LA-13639-MS

SOURCES 4A: A Code for Calculating (α,n), Spontaneous Fission, and Delayed Neutron Sources and Spectra

Los Alamos
NATIONAL LABORATORY

Los Alamos National Laboratory is operated by the University of California for the United States Department of Energy under contract W-7405-ENG-36.

Edited by Maco Stewart, Group CIC-1

An Affirmative Action/Equal Opportunity Employer

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

LA-13639-MS

Issued: September 1999

SOURCES 4A: A Code for Calculating (α,n), Spontaneous Fission, and Delayed Neutron Sources and Spectra

W. B. Wilson
R. T. Perry
W. S. Charlton
T. A. Parish*
G. P. Estes
T. H. Brown
E. D. Arthur
M. Bozoian
T. R. England
D. G. Madland
J. E. Stewart

*Consultant at Los Alamos. Texas A&M University, College Station, TX 77843

Los Alamos
NATIONAL LABORATORY
Los Alamos, New Mexico 87545

TABLE OF CONTENTS

SOURCES 4A: A Code for Calculating (α,n), Spontaneous Fission, and Delayed Neutron Sources and Spectra

by

W. B. Wilson, R. T. Perry, W. S. Charlton, T. A. Parish, G. P. Estes, T. H. Brown, E. D. Arthur, M. Bozoian, T. R. England, D. G. Madland, and J. E. Stewart

ABSTRACT

SOURCES 4A is a computer code that determines neutron production rates and spectra from (α,n) reactions, spontaneous fission, and delayed neutron emission due to the decay of radionuclides. The code is capable of calculating (α,n) source rates and spectra in four types of problems: homogeneous media (i.e., a mixture of α-emitting source material and low-Z target material), two-region interface problems (i.e., a slab of α-emitting source material in contact with a slab of low-Z target material), three-region interface problems (i.e., a thin slab of low-Z target material sandwiched between α-emitting source material and low-Z target material), and (α,n) reactions induced by a monoenergetic beam of α-particles incident on a slab of target material. Spontaneous fission spectra are calculated with evaluated half-life, spontaneous fission branching, and Watt spectrum parameters for 43 actinides. The (α,n) spectra are calculated using an assumed isotropic angular distribution in the center-of-mass system with a library of 89 nuclide decay α-particle spectra, 24 sets of measured and/or evaluated (α,n) cross sections and product nuclide level branching fractions, and functional α-particle stopping cross sections for Z<106. The delayed neutron spectra are taken from an evaluated library of 105 precursors. The code outputs the magnitude and spectra of the resultant neutron source. It also provides an analysis of the contributions to that source by each nuclide in the problem.

I. INTRODUCTION

In many systems, it is imperative to have accurate knowledge of all significant sources of neutrons due to the decay of radionuclides. These sources can include neutrons resulting from the spontaneous fission of actinides, the interaction of actinide decay α-particles in (α,n) reactions with low- or medium-Z nuclides, and/or delayed neutrons from

the fission products of actinides. Numerous systems exist in which these neutron sources could be important. These include, but are not limited to, clean and spent nuclear fuel (UO_2, ThO_2, MOX, etc.), enrichment plant operations (UF_6, PuF_4), waste tank studies, waste products in borosilicate glass or glass-ceramic mixtures, and weapons-grade plutonium (WPu) in storage containers. The SOURCES 4A code was designed to calculate neutron sources (magnitude and spectra) resulting from any of the aforementioned interactions and decays.

The spontaneous fission spectra are calculated with evaluated half-life, spontaneous fission branching, and ν data using Watt spectrum parameters for 43 actinides. The (α,n) spectra are calculated with a library of 89 nuclide decay α–particle spectra, 24 sets of evaluated (α,n) cross sections and product nuclide level branching fractions, and 105 functional α stopping cross sections using an assumed isotropic neutron angular distribution in the center-of-mass system. A maximum α-particle energy of 6.5 MeV is allowed by SOURCES 4A. This restriction is required because of the limitations of the cross section libraries. The delayed neutron sources are calculated from a library of evaluated delayed neutron branching fractions and half-lives for 105 precursors.

The SOURCES 4A code is capable of calculating neutron sources in homogeneous problems (i.e., homogeneous mixtures of α-emitting and low-Z materials), interface problems (i.e., composite material consisting of two separate slab regions), α-beam problems (i.e., a monoenergetic α-beam incident on a low-Z slab), and three-region interface problems (i.e., a thin slab of low-Z target material sandwiched between α-emitting source material and low-Z target material). However, systems that include combinations of these problems must be run separately and then compiled by the user.

SOURCES 4A consists of a FORTRAN 77 (F77) source code, a user-created input file, up to five output files, and four library files. The SOURCES 4A code has been under development for several years with continuing improvements made in methods and data. The original version of SOURCES (SOURCES 1x) was actually named POFEAL and was primarily used for calculating P_i OF E-ALpha [i.e., the probability of an (α,n) interaction with nuclide i by an α-particle prior to stopping in the material].[1] SOURCES 2x was an improvement of the original POFEAL code which included spectra calculations.[2] Also,

improvements in the calculational algorithm were implemented in this version. The previous addition to the code (SOURCES-3A) was the ability to handle two-region interface problems. Recently the capability to calculate (α,n) source rates and spectra for three-region interface problems was added (SOURCES 4A). SOURCES will continue to be updated and improved as more experimental data and computational methods become available. SOURCES 4A exists for both Unix and PC platforms. The information in this manual is applicable to both the Unix and PC versions; however, installation may vary.

Several works related to SOURCES that are not expressly referenced in the text of this manual are included in Appendix C.

II. THEORY

The SOURCES 4A code is capable of calculating neutron production rates for four different problem configurations (interface, homogeneous, beam, and three-region interface problems) with three different neutron sources: (α,n), spontaneous fission, and delayed neutrons. In the following section, the theory leading to each of these sources and problems is derived. Also, the methodology used in generating the neutron production functions is described in detail.

A. Homogeneous Mixture Problems

A homogeneous mixture problem is one in which the α-emitting material and spontaneous fission sources are intimately mixed with the low-Z target material (i.e., atoms of α-emitting material are directly adjacent to the target atoms). Three sources of neutrons exist in these problems, namely spontaneous fission neutrons, delayed neutrons, and neutrons emitted as a result of (α,n) reactions during the slowing down of α-particles. The theory pertaining to calculations for each of these neutron sources is described below. For homogeneous mixture problems, the neutron source (spectra and magnitude) is output as neutrons produced per second per unit volume. It is assumed in all homogeneous mixture calculations that the target is thick (i.e., that the dimensions of the target are much smaller than the range of the α-particles); and thus, all α-particles are stopped within the mixture.

1. (α,n) Sources

The calculation of the (α,n) neutron production in a material requires accurate knowledge of the slowing down of the α-particles, as well as the probability of neutron production from an α-particle at energy E_α. The slowing and stopping of α-particles in a material are described by the material's stopping power,

$$SP(E) = -\frac{dE}{dx} \tag{1}$$

which yields the energy loss of α-particles of energy E per unit path length x.[3] The energy loss of an α-particle of initial energy E_α in traveling a distance L can be determined from the stopping power as

$$\Delta E = E_\alpha - E'_\alpha = \int_0^L \left(-\frac{dE}{dx}\right) dx . \tag{2}$$

Similarly, the distance traveled in slowing from E_α to E'_α is

$$L = \int_{E_\alpha}^{E'_\alpha} \frac{1}{\left(\frac{dE}{dx}\right)} dE = \int_{E'_\alpha}^{E_\alpha} \frac{1}{\left(-\frac{dE}{dx}\right)} dE . \tag{3}$$

During the slowing down of the α-particles within the material, neutrons may be produced by (α,n) reactions with the nuclides contained in the material. The probability of an (α,n) interaction with nuclide i by an α-particle of energy E traveling from x to x+dx is

$$N_i \sigma_i(E) dx = \frac{N_i \sigma_i(E) dE}{\left(\frac{dE}{dx}\right)} \tag{4}$$

where N_i is the atom density of nuclide i and σ_i is the microscopic (α,n) cross section for nuclide i. The probability of (α,n) interaction with nuclide i by an α-particle that slowed from E_α to E'_α is then

$$p_i(E_\alpha \rightarrow E'_\alpha) = \int_{E_\alpha}^{E'_\alpha} \frac{N_i \sigma_i(E)}{\left(\frac{dE}{dx}\right)} dE = \int_{E'_\alpha}^{E_\alpha} \frac{N_i \sigma_i(E)}{\left(-\frac{dE}{dx}\right)} dE . \tag{5}$$

Thus, the probability of an α-particle undergoing an (α,n) reaction with nuclide i before stopping in the material is given by the thick-target neutron production function,

$$P_i(E_\alpha) = \int_0^{E_\alpha} \frac{N_i \sigma_i(E)}{\left(-\frac{dE}{dx}\right)} dE . \tag{6}$$

The stopping cross section (ε) is defined as,

$$\varepsilon(E) = -\frac{1}{N}\frac{dE}{dx} \tag{7}$$

where N is the total atom density of the material. The quantities p_i and P_i can now be expressed in terms of the stopping cross section

$$p_i(E_\alpha \rightarrow E'_\alpha) = \frac{N_i}{N} \int_{E'_\alpha}^{E_\alpha} \frac{\sigma_i(E)}{\varepsilon(E)} dE \tag{8}$$

and

$$P_i(E_\alpha) = \frac{N_i}{N} \int_0^{E_\alpha} \frac{\sigma_i(E)}{\varepsilon(E)} dE . \tag{9}$$

In general, any material involved in a homogeneous problem will be composed of any number of different elements (e.g., H, C, and O). The stopping cross section ε(E) of a material composed of J elemental constituents may be calculated using the Bragg-Kleeman[4] relationship

$$\varepsilon(E) \cong \frac{1}{N} \sum_{j=1}^{J} N_j \varepsilon_j(E) \tag{10}$$

where

$$N = \sum_{j=1}^{J} N_j . \tag{11}$$

A fraction of the decays of nuclide k within a material may be via α-particle emission. This fraction (F^{α}_k) of alpha decays may occur with the emission of one of L possible α-particle energies. The intensity f^{α}_{kl} is the fraction of all decays of nuclide k resulting in an α-particle of energy E_l; and thus,

$$F_k^{\alpha} = \sum_{l=1}^{L} f_{kl}^{\alpha} . \tag{12}$$

Therefore, the fraction of nuclide k decays resulting in an (α,n) reaction in a thick-target material containing I nuclides with non-negligible (α,n) cross sections is

$$R_k(\alpha,n) = \sum_{l=1}^{L} f_{kl}^{\alpha} \sum_{i=1}^{I} P_i(E_l) . \tag{13}$$

The value for $P_i(E_l)$ will be determined using the discrete form of Eq. (9),

$$P_i(E_l) = \frac{N_i}{N} \sum_{g=1}^{G-1} \frac{1}{2} \left[\frac{\sigma_i^{g+1}}{\varepsilon^{g+1}} + \frac{\sigma_i^{g}}{\varepsilon^{g}} \right] \left(E^{g+1} - E^{g} \right) \tag{14}$$

where $\sigma^1_i=\sigma_i(0)$, $\sigma^G_i=\sigma_i(E_l)$, $\varepsilon^1=\varepsilon(0)$, and $\varepsilon^G=\varepsilon(E_l)$ (i.e., the energy range has been discretized into G-1 energy groups). It is important to note that to calculate the (α,n)

neutron source per decay of nuclide k, it is necessary to have accurate knowledge of the discrete-energy (α,n) cross section for each target nuclide (σ^g_i), discrete energy stopping cross section (ε^g) for all elemental constituents, atom fraction (N_i/N) for each target nuclide, the intensity for emission of each of L α-particles (f^{α}_{kl}), and the energy of each of the L α-particles (E_l). The atom fractions are provided by the user in a file named *tape1*. The other quantities are available to SOURCES from a number of library files (see Section III).

The (α,n) spectra are determined assuming an isotropic neutron angular distribution in the center-of-mass (COM) system[5] with a library of 89 nuclide decay α spectra and 24 sets of product-nuclide level branching fractions. Figure 1 shows an illustration of a general (α,n) reaction in the laboratory system where any associated gamma ray is assumed to be emitted after the neutron is emitted. This assumption is identical to neglecting the momentum of any associated gamma ray, but it accounts for its energy.

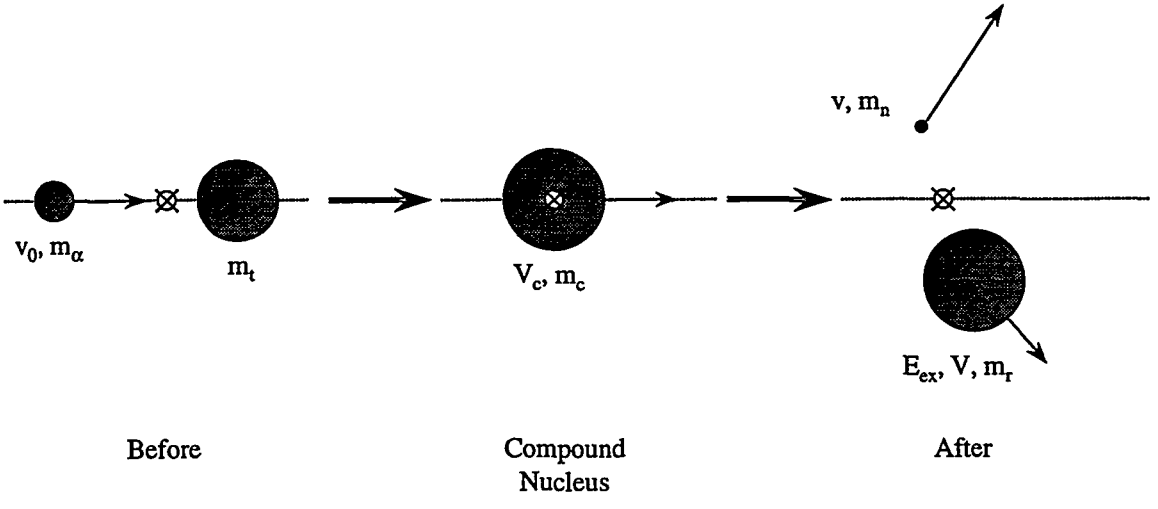


Fig. 1. (α,n) Reaction in the Laboratory System.

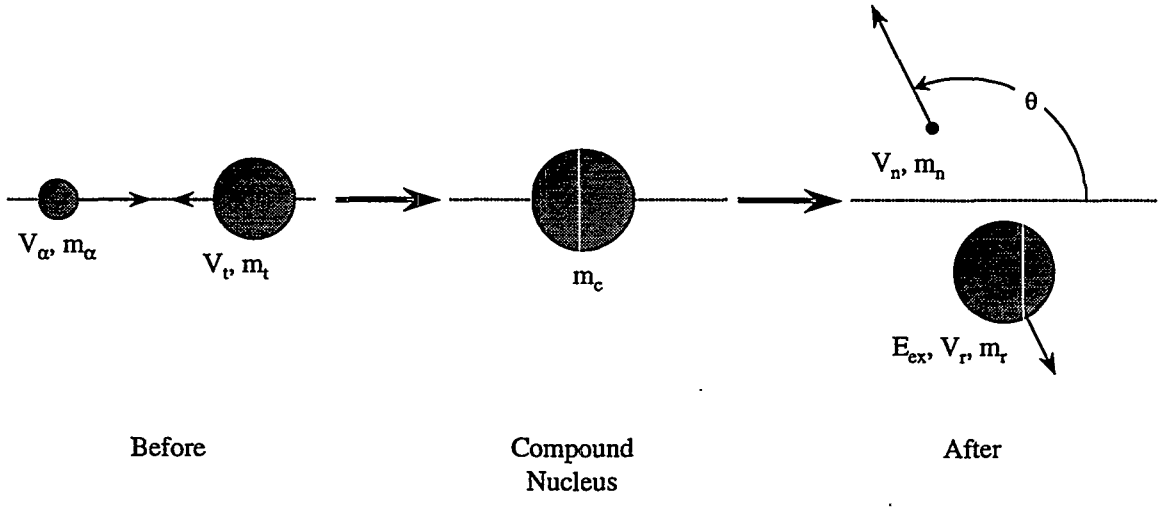


Fig. 2. (α,n) Reaction in the Center-of-Mass System.

It is readily apparent from conservation of momentum that the velocity of the center-of-mass (V_c) is simply given by

$$V_c = \left(\frac{m_\alpha}{m_\alpha + m_t}\right) v_0. \tag{15}$$

This is also equivalent to the velocity of the compound nucleus, assuming the compound nucleus is not in an excited state. Subtracting the velocity of the COM from the particle velocities displayed in Fig. 1 allows for a transformation to the COM coordinate system (Fig. 2). The α-particle velocity in the COM system is given by

$$V_\alpha = v_0\left(\frac{m_t}{m_\alpha + m_t}\right). \tag{16}$$

The target nuclide velocity in the COM system is

$$V_t = -v_0\left(\frac{m_\alpha}{m_\alpha + m_t}\right). \tag{17}$$

From conservation of energy in the COM system, we find

$$KE_n = (Q - E_{ex}) + KE_\alpha + KE_t - KE_r \tag{18}$$

where KE_n is the neutron kinetic energy, KE_α is the α-particle kinetic energy, KE_t is the target nucleus kinetic energy, KE_r is the recoil nucleus kinetic energy, E_{ex} is the excitation level of the recoil nucleus, and Q is the reaction Q-value (all variables in the COM

system). It is customary to define the reaction Q-value for production of product-nuclide level m as

$$Q_m = Q - E_{ex}. \tag{19}$$

From conservation of momentum in the vertical direction we see

$$V_r = \left(\frac{m_n}{m_r}\right) V_n. \tag{20}$$

Using Eq.'s (16) and (17), it can be shown that

$$KE_\alpha + KE_t = E_\alpha \left(\frac{m_t}{m_\alpha + m_t}\right) \tag{21}$$

where E_α is the α-particle kinetic energy in the laboratory system. The recoil nuclei kinetic energy is given by

$$KE_r = \frac{1}{2} m_r V_r^2 = \frac{1}{2} \frac{m_n^2}{m_r} V_n^2 = KE_n \left(\frac{m_n}{m_r}\right) \tag{22}$$

where we have made use of Eq. (20) and the definition of the neutron kinetic energy. Substituting Eq.'s (19), (21), and (22) into Eq. (18) yields

$$KE_n = Q_m + E_\alpha \left(\frac{m_t}{m_\alpha + m_t}\right) - KE_n \left(\frac{m_n}{m_r}\right). \tag{23}$$

Solving Eq. (23) for the neutron kinetic energy yields

$$KE_n = Q_m \left(\frac{m_r}{m_r + m_n}\right) + E_\alpha \left(\frac{m_t}{m_\alpha + m_t}\right)\left(\frac{m_r}{m_r + m_n}\right). \tag{24}$$

Using the definition of kinetic energy, the neutron velocity in COM system can be acquired as

$$V_n = \pm \sqrt{\frac{Q_m}{m_n} \frac{2m_r}{m_r + m_n} + \frac{2E_\alpha}{m_n} \frac{m_t}{m_t + m_\alpha} \frac{m_r}{m_r + m_n}}. \tag{25}$$

This can be converted to the neutron velocity in the laboratory system by adding the velocity of the COM

$$v = \sqrt{\frac{2E_\alpha}{m_\alpha}} \left(\frac{m_\alpha}{m_\alpha + m_t}\right) \pm \sqrt{\frac{Q_m}{m_n} \frac{2m_r}{m_r + m_n} + \frac{2E_\alpha}{m_n} \frac{m_t}{m_t + m_\alpha} \frac{m_r}{m_r + m_n}}. \tag{26}$$

Eq. (26) can be expressed easily in terms of the square root of the neutron kinetic energy as

$$\sqrt{E_{n,m}} = \pm\sqrt{\frac{1}{2}m_n}v \tag{27}$$

where $E_{n,m}$ is the neutron kinetic energy in the laboratory frame of reference from an incident α-particle of energy E_α and generating a product nuclei of level m. Thus the neutron kinetic energy is

$$E^{\pm}_{n,m} = \left(\sqrt{E_\alpha a_1}\left(\frac{1}{1+a_2}\right) \pm \sqrt{Q_m \frac{1}{1_r + a_3} + E_\alpha \frac{a_2}{1+a_2}\frac{1}{1+a_3}}\right)^2 \tag{28}$$

where we have defined

$$a_1 = \frac{m_n}{m_\alpha} \tag{29}$$

$$a_2 = \frac{m_t}{m_\alpha} \tag{30}$$

and

$$a_3 = \frac{m_n}{m_r}. \tag{31}$$

Equation (28) relates the maximum (+ second term) and minimum (– second term) permissible neutron kinetic energies from an incident α-particle of energy E_α generating a product nuclide with level m.

For each target nuclide and each source α-particle, the code can read the number of product-nuclide levels (M_i), the number of product level branching data points (M'_i), the (α,n) reaction Q-value (Q^i), the excitation energy of each product-nuclide level [$E^i_{ex}(m)$], and the fraction of (α,n) reactions at energy E(m') resulting in the production of product level m [$f_i(m,m')$] from the library files. The neutron energy spectra will be discretized into a user-defined energy group structure. The fraction of target i product level m reactions of source k α-particles occurring in α-particle energy group g is

$$H^l_{i,k}(m) = \frac{P_i(E_{l+1}) - P_i(E_l)}{P_i(E_\alpha)}, \tag{32}$$

where $P_i(E_l)$ was defined in Eq. (14). The branching fraction of α-particles at E_α reacting with target nuclide i and producing product level m is

$$S_{i,k}(m) = f_i(m,m'-1) + \left(f_i(m,m') - f_i(m,m'-1)\right)\frac{E_\alpha - E(m'-1)}{E(m') - E(m'-1)}. \quad (33)$$

Thus, the fraction of α-particles at E_α reacting with target nuclide i and resulting in product level m reactions occurring in α-particle energy group g is simply the product of Eq.'s (32) and (33):

$$F^l_{i,k}(m) = S_{i,k}(m) H^l_{i,k}(m). \quad (34)$$

It will be assumed that the neutrons are isotropically emitted from the compound nucleus; therefore, they will contribute evenly to all groups between $E^+_{n,m}$ and $E^-_{n,m}$. The contribution per decay of source nuclide k to neutron energy group g is given by

$$\chi^{(\alpha,n)}_k(E_g) = R_k(\alpha,n) F^l_{i,k}(m) \frac{E_{g+1} - E_g}{E^+_{n,m} - E^-_{n,m}}, \quad (35)$$

where E_{g+1} and E_g are between $E^+_{n,m}$ and $E^-_{n,m}$.

2. *Spontaneous Fission Sources*

The spontaneous fission of an actinide nuclide k is accompanied by the emission of an average $\nu_k(SF)$ neutrons. The fraction of nuclide k decays that are spontaneous fission events are given by the SF branching fraction

$$F^{SF}_k = \frac{\lambda^{SF}_k}{\lambda_k}. \quad (36)$$

Thus, the average number of SF neutrons emitted per decay of nuclide k (by any mode) is

$$R_k(SF) = F^{SF}_k \nu_k(SF). \quad (37)$$

Therefore, to compute the neutron production due to spontaneous fission per decay of nuclide k, the SF branching fraction and average number of neutrons per spontaneous fission must be known. These quantities are available to SOURCES from a library file named *tape5* (see Section III).

The spontaneous fission neutron spectra are approximated by a Watt's fission spectra using two evaluated parameters (a and b):

$$\chi^{SF}_k(E) = R_k(SF) e^{-E/a} \sinh\sqrt{bE}. \quad (38)$$

Evaluated parameters are provided for 43 fissioning nuclides in the *tape5* library file (see Section III below).

3. Delayed Neutron Sources

During the fissioning process, a number of products are formed including neutrons, gamma rays, beta rays, neutrinos, fission products, and an appreciable amount of energy. Some of the fission products formed as a result of fission can decay by β^- emission to a highly excited state, which can then decay by emitting a neutron. These neutrons are called "delayed neutrons" because they appear within the system with some appreciable time delay. The nuclide emitting the neutron is referred to as the "delayed neutron emitter," and the nuclide which β^- decays to the emitter is referred to as a "delayed neutron precursor." It is customary to assume that one neutron is emitted per decay and that the emitter decays almost instantaneously. Thus, the fraction of decays by nuclide k (by any mode) leading to the emission of a delayed neutron is given by the product of the DN branching fraction (F^{DN}_k):

$$R_k(DN) = F_k^{DN}. \tag{39}$$

Computing the neutron production rate due to delayed neutron emission requires knowledge of the DN branching fraction. The value for F^{DN}_k is provided to SOURCES in a library file (see Section III).

A series of evaluated delayed neutron spectra are provided in a library file for 105 precursor nuclides [φ_k(E)]. These evaluated spectra are provided in a discretized form. They are read directly into SOURCES and then adjusted so that the default spectra energy mesh correlates with the user-desired energy mesh. The energy spectra is then renormalized by multiplying through by the quantity R_k(DN), such that

$$\chi_k^{DN}(E) = R_k(DN)\varphi_k(E). \tag{40}$$

4. Total Neutron Source

The average total number of neutrons per decay emitted due to (α,n) reactions, spontaneous fission, and delayed neutron emission is given by

$$R_k = R_k(\alpha,n) + R_k(SF) + R_k(DN). \tag{41}$$

Therefore, the total neutron source from (α,n) reactions, spontaneous fission, and delayed neutron emission within a homogenous problem consisting of K pertinent radionuclides is

$$S = \sum_{k=1}^{K} \lambda_k N_k R_k \qquad (42)$$

where λ_k is the decay constant for nuclide k and N_k is the atom density of nuclide k. A similar expression will be used for the source produced in interface and beam problems.

The energy-dependent neutron source spectra are calculated using the absolute (α,n), SF, and DN spectra calculated above for each nuclide k by an expression similar to that given in Eq. (42):

$$S_g = \lambda_k N_k \chi_k(E_g) \qquad (43)$$

where

$$\chi_k(E_g) = \chi_k^{(\alpha,n)}(E_g) + \chi_k^{SF}(E_g) + \chi_k^{DN}(E_g). \qquad (44)$$

B. Beam Problems

A beam problem is one in which a monoenergetic α-beam is incident upon a slab containing low-Z target material (see Fig. 3). The slab could also contain higher mass isotopes; however, actinides (i.e., α-emitting or spontaneous fissioning material) will not be used to calculate a source. It is a necessary condition that the thickness of the slab of target material (t) be significantly larger than the range of the α-particles in the beam (i.e., that all α-particles come to rest within the target slab).

The neutron production rate within the slab per incident α-particle is a function of the α-particle beam energy (E_α) and the probability of an (α,n) interaction with any nuclide i within the slab by an α-particle from the beam prior to stopping in the material,

$$S = \sum_{i=1}^{I} P_i(E_\alpha). \qquad (45)$$

The beam energy (E_α) must be supplied by the user. The thick-target neutron production function [$P_i(E_\alpha)$] is calculated using Eq. (14) above. The neutron spectra are calculated using the same procedure as described in Section II.A.1.

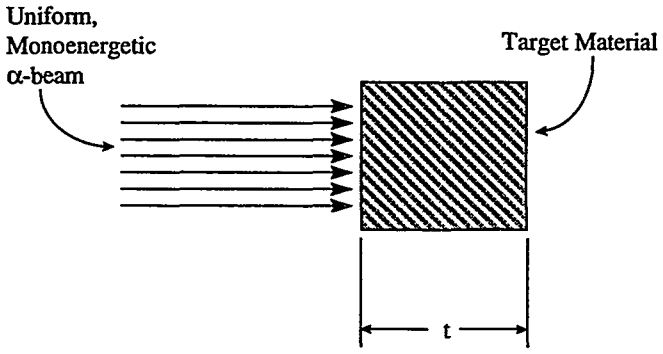


Fig. 3. General Schematic for Beam Problems.

C. Interface Problems

Interface problems (Fig. 4) exist when a slab of α-emitting material (such as Pu, Po, or Am) is in close contact with a low-Z target material (such as Be, C, or Al). In these problems, α-particles are emitted from the Region I materials and travel across the interface junction into the Region II materials. In Region II, the α-particles can interact through (α,n) reactions and generate a neutron source. It is necessary to assume that in all interface problems the thickness of each region is significantly larger than the range of the α-particles within it. Also, it will be assumed that all α-particles travel in a straight-line trajectory from their point of emission (generally an excellent assumption).

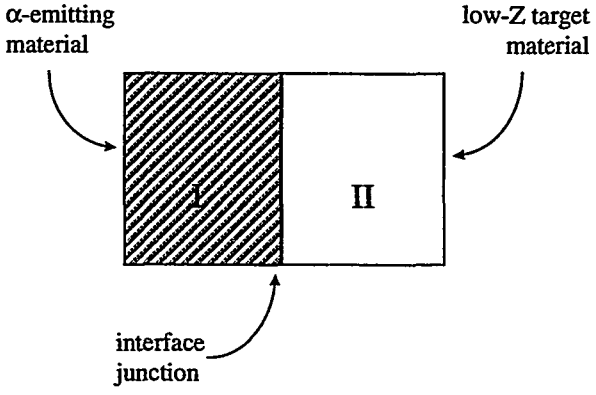


Fig. 4. General Schematic for Interface Problems.

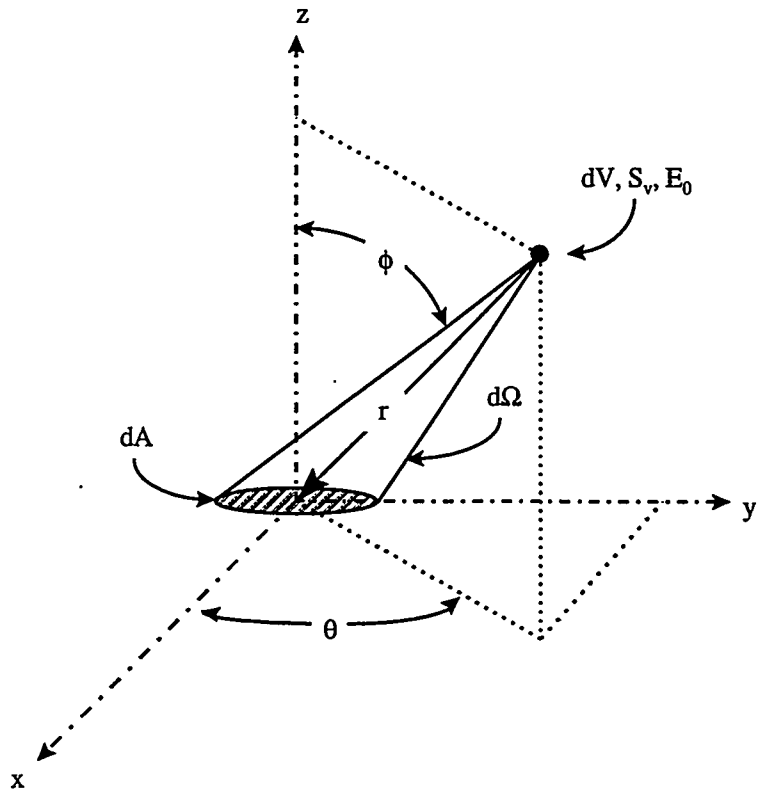


Fig. 5. The α-Particle Solid Angle to Differential Area Due to Generalized α Source.

