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CODE ALICE B & B A PRECOMPOUND/EVAPORATION
FISSION CODE EMPHASIZING NEUTRON YIELDS

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Livermore
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CODE ALICE B & B

A PRECOMPOUND/EVAPORATION/FISSION CODE EMPHASIZING NEUTRON YIELDS

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ABSTRACT

This report describes a version of the code ALICE/LIVERMORE 82 which has been modified to output neutron yield information of the type one would measure in an experiment gated by measuring a fission fragment. The modified code does the same precompound/compound decay calculations as the code ALICE/LIVERMORE 82, but has additional buffers and algorithms for the neutron yield calculation. We calculate the pre-fission neutron spectra and multiplicities, and post-fission neutron spectra and multiplicities, predicted for measurements which are gated on fission events. In this report we describe the additional output, referring the reader to the more complete report on the original code for further details.

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I. INTRODUCTION

We are concerned with the interpretation of some experiments in which neutron spectra and/or multiplicities are measured in coincidence with fission. In particular we have modified the nuclear reactions code ALICE/LIVERMORE 82¹ (hereafter referred to as 'ALICE') to provide additional predictions for neutron spectra and multiplicities for gated fission experiments which are not available in the code in its standard form.

The ALICE code is designed to predict results of the de-excitation of nuclei formed at excitations of up to 200 (or 300) MeV, when the principal decay channels are (multiple) precompound decay, fission, or evaporation of neutrons, protons, deuterons or alpha particles. Precompound decay is treated via the hybrid/geometry dependent hybrid models,^{2,3} fission via the Bohr-Wheeler approach,⁴ and compound nucleus decay by the Weisskopf-Ewing evaporation model.⁵ More details on the code, the physics of the code, references to the physics, and descriptions of subroutines which provide most of the input to the code, are to be found in a recent report.¹ We do not attempt to duplicate that report here, but attempt to describe the new buffers and algorithms used to deduce neutron yield information relevant to fission-gated measurements.

Because many of the algorithms we have adopted for the reaction ($^{232}\text{Th} + n$)⁶ which prompted this work may need modification for other mass regions, we will refer to relevant buffers using their FORTRAN source code designations. In Section 2 we will describe the type of information that is ordinarily available from ALICE, and the general flow of the computation. This should help in understanding the descriptions of the additional output which will be given in Section 3.

II. GENERAL COMMENTS ON CALCULATIONS WITH ALICE

In Fig. 1 we have drawn boxes which might correspond to a portion of the chart of the nuclides with the composite nucleus A, Z , which is to be formed in a nuclear reaction indicated in the upper right hand corner. In the course of the de-excitation of this nucleus A, Z , any of the other residual nuclides indicated might be formed. We indicate the number of neutrons and/or protons which would be emitted either as nucleons or as part of clusters to reach a product corresponding to any column or row in the Table. We will refer to this 'organization chart' in later descriptions. All calculations are performed with a 1 MeV mesh size in energy.

The ALICE code gives the particle spectra emitted to populate any nuclide in the array shown in Fig. 1, the reaction yield for each nuclide, and the fission cross section for each nuclide. This information does not preserve relationships between particle spectra populating those nuclides which undergo fission, nor the excitation spectra of the nuclides undergoing fission. These features are necessary in estimating pre-fission and post-fission neutron yields. The manner of extracting these quantities is discussed in Section 3.

III. CALCULATION OF NEUTRON SPECTRA AND MULTIPLICITIES FOR FISSION GATED RESULTS

A. General Remarks

In Fig. 2 we show the region of Fig. 1 corresponding to nuclides which could be populated directly by the multiple precompound decay treated in the ALICE code. It is assumed that precompound decay precedes fission (this assumption is supported by analysis of actinide region excitation functions).⁷ Therefore a part of the initial composite nucleus excitation will be distributed by the precompound mechanism into the residual nuclides as indicated, while a portion will not undergo precompound decay and will form equilibrated compound nuclei. These equilibrated nuclei may either evaporate particles, or undergo fission.

If the equilibrated nucleus fissions, the fission event is preceded by zero neutrons, and the excitation energy of the nuclei with the cross section which undergoes fission is the compound nucleus excitation. Nuclear masses and fission systematics may be used to estimate the average excitation energy of each fragment, and evaporation theory may then be used to estimate the post fission neutron spectra and multiplicities. This channel is the easiest part of the total to calculate, as there are no prefission neutrons. Next we must consider other paths which may be some combination of precompound decay and/or evaporation followed by fission. These paths originate from the composite nucleus emission just referred to, and evaporation from all populations shown in Fig. 2, those daughter products, etc.

The method to be used begins by storing explicitly the cross section for fission of each nuclide A', Z' in the de-excitation cascade, as a function of

the excitation energy U of the nuclide which is calculated to decay by fission. This defines an array $FISE(Z',A',U)$ which gives a fission cross section in mb/MeV for fission from all possible nuclides and at all possible excitations U . We will use this array of fissioning species populations to deduce pre- and post- fission multiplicities and spectra.

B. Generation of Pre-fission Spectra and Multiplicities

The FISE buffer contains the cross section (mb/MeV) undergoing fission from all nuclei A', Z' of interest, as a function of excitation. We would like to convert this information first to the total neutron evaporation cross section coming from each excitation energy interval, and then convert the latter result to the total spectrum of neutrons preceding fission. If there were no precompound decay, the first goal would be accomplished by defining the buffer

$$CNEUT(U) = \sum_{A', Z'} M * FISE(Z', A', U) \quad (1)$$

where M is the number of prefission evaporation neutrons required to populate A', Z' as indicated e.g. in Fig. 1.

However the ALICE code does treat precompound decay, and so we wish to treat the emission of the first neutron from A, Z as being ambiguous as to origin (i.e. it may be either precompound or evaporation) and explicitly model its contribution to all the fissioning species, i.e. all A' for Z . For Z' not equal to Z we assume that all neutrons are evaporation in origin. For this reason, when taking the sum of Eq. (1) to define the $CNEUT(U)$ buffer, we use $M' = M-1$ when $Z' = Z$ in the sum; for $Z' \neq Z$ we use M defined as in Fig. 1. We next try to get the spectrum of the excluded first neutron by the following arguments.

For the population undergoing fission after the emission of one and only one neutron, the recovery of the prefission neutron spectrum ($PREN(\epsilon)$) from $FISE(Z', A', U)$ is trivial and direct

$$PREN(\epsilon) = FISE(Z', A-1, U) \quad (2)$$

where $U = E - B_n - \epsilon$.

