
DO 629 I=1,27
DO 629 J=1,72
629 SG(J,I)=0.0

C
WRITE(9,318)
READ(10,104) IGTOT

```

```

WRITE(9,104) IGTOT
REWIND 3
REWIND 4
K=1
READ(10,104) ITAPE,ITOTAP
WRITE(9,104) ITAPE,ITOTAP
C ALL OF TAPE 3 WRITTEN ON TAPE 4.
IF (ITAPE-2) 552,553,553
553 DO 561 ITA=1,ITAPE
K=ITA
552 READ( 3,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A7(I,K),I=1,30)
WRITE(9,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A7(I,K),I=1,30)
READ(3,122) (CHI(I),I=1,72)
READ(3,125) (V(I),I=1,72)
READ( 3,122) ((SG(I,J),J=1,27),I=1,72)
WRITE(4,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A7(I,K),I=1,30)
WRITE(4,122) (CHI(I),I=1,72)
WRITE(4,125) (V(I),I=1,72)
WRITE(4,122) ((SG(I,J),J=1,27),I=1,72)
READ( 3,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A7(I,K),I=1,30)
WRITE(9,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A7(I,K),I=1,30)
READ(3,122) (CHI(I),I=1,72)
READ(3,125) (V(I),I=1,72)
READ( 3,122) ((SB(I,J),J=1,27),I=1,72)
WRITE(4,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A7(I,K),I=1,30)
WRITE(4,122) (CHI(I),I=1,72)
WRITE(4,125) (V(I),I=1,72)
WRITE(4,122) ((SB(I,J),J=1,27),I=1,72)
READ( 3,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A7(I,K),I=1,30)
WRITE(9,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A7(I,K),I=1,30)
READ(3,122) (CHI(I),I=1,72)
READ(3,125) (V(I),I=1,72)
READ( 3,122) ((SC(I,J),J=1,27),I=1,72)
WRITE(4,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A7(I,K),I=1,30)
WRITE(4,122) (CHI(I),I=1,72)
WRITE(4,125) (V(I),I=1,72)
WRITE(4,122) ((SC(I,J),J=1,27),I=1,72)
IF (ITAPE-2) 565,561,561
561 CONTINUE
565 IF (ITAPE-ITOTAP) 563,554,554
563 N=3*(ITOTAP-ITAPE)
564 DO 562 L=1,N
READ( 3,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A8(I),I=1,30)
READ(3,122) (CHI(I),I=1,72)
READ(3,125) (V(I),I=1,72)
READ( 3,122) ((SD(I,J),J=1,27),I=1,72)
WRITE(9,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A8(I),I=1,30)
WRITE(4,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A8(I),I=1,30)
WRITE(4,122) (CHI(I),I=1,72)
WRITE(4,125) (V(I),I=1,72)
WRITE(4,122) ((SD(I,J),J=1,27),I=1,72)
IF (ITAPE-ITOTAP) 562,554,554
562 CONTINUE
554 SIO=0.0
DO 1 N=1,IR
SREZ(N)=SQRT(EZ(N))
GE(N)=.001*GE(N)
GG(N)=.001*GG(N)+GE(N)

```

NOTE GAMMA N ZERO OR NOT.

```

GN(N)=.001*GN(N)*SREZ(N)
GT(N)=GG(N)+GN(N)
G=AG(N)
SCZ(N)=2.601*10.**6.*G *GN(N)*GG(N)/GT(N)/EZ(N)/GT(N)
SSZ(N)=SCZ(N)*GN(N)/GG(N)
ALZ(N)=4.56*10.**2./SREZ(N)
BI(N)=16.*3.1416*ALZ(N)*G *GN(N)*RAD
SZ(N)=SCZ(N)*GT(N)/GG(N)
AIO =3.1416*GG(N)/EZ(N)*SZ(N)/2.0
SIO=SIO+AIO
SZGN1(N)=SZ(N)*GT(N)
SZGN2(N)=SZ(N)*GT(N)*GT(N)
1 CONTINUE
WRITE(9,318)
WRITE(9,315)
WRITE(9,319)
WRITE(9,200) (N,EZ(N),SZ(N),SCZ(N),SZGN1(N),SZGN2(N),N=1,IR)
WRITE(9,319)
PRINT 117,SIO
WRITE(9,318)

```

```

C
SGA(1)=0.0
SGF(1)=0.0
SGT(1)=0.0

```

CHOOSE TEMPERATU

```

C
C
C
IT=1
ET=T(IT)
DO 404 N=1,IR
404 DELT(N)=2.0*SQRT(ET *EZ(N)/AMAS)
F(1)=0.0
DE=0.00063
DO 2 M=2,11
2 E(M)=F(M-1)+DE
DF=0.002
DO 6 M=12,100
6 E(M)=E(M-1)+DE
DO 20 M=101,10000
20 E(M)=F(M-1)+0.01
IG=48
PRINT 117,(E(M),M=1,91,10)

```

```

C
C
C
M=2
N=1
DO 3 ML=2,1000
SQTE=SQRT(E(M))
IF(E(M)-(EZ(N)-10.0*(GT(N)+DELT(N)))) 4,5,5
4 SGA(M)=0.159*STHA/SQTE
SGE(M)=0.159*STHE/SQTE
SGT(M)=SGA(M)+4.0*3.1416*RAD**2
3 M=M+1
5 MLOOP=1
NLOOP=712
TOT=0.0
NLO=1
NHI=4
DO 19 IM=MLOOP,NLOOP
16 IF(E(M)-EZ(NLO+3)) 7,7,8
8 NLO=NLO+1
7 IF(EZ(NHI-3)-E(M)) 10,17,17
10 NHI=NHI+1
GO TO 7
17 IF (NHI-IR) 11,11,12
12 NHI=NHI-1
11 IF (NHI-NLO) 13,13,14
14 ISTEP=M
IF(NHI-IR) 22,23,13
23 JSTEP=M+59

```

```

GO TO 24
22 JSTEP=M+97
24 DO 18MI=ISTEP,JSTEP
   SQTE=SQRT(E(M))
   S1=0.0
   S2=0.0
   S3=0.0
   DO 151 N=NLC,NHI
   G=AG(N)
   CALL VGT
   S1=S1+SIGA*W(N)
   S2=S2+SIGT*W(N)
   S3=S3+SIGA*GE(N)/GG(N)*W(N)
151 CONTINUE
   SGA(M)=0.159*STHA/SQTE+S1
   SGE(M)=0.159*STHE/SQTE+S3
   SIGT=S2+4.0*3.1416*RAD**2
   SGT(M)=0.159*STHA/SQTE+SIGT
   IF (SGT(M)-1.0E+8) 18,13,13
18 M=M+1
19 CONTINUE
13 IMAX=M-2
   PRINT 117,(SGA(I),I=1,91,10)
   PRINT 117,(SGT(I),I=1,91,10)
   PRINT 119,IMAX

```

NOTE, 1200 POINT PLOT.