To derive the α-particle source rate at the interface, consider the half-space above the x-y plane shown in Fig. 5. .There exists a uniform volumetric source (S_v) of α-particles in this half-space. The α-particles are assumed to be emitted isotropically with initial energy E_0. The differential area dA subtends a solid angle dΩ when viewed from the source point dV. Thus,

$$d\Omega = \frac{\cos\phi}{r^2} dA \tag{46}$$

and

$$dV = r^2 \sin\phi \cdot d\theta \cdot d\phi \cdot dr\,. \tag{47}$$

The rate at which α-particles are born in dV is equal to

$$S_v dV = S_v r^2 \sin\phi \cdot d\theta \cdot d\phi \cdot dr\,. \tag{48}$$

The solid angle subtended by dA relative to the total solid angle into which α-particles are emitted is given by

$$\frac{d\Omega}{4\pi} = \frac{\cos\phi \cdot dA}{4\pi \cdot r^2}. \tag{49}$$

Multiplying Eq. (48) by Eq. (49) yields the number of α-particles per unit time originating within dV that can pass through dA provided that r is less than the α-particle range, or

$$dU = \frac{S_v}{4\pi}\sin\phi\cos\phi \cdot dA \cdot d\theta \cdot d\phi \cdot dr. \tag{50}$$

The rate at which α-particles pass through dA as a result of having been born in a hemispherical shell centered about dA whose radius is r and thickness is dr is acquired by integrating Eq. (50) over θ and ϕ, or

$$dU' = \frac{S_v}{4\pi} dA \cdot dr \int_0^{2\pi} d\theta \int_0^{\pi/2} \sin\phi\cos\phi \cdot d\phi. \tag{51}$$

Performing the integration yields

$$dU' = \frac{S_v}{4} dA \cdot dr. \tag{52}$$

From Eq. (7) we see

$$dr = -\frac{1}{N}\frac{dE}{\varepsilon(E)} \tag{53}$$

where ε(E) is the stopping cross section and N is the total atom density of the material in the region. Thus, the rate at which α-particles pass through the interface per unit area is given by

$$\frac{dU'}{dA} = -\frac{S_v}{4N}\frac{1}{\varepsilon(E)} dE. \tag{54}$$

Therefore, the rate at which α-particles with energies between E_g and E_{g+1} pass through the interface per unit area (Φ) is

$$\Phi^g = \frac{U}{A} = \frac{S_v}{4N}\int_{E_g}^{E_{g+1}} \frac{dE}{\varepsilon(E)}. \tag{55}$$

The volumetric source (S_v) can be expressed as

$$S_v = \lambda_k N_k f_{kl}^{\alpha} \tag{56}$$

where λ_k is the decay constant for source nuclide k, N_k is the atom density of source nuclide k, and f^{α}_{kl} is the fraction of all decays of nuclide k resulting in an α-particle of energy E_{kl}. Thus we see that

$$\Phi^g = \frac{\lambda_k f_{kl}^{\alpha}}{4} \frac{N_k}{N} \int_{E_g}^{E_{g+1}} \frac{dE}{\varepsilon(E)}. \tag{57}$$

This quantity (Φ^g) is the source of α-particles between energies E_g and E_{g+1} passing into the low-Z target material (Region II) per unit area and per unit time. The quantity Φ^g is then used by SOURCES as the source strength of a monoenergetic beam with energy:

$$E^g_{beam} = \frac{E_g + E_{g+1}}{2}. \tag{58}$$

SOURCES can then use the same procedure developed in Section II.B to solve for the neutron production rate due to the α-particles crossing the junction with energies between E_g and E_{g+1}. SOURCES then repeats this procedure for all α-particle energies and all source nuclides.

D. Three-Region Interface Problems

A three-region (α,n) problem consists of an α-emitting slab (such as Pu, Po, or Am) in direct contact with a thin slab of low-Z target material (such as Be, C, or Al) which is itself in contact with a thick (α,n) target (Fig. 6). In this particular problem, α-particles born in region A, can slow through region A to interface ab, slow through region B to interface bc, and slow to a stop in region C. Thus, neutrons can be produced in both region B and region C due to the slowing α-particles.

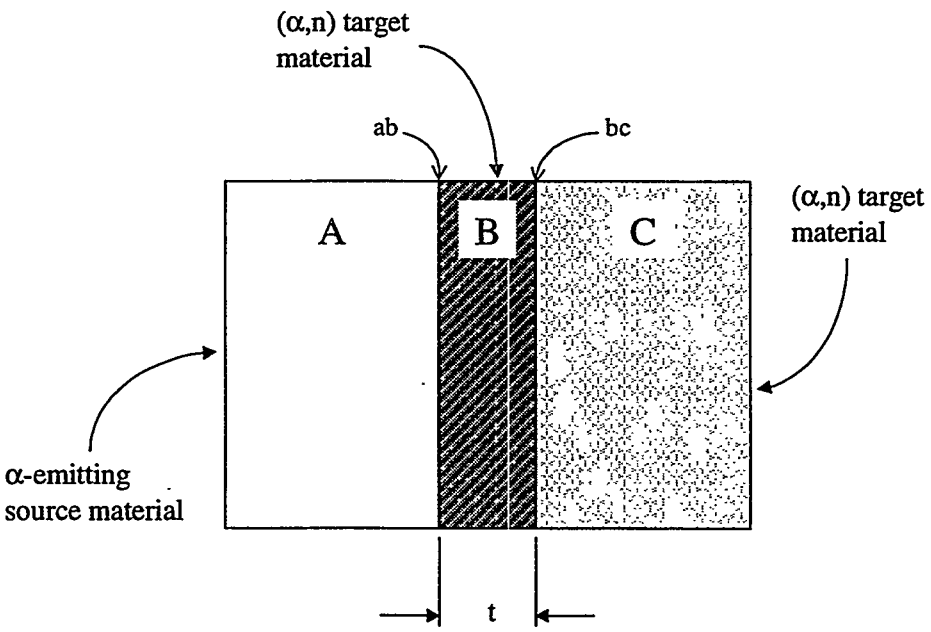


Fig. 6. General Schematic for Three-Region Interface Problem.

This problem is significantly more complicated than the two-region variety due to the dependence on the thickness of the track through region B and thus an inherent angular dependence. It is necessary to assume that in all interface problems the thickness of regions A and C is significantly larger than the range of the α-particles within it. Region B can have any thickness. It will also be assumed that all α-particles travel in a straight-line trajectory from their point of emission (generally an excellent assumption).

The number of α-particles crossing interface ab with energies between E_g and E_{g+1} is given by:

$$\Phi_{ab}^{g} = \sum_{k=1}^{NN} \sum_{l=1}^{NL_k} \frac{\lambda_k f_{kl}^{\alpha}}{4} \left(\frac{N_k}{N} \right)_A m_{kl}^{g} \left(\frac{1}{\varepsilon_A^{g}} + \frac{1}{\varepsilon_A^{g+1}} \right) \left(\frac{E_{g+1} - E_g}{2} \right) \qquad (59)$$

where λ_k is the decay constant of α-emitting source nuclide k, f^{α}_{kl} is the fraction of all nuclide k decays that result in the production of an α-particle at energy E^{α}_l, $(N_k/N)_A$ is the atom fraction of α-emitting source nuclide k in region A, ε^{g}_A is the stopping cross section of region A at energy E_g, and m^{g}_{kl} is a calculational factor given by:

$$m_{kl}^{g} = \begin{cases} 0 & if \quad E_g > E_{kl}^{\alpha} \\ 1 & if \quad E_{g+1} < E_{kl}^{\alpha} \\ \dfrac{E_{kl}^{\alpha} - E_g}{E_{g+1} - E_g} & if \quad E_g < E_{kl}^{\alpha} < E_{g+1} \end{cases} . \tag{60}$$

The only assumption made in equation (59) is that the function $1/\varepsilon^{g}_{A}$ is linear between E_g and E_{g+1}.

To calculate the α-particle source rate at the interface bc, we first must determine the energies at which the α-particles transition from region A to region B at an angle between ϕ_i and ϕ_{i+1} and end up at interface bc with energies between E_g and E_{g+1}. We will calculate these energies by performing the following:

$$\min_{itrans_{i,g}} \left[\sum \left(\frac{1}{\varepsilon_B^{g'}} + \frac{1}{\varepsilon_B^{g'+1}} \right) \left(\frac{E_{g'+1} - E_{g'}}{2} \right) > \frac{t}{\cos(\phi_i)} \right] \quad for \ all \ i \ and \ g \tag{61}$$

where $itrans_{i,g}$ is the transition energy index and t is the thickness of the intermediate region B. The number of α-particles crossing interface bc with energies between E_g and E_{g+1} is given by:

$$\Phi_{bc}^{g} = \sum_{k=1}^{NN} \sum_{l=1}^{NL_k} \sum_{i=1}^{I} P_{g,i,k,l} \frac{\lambda_k f_{kl}^{\alpha}}{8} \left(\frac{N_k}{N} \right)_A \left(\cos(2\phi_i) - \cos(2\phi_{i+1}) \right) m_{kl}^{g} \left(\frac{1}{\varepsilon_A^{g}} + \frac{1}{\varepsilon_A^{g+1}} \right) \left(\frac{E_{g+1} - E_g}{2} \right) \tag{62}$$

where λ_k is the decay constant of α-emitting source nuclide k, f^{α}_{kl} is the fraction of all nuclide k decays that result in the production of an α-particle at energy E^{α}_{l}, $(N_k/N)_A$ is the atom fraction of α-emitting source nuclide k in region A, ε^{g}_{A} is the stopping cross section of region A at energy E_g, m^{g}_{kl} is a calculational factor given in equation (2) above, and $P_{g,i,k,l}$ is a calculational factor given by:

$$P_{g,i,k,l} = \begin{cases} 0 & if \quad E_{itrans_{i,g}} > E_{kl}^{\alpha} \\ 1 & if \quad E_{itrans_{i,g}} < E_{kl}^{\alpha} \end{cases} . \tag{63}$$

These α-particle source rates can now be used to calculate the neutron production rates in a manner not dissimilar to what was used in section II.C. above. To accomplish this we will calculate the neutron production rates and spectra from the α-particle source at interface ab due to material B assuming region B is infinitely thick ($\Psi^{g}_{ab,B}$). We will then

calculate the neutron production rates and spectra from the α-particle source at interface bc due to material B assuming region C is infinitely thick ($\Psi^g_{bc,B}$). Then, we will calculate the neutron production rates and spectra from the α-particle source at interface bc due to material C assuming region C is infinitely thick ($\Psi^g_{bc,C}$). The total neutron production rates and spectra due to the interface is then given by:

$$\Psi^g = \Psi^g_{bc,C} + \left(\Psi^g_{ab,B} - \Psi^g_{bc,B}\right). \tag{64}$$

These multigroup neutron source rates are then output to a file for the user.

III. FILE STRUCTURE

The SOURCES 4A code system is composed of an F77 source code, an executable, an input file, several output files, and a series of library files. All of these files (except for the output files which SOURCES will generate) are necessary for proper execution of the SOURCES code. The name and a short description of each file included in the SOURCES 4A code system are included below:

tape1 = user input file
tape2 = stopping cross section expansion coefficients library
tape3 = target (α,n) cross section library
tape4 = target (α,n) product level branching library
tape5 = sources decay data library
tape6 = neutron source magnitudes output file
tape7 = absolute neutron spectra output file
tape8 = normalized neutron spectra output file
tape9 = neutrons spectra output file by product level
outp = summary output file
SOURCE4A.for = F77 source code
SOURCE4A.exe = executable code.

Fig. 7 illustrates the sources code structure and how each file interacts with the executable file.

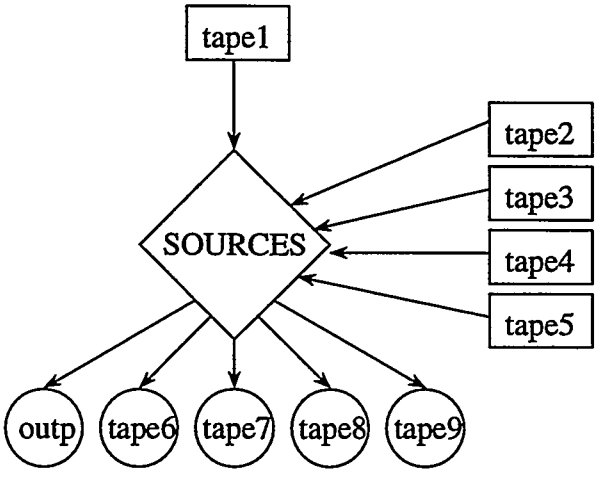


Fig. 7. Schematic Diagram of the SOURCES 4A File Structure.

The data necessary for computing the magnitude of the neutron source due to (α,n) reactions, spontaneous fission, and delayed neutron emission are:

1. The energy-dependent α-particle stopping cross section for all elemental constituents (ε^g_i).
2. The energy dependent (α,n) cross section for all target nuclides (σ^g_i).
3. The intensity for emission of each of the L α-particles (f^{α}_{kl}).
4. The energy of each of the L α-particles (E_l).
5. The SF branching fractions for each source nuclide k (F^{SF}_k).
6. The average number of neutrons per SF of nuclide k [ν_k(SF)].
7. The DN branching fraction for each nuclide k (F^{DN}_k).
8. The source nuclide decay constants (λ_k).

To calculate the neutron source spectrum, it is necessary to have data for:

1. The number of product nuclide levels (M) for all target nuclides.
2. The number of product nuclide level branching data points (M') for all target nuclides.
3. The (α,n) reaction Q-value for all target nuclides.
4. The excitation energy [E_{ex}(m)] of product nuclide level m for all target nuclides.
5. The fraction of (α,n) reactions with target i at energy E(m) resulting in the production of product level m.
6. The α-particle, neutron, target, and product nuclei masses.
7. Watt's fission spectrum parameters (a and b) for each source nuclide k.
8. Delayed neutron energy spectrum for each source nuclide k.

All of these parameters are included in the library files. The library files contain (α,n) target nuclide cross section parameters for all the nuclei listed in Table I and source parameters for all the nuclei listed in Table II. These nuclides are listed in ZAID format and are defined as ZAID = state + (10·A) + (10000·Z), where Z is the atomic number, A is the atomic mass, and the state is either 0 or 1 for ground or metastable, respectively.

TABLE I
(α,n) Target Isotopes Available in SOURCES 4A.

Isotope	ZAID	Level Branching Fraction Data Source	Cross Section Data Source
Li-7	030070	GNASH	Gibbons and Macklin[6]
Be-9	040090	Geiger and Van der Zwan[7]	Geiger and Van der Zwan[7]
B-10	050100	GNASH	Bair *et al.*[8]
B-11	050110	GNASH	Bair *et al.*[8]
C-13	060130	GNASH[a]	Bair and Haas[9]
N-14	070140	N/A[b]	GNASH
O-17	080170	Lesser and Schenter[10]	Perry and Wilson[1]
O-18	080180	Lesser and Schenter[10]	Perry and Wilson[1]
F-19	090190	Lesser and Schenter[10]	Balakrishnan *et al.*[11]
Ne-21	100210	N/A[b]	GNASH
Ne-22	100220	N/A[b]	GNASH
Na-23	110230	GNASH	GNASH[a]
Mg-25	120250	GNASH	GNASH
Mg-26	120260	GNASH	GNASH
Al-27	130270	GNASH	GNASH[a]
Si-29	140290	GNASH	GNASH[a]
Si-30	140300	GNASH	GNASH[a]
P-31	150310	GNASH	GNASH
Cl-37	170370	GNASH	Woosley *et al.*[12]

[a] GNASH calculated and measured data (in that order) are available for these nuclides in the library file. By default, the GNASH calculation is used (actually SOURCES uses the first data set that it encounters during the reading of the file). To use an alternate data set, the library file must be altered by reversing the order in which these data sets occur in the file.

[b] Nuclide level branching data for these isotopes are absent from the library files. Thus, problems containing these isotopes can be executed only for neutron source magnitudes (*id*=1) and not for neutron source spectra (*id*=2).

TABLE II

Actinide Isotopes Available as Decay Sources in SOURCES 4A.

Isotope	ZAID
Ce-142	581420
Nd-144	601440
Sm-146	621460
Sm-147	621470
Sm-148	621480
Sm-149	621490
Gd-152	641520
Pb-210	822100
Bi-210	832100
Bi-211	832110
Bi-212	832120
Bi-213	832130
Bi-214	832140
Po-210	842100
Po-211	842110
Po-212	842120
Po-213	842130
Po-214	842140
Po-215	842150
Po-216	842160
Po-218	842180
At-215	852150
At-217	852170
At-218	852180
At-219	852190
Rn-217	862170
Rn-218	862180
Rn-219	862190
Rn-220	862200
Rn-222	862220
Fr-221	872210
Fr-222	872220
Fr-223	872230
Ra-222	882220
Ra-223	882230

Isotope	ZAID
Ra-224	882240
Ra-226	882260
Ac-225	892250
Ac-226	892260
Ac-227	892270
Th-226	902260
Th-227	902270
Th-228	902280
Th-229	902290
Th-230	902300
Th-232	902320
Pa-230	912300
Pa-231	912310
U-230	922300
U-232	922320
U-233	922330
U-234	922340
U-235	922350
U-236	922360
U-237	922370
U-238	922380
U-239	922390
Np-236	932360
Np-237	932370
Np-238	932380
Np-239	932390
Pu-235	942350
Pu-236	942360
Pu-237	942370
Pu-238	942380
Pu-239	942390
Pu-240	942400
Pu-241	942410
Pu-242	942420
Pu-243	942430

Isotope	ZAID
Pu-244	942440
Am-240	952400
Am-241	952410
Am-242	952420
Am-242m	952421
Am-243	952430
Am-244	952440
Am-244m	952441
Cm-240	962400
Cm-241	962410
Cm-242	962420
Cm-243	962430
Cm-244	962440
Cm-245	962450
Cm-246	962460
Cm-247	962470
Cm-248	962480
Cm-250	962500
Bk-249	972490
Cf-248	982480
Cf-249	982490
Cf-250	982500
Cf-251	982510
Cf-252	982520
Cf-253	982530
Cf-254	982540
Es-253	992530
Es-254	992540
Es-254m	992541
Es-255	992550
Fm-254	1002540
Fm-255	1002550
Fm-256	1002560
Fm-257	1002570

Stopping-power coefficients (which are a function of atomic number only) are included for all elemental constituents with $Z \leq 105$. The data by Ziegler *et al.*[13] was used for all $Z \leq 92$. The stopping power coefficients calculated by Perry and Wilson[2] were used for $92 < Z \leq 105$.

IV. INPUT AND EXECUTION

The SOURCES input is designed to be relatively simple; however, its length can vary over a wide range from exceptionally short ($\leq$ 10 lines) to very long ($\geq$ 50 lines). This large range depends on the number of nuclides (source and target) contained in the problem. Appropriate knowledge of the physics (both macroscopic and microscopic) present in a problem is vital for proper execution of SOURCES (see the Po-Be sample problems in Section VI). All SOURCES input is free format with spaces as delimiters (or commas; however, spaces look better). The input deck should be created in a file named *tape1* for use by the SOURCES executable. Every SOURCES problem begins with the same two cards:

card 1: title

card 2: idd id

The first card is a title card with a maximum length of 77 characters. The second card contains two records (*idd* and *id*), which define the type of problem to be considered (homogeneous, interface, or beam) and the type of neutron source output to be produced (magnitudes only or magnitudes and spectra), respectively. The record *idd* can be either a 1 (for a homogeneous problem), a 2 (for an interface problem), or a 3 (for a beam problem). The record *id* can be either a 1 (for magnitudes only) or a 2 (for magnitudes and spectra). The remaining input cards depend upon the type of problem being considered.

A. Homogeneous Problems (*idd*=1)

A homogeneous problem must contain at least 8 cards which describe the elemental constituents in the material, the neutron energy group structure to be used in the output, the source nuclides present, the type of stopping cross sections to be used, and the (α,n) target nuclides present. If multiple materials are present or neutron energy spectra are requested, then more cards can exist in the input deck.

Cards 1-9 for a homogeneous problem are as follows:

card 1: *title*

card 2: *idd id*

card 3: *nz isg*

card 4.1 - 4.nz: *jzm(j) azm(j)*

card 5: *nng enmax enmin* (if necessary)

card 5.1 - 5.nng: *en(n)* (if necessary)

card 6: *nq*

card 7.1 - 7.nq: *jq(k) aq(k)*

card 8: *nt nag*

card 9.1 - 9.nt: *idt(i) at(i)*

Multiple cards are designated by subcards (i.e., if *nz*=3, then the input deck would include card 3 and subcards 4.1, 4.2, and 4.3). Each card must be entered on a new line (the exceptions are subcards 5.1 through 5.*nng* where all records *en(n)* can be entered on the same line or multiple lines). Several example input decks are included below to illustrate the procedure described above. Each record is defined as follows:

nz = the number of stopping cross section elemental constituents present in the material (must be an integer between 0 and 20).

isg = the type of stopping cross sections to be used (0 for solid stopping cross sections, 1 for gas stopping cross sections).

jzm(j) = the atomic number of each stopping cross section elemental constituent from j=1 to *nz*.

azm(j) = the fraction of all atoms that are element j.

nng = the number of neutron spectrum energy groups (integer between 1 and 750 or between -1 and -750); read only if *id*=2, otherwise omitted. If *nng* is positive, then the energy group structure will be determined by a linear interpolation between *enmax* and *enmin* (cards 5.1 through 5.nng are omitted). If *nng* is negative, then the energy group upper bounds must be specified on cards 5.1 through 5.*nng* (however, *enmax* and *enmin* must still be included).

enmax = the maximum neutron energy in MeV (read only if *id*=2, otherwise omitted).

enmin = the minimum neutron energy in MeV (read only if *id*=2, otherwise omitted).

en(n) = the upper energy bound in MeV of neutron groups, listed in descending order (must contain *nng* records in any format); read only if *id*=2 and *nng* is negative (otherwise omitted).

nq = the number of source nuclides to be evaluated (integer between 1 and 300).

jq(k) = the source nuclide k identification in ZAID format (see Section III).

aq(k) = the atom density (atoms/cm^3) of source nuclide k.

nt = the number of target nuclides (integer value between 1 and 20).

nag = the number of α-particle energy groups to be used in calculation (integer value between 1 and 4000).

idt(i) = the target nuclide i identification in ZAID format (see Section III).

at(i) = the fraction of all atoms that are target nuclide i.

The first example demonstrates the neutron sources produced via decay in clean UO_2 fuel with 3% enriched U-235. This problem solves for the neutron source magnitudes from the homogeneous mixture. The first card is simply the title. The second card has records for *idd*=1 (homogeneous problem) and *id*=1 (neutron source magnitudes only). The third card has records for *nz*=2 (two elemental constituents: uranium and oxygen) and *isg*=0 (solid stopping power coefficients). Cards 4.1 and 4.2 include the Z-values for oxygen and uranium [*jzm*(1)=8 and *jzm*(2)=92], as well as their atom fractions [*azm*(1)=$^2/_3$ and *azm*(2)=$^1/_3$]. Two source nuclides are included (*nq*=2) for U-235 and U-238 with atom densities of 6.77 x 10^{20} and 2.16 x 10^{22} atoms/cm^3, respectively. Card 8 includes records for *nt*=2 and *nag*=4000. Card 9.1 and 9.2 include the atom fractions for O-17 and O-18 [the (α,n) targets in natural oxygen].

Example Problem #1 - 3% Enriched Uranium Dioxide Fuel for Neutron Source Magnitude.

```
Example 1 - Clean UO2 Fuel (3% enriched)
1 1
2 0
    8 0.6666667
   92 0.3333333
2
   922350 6.77e+20
   922380 2.16e+22
2 4000
    80170 0.000253
    80180 0.001333
```

A second example problem using the identical material characteristics used in Example Problem #1 is shown in Example Problem #2. This problem illustrates the input necessary to develop neutron spectra outputs. The value of *id* (Card 2) has been changed to 2, and cards 5, 5.1, and 5.2 have been included to define the neutron energy spectra (these cards were omitted in Example Problem #1). The spectra have been established by user input (i.e., *nng* is negative) to span from 0.0 to 10.0 MeV in 1.0 MeV bins. The energy group width can be of any magnitude and can vary from group to group.

Example Problem #2 - 3% Enriched Uranium Dioxide Fuel for Neutron Source Magnitude and Spectra.

```
Example 2 - Clean UO2 Fuel (3% enriched) for Spectra
1 2
2 0
    8 0.6666667
   92 0.3333333
-10 10.0 0.0
   10.0 9.0 8.0 7.0 6.0
    5.0 4.0 3.0 2.0 1.0
2
   922350 6.77e+20
   922380 2.16e+22
2 4000
   80170 0.000253
   80180 0.001333
```

Example Problem #3 - PuF_4 Gas for Neutron Source Magnitude and Spectra.

```
Example 3 - PuF4 Gaseous Problem
1 2
2 1
        9 0.8
       94 0.2
20 15.0 0.0
6
   942380 2.13e+17
   942390 2.54e+21
   942400 1.65e20
   942410 7.39e18
   942420 8.99e17
   952410 4.37e18
1 2000
    90190 0.8
```

Example Problem #3 illustrates the usage of a linearly interpolated energy structure (*nng*>0) and gas stopping power coefficients (*isg*=1). In this problem, the neutron source spectra and magnitudes (*id*=2) are determined for a PuF_4 gas. This problem is reminiscent of possible criticality conditions in enrichment operations. The energy spectra have been established to include 20 groups (*nng*=20) linearly interpolated from 15.0 (*enmax*) to 0.0 (*enmin*) MeV. The problem includes five isotopes of plutonium (Pu-238, Pu-239. Pu-240, Pu-241, and Pu-242) and one isotope of americium (Am-241) as sources (*nq*=6). Also one isotope of fluorine (F-19) is included as an (α,n) target (*nt*=1). Note that due to its low concentration, Am was neglected as an elemental constituents. Thus, only two elemental constituents are present (Pu and F, *nz*=2) to slow the α-particles. Both of these elements will use gas stopping power coefficients in all calculations. The number of α-particle energy groups (*nag*) used was 2000.

The free-form input allows for the atom densities to be entered in any format (i.e., decimal, scientific with the + or – sign on the exponent, or scientific without the + or – on the exponent) and for spaces to be used freely to allow for easier reading by the user. Also, nuclides can be included as sources or targets and not appear as an elemental constituent.

The outputs for some of the above example problems will be presented in Appendix A. Also, several sample problems that show other input decks are described in Section VI.

B. Interface Problems (*idd*=2)

An interface problem input deck is divided into two sections, one for the source side and one for the target side. The deck must contain at least 13 cards that are used to describe the material constituents of both the source and target sides, the source nuclides present, the target nuclides present, the types of stopping cross sections to be used, the α-particle energy group structure to be used at the interface, and the neutron source energy group structure to be used in the output. The cards are described as follows:

card 1:	*title*
card 2:	*idd id*
card 3:	*nzq isgq eamax eamin*
card 4.1 - 4.nz:	*jzq(j) azq(j)*
card 5:	*naq*
card 6:	*nq*
card 7.1 - 7.nq:	*jq(k) aq(k)*
card 8:	*title2*
card 9:	*nzt isgt*
card 10:	*jzt(k) azt(k)*
card 11:	*nng enmax enmin* (if necessary)
card 11.1 - 11.nng:	*en(n)* (if necessary)
card 12:	*nt nag*
card 13.1 - 13.nt:	*idt(i) at(i).*

Multiple cards are designated by subcards (i.e., if *nz*=3, then the input deck would include cards 3, 4.1, 4.2, and 4.3). Each subcard must be entered on a new line (the exception being cards 11.1 through 11.*nng* where all records *en(n)* can be entered on the same line or multiple lines). Each record is defined as follows:

nzq = the number of stopping cross section elemental constituents present in the source material (must be an integer between 0 and 20).

isgq = the type of stopping cross sections to be used for source side (0 for solid stopping cross sections, 1 for gas stopping cross sections).

eamax = the maximum α-particle energy for α-particle source at the interface.

eamin = the minimum α-particle energy for α-particle source at interface.

jzq(k) = the atomic number of each stopping cross section elemental constituent from j=1 to *nzq* for the source side.

azq(k) = the fraction of all atoms on source side that are element k.

naq = the number of α-particle energy groups (integer between 1 and 4000) for α-particle source at the interface.

nq = the number of source nuclides to be evaluated (integer value between 1 and 300).

jq(k) = the source nuclide k identification in ZAID format (see Section III).

aq(k) = the fraction of all atoms on source side that are source nuclide k.

title2 = title record for the target side (maximum of 77 characters).

nzt = the number of stopping cross section elemental constituents present in the target material (must be an integer between 0 and 20).

isgt = the type of stopping cross sections to be used for the target side (0 for solid stopping cross sections, 1 for gas stopping cross sections).

jzt(k) = the atomic number of each stopping cross section elemental constituent from j=1 to *nzt* for the target side.

azt(k) = the fraction of all atoms on the target side that are element k.

nng = the number of neutron spectrum energy groups (integer between 1 and 750 or between -1 and -750); read only if *id*=2,

otherwise omitted. If *nng* is positive, then the energy group structure will be determined by a linear interpolation between *enmax* and *enmin* (cards 5.1 through 5.*nng* are omitted). If *nng* is negative, then the energy group upper bounds must be specified on cards 5.1 through 5.*nng* (however *enmax* and *enmin* must still be included).

enmax = the maximum neutron energy in MeV (read only if *id*=2, otherwise omitted).

enmin = the minimum neutron energy in MeV (read only if *id*=2, otherwise omitted).

en(n) = the upper energy bound in MeV of neutron groups, listed in descending order (must contain *nng* records in any format); read only if *id*=2 and *nng* is negative (otherwise omitted).

nt = the number of target nuclides (integer value between 1 and 20).

nag = the number of α-particle energy groups to be used in calculation (integer value between 1 and 4000).

idt(i) = the target nuclide i identification in ZAID format (see Section III).

at(i) = the fraction of all atoms on target side that are nuclide i.

Two example input decks are listed below. These examples illustrate the proper usage of the cards and records described above.

Example Problem #4 consists of a slab of weapons grade plutonium (WPu) adjacent to a slab of Be. The problem has *idd*=2 to signify an interface problem and *id*=2 for a magnitudes and spectra solution. The WPu consists of 5 isotopes of Pu (Pu-238, Pu-239, Pu-240, Pu-241, and Pu-242) and one isotope of Am (Am-241) as a contaminant. Thus, *nzq*=2 (for Pu and Am), and *nq*=6 (for the six isotopes of Pu and Am). Solid-stopping cross sections were used for both the source and target side (*isgq*=*isgt*=0). The α-particle energy structure at the interface consists of 100 groups linearly interpolated between 6.50 and 0.0000001 MeV. The target is composed of beryllium metal, thus *nzt*=1

and *nt*=1 (for Be-9 only). The neutron energy group structure is defined to contain 20 groups linearly interpolated between 10.0 and 0.0 MeV.

Example Problem #5 models a problem with a pure Am-241 source material interfaced with an AlB_2 plate. This example solves for the neutron source magnitudes only (*id*=1) using only Am-241 as the source material (*nzq*=1). The target material is made of Al and B (*nzt*=2). Note that the elemental constituents can be entered in any order [i.e., Al (Z=13) before B (Z=5)]; however, the target isotopes must be in increasing ZAID order. Three (α,n) target isotopes are present: B-10, B-11, and Al-27. The outputs for all of the above examples will be presented in Appendix A. Several sample problems that show other input decks are described in Section VI.