For the contribution of precompound neutrons preceding fission where more than one neutron is emitted, the extraction of a precompound spectrum preceding fission is less direct. At present we make the simple assumption that each evaporated neutron following emission of the first neutron (of whatever origin) removes its binding energy plus 2 MeV in kinetic energy. We therefore deduce the kinetic energy of the one 'special' neutron in the xn channel as the difference between $(B_n + 2) \cdot (n-1)$ MeV removed from the total excitation energy, and the fissioning species excitation less binding energy of the last neutron,

$$\text{PRENM}(\epsilon) = \sum_{A', Z} \text{FISE}(Z, A', U) \quad (3)$$

where

$$\epsilon = E_{\text{CN}} - U - \sum_{A=A_{\text{CN}}}^{A=A'-1} (B_n + 2) \quad (4)$$

The most obvious shortcoming, and first area where improvements should be made involves failure of these algorithms to adequately treat multiple precompound decay. This failure should start to become important as multiple precompound decay becomes important. The output of this code divides the cross section between single and multiple precompound decay; careful study therefore signals a warning as to when this approximation becomes poor or invalid. The effect of the approximation, as it becomes poorer, will be to overestimate the evaporation neutron intensity (multiplicity) and to underestimate the precompound contribution.

We integrate (as additional information) the cross section for the emission of one neutron which is followed by fission (SUM2N) and the total of

all other precompound neutrons followed by evaporation and eventually by fission (SUMPN).

To this point we have described the extraction of a buffer CNEUT(U) giving the total mb/MeV of prefission evaporation neutrons versus the excitation energy of the residual nucleus following their emission and two buffers giving the spectrum of the first neutron preceding fission (PREN(ϵ)) and of the first neutron preceding additional neutron evaporation and fission (PRENM(ϵ)). We wish next to convert the CNEUT(U) spectrum to an evaporated neutron spectrum (EVAP(ϵ)) so that we have three neutron spectra buffers which should, as a sum, give the total average prefission neutron spectrum. The sum over this spectrum divided by the reaction cross section will give the average prefission neutron multiplicity.

The conversion from spectrum of species which have emitted neutrons to neutron spectra is accomplished by multiplication by a unity normalized evaporation spectrum library which is calculated from evaporation theory and stored according to the excitation energy of the residual nucleus, with a 10 MeV mesh size. Improved accuracy for these algorithms can be achieved in a straightforward manner.

The library spectra are defined by

$$\text{PREFIS}(\epsilon, U) = \frac{\sigma_n(\epsilon) * \rho(U')}{\sum_{\epsilon=0.1} \sigma_n(\epsilon) \rho(U')} \quad (5)$$

where $U' = U - \epsilon$, where the $\rho(U')$ represent the Fermi gas level density expressions used in ALICE, and $\sigma_n(\epsilon)$ are the inverse reaction cross sections (which are generated in the code either from an optical model subroutine or from a classical sharp-cutoff algorithm). The library PREFIS

spectra, since they are unity normalized, may be multiplied by a cross section at a given excitation and will return that cross section distributed over a 10 MeV neutron kinetic energy range. As an aid to seeing if the number of energies (10) per library entry and the parent excitation mesh (10 MeV) are adequate, the library is printed in its entirety.

Having formed the library of neutron evaporation spectra versus excitation energy, the prefission multiplicity-times-cross section spectra (CNEUT(U)) are converted into total prefission neutron spectra,

$$EVAP(\epsilon) = \sum_{U=1}^{\max} CNEUT(U) * PREFIS(\epsilon, U) \quad (6)$$

where U is the excitation of the nucleus undergoing fission rounded upwards to the nearest 10 MeV increment, i.e. excitations from 0 to 10 MeV are treated by library entry for 10 MeV, 10 to 20 MeV by entry for 20 MeV, etc.

Having created the total prefission evaporation spectrum, EVAP(ϵ), we also sum this result over all neutron kinetic energies ϵ , to produce the variable SUMN.

This completes the description of prefission neutron spectra and multiplicities as treated within the modified ALICE code, which we refer to as ALICE B&B. We next describe the treatment of the post-fission neutron multiplicities and spectra.

C. Post fission neutrons

The FISE(Z', A', U) buffer gives the spectrum of species undergoing fission for all A', Z' of interest. Algorithms of various degrees of sophistication could be applied quite easily to the question of post fission neutrons, beginning with the data in the FISE buffer. One could consider energy dependent A and Z distributions for each fissioning species, apply fission systematics to each distribution to get fragment excitations, and then apply evaporation models to deduce neutron spectra and multiplicities. Since calculations are performed over each entrance channel partial wave, the S-wave approximation¹ could be adopted to give enhanced γ ray de-excitation due to rotational energy. We have implemented a simpler approach, which applies only in the mass region $A \approx 235$ which interested us. The algorithms are easily identifiable and easily modified when greater sophistication is required.

We began with the simplification (not necessary) that the postfission neutron spectra and multiplicity to first order depend only on the excitation of the fissioning species, and not on the A,Z. This assumes that not too broad a range of A,Z contribute to the total fission cross section, and that the relationship of initial excitation to fragment excitation remains fairly constant over this range. One must pay attention to the A,Z region of the particular application to decide if this is a satisfactory approximation. If the approximation is acceptable, we can sum over all species versus excitation,

$$\text{POSTF}(U) = \sum_{A', Z'} \text{FISE}(Z', A', U)$$

The POSTF(U) buffer now gives the total fission cross section as a function of the excitation undergoing fission. Each fission event will convert part of

the binding energy into fragment thermal excitation, part into fragment kinetic energy. Additional excitation (U) of the parent fissioning nucleus will for the most part go into additional fragment excitation, and therefore into additional neutron emission. The average number of neutrons emitted versus parent nucleus excitation "U" will therefore (to first order) have a linear energy dependence. Expressed in slope-intercept form, the intercept will give the average post fission neutron multiplicity for fission of a 'cold' nucleus; the slope versus excitation will reflect the average neutron binding energy plus twice the average fragment temperature.

For our case we convert the POSTF(U) buffer to one giving the post fission neutron yield cross section versus parent excitation by multiplication by an algorithm satisfactory for Th nuclei:

$$\text{POSTFN}(U) = \text{POSTF}(U) \times (0.97 + U/6) \quad (8)$$

The next goal of generating the spectrum of post fission neutrons is accomplished by multiplication of each element of the POSTFN(U) buffer by a normalized library of postfission neutron shapes, analogous to the treatment of prefission evaporation neutrons.