```

C
C
C
M=1
LIMP=IMAX/200
LIM=200
CALL ADV(1)
DO 2000 K=1,LIMP
DO 2002 L=1,LIM
M1=M+L
X(L)=E(M1)
2002 Y(L)=SGT(M1)
CALL SPLIT(1,LIM,X,Y,42,1)
M=M+LIM
2000 CONTINUE
C
READ(10,322)((A7(I,K),I=1,30),K=1,3)
IJK=1
C
FOIL PRESENT IF SP1(3) IS LESS THAN 1.0
IF (SP1(2)-1.0) 504,405,405
504 IF1=2
   IF2=NSP1
   GO TO 406
405 IF1=1
   IF2=NSP1
C
406 DO 402 IP=IF1,IF2
   IF (SP1(2)-1.0) 505,505,506
505 SP=SP1(IP)
   DFOIL=SP1(IP)
   GO TO 507
506 SP=SP1(IP)
C
ITOP IS BOTTOM OF TOP RES. GROUP.
C
507 IG=48
   IRES=IG
   DO 501 I=1,IG
   QSFT(I)=0.0
   QSAT(I)=0.0
   QSE(I)=0.0
   QSA(I)=0.0
   QDSA(I)=0.0001
   QDST(I)=0.0001
   QASA(I)=0.0
   QASE(I)=0.0
   QAST(I)=0.0

```

```

      QEST(I)=0.0
      SA(I)=0.0
      SE(I)=0.0
      SAT(I)=0.0
      SFT(I)=0.
      ASA(I)=0.0
      ASF(I)=0.0
      AST(I)=0.0
      EST(I)=0.0
      DST(I)=0.0001
501 DSA(I)=0.0001
      Q1=1000.

C
      TOT=0.0
      DO 303 M=2,IMAX
304 IF (EB(IG-1)-E(M)) 301,301,302
301 SA(IG)=ASA(IG)/DSA(IG)
      SF(IG)=ASE(IG)/DSA(IG)
      SAT(IG)=AST(IG)/DST(IG)
      SFT(IG)=EST(IG)/DST(IG)
      QSA(IG)=QASA(IG)/QDSA(IG)
      QSE(IG)=QASE(IG)/QDSA(IG)
      QSFT(IG)=QEST(IG)/QDST(IG)
      QSAT(IG)=QAST(IG)/QDST(IG)
      IG=IG-1
      IF (IG-1) 320,320,304
302 DSA1=(E(M+1)-E(M-1))/(E(M)*(1.0+SGA(M)/SP))
      IF (DSA1-3.0) 3002,320,320
3002 DST1=(E(M+1)-E(M-1))/(E(M)*(1.0+SGT(M)/SP))
      C1=(BSQ/(COS1*SP*RHOA)+RHOA*SGA(M))*(E(M)-E(M-1))/(2.0*
      1XI*(RHOA*(SP+SGT(M)))*E(M))
      IF (F(M)-0.005) 801,802,802
801 TWT=E(M)*EXP(-E(M)/ET)
802 TWT=TWT
803 Q2=(1.0+C1)/(1.0-C1)*Q1*TWT
      IF (Q2-0.0) 900,901,901

C
      Q2=Q1
      Q2 NEGATIVE PRINT.
900 Q2=Q1
901 IF (SP1(2)-1.0) 407,408,408
408 QASA(IG)=QASA(IG)+SGA(M)*Q2*DSA1
      QAST(IG)=QAST(IG)+SGA(M)*Q2*DST1
      QASE(IG)=QASE(IG)+SGE(M)*Q2*DSA1
      QEST(IG)=QEST(IG)+SGE(M)*Q2*DST1
      QDSA(IG)=QDSA(IG)+Q2*DSA1
      QDST(IG)=QDST(IG)+Q2*DST1
      ASA(IG)=ASA(IG)+SGA(M)*DSA1
      ASE(IG)=ASE(IG)+SGE(M)*DSA1
      AST(IG)=AST(IG)+SGA(M)*DST1
      EST(IG)=EST(IG)+SGE(M)*DST1
      DSA(IG)=DSA(IG)+DSA1
      DST(IG)=DST(IG)+DST1
      TOT=TOT+(DSA1*SGA(M)/(1.0+SGA(M)/SP))
      IF (ASA(IG)-1.0E+7) 325,326,325
326 GO TO 2001
325 GO TO 323

C
      FOIL SECTION.
407 DSA1=(E(M+1)-E(M-1))/E(M)
      IF (DSA1-3.0) 3001,320,320
3001 DST1=DSA1
C
      POFACF CALC. FROM E3 TABLE
      AOL= ANFOIL*DFOIL*SGA(M)
      IF (AOL-1.74) 409,410,410
410 PZERO=1.0-0.5/AOL
      GO TO 411
409 NTAB=AOL*100.0+1.0
      ANTAB=NTAB
      E3INT=(AOL*100.0+1.0-ANTAB)*(E3(NTAB)-E3(NTAB+1))*(-1.)+E3(NTAB)
      IF (E3INT-0.0) 902,902,903
902 PRINT 19, M,AOL,ANTAB,E3(NTAB),E3(NTAB+1)
903 PZERO=1.0-(0.5-E3INT)/AOL
411 PWT = (1.0-PZERO*(1.0-(SGT(M)-SGA(M))/SGT(M)))*SGA(M)

```

```

      IF (PWT - 0.0) 904,904,905
904 PWT = 0.0
905 AST(IG) = AST(IG) + Q2 * PWT * DST1
      TOT = TOT + PWT * DST1
      DST(IG) = DST(IG) + Q2 * DST1
      EST(IG) = EST(IG) + SGE(M) * Q2 * PWT * DST1 / SGA(4)
323 Q1 = Q2 / TWT
303 CONTINUE
      WRITE(9,319)
      WRITE(9,319)
      PRINT 117, TOT