Example Problem #4 - Weapons Grade Pu-Be Interface Source Calculation for Magnitudes and Spectra.

```
Example 4 - WPu-Be Interface Problem
2 2
2 0 6.50 0.0000001
  94 0.9998
  95 0.0002
100
6
  942380 0.0005
  942390 0.9233
  942400 0.0650
  942410 0.0100
  942420 0.0010
  952410 0.0002
target is composed of Be
1 0
  4 1.0
20 10.0 0.0
1 4000
  40090 1.0
```

Example Problem #5 - Am-AlB_2 Interface Calculation for Neutron Source Magnitudes and Spectra.

```
Example 5 - Am-AlB2 Interface Problem
2 1
1 0 6.50 0.000001
  95 1.0
52
1
  952410 1.00
target is composed of AlB2
2 0
  13 0.333333
  5  0.666667
3 4000
   50100 0.132667
   50110 0.534000
  130270 0.333333
```

C. Beam Problems (*idd*=3)

The input deck for a beam problem is traditionally simpler than that for an interface or homogeneous problem because the problem is devoid of any source nuclides to describe. A beam problem must contain at least eight cards that describe the elemental constituents in the material, the neutron energy group structure to be used in the output, the α-particle beam energy, the type of stopping cross sections to be used, and the (α,n) target nuclides present. If multiple materials are present or a neutron energy spectrum is requested, then more cards can exist in the input deck. Cards 1–8 are as follows:

card 1:	*title*
card 2:	*idd id*
card 3:	*nz isg*
card 4.1 - 4.nz:	*jzm(j) azm(j)*
card 5:	*nng enmax enmin* (if necessary)
card 5.1 - 5.nng:	*en(n)* (if necessary)
card 6:	*ebeam*
card 7:	*nt nag*
card 8.1 - 8.nt:	*idt(i) at(i)*

Note that multiple cards are designated by subcards (i.e., if *nz*=3, then the input deck would include cards 3, 4.1, 4.2, and 4.3). Each subcard must be entered on a new line

(the exception being cards 5.1 through 5.*nng* where all records *en(n)* can be entered on the same line or multiple lines). Each record is defined as follows:

nz = the number of stopping cross section elemental constituents present in the material (must be an integer between 0 and 20).

isg = the type of stopping cross sections to be used (0 for solid stopping cross sections, 1 for gas stopping cross sections).

jzm(j) = the atomic number of each stopping cross section elemental constituent from j = 1 to *nz*.

azm(j) = the fraction of all atoms that are element j.

nng = the number of neutron spectrum energy groups (integer between 1 and 750 or between -1 and -750); read only if *id*=2, otherwise omitted. If *nng* is positive, then the energy group structure will be determined by a linear interpolation between *enmax* and *enmin* (cards 5.1 through 5.*nng* are omitted). If *nng* is negative, then the energy group upper bounds must be specified on cards 5.1 through 5.*nng* (however *enmax* and *enmin* must still be included).

enmax = the maximum neutron energy in MeV (read only if *id*=2, otherwise omitted).

enmin = the minimum neutron energy in MeV (read only if *id*=2, otherwise omitted).

en(n) = the upper energy bound in MeV of neutron groups, listed in descending order (must contain *nng* records in any format); read only if *id*=2 and *nng* is negative (otherwise omitted).

ebeam = the α-particle beam energy in MeV.

nt = the number of target nuclides (integer value between 1 and 20).

nag = the number of α-particle energy groups to be used in calculation (integer value between 1 and 4000).

idt(i) = the target nuclide i identification in ZAID format (see Section III).

at(i) = the fraction of all atoms that are target nuclide i.

Example Problem #6 below illustrates the procedure described above. This beam problem (*idd*=3) consists of a slab of silicon dioxide bombarded by 5.5 MeV α-particles (*ebeam*=5.5). Solid stopping cross section values (*isg*=0) are used for the two elemental constituents (*nz*=2) present in the problem (Si and O). The problem solves for the neutron source magnitudes and spectra (*id*=2) resulting from four (*nt*=4) target isotopes (O-17, O-18, Si-29, and Si-30). The outputs for this problem will be presented in Section V. Also several sample problems that show other input decks are described in Section VI.

Example Problem #6 - 5.5 MeV α-particle Beam Incident on a Slab of Silicon Dioxide.

```
Example 6 - Alpha Beam (5.5 MeV) on SiO2
3 2
2 0
       8 0.666667
      14 0.333333
-22 10.0 0.0
  10.00 7.00 6.00 5.50 5.00 4.50 4.00 3.50 3.25
   3.00 2.75 2.50 2.25 2.00 1.75 1.50 1.25 1.00
   0.75 0.50 0.25 0.10
5.5
4 4000
   80170 0.000253
   80180 0.001333
  140290 0.015567
  140300 0.010333
```

D. Three Region Interface Problems (*idd*=4)

A three-region interface problem input deck is divided into four sections. The first section contains information regarding the energy and angular grids to be used in the calculations. The remaining three sections pertain to each of the three slab regions. The deck must contain at least 15 cards that are used to describe the α-particle energy grid at each interface, the neutron energy grid for the output, the angular grid, material constituents for all regions, the source nuclides present, the target nuclides present, and the types of stopping cross sections to be used. The cards are described as follows:

card 1: *title*

card 2:	*idd id*
card 3:	*nag eamax eamin*
card 4:	*nng enmax enmin* (if necessary)
card 4.1 - 4.nng:	*en(n)* (if necessary)
card 5:	*ncg*
card 6:	*title1*
card 7:	*nza isga*
card 8.1 - 8.nza:	*jza(j) aza(j)*
card 9:	*nq*
card 10.1 - 10.nq:	*jq(k) aq(k)*
card 11:	*title2*
card 12:	*nzb isgb anumb t*
card 13:	*jzb(k) azb(k)*
card 14:	*ntb*
card 15.1 - 15.ntb:	*idb(i) atb(i)*
card 11:	*title3*
card 12:	*nzc isgc*
card 13:	*jzc(k) azc(k)*
card 14:	*ntc*
card 15.1 - 15.ntc:	*idc(i) atc(i)*

Multiple cards are designated by subcards (i.e., if *nza*=3, then the input deck would include cards 7, 8.1, 8.2, and 8.3). Each subcard must be entered on a new line (the exception being cards 4.1 through 4.*nng* where all records *en(n)* can be entered on the same line or multiple lines). Each record is defined as follows:

nag = the number of α-particle energy groups (integer between 1 and 4000) for α-particle source at each interface.

eamax = the maximum α-particle energy for α-particle source at each interface.

eamin = the minimum α-particle energy for α-particle source at each interface.

nng = the number of neutron spectrum energy groups (integer between 1 and 750 or between -1 and -750); read only if *id*=2, otherwise omitted. If *nng* is positive, then the energy group structure will be determined by a linear interpolation between *enmax* and *enmin* (cards 4.1 through 4.*nng* are omitted). If *nng* is negative, then the energy group upper bounds must be specified on cards 4.1 through 4.*nng* (however *enmax* and *enmin* must still be included).

enmax = the maximum neutron energy in MeV (read only if *id*=2, otherwise omitted).

enmin = the minimum neutron energy in MeV (read only if *id*=2, otherwise omitted).

en(n) = the upper energy bound in MeV of neutron groups, listed in descending order (must contain *nng* records in any format); read only if *id*=2 and *nng* is negative (otherwise omitted).

ncg = the number of α-particle angular groups (integer between 1 and 4000) for α-particle source at each interface.

title1 = title record for the region A (maximum of 77 characters).

nza = the number of stopping cross sections elemental constituents in region A (up to 20).

jza(k) = the atomic number of each stopping cross section elemental constituent from j = 1 to *nza* for the region A.

aza(k) = the fraction of all atoms in region A that are element k.

nq = the number of source nuclides to be evaluated (integer value between 1 and 300).

jq(k) = the source nuclide k identification in ZAID format (see Section III).

aq(k) = the fraction of all atoms in region A that are source nuclide k.

title2 = title record for the region B (maximum of 77 characters).

nzb = the number of stopping cross section elemental constituents present in region B (must be an integer between 0 and 20).

isgb = the type of stopping cross sections to be used for the target side (0 for solid stopping cross sections, 1 for gas stopping cross sections).

anumb = the atomic number density of all materials in region B (in atoms/b-cm)

t = the thickness (in cm) of region B.

jzb(k) = the atomic number of each stopping cross section elemental constituent from j=1 to *nzb* in region B.

azb(k) = the fraction of all atoms in region B that are element k.

ntb = the number of target nuclides (integer value between 1 and 20).

idb(i) = the target nuclide i identification in ZAID format (see Section III) for targets in region B.

atb(i) = the fraction of all atoms in region B that are nuclide i.

title3 = title record for the region B (maximum of 77 characters).

nzc = the number of stopping cross section elemental constituents present in region B (must be an integer between 0 and 20).

isgc = the type of stopping cross sections to be used for the target side (0 for solid stopping cross sections, 1 for gas stopping cross sections).

jzc(k) = the atomic number of each stopping cross section elemental constituent from j=1 to *nzb* in region B.

azc(k) = the fraction of all atoms in region B that are element k.

ntc = the number of target nuclides (integer value between 1 and 20).

idc(i) = the target nuclide i identification in ZAID format (see Section III) for targets in region B.

atc(i) = the fraction of all atoms in region B that are nuclide i.

Two example input decks are listed below. These examples illustrate the proper usage of the cards and records described above.

The first example problem consists of a slab of weapons grade plutonium (WPu) adjacent to a slab of Be with a thin layer of Al for region B. The problem has *idd*=4 to signify a three-region interface problem and *id*=2 for a magnitudes and spectra solution. The WPu consists of 5 isotopes of Pu (Pu-238, Pu-239, Pu-240, Pu-241, and Pu-242) and one isotope of Am (Am-241) as a contaminant. Thus, *nzq*=2 (for Pu and Am), and *nq*=6 (for the six isotopes of Pu and Am). Solid-stopping cross sections were used for all regions (*isga*=*isgb*=*isgc*=0). The α-particle energy structure at each interface consists of 400 groups linearly interpolated between 6.50 and 0.0000001 MeV. Region B is composed of Al metal [*nzb*=1 and *ntb*=1 (for Al-27 only)] with a density of 0.15 atoms/b-cm and a thickness of 1 mm. Region C is composed of beryllium metal, thus *nzc*=1 and *ntc*=1 (for Be-9 only). The neutron energy group structure is defined to contain 20 groups linearly interpolated between 10.0 and 0.0 MeV. Forty angular groups are used at each interface (*ncg*=40).

Example Problem #7 - Weapons Grade Pu-Al-Be Interface Source Calculation for Magnitudes and Spectra.

```
Example #7 (WPu-Al-Be)
4 2
400 6.5 0.0000001
20 10.0 0.0
40
WPu region
2 0
  94 0.9998
  95 0.0002
6
  942380 0.0005
  942390 0.9233
  942400 0.0650
  942410 0.0100
  942420 0.0010
  952410 0.0002
Al interface
1 0 0.5 0.1
  13 1.0
1
 130270 1.0
Be reflector
1 0
  4 1.0
1
  40090 1.0
```

Example Problem #8 models a problem with a pure Am-241 source material interfaced with an AlB_2 plate with a small (3.0 cm thick) CO_2 gap. This example solves

for the neutron source magnitude and spectra (*id*=2) using only Am-241 as the source material (*nza*=1). Region B consists of two elements (*nzb*=2) and three (α,n) target nuclides (*ntb*=3). Gas stopping powers are used in region B (*isgb*=1). The target material in region C is made of Al and B (*nzc*=2). Note that the elemental constituents can be entered in any order [i.e., Al (Z=13) before B (Z=5)]; however, the target isotopes must be in increasing ZAID order. Three (α,n) target isotopes are present in region C: B-10, B-11, and Al-27. The outputs for all of the above examples will be presented in Appendix A.

Example Problem #8 - Am-CO_2-AlB_2 Interface Calculation for Neutron Source Magnitudes and Spectra.

```
Example 8 - Am-CO2-AlB2 Interface Problem
4 2
400 6.5 0.0000001
20 10.0 0.0
60
Pure Am-241 in region A
1 0
  95 1.0
1
  952410 1.0
CO2 gas in region B
2 1 0.004 3.0
   6 0.333
   8 0.667
3
  60130 0.0073333
  80170 0.0002667
  80180 0.0013333
AlB2 shield in region C
2 0
  13 0.33333
   5 0.66667
3
   50100 0.132667
   50110 0.534000
  130270 0.333333
```

E. Execution

To execute the SOURCES 4A code, one needs to run the executable file. The input deck must be named *tape1*, and the library files must be named *tape2* through *tape5*. After execution, SOURCES 4A will display a STOP message informing the user whether

the code was executed normally or if any errors existed during execution. In most cases, the SOURCES 4A STOP message will inform the user of the cause of any execution error.

If any errors occur during execution, the user should first check the file *outp* that is created during SOURCES 4A execution (see Section V). *outp* contains a summary of all input read from *tape1*. The majority of all errors from SOURCES 4A are from an improperly constructed input deck. If no errors exist in the input deck, then the next most common error occurs when SOURCES 4A attempts to run a problem for which isotopic data does not exist. The user should recheck the ZAID specified in *tape1* with the isotopes listed in Tables I and II. The users should only make modifications to the library files as a last resort to correcting any errors.

V. DESCRIPTION OF OUTPUT

Depending upon the neutron source output requested, SOURCES 4A can create between two and five output files. If the input deck specifies a magnitudes only problem (*id*=1), then only two output files, *outp* and *tape6,* are created. If the neutron source spectra (*id*=2) is requested in the input deck (*tape1*), then all the files specified in Section III are created (*tape6* to *tape9* and *outp*). The exception being for interface problems in which the file *tape9* is not created. This is because the source α-particles on the target side may have numerous groups and would make the file extremely large. Each output file has a header summarizing the contents of each file and a card listing the problem title.

The file *outp* contains a summary of the input deck as read by SOURCES 4A and of all output decks created by SOURCES 4A. The user should always check this file after running SOURCES 4A to ensure that all input was read as intended. The file *outp* also lists the neutron source strengths and average energies by decay mode and total for all modes. Lastly, *outp* shows the portion of the neutron source rates included in the neutron spectrum calculations [for total, (α,n), spontaneous fission, and delayed neutron, when necessary]. In other words, the ratio of the neutron source magnitude, calculated by a summation over all energy groups, to the total calculated neutron source magnitude. This value is a measure of how well the energy structure was chosen. If the value is below a certain percentage (~90%), the energy structure may have been chosen with *enmax* significantly less than the maximum neutron energy produced in the system. In this case, the user may want to choose a different energy structure with a larger group 1 upper-bound.

The *tape6* file lists the neutron source magnitudes by target nuclide and by source α-particle. The file contains up to three tables labeled Table I, Table II, and Table III. These tables report the neutron production rates from (α,n) reactions, spontaneous fission, and delayed neutron emission in that order. The tables have appropriate headings for the neutron production parameters with units included. The *outp* and *tape6* files for the input deck in Example Problems #1, #2, #4, and #6 are listed in Appendix A.

The *tape7* file lists the absolute neutron spectra (i.e., neutrons per second per unit volume per unit energy). This file first lists the multigroup neutron spectra (i.e., the energy bounds) used in the calculations in decreasing order. The absolute neutron spectra listed by neutron/target combination is then displayed in order coinciding with the group structure specified at the beginning of the file. Totals per target nuclide for (α,n) reactions, totals for all (α,n) reactions, totals for spontaneous fission neutrons, and totals for delayed neutrons are also listed (if applicable). The file *tape8* is similar to *tape7* except that normalized neutron spectra are reported. *Tape9* lists the energy dependent neutron production rates by target nuclide and by product nuclide level. Again, the neutron energy group structure is first listed and then a breakdown of the neutron energy spectra with target nuclide totals is reported. The *tape7* output files corresponding to the input files listed in Example Problems #2, #4, and #6 are included in Appendix A, as well as the *tape9* output files for Example Problems #2 and #4.

VI. SAMPLE PROBLEMS

Several sample problems are listed below. These problems are available to aid the user in construction of the input deck (*tape1*). The problems were executed to model experimental arrangements, and measured data are reported with the SOURCES 4A calculation when available.

A. Homogeneous Mixtures

1. Sample Problem #1

This problem illustrates the neutron source magnitudes and spectra from a $PuBe_{13}$ source (elemental constituents are $^{13}/_{14}$ Be and $^{1}/_{14}$ Pu) with six isotopes of Pu (Pu-237, Pu-238, Pu-239, Pu-240, Pu-241, and Pu-242) and one isotope of Be (Be-9) present. The example solves for the magnitude and spectra (*id*=2) and uses a 48 group neutron energy structure (*nng*=48) which is linearly interpolated between 12.0 (*enmax*) and 0.0 (*enmin*) MeV. The six Pu isotopes are used as sources (*nq*=6), and the one beryllium isotope is the target (*nt*=1). The atom fractions and densities can be entered in scientific notation or decimal notation. This input deck is an appropriate model of the experimental measurement performed by L. Stewart.[14]

```
Sample 1 - PuBe13 Source (Stewart, 1953)
1 2
2 0
    4 0.928571
   94 0.071429
48 12.0 0.0
6
   942370 13144.0
   942380 7.08e+17
   942390 5.82e+21
   942400 3.74e+20
   942410 1.69e+19
   942420 1.22e+18
1 4000
   40090 0.928571
```

Fig. 8. Sample Problem #1 Input Deck.

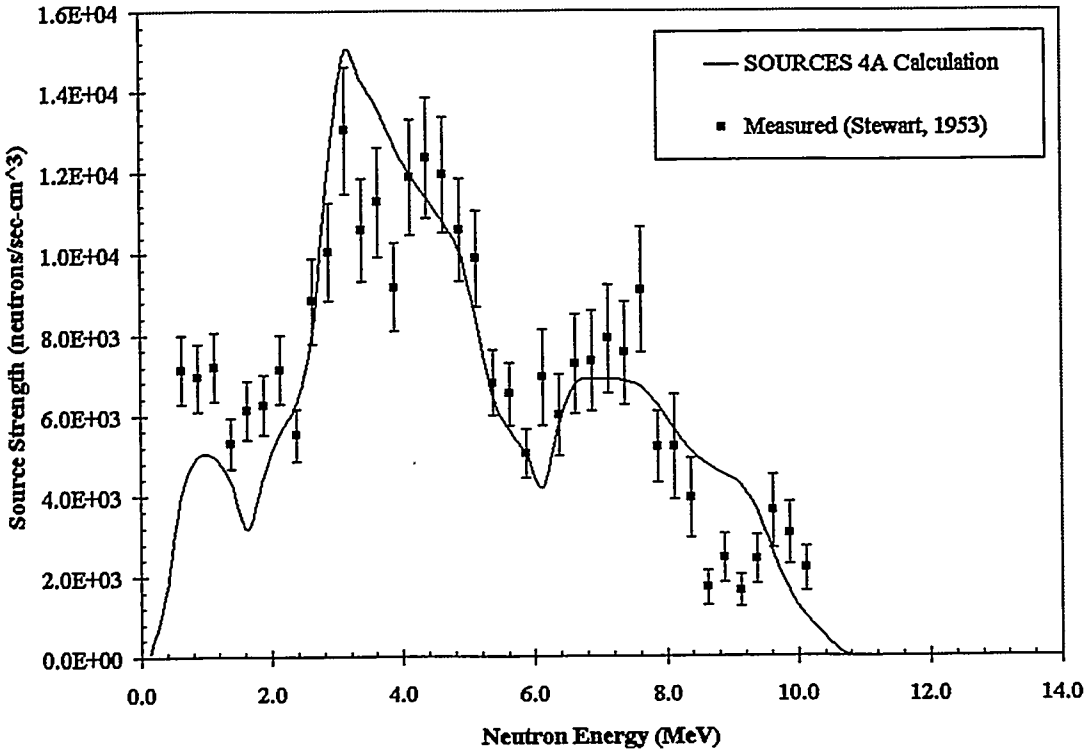


Fig. 9. Energy-Dependent Neutron Source Strength in $PuBe_{13}$ Homogeneous Problem as Calculated by SOURCES 4A and Compared with Measured Data.

A comparison of the data measured by the experimenters and the SOURCES 4A calculation is presented in Fig. 9. To construct this plot, the histogram output from SOURCES 4A was converted to a continuous distribution using the midpoint energy for each energy group. This conversion was repeated for all energy-dependent neutron source plots in this section (i.e., Figs. 8, 11, 14, 16, 18, and 20). The total neutron source magnitude calculated by SOURCES 4A was 2.69×10^5 neutrons/s-cm^3, whereas the experimenters reported a total neutron source rate of 2.28×10^5 neutrons/s-cm^3. This magnitude of agreement (±17%) is standard for a SOURCES 4A calculation. From Fig. 9, reasonable agreement between the SOURCES 4A spectrum calculation and the measured values is found. The calculation neglected any source contaminants (esp., Am-241), because they were not specified in the published experiment.

2. Sample Problem #2

Several experiments have been performed to analyze criticality accidents involving uranium-containing solutions. These experiments have been primarily interested in gross measurements of k_{eff}. However, in 1991 a measurement was performed by Seale and Andersen[15] to record neutron production rates from uranyl fluoride and uranyl nitrate. The researchers' data for uranyl fluoride was found to be reproducible in several different samples, all of different total volumes. Thus, the uranyl fluoride experiment was chosen for modeling by SOURCES 4A. A calculation was performed using the data listed by the experimenters. The solution was uranyl fluoride (UO_2F_2) with a density of 2.16 g/ml. The uranium concentration consisted of 5% U-235 by weight. The SOURCES 4A input deck for this problem is listed in Fig. 10. The source nuclides included U-234, U-235, and U-238. No contaminants were listed by the researchers; thus, none were included in the calculation. The three (α,n) target nuclides shown are O-17, O-18, and F-19. O-16 was not included because of its negligible (α,n) reaction cross section. The problem was executed, and SOURCES 4A reported a neutron production rate of 0.0469 neutrons/s-cm^3. The experimenters had measured a neutron production rate of 0.0421 $\pm$ 0.0016 neutrons/s-cm^3. This yields a discrepancy between the SOURCES 4A calculation and the experimentally measured value of 11.5%. This is generally considered good agreement for this type of calculation.

```
Sample 2 - Uranyl-Fluoride Solution (Seale, 1991)
1 1
3 0
    8 0.4
    9 0.4
   92 0.2
3
   922340 1.32e+17
   922350 2.11e+20
   922380 4.01e+21
3 4000
   80170 0.000152
   80180 0.0008
   90190 0.4
```

Fig. 10. Sample Problem #2 Input Deck.

```
Sample 3 - PoBe Source (Speck, 1944)
1 2
2 0
        4 0.99999886
       84 0.00000114
60 12.0 0.0
1
   842100 6.38e+16
1 4000
    40090 0.99999886
```

Fig. 11. Sample Problem #3 Input Deck.

3. *Sample Problem #3*

Using the photographic emulsion method, researchers at Los Alamos National Laboratory measured the neutron energy spectrum from a Po-Be source.[16] The source was a mixture of Po and Be metals in the shape of a cylinder $^3/_8$" in diameter and ¼" in height. The total source activity was reported by the researchers as 100 mCi. This source configuration was modeled as a homogeneous source problem using SOURCES 4A. The input deck (*tape1*) used in this calculation is shown in Fig. 11. The problem was executed to acquire the neutron source energy spectrum plotted in Fig. 12.

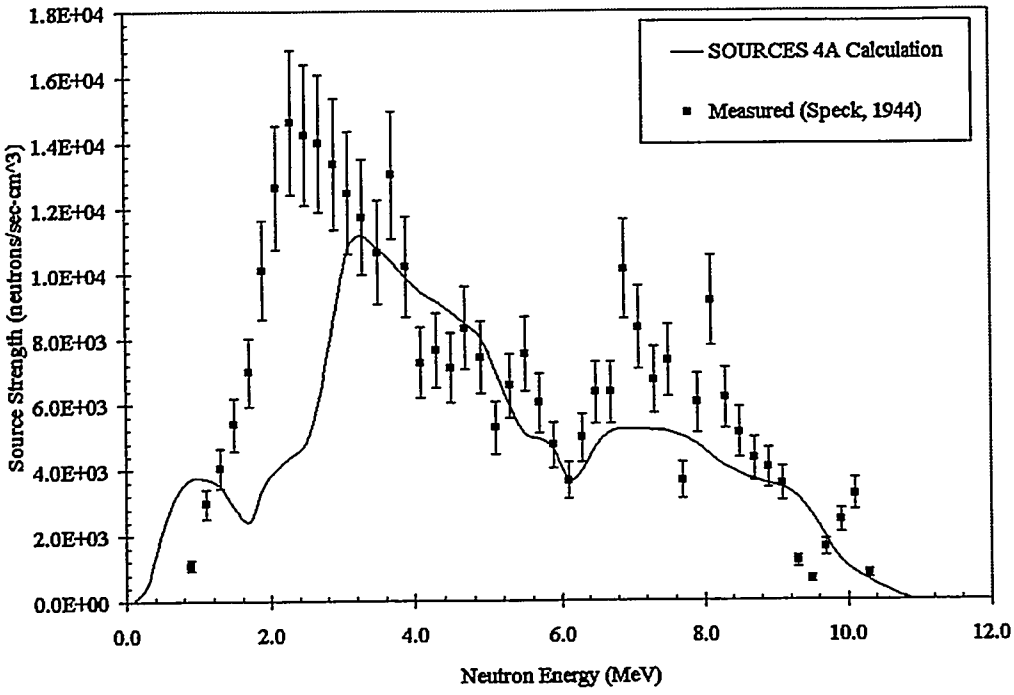


Fig. 12. Energy-Dependent Neutron Source Strength in Po-Be Homogeneous Problem as Calculated by SOURCES 4A and Compared to Measured Data.

An analysis of the data in Fig. 12 shows that the SOURCES 4A calculation appears to overestimate the average neutron energy produced from the sample. On further analysis, it can be found that a Po-Be source, though a mixture of α-emitting material and (α,n) target material, is composed of grains of Po and grains of Be. These grains have an average diameter significantly larger than the α-particle range. Thus, it is postulated that a Po-Be source is more properly modeled as an interface problem of Po and Be. This theory is supported by the calculations performed in Sample Problem #5.

The outcome of this problem is extremely important. A SOURCES 4A user must be aware of the physics inherent in any problem being modeled. In the case of Sample Problems #1 and #2, the materials were compounds ($PuBe_{13}$ or UO_2F_2). Thus, the α-emitting nuclides and (α,n) target nuclides are intimately mixed. For Po-Be, the material has a tendency to clump into grains, and the grain size of the metals can and will affect the outcome of the calculations. Therefore, it is imperative that a user analyze the chemical nature of any problem under consideration prior to constructing the input deck.

B. Interface Problem Examples

1. Sample Problem #4

In 1944, a study was conducted by Perlman *et al.*[17] at Los Alamos National Laboratory to explore the possibility of using an (α,n) neutron source to simulate a fission neutron spectra. In this study, a series of platinum foils (3 x 3 cm in size) were coated with 180 mCi of Po and then interleaved between sintered B_4C slabs. The entire Po-B_4C assembly was placed in a brass box and sealed under a slight vacuum. The resultant neutron energy spectra from the source was measured using the photographic emulsion method. A schematic of the experimental setup is shown in Fig. 13.

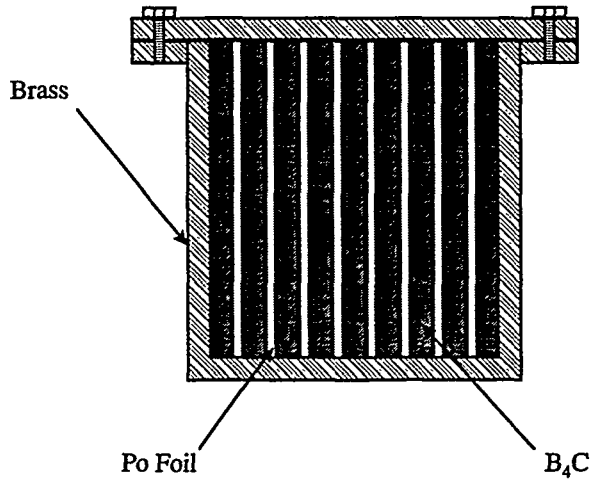


Fig. 13. Po-B_4C Source Arrangement for Sample Problem #4.

To model this arrangement, a SOURCES 4A input deck (Fig. 14) was constructed which employed the interface problem capabilities of SOURCES 4A (*idd*=2). The atomic fractions listed in the input deck show that natural boron (19.9% B-10 and 80.1% B-11) and carbon (98.9% C-12 and 1.1% C-13) were used. Also the α-particle source was set to include 100% Po-210. This input deck was executed to solve for the neutron source spectra and magnitudes (*id*=2) resulting from the (α,n) interactions in the boron carbide.

```
Sample 4 - Po-B4C Interface Experiment (Perlman, 1944)
2 2
1 0 10.0 0.0000001
     84 1.0
50
1
  842100 1.00
target is a B4C slab
2 0
       5 0.8
       6 0.2
50 10.0 0.0
3 4000
   50100 0.1592
   50110 0.6408
   60130 0.0022
```

Fig. 14. Sample Problem #4 Input Deck.

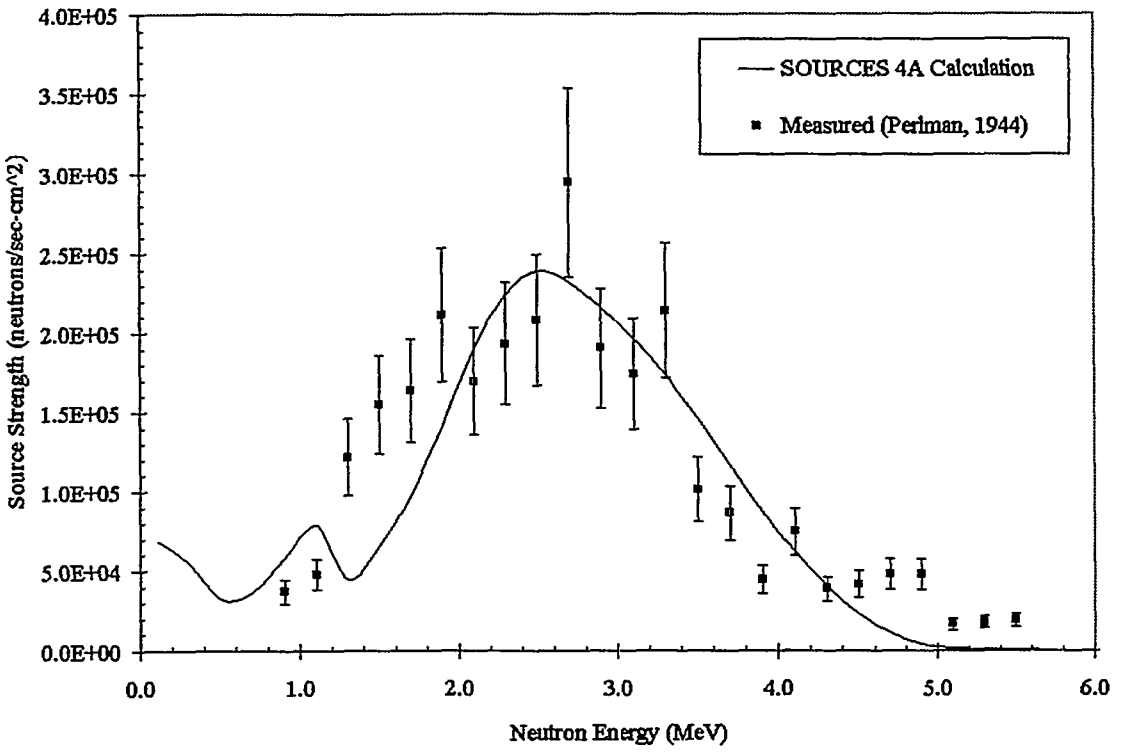


Fig. 15. Energy-Dependent Neutron Source Strength in Po-B_4C Interface Problem as Calculated by SOURCES 4A and Compared to Measured Data.

The energy spectrum of the calculated Po-B_4C neutrons is plotted in Fig. 15, along with the measured data. The agreement between the measured data and the SOURCES 4A calculation is reasonable and typical of an interface problem. The addition of contaminants and other Po isotopes could greatly affect the shape of this spectrum. It is important to note that the researchers did not specify the presence of any contaminants in the samples.

2. Sample Problem #5

In Sample Problem #3, a Po-Be source was investigated as a possible homogeneous problem. It was discovered that the energy spectrum of the neutrons calculated by SOURCES 4A was shifted to a higher average energy than what was reported by the experimenters.[16] It was suggested that this shift was due to the grain structure of the materials in Po-Be sources. To verify this hypothesis, SOURCES 4A was used to model

the identical experiment using its interface capabilities. The Po-Be input deck used for this model is shown in Fig. 16. The figure shows that the entire source side is composed of Po-210, and the entire target side is composed of a Be-9. The neutron energy group structure is the same as that used in Sample Problem #3.

```
Sample 5 - Po-Be Interface Experiment (Speck, 1944)
2 2
1 0 10.0 0.000001
      84 1.0
50
1
  842100 1.00
target is a Be slab
1 0
       4 1.0
60 12.0 0.0
1 4000
   40090 1.0
```

Fig. 16. Sample Problem #5 Input Deck.