The posfission library which is presently used in the code, but which can easily be changed, is based on results of J. Terrell (IAEA Fission Symposium/ Salzburg 1965)⁸ and Doyas and Howerton (UCRL-50755 (1969)).⁹ The postfission library is, as for prefission, stored as a function of excitation of the fissioning nucleus:

$$\text{POSTFS}(\epsilon, U) = \frac{\epsilon^{1/2} \cdot \exp^{-\epsilon/(0.5 + 0.17\sqrt{12.5 + U/2}}}{\frac{(0.5 + 0.17\sqrt{12.5 + U/2})^{3/2}}{9.1} \sum_{\epsilon=0.1}^{\epsilon=9.1} (\text{numerator})} \quad (9)$$

where the summation in the denominator represents the sum of the expression in the numerator over ten values of neutron kinetic energy from 0.1 to 9.1 MeV. The library is stored in a 10 MeV wide mesh size on fissioning nucleus excitation, analogous to the prefission neutron library spectra.

The conversion of postfission neutrons per fission versus excitation to the postfission neutron spectrum is completed by multiplying the mb/MeV of neutrons undergoing fission from each excitation ($\text{POSTFN}(U)$) by the library function appropriate to the fissioning nucleus excitation,

$$\text{FISNUT}(\epsilon) = \sum_{U=0}^{\max} \text{POSTFN}(U) \cdot \text{POSTFS}(\epsilon, U) \quad (10)$$

This completes the discussion of the algorithms and assumptions made in extracting predictions of neutron multiplicities and spectra of pre- and post-fission neutrons which would be expected in an experiment gated on detection of a fission fragment. A few operational comments on dimension limits may be beneficial. Versions of ALICE/LIVERMORE 82 exist which are dimensioned for excitations up to 200 MeV and 300 MeV respectively. The changes discussed in this report were made to a 200 MeV version. However the various quantities related to neutron spectra are presently dimensioned only for excitation energies up to 100 MeV, and for the compound nucleus Z, Z-1, and Z-2.

Extension of these dimensions in energy and Z space is trivial, but one must be aware of the existing dimensioned limits when using the present code without alteration.

We have also indicated that mesh sizes of library spectra may easily be reduced, and algorithms may easily be altered. A better treatment of multiple precompound decay on pre-fission neutron multiplicities could be done, but the programming changes are beyond the trivial level. A listing of output for a calculation follows, with added commentary explaining the output beyond the Hollerith labels provided in the standard output.

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FIGURE CAPTIONS

Fig. 1 This figure represents the area in nuclidic mass number A and charge Z over which the ALICE code performs the fission/evaporation calculation. The range of calculation given by ' A ' and ' Z ' is provided by input variables. The compound or composite nucleus is represented by A, Z , as in a chart of the nuclides display. One neutron emission gives the daughter $A-1, Z$ etc. The number of neutrons emitted to give any product in a given column is shown at the top of the figure. The number of protons to populate any product in a given row is shown in the right hand column. Several boxes have been identified by their ' A, Z ' designations.

Fig. 2 Representation of the nuclides which may be populated directly by precompound decay in the ALICE code. Part of the initial composite nucleus population is shown to form an equilibrated compound nucleus (rather than undergoing precompound decay) which may undergo fission, or evaporate light particles. The daughter nuclides populated by precompound decay may similarly evaporate light particles or may undergo fission.

$\nu = a$	3	2	1	0	p
A-a Z		A-2 Z	A-1 Z	A, Z	0
			A-2 Z-1	A-1 Z-1	1
				A-2 Z-2	2
				A-a Z-z	z

Figure 1

Figure 2

SAMPLE OUTPUT

EVAPORATION CODE ALICE88

THIS CODE IS DESCRIBED IN REPORT ALICE88 LLNL/1981

USERS SENDING A LETTER TO M.BLANN WILL BE ADVISED OF FUTURE CORRECTIONS OR IMPROVEMENS TO THIS CODE.
MAILING ADDRESS IS

M.BLANN

E DIVISION-L485

LAWRENCE LIVERMORE LABORATORY

LIVERMORE, CALIFORNIA, 94550, USA
PORTIONALLY WITH RLOM VS. J

FISSION BARRIERS USER SUPPLIED AND SCALED PRO

6.3	6.3	6.3	6.3	6.3	6.3	6.3	6.3	6.3
7.7	7.7	7.7	7.7	7.7	7.7	7.7	7.7	7.7
8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5

TH232+NEUTS

BARRIER=LIQUID DROP BARRIER*1.000

QVAL = 0.0 AP = 1.0 AT = 232.0 ZP = 0.0 ZT = 90.0
CID = 1.1 NA = 8 NZ = 3 MC = 10 MP = 1 INVER = 0 IKE = 4 IPCH = 2 PLO = 9.0 KPLT = 0 M3 = 3
QVAL FROM MYERS SWIATECKI MASS FORMULA = 5.0

MASS OPTIONS

LIQUID DROP WITH SHELL CORRECTION TERM

WITH PAIRING, LEVEL DENSITY PAIRING SHIFT CALC. FROM MSL FORMULA AND APPLIED IN BACKSHIFTED CONVENTION

WITHOUT PAIRING, I.E. LEVEL DENSITY PAIRING SHIFT ABSORBED IN BINDING ENERGIES

QVAL USED (CALCULATED FROM MASS TABLE) = 4.8

BINDING ENERGIES CALCULATED FROM MASS TABLE WHERE POSSIBLE.