C
320 QSA(IG) = QASA(IG) / QDSA(IG)
      QSE(IG) = QASE(IG) / QDSA(IG)
      QSFT(IG) = QEST(IG) / QDST(IG)
      QSAT(IG) = QAST(IG) / QDST(IG)
      SA(IG) = ASA(IG) / DSA(IG)
      SE(IG) = ASE(IG) / DSA(IG)
      SAT(IG) = AST(IG) / DST(IG)
      SFT(IG) = EST(IG) / DST(IG)
      WRITE(9,318)
      WRITE(9,106) (AB(I), I=1,9), SP1(IP), T(IT)
      WRITE(9,316)
      WRITE(9,306) (I, EB(I), SA(I), SAT(I), SE(I), SFT(I), I=1, IRES)
      WRITE(9,319)
      IF (SP1(3) - 1.0) 611,611,612
612 WRITE(9,317)
      WRITE(9,306) (I, EB(I), QSA(I), QSAT(I), QSE(I), QSFT(I), I=1, IRES)
611 ET = T(IT)
      IET = 10000. * ET
      ET = IET / 10000
      WRITE(9,319)
      IF (SP1(3) - 1.0) 631,631,632
631 SP = 1.0 / (2.0 * DFOIL * ANFOIL)
632 ISPD = SP * 10.
      SP = I SP / 10
      DO 701 I = 1, 16
      IETD = 10000. * TD(I)
      TD(I) = IETD / 10000
      IF (ET - TD(I)) 702,703,704
702 WRITE(9,110)
      GO TO 701
703 RT = 0.0
      ITD = 1
      GO TO 705
704 IF (ET - TD(3)) 707,706,702
706 RT = 1.0
      ITD = 2
      GO TO 705
707 CALL SEARCH (ET, TD(1), 3, ITD, MFLAG)
      RT = (ET - TD(ITD)) / (TD(ITD+1) - TD(ITD))
705 IF (SP - SP1D(1)) 708,709,710
708 WRITE(9,111)
      WRITE(9,109) SP
      WRITE(9,109) SP1D(1)
      WRITE(9,109) SP1D(3)
      GO TO 701
709 ISPD = 1
      GO TO 714
710 IF (SP - SP1D(3)) 713,712,708
712 ISPD = 2
      GO TO 714
713 CALL SEARCH (SP, SP1D(1), 3, ISPD, MFLAG)
714 D = (SAD(I, ITD, I SPD+1) - SAD(I, ITD, ISPD)) / ALOG(SP1D( ISPD+1) /
      1 SP1D( ISPD))
      C = SAD(I, ITD, ISPD) - D * ALOG(SP1D( ISPD))
      S1 = C + D * ALOG(SP)
      D = (SAD(I, ITD+1, I SPD+1) - SAD(I, ITD+1, ISPD)) / ALOG(SP1D( ISPD+1) /
      1 SP1D( ISPD))
      C = SAD(I, ITD+1, ISPD) - D * ALOG(SP1D( ISPD))
      S2 = C + D * ALOG(SP)
      SG3(I+20,3) = SG(I+20,3) - SG(I+20,1)
      SG(I+20,3) = SG(I+20,1) + SG3(I+20,3)

```



```

      SG(I+20,1)=((S2-S1)*RT+S1)
      D=(SFD(I,ITD,ISPD+1)-SFD(I,ITD,ISPD))/ALOG(SP1D(ISPD+1)/
      SP1D(ISPD))
      C=SFD(I,ITD,ISPD)-D*ALOG(SP1D(ISPD))
      S1=C+D*ALOG(SP)
      D=(SFD(I,ITD+1,ISPD+1)-SFD(I,ITD+1,ISPD))/ALOG(SP1D
      (ISPD+1)/SP1D(ISPD))
      C=SFD(I,ITD+1,ISPD)-D*ALOG(SP1D(ISPD))
      S2=C+D*ALOG(SP)
      SG(I+20,2)=2.43*((S2-S1)*RT+S1)
701 CONTINUE
C
      DO 610 I=ITCP,IGTOT
      SG3(I)=SG(I,3)-SG(I,1)
      J=I-24
      SG(I,1)=SA(J)
      SG(I,2)=ANU*SFT(J)
610 SG(I,3)=SG3(I)+SG(I,1)
C
      FWT=0.0
      AVA=0.0
      AVF=0.0
      TFWT=0.0
      DO 602 M=2,30
      AVEE=(E(M-1)+E(M))*0.5
      DE=E(M+1)-E(M)
      IF(SP1(3)-1.0) 618,618,619
618 AOL=ANFOIL*DFQIL*SGA(M)
      IF(AOL-1.74) 620,621,621
621 PZERO=1.0-0.5/AOL
      GO TO 622
620 NTAB=AOL*100.0+1.0
      ANTAB=NTAB
      E3INT=(ACL*100.0+1.0-ANTAB)*(E3(NTAB)-E3(NTAB+1))*(-1.)+E3(NTAB)
      IF(E3INT-0.0) 623,623,624
623 PRINT119,M,AOL,ANTAB,E3(NTAB),E3(NTAB+1)
624 PZERO=1.0-(0.5-E3INT)/AOL
622 PWT=(1.0-PZERO*(1.0-(SGT(M)-SGA(M))/SGT(M)))*SGA(M)
      IF(PWT-0.0) 625,625,626
625 PRINT119,M,DSA1,PZERO,SGT(M),SGA(M),AST(IG),DST(IG)
626 AOLCR=PWT
      GO TO 617
619 AOLCR=1.0
      PWT=SGA(M)
617 FWT=AVEE*EXP(-AVEE/ET)*DE/AVEE
      TFWT=TFWT+FWT
      AVA=AVA+(SGA(M)+SGA(M+1))/2.0*FWT*PWT /SGA(M)
      AVF=AVF+(SGE(M)+SGE(M+1))/2.0*FWT*PWT /SGA(M)
602 CONTINUE
      SG(72,1)=AVA/TFWT
      SG(72,2)=ANU*AVF/TFWT
      SG(72,3)=SG(72,1)+SG(72,4)
600 WRITE(9,318)
C
      CALCULATE OCTAL NUMBER OF ELEMENT.
      K=ITAPE
      N=NOEL
      IB(1,K)=N/4096
      N=N-4096*IB(1,K)
      IB(2,K)=N/512
      IB(3,K)=(N-IB(2,K)*512)/64
      IB(4,K)=(N-IB(2,K)*512-IB(3,K)*64)/8
      IB(5,K)=N-IB(2,K)*512-IB(3,K)*64-IB(4,K)*8
      N=NOEL+1
      WRITE(9,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(IB(I,K),I=1,5),(A7(I,IJK),I=1,30)
      WRITE(9,319)
      WRITE(4,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(IB(I,K),I=1,5),(A7(I,IJK),I=1,30)
      WRITE(4,122) CHI(I),I=1,72)

```