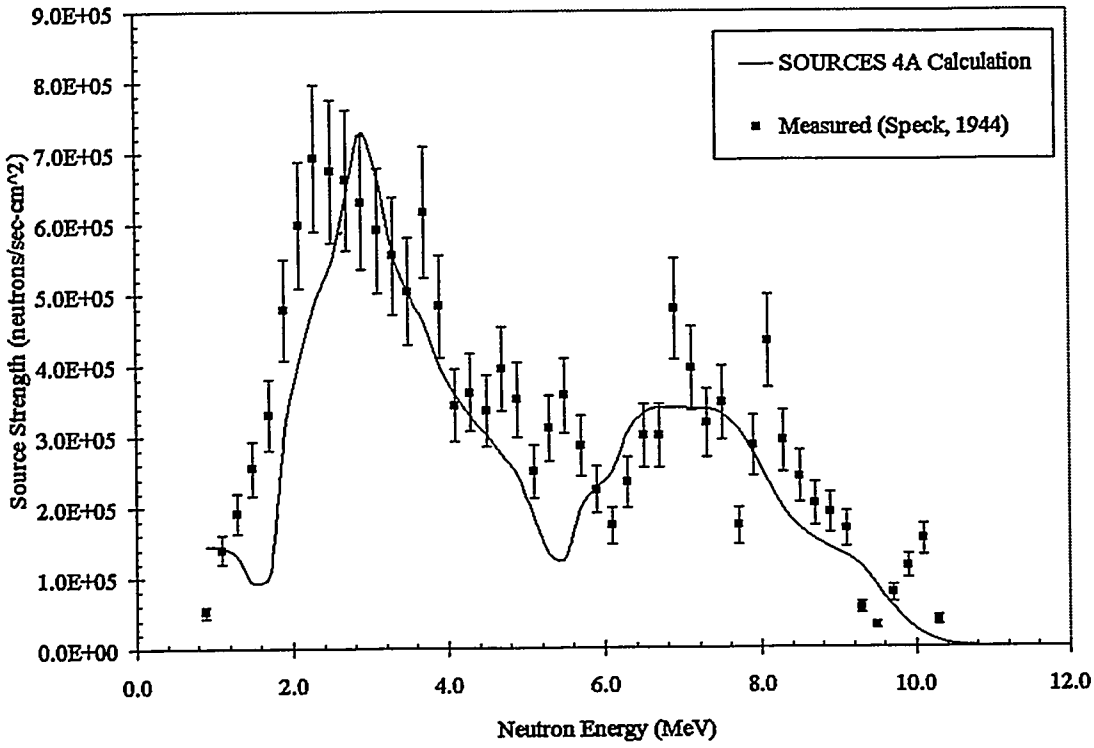


Fig. 17. Energy-Dependent Neutron Source Strength in Po-Be Interface Problem as Calculated by SOURCES 4A and Compared to Measured Data.

The output from the execution of Sample Problem #5 was plotted versus the data given by the experimenters (Fig. 17). The SOURCES 4A calculation and the measured values appear to agree to within a good degree of accuracy. This yields confirmation that a Po-Be (α,n) source is affected by the grain structure of its metal components. It is important that any SOURCES 4A users consider this type of problem when modeling any realistic (α,n) sources.

C. Beam Problem Examples

In 1983, researchers in Belgium used a 7 MV Van de Graaff accelerator and an NE213 liquid scintillator to analyze the energy spectra of neutrons produced by 4 to 5.5 MeV α-particles on thick targets of light elements (e.g., C, O, Mg, F, Al, Si, Al_2O_3, and SiO_2).[18] The experimenters measured the energy spectra of neutrons in 0.1 MeV bins and reported these spectra, the total neutron yields per incident α-particle, and the average neutron energies for each sample. Two samples bombarded by the experimenters are modeled below using SOURCES 4A. The first sample was aluminum oxide (Sample Problem #6) and the second sample was pure magnesium (Sample Problem #7). The measured and calculated data for each are plotted in Figs. 19 and 21. Also, the average neutron energy and the total neutron yield per incident α-particle is reported.

1. Sample Problem #6

The input deck used for modeling the bombardment of aluminum oxide by 5.0 MeV α-particles is listed in Fig. 18. The energy structure is divided into 45 bins of 0.1 MeV width; however, the lower energy cutoff is above 0.0 MeV.

```
Sample 6 - Alpha Beam (5.0 MeV) on Al2O3
3 2
2 0
       8 0.6
      13 0.4
55 5.55 0.05
5.0
3 4000
   80170 0.000228
   80180 0.0012
  130270 0.4
```

Fig. 18. Sample Problem #6 Input Deck.

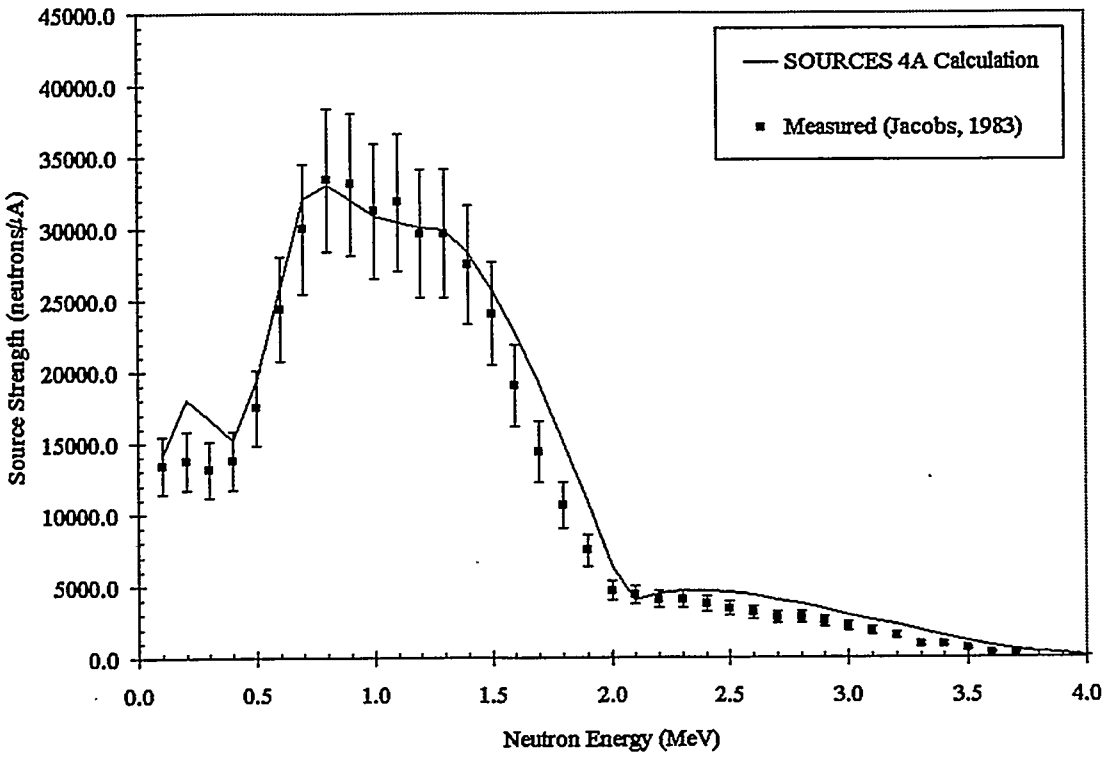


Fig. 19. Energy-Dependent Neutron Source Strength from 5.0 MeV α-Particles Incident on Aluminum Oxide Slab as Calculated by SOURCES 4A and Compared to Measured Data.

The total neutron yield per incident α-particle was reported by the experimenters to be 1.58×10^{-7} neutrons/α-particle.[18] The SOURCES 4A calculation output a value of 1.63×10^{-7} neutrons/α-particle. The measured average neutron energy was 1.14 ± 0.04 MeV whereas SOURCES 4A reported an average energy of 1.35 MeV. As can be seen, the total neutron yields agree to within 4%. The energy-dependent neutron spectra are plotted in Fig. 18. The neutron spectra have a few small discrepancies, and the average neutron energies show an 18% difference; however the agreement is generally very good. The SOURCES 4A total neutron yields consistently have better agreement to measured data than the spectral calculations. However, as is shown here and in Sample Problem #7, for beam problems the SOURCES 4A calculations are excellent for both magnitudes and spectra.

2. *Sample Problem #7*

The input deck used for modeling the magnesium irradiation by 5.5 MeV α-particles is shown in Fig. 20. The magnesium sample contains two naturally occurring isotopes (Mg-25 and Mg-26) as (α,n) target nuclides. The isotope Mg-24 was neglected due to its negligible (α,n) cross section. The neutron energy group structure consisted of 81 energy groups in 0.1 MeV bins.

```
Sample 7 - Alpha Beam (5.5 MeV) on Mg
3 2
1 0
     12 1.0
81 8.15 0.05
5.5
2 4000
  120250 0.10
  120260 0.1101
```

Fig. 20. Sample Problem #7 Input Deck.

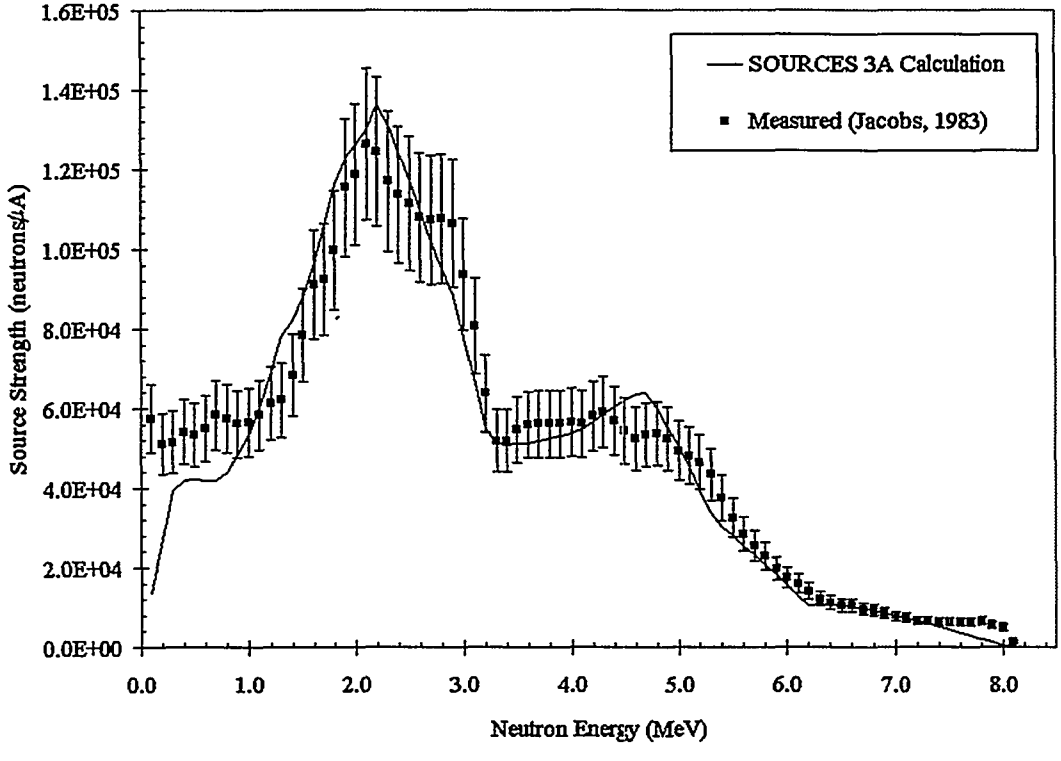


Fig. 21. Energy-Dependent Neutron Source Strength from 5.5 MeV α-Particles Incident on Magnesium Slab as Calculated by SOURCES 4A and Compared to Measured Data.

The total neutron yield per incident α-particle was reported by the experimenters to be 1.33×10^{-6} neutrons/α-particle.[18] The SOURCES 4A calculation output a value of 1.27×10^{-6} neutrons/α-particle. The measured average neutron energy was 2.85 ± 0.12 MeV, where as SOURCES 4A reported an average energy of 3.04 MeV. The total neutron yields agree to within 5%. The energy-dependent neutron spectra are plotted in Fig. 21. The measured and calculated neutron energy spectra have excellent agreement (within experimental error).

VII. ACKNOWLEDGMENTS

The authors would like to acknowledge the aid given by J. Devaney in the derivation of the theory behind the (α,n) interface problem.

VIII. REFERENCES

1. R. T. PERRY and W. B. WILSON, "Neutron Production from (α,n) Reactions and Spontaneous Fission in ThO_2, UO_2 and $(U,Pu)O_2$ Fuels," Los Alamos National Laboratory report LA-8869-MS (1981). [included in Appendix D]

2. W. B. WILSON, R. T. PERRY, J. E. STEWART, T. R. ENGLAND, D. G. MADLAND, and E. D. ARTHUR, "Development of the SOURCES Code and Data Library for the Calculation of Neutron Sources and Spectra from (α,n) Reactions, Spontaneous Fission, and β^- Delayed Neutrons," Los Alamos National Laboratory report LA-9841-PR, page 65 (1983).

3. H. J. HIRSCH and H. J. MATZKE, "Stopping Power and Range of α-Particles in (U,Pu)C and UC and Application to Self-Diffusion Measurements Using Alpha Spectroscopy," *J. of Nucl. Mater.* **45**, page 29 (1972).

4. W. H. BRAGG and R. KLEEMAN, "On the Alpha Particles of Radium and their Loss of Range in Passing through Various Atoms and Molecules," *Phil. Mag.* **10**, page 318 (1905).

5. B. G. WHITMORE and W. B. BAKER, "The Energy Spectrum of Neutrons from a Po-Be Source," *Phys. Rev.* **78**, page 799 (1950).

6. J. H. GIBBONS and R. L. MACKLIN, "Total Neutron Yields from Light Elements Under Proton and Alpha Bombardment," *Phys. Rev.*, **114**, page 571 (1959).

7. K. W. GEIGER and L. VAN DER ZWAN, "An Evaluation of the $^9Be(\alpha,n)$ Cross Section Data," *Nucl. Instr. and Methods* **131**, page 315 (1975).

8. J. K. BAIR and J. GOMEZ DEL CAMPO, "Neutron Yields from Alpha Particle Bombardment," *Nucl. Sci. and Eng.* **71**, page 18 (1979).

9. J. K. BAIR and F. X. HAAS, "Total Neutron Yield from the Reaction $^{13}C(\alpha,n)^{16}O$ and $^{17,18}O(\alpha,n)^{20,21}Ne$," *Phys. Rev. C* **7**, page 1356 (1973).

10. D. L. LESSOR and R. E. SCHENTER, "Neutron Spectra from (α,n) Reactions in Plutonium Compounds Calculated from Hauser-Feshbach Reaction Theory," Brookhaven National Laboratory report BNWL-B-109 (1971).

11. M. BALAKRISHNAN, S. KAILAS, and M. K. MEHTA, "A Study of the Reaction $^{19}F(\alpha,n)^{22}Na$ in the Bombarding Energy Range 2.6 to 5.1 MeV," *Pramana* **10**, page 329 (1978).

12. S. E. WOOSLEY, W. A. FOWLER, J. A. HOLMES, and B. A. ZIMMERMAN, OAP-422 (1975).

13. J. F. ZIEGLER, "Helium Stopping Powers and Ranges in All Elemental Matter," Vol. 4 of The Stopping Power and Ranges of Ions in Matter Series, Pergammon Press, New York (1977).

14. L. STEWART, "Determination of the Neutron Spectrum and Absolute Yield of a Plutonium-Beryllium Source," Master's Thesis, University of Texas (1953).

15. R. L. SEALE and R. E. ANDERSEN, "Intrinsic Neutron Source Strengths in Uranium Solutions," *Trans. Am. Nucl. Soc.* **63**, page 226 (1991).

16. L. SPECK and H. RICHARDS, "Energy Spectrum of Po-Be Neutrons," Los Alamos Scientific Laboratory report LA-111 (1944).

17. T. H. PERLMAN, H. T. RICHARDS, and J. H. WILLIAMS, "The Energies of the Neutrons from Polonium Alphas on Boron," Los Alamos Scientific Laboratory report LA-66 (1944).

18. G. J. H. JACOBS and H. LISKIEN, "Energy Spectra of Neutrons Produced by α-Particles in Thick Targets of Light Elements," *Ann. Nucl. Energy* **10**, page 541 (1983).

APPENDIX A

Output Files for Example Problems

Example Problem #1 (file: *outp*)

Summary of Input

Title: Example 1 - Clean UO2 Fuel (3% enriched)
Homogeneous problem input (idd= 1)
Magnitudes only computed.
Number of elemental constituents: 2
Solid stopping cross sections used (isg= 0)

Elemental Constituents:

Z-value	Atom Fraction
8	.6666667000
92	.3333333000

Number of source nuclides to be evaluated: 2

Source Nuclides:

ZAID	Atom Density (g/cc)
922350	6.770E+20
922380	2.160E+22

Number of target nuclides to be used: 2
4000 Alpha energy groups used.

Target Nuclides:

ZAID	Atom Fraction
80170	2.530E-04
80180	1.333E-03

Summary of Output

Total (alpha,n) neutron source from all sources and targets: 9.030E-04 n/sec-cm^3.

Total spontaneous fission neutron source from all sources and targets: 1.164E-01 n/sec-cm^3.

Total delayed neutron source from all sources and targets: 0.000E+00 n/sec-cm^3.

Total neutron source from all sources and targets: 1.173E-01 n/sec-cm^3.

Example Problem #1 (file: *tape6*)

Title: Example 1 - Clean UO2 Fuel (3% enriched)

1 Table I

(alpha,n) Neutron Production by Target Per Source Alpha

	target	target atom frac.	alpha source	alpha energy	alphas/sec /cm**3	p(e) neut/alpha	neuts/sec /cm**3
+	o 17	2.5300E-04	u235	4.154	1.9016E+02	6.4793E-10	1.2321E-07
				4.217	1.2043E+03	6.9694E-10	8.3933E-07
				4.227	1.9016E+02	7.0215E-10	1.3352E-07
				4.271	8.4513E+01	7.2387E-10	6.1177E-08
				4.324	9.9303E+02	7.5301E-10	7.4776E-07
				4.364	3.5918E+03	7.7551E-10	2.7855E-06
				4.398	1.1832E+04	7.9048E-10	9.3529E-06
				4.416	4.4370E+02	7.9781E-10	3.5399E-07
				4.439	1.4790E+02	8.0800E-10	1.1950E-07
				4.502	3.5918E+02	8.3379E-10	2.9948E-07
				4.556	9.5078E+02	8.5739E-10	8.1519E-07
				4.598	1.1409E+03	8.8577E-10	1.0106E-06
+						Total:	1.6642E-05
	o 17	2.5300E-04	u238	4.039	2.4366E+02	5.5553E-10	1.3536E-07
				4.149	2.4366E+04	6.4340E-10	1.5677E-05
				4.196	8.1574E+04	6.8428E-10	5.5819E-05
+						Total:	7.1632E-05
+					Total (this target):		8.8274E-05
	o 18	1.3330E-03	u235	4.154	1.9016E+02	5.8662E-09	1.1155E-06
				4.217	1.2043E+03	6.2443E-09	7.5201E-06
				4.227	1.9016E+02	6.3249E-09	1.2027E-06
				4.271	8.4513E+01	6.8207E-09	5.7644E-07
				4.324	9.9303E+02	7.3636E-09	7.3123E-06
				4.364	3.5918E+03	7.7930E-09	2.7991E-05
				4.398	1.1832E+04	8.2848E-09	9.8025E-05
				4.416	4.4370E+02	8.5493E-09	3.7933E-06
				4.439	1.4790E+02	8.8354E-09	1.3067E-06
				4.502	3.5918E+02	9.2866E-09	3.3356E-06
				4.556	9.5078E+02	9.7749E-09	9.2937E-06
				4.598	1.1409E+03	1.0321E-08	1.1776E-05
+						Total:	1.7325E-04
	o 18	1.3330E-03	u238	4.039	2.4366E+02	5.1864E-09	1.2637E-06
				4.149	2.4366E+04	5.8398E-09	1.4229E-04
				4.196	8.1574E+04	6.1034E-09	4.9788E-04
+						Total:	6.4143E-04
+					Total (this target):		8.1468E-04
+							

Total (all targets): 9.0295E-04

2

Table II

Spontaneous Fission Neutron Production

source nuclide	source atoms per cm**3	dk constant (/second)	sf decay branching	nu bar	neutrons sec/cm**3
u235	6.7700E+20	3.1209E-17	2.011E-09	1.860	7.903E-05
u238	2.1600E+22	4.9159E-18	5.450E-07	2.010	1.163E-01
				Total:	1.164E-01

Example Problem #2 (file: *outp*)

Summary of Input

Title: Example 2 - Clean UO2 Fuel (3% enriched) for Spectra
Homogeneous problem input (idd= 1)
Magnitudes and spectra computed.
Number of elemental constituents: 2
Solid stopping cross sections used (isg= 0)

Elemental Constituents:

Z-value	Atom Fraction
8	.6666667000
92	.3333333000

Number of neutron spectrum energy groups: 10
Maximum neutron energy is 1.000E+01 MeV.
Minimum neutron energy is 0.000E+00 MeV.

Energy Group Structure:

Group	Upper-Bound	Lower-Bound
1	1.000E+01	9.000E+00
2	9.000E+00	8.000E+00
3	8.000E+00	7.000E+00
4	7.000E+00	6.000E+00
5	6.000E+00	5.000E+00
6	5.000E+00	4.000E+00
7	4.000E+00	3.000E+00
8	3.000E+00	2.000E+00
9	2.000E+00	1.000E+00
10	1.000E+00	0.000E+00

Number of source nuclides to be evaluated: 2

Source Nuclides:

ZAID	Atom Density (g/cc)
922350	6.770E+20
922380	2.160E+22

Number of target nuclides to be used: 2
4000 Alpha energy groups used.

Target Nuclides:

ZAID	Atom Fraction
80170	2.530E-04
80180	1.333E-03

Summary of Output

Total (alpha,n) neutron source from all sources and targets: 9.030E-04 n/sec-cm^3.

Total spontaneous fission neutron source from all sources and targets: 1.164E-01 n/sec-cm^3.

Total delayed neutron source from all sources and targets: 0.000E+00 n/sec-cm^3.

Total neutron source from all sources and targets: 1.173E-01 n/sec-cm^3.

Average (alpha,n) neutron energy: 2.012E+00 MeV.

Average spontaneous fission neutron energy: 1.68826E+00 MeV.

Average delayed neutron energy: 0.000E+00 MeV.

Average neutron energy from all sources: 1.691E+00 MeV.

Normalized Neutron Energy Spectrum by Energy Group for All Sources:

Group	Contribution
1	1.934E-04
2	5.838E-04
3	1.715E-03
4	4.879E-03
5	1.333E-02
6	3.458E-02
7	8.364E-02
8	1.848E-01
9	3.299E-01
10	3.463E-01

Portion of Total Neutron Source Rate Accounted for in the Total Energy Spectrum: 100.0%.

Portion of Spontaneous Fission Neutron Source Rate Accounted For in the Spontaneous Fission Energy Spectrum: 100.0%.

Example Problem #2 (file: *tape7*)

```
                    Absolute Neutron Source Spectra
                    -------------------------------

Title: Example 2 - Clean UO2 Fuel (3% enriched) for Spectra

 Neutron Multigroup Structure (MeV)
1.000E+01 9.000E+00 8.000E+00 7.000E+00 6.000E+00 5.000E+00 4.000E+00 3.000E+00
2.000E+00 1.000E+00 0.000E+00

-------------------------------------------------------------------------------


 (a,n) neutrons/sec-cc from  6.77000E+20 at/cc  u235  alphas on  o 17  in target
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 6.740E-07 2.450E-06 6.312E-06
6.670E-06 5.358E-07
                                  Total (all groups):  1.664E-05 neutrons/sec-cm**3
.
                                Average Neutron Energy:  2.266E+00 MeV.


 (a,n) neutrons/sec-cc from  2.16000E+22 at/cc  u238  alphas on  o 17  in target
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 2.254E-06 1.076E-05 2.436E-05
3.158E-05 2.683E-06
                                  Total (all groups):  7.163E-05 neutrons/sec-cm**3
.
                                Average Neutron Energy:  2.196E+00 MeV.


 Total (alpha,n) neutron spectrum this target
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 2.928E-06 1.321E-05 3.067E-05
3.825E-05 3.219E-06
                                  Total (all groups):  8.827E-05 neutrons/sec-cm**3
.
                                Average Neutron Energy:  2.209E+00 MeV.

-------------------------------------------------------------------------------


 (a,n) neutrons/sec-cc from  6.77000E+20 at/cc  u235  alphas on  o 18  in target
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 7.825E-06 9.321E-05
6.171E-05 1.049E-05
                                  Total (all groups):  1.732E-04 neutrons/sec-cm**3
.
                                Average Neutron Energy:  2.066E+00 MeV.


 (a,n) neutrons/sec-cc from  2.16000E+22 at/cc  u238  alphas on  o 18  in target
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 7.453E-06 3.164E-04
2.837E-04 3.388E-05
                                  Total (all groups):  6.414E-04 neutrons/sec-cm**3
.
                                Average Neutron Energy:  1.971E+00 MeV.


 Total (alpha,n) neutron spectrum this target
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.528E-05 4.096E-04
```

3.454E-04 4.437E-05

Total (all groups): 8.146E-04 neutrons/sec-cm**3

Average Neutron Energy: 1.991E+00 MeV.

Grand total (alpha,n) neutron spectrum, all targets, all sources
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 2.928E-06 2.849E-05 4.402E-04
3.837E-04 4.758E-05

Total (all groups): 9.029E-04 neutrons/sec-cm**3

Average Neutron Energy: 2.012E+00 MeV.

S.F. neutrons/cc from 6.77000E+20 at/cc u235
5.043E-08 1.268E-07 3.116E-07 7.451E-07 1.722E-06 3.808E-06 7.930E-06 1.509E-05
2.454E-05 2.467E-05

Total (all groups): 7.900E-05 neutrons/sec-cm**3

Average Neutron Energy: 1.890E+00 MeV.

S.F. neutrons/cc from 2.16000E+22 at/cc u238
2.264E-05 6.835E-05 2.009E-04 5.716E-04 1.562E-03 4.049E-03 9.775E-03 2.122E-02
3.829E-02 4.055E-02

Total (all groups): 1.163E-01 neutrons/sec-cm**3

Average Neutron Energy: 1.688E+00 MeV.

Total S.F. neutron spectrum
2.269E-05 6.848E-05 2.012E-04 5.723E-04 1.563E-03 4.053E-03 9.783E-03 2.124E-02
3.832E-02 4.057E-02

Total (all groups): 1.164E-01 neutrons/sec-cm**3

Average Neutron Energy: 1.688E+00 MeV.

Total Neutron Spectrum
2.269E-05 6.848E-05 2.012E-04 5.723E-04 1.563E-03 4.056E-03 9.812E-03 2.168E-02
3.870E-02 4.062E-02

Total (all groups): 1.173E-01 neutrons/sec-cm**3

Average Neutron Energy: 1.691E+00 MeV.

Example Problem #2 (file: *tape9*)

```
                    Neutron Source Spectra by Nuclide Energy Level
                    ----------------------------------------------

Title: Example 2 - Clean UO2 Fuel (3% enriched) for Spectra

 Neutron Multigroup Structure (MeV)
 1.000E+01 9.000E+00 8.000E+00 7.000E+00 6.000E+00 5.000E+00 4.000E+00 3.000E+00
 2.000E+00 1.000E+00 0.000E+00

-------------------------------------------------------------------------------


 Neutron spectrum from  u235  alphas on  o 17  via the     .00-MeV product level
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 6.740E-07 2.428E-06 9.409E-07
 3.010E-08 0.000E+00
                                  Total (all groups):  4.073E-06 neutrons/sec-cm**3
.
                                Average Neutron Energy:  3.405E+00 MeV.


 Neutron spectrum from  u235  alphas on  o 17  via the    1.63-MeV product level
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 2.258E-08 5.371E-06
 6.640E-06 5.338E-07
                                  Total (all groups):  1.257E-05 neutrons/sec-cm**3
.
                                Average Neutron Energy:  1.897E+00 MeV.


 Neutron spectrum from  u235  alphas on  o 17  via the    4.25-MeV product level
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.000E+00 1.965E-09
                                  Total (all groups):  1.965E-09 neutrons/sec-cm**3
.
                                Average Neutron Energy:  7.859E-02 MeV.


 Neutron spectrum from  u238  alphas on  o 17  via the     .00-MeV product level
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 2.254E-06 1.076E-05 4.729E-06
 1.513E-07 0.000E+00
                                  Total (all groups):  1.790E-05 neutrons/sec-cm**3
.
                                Average Neutron Energy:  3.333E+00 MeV.


 Neutron spectrum from  u238  alphas on  o 17  via the    1.63-MeV product level
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.963E-05
 3.143E-05 2.683E-06
                                  Total (all groups):  5.374E-05 neutrons/sec-cm**3
.
                                Average Neutron Energy:  1.817E+00 MeV.


-------------------------------------------------------------------------------


 Neutron spectrum from  u235  alphas on  o 18  via the     .00-MeV product level
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 7.464E-06 5.746E-05
 2.497E-05 2.579E-07
```

Total (all groups): 9.015E-05 neutrons/sec-cm**3

Average Neutron Energy: 2.291E+00 MeV.

Neutron spectrum from u235 alphas on o 18 via the .35-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 3.602E-07 3.575E-05
3.364E-05 1.697E-06

Total (all groups): 7.145E-05 neutrons/sec-cm**3

Average Neutron Energy: 2.000E+00 MeV.

Neutron spectrum from u235 alphas on o 18 via the 1.75-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
3.098E-06 6.409E-06

Total (all groups): 9.507E-06 neutrons/sec-cm**3

Average Neutron Energy: 8.581E-01 MeV.

Neutron spectrum from u235 alphas on o 18 via the 2.79-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 1.598E-06

Total (all groups): 1.598E-06 neutrons/sec-cm**3

Average Neutron Energy: 1.285E-01 MeV.

Neutron spectrum from u235 alphas on o 18 via the 2.79-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 5.267E-07

Total (all groups): 5.267E-07 neutrons/sec-cm**3

Average Neutron Energy: 1.286E-01 MeV.

Neutron spectrum from u238 alphas on o 18 via the .00-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 7.453E-06 2.138E-04
1.255E-04 1.297E-06

Total (all groups): 3.481E-04 neutrons/sec-cm**3

Average Neutron Energy: 2.157E+00 MeV.

Neutron spectrum from u238 alphas on o 18 via the .35-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.025E-04
1.547E-04 8.531E-06

Total (all groups): 2.657E-04 neutrons/sec-cm**3

Average Neutron Energy: 1.858E+00 MeV.

Neutron spectrum from u238 alphas on o 18 via the 1.75-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
3.543E-06 2.405E-05

Total (all groups): 2.759E-05 neutrons/sec-cm**3

Average Neutron Energy: 6.986E-01 MeV.

Example Problem #4 (file: *outp*)

Summary of Input

Title: Example 4 - WPu-Be Interface Problem

Interface problem input (idd= 2)
Magnitudes and spectra computed.
Number of elemental constituents on source side: 2
Solid stopping cross sections used (isg= 0) on source side.
Maximum energy for alpha spectra: 6.500E+00 MeV.
Minimum energy for alpha spectra: 1.000E-07 MeV.

Elemental Constituents on Source Side:

Z-value	Atom Fraction
94	.9998000000
95	.0002000000

100 alpha energy groups used at interface.
Number of source nuclides to be evaluated: 6

Source Nulcides:

ZAID	Atom Fraction
942380	5.000E-04
942390	9.233E-01
942400	6.500E-02
942410	1.000E-02
942420	1.000E-03
952410	2.000E-04

Target title: target is composed of Be
Number of elemental constituents on target side: 1
Solid stopping cross sections used (isg= 0) on target side.

Elemental Constituents on Target Side:

Z-value	Atom Fraction
4	1.0000000000

Number of neutron spectrum energy groups: 20
Maximum neutron energy is 1.00000E+01 MeV.
Minimum neutron energy is 0.00000E+00 MeV.