BINDING ENERGIES AND LEVEL DENSITY GROUND STATE SHIFTS USED

IZ = Z-INDEX OF NUCLEUS, IA = A-INDEX OF NUCLEUS IN PROGRAM ISOTOPE TABLE

MSL = CALCULATED BY MYERS SWIATECKI LYSEKILL MASS FORMULA

TAB = TAKEN FROM 1971 MASS TABLE

INP = PROVIDED BY USER

ABE = ABSORBED IN BINDING ENERGY

IZ	IA	NEUTRON	PROTON	ALPHA	DEUTERON	SOURCE	GS SHIFT	SOURCE
1	1	4.78669	7.70692	-3.50736	10.17839	TAB	-72063	MSL
1	2	6.43669	7.75202	-4.08126	11.54667	TAB	0.00000	MSL
1	3	5.12079	7.23702	-4.20226	10.01440	TAB	-72375	MSL
1	4	6.79129	7.14792	-4.77066	11.37463	TAB	0.00000	MSL
1	5	5.24079	6.68332	-5.16866	9.80989	TAB	-72690	MSL
1	6	7.11999	6.38122	-5.52006	11.11955	TAB	0.00000	MSL
1	7	5.45439	5.70792	-6.14556	9.30636	TAB	-73010	MSL
1	8	7.18569	5.72622	-6.44306	10.20401	TAB	0.00000	MSL
1	9	0.00000	0.00000	0.00000	0.00000		-73333	MSL
1	10	0.00000	0.00000	0.00000	0.00000		0.00000	MSL
2	1	4.82137	6.42994	-3.20439	8.97354	MSL	-1.44437	MSL
2	2	5.92169	5.93922	-3.90506	10.33272	TAB	-72375	MSL
2	3	5.03169	6.24922	-3.07506	8.70115	TAB	-1.45064	MSL
2	4	6.24669	5.51022	-4.50506	10.11916	TAB	-72690	MSL
2	5	5.02669	5.57922	-4.75006	8.50326	TAB	-1.45699	MSL
2	6	6.52269	5.10422	-5.04276	9.69177	TAB	-73010	MSL
2	7	5.39669	4.97502	-5.63806	7.61324	TAB	-1.46342	MSL
2	8	6.66469	4.47622	-5.93606	8.50896	TAB	-73333	MSL
2	9	0.00000	0.00000	0.00000	0.00000		-1.46994	MSL
2	10	0.00000	0.00000	0.00000	0.00000		-73661	MSL
3	1	4.60360	8.32960	-2.81077	10.55211	MSL	-72375	MSL
3	2	6.10056	8.10061	-3.11693	12.00200	MSL	0.00000	MSL
3	3	4.04864	7.06152	-3.60097	10.42390	MSL	-72690	MSL
3	4	6.41726	7.63534	-4.35113	11.71516	MSL	0.00000	MSL
3	5	5.04261	7.35790	-5.10041	9.95642	MSL	-73010	MSL
3	6	6.39329	7.41352	-4.07076	10.02531	TAB	0.00000	MSL
3	7	4.89739	7.01192	-5.18236	8.02080	TAB	-73333	MSL
3	8	6.49349	6.85062	-5.78906	10.11897	TAB	0.00000	MSL
3	9	0.00000	0.00000	0.00000	0.00000		-73661	MSL
3	10	0.00000	0.00000	0.00000	0.00000		0.00000	MSL
4	1	0.00000	0.00000	0.00000	0.00000		-1.45064	MSL
4	2	0.00000	0.00000	0.00000	0.00000		-72690	MSL
4	3	0.00000	0.00000	0.00000	0.00000		-1.45699	MSL
4	4	0.00000	0.00000	0.00000	0.00000		-73010	MSL
4	5	0.00000	0.00000	0.00000	0.00000		-1.46342	MSL
4	6	0.00000	0.00000	0.00000	0.00000		-73333	MSL
4	7	0.00000	0.00000	0.00000	0.00000		-1.46994	MSL
4	8	0.00000	0.00000	0.00000	0.00000		-73661	MSL
4	9	0.00000	0.00000	0.00000	0.00000		-1.47654	MSL
4	10	0.00000	0.00000	0.00000	0.00000		-73994	MSL
5	1	0.00000	0.00000	0.00000	0.00000		-72690	MSL
5	2	0.00000	0.00000	0.00000	0.00000		0.00000	MSL

OPTICAL MODEL NEUTRON INVERSE CROSS SECTIONS

INVERSE	REACTION	CROSS	SECT.	FOR E=1	TO 90MEV				
σ	σ	σ	σ	σ	σ	σ	σ	σ	σ
6384.	4885.	3469.	3849.	2978.	2999.	3084.	2955.	2889.	2835.
2815.	2798.	2758.	2685.	2638.	2597.	2593.	2572.	2553.	2524.
2494.	2478.	2456.	2448.	2439.	2427.	2413.	2398.	2387.	2378.
2371.	2361.	2348.	2333.	2318.	2306.	2298.	2292.	2285.	2275.
2264.	2251.	2237.	2225.	2215.	2207.	2199.	2191.	2181.	2178.
2158.	2147.	2137.	2128.	2119.	2111.	2103.	2094.	2084.	2074.
2064.	2055.	2046.	2037.	2029.	2021.	2013.	2005.	1996.	1988.
1979.	1971.	1962.	1954.	1947.	1939.	1931.	1923.	1916.	1908.
1908.	1892.	1884.	1876.	1869.	1861.	1853.	1845.	1837.	1830.

OPTICAL MODEL PROTON INVERSE CROSS SECTIONS

INVERSE	REACTION	CROSS	SECT.	FOR E=1	TO 90MEV				
σ	σ	σ	σ	σ	σ	σ	σ	σ	σ
85.	172.	291.	433.	582.	725.	859.	979.	1089.	1188.
1276.	1355.	1426.	1492.	1552.	1606.	1652.	1692.	1730.	1768.
1805.	1839.	1867.	1889.	1908.	1925.	1944.	1956.	1988.	2009.
2025.	2037.	2046.	2055.	2064.	2074.	2085.	2094.	2102.	2108.
2113.	2118.	2124.	2130.	2135.	2140.	2143.	2144.	2144.	2144.
2145.	2146.	2149.	2151.	2153.	2154.	2154.	2153.	2151.	2149.
2147.	2145.	2144.	2142.	2141.	2140.	2138.	2136.	2134.	2131.
2127.	2124.	2120.	2116.	2112.	2108.	2104.	2100.	2096.	2091.

OPTICAL MODEL ALPHA INVERSE CROSS SECTIONS

INVERSE	REACTION	CROSS	SECT.	FOR E=1	TO 90MEV				
σ	σ	σ	σ	σ	σ	σ	σ	σ	σ
19.	52.	116.	213.	330.	452.	571.	683.	709.	887.
978.	1052.	1141.	1215.	1284.	1348.	1409.	1466.	1519.	1558.
1617.	1662.	1704.	1744.	1782.	1818.	1852.	1885.	1915.	1944.
1972.	1997.	2022.	2045.	2067.	2087.	2106.	2123.	2139.	2153.
2165.	2176.	2184.	2199.	2192.	2190.	2185.	2175.	2168.	2142.
2121.	2098.	2073.	2048.	2023.	1998.	1973.	1948.	1924.	1908.
1876.	1853.	1831.	1809.	1787.	1766.	1746.	1726.	1706.	1687.