```

WRITE(4,125) (V(I),I=1,72)
PRINT 119,(I,(SG(I,J),J=1,10),I=1,72)
WRITE(4,122) ((SG(I,J),J=1,27),I=1,72)
IB(1,K)=N/4096
N=N-4096*IB(1,K)
IB(2,K)=N/512
IB(3,K)=(N-IB(2,K)*512)/64
IB(4,K)=(N-IB(2,K)*512-IB(3,K)*64)/8
IB(5,K)=N-IB(2,K)*512-IB(3,K)*64-IB(4,K)*8
N=NOEL+2
WRITE(4,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(IB(I,K),I=1,5),(A7(I,IJK),I=1,30)
WRITE(4,122) (CHI(I),I=1,72)
WRITE(4,125) (V(I),I=1,72)
WRITE(4,122) ((SB(I,J),J=1,27),I=1,72)
IB(1,K)=N/4096
N=N-4096*IB(1,K)
IB(2,K)=N/512
IB(3,K)=(N-IB(2,K)*512)/64
IB(4,K)=(N-IB(2,K)*512-IB(3,K)*64)/8
IB(5,K)=N-IB(2,K)*512-IB(3,K)*64-IB(4,K)*8
WRITE(4,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(IB(I,K),I=1,5),(A7(I,IJK),I=1,30)
WRITE(4,122) (CHI(I),I=1,72)
WRITE(4,125) (V(I),I=1,72)
WRITE(4,122) ((SC(I,J),J=1,27),I=1,72)
ITOTAP=ITOTAP+1
NOEL=NOEL+3
IJK=IJK+1
402 CONTINUE
C
GO TO 2001
214 WRITE(9,321)
2001 WRITE(9,318)
READ(10,131) ENDT
WRITE(9,131) ENDT
WRITE(4,131) ENDT
END FILE 4
REWIND 3
REWIND 4
WRITE(9,118) ITOTAP
RETURN
END

```

PE.FORTRAN

```

SUBROUTINE VGT
  DIMENSION AI(60),AKJ(50),DELT(1000),E(10000),GG(1000),GN(1000),
  1GT(1000),BI(1000),RIJ(60,46),WI(60),EZ(1000),A(1),HI(60)
  DIMENSION SSZ(1000),SCZ(1000)
  COMMON RAD,G,AI,AKJ,DELT,E,GG,GN,GT,BI,RIJ,WI,EZ,HI,SIGA,SIGS,
  1SIGT,SSZ,SCZ
  COMMON I,J,K,L,M,N
101 FORMAT (10(1X ,E10.5) )
104 FORMAT(3I5,8F12.4)
119 FORMAT(I5,10E12.4)
  A(1)=GT(N)/(2.0*DELT(N))
C
  IF (AI(60)-A(1))3,2,2
2  CALL SEARCH (A(1),AI(1),60,I,MFLAG)
  AR=(A(1)-AI(I))/(AI(I+1)-AI(I))
  H=HI(I)-AR*(HI(I)-HI(I+1))
  BK=K
  X=WI(I)+AR*(WI(I+1)-WI(I))
  V=(E(M)-EZ(N))/DELT(N)
  AK=V/X*.99999
202 IF (AK-0.0) 5,7,7
  5 AK=-AK
  7 IF (20.-AK) 8,10,10
  8 IF (AK-40.0) 16,16,17
  17 R=0.0
  GO TO 9
  16 R=RIJ(I,46)*EXP(-0.14*(AK-20.))
  GO TO 9
  10 CALL SEARCH (AK ,AKJ(1),46,J,MFLAG)
  AKR=(AKJ(J)-AK)/(AKJ(J)-AKJ(J+1))
204 RKM=RIJ(I,J)-AKR*(RIJ(I,J)-RIJ(I+1,J))
  RKP=RIJ(I,J+1)-AKR*(RIJ(I,J+1)-RIJ(I+1,J+1))
  R=RKM-(RKM-RKP)*AKR
  9 HFN=R*H
206 PSI=1.772*A(1)*HFN
  AK =1.00001*V/X
  IF (AK-0.0) 20,20,21
  20 AK=-AK
  21 IF (20.-AK) 11,12,12
  11 IF (AK-40.) 18,18,19
  19 R=0.0
  210 GO TO 13
  18 R=RIJ(I,46)*EXP(-0.14*(AK-20.))
212 GO TO 13
  12 AKR=(AKJ(J)-AK)/(AKJ(J)-AKJ(J+1))
  PKM=RIJ(I,J)-AKR*(RIJ(I,J)-RIJ(I+1,J))
  RKP=RIJ(I,J+1)-AKR*(RIJ(I,J+1)-RIJ(I+1,J+1))
  R=RKM-(RKM-RKP)*AKR
  13 HFN=R*H
  PSIP=1.772*A(1)*HFN
  DPDX=A(1)/X*(PSI-PSIP)/(0.00002*AK)
  PHI= .5/(A(1)**2.0)*DPDX+2.0/GT(N)*(E(M)-EZ(N))*PSI
  SIGS=SSZ(N)*PSI
  SIGA=SCZ(N)*PSI*SQRT(EZ(N)/E(M))
  SIGT= SIGA+SIGS +BI(N)/(2.0*GT(N))*PHI
214 IF (SIGT) 14,15,15
  14 SIGT=SIGA
  15 GO TO 4
  3 SIGT=2.6*10.**6.*G/E(M)*GN(N)*GG(N)/GT(N)/GT(N)/(1.+4./GT(N)
  1/GT(N)*(E(M)-EZ(N))**2.0) *(SQRT(EZ(N)/E(M))+GN(N)/GG(N))
  SIGA=2.6*10.**6.*G/E(M)*GN(N)*GG(N)/GT(N)/GT(N)/(1.+4./GT(N)
  1/GT(N)*(E(M)-EZ(N))**2.0) * SQRT(EZ(N)/E(M))
  4 RETURN
END

```