Energy Group Structure:

Group	Upper-Bound	Lower-Bound
1	1.00000E+01	9.50000E+00
2	9.50000E+00	9.00000E+00
3	9.00000E+00	8.50000E+00
4	8.50000E+00	8.00000E+00
5	8.00000E+00	7.50000E+00
6	7.50000E+00	7.00000E+00
7	7.00000E+00	6.50000E+00
8	6.50000E+00	6.00000E+00
9	6.00000E+00	5.50000E+00

10	5.50000E+00	5.00000E+00
11	5.00000E+00	4.50000E+00
12	4.50000E+00	4.00000E+00
13	4.00000E+00	3.50000E+00
14	3.50000E+00	3.00000E+00
15	3.00000E+00	2.50000E+00
16	2.50000E+00	2.00000E+00
17	2.00000E+00	1.50000E+00
18	1.50000E+00	1.00000E+00
19	1.00000E+00	5.00000E-01
20	5.00000E-01	0.00000E+00

Number of target nuclides to be used: 1
4000 alpha energy groups used in target calculation.

Target Nuclides:

ZAID	Atom Fraction
40090	1.00000E+00

Summary of Output

Total (alpha,n) neutron source from all sources and targets: 2.35281E+02 n/sec-cm^2.

Average (alpha,n) neutron energy: 4.66321E+00 MeV.

Normalized Neutron Energy Spectrum by Energy Group for All Sources:

Group	Contribution
1	9.08200E-03
2	2.08155E-02
3	2.73146E-02
4	3.83261E-02
5	5.65617E-02
6	6.38810E-02
7	6.41606E-02
8	5.49906E-02
9	3.58767E-02
10	2.43732E-02
11	4.61727E-02
12	5.98003E-02
13	8.03417E-02
14	1.10000E-01
15	1.27823E-01
16	9.04486E-02
17	3.38045E-02
18	2.29970E-02
19	2.51460E-02
20	6.39882E-03

Portion of Total Neutron Source Rate Accounted for in the Energy Spectrum: 99.8%.

Example Problem #4 (file: *tape6*)

Neutron Source Magnitudes

Title: Example 4 - WPu-Be Interface Problem

1 Table I

(alpha,n) Neutron Production by Target and Alpha Energy

+ target	target atom frac.		alpha energy	alphas/sec /cm^2	p(e) neut/alpha	neuts/sec /cm^2
be 9	1.0000E+00		.033	3.5424E+08	0.0000E+00	0.0000E+00
			.098	2.7970E+05	0.0000E+00	0.0000E+00
			.163	2.0307E+05	0.0000E+00	0.0000E+00
			.228	1.6822E+05	0.0000E+00	0.0000E+00
			.293	1.4791E+05	0.0000E+00	0.0000E+00
			.358	1.3479E+05	0.0000E+00	0.0000E+00
			.423	1.2587E+05	0.0000E+00	0.0000E+00
			.488	1.1968E+05	0.0000E+00	0.0000E+00
			.553	1.1538E+05	0.0000E+00	0.0000E+00
			.618	1.1248E+05	0.0000E+00	0.0000E+00
			.683	1.1063E+05	0.0000E+00	0.0000E+00
			.748	1.0959E+05	0.0000E+00	0.0000E+00
			.813	1.0920E+05	0.0000E+00	0.0000E+00
			.878	1.0933E+05	0.0000E+00	0.0000E+00
			.943	1.0988E+05	0.0000E+00	0.0000E+00
			1.008	1.1077E+05	1.0799E-11	1.1962E-06
			1.072	1.1194E+05	1.0151E-09	1.1363E-04
			1.138	1.1334E+05	3.6938E-09	4.1866E-04
			1.203	1.1493E+05	8.1056E-09	9.3159E-04
			1.268	1.1667E+05	1.4311E-08	1.6697E-03
			1.332	1.1854E+05	2.2371E-08	2.6519E-03
			1.398	1.2050E+05	3.2350E-08	3.8981E-03
			1.463	1.2254E+05	4.4309E-08	5.4296E-03
			1.528	1.2464E+05	5.8975E-08	7.3507E-03
			1.592	1.2679E+05	8.8259E-08	1.1190E-02
			1.658	1.2897E+05	1.5772E-07	2.0341E-02
			1.723	1.3117E+05	3.0447E-07	3.9935E-02
			1.788	1.3338E+05	6.0884E-07	8.1207E-02
			1.852	1.3560E+05	1.2131E-06	1.6449E-01
			1.918	1.3782E+05	2.0683E-06	2.8504E-01
			1.983	1.4003E+05	2.9435E-06	4.1218E-01
			2.048	1.4223E+05	3.6538E-06	5.1969E-01
			2.112	1.4443E+05	4.2536E-06	6.1433E-01
			2.178	1.4660E+05	4.8022E-06	7.0401E-01
			2.243	1.4876E+05	5.3915E-06	8.0202E-01
			2.307	1.5089E+05	6.0026E-06	9.0576E-01
			2.372	1.5301E+05	6.5674E-06	1.0049E+00
			2.438	1.5510E+05	7.1336E-06	1.1064E+00
			2.503	1.5718E+05	7.7294E-06	1.2149E+00
			2.568	1.5922E+05	8.3951E-06	1.3367E+00
			2.632	1.6125E+05	9.1060E-06	1.4683E+00
			2.697	1.6325E+05	9.7897E-06	1.5981E+00
			2.763	1.6523E+05	1.0404E-05	1.7190E+00
			2.827	1.6718E+05	1.0961E-05	1.8325E+00

2.892	1.6911E+05	1.1489E-05	1.9430E+00
2.957	1.7102E+05	1.1973E-05	2.0476E+00
3.023	1.7290E+05	1.2403E-05	2.1445E+00
3.088	1.7477E+05	1.2793E-05	2.2358E+00
3.152	1.7661E+05	1.3188E-05	2.3291E+00
3.217	1.7843E+05	1.3574E-05	2.4221E+00
3.283	1.8024E+05	1.3945E-05	2.5133E+00
3.347	1.8202E+05	1.4323E-05	2.6070E+00
3.412	1.8378E+05	1.4736E-05	2.7082E+00
3.477	1.8553E+05	1.5182E-05	2.8167E+00
3.543	1.8725E+05	1.5686E-05	2.9373E+00
3.608	1.8896E+05	1.6294E-05	3.0789E+00
3.672	1.9065E+05	1.7011E-05	3.2433E+00
3.737	1.9233E+05	1.7787E-05	3.4210E+00
3.803	1.9399E+05	1.8665E-05	3.6209E+00
3.867	1.9564E+05	1.9665E-05	3.8472E+00
3.932	1.9727E+05	2.0859E-05	4.1148E+00
3.997	1.9888E+05	2.2691E-05	4.5128E+00
4.063	2.0049E+05	2.5053E-05	5.0228E+00
4.127	2.0207E+05	2.7233E-05	5.5032E+00
4.193	2.0365E+05	2.9308E-05	5.9686E+00
4.257	2.0521E+05	3.1336E-05	6.4307E+00
4.323	2.0677E+05	3.3411E-05	6.9083E+00
4.387	2.0831E+05	3.5508E-05	7.3965E+00
4.453	2.0984E+05	3.7638E-05	7.8978E+00
4.517	2.1135E+05	3.9816E-05	8.4152E+00
4.582	2.1286E+05	4.1969E-05	8.9335E+00
4.648	2.1436E+05	4.4051E-05	9.4427E+00
4.712	2.1584E+05	4.6085E-05	9.9470E+00
4.778	2.1731E+05	4.8202E-05	1.0475E+01
4.842	2.1877E+05	5.0455E-05	1.1038E+01
4.908	2.2018E+05	5.2914E-05	1.1650E+01
4.972	2.2159E+05	5.5568E-05	1.2313E+01
5.037	2.2297E+05	5.8314E-05	1.3002E+01
5.102	2.1388E+05	6.1134E-05	1.3075E+01
5.167	8.0625E+04	6.3941E-05	5.1552E+00
5.233	2.5755E+04	6.6902E-05	1.7231E+00
5.297	2.5914E+04	7.0015E-05	1.8144E+00
5.363	2.6053E+04	7.3170E-05	1.9063E+00
5.427	2.5732E+04	7.6386E-05	1.9656E+00
5.493	1.1151E+04	7.9681E-05	8.8854E-01
5.557	2.0740E+00	8.3278E-05	1.7272E-04
		Total:	2.3528E+02

Example Problem #4 (file: *tape7*)

```
                    Absolute Neutron Source Spectra
                    -------------------------------

Title: Example 4 - WPu-Be Interface Problem


 Neutron Multigroup Structure (MeV)
 1.000E+01 9.500E+00 9.000E+00 8.500E+00 8.000E+00 7.500E+00 7.000E+00 6.500E+00
 6.000E+00 5.500E+00 5.000E+00 4.500E+00 4.000E+00 3.500E+00 3.000E+00 2.500E+00
 2.000E+00 1.500E+00 1.000E+00 5.000E-01 0.000E+00

 (a,n) neutrons from alphas on be  9  in target.
 2.137E+00 4.897E+00 6.427E+00 9.017E+00 1.331E+01 1.503E+01 1.510E+01 1.294E+01
 8.441E+00 5.735E+00 1.086E+01 1.407E+01 1.890E+01 2.588E+01 3.007E+01 2.128E+01
 7.954E+00 5.411E+00 5.916E+00 1.506E+00
                            Total (all groups):  2.349E+02 neutrons/sec-cm^2.
                        Average Neutron Energy:  4.663E+00 MeV.

-------------------------------------------------------------------------------



 Total Neutron Spectrum
 2.137E+00 4.897E+00 6.427E+00 9.017E+00 1.331E+01 1.503E+01 1.510E+01 1.294E+01
 8.441E+00 5.735E+00 1.086E+01 1.407E+01 1.890E+01 2.588E+01 3.007E+01 2.128E+01
 7.954E+00 5.411E+00 5.916E+00 1.506E+00
                            Total (all groups):  2.349E+02 neutrons/sec-cm^2.
                        Average Neutron Energy:  4.663E+00 MeV.
```

Example Problem #6 (file: *outp*)

Summary of Input

Title: Example 6 - Alpha Beam (5.5 MeV) on SiO2
Beam problem input (idd= 3)
Magnitudes and spectra computed.
Number of elemental constituents: 2
Solid stopping cross sections used (isg= 0)

Elemental Constituents:

Z-value	Atom Fraction
8	.6666670000
14	.3333330000

Number of neutron spectrum energy groups: 22
Maximum neutron energy is 1.000E+01 MeV.
Minimum neutron energy is 0.000E+00 MeV.

Energy Group Structure:

Group	Upper-Bound	Lower-Bound
1	1.000E+01	7.000E+00
2	7.000E+00	6.000E+00
3	6.000E+00	5.500E+00
4	5.500E+00	5.000E+00
5	5.000E+00	4.500E+00
6	4.500E+00	4.000E+00
7	4.000E+00	3.500E+00
8	3.500E+00	3.250E+00
9	3.250E+00	3.000E+00
10	3.000E+00	2.750E+00
11	2.750E+00	2.500E+00
12	2.500E+00	2.250E+00
13	2.250E+00	2.000E+00
14	2.000E+00	1.750E+00
15	1.750E+00	1.500E+00
16	1.500E+00	1.250E+00
17	1.250E+00	1.000E+00
18	1.000E+00	7.500E-01
19	7.500E-01	5.000E-01
20	5.000E-01	2.500E-01
21	2.500E-01	1.000E-01
22	1.000E-01	0.000E+00

Alpha beam energy is 5.500E+00 Mev.

Number of target nuclides to be used: 4
4000 Alpha energy groups used.

Target Nuclides:

ZAID	Atom Fraction
80170	2.530E-04
80180	1.333E-03

140290	1.557E-02
140300	1.033E-02

Summary of Output

Total (alpha,n) neutron source from all sources and targets: 3.261E+05 n/sec-microamp.

Average (alpha,n) neutron energy: 1.965E+00 MeV.

Normalized Neutron Energy Spectrum by Energy Group for All Sources:

Group	Contribution
1	0.000E+00
2	0.000E+00
3	9.134E-05
4	4.811E-04
5	8.194E-04
6	7.903E-03
7	3.521E-02
8	3.882E-02
9	5.565E-02
10	7.192E-02
11	7.803E-02
12	7.718E-02
13	6.985E-02
14	6.158E-02
15	4.588E-02
16	4.422E-02
17	7.885E-02
18	1.131E-01
19	1.163E-01
20	7.256E-02
21	2.385E-02
22	7.673E-03

Portion of Total Neutron Source Rate Accounted for in the Total Energy Spectrum: 100.0%.

Example Problem #6 (file: *tape6*)

Neutron Source Magnitudes

Title: Example 6 - Alpha Beam (5.5 MeV) on SiO2

1 Table I

(alpha,n) Neutron Production by Target Per Source Alpha

target	target atom frac.	alpha source	alpha energy	alphas/sec /microamp	p(e) neut/alpha	neuts/sec /microamp
o 17	2.5300E-04	beam	5.500	3.1209E+12	3.2813E-09	1.0240E+04
o 18	1.3330E-03	beam	5.500	3.1209E+12	3.7792E-08	1.1795E+05
si 29	1.5567E-02	beam	5.500	3.1209E+12	4.2853E-08	1.3374E+05
si 30	1.0333E-02	beam	5.500	3.1209E+12	2.0548E-08	6.4127E+04
				Total (all targets):		3.2605E+05

Example Problem #6 (file: *tape7*)

```
                    Absolute Neutron Source Spectra
                    ~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~

Title: Example 6 - Alpha Beam (5.5 MeV) on SiO2

 Neutron Multigroup Structure (MeV)
 1.000E+01 7.000E+00 6.000E+00 5.500E+00 5.000E+00 4.500E+00 4.000E+00 3.500E+00
 3.250E+00 3.000E+00 2.750E+00 2.500E+00 2.250E+00 2.000E+00 1.750E+00 1.500E+00
 1.250E+00 1.000E+00 7.500E-01 5.000E-01 2.500E-01 1.000E-01 0.000E+00

------------------------------------------------------------------------------


 (a,n) neutrons/sec-microamp  5.500 mev a on  o 17  in target
 0.000E+00 0.000E+00 2.978E+01 1.569E+02 2.672E+02 4.388E+02 9.911E+02 6.879E+02
 7.900E+02 8.636E+02 1.007E+03 1.039E+03 8.887E+02 7.442E+02 6.080E+02 3.470E+02
 2.388E+02 2.687E+02 3.603E+02 3.538E+02 1.201E+02 3.822E+01
                                  Total (all groups):  1.024E+04 neutrons/sec-
microamp.
                               Average Neutron Energy:  2.522E+00 MeV.


 (a,n) neutrons/sec-microamp  5.500 mev a on  o 18  in target
 0.000E+00 0.000E+00 2.908E-03 1.532E-02 2.609E-02 2.138E+03 9.316E+03 8.074E+03
 1.056E+04 1.293E+04 1.372E+04 1.337E+04 1.171E+04 9.322E+03 6.951E+03 5.202E+03
 3.950E+03 3.247E+03 2.911E+03 2.516E+03 1.438E+03 5.850E+02
                                  Total (all groups):  1.179E+05 neutrons/sec-
microamp.
                               Average Neutron Energy:  2.385E+00 MeV.


 (a,n) neutrons/sec-microamp  5.500 mev a on si 29  in target
 0.000E+00 0.000E+00 2.466E-08 1.299E-07 2.212E-07 1.813E-02 1.173E+03 3.895E+03
 6.793E+03 9.658E+03 1.071E+04 1.076E+04 1.018E+04 1.001E+04 5.933E+03 2.317E+03
 8.670E+03 1.789E+04 1.980E+04 1.251E+04 2.912E+03 5.313E+02
                                  Total (all groups):  1.337E+05 neutrons/sec-
microamp.
                               Average Neutron Energy:  1.993E+00 MeV.


 (a,n) neutrons/sec-microamp  5.500 mev a on si 30  in target
 0.000E+00 0.000E+00 1.844E-13 9.710E-13 1.654E-12 1.355E-07 8.770E-03 2.912E-02
 5.079E-02 7.221E-02 8.010E-02 8.046E-02 7.612E-02 7.487E-02 1.469E+03 6.552E+03
 1.285E+04 1.548E+04 1.485E+04 8.277E+03 3.308E+03 1.347E+03
                                  Total (all groups):  6.413E+04 neutrons/sec-
microamp.
                               Average Neutron Energy:  1.761E+00 MeV.

------------------------------------------------------------------------------


 Grand total (alpha,n) neutron spectrum, all targets, all sources
 0.000E+00 0.000E+00 2.978E+01 1.569E+02 2.672E+02 2.577E+03 1.148E+04 1.266E+04
 1.815E+04 2.345E+04 2.544E+04 2.516E+04 2.278E+04 2.008E+04 1.496E+04 1.442E+04
 2.571E+04 3.688E+04 3.792E+04 2.366E+04 7.777E+03 2.502E+03
                                  Total (all groups):  3.260E+05 neutrons/sec-
microamp.
                               Average Neutron Energy:  1.965E+00 MeV.


------------------------------------------------------------------------------
```

Example Problem #6 (file: *tape9*)

```
                    Neutron Source Spectra by Nuclide Energy Level
                    ----------------------------------------------

Title: Example 6 - Alpha Beam (5.5 MeV) on SiO2

 Neutron Multigroup Structure (MeV)
 1.000E+01 7.000E+00 6.000E+00 5.500E+00 5.000E+00 4.500E+00 4.000E+00 3.500E+00
 3.250E+00 3.000E+00 2.750E+00 2.500E+00 2.250E+00 2.000E+00 1.750E+00 1.500E+00
 1.250E+00 1.000E+00 7.500E-01 5.000E-01 2.500E-01 1.000E-01 0.000E+00

 ------------------------------------------------------------------------------


 Neutron spectrum from  5.50 MeV alphas on  o 17  via the    .00-MeV product level
 0.000E+00 0.000E+00 2.978E+01 1.569E+02 2.672E+02 4.260E+02 4.990E+02 2.171E+02
 1.847E+02 1.265E+02 7.294E+01 4.086E+01 1.717E+01 7.291E+00 7.421E-01 6.503E-02
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
                                  Total (all groups):  2.046E+03 neutrons/sec-
microamp.
                               Average Neutron Energy:  3.884E+00 MeV.


 Neutron spectrum from  5.50 MeV alphas on  o 17  via the   1.63-MeV product level
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.283E+01 4.921E+02 4.707E+02
 6.053E+02 7.372E+02 9.338E+02 9.983E+02 8.716E+02 7.369E+02 6.073E+02 3.469E+02
 1.964E+02 8.404E+01 4.867E+01 1.092E+01 6.001E-01 8.262E-02
                                  Total (all groups):  7.154E+03 neutrons/sec-
microamp.
                               Average Neutron Energy:  2.422E+00 MeV.


 Neutron spectrum from  5.50 MeV alphas on  o 17  via the   4.25-MeV product level
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 4.240E+01 1.846E+02 3.116E+02 3.429E+02 1.133E+02 3.016E+01
                                  Total (all groups):  1.025E+03 neutrons/sec-
microamp.
                               Average Neutron Energy:  5.362E-01 MeV.


 Neutron spectrum from  5.50 MeV alphas on  o 17  via the   4.97-MeV product level
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 6.171E+00 7.983E+00
                                  Total (all groups):  1.415E+01 neutrons/sec-
microamp.
                               Average Neutron Energy:  9.355E-02 MeV.


 Neutron spectrum from  5.50 MeV alphas on  o 18  via the    .00-MeV product level
 0.000E+00 0.000E+00 2.978E+01 1.569E+02 2.672E+02 2.437E+03 6.435E+03 5.097E+03
 6.280E+03 7.238E+03 7.053E+03 6.253E+03 4.737E+03 3.345E+03 1.935E+03 1.088E+03
 4.876E+02 5.727E+01 9.773E+00 2.367E+00 6.605E-02 0.000E+00
                                  Total (all groups):  5.086E+04 neutrons/sec-
microamp.
                               Average Neutron Energy:  2.817E+00 MeV.


 Neutron spectrum from  5.50 MeV alphas on  o 18  via the    .35-MeV product level
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.400E+02 3.872E+03 3.665E+03
 5.074E+03 6.555E+03 7.610E+03 7.493E+03 6.637E+03 5.058E+03 3.470E+03 1.903E+03
 1.041E+03 4.832E+02 1.042E+02 1.578E+01 1.292E+00 1.166E-01
```

Total (all groups): 4.597E+04 neutrons/sec-microamp.

Average Neutron Energy: 2.509E+00 MeV.

Neutron spectrum from 5.50 MeV alphas on o 18 via the 1.75-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 6.746E+01 6.589E+02 1.222E+03 1.663E+03 2.154E+03 2.369E+03
1.752E+03 1.410E+03 9.684E+02 6.335E+02 1.586E+02 3.654E+01
Total (all groups): 1.207E+04 neutrons/sec-microamp.

Average Neutron Energy: 1.397E+00 MeV.

Neutron spectrum from 5.50 MeV alphas on o 18 via the 2.79-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.232E+02
5.985E+02 1.046E+03 1.491E+03 1.559E+03 9.184E+02 3.482E+02
Total (all groups): 6.071E+03 neutrons/sec-microamp.

Average Neutron Energy: 5.627E-01 MeV.

Neutron spectrum from 5.50 MeV alphas on o 18 via the 2.79-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 6.528E+01
3.094E+02 5.195E+02 6.980E+02 6.325E+02 3.292E+02 1.162E+02
Total (all groups): 2.670E+03 neutrons/sec-microamp.

Average Neutron Energy: 6.061E-01 MeV.

Neutron spectrum from 5.50 MeV alphas on o 18 via the 3.66-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 0.000E+00 2.647E+01 1.501E+02 1.221E+02
Total (all groups): 2.987E+02 neutrons/sec-microamp.

Average Neutron Energy: 1.303E-01 MeV.

Neutron spectrum from 5.50 MeV alphas on si 29 via the .00-MeV product level
0.000E+00 0.000E+00 2.978E+01 1.569E+02 2.672E+02 2.437E+03 7.608E+03 8.992E+03
1.307E+04 1.690E+04 1.777E+04 1.701E+04 1.492E+04 1.336E+04 7.867E+03 2.668E+03
7.294E+02 1.509E+02 5.791E+01 2.923E+01 7.119E+00 1.266E+00
Total (all groups): 7.111E+04 neutrons/sec-microamp.

Average Neutron Energy: 2.594E+00 MeV.

Neutron spectrum from 5.50 MeV alphas on si 29 via the 2.24-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.400E+02 3.872E+03 3.665E+03
5.074E+03 6.555E+03 7.610E+03 7.493E+03 6.637E+03 5.058E+03 3.470E+03 2.641E+03
9.469E+03 1.828E+04 1.985E+04 1.250E+04 2.906E+03 5.302E+02
Total (all groups): 6.263E+04 neutrons/sec-microamp.

Average Neutron Energy: 1.527E+00 MeV.

Neutron spectrum from 5.50 MeV alphas on si 29 via the 3.78-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 6.746E+01 6.589E+02 1.222E+03 1.663E+03 2.154E+03 2.369E+03
1.752E+03 1.410E+03 9.684E+02 6.335E+02 1.586E+02 3.654E+01
Total (all groups): 0.000E+00 neutrons/sec-microamp.

Average Neutron Energy: 1.397E+00 MeV.

Neutron spectrum from 5.50 MeV alphas on si 30 via the .00-MeV product level
0.000E+00 0.000E+00 2.978E+01 1.569E+02 2.672E+02 2.437E+03 7.608E+03 8.992E+03
1.307E+04 1.690E+04 1.777E+04 1.701E+04 1.492E+04 1.336E+04 9.336E+03 9.220E+03
1.358E+04 1.563E+04 1.413E+04 6.336E+03 2.426E+03 1.108E+03
Total (all groups): 6.025E+04 neutrons/sec-microamp.
Average Neutron Energy: 2.022E+00 MeV.

Neutron spectrum from 5.50 MeV alphas on si 30 via the .84-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.400E+02 3.872E+03 3.665E+03
5.074E+03 6.555E+03 7.610E+03 7.493E+03 6.637E+03 5.058E+03 3.470E+03 2.641E+03
9.469E+03 1.828E+04 2.063E+04 1.447E+04 3.795E+03 7.707E+02
Total (all groups): 3.874E+03 neutrons/sec-microamp.
Average Neutron Energy: 1.489E+00 MeV.

Neutron spectrum from 5.50 MeV alphas on si 30 via the 1.97-MeV product level
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 6.746E+01 6.589E+02 1.222E+03 1.663E+03 2.154E+03 2.369E+03
1.752E+03 1.410E+03 9.684E+02 6.335E+02 1.586E+02 3.654E+01
Total (all groups): 0.000E+00 neutrons/sec-microamp.
Average Neutron Energy: 1.397E+00 MeV.

Example Problem #7 (file: *outp*)

Summary of Input

Title: Three Region Test Problem #1 (WPu-Al-Be)

Three region problem input (idd= 4)
Magnitudes and spectra computed.

400 alpha energy groups used at each interface.
Maximum energy for alpha spectra: 6.500E+00 MeV.
Minimum energy for alpha spectra: 1.000E-07 MeV.

Number of neutron spectrum energy groups: 20
Maximum neutron energy is 1.00000E+01 MeV.
Minimum neutron energy is 0.00000E+00 MeV.

Neutron Energy Group Structure:

Group	Upper-Bound	Lower-Bound
1	1.00000E+01	9.50000E+00
2	9.50000E+00	9.00000E+00
3	9.00000E+00	8.50000E+00
4	8.50000E+00	8.00000E+00
5	8.00000E+00	7.50000E+00
6	7.50000E+00	7.00000E+00
7	7.00000E+00	6.50000E+00
8	6.50000E+00	6.00000E+00
9	6.00000E+00	5.50000E+00
10	5.50000E+00	5.00000E+00
11	5.00000E+00	4.50000E+00
12	4.50000E+00	4.00000E+00
13	4.00000E+00	3.50000E+00
14	3.50000E+00	3.00000E+00
15	3.00000E+00	2.50000E+00
16	2.50000E+00	2.00000E+00
17	2.00000E+00	1.50000E+00
18	1.50000E+00	1.00000E+00
19	1.00000E+00	5.00000E-01
20	5.00000E-01	0.00000E+00

Number of angular groups: 40

Angular Group Structure:

Group	Upper-Bound	Lower-Bound
1	3.92699E-02	0.00000E+00
2	7.85398E-02	3.92699E-02
3	1.17810E-01	7.85398E-02
4	1.57080E-01	1.17810E-01
5	1.96350E-01	1.57080E-01
6	2.35619E-01	1.96350E-01
7	2.74889E-01	2.35619E-01
8	3.14159E-01	2.74889E-01
9	3.53429E-01	3.14159E-01
10	3.92699E-01	3.53429E-01
11	4.31969E-01	3.92699E-01
12	4.71239E-01	4.31969E-01
13	5.10509E-01	4.71239E-01
14	5.49779E-01	5.10509E-01
15	5.89049E-01	5.49779E-01
16	6.28319E-01	5.89049E-01

17	6.67589E-01	6.28319E-01
18	7.06859E-01	6.67589E-01
19	7.46128E-01	7.06859E-01
20	7.85398E-01	7.46128E-01
21	8.24668E-01	7.85398E-01
22	8.63938E-01	8.24668E-01
23	9.03208E-01	8.63938E-01
24	9.42478E-01	9.03208E-01
25	9.81748E-01	9.42478E-01
26	1.02102E+00	9.81748E-01
27	1.06029E+00	1.02102E+00
28	1.09956E+00	1.06029E+00
29	1.13883E+00	1.09956E+00
30	1.17810E+00	1.13883E+00
31	1.21737E+00	1.17810E+00
32	1.25664E+00	1.21737E+00
33	1.29591E+00	1.25664E+00
34	1.33518E+00	1.29591E+00
35	1.37445E+00	1.33518E+00
36	1.41372E+00	1.37445E+00
37	1.45299E+00	1.41372E+00
38	1.49226E+00	1.45299E+00
39	1.53153E+00	1.49226E+00
40	1.57080E+00	1.53153E+00

Region A Title: WPu region

Number of elemental constituents in region A: 2
Solid stopping cross-sections used (isga= 0) in region A.
Elemental Constituents in Region A:

Z-value	Atom Fraction
94	0.9998000264
95	0.0002000000

Number of source nuclides to be evaluated: 6

Source Nuclides in Region A:

ZAID	Atom Fraction
942380	5.000E-04
942390	9.233E-01
942400	6.500E-02
942410	1.000E-02
942420	1.000E-03
952410	2.000E-04

Region B Title: Al interface

Number of elemental constituents in region B: 1
Solid stopping cross-sections used (isgb= 0) in region B.
Material B atom density: 1.50000E-01 atoms/cc.
Interface region thickness: 1.00000E-01 cm.

Elemental Constituents in Region B:

Z-value	Atom Fraction
13	1.0000000000

Number of target nuclides in region B: 1

Target Nuclides in Region B:

ZAID	Atom Fraction
130270	1.000E+00

Region C Title: Be reflector

Number of elemental constituents in region C: 1
Solid stopping cross-sections used (isgc= 0) in region C.

Elemental Constituents in Region C:

Z-value	Atom Fraction
4	1.0000000000

Number of target nuclides in region C: 1

Target Nuclides in Region C:

ZAID	Atom Fraction
40090	1.000E+00

Summary of Output

Total (alpha,n) neutron source from all sources and targets: 1.21575E+02 n/sec-cm^2.

Average (alpha,n) neutron energy: 4.54836E+00 MeV.

Normalized Neutron Energy Spectrum by Energy Group for All Sources:

Group	Contribution
1	4.47585E-03
2	1.42866E-02
3	2.12089E-02
4	3.56358E-02
5	6.08200E-02
6	7.12515E-02
7	7.17967E-02
8	6.62933E-02
9	4.12643E-02
10	8.90019E-03
11	2.58705E-02
12	4.10487E-02
13	6.74781E-02
14	1.08942E-01
15	1.58448E-01
16	1.24117E-01
17	3.98306E-02
18	1.46272E-02
19	1.75079E-02
20	6.19659E-03

Example Problem #7 (file: *tape7*)

```
                    Absolute Neutron Source Spectra
                    -------------------------------

Title: Three Region Test Problem #1 (WPu-Al-Be)


 Neutron Multigroup Structure (MeV)
 1.000E+01 9.500E+00 9.000E+00 8.500E+00 8.000E+00 7.500E+00 7.000E+00 6.500E+00
 6.000E+00 5.500E+00 5.000E+00 4.500E+00 4.000E+00 3.500E+00 3.000E+00 2.500E+00
 2.000E+00 1.500E+00 1.000E+00 5.000E-01 0.000E+00

================================================================================

Title: Alphas at interface ab using region B materials for neutron production

 Neutron Spectrum (neuts/cm^2-sec)
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.458E-06 4.946E-03
 7.175E-02 1.856E-01 2.212E-01 1.113E-01
                              Total (all groups):  5.949E-01 neutrons/sec-cm^2.
                          Average Neutron Energy:  9.430E-01 MeV.

--------------------------------------------------------------------------------


 Total Neutron Spectrum
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.458E-06 4.946E-03
 7.175E-02 1.856E-01 2.212E-01 1.113E-01
                              Total (all groups):  5.949E-01 neutrons/sec-cm^2.
                          Average Neutron Energy:  9.430E-01 MeV.

================================================================================

Title: Alphas at interface bc using region B materials for neutron production

 Neutron Spectrum (neuts/cm^2-sec)
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.784E-05
 3.618E-03 3.111E-02 5.597E-02 2.727E-02
                              Total (all groups):  1.180E-01 neutrons/sec-cm^2.
                          Average Neutron Energy:  7.960E-01 MeV.

--------------------------------------------------------------------------------


 Total Neutron Spectrum
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.784E-05
 3.618E-03 3.111E-02 5.597E-02 2.727E-02
                              Total (all groups):  1.180E-01 neutrons/sec-cm^2.
                          Average Neutron Energy:  9.186E-01 MeV.
```

Title: Alphas at interface bc using region C materials for neutron production

Neutron Spectrum (neuts/cm^2-sec)
5.442E-01 1.737E+00 2.578E+00 4.332E+00 7.394E+00 8.662E+00 8.729E+00 8.060E+00
5.017E+00 1.082E+00 3.145E+00 4.991E+00 8.204E+00 1.324E+01 1.926E+01 1.508E+01
4.774E+00 1.624E+00 1.963E+00 6.693E-01
Total (all groups): 1.211E+02 neutrons/sec-cm^2.
Average Neutron Energy: 4.565E+00 MeV.