DELTA = 7.1 ALT = 18.1 CLD = 1.13800

LABORATORY BOMBARDMENT ENERGY = 39.2MEV ET MASS = 232.0
 COMPOUND NUCLEUS MASS = 233.0 COMPOUND NUCLEUS FORMATION Q VALUE = 4.8

JCAL = 0 START J = 0 STOP J = 0 JANG = 0

CALCULATION INCLUDES FISSION COMPETITION
 HYBRID CALCULATION HAS BEEN SELECTED

REACTION CROSS SECTION COMPUTED BY OPTICAL MODEL SUBROUTINE 2276.67 MB

PARTIAL REACTION CROSS SECTIONS, L=0 TO LMAX

10.	32.	51.	75.	95.	119.	151.	164.	206.	249.
250.	304.	277.	230.	52.	10.	2.	1.	0.	0.
0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

COMPOUND NUCLEUS ATOMIC NUMBER = 90 MASS NUMBER = 233

TOTAL REACTION CROSS SECTION = 2276.7

EXCITATION ENERGY OF COMPOUND NUCLEUS = 43.8MEV J = 20

REACTION CROSS SECTION .0

PRECOMPOUND PARAMETERS HAVE BEEN SELECTED INTERNALLY

GEOMETRY DEPENDENT HYBRID MODEL SELECTED

THE FOLLOWING PARAMETER VALUES WERE SELECTED FOR THE PRECOMPOUND CALCULATION

MEAN FREE PATH MULTIPLIER IS = 1.000

TRANSITION RATES ARE FROM NUCLEON-NUCLEON SCATTERING

TD = 3.00 EX1 = 1.34 EX2 = .66 TMX = 1.00 AV = 1.00 GAV = 1.00 COST = 0.00 GDO = 1.00 IJ = 0

EFERMI = 32.7 MEV

EFERMI = 0 MEV

PRECOMPOUND CROSS SECTIONS FOR PARTIAL WAVE L=20. ONLY

TOTAL PRECOMPOUND NEUTRON CROSS SECTION = .860E-02, TOTAL PRECOMPOUND PROTON CROSS SECTION = .331E-02

KINETIC ENERGY// OS/DE= NEUTRONS PROTONS (PRECOMPOUND DISTRIBUTION)

	1	2	3	4	5	6	7	8	9	10
0 DS/DE NEUTRONS	2.16E-04	2.22E-04	2.22E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04
PROTONS	0.	0.	0.	0.	0.	0.	0.	6.53E-05	9.00E-05	1.06E-04
10 DS/DE NEUTRONS	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04
PROTONS	1.08E-04	1.08E-04	1.08E-04	1.08E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04
20 DS/DE NEUTRONS	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04
PROTONS	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04
30 DS/DE NEUTRONS	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04
PROTONS	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04
40 DS/DE NEUTRONS	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
PROTONS	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

HYBRID FOLLOWING GDH SELECTED

EFERMI = 32.7 MEV

TOTAL PRECOMPOUND NEUTRON CROSS SECTION = .972E-02, TOTAL PRECOMPOUND PROTON CROSS SECTION = .334E-02										
KINETIC ENERGY// DS/DE = NEUTRONS (PRECOMPOUND DISTRIBUTION)										
	1	2	3	4	5	6	7	8	9	10
DS/DE NEUTRONS	2.59E-04	3.62E-04	3.58E-04	3.36E-04	3.28E-04	3.86E-04	2.94E-04	2.82E-04	2.72E-04	2.64E-04
PROTONS	0.	0.	0.	0.	0.	0.	0.	6.54E-05	9.81E-05	1.06E-04
10 DS/DE NEUTRONS	2.57E-04	2.52E-04	2.47E-04	2.43E-04	2.40E-04	2.37E-04	2.35E-04	2.33E-04	2.31E-04	2.30E-04
PROTONS	1.08E-04	1.10E-04	1.10E-04	1.11E-04	1.11E-04	1.11E-04	1.11E-04	1.11E-04	1.11E-04	1.11E-04
20 DS/DE NEUTRONS	2.20E-04	2.27E-04	2.27E-04	2.26E-04	2.25E-04	2.25E-04	2.24E-04	2.24E-04	2.24E-04	2.23E-04
PROTONS	1.10E-04	1.10E-04	1.10E-04	1.10E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04
30 DS/DE NEUTRONS	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	2.23E-04	0.
PROTONS	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	1.09E-04	0.	0.
40 DS/DE NEUTRONS	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
PROTONS	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

CSN= .116E-02 CSP= .209E-04 CSNN= .523E-02 CSPP= .201E-06 CSPN= .332E-02

TYP= 2CNCS HYB ROUTINE = 00

COMPOUND NUCLEUS (EQUILIBRIUM) CROSS SECTION THIS PARTIAL WAVE = .262E-03 MB
GAMF/GAMTOT = .314E+00

ATOMIC NUMBER	90	90	90	90	90	90	90	90	90	90
MASS NUMBER	233	232	231	230	229	228	227	226	225	224
ER XSECT S(J)	.348E+03	.604E+02	.929E+02	.126E+03	.176E+03	.399E+03	.120E+03	.719E+00		
ER XSECT J	.262E-03	.150E-03	.565E-03	.400E-03	.333E-03	.296E-03	.450E-04	.126E-07		
FISS X SECT	.026E-04	.352E-03	.003E-03	.810E-03	.192E-03	.115E-03	.160E-07	0.		
DEL SADDLE PT	.720E+01	.720E+01	.721E+01	.721E+01	.721E+01	.722E+01	.722E+01	.722E+01		
FISS BARRIER	.567E+01	.566E+01	.566E+01	.565E+01	.564E+01	.563E+01	.562E+01	.562E+01		
DEL ROTATION	.153E+01	.154E+01	.155E+01	.156E+01	.157E+01	.158E+01	.160E+01	.161E+01		
ROTATIONAL EN	.157E+01	.158E+01	.159E+01	.160E+01	.161E+01	.162E+01	.164E+01	.165E+01		
FISSION CROSS SECTION TO HERE =	.113E+04					.115E+04			.120E+01	
ER XSECT TO HERE =										
MISSING CROSS SECTION =										
ATOMIC NUMBER	89	89	89	89	89	89	89	89	89	89
MASS NUMBER	232	231	230	229	228	227	226	225	224	223
ER XSECT S(J)	.108E+02	.533E+02	.468E+02	.414E+02	.124E+02	.783E+01	.301E-01	0.		
ER XSECT J	.341E-05	.796E-03	.687E-03	.837E-03	.573E-03	.175E-03	.914E-07	0.		
FISS X SECT	.039E-06	.683E-04	.562E-04	.133E-04	.192E-05	.569E-09	0.	0.		
DEL SADDLE PT	.852E+01	.852E+01	.852E+01	.852E+01	.852E+01	.851E+01	.851E+01	.851E+01		
FISS BARRIER	.698E+01	.697E+01	.696E+01	.695E+01	.693E+01	.692E+01	.690E+01	.689E+01		
DEL ROTATION	.154E+01	.155E+01	.156E+01	.157E+01	.159E+01	.160E+01	.161E+01	.162E+01		
ROTATIONAL EN	.158E+01	.159E+01	.160E+01	.161E+01	.162E+01	.164E+01	.165E+01	.166E+01		
FISSION CROSS SECTION TO HERE =	.113E+04					.115E+04			.127E+01	
ER XSECT TO HERE =										
MISSING CROSS SECTION =										
ATOMIC NUMBER	88	88	88	88	88	88	88	88	88	88
MASS NUMBER	231	230	229	228	227	226	225	224	223	222
ER XSECT S(J)	.130E+01	.173E+01	.730E+00	.181E+00	.526E-02	.703E-01	.320E-01	.300E-03		
ER XSECT J	.212E-07	.343E-07	.229E-07	.232E-07	.254E-08	.331E-07	.159E-07	.635E-10		
FISS X SECT	.512E-09	.174E-09	.604E-09	.674E-09	.319E-09	.841E-11	0.	0.		
DEL SADDLE PT	.931E+01	.931E+01	.931E+01	.931E+01	.931E+01	.931E+01	.931E+01	.931E+01		
FISS BARRIER	.776E+01	.776E+01	.774E+01	.773E+01	.772E+01	.768E+01	.769E+01	.768E+01		
DEL ROTATION	.155E+01	.156E+01	.158E+01	.159E+01	.160E+01	.161E+01	.162E+01	.163E+01		
ROTATIONAL EN	.159E+01	.160E+01	.161E+01	.162E+01	.164E+01	.165E+01	.166E+01	.167E+01		
FISSION CROSS SECTION TO HERE =	.113E+04					.115E+04			.127E+01	
ER XSECT TO HERE =										
MISSING CROSS SECTION =										