```

PROGRAM ADLER (INPUT,TAPE 10=INPUT,OUTPUT,
1TAPE 9=OUTPUT,FILM)

C
C ADLER MULTILEVEL-MULTICHANNEL FORMALISM.
C NATL. TOPICAL ON REACTOR PHYSICS IN RES. AND TH. REGION
C FEB. 7-9 1966 SAN DIEGO.
C DEFINITIONS:
C SIG(X) AT E. CNST=6.52X105 BARNS EV. V(J)=NU(J),
C GA=G(J,X), U(J)=MU(J), E=ENEGY(VOLTS), GT=G(J,T),
C H=H(J,X), SIGP0=SIGMA SUB P ZERO
C
000003 DIMENSION SIGA(5000),SIGF(5000),SIGT(5000),E(5000),
1V(100),GT(100),U(100),HT(100), A(100),R(100),C(100),D(100),
2GF(100),HF(100),SGFV(1000)
000003 100 FORMAT(6E12.4)
000003 101 FORMAT(1X,1P10E12.4)
000003 102 FORMAT(6I5)
000003 103 FORMAT(15,1P10E12.4)
000003 READ(10,102) NRES
000011 DO 1 I=1,NRES
000013 READ(10,100) U(I),V(I),A(I),R(I),C(I),D(I)
000032 WRITE(9,100) U(I),V(I),A(I),R(I),C(I),D(I)
000052 GT(I)=A(I)*V(I)
000055 HT(I)=B(I)*V(I)
000057 GF(I)=C(I)*V(I)
000061 HF(I)=D(I)*V(I)
000063 1 CONTINUE
000065 DO 5 I=1,NRES.
000067 5 PRINT 103, I,U(I),V(I),GT(I),HT(I),GF(I),HF(I)
C SUM OVER J RESONANCES AT EACH E(I).
000113 E(I)=0.0
000114 AREA=4.*3.1416*0.91**2
000117 CNST=65.2
000120 DE=U(NRES)*0.001
000123 DO 2 I=2,1000
000124 E(I)=E(I-1)+DE
000127 SQTE=SQRT(E(I))
000131 SIGT(I)=0.0
000132 SIGA(I)=0.0
000133 SIGF(I)=0.0
C SUM OVER J.
000134 DO 3 J=1,NRES
000136 SIGT(I)=SIGT(I)+CNST/SQTE*(V(J)*GT(J)+(U(J)-E(I))
1*HT(J))/((E(I)-U(J))**2+V(J)**2)
000154 SIGF(I)=SIGF(I)+CNST/SQTE*(V(J)*GF(J)+(U(J)-E(I))
1*HF(J))/((E(I)-U(J))**2+V(J)**2)
000172 3 CONTINUE
000174 SIGT(I)=SIGT(I)+AREA
000177 2 CONTINUE
C
000201 SIGF(I)=SIGF(2)
000202 SIGT(I)=SIGT(2)
000204 DO 4 I=1,1000,10
000205 4 PRINT 101,E(I),SIGF(I),SIGT(I)
C
000220 CALL SPLOT(1, 200,E,SIGT,42,1)
000224 CALL SPLOT(1, 200,E,SIGF,42,1)
C DOPPLER BROADENING KERNEL NOT USED.
000230 STOP
000232 END

```

APPENDIX I. CODE FISH

Code FISH integrates the product of neutron flux spectrum, $\phi(r,E)\Delta E$, and macroscopic neutron

cross sections to obtain the averaged cross sections and total absorption values for the isotopes. (See Appendix B for group-averaged microscopic cross sections.)

```

PROGRAM FISH (INPUT,TAPE 10=INPUT, OUTPUT,TAPE 9=OUTPUT,FILM,
1PUNCH)
C  FISSION AND CAPTURE, GIVEN FLUX AND SIG.
  DIMENSION FLUX(25,50),A(12),RHON(50,30),R(12,50),
  1S1(25,30,10),SF(50,30),C(12),
  2R(50),AR(50),FR(50), F(30),SNUF(50),ANUFR(50),FD(50)
100 FORMAT(6F12.6)
101 FORMAT(1X,9E14.4)
102 FORMAT(1X,10F13.4)
103 FORMAT(5I5,5F10.4)
104 FORMAT(12A6)
105 FORMAT(1H0)
  READ(10,104) (A(I),I=1,12)
  WRITE(9,104) (A(I),I=1,12)
  READ(10,103) IGROUP,IRCOR,IRTOT,IGEOM,NOEL,RADC,RADT
  WRITE(9,103) IGROUP,IRCOR,IRTOT,IGEOM,NOEL,RADC,RADT
  AIT=IRTOT
  AIC=IRCOR
  R(1)=0.0
  DR=RADC/AIC
  DO 5 I=1,IRCOR
5  R(I+1)=R(I)+DR
  IF (IRTOT-IRCOR) 7,8,7
  7  DR=(RADT-RADC)/(AIT-AIC)
  DO 6 I=IRCOR,IRTOT
6  R(I+1)=R(I)+DR
  8  WRITE(9,105)
  READ(10,100) ((FLUX(I,J),J=1,IRTOT),I=1,IGROUP)
  WRITE(9,101) (FLUX(I,1),I=1,21)
  WRITE(9,101) (FLUX(I,1),I=1,24)
C  READ CROSS SECTIONS.
  DO 2 K=1,NOEL
  READ(10,104) (B(I,K),I=1,12)
  WRITE(9,104) (B(I,K),I=1,12)
  READ(10,100) (RHON(J,K),J=1,IRTOT)
  WRITE(9,101) (RHON(J,K),J=1,IRTOT)
  WRITE(9,105)
  READ(10,100) (SF(I,K),I=1,24)
  WRITE(9,101) (SF(I,K),I=1,24)
  WRITE(9,105)
  F(K)=0.0
  SUMWT=0.0
  SUMSIG=0.0
  DO 3 J=1,IRTOT
  AR=0.0
  FR=0.0
  FD(J)=0.0
  DO 4 I=1,IGROUP
  FR1=FR1+FLUX(I,J)*RHON(J,K)*4.189*SF(I,K)*
  1(R(J+1)**3-R(J)**3)
  FR2= FLUX(I,J)*RHON(J,K)*4.189*SF(I,K)*
  1(R(J+1)**3-R(J)**3)
  SUMSIG=SUMSIG+FR2
  FD(J)=FD(J)+FLUX(I,J)*RHON(J,K)*SF(I,K)
  SUMWT=SUMWT+FLUX(I,J)*RHON(J,K)*(R(J+1)**3-R(J)**3)*4.189
  4  CONTINUE
  FR(J)=FR1
  F(K)=F(K)+FR1
  3  CONTINUE
  PRINT 101,(FR(J),J=1,IRTOT)
  WRITE(9,105)
  PRINT 101,(FD(J),J=1,IRTOT)
  WRITE(9,105)
  AVSIG=SUMSIG/SUMWT
  PRINT 101,AVSIG
  WRITE(9,105)
  2  CONTINUE
  PRINT 101,(F(K),K=1,NOEL)
C  RETURN
  END

```

APPENDIX J. CODE COPY 1

The tape correct code COPY 1 simply changes the Master Data Tape output to DTF-IV tape format, including the proper content of the name card and X's and v's specific to the group energy bounds used in the Master Data Tape program. X_1 is the prompt neutron energy spectrum to be used in a DTF calculation ($\sum_1 X_1 = 1$), and v_1 is the average neutron velocity in the energy interval ΔE_1 .

PE,FORTTRAN,UMAP