Total Neutron Spectrum
5.442E-01 1.737E+00 2.578E+00 4.332E+00 7.394E+00 8.662E+00 8.729E+00 8.060E+00
5.017E+00 1.082E+00 3.145E+00 4.991E+00 8.204E+00 1.324E+01 1.926E+01 1.508E+01
4.774E+00 1.624E+00 1.963E+00 6.693E-01
Total (all groups): 1.211E+02 neutrons/sec-cm^2.
Average Neutron Energy: 4.544E+00 MeV.

Title: Total neutron production from all interfaces

Total Neutron Spectrum
5.442E-01 1.737E+00 2.578E+00 4.332E+00 7.394E+00 8.662E+00 8.729E+00 8.060E+00
5.017E+00 1.082E+00 3.145E+00 4.991E+00 8.204E+00 1.324E+01 1.926E+01 1.509E+01
4.842E+00 1.778E+00 2.129E+00 7.534E-01
Total (all groups): 1.216E+02 neutrons/sec-cm^2.
Average Neutron Energy: 4.548E+00 MeV.

Example Problem #8 (file: *outp*)

Summary of Input

Title: Example 8 - Am-CO2-AlB2 Interface Problem

Three region problem input (idd= 4)
Magnitudes and spectra computed.

400 alpha energy groups used at each interface.
Maximum energy for alpha spectra: 6.500E+00 MeV.
Minimum energy for alpha spectra: 1.000E-07 MeV.

Number of neutron spectrum energy groups: 20
Maximum neutron energy is 1.00000E+01 MeV.
Minimum neutron energy is 0.00000E+00 MeV.

Neutron Energy Group Structure:

Group	Upper-Bound	Lower-Bound
1	1.00000E+01	9.50000E+00
2	9.50000E+00	9.00000E+00
3	9.00000E+00	8.50000E+00
4	8.50000E+00	8.00000E+00
5	8.00000E+00	7.50000E+00
6	7.50000E+00	7.00000E+00
7	7.00000E+00	6.50000E+00
8	6.50000E+00	6.00000E+00
9	6.00000E+00	5.50000E+00
10	5.50000E+00	5.00000E+00
11	5.00000E+00	4.50000E+00
12	4.50000E+00	4.00000E+00
13	4.00000E+00	3.50000E+00
14	3.50000E+00	3.00000E+00
15	3.00000E+00	2.50000E+00
16	2.50000E+00	2.00000E+00
17	2.00000E+00	1.50000E+00
18	1.50000E+00	1.00000E+00
19	1.00000E+00	5.00000E-01
20	5.00000E-01	0.00000E+00

Number of angular groups: 60

Angular Group Structure:

Group	Upper-Bound	Lower-Bound
1	2.61799E-02	0.00000E+00
2	5.23599E-02	2.61799E-02
3	7.85398E-02	5.23599E-02
4	1.04720E-01	7.85398E-02
5	1.30900E-01	1.04720E-01
6	1.57080E-01	1.30900E-01
7	1.83260E-01	1.57080E-01
8	2.09440E-01	1.83260E-01
9	2.35619E-01	2.09440E-01
10	2.61799E-01	2.35619E-01
11	2.87979E-01	2.61799E-01
12	3.14159E-01	2.87979E-01
13	3.40339E-01	3.14159E-01
14	3.66519E-01	3.40339E-01
15	3.92699E-01	3.66519E-01

16	4.18879E-01	3.92699E-01
17	4.45059E-01	4.18879E-01
18	4.71239E-01	4.45059E-01
19	4.97419E-01	4.71239E-01
20	5.23599E-01	4.97419E-01
21	5.49779E-01	5.23599E-01
22	5.75959E-01	5.49779E-01
23	6.02139E-01	5.75959E-01
24	6.28318E-01	6.02139E-01
25	6.54498E-01	6.28318E-01
26	6.80678E-01	6.54498E-01
27	7.06858E-01	6.80678E-01
28	7.33038E-01	7.06858E-01
29	7.59218E-01	7.33038E-01
30	7.85398E-01	7.59218E-01
31	8.11578E-01	7.85398E-01
32	8.37758E-01	8.11578E-01
33	8.63938E-01	8.37758E-01
34	8.90118E-01	8.63938E-01
35	9.16297E-01	8.90118E-01
36	9.42477E-01	9.16297E-01
37	9.68657E-01	9.42477E-01
38	9.94837E-01	9.68657E-01
39	1.02102E+00	9.94837E-01
40	1.04720E+00	1.02102E+00
41	1.07338E+00	1.04720E+00
42	1.09956E+00	1.07338E+00
43	1.12574E+00	1.09956E+00
44	1.15192E+00	1.12574E+00
45	1.17810E+00	1.15192E+00
46	1.20428E+00	1.17810E+00
47	1.23046E+00	1.20428E+00
48	1.25664E+00	1.23046E+00
49	1.28282E+00	1.25664E+00
50	1.30900E+00	1.28282E+00
51	1.33518E+00	1.30900E+00
52	1.36136E+00	1.33518E+00
53	1.38754E+00	1.36136E+00
54	1.41372E+00	1.38754E+00
55	1.43990E+00	1.41372E+00
56	1.46608E+00	1.43990E+00
57	1.49226E+00	1.46608E+00
58	1.51844E+00	1.49226E+00
59	1.54462E+00	1.51844E+00
60	1.57080E+00	1.54462E+00

Region A Title: Pure Am-241 in region A

Number of elemental constituents in region A: 1
Solid stopping cross-sections used (isga= 0) in region A.
Elemental Constituents in Region A:

Z-value	Atom Fraction
95	1.0000000000

Number of source nuclides to be evaluated: 1

Source Nuclides in Region A:

ZAID	Atom Fraction
952410	1.000E+00

Region B Title: CO2 gas in region B

Number of elemental constituents in region B: 2
Gas stopping cross-sections used (isgb= 1) in region B.
Material B atom density: 4.00000E-03 atoms/cc.
Interface region thickness: 3.00000E+00 cm.

Elemental Constituents in Region B:

Z-value	Atom Fraction
6	0.3330000043
8	0.6669999957

Number of target nuclides in region B: 3

Target Nuclides in Region B:

ZAID	Atom Fraction
60130	7.333E-03
80170	2.667E-04
80180	1.333E-03

Region C Title: AlB2 shield in region C

Number of elemental constituents in region C: 2
Solid stopping cross-sections used (isgc= 0) in region C.

Elemental Constituents in Region C:

Z-value	Atom Fraction
13	0.3333300054
5	0.6666700244

Number of target nuclides in region C: 3

Target Nuclides in Region C:

ZAID	Atom Fraction
50100	1.327E-01
50110	5.340E-01
130270	3.333E-01

Summary of Output

Total (alpha,n) neutron source from all sources and targets: 1.11347E+03 n/sec-cm^2.

Average (alpha,n) neutron energy: 2.56872E+00 MeV.

Normalized Neutron Energy Spectrum by Energy Group for All Sources:

Group	Contribution
1	0.00000E+00
2	0.00000E+00
3	0.00000E+00
4	0.00000E+00
5	6.81977E-12
6	3.68090E-05
7	1.42735E-04

8	1.90162E-04
9	2.92516E-04
10	1.01406E-03
11	6.59448E-03
12	3.79637E-02
13	1.00384E-01
14	1.72817E-01
15	2.26264E-01
16	2.22286E-01
17	1.19382E-01
18	4.91691E-02
19	3.98285E-02
20	2.36355E-02

Example Problem #8 (file: *tape7*)

Absolute Neutron Source Spectra

Title: Example 8 - Am-CO2-AlB2 Interface Problem

Neutron Multigroup Structure (MeV)
1.000E+01 9.500E+00 9.000E+00 8.500E+00 8.000E+00 7.500E+00 7.000E+00 6.500E+00
6.000E+00 5.500E+00 5.000E+00 4.500E+00 4.000E+00 3.500E+00 3.000E+00 2.500E+00
2.000E+00 1.500E+00 1.000E+00 5.000E-01 0.000E+00

Title: Alphas at interface ab using region B materials for neutron production

Neutron Spectrum (neuts/cm^2-sec)
0.000E+00 0.000E+00 0.000E+00 0.000E+00 7.594E-09 4.475E-02 2.159E-01 3.568E-01
5.042E-01 1.153E+00 1.757E+00 2.490E+00 2.402E+00 1.261E+00 8.374E-02 4.120E-03
0.000E+00 0.000E+00 2.012E-03 9.204E-02
Total (all groups): 1.037E+01 neutrons/sec-cm^2.
Average Neutron Energy: 4.366E+00 MeV.

Neutron Spectrum (neuts/cm^2-sec)
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
1.829E-04 2.754E-03 8.265E-03 2.107E-02 3.972E-02 6.444E-02 8.997E-02 1.321E-01
1.287E-01 6.632E-02 2.634E-02 1.549E-02
Total (all groups): 5.954E-01 neutrons/sec-cm^2.
Average Neutron Energy: 2.314E+00 MeV.

Neutron Spectrum (neuts/cm^2-sec)
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 1.748E-08 1.772E-02 1.667E-01 6.083E-01 1.290E+00 1.689E+00
1.295E+00 6.646E-01 2.801E-01 2.183E-01
Total (all groups): 6.230E+00 neutrons/sec-cm^2.
Average Neutron Energy: 2.148E+00 MeV.

Total Neutron Spectrum
0.000E+00 0.000E+00 0.000E+00 0.000E+00 7.594E-09 4.475E-02 2.159E-01 3.568E-01
5.044E-01 1.156E+00 1.766E+00 2.528E+00 2.608E+00 1.934E+00 1.463E+00 1.825E+00
1.424E+00 7.310E-01 3.084E-01 3.258E-01
Total (all groups): 1.719E+01 neutrons/sec-cm^2.
Average Neutron Energy: 3.491E+00 MeV.

Title: Alphas at interface bc using region B materials for neutron production

Neutron Spectrum (neuts/cm^2-sec)
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 3.763E-03 5.700E-02 1.450E-01
2.563E-01 7.799E-01 1.302E+00 2.053E+00 2.023E+00 1.084E+00 7.342E-02 3.688E-03
0.000E+00 0.000E+00 7.850E-08 4.935E-03
Total (all groups): 7.786E+00 neutrons/sec-cm^2.
Average Neutron Energy: 4.254E+00 MeV.

Neutron Spectrum (neuts/cm^2-sec)
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
3.188E-06 7.044E-04 3.915E-03 1.322E-02 2.449E-02 3.945E-02 5.654E-02 9.193E-02
9.888E-02 5.419E-02 1.809E-02 7.014E-03
Total (all groups): 4.084E-01 neutrons/sec-cm^2.
Average Neutron Energy: 2.242E+00 MeV.

Neutron Spectrum (neuts/cm^2-sec)
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 0.000E+00 1.406E-03 5.245E-02 3.122E-01 8.077E-01 1.198E+00
9.826E-01 5.012E-01 1.802E-01 1.337E-01
Total (all groups): 4.170E+00 neutrons/sec-cm^2.
Average Neutron Energy: 2.074E+00 MeV.

Total Neutron Spectrum
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 3.763E-03 5.700E-02 1.450E-01
2.563E-01 7.806E-01 1.306E+00 2.068E+00 2.100E+00 1.435E+00 9.376E-01 1.294E+00
1.082E+00 5.553E-01 1.983E-01 1.457E-01
Total (all groups): 1.236E+01 neutrons/sec-cm^2.
Average Neutron Energy: 3.475E+00 MeV.

==========

Title: Alphas at interface bc using region C materials for neutron production

Neutron Spectrum (neuts/cm^2-sec)
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.101E-07
7.760E-02 7.527E-01 1.964E+00 2.978E+00 4.131E+00 4.716E+00 4.156E+00 2.847E+00
2.816E+00 5.320E+00 7.118E+00 5.952E+00
Total (all groups): 4.283E+01 neutrons/sec-cm^2.
Average Neutron Energy: 2.180E+00 MeV.

Neutron Spectrum (neuts/cm^2-sec)
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 8.456E-04 4.919E+00 3.883E+01 1.071E+02 1.872E+02 2.473E+02 2.441E+02
1.285E+02 4.574E+01 3.291E+01 1.807E+01
Total (all groups): 1.055E+03 neutrons/sec-cm^2.
Average Neutron Energy: 2.597E+00 MeV.

Neutron Spectrum (neuts/cm^2-sec)
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 6.760E-02
1.316E+00 3.510E+00 4.209E+00 2.118E+00
Total (all groups): 1.122E+01 neutrons/sec-cm^2.
Average Neutron Energy: 9.362E-01 MeV.

Total Neutron Spectrum
0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.101E-07
7.760E-02 7.535E-01 6.883E+00 4.181E+01 1.113E+02 1.919E+02 2.514E+02 2.470E+02
1.326E+02 5.457E+01 4.424E+01 2.614E+01
Total (all groups): 1.109E+03 neutrons/sec-cm^2.
Average Neutron Energy: 2.588E+00 MeV.

Title: Total neutron production from all interfaces

Total Neutron Spectrum
0.000E+00 0.000E+00 0.000E+00 0.000E+00 7.594E-09 4.099E-02 1.589E-01 2.117E-01
3.257E-01 1.129E+00 7.343E+00 4.227E+01 1.118E+02 1.924E+02 2.519E+02 2.475E+02
1.329E+02 5.475E+01 4.435E+01 2.632E+01

Total (all groups): 1.113E+03 neutrons/sec-cm^2.
Average Neutron Energy: 2.569E+00 MeV.

APPENDIX B

Energy-Dependent, Thick-Target Yields for Various Target Materials

This appendix contains thick-target yield data (neutrons/incident α-particle) for several α-particle energies and numerous materials. This section includes both the SOURCES 4A calculated data and data from experiments contained in the literature. Plots of the calculated and measured energy-dependent thick-target yields for various materials are shown in Figs. B-1 through B-11. Following these plots are tables showing the actual data used to construct the figures. The measured data are referenced in both Section VIII and Appendix C.

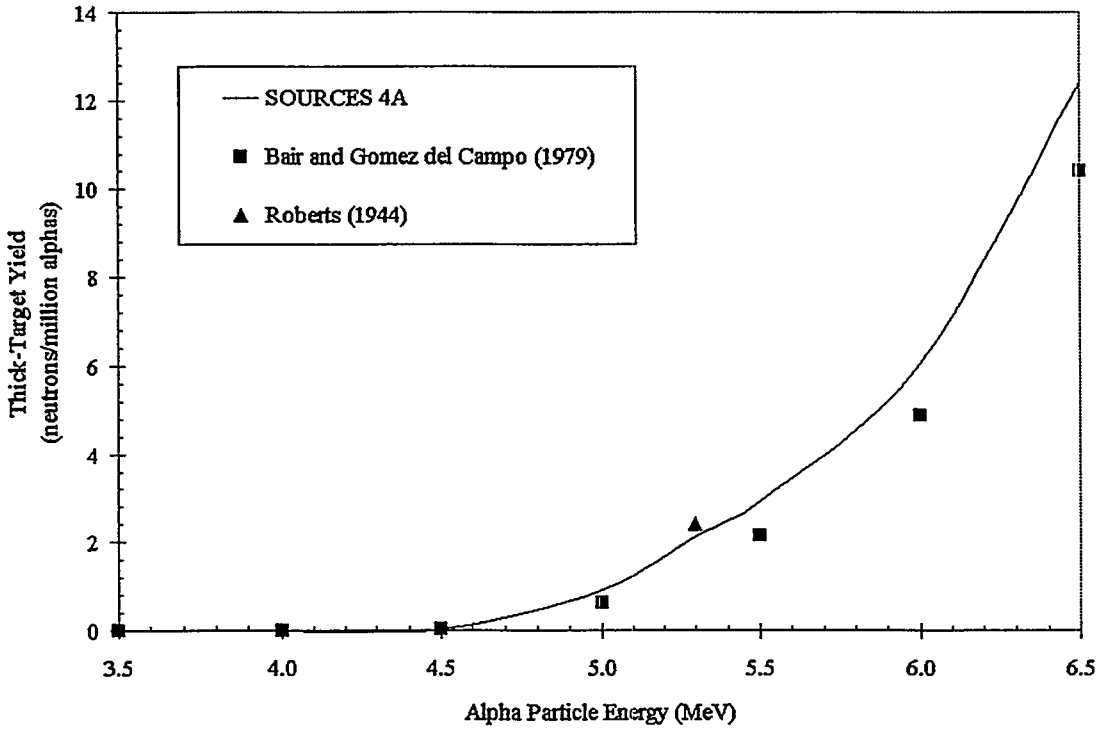


Fig. B-1. Energy-Dependent Thick-Target Yields as Calculated by SOURCES 4A and Compared to Measured Data for Lithium.

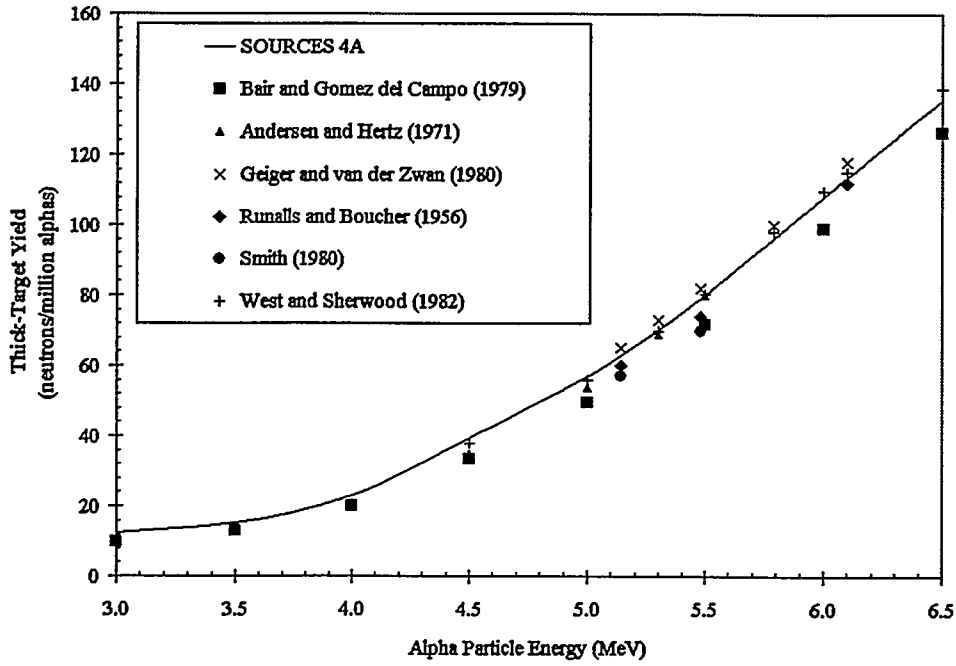


Fig. B-2. Energy-Dependent Thick-Target Yields as Calculated by SOURCES 4A and Compared to Measured Data for Beryllium.

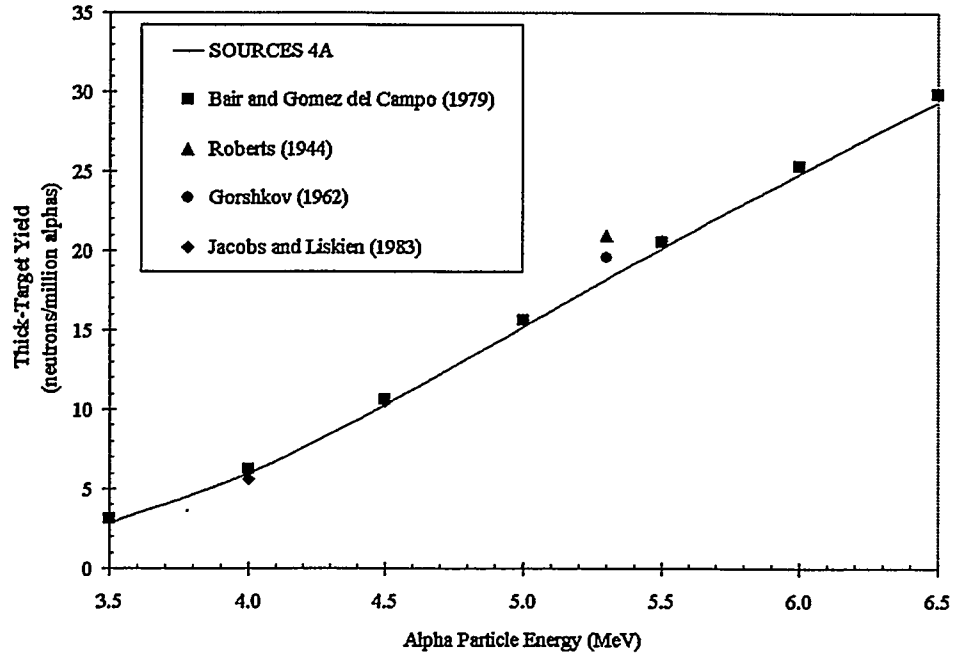


Fig. B-3. Energy-Dependent Thick-Target Yields as Calculated by SOURCES 4A and Compared to Measured Data for Boron.

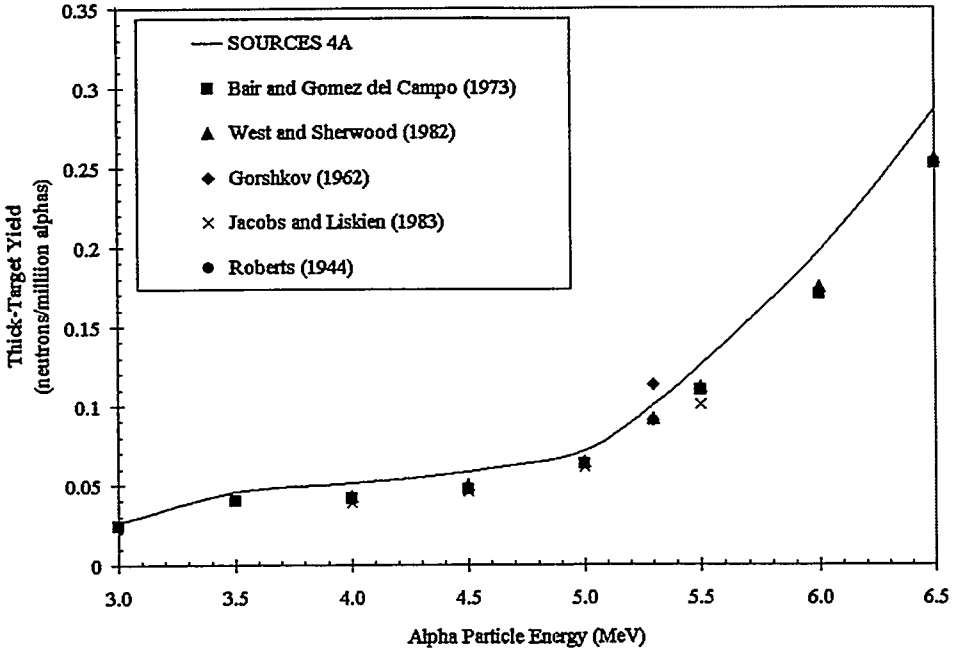


Fig. B-4. Energy-Dependent Thick-Target Yields as Calculated by SOURCES 4A and Compared to Measured Data for Carbon.

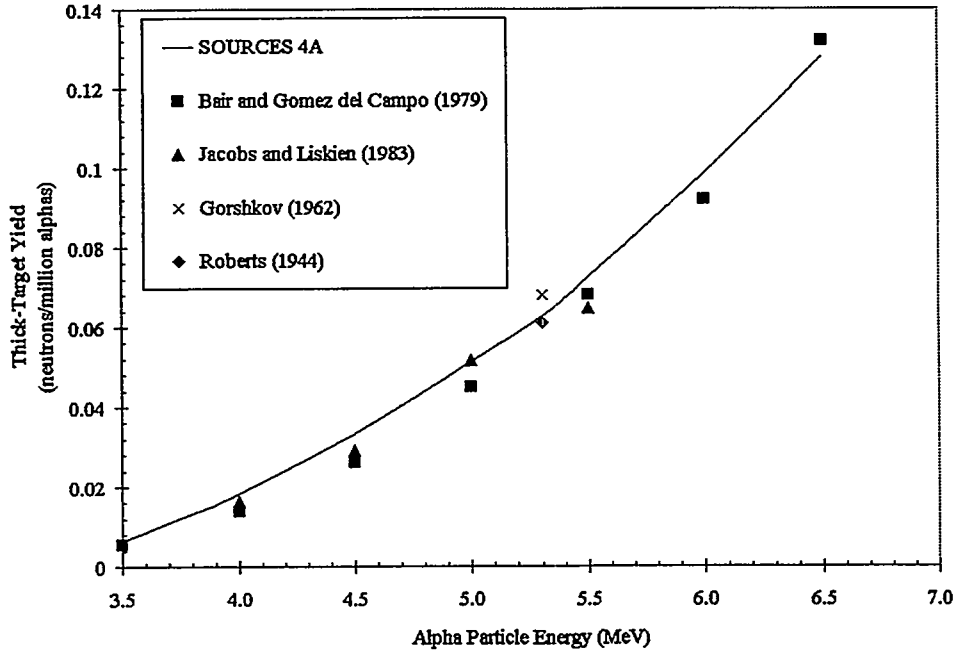


Fig. B-5. Energy-Dependent Thick-Target Yields as Calculated by SOURCES 4A and Compared to Measured Data for Oxygen.

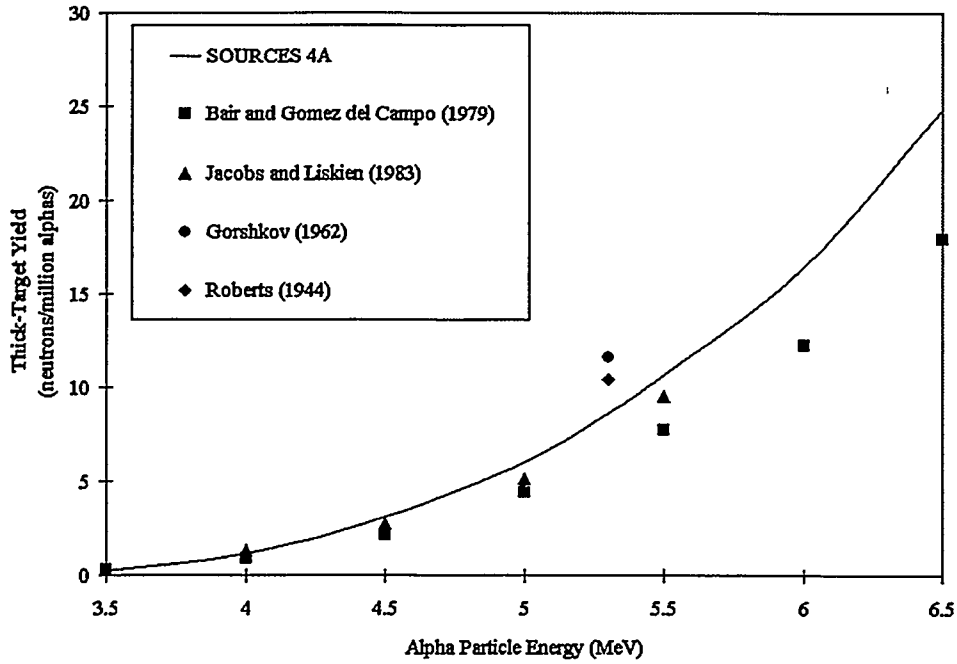


Fig. B-6. Energy-Dependent Thick-Target Yields as Calculated by SOURCES 4A and Compared to Measured Data for Fluorine.

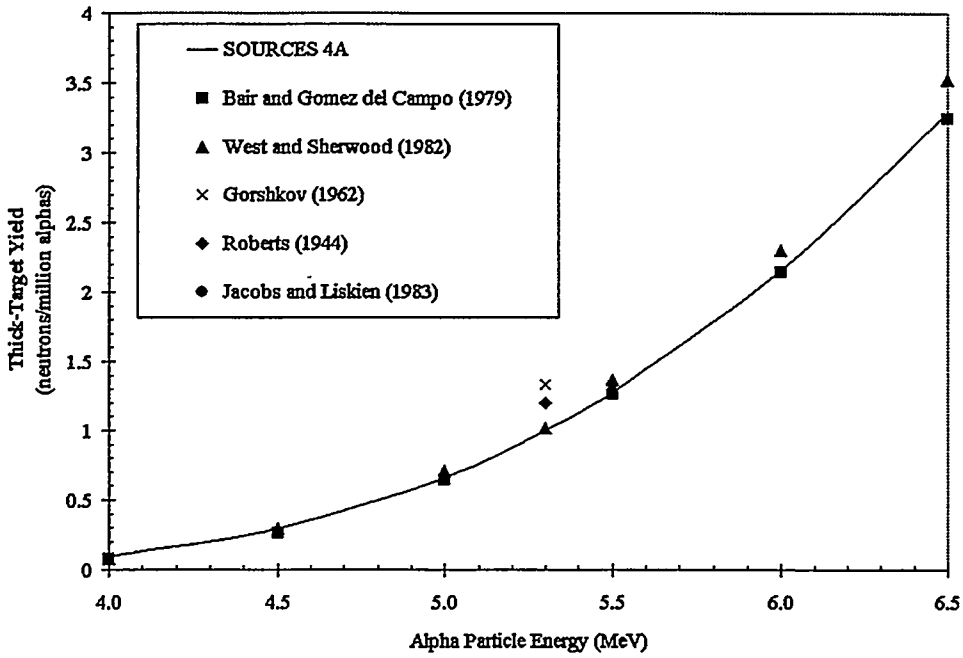


Fig. B-7. Energy-Dependent Thick-Target Yields as Calculated by SOURCES 4A and Compared to Measured Data for Magnesium.

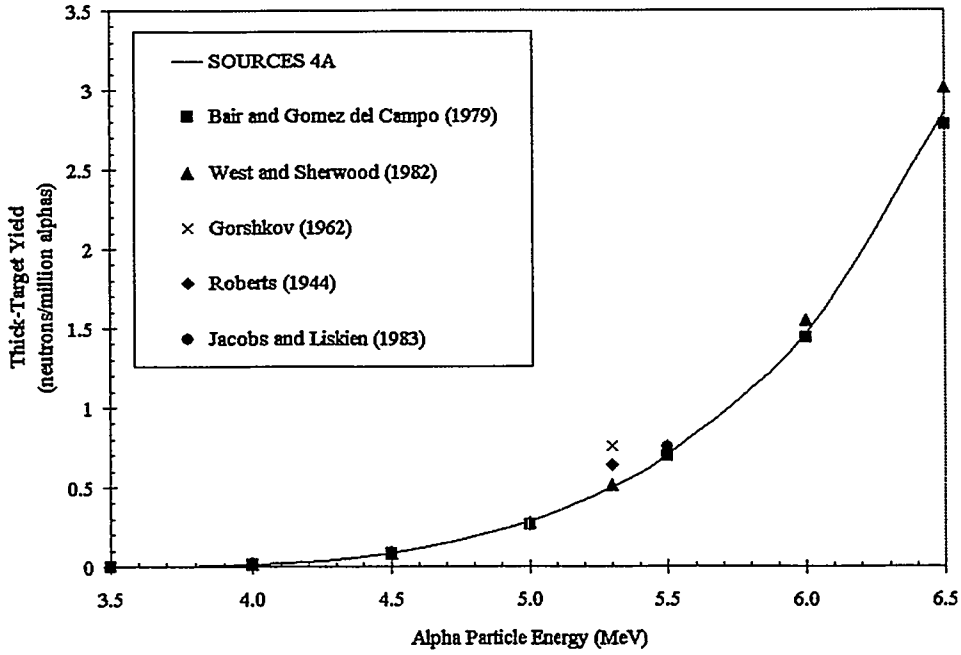


Fig. B-8. Energy-Dependent Thick-Target Yields as Calculated by SOURCES 4A and Compared to Measured Data for Aluminum.