PARTICLE SPECTRA SUMMED OVER ALL EMITTING NUCLEI (AND PARTIAL WAVES IF APPLICABLE) WITH PRECOMPOUND INCLUDED.

KINETIC ENERGY SPECTRA FROM A = 224.0 Z = 88.0 (NOTE THAT ACTUAL CHANNEL ENERGY EQUALS KE INDEX LESS 0.9 MEV)

KE	DS/DE(MB/MEV)=	NEUTRONS	PROTONS	ALPHAS	DEUTERONS
1		.127E+04	0.	0.	0.
2		.209E+04	0.	0.	0.
3		.112E+04	0.	0.	0.
4		.556E+03	0.	0.	0.
5		.317E+03	0.	0.	0.
6		.224E+03	0.	0.	0.
7		.143E+03	0.	0.	0.
8		.108E+03	.427E-01	0.	0.
9		.867E+02	.191E+00	0.	0.
10		.726E+02	.595E+00	0.	0.
11		.629E+02	.138E+01	0.	0.
12		.554E+02	.249E+01	0.	0.
13		.490E+02	.377E+01	0.	0.
14		.435E+02	.501E+01	0.	0.
15		.391E+02	.607E+01	0.	0.
16		.355E+02	.687E+01	0.	0.
17		.328E+02	.743E+01	0.	0.
18		.305E+02	.791E+01	0.	0.
19		.284E+02	.807E+01	.133E-01	0.
20		.266E+02	.823E+01	.172E-01	0.
21		.251E+02	.831E+01	.199E-01	0.
22		.237E+02	.835E+01	.199E-01	0.
23		.225E+02	.837E+01	.165E-01	0.
24		.218E+02	.838E+01	.111E-01	0.
25		.210E+02	.836E+01	.627E-02	0.
26		.203E+02	.830E+01	.309E-02	0.
27		.195E+02	.824E+01	.139E-02	0.
28		.190E+02	.817E+01	.566E-03	0.
29		.185E+02	.810E+01	.234E-03	0.
30		.180E+02	.754E+01	.698E-04	0.
31		.175E+02	.772E+01	.332E-04	0.
32		.168E+02	.749E+01	.119E-04	0.
33		.161E+02	.723E+01	.411E-05	0.
34		.153E+02	.665E+01	.138E-05	0.
35		.144E+02	.595E+01	.447E-06	0.
36		.129E+02	.403E+01	.148E-06	0.
37		.113E+02	.314E+01	.421E-07	0.
38		.810E+01	.100E+01	.121E-07	0.
39		.420E+01	0.	.332E-08	0.
40	0.	0.	0.	.862E-09	0.
41	0.	0.	0.	.209E-09	0.
42	0.	0.	0.	.471E-10	0.
43	0.	0.	0.	.966E-11	0.
44	0.	0.	0.	.177E-11	0.
45	0.	0.	0.	.280E-12	0.
46	0.	0.	0.	.362E-13	0.
47	0.	0.	0.	.339E-14	0.
48	0.	0.	0.	.102E-15	0.

FISSIONING SPECIES SPECTRUM

ATOMIC NUMBER OF FISSIONING SPECIES = 98

6

[illegible]

Spectra of species which fission

SUM/A .861E+02 .382E+03 .283E+03 .211E+03 .957E+02 .632E+02 .185E-01 0.

ATOMIC NUMBER OF FISSIONING SPECIES = 82

EXC	MAT=232	231	233	229	228	227	226	225
6	183E-08	0.	0.	0.	0.	0.	0.	0.
7	478E-03	0.	234E-02	0.	372E-03	0.	0.	0.
8	497E-02	140E-02	127E-01	101E-02	348E-02	200E-04	0.	0.
9	123E-01	612E-02	262E-01	370E-02	698E-02	686E-05	0.	0.
10	221E-01	120E-01	309E-01	597E-02	846E-02	0.	0.	0.
11	304E-01	208E-01	464E-01	901E-02	713E-02	0.	0.	0.
12	408E-01	296E-01	602E-01	130E-01	395E-02	0.	0.	0.
13	519E-01	391E-01	496E-01	180E-01	124E-02	0.	0.	0.
14	638E-01	487E-01	461E-01	235E-01	163E-03	0.	0.	0.
15	762E-01	577E-01	422E-01	284E-01	0.	0.	0.	0.
16	887E-01	654E-01	407E-01	310E-01	0.	0.	0.	0.
17	102E+00	708E-01	435E-01	299E-01	0.	0.	0.	0.
18	115E+00	725E-01	506E-01	294E-01	0.	0.	0.	0.
19	127E+00	701E-01	608E-01	673E-02	0.	0.	0.	0.
20	139E+00	635E-01	728E-01	165E-02	0.	0.	0.	0.
21	149E+00	544E-01	846E-01	498E-04	0.	0.	0.	0.