```

        DIMENSION V(24),CHI(24,3),S(108),S1(72,27),BNAME(36)
        DIMENSION A1(15),I2(3),A3(4),I4(2),A5(13),I8(5,30),A7(30,30)
100  FORMAT(72A1)
101  FORMAT(36A1)
103  FORMAT(215)
104  FORMAT(1P6E12.5)
121  FORMAT(15A1,2I1,4A1,2I1,13A1,5I1,30A1)
        READ(10,104) ((CHI(I,J),I=1,24),J=1,3)
        READ(10,104) (V(I),I=1,24)
        WRITE(9,104) (V(I),I=1,24)
        DO 30 K=1,24
        READ(10,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1 (I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A7(I,K),I=1,30)
30  CONTINUE
        NO=1
        DO 10 K=1,19
        L=1
        IF (K-8) 1,2,1
2  READ(3,101) BNAME
        WRITE(9,101) BNAME
        DO 200 ILOOP=1,324
        READ(3,100) (S(I1),I1=1,72)
200  CONTINUE
        DO 13 M=1,3
        READ(10,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1 (I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A7(I,K),I=1,30)
        READ(10,104) ((S1(I,J),J=1,27),I=1,24)
        N=NO
        IB(1,K)=N/4096
        N=N-4096*IB(1,K)
        IB(2,K)=N/512
        IB(3,K)=(N-IB(2,K)*512)/64
        IB(4,K)=(N-IB(2,K)*512-IB(3,K)*64)/8
        IB(5,K)=N-IB(2,K)*512-IB(3,K)*64-IB(4,K)*8
        WRITE(9,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1 (I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A7(I,K),I=1,30)
        WRITE(9,104) (CHI(I,L),I=1,24)
        WRITE(4,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1 (I4(I),I=1,2),(A5(I),I=1,13),(I8(I,K),I=1,5),(A7(I,K),I=1,30)
        WRITE(4,104) (CHI(I,L),I=1,24)
        WRITE(4,104) (V(I),I=1,24)
        WRITE(4,104) ((S1(I,J),J=1,27),I=1,24)
        NO=NO+1
13  CONTINUE
        GO TO 14
2  READ(3,101) BNAME
        WRITE(9,101) BNAME
        IF(K-1) 7,5,6
5  L=2
        GO TO 7
6  IF(K-3) 7,8,9
8  L=3
        GO TO 7
9  IF(K-4) 7,11,7
11 L=3

```

```

7 DO 12 M=1,3
  N=NO
  IB(1,K)=N/4096
  N=N-4096*IB(1,K)
  IB(2,K)=N/512
  IB(3,K)=(N-IB(2,K)*512)/64
  IB(4,K)=(N-IB(2,K)*512-IB(3,K)*64)/8
  IB(5,K)=N-IB(2,K)*512-IB(3,K)*64-IB(4,K)*8
  WRITE(9,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(IB(I,K),I=1,5),(A7(I,K),I=1,30)
  WRITE(4,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(IB(I,K),I=1,5),(A7(I,K),I=1,30)
  WRITE(4,104) (CHI(I,L),I=1,24)
  WRITE(4,104) (V(I),I=1,24)
  DO 201 ILOOP=1,108
    READ(3,100) (S(IL),IL=1,72)
    WRITE(4,100) (S(IL),IL=1,72)
201 CONTINUE
C
C   WRITE LAST OF LOOP
  NO=NO+1
12 CONTINUE
14 L=1
10 CONTINUE
  L=L+1
  DO 20 K=20,24
    DO 21 M=1,3
      READ(10,100) BNAME
      N=NO
      IB(1,K)=N/4096
      N=N-4096*IB(1,K)
      IB(2,K)=N/512
      IB(3,K)=(N-IB(2,K)*512)/64
      IB(4,K)=(N-IB(2,K)*512-IB(3,K)*64)/8
      IB(5,K)=N-IB(2,K)*512-IB(3,K)*64-IB(4,K)*8
      WRITE(9,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(IB(I,K),I=1,5),(A7(I,K),I=1,30)
      WRITE(4,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(IB(I,K),I=1,5),(A7(I,K),I=1,30)
      WRITE(4,104) (CHI(I,L),I=1,24)
      WRITE(4,104) (V(I),I=1,24)
      DO 202 N=1,108
        READ(10,100) (S(I),I=1,72)
        WRITE(4,100) (S(I),I=1,72)
202 CONTINUE
      NO=NO+1
21 CONTINUE
20 CONTINUE
  K=K+1
  READ(10,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(IB(I,K),I=1,5),(A7(I,K),I=1,30)
  WRITE(9,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(IB(I,K),I=1,5),(A7(I,K),I=1,30)
  WRITE(4,131) (A1(I),I=1,15),(I2(I),I=1,3),(A3(I),I=1,4),
1(I4(I),I=1,2),(A5(I),I=1,13),(IB(I,K),I=1,5),(A7(I,K),I=1,30)
  END FILE 4
  REWIND 3
  REWIND 4
  RETURN
  END

```

APPENDIX K. CODE FABELL

This code computes the free atom thermalization model group-averaged neutron cross sections. The form of the output is suitable for isotropic scattering, four groups up and four groups down from each thermal group.