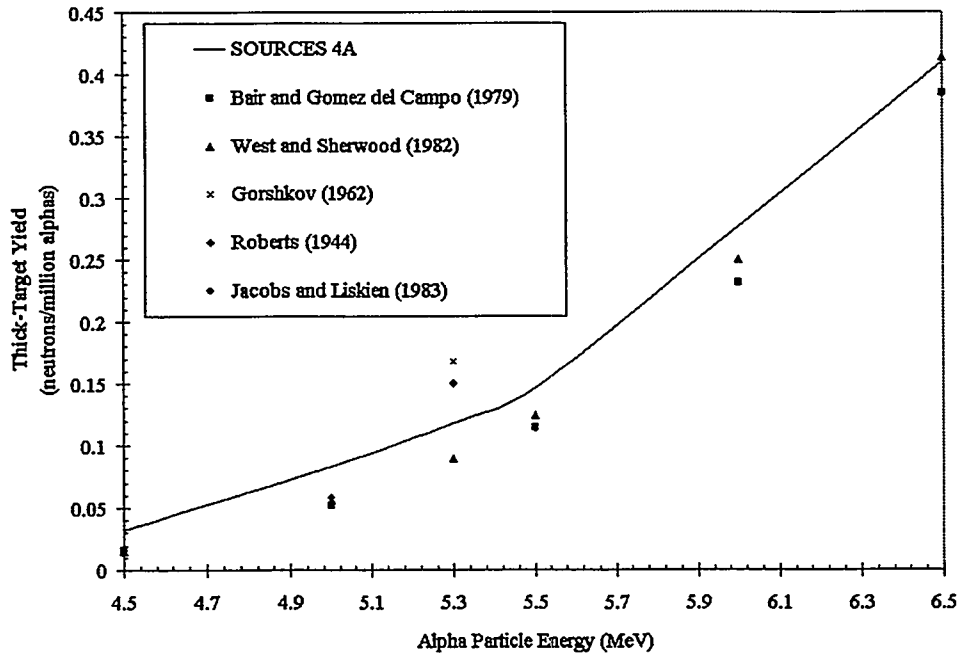


Fig. B-9. Energy-Dependent Thick-Target Yields as Calculated by SOURCES 4A and Compared to Measured Data for Silicon.

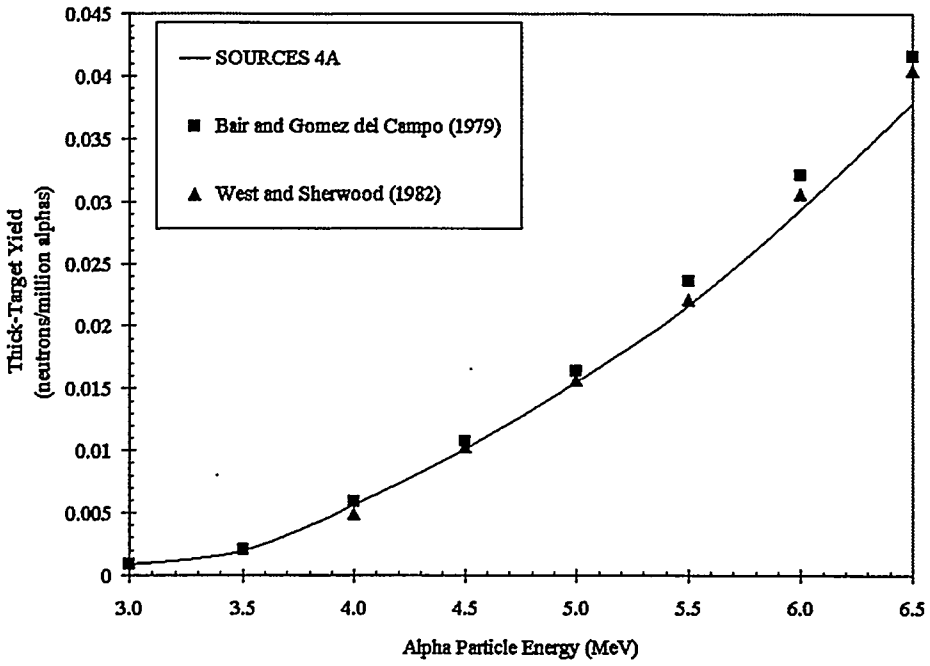


Fig. B-10. Energy-Dependent Thick-Target Yields as Calculated by SOURCES 4A and Compared to Measured Data for Uranium Dioxide.

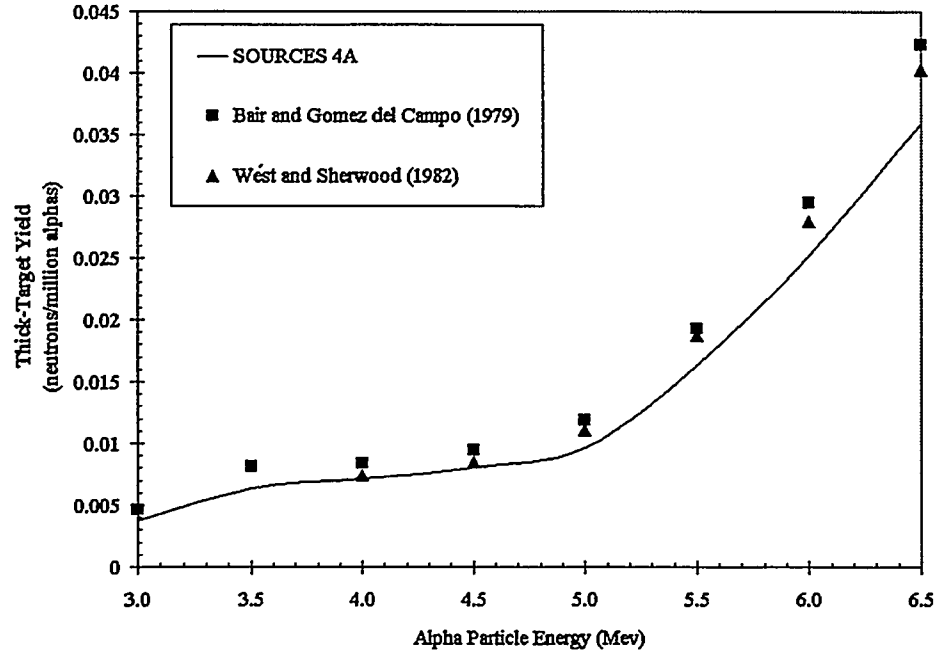


Fig. B-11. Energy-Dependent Thick-Target Yields as Calculated by SOURCES 4A and Compared to Measured Data for Uranium Carbide.

Thick Target Yield Data

(α-Particle Energies in MeV and Yields in neutrons/million α-particles)

Natural Lithium Target

E_a	SOURCES 4A	Ref. 8	Ref. C-5
3.5	0	0.001	
4	0	0.002	
4.5	0.033964562	0.028	
5	0.909993912	0.629	
5.3	2.137204012		2.4
5.5	2.919029767	2.15	
6	6.039924381	4.873	
6.5	12.39706495	10.41	

Be-9 Target

E_a	SOURCES 4A	Ref. 8	Ref. C-1	Ref. 7	Ref. C-6	Ref. C-8	Ref. C-7
3	12.25928418	9.79	10.05				
3.5	15.34813676	12.97	14.4				
4	22.78509404	19.88	21.1				
4.5	39.22906854	33.27	34.8			37.63	
5	56.71440931	49.43	53.5			55.87	
5.14	62.74151687			65	60		57
5.3	70.13681951		69	73		69.77	
5.48	79.04130219			82	74		70
5.5	80.06344324	71.81	80			80.46	
5.79	96.26710244			100		98.17	
6	108.1450864	99.16				109.7	
6.1	113.5826204			118	112	115.3	
6.5	135.6019097	126.2				138.7	

Natural Boron Target

E_n	SOURCES 4A	Ref. 8	Ref. C-5	Ref. C-2	Ref. 18
3.5	2.8645583	3.15			
4	6.004678138	6.238			5.6
4.5	10.25665673	10.63			10.5
5	15.21676439	15.64			15.6
5.3	18.20308244		21	19.6	
5.5	20.14162581	20.59			20.6
6	24.8774392	25.35			
6.5	29.36332468	29.85			

Natural Carbon Target

E_n	SOURCES 4A	Ref. 8	Ref. C-8	Ref. C-2	Ref. 18	Ref. C-5
3	0.02608222	0.024				
3.5	0.045820116	0.04				
4	0.051267263	0.042	0.04329		0.039	
4.5	0.058316511	0.047	0.0497		0.046	
5	0.071453747	0.063	0.06468		0.061	
5.3	0.100932423		0.09136	0.113		0.09
5.5	0.125604793	0.11	0.1116		0.101	
6	0.197058541	0.17	0.1748			
6.5	0.285814989	0.252	0.2555			

Natural Oxygen Target

E_a	SOURCES 4A	Ref. 8	Ref. 18	Ref. C-2	Ref. C-5
3.5	0.006376366	0.0056			
4	0.018360088	0.014	0.0164		
4.5	0.033323721	0.026	0.0293		
5	0.051267263	0.045	0.0518		
5.3	0.062802397			0.068	0.061
5.5	0.072415008	0.068	0.0646		
6	0.098689481	0.0919			
6.5	0.128168157	0.132			

Fluorine Target

E_a	SOURCES 4A	Ref. 8	Ref. 18	Ref. C-2	Ref. C-5
3.5	0.209875356	0.31			
4	1.15351341	0.879	1.28		
4.5	3.076035759	2.159	2.76		
5	6.033515973	4.394	5.09		
5.3	8.612900125			11.6	10.4
5.5	10.62514018	7.746	9.5		
6	16.50485437	12.26			
6.5	24.84860136	17.95			

Natural Magnesium Target

E_n	SOURCES 4A	Ref. 8	Ref. C-8	Ref. C-2	Ref. C-5	Ref. 18
4	0.097407799	0.077	0.08272			0.073
4.5	0.291902977	0.263	0.2932			0.26
5	0.660066007	0.644	0.7039			0.665
5.3	0.999711622		1.018	1.33	1.2	
5.5	1.278477362	1.262	1.368			1.33
6	2.159633439	2.141	2.298			
6.5	3.293921625	3.25	3.523			

Natural Aluminum Target

E_n	SOURCES 4A	Ref. 8	Ref. C-8	Ref. C-2	Ref. C-5	Ref. 18
3.5	0.000839501	0.0012				
4	0.015828767	0.0169	0.01655			0.019
4.5	0.082988881	0.0802	0.08124			0.087
5	0.281329104	0.2643	0.2812			0.26
5.3	0.496651607		0.5119	0.76	0.64	
5.5	0.698516454	0.6967	0.7555			0.747
6	1.467525393	1.438	1.549			
6.5	2.854945689	2.78	3.015			

Natural Silicon Target

E_n	SOURCES 4A	Ref. 8	Ref. C-8	Ref. C-2	Ref. C-5	Ref. 18
4.5	0.031785703	0.016	0.0156			0.0138
5	0.083309302	0.052	0.05649			0.0581
5.3	0.117594284		0.08954	0.168	0.15	
5.5	0.146432119	0.114	0.1245			0.113
6	0.277163639	0.231	0.2504			
6.5	0.410138101	0.385	0.4131			

Uranium Dioxide

E_a	SOURCES 4A	Ref. 8	Ref. C-8
3	0.000852318	0.00091	
3.5	0.001960973	0.00208	
4	0.005607357	0.00593	0.004935
4.5	0.010125284	0.0107	0.0103
5	0.015476305	0.0164	0.01568
5.5	0.021660418	0.0236	0.02208
6	0.029318466	0.0321	0.03061
6.5	0.037809606	0.0416	0.04044

Uranium Carbide

E_a	SOURCES 4A	Ref. 8	Ref. C-8
3	0.003716877	0.00468	
3.5	0.006376366	0.0081	
4	0.007113333	0.00842	0.007432
4.5	0.00801051	0.00943	0.008472
5	0.009676696	0.0119	0.01104
5.5	0.016373482	0.0193	0.01879
6	0.025153001	0.0295	0.02793
6.5	0.035887084	0.0423	0.04023

APPENDIX C

General Bibliography

C-1. M. E. ANDERSON and M. R. HERTZ, "Thick Target Yields for the Be(α,n) Reaction," *Nucl. Sci. Eng.* **44**, page 437 (1971).

C-2. G. V. GORSHKOV, V. A. ZYABKIN, and O. S. TSVETKOV, "Neutron Yield from the Reaction (α,n) in Be, C, O, F, Mg, Al, Si, and Granite Irradiated with Polonium α-Particles," *Atomnaya Energiya* **13**, page 65 (1962).

C-3. R. T. PERRY and W. B. WILSON, "Alpha-N and Spontaneous Fission Sources from Individual Plutonium Isotopes in PuF_4 and PuO_2," Los Alamos National Laboratory document LA-UR-96-1859 (1996).

C-4. R. T. PERRY, W. B. WILSON, T. H. BROWN, and T. A. PARISH, "(Alpha,n) Neutron Sources and Spectra at the Interface of an Alpha Source Region and a Neutron Producing Region," *Trans. Am. Nucl. Soc.* **76**, page 343 (1997).

C-5. J. H. ROBERTS, "Neutron Yields of Several Light Elements Bombarded with Polonium Alpha Particles," MMDC-731 (1944).

C-6. O. J. C. RUNALLS and R. R. BOUCHER, "Neutron Yields from Actinide-Beryllium Alloys," *Can. J. Phys.* **34**, page 949 (1956).

C-7. A. B. SMITH, "Neutron Sources in Perspective," INDC(NDS)-114/GT, page 19 (1980).

C-8. D. WEST and A. C. SHERWOOD, "Measurements of Thick-Target (α,n) Yields from Light Elements," *Ann. Nucl. Energy* **9**, page 551 (1982).

C-9. W. B. WILSON, T. R. ENGLAND, W. C. HOPKINS, and R. T. PERRY, "SOURCES Calculation of TMI-2 Spontaneous-Fission and (α,n) Neutron Sources," Los Alamos National Laboratory report LA-10288-PR, page 61 (1984).

C-10. R. T. PERRY and W. B. WILSON, "Neutron Production from (alpha,n) Reactions and Spontaneous Fission in ThO_2, UO_2, and (U, Pu)O_2 Fuels," Los Alamos Scientific Laboratory report LA-8869-MS (1981).

C-11. W. B. WILSON, R. T. PERRY, J. E. STEWART, T. R. ENGLAND, D. G. MADLAND, and E. D. ARTHUR, "Development of Sources Code and Data Library for the Calculation of Neutron Sources and Spectra from (alpha,n) Reactions, Spontaneous Fission, (ß⁻, n) and Delayed Neutrons," Los Alamos National Laboratory report LA-9841-PR, pages 65-66 (1983).

C-12. W. B. WILSON, R. T. PERRY, J. E. STEWART, T. R. ENGLAND, D. G. MADLAND, and E. D. ARTHUR, "Development of Methodology for Calculating (alpha, n), Spontaneous Fission, and ß⁻ Delayed Neutron Sources and Spectra in Reactor Fuels," Oak Ridge National Laboratory *Actinide Newsletter,* issue 7, page 40 (1984).

C-13. W. B. WILSON, R. T. PERRY, J. E. STEWART, T. R. ENGLAND, E. D. ARTHUR, and D. G. MADLAND, "Development of SOURCES Code and Data Library for the Calculation of (alpha, n), Delayed Neutrons, (gamma,n) and Spontaneous Fission Decay Neutron Sources and Spectra," in *Reports to the DOE Nuclear Data Committee,* Brookhaven National Laboratory report BNL-NCS-34291, pages 113-114 (1984).

C-14. W. B. WILSON and R. T. PERRY, "Evaluation of ^{10,11}B(alpha, n) Cross Sections," in *Applied Nuclear Science Research and Development Progress Report, June 1, 1985-November 30, 1985,* Los Alamos National Laboratory progress report LA-10689-PR, pages 96-98 (1986).

C-15. W. B. WILSON, M. BOZOIAN, and R. T. PERRY, "Calculated Alpha-Induced Thick Target Neutron Yields and Spectra, with Comparison to Measured Data,"

Proceedings of the International Conference on Nuclear Data for Science and Technology, Mito, Japan, May 30–June 3, 1988, pages 1193-1197 (1988).

C-16. R. T. PERRY, W. B. WILSON, T. H. BROWN, and T. PARISH, "(Alpha,n) Neutron Sources and Spectra at the Interface of an Alpha Source Region and a Neutron Producing Region," *Trans. Am. Nucl. Soc.* **76**, page 343 (1997).

C-17. R. T. PERRY and W. B. WILSON, "Neutron Production from (alpha,n) Reactions in PuO_2, UO_2, and ThO_2 Fuels," *Trans. Am. Nucl. Soc.* **35**, page 549 (1980).

C-18. W. B. WILSON, J. E. STEWART, and R. T. PERRY, "Neutron Production in UF_6 from the Decay of Uranium Nuclides," *Trans. Am. Nucl. Soc.* **38**, page 176 (1981).

C-19. R. T. PERRY and W. B. WILSON, "On the (alpha,n) Neutron Production in Boron-Containing Systems," *Trans. Am. Nucl. Soc.* **46**, page 763 (1984).

C-20. R. T. PERRY, T. A. PARISH, and W. B. WILSON, "A 14 MeV Neutron Source," *Fusion Technology* **8(1)**, page 1454 (1985).

C-21. W. B. WILSON, J. E. STEWART, T. R. ENGLAND, and R. T. PERRY, "Calculations of Neutron Sources and Spectra in a PWR Fuel," *Trans. Am. Nucl. Soc.* **50**, page 569 (1985).

C-22. T. A. PARISH, R. T. PERRY, and W. B. WILSON, "Neutron Sources and Spectra from Cold Fusion," *Journal of Fusion Energy* **9(4)**, page 479 (1990).

C-23. R. T. PERRY and W. B. WILSON, "Neutron Sources and Spectra from Cold Fusion," Los Alamos National Laboratory document LA-UR-89-1678 (Revised).

C-24. R. T. PERRY, "Neutron Radiation Damage from Actinide Decay and Fission in the Hanford Waste Tanks," Los Alamos National Laboratory publication LALP 93-84, page 184 (1983).

C-25. R. T. PERRY, "Neutron and Gamma Sources in Hanford Waste Tank 241-101-SY," *Proceedings 8th International Conference on Radiation Shielding,* Arlington, Texas, April 24–28, 1994, pages 523-527 (1994).

C-26. R. T. PERRY, "Estimation of Neutron Radiation Damage from Actinide Decay and Fission in the Hanford Waste Tanks," *Journal of Alloys and Compounds* **213/214**, page 560 (1994).

C-27. R. T. PERRY and W. B. WILSON, "Alpha-n and Spontaneous Fission Sources and Spectra from Individual Plutonium Isotopes in PuF_4 and PuO_2," *Trans. Am. Nucl. Soc.* **75**, page 299 (1996).

C-28. D. E. KORNREICH, S. T. BOERIGTER, M. G. HOUTS, R. T. PERRY, and D. I. POSTON, "Analysis of Salt-to Oxide Procession Schedule for PF-4 Vault Storage Containers Based on Accumulated Neutron Dose Equivalent," Los Alamos National Laboratory report LA-CP-96-172 (1996).

C-29. W. B. WILSON, M. BOZOIAN, and R. T. PERRY, "Calculated Alpha-Induced Thick-Target Neutron Yields and Spectra," *Reports to the DOE Nuclear Data Committee*, Brookhaven National Laboratory report BNL-NCS-40911, p. 108 (1988).

C-30. W. B. WILSON, R. T. PERRY, J. E. STEWART, and T. R. ENGLAND, "Calculation of Neutron Sources in PWR Fuel," *Applied Nuclear Science Research and Development Progress Report, June 1, 1984–May 31, 1985,* Los Alamos National Laboratory progress report LA-10513-PR, pages 42-44 (1985).

C-31. R. T. PERRY, T. A. PARISH, and W. B. WILSON, "Feasibility Study of a Thermal Neutron Driven 14-MeV Neutron Source," *Applied Nuclear Science Research and Development Progress Report, June 1, 198–May 31, 1985,* Los Alamos National Laboratory progress report LA-10513-PR, pages 49-51 (1985).

C-32. W. B. WILSON, R. T. PERRY, J. E. STEWART, and T. R. ENGLAND, "Neutron Sources in Reactor Fuel," *Reports to the DOE Nuclear Data Committee,* Brookhaven National Laboratory report BNL-NCS-36272, pages 99-100 (1985).

C-33. W. B. WILSON, T. R. ENGLAND, W. C. HOPKINS, and R. T. PERRY, "SOURCES Calculation of TMI-2 Spontaneous-Fission and (α,n) Neutron Sources," *Applied Nuclear Science Research and Development Semiannual Progress Report, October 1, 1983–May 31, 1984,* Los Alamos National Laboratory progress report LA-10288-PR, pages 61-62 (1985).

C-34. R. T. PERRY and W. B. WILSON, "(α,n), Neutron production in Boron-Containing Systems," *Applied Nuclear Data Research and Development, Semiannual Progress Report, April 1, 1983–September 30, 1983,* Los Alamos National Laboratory progress report LA-10069-PR, pages 86-87 (1984).

C-35. W. B. WILSON, R. T. PERRY, and J. E. STEWART, "Neutron Production in UO_2F_2 from the Spontaneous Fission and Alpha Decay of U Nuclides and Subsequent $^{17,18}O(\alpha,n)$ and $^{19}F(\alpha,n)$ Reactions," *Applied Nuclear Data Research and Development Progress Report, April 1, 1981–June 30, 1981,* Los Alamos National Laboratory progress report LA-9060-PR, pages 50-51 (1981).

C-36. W. B. WILSON, R. T. PERRY, D. G. MADLAND, N. ENSSLIN, and J. E. STEWART, "Calculated Neutron Sources in Pu Process Solutions," *Applied Nuclear Data Research and Development Progress Report, October 1, 1981–March 31, 1982,* Los Alamos National Laboratory progress report LA-9468-PR, pages 75-79 (1982).

C-37. W. B. WILSON and R. T. PERRY, "Calculated Neutron Sources in Plutonium Oxalate," *Applied Nuclear Data Research and Development Progress Report, April 1, 1982–September 30, 1982,* Los Alamos National Laboratory progress report LA-9647-PR, pages 88-91 (1983).

C-38. W. B. WILSON, J. E. STEWART, and R. T. PERRY, "The ^{19}F (α,n) Neutron Production from the Decay of U Nuclides in UF_6," *Applied Nuclear Data Research and Development Progress Report October 1, 1980–December 31, 1980,* Los Alamos National Laboratory progress report LA-8757-PR, pages 40-48 (1981).

C-39. W. B. WILSON, R. T. PERRY, T. R. ENGLAND, R. J. LABAUVE, M. E. BATTAT, and N. L. WHITTEMORE, "Neutron Production from Actinide Decay in Oxide Fuels," *Applied Nuclear Data Research and Development, July 1–September 30, 1980,* Los Alamos National Laboratory progress report LA-8630-PR, pages 23-29 (1980).

C-40. R. T. PERRY and W. B. WILSON, "The (α,n) Neutron Production by Alpha Particles in PuO_2, UO_2, ThO_2 Fuels," *Applied Nuclear Data Research and Development Progress Report, April–June 30, 1980,* Los Alamos National Laboratory progress report LA-8524-PR, pages 20-22 (1980).

C-41. G. P. ESTES, "Benchmarking of the SOURCES 3A Code for PuBe Sources," *Am. Nucl. Soc. Trans.* **79**, page 270 (1998).

C-42. T. H. BROWN, W. B. WILSON, R. T. PERRY, and W. S. CHARLTON, "Comparison of (α,n) Thick-Target Neutron Yields and Spectra from ORIGEN-S and SOURCES," *Am. Nucl. Soc. Trans.* **79**, page 272 (1998).

C-43. W. S. CHARLTON and T. A. PARISH, "Alpha-Particle Source Rate and Spectra at the Surface of a Spherical Source Particle," *Am. Nucl. Soc. Trans.* **79**, page 274 (1998).

C-44. W. S. CHARLTON, R. T. PERRY, G. P. ESTES, and T. A. PARISH, "Derivation and Implementation of the Three Region Problem in the Sources Code System," Los Alamos National Laboratory document LA-UR-98-5983 (1998).

APPENDIX D

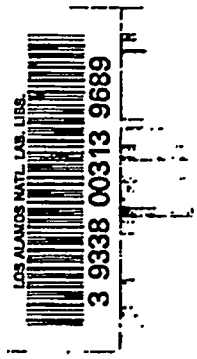
Los Alamos Scientific Laboratory Report LA-8869-MS

LA-8869-MS

UC-34c
Issued: June 1981

Neutron Production from (a,n) Reactions and Spontaneous Fission in ThO_2, UO_2, and $(U,Pu)O_2$ Fuels

R. T. Perry*
W. B. Wilson



*LASL Consultant. University of Wisconsin, 329 Engineering Research Building, Madison, WI 53706.

NEUTRON PRODUCTION FROM (α,n) REACTIONS AND SPONTANEOUS FISSION IN ThO_2, UO_2, AND $(U,Pu)O_2$ FUELS

by

R. T. Perry and W. B. Wilson

ABSTRACT

Available alpha-particle stopping cross-section and $^{17,18}O(\alpha,n)$ cross-section data were adjusted, fitted, and used in calculating the thick-target neutron production function for alpha particles below 10 MeV in oxide fuels. The spent UO_2 function produced was folded with actinide decay spectra to determine (α,n) neutron production by each of 89 actinides. Spontaneous-fission (SF) neutron production for 40 actinides was calculated as the product of $\bar{\nu}$(SF) and SF branching-fraction values accumulated or estimated from available data. These contributions and total neutron production in spent UO_2 fuel are tabulated and, when combined with any calculated inventory, describe the spent UO_2 neutron source. All data are tabulated and methodology is described to permit easy extension to specialized problems.

I. INTRODUCTION

Neutron sources are present in reactor fuel from the spontaneous-fission (SF) decay of actinide nuclides and from the interaction of their decay alpha particles with low- and medium-Z nuclides in (α,n) reactions. The (α,n) source in oxide fuels is dominated by reactions with ^{17}O and ^{18}O, which are present in ^{NAT}O in 0.038 and 0.204 atom percent abundancies, respectively.

The probability of neutron production by an alpha particle emitted at energy E_α in the fuel is given by the thick-target neutron production function $P(E_\alpha)$, which we have evaluated for four fuel compositions--clean ThO_2 thermal

reactor fuel, clean and spent UO_2 thermal reactor fuel, and clean $(U,Pu)O_2$ fast reactor fuel. The (α,n) neutron production function has been evaluated at the Hanford Engineering Development Laboratory (HEDL) by Ombrellaro and Johnson for alpha particles in FFTF fuel;[1] however, $P(E_\alpha)$ has not been calculated for the fuels of interest here, and the change in $P(E_\alpha)$ with exposure has not been evaluated. We have employed the methodology and data used in the HEDL work[1] with minor exceptions in data and energy range of calculation.

The equations describing (α,n) and SF neutron production and the data quantities used in the calculations are given in Sec. II. The available data sources and adjustments made to the data are described in Sec. III. Details of the (α,n) calculations are briefly discussed in Sec. IV. Resulting (α,n), SF, and total neutron production values are given in Sec. V for each of a variety of actinide nuclides produced in reactor fuels.

Selected results of these calculations have been reported previously.[2-5]

II. THEORY

The slowing and stopping of alpha particles in a material are described by the material's alpha-particle stopping power,

$$SP(E) = -\frac{dE}{dx} \quad , \tag{1}$$

which gives the energy-dependent energy loss of alpha particles of energy E per unit path length x.[6] The energy loss of an alpha particle of initial energy E_α in traveling a distance X can be determined from the stopping power as

$$\Delta E = E_\alpha - E'_\alpha = \int_0^X \left(-\frac{dE}{dx}\right) dx \quad . \tag{2}$$

Similarly, the distance traveled in slowing from E_α to E'_α is

$$X = \int_{E_\alpha}^{E'_\alpha} \frac{1}{\left(\frac{dE}{dx}\right)} dE = \int_{E'_\alpha}^{E_\alpha} \frac{1}{\left(-\frac{dE}{dx}\right)} dE \quad . \tag{3}$$

Neutrons may be produced within the material by (α,n) reactions with nuclide i, which has atom density N_i and microscopic (α,n) cross section $\sigma_i(E)$. The probability of (α,n) interaction with nuclide i by an alpha particle of energy E traveling from x to x + dx is

$$N_i \sigma_i(E)dx = \frac{N_i \sigma_i(E)dE}{\left(\frac{dE}{dx}\right)} \quad . \tag{4}$$

The probability of (α,n) interaction with nuclide i by an alpha particle in lieu of slowing from E_α to E'_α is then

$$p_i(E_\alpha, E'_\alpha) = \int_{E_\alpha}^{E'_\alpha} \frac{N_i \sigma_i(E)dE}{\left(\frac{dE}{dx}\right)} = \int_{E'_\alpha}^{E_\alpha} \frac{N_i \sigma_i(E)dE}{\left(-\frac{dE}{dx}\right)} \quad . \tag{5}$$

The probability of (α,n) interaction with nuclide i by an alpha particle prior to stopping in the material is given by the thick-target neutron production function

$$P_i(E_\alpha) = \int_0^{E_\alpha} \frac{N_i \sigma_i(E)dE}{\left(-\frac{dE}{dx}\right)} \quad . \tag{6}$$

In addition to that of the above definition of Eq. (1), a variety of quantities are referred to as "stopping powers" or often alternately "stopping cross sections." These include (typically without explicit regard to sign) the quantities $\frac{dE}{d\chi} = \frac{dE}{d(\rho x)} = \frac{dE}{\rho dx}$,[7] $\frac{dE}{Z^2 dx}$,[8] $\rho \frac{dE}{dx}$, and $\frac{dE}{Ndx}$.[9] Here χ is material thickness (mg/cm^2), Z is atomic number, ρ is material density (g/cm^3), and N is the total atom density of the material ($atoms/cm^3$). The last quantity is also called the stopping cross section,

$$\varepsilon(E) = -\frac{1}{N}\frac{dE}{dx} \quad , \tag{7}$$

a notation adopted here. Equations above defining p_i and P_i may now be written in terms of ε as

$$p_i(E_\alpha, E'_\alpha) = \frac{N_i}{N} \int_{E'_\alpha}^{E_\alpha} \frac{\sigma_i(E)}{\varepsilon(E)}\, dE \quad . \tag{8}$$

and

$$P_i(E_\alpha) = \frac{N_i}{N} \int_0^{E_\alpha} \frac{\sigma_i(E)}{\varepsilon(E)}\, dE \quad . \tag{9}$$

Note that p_i and P_i are related by

$$p_i(E_\alpha, E'_\alpha) = P_i(E_\alpha) - P_i(E'_\alpha) \quad . \tag{10}$$

The stopping cross section $\varepsilon(E)$ of a material composed of J elemental constituents may be calculated using the Bragg-Kleeman[10] relationship, which may be written as

$$\varepsilon(E) = \frac{1}{N} \sum_{j=1}^{J} N_j \varepsilon_j(E) \quad , \tag{11}$$

where

$$N = \sum_{j=1}^{J} N_j \quad . \tag{12}$$

The accuracy of the approximation of Eq. (11) will be discussed in Sec. III.