22	.155E+00	.454E-01	.936E-01	0.	0.	0.	0.	0.	0.	0.	0.
23	.157E+00	.398E-01	.931E-01	0.	0.	0.	0.	0.	0.	0.	0.
24	.150E+00	.396E-01	.696E-01	0.	0.	0.	0.	0.	0.	0.	0.
25	.134E+00	.446E-01	.215E-01	0.	0.	0.	0.	0.	0.	0.	0.
26	.108E+00	.532E-01	.318E-03	0.	0.	0.	0.	0.	0.	0.	0.
27	.767E-01	.641E-01	0.	0.	0.	0.	0.	0.	0.	0.	0.
28	.451E-01	.765E-01	0.	0.	0.	0.	0.	0.	0.	0.	0.
29	.206E-01	.923E-01	0.	0.	0.	0.	0.	0.	0.	0.	0.
30	.699E-02	.114E+00	0.	0.	0.	0.	0.	0.	0.	0.	0.
31	.158E-02	.123E+00	0.	0.	0.	0.	0.	0.	0.	0.	0.
32	0.	.596E-02	0.	0.	0.	0.	0.	0.	0.	0.	0.

SUM/A .108E+01 .131E+01 .945E+00 .193E+00 .318E-01 .268E-04 0.

ATOMIC NUMBER OF FISSIONING SPECIES = 88

EXC	MASS-231	238	229	228	227	226	225	224			
8	.472E-06	0.	.353E-08	0.	.127E-09	0.	.172E-12	0.	0.	0.	0.
9	.922E-05	.264E-06	.145E-05	.220E-08	.774E-06	.117E-07	.768E-11	0.	0.	0.	0.
10	.461E-04	.172E-04	.244E-05	.487E-07	.510E-05	.694E-06	0.	0.	0.	0.	0.
11	.100E-03	.531E-04	.158E-05	.263E-06	.137E-04	.133E-05	0.	0.	0.	0.	0.
12	.106E-03	.054E-04	.505E-06	.868E-06	.254E-04	.101E-05	0.	0.	0.	0.	0.
13	.273E-03	.101E-03	.120E-06	.231E-05	.365E-04	.405E-06	0.	0.	0.	0.	0.
14	.363E-03	.947E-04	.116E-07	.545E-05	.425E-04	.703E-07	0.	0.	0.	0.	0.
15	.446E-03	.712E-04	.859E-09	.110E-04	.403E-04	.394E-08	0.	0.	0.	0.	0.
16	.510E-03	.421E-04	.299E-07	.233E-04	.322E-04	0.	0.	0.	0.	0.	0.
17	.539E-03	.109E-04	.969E-07	.423E-04	.165E-04	0.	0.	0.	0.	0.	0.
18	.517E-03	.620E-05	.295E-06	.650E-04	.566E-05	0.	0.	0.	0.	0.	0.
19	.442E-03	.126E-05	.344E-06	.995E-04	.914E-06	0.	0.	0.	0.	0.	0.
20	.326E-03	.928E-07	.226E-05	.124E-03	0.	0.	0.	0.	0.	0.	0.
21	.198E-03	0.	.567E-05	.132E-03	0.	0.	0.	0.	0.	0.	0.
22	.925E-04	0.	.131E-04	.119E-03	0.	0.	0.	0.	0.	0.	0.
23	.316E-04	0.	.273E-04	.081E-04	0.	0.	0.	0.	0.	0.	0.
24	.740E-05	0.	.498E-04	.471E-04	0.	0.	0.	0.	0.	0.	0.
25	0.	0.	.760E-04	.110E-04	0.	0.	0.	0.	0.	0.	0.
26	0.	0.	.947E-04	0.	0.	0.	0.	0.	0.	0.	0.
27	0.	0.	.963E-04	0.	0.	0.	0.	0.	0.	0.	0.
28	0.	0.	.923E-04	0.	0.	0.	0.	0.	0.	0.	0.
29	0.	0.	.607E-04	0.	0.	0.	0.	0.	0.	0.	0.
30	0.	0.	.216E-04	0.	0.	0.	0.	0.	0.	0.	0.

SUM/A .410E-02 .491E-03 .537E-03 .777E-03 .210E-03 .353E-05 .705E-11 0.

LIBRARY SPECTRA FOLLOW VS. COMPOUND EXCITATION FIRST PREFISSION, THEN POSTFISSION, THEN CHANGE C.N. EXC

KE= 1 2 3 4 5 6 7 9 9 19

EMITTING NUCLEUS EXCITATION = 10 MEV

.278E+00	.474E+00	.182E+00	.505E-01	.127E-01	.280E-02	.508E-03	.722E-04	.723E-05	.323E-06
.235E+00	.342E+00	.207E+00	.110E+00	.554E-01	.271E-01	.130E-01	.613E-02	.207E-02	.133E-02

EMITTING NUCLEUS EXCITATION = 20 MEV

.167E+00	.414E+00	.239E+00	.106E+00	.452E-01	.102E-01	.679E-02	.233E-02	.749E-03	.227E-03
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9

.213E+00	.320E+00	.211E+00	.119E+00	.638E-01	.331E-01	.168E-01	.845E-02	.420E-02	.207E-02
EMITTING NUCLEUS EXCITATION = 30 MEV									
.122E+00	.359E+00	.250E+00	.134E+00	.706E-01	.356E-01	.169E-01	.751E-02	.320E-02	.132E-02
.196E+00	.316E+00	.213E+00	.126E+00	.707E-01	.384E-01	.205E-01	.108E-01	.561E-02	.290E-02
EMITTING NUCLEUS EXCITATION = 40 MEV									
.971E-01	.317E+00	.246E+00	.149E+00	.883E-01	.506E-01	.274E-01	.140E-01	.690E-02	.332E-02
.102E+00	.305E+00	.214E+00	.131E+00	.765E-01	.432E-01	.239E-01	.131E-01	.707E-02	.379E-02
EMITTING NUCLEUS EXCITATION = 50 MEV									
.811E-01	.285E+00	.239E+00	.156E+00	.101E+00	.627E-01	.371E-01	.208E-01	.112E-01	.597E-02
.171E+00	.296E+00	.213E+00	.136E+00	.815E-01	.475E-01	.271E-01	.153E-01	.864E-02	.473E-02
EMITTING NUCLEUS EXCITATION = 60 MEV									
.699E-01	.268E+00	.231E+00	.166E+00	.109E+00	.725E-01	.456E-01	.273E-01	.156E-01	.899E-02
.161E+00	.287E+00	.213E+00	.139E+00	.857E-01	.514E-01	.302E-01	.175E-01	.100E-01	.570E-02
EMITTING NUCLEUS EXCITATION = 70 MEV									
.610E-01	.240E+00	.222E+00	.161E+00	.115E+00	.803E-01	.531E-01	.334E-01	.203E-01	.122E-01
.153E+00	.279E+00	.212E+00	.141E+00	.895E-01	.549E-01	.330E-01	.196E-01	.115E-01	.669E-02
EMITTING NUCLEUS EXCITATION = 80 MEV									
.556E-01	.223E+00	.215E+00	.161E+00	.120E+00	.867E-01	.595E-01	.389E-01	.247E-01	.155E-01
.146E+00	.271E+00	.211E+00	.144E+00	.927E-01	.580E-01	.356E-01	.216E-01	.129E-01	.770E-02
EMITTING NUCLEUS EXCITATION = 90 MEV									
.505E-01	.209E+00	.207E+00	.161E+00	.123E+00	.919E-01	.652E-01	.441E-01	.289E-01	.187E-01
.140E+00	.265E+00	.265E+00	.145E+00	.956E-01	.609E-01	.381E-01	.235E-01	.144E-01	.870E-02
EMITTING NUCLEUS EXCITATION = 100 MEV									
.465E-01	.198E+00	.201E+00	.160E+00	.125E+00	.962E-01	.701E-01	.487E-01	.329E-01	.219E-01
.134E+00	.259E+00	.208E+00	.147E+00	.981E-01	.636E-01	.404E-01	.254E-01	.158E-01	.971E-02