```

PROGRAM FABELL(INPUT,TAPE 10=INPUT, OUTPUT,TAPE 9=OUTPUT,FILM,
1PUNCH)
C   FREE ATOM THERMALIZATION,MODIFIED BY G.BELL 12/66. TWO CODES. ONE USES
C   USES OLD 25 GROUP FORMAT. THE OTHER USES THE HANSEN,SANDMEIER,
C   LAZARUS FORMAT FOR THE CROSS SECTION LIST TO WHICH IS ADDED
C   THE SPEED-UP CROSS SECTION TABLE OF NUMBER IG-1. THE TOTAL
C   NUMBER OF GROUPS IS IGTOT. OTHER DATA TO BE PROVIDED IS GROUP
C   BOUNDS FOR THE THERMALIZED SET AND THE SCATTERING CROSS SECTIONS.
C   AVERAGED IN THE IG-1 GROUPS.
C   THE COMPUTING TIME IS 5 MIN. RFT TEMPERATURE.
C
C   WHEN DESIRED, THE OLD CROSS SECTION FORMAT CODE USES REACTOR FLUX
C   WEIGHTING FOR THE THERMAL GROUPS.
C
  DIMENSION E(20),Z(20),FZ(999),nZ(20),SZ(20),S1(40,40),SG(30,30),
1    S(30,30),GFG(30,30),SGFG(20),T(5)
  DIMENSION AA(12)
  DIMENSION DPLSG(20)
  DIMENSION AN(999),AVZ(30),SNZ(20),V(30)
  DIMENSION FLUX(100,30),ENDFLN(30)
100  FORMAT(2I5)
101  FORMAT(7F9.4)
102  FORMAT(1X,7F9.4)
103  FORMAT(7F10.4)
104  FORMAT(1X,7F14.4)
105  FORMAT(2I5,8F14.0)
106  FORMAT(1X,10F12.4)
107  FORMAT(10F12.4)
108  FORMAT(3I5,10E10.4)
109  FORMAT(1X,10F10.4)
110  FORMAT(7F10.4)
111  FORMAT(6E12.4)
112  FORMAT(13F10.4)
113  FORMAT(11,7(F8.4,1X),F8.4/11,7(F8.4,1X),F8.4/11,7(F8.4,1X),F8.4/
114  FORMAT(1X,15E8.3)
115  FORMAT(15)
116  FORMAT(9A6,F10.4)
117  FORMAT(1X,13F9.4)
118  FORMAT(1H1)
119  FORMAT(6E12.4)
120  FORMAT(6F12.0)
121  FORMAT(6F12.4)
122  FORMAT(30A8)
123  FORMAT(1X,9E14.6)
124  FORMAT(1X,13F10.4)
125  FORMAT(6E12.4)
126  FORMAT(6(3X,E9.4))
127  FORMAT(1X,10E12.4)
  CONST=SQRT(1.6*10.**12/1.672)/100000000.
  WRITE(9,118)
C
C   IF IWGT=WEIGHTING CONSTANT IS 1, USE FLUX WEIGHT , A FLUX DUMP
C   FROM A REACTOR. IF 2 USE A MAXWELLIAN VELOCITY DISTRIBUTION WEIGHT.
C
  READ(10,108) IWGT
  WRITE(9,108) IWGT
  IF(IWGT-1) 250,250,251
250  READ(10,108) IGTOT,IR,IFLUX
  WRITE(9,108) IGTOT,IR,IFLUX
  READ(10,119) ((FLUX(I,J),I=1,10),J=1,IGTOT)
  WRITE(9,119) ((FLUX(I,J),I=1,10),J=1,IGTOT)

```



```

C
C
C      NUMBER OF THERMAL GROUP BOUNDS.
251 READ(10,100) IG ,IGTOT
    WRITE(9,100) IG ,IGTOT
    READ(10,103) (E(I),I=1,IG)
    WRITE(9,104) (E(I),I=1,IG)

C
C      ENERGY LIMITS
C
C      AT. WT.. (1-MU(AVE), T(I),I=1,3
    READ(10,101) AM,AMU,(T(I),I=1,3)
    WRITE(9,121) AM,AMU,(T(I),I=1,3)

C
    READ(10,110) (SZ(I),I=2,IG)
    WRITE(9,124) (SZ(I),I=2,IG)

C
C      ELEMENT NAME
    READ(10,116) (AA(I),I=1,9)
    WRITE(9,116) (AA(I),I=1,9)
                                     WRIT

C
C      STANDARD NEUTRON CROSS SECTION SET TO BE MODIFIED.
C      GROUP AVERAGE SCATTERING CROSS SECTION (BARNS) FROM LAZARUS.
    READ(10,111) ((SG(I,J),J=1,13),I=1,IGTOT)
    WRITE(9,124) ((SG(I,J),J=1,13),I=1,IGTOT)

C
C      OUTER LOOP FOR ELEMENT TEMPERATURE.
    DO 1002 IA=1,3
    DO 29 J=1,30
    DO 29 I=1,30
    GFG(I,J)=0.0
    S(I,J)=0.0
29 S1(I,J)=0.0
    VT=T(IA)
    DO 1001 I=1,IG
    SGFG(I)=0.0001
1001 Z(I)=SQRT(2.0*E(I))
    WRITE(9,104) (Z(I),I=1,IG)
    BETA=SQRT(0.5/VT)
    RTM=SQRT(AM)
    THETA=(AM+1.0)/(2.0*RTM)
    A=THETA**2.0
    C=BETA*THETA
    D=BETA*(RTM-THETA)
    G=BETA**2.0
    WRITE(9,117) A,C,D,G
                                     WRITE
    K=2
    FZ(1)=0.0