KE	TOTAL NEUTRON SPECTRUM	FIRST NEUTRON	HIGHER ORDER PRECOMPOUND	PREFISSION EVAPORATION	POST FISSION EVAPORATION
1	.140E+04	.188E+02	.639E+02	.212E+03	.110E+04
2	.243E+04	.605E+02	.657E+02	.507E+03	.180E+04
3	.164E+04	.537E+02	.616E+02	.289E+03	.123E+04
4	.969E+03	.409E+02	.545E+02	.139E+03	.740E+03
5	.563E+03	.324E+02	.464E+02	.618E+02	.422E+03
6	.327E+03	.263E+02	.386E+02	.293E+02	.234E+03
7	.192E+03	.215E+02	.315E+02	.124E+02	.127E+03
8	.116E+03	.176E+02	.246E+02	.525E+01	.682E+02
9	.716E+02	.145E+02	.186E+02	.216E+01	.363E+02
10	.479E+02	.122E+02	.157E+02	.877E+00	.192E+02
11	.237E+02	.104E+02	.133E+02		
12	.203E+02	.895E+01	.113E+02		
13	.174E+02	.772E+01	.969E+01		
14	.150E+02	.666E+01	.834E+01		
15	.130E+02	.500E+01	.715E+01		

NEUTRON SPECTRA:
TOTAL, FIRST NEUTRON
HIGHER ORDER
PRECOMPOUND, ALL
PREFISSION EVAPORATION,
POST FISSION EVAPORATION

16	.118E+02	.511E+01	.586E+01
17	.658E+01	.456E+01	.402E+01
18	.692E+01	.489E+01	.282E+01
19	.619E+01	.368E+01	.250E+01
20	.553E+01	.332E+01	.221E+01
21	.494E+01	.301E+01	.193E+01
22	.438E+01	.273E+01	.163E+01
23	.388E+01	.250E+01	.138E+01
24	.299E+01	.229E+01	.781E+00
25	.210E+01	.210E+01	0.
26	.192E+01	.192E+01	0.
27	.173E+01	.173E+01	0.
28	.156E+01	.156E+01	0.
29	.138E+01	.138E+01	0.
30	.120E+01	.120E+01	0.
31	.988E+00	.988E+00	0.
32	.728E+00	.728E+00	0.
33	.792E+00	.792E+00	0.
SUM=	.791E+04	.382E+03	.494E+03 .125E+04 .578E+04

TOTAL NEUTRON C.S. COINCIDENT WITH FISSION= .791E+04

÷ BY 0.2 TO GET MULTIPLICITY.

EXC C.S. AT EXCITATION 'EXC'
EMITTING ONE NEUTRON
PRIOR TO FISSION

C.S. UNDERGOING
FISSION(CSF) VS EXC OF
FISSIONING NUCLEUS

CSF NO. OF POST FISSION
NEUTRONS AT EXCITATION
'EXC'

1	0.	0.	0.
2	0.	0.	0.
3	0.	0.	0.
4	0.	0.	0.
5	0.	0.	0.
6	0.	0.	0.
7	.126E+02	.183E+00	.280E+00
8	.146E+03	.548E+01	.115E+02
9	.807E+02	.485E+02	.912E+02
10	.688E+02	.256E+02	.617E+02
11	.611E+02	.238E+02	.611E+02
12	.567E+02	.238E+02	.628E+02
13	.555E+02	.231E+02	.667E+02
14	.554E+02	.239E+02	.729E+02
15	.539E+02	.249E+02	.799E+02
16	.501E+02	.254E+02	.857E+02
17	.453E+02	.252E+02	.891E+02
18	.423E+02	.249E+02	.919E+02
19	.431E+02	.254E+02	.978E+02
20	.471E+02	.274E+02	.110E+03
21	.517E+02	.306E+02	.120E+03
22	.545E+02	.340E+02	.147E+03
23	.533E+02	.365E+02	.164E+03
24	.474E+02	.374E+02	.174E+03
25	.384E+02	.363E+02	.174E+03
26	.385E+02	.348E+02	.169E+03
27	.276E+02	.327E+02	.160E+03
28	.289E+02	.342E+02	.181E+03
29	.316E+02	.381E+02	.208E+03
30	.385E+02	.419E+02	.235E+03
31	.238E+02	.427E+02	.246E+03
32	.110E+02	.376E+02	.223E+03
33	0.	.285E+02	.174E+03
		.235E+02	.147E+03

34	0.	.263E+02	.169E+03
35	0.	.324E+02	.213E+03
36	0.	.409E+02	.275E+03
37	0.	.537E+02	.370E+03
38	0.	.605E+02	.426E+03
39	0.	.188E+02	.136E+03
40	0.	0.	0.
41	0.	0.	0.
42	0.	0.	0.
43	0.	.861E+02	.676E+03

TOTAL PRE FISSION EVAPORATED NEUTRON CROSS SECTION = .125E+04 .125E+04

CROSS SECTION NEUTRONS COINCIDENT WITH SECOND CHANCE FISSION = .392E+03

C.S. PRECOMPOUND NEUTRONS WITH G.T. SECOND CHANCE FISSION = .494E+03
 TOTAL POST FISSION NEUTRON CROSS SECTION = .578E+04 .578E+04
 TOTAL FISSION CROSS SECTION = .113E+04

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