C
C      ENDFLN IS ACTUALLY FLUX/VELOCITY.
    FLUX(IFLUX,IGTOT+1)=0.0
    ENDFLN(1)=0.0
    AN(1)=0.0
    DO 3 L=2,IG
    I=IG-L+1
    M=IGTOT-L+2
    DZ(I)=(Z(I)-Z(I+1))/40.
    SANK=0.0
    SANXZ=0.
    IF (IWGT-1) 21,21,22
21 ENDFLN(L)=FLUX(IFLUX,M)/(Z(L)+Z(L-1))
    DN=(ENDFLN(L)-ENDFLN(L-1))/40.
    DO20 J=1,40
    FZ(K)=FZ(K-1)+DZ(I)
    AN(K)=AN(K-1)+DN
    SANK=SANK+AN(K)*DZ(I)
    SANXZ=SANXZ+AN(K)*FZ(K)*DZ(I)
20 K=K+1
    GO TO 24
22 DO23 J=1,40
    FZ(K)=FZ(K-1)+DZ(I)
    AN(K)=FZ(K)**2/(SQRT(VT)**3)*EXP(-FZ(K)**2/2.0/VT)

```

```

      SANK=SANK+AN(K)*DZ(I)
      SANXZ=SANXZ+AN(K)*FZ(K)*DZ(I)
23  K=K+1
24  AVZ(I+1)=SANXZ/SANK
      V(I+1)=AVZ(I+1)*CONST
      SNZ(I+1)=SANK
3   CONTINUE
      IGR=IG-1
      IGFR=40*IGR
      PRINT 127, (DZ(I),I=1,IGR)
      PRINT 127, (FZ(I),I=2,IGFR,10)
      PRINT 127, (AN(I),I=2,IGFR,10)
      PRINT 127, (AVZ(I),I=2,IG)
      PRINT 127, (SNZ(I),I=2,IG)
      PRINT 127, (V(I),I=2,IG)
      PUNCH 126, (V(I),I=2,IG)
      ITEST=1
      WRITE(9,115) ITEST
      KMAX=40*(IG-1)
      M=KMAX
      J=IG
      DO 1 L=2,M
      I=IG
      DO 2 K=2,M
      IF (FZ(L)-FZ(K))5,5,4
4   H=1.0
      GO TO 14
5   H=-1.0
14  E11= (C*FZ(K)-D*FZ(L))
      E22=(C*FZ(K)+D*FZ(L))
      E33= (C*FZ(L)-D*FZ(K))
      E44= (C*FZ(L)+D*FZ(K))
      IF (ABS(E11)-3.0) 6,6,7
7   E1=1.0
      GO TO 15
6   E1=ERF(E11)
15  IF (ABS(E22)-3.0) 8,8,9
9   E2=1.0
      GO TO 16
8   E2=ERF(E22)
16  IF (ABS(E33)-3.0) 10,10,11
11  E3=1.0
      GO TO 17
10  E3=ERF(E33)
17  IF (ABS(E44)-3.0) 12,12,13
13  E4=1.0
      GO TO 54
12  E4=ERF(E44)
54  FG =A*FZ(K)/FZ(L)*(E1+H*E2
      1+EXP(G*(FZ(L)**2.0-FZ(K)**2.0))*(E3-H*E4))
C   SCATTERING FROM L TO K
201 IF (FZ(L)-Z(J-1)) 202,202,200
200 J=J-1
202 IF (FZ(K)-Z(I-1)) 204,204,203
203 I=I-1
204 DI=FG*DZ(I-1)*DZ(J-1)*AN(L)
      IF (DI-1.0E+6) 205,205,2
205 GFG(I,J)=GFG(I,J)+DI
2   CONTINUE
1   CONTINUE
      DO 18 J=2,IG
      DO 18 I=2,IG
18  GFG(I,J)=SZ(J)/AVZ(J)/SNZ(J)*GFG(I,J)
      WRITE(9,117) ((GFG(I,J),I=2,IG),J=2,IG)
      GFG(3,2)=SG(IG+2,10)
      DO 30 I=2,IG
      DO 30 J=2,IG
30  S(I,J)=GFG(J,I)
      I=1
      DO 31 I=2,IG
      K=I+IGTOT-IG

```

WRITE

```

DO 31 J=2,IG
L=J-1+15
31 S1(K,L)=S(I,J)
LIM1=IGTOT-IG+2
LIM2=IGTOT
WRITE(9,109) ((S1(K,L),L=11,20),K=LIM1,LIM2)
DO 32 K=LIM1,LIM2
X=0.0
DO 33 L=3,11
X=X+S1(K,L)
33 S1(K,L)=0.0
32 S1(K,11)=X
DO 34 K=LIM1,LIM2
X=0.0
DO 35 L=19,27
X=X+S1(K,L)
35 S1(K,L)=0.0
34 S1(K,19)=X
DO 36 K=LIM1,LIM2
DO 36 L=11,19
M=L-6
N=K+L-15
36 SG(N,M)=S1(K,L)
WRITE(9,109) ((S1(K,L),L=11,20),K=LIM1,LIM2)
DO 40 I=LIM1,LIM2
J=I-IGTOT+IG
SG(I,3)=AMU*SZ(J)+SG(I,1)
SG(I,4)=0.0
X2=0.0
DO 37 L=11,19
37 X2=X2+S1(I,L)
SG(I,9)=AMU*SZ(J)-X2+S1(I,15)
40 SG(I,2)=0.0
DO 500 I=1,IGTOT
IF(SG(I,9)-0.0) 501,500,500
501 SG(I,3)=SG(I,3)-SG(I,9)
SG(I,9)=0.0
500 CONTINUE
DO 506 I=1,5
DO 506 J=1,13
506 S1(I,J)=0.0
DO 503 I=1,IGTOT
K=I+5
DO 503 J=1,13
503 S1(K,J)=SG(I,J)
L=IGTOT+5
DO 502 I=6,L
S1(I,4)=0.0
DO 502 J=1,4
502 S1(I,4)=S1(I,4)+S1(I+J-5,J+4)
DO 507 I=1,IGTOT
507 SG(I,4)=S1(I+5,4)
WRITE(9,116) (AA(I),I=1,9), T(IA) WRITE
PUNCH 116, (AA(I),I=1,9), T(IA)
WRITE(9,124) ((SG(I,J),J=1,13),I=1,IGTOT)
PUNCH 119, ((SG(I,J),J=1,13),I=1,IGTOT)
CALL CLOCK1(T)
PRINT 120,T
PRINT 122,T
ITEST=6
WRITE(9,115) ITEST WRITE
1002 CONTINUE
STOP
END

```

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