A fraction of the decays of nuclide k within the material may be by alpha-particle emission. This fraction F_k^α of alpha decays may occur with the emission of one of L possible alpha-particle energies. The intensity $f_{k\ell}^\alpha$ is the fraction of all decays of nuclide k resulting in an alpha particle of energy $E_{k\ell}$, and

$$F_k^\alpha = \sum_{\ell=1}^{L} f_{k\ell}^\alpha \quad . \tag{13}$$

The fraction of nuclide k decays resulting in (α,n) neutron production in a thick-target material containing I nuclides with (α,n) cross sections is thus

$$R_k(\alpha,n) = \sum_{\ell=1}^{L} f_{k\ell}^{\alpha} \sum_{i=1}^{I} P_i(E_{k\ell}) \quad . \tag{14}$$

The SF of an actinide nuclide k is accompanied by the emission of an average $\overline{\nu}_k(SF)$ neutrons. The SF activity A_k^{SF} of nuclide k, having atom density N_k, is

$$A_k^{SF} = \lambda_k^{SF} N_k \quad . \tag{15}$$

Here, λ_k^{SF} is the SF decay constant defined by

$$\lambda_k^{SF} = \ell n 2 / T_{1/2}^k(SF) \quad , \tag{16}$$

where $T_{1/2}^k$ (SF) is the SF half-life of nuclide k. SF is typically only one of M modes of decay; the total activity due to nuclide k is

$$A_k = \lambda_k N_k = \sum_{m=1}^{M} A_k^m \quad , \tag{17}$$

where λ_k is the total decay constant of nuclide k,

$$\lambda_k = \sum_{m=1}^{M} \lambda_k^m = \ell n 2 / T_{1/2}^k \quad , \tag{18}$$

and $T_{1/2}^k$ is the total half-life of nuclide k. The fraction of nuclide k decays by SF is given by the SF branching fraction

$$F_k^{SF} = A_k^{SF}/A_k = \lambda_k^{SF}/\lambda_k = T_{1/2}^k / T_{1/2}^k(SF) \quad . \tag{19}$$

The average number of SF neutrons emitted per decay (by any mode) of nuclide k is then

$$R_k(SF) = F_k^{SF} \bar{\nu}_k(SF) \quad . \tag{20}$$

The total number of neutrons, on the average, emitted due to (α,n) reactions and SF is

$$R_k = R_k(\alpha,n) + R_k(SF) \quad . \tag{21}$$

The total neutron source S from (α,n) reactions and SF within a material containing K pertinent radionuclides is then

$$S = \sum_{k=1}^{K} \lambda_k N_k R_k \quad . \tag{22}$$

The evaluation of the quantities $R_k(\alpha,n)$, $R_k(SF)$, and R_k for a number of actinide nuclides is described in the following sections.

III. DATA

The data quantities required to compute the neutron production fractions $R_k(\alpha,n)$ and $R_k(SF)$ for each of the four fuels of interest include the following.

- For each major elemental constituent j of the material: N_j, the atom density; and $\varepsilon_j(E)$, the alpha-particle stopping cross section.

- For each nuclide i within the material having an (α,n) cross section: N_i, the atom density; and $\sigma_i(E)$, the microscopic (α,n) cross section.

- For each nuclide k decaying by alpha decay: $f_{k\ell}^{\alpha}$, the intensity for emission of each L alpha particles; and $E_{k\ell}$, the energy of each of L alpha particles.

- For each nuclide k decaying by SF: F_k^{SF}, the SF branching fraction; and $\bar{\nu}_k(SF)$, the average number of neutrons emitted per SF.

A. Stopping Cross Section $\varepsilon(E)$

Densities of each constituent of each fuel type are given in Table I. The fuel compositon of UO_2 LWR fuel is given for clean and spent conditions for the evaluation of the effect of exposure-dependent fuel composition on stopping cross section ε; here, $_{41}Nb$ and $_{59}Pr$ represent the low- and high-mass fission products, respectively. Concentrations of $_{93}Np$, $_{95}Am$, and $_{96}Cm$ are given for the spent UO_2 fuel, although the minor contributions to ε from these nuclides are included as plutonium. Elements contributing to the material stopping cross sections are thus O, Nb, Pr, Th, U, and Pu.

A bibliography of experimental and theoretical stopping-power references by Anderson[11] notes that some 900 papers have been published on the subject of ion energy loss in matter. Anderson, noting the observation by Bichsel[12] that stopping powers measured by different groups often did not agree within stated uncertainties, was unable to resolve discrepancies after careful analysis and cautioned that stopping-power data sources should be selected carefully. We have chosen as the major stopping cross-section data source the comprehensive volume edited by Ziegler,[13] which gives tabulated alpha stopping cross-section values and functional fits for elements in the range $1 \leq Z \leq 92$.

No values of the alpha-stopping cross section for plutonium were identified, although values for plutonium compounds were found.[7] Northcliffe and Schilling[8] have tabulated values of the stopping power $dE/d\chi$ for $Z \leq 92$. They have shown graphically, for each Z including $Z = 94$, the energy-dependent ratio $(dE/d\chi)_Z:(dE/d\chi)_{A\ell}$. In order to form a stopping cross section for plutonium consistent with the data of Ziegler,[13] we have used the stopping power ratio of Ref. 8 in the expression

$$\varepsilon_{Pu} = \varepsilon_u \frac{A_{Pu}}{A_U} \left[(dE/d\chi)_{Pu}:(dE/d\chi)_{A\ell} \frac{(dE/d\chi)_{A\ell}}{(dE/d\chi)_U} \right] , \tag{23}$$

where all quantities enclosed in brackets [] were taken from Ref. 8. Values used and produced in this calculation are given in Table II.

Fourth-degree polynomial functions of the form

$$\ell n\varepsilon = C_0 + C_1 \ell nE + C_2 \ell n^2 E + C_3 \ell n^3 E + C_4 \ell n^4 E \tag{24}$$

TABLE I

PROPERTIES OF OXIDE FUELS

	Thermal Reactor Fuels			Fast Reactor Fuel
	UO_2 Clean	UO_2 Spent	ThO_2 Clean	$(U,Pu)O_2$ Clean
Fuel Density (g/cm^3)	9.95	9.95	9.17	9.62
Exposure GWd/t	0	34	0	0
Atom Densities (atoms/b-cm)				
$^{NAT}_{8}O$	0.04372	0.04372	0.04184	0.04215
^{16}O	0.04361	0.04361	0.04174	0.04205
^{17}O	1.6614-5	1.6614-5	1.5899-5	1.6017-5
^{18}O	8.9189-5	8.9189-5	8.5354-5	8.5986-5
$_{41}Nb$	0	7.893-4	0	0
$_{59}Pr$	0	7.893-4	0	0
$_{90}Th$	0	0	0.02025	0
$_{92}U$	0.02186	0.02085	6.724-4	0.01887
$_{93}Np$	0	1.043-5	0	0
$_{94}Pu$	0	2.037-4	0	0.002634
$_{95}Am$	0	5.692-6	0	0
$_{96}Cm$	0	1.131-6	0	0

TABLE II

DATA OF NORTHCLIFFE AND SCHILLING[a] AND ZIEGLER[b] USED IN CALCULATING THE ALPHA PARTICLE STOPPING CROSS SECTION OF PLUTONIUM

	Stopping Power Ratios and Values from Northcliffe and Schilling				ε(E) Stopping Cross Section	
E_α MeV	$\frac{(dE/d\chi)_{Pu}}{(dE/d\chi)_{A\ell}}$	(MeV/mg/cm^2) $(dE/d\chi)_{A\ell}$	$(dE/d\chi)_U$	$\frac{(dE/d\chi)_{Pu}}{(dE/d\chi)_U}$	eV/(10^{15} atoms/cm^2) U(Ziegler)	Pu(Calculated)
0.100	0.150	0.752	0.135	0.837	75.80	63.74
0.320	0.188	1.219	0.243	0.942	139.93	132.48
0.500	0.214	1.317	0.286	0.986	165.64	164.08
0.805	0.235	1.299	0.312	0.978	178.59	175.40
1.281	0.256	1.170	0.307	0.977	166.77	163.72
2.402	0.291	0.904	0.269	0.978	129.15	126.86
4.003	0.322	0.682	0.223	0.982	100.57	99.23
6.404	0.350	0.512	0.183	0.978	78.65	77.29
10.007	0.382	0.379	0.148	0.980	60.67	59.71
16.010	0.418	0.270	0.114	0.991	47.09	46.90
24.016	0.448	0.200	0.090	1.000	37.01	37.18
48.031	0.490	0.118	0.059	0.983	23.64	23.35

[a]Northcliffe and Schilling, Nucl. Data Tables A7, 233 (1970)

[b]J. F. Ziegler, Helium Stopping Powers and Ranges in All Elemental Matter, Vol. 4 of The Stopping and Ranges of Ions In Matter Series (Pergamon Press, New York, 1977).

were fit to each set of tabulated stopping cross-section values, representing the values within 1% at any energy over the range 0.5 MeV $\leq E_\alpha \leq$ 10 MeV. These functional stopping cross sections are shown in Fig. 1. Coefficients of the polynomial functions are given in Table III. Stopping cross sections of the oxide fuels were formed from these component stopping cross-section functions using the Bragg-Kleeman relationship of Eq. (11) and component densities given in Table I.

Stopping cross-section values of UO_2, ThO_2, and $(U_{.8}Pu_{.2})O_2$ were computed over the range 2 MeV $\leq E_\alpha \leq$ 8 MeV and compared in Table IV with values of ε converted from experimentally measured values of dE/dX reported by Nitzki and Matzke.[7] The measured and calculated values of ε agree within 9% over this range, with calculated values generally lower than measured values.

B. (α,n) Cross Sections

The cross sections for the $^{17,18}O(\alpha,n)$ reactions have been reported over four limited ranges of E_α, although no single measurement extends over the entire range of our interest. Bair and Willard[14] plotted their measured $^{18}O(\alpha,n)^{21}Ne$ cross-section values over the range 2.37 MeV $\leq E_\alpha \leq$ 5.15 MeV. Bair and Hass[15] extended the range of these data down to 1.14 MeV and plotted the $^{17}O(\alpha,n)^{20}Ne$ cross section over the range 1.31 MeV $\leq E_\alpha \leq$ 5.31 MeV. Bair and del Campo[16] later plotted the $^{NAT}O(\alpha,n)$ cross section over the range 3.1 MeV $\leq E_\alpha \leq$ 8 MeV and, based on their measured $^{NAT}O(\alpha,n)$ neutron production by alpha particles in the range 4.62 MeV $\leq E_\alpha \leq$ 4.8 MeV, recommended that the $^{17,18}O(\alpha,n)$ cross sections reported in Refs. 14 and 15 be increased by 35%.

Differential cross sections $d\sigma(E)/d\Omega$ for $^{17,18}O(\alpha,n)$ reactions were measured at higher energies by Hansen et al.,[17] who fit their measured angular distributions with Legendre polynomial expansions that they integrated to yield total $\sigma(\alpha,n)$ values. These values were plotted for the range 4.3 MeV $\leq E_\alpha \leq$ 12.3 MeV, and smooth curves were plotted approximating each set of data.

Except for cross-section values given by Hansen et al.[17] at 9.8, 11.6, and 12.3 MeV, no data were available in other than graphic form--despite the best efforts of Bair,[18] del Campo,[19] and Hansen[20] to resurrect their numerical data. Data taken from the $^{17,18}O(\alpha,n)$ cross-section curves of Refs. 14 and 15 for the earlier HEDL work[1] were supplied to us.[21] These data were thinned to 744 values of the $^{17}O(\alpha,n)$ cross section and 687 values of the $^{18}O(\alpha,n)$ cross section. Fourth-degree polynomial fits were made to data taken from the

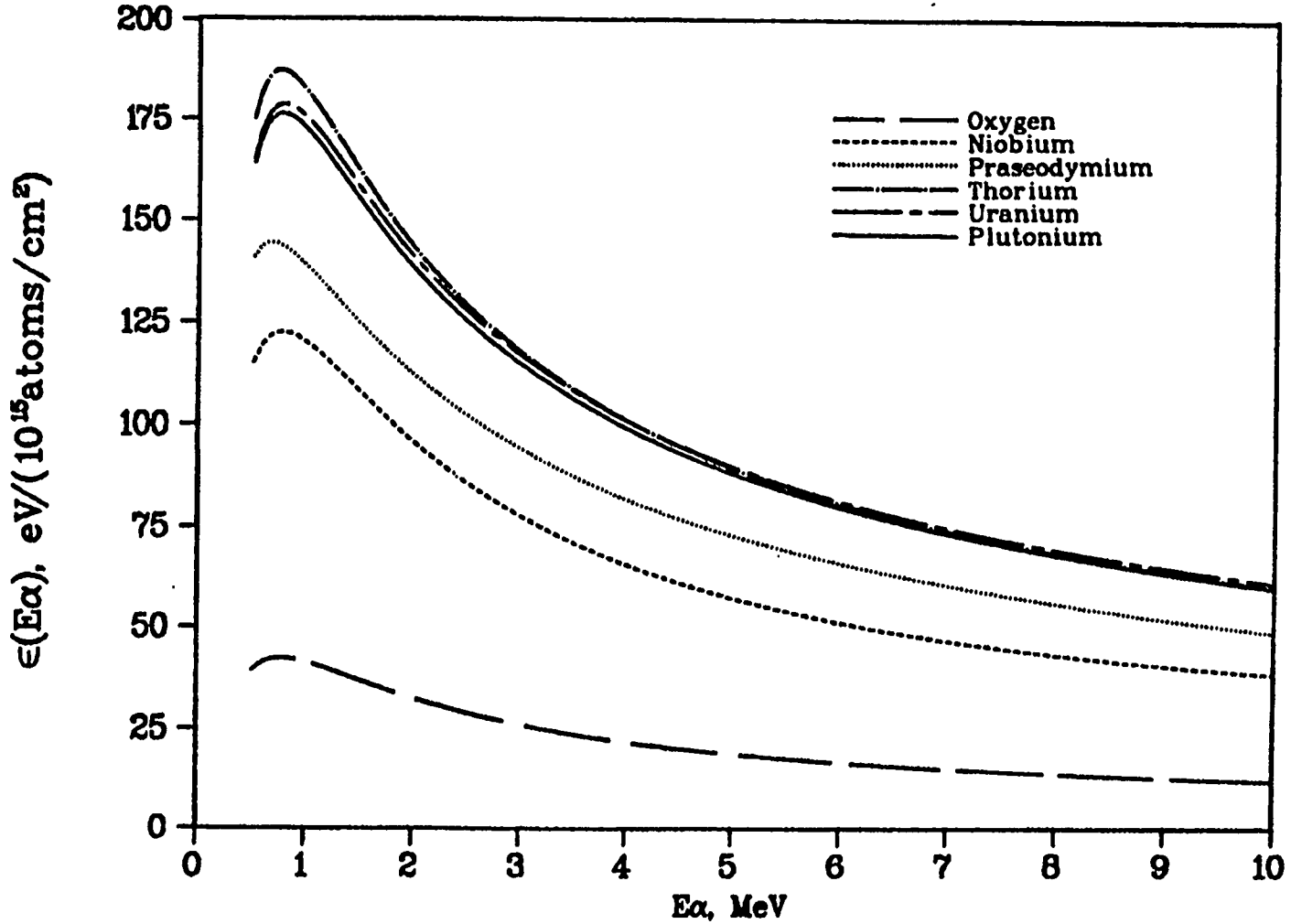


Fig. 1.
Stopping cross sections $\varepsilon(E_\alpha)$ of O, Nb, Pr, Th, U, and Pu.

TABLE III

COEFFICIENTS OF POLYNOMIAL FITS TO STOPPING CROSS SECTIONS[a]

Element	C_0	C_1	C_2	C_3	C_4
O	3.7213	-0.168700	-0.300138	0.0700466	-0.00377296
Ni	4.7872	-0.156294	-0.278932	0.0533399	0.00186590
Pr	4.9321	-0.192312	-0.199561	0.0592391	-0.00940776
Th	5.2027	-0.195369	-0.278809	0.105037	-0.0163945
U	5.1648	-0.161478	-0.279242	0.099232	-0.0146254
Pu	5.1486	-0.171158	-0.272723	0.100975	-0.0160365

[a] $\ell n\varepsilon = C_0 + C_1 \ell nE + C_2 \ell n^2 E + C_3 \ell n^3 E + C_4 \ell n^4 E$,
E is alpha-particle energy in MeV, $0.5 \leq E\ (MeV) \leq 10.0$, and
ε is stopping cross section in $eV/(10^{15}\ atoms/cm^2)$.

TABLE IV

COMPARISON OF CALCULATED AND MEASURED
ALPHA STOPPING CROSS SECTIONS FOR OXIDE FUELS

	$\varepsilon(E)$ for ThO_2			$\varepsilon(E)$ for UO_2			$\varepsilon(E)$ for $(U_{.8}Pu_{.2})O_2$		
E MeV	From a	From Table III and Eq. (11)	% Dif	From a	From Table III and Eq. (11)	% Dif	From a	From Table III and Eq. (11)	% Dif
2	68.96	69.40	0.6	71.10	68.73	-3.3	72.17	68.55	-5.0
3	59.38	56.27	-5.2	59.91	55.93	-6.6	60.48	55.84	-7.7
4	52.13	48.67	-6.6	51.76	48.13	-7.0	52.05	48.01	-7.8
5	46.46	42.43	-8.7	45.56	42.53	-6.6	45.69	42.43	-7.1
6	41.91	38.20	-8.9	40.69	38.37	-5.7	40.71	38.27	-6.0
7	38.16	34.87	-8.6	36.76	35.10	-4.5	36.71	35.00	-4.7
8	35.03	32.23	-8.0	33.52	32.50	-3.0	33.42	32.41	-3.0

[a] Nitzki and Matzke, Phys. Rev. B8, 1894 (1973).

$^{NAT}O(\alpha,n)$ cross-section plot of Ref. 16 and to data taken from the $^{17,18}O(\alpha,n)$ cross-section plots of Ref. 17. These five cross-section descriptions are shown in Fig. 2.

The $^{17,18}O(\alpha,n)$ cross sections used in the present calculations were composed of the lower energy data of Refs. 14 and 15 increased by 35% as recommended in Ref. 16 and joined with the adjusted higher energy data of Ref. 17. This adjustment, amounting to a 9.2% reduction, was determined by normalizing the integral of the $^{NAT}O(\alpha,n)$ cross section formed from the functional fits to $^{17,18}O(\alpha,n)$ cross sections of Ref. 17 to the integral of the $^{NAT}O(\alpha,n)$ cross section of Ref. 16 over the range 5.15 MeV $\leq E_\alpha \leq$ 8 MeV. The resulting adjusted cross sections are shown in Fig. 3. The adjusted $^{17}O(\alpha,n)$ cross section is given in Table V, and the adjusted $^{18}O(\alpha,n)$ cross section is given in Table VI; cross sections are defined there by interpolation points at low energies ($\lesssim$ 5 MeV) and by polynomial functions at higher energies.

C. Alpha-Decay Data

A total of 144 actinide nuclides produced in reactor fuel have been identified,[22] using data of ENDF/B-V and Refs. 23-25. Of these, 89 decay at least partly by alpha decay. Each nuclide has some L different alpha-particle energies with $1 \leq L \leq 26$ for the data collection used. Alpha-particle energies in the data collection fall in the range 3.71 MeV $\leq E_\alpha \leq$ 8.78 MeV. TABLE VII lists the alpha-particle energies and intensities for each nuclide.

D. Spontaneous-Fission Data

Of the 144 actinide nuclides identified, 40 decay at least partly by spontaneous fission. Values of $\bar{\nu}_p(SF)$, the major prompt contribution to $\bar{\nu}(SF)$, are given by Manero and Konshin[26] for many of these. These values were used in Fig. 4 to estimate values of $\bar{\nu}_p(SF)$ for nuclides without data.

Branching fractions F^{SF}, if not given in a data reference, were constructed from total and SF half-life values $T_{1/2}(SF)$ using Eq. (19). Values of $T_{1/2}(SF)$ given as limiting values were used and quoted without qualification. The values of $\bar{\nu}(SF)$, F^{SF}, and R(SF) for each of the 40 nuclides are given in Table VIII.

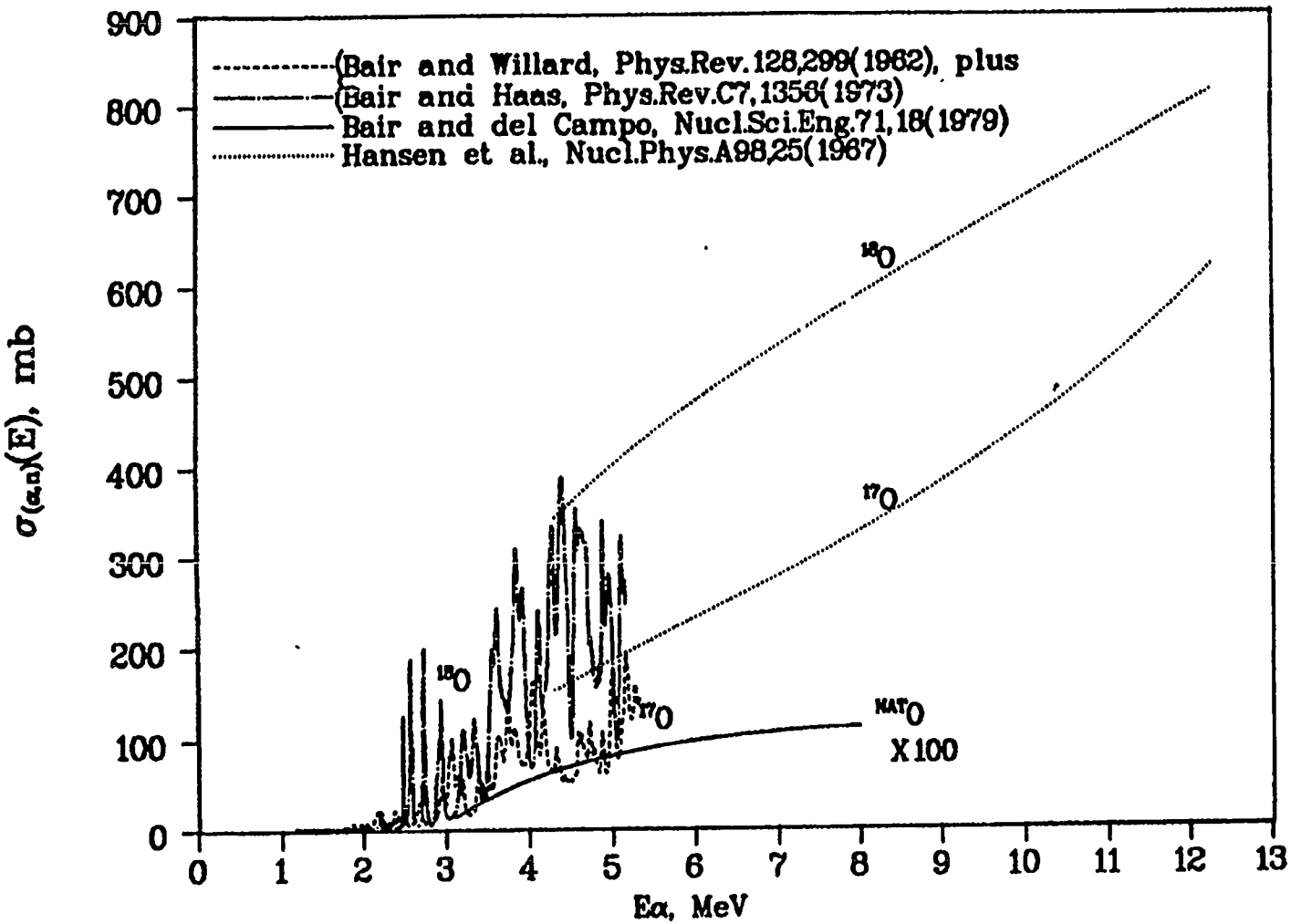


Fig. 2.
^{17}O, ^{18}O, and ^{NAT}O (α,n) cross-section data.

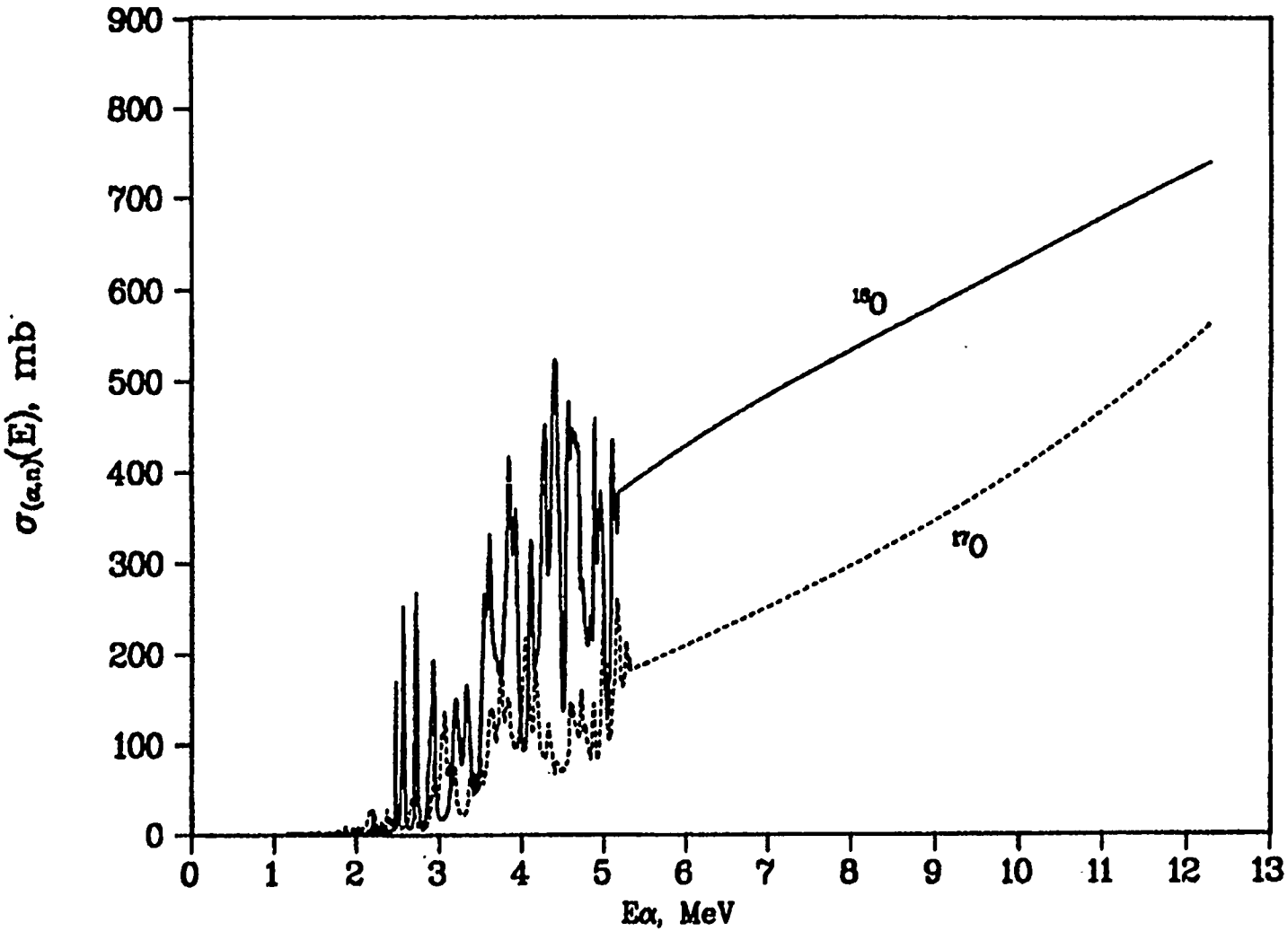


Fig. 3.
^{17}O and ^{18}O adjusted (α,n) cross sections.

TABLE V

ADJUSTED ^{17}O (α,n) CROSS SECTION

E(MEV)	CX(MB)	E(MEV)	CX(MB)	E(MEV)	CX(MB)	E(MEV)	CX(MB)	E(MEV)	CX(MB)	E(MEV)	CX(MB)	E(MEV)	CX(MB)	E(MEV)	CX(MB)	E(MEV)	CX(MB)	E(MEV)	CX(MB)
1.3117	.03	1.5885	.04	1.8131	.94	2.1579	25.10	2.4845	13.61	2.8901	23.54	3.3322	27.73	3.7914	127.44	4.2865	85.87	4.8113	93.77
1.3139	.03	1.5927	.12	1.8177	.78	2.1628	26.32	2.4901	18.44	2.8960	38.77	3.3363	28.88	3.7978	131.57	4.2933	90.84	4.8184	91.62
1.3158	0.00	1.5968	.13	1.8221	.61	2.1677	26.23	2.4951	20.25	2.9016	45.71	3.3444	32.44	3.8041	140.13	4.3002	96.50	4.8260	85.17
1.3178	.12	1.6007	.16	1.8265	.73	2.1726	25.62	2.5005	18.22	2.9072	44.74	3.3503	36.07	3.8105	144.46	4.3070	100.64	4.8331	84.66
1.3195	.12	1.6051	.22	1.8311	.62	2.1775	25.09	2.5056	14.93	2.9128	47.46	3.3564	41.15	3.8171	150.16	4.3139	120.84	4.8401	84.15
1.3214	.03	1.6094	.26	1.8356	.68	2.1824	25.23	2.5110	11.81	2.9241	47.80	3.3625	51.87	3.8234	150.04	4.3207	121.73	4.8475	96.54
1.3234	.04	1.6135	.13	1.8400	2.07	2.1872	27.24	2.5161	9.09	2.9298	66.26	3.3686	64.62	3.8288	143.59	4.3276	113.21	4.8545	118.76
1.3250	.07	1.6177	.26	1.8447	1.36	2.1921	26.45	2.5212	7.99	2.9354	83.75	3.3744	63.36	3.8361	145.39	4.3345	105.07	4.8621	135.24
1.3272	0.00	1.6218	.49	1.8490	1.34	2.1970	25.34	2.5266	7.01	2.9464	71.05	3.3805	60.05	3.8427	141.48	4.3413	102.42	4.8698	144.37
1.3289	.03	1.6232	.26	1.8533	1.46	2.2019	23.35	2.5317	6.98	2.9522	55.97	3.3867	58.85	3.8491	135.55	4.3481	101.61	4.8765	133.54
1.3306	0.00	1.6246	.31	1.8580	1.30	2.2068	22.38	2.5371	6.52	2.9582	46.58	3.3925	56.66	3.8554	124.56	4.3550	94.84	4.8838	125.50
1.3328	.03	1.6260	.39	1.8625	1.46	2.2117	21.25	2.5422	7.57	2.9638	39.77	3.3986	61.05	3.8622	118.17	4.3618	92.32	4.8913	110.55
1.3345	.01	1.6301	.26	1.8671	1.32	2.2169	23.67	2.5474	8.64	2.9694	35.38	3.4047	67.86	3.8683	109.98	4.3686	86.66	4.8981	94.06
1.3366	.07	1.6304	.53	1.8715	1.42	2.2217	19.08	2.5531	8.99	2.9753	33.84	3.4108	70.08	3.8749	109.59	4.3755	84.38	4.9054	88.82
1.3385	.03	1.6343	.38	1.8761	.99	2.2265	16.21	2.5582	8.72	2.9809	35.05	3.4169	69.48	3.8813	105.83	4.3823	78.26	4.9127	83.89
1.3424	.03	1.6345	.43	1.8805	1.32	2.2314	14.46	2.5636	8.91	2.9846	35.65	3.4230	65.08	3.8879	102.55	4.3891	74.88	4.9203	84.62
1.3441	0.00	1.6385	.43	1.8852	1.16	2.2363	12.96	2.5689	8.06	2.9883	38.91	3.4291	64.44	3.8942	102.10	4.3962	68.86	4.9276	85.16
1.3461	.03	1.6388	.58	1.8899	1.44	2.2414	11.87	2.5741	8.72	2.9921	41.09	3.4352	62.23	3.9008	100.10	4.4033	68.67	4.9350	95.01
1.3498	.01	1.6426	.35	1.8943	.93	2.2463	9.79	2.5792	8.30	2.9981	48.18	3.4413	59.79	3.9072	96.74	4.4104	76.87	4.9418	104.46
1.3537	.04	1.6431	.32	1.8987	1.17	2.2512	8.40	2.5846	8.84	3.0037	54.88	3.4474	54.86	3.9138	96.88	4.4170	80.24	4.9491	120.67
1.3557	.04	1.6467	.38	1.9036	.92	2.2564	5.94	2.5897	10.07	3.0093	65.73	3.4535	55.12	3.9201	100.08	4.4231	80.81	4.9567	134.85
1.3576	.04	1.6475	.34	1.9079	1.03	2.2613	5.08	2.5951	10.79	3.0152	77.73	3.4596	52.54	3.9267	97.51	4.4307	79.43	4.9638	151.05
1.3618	.01	1.6509	.34	1.9125	1.42	2.2662	3.73	2.6004	12.30	3.0208	95.28	3.4658	51.58	3.9335	99.37	4.4380	77.36	4.9711	176.05
1.3655	0.00	1.6516	.45	1.9172	1.15	2.2713	3.56	2.6058	12.31	3.0264	107.08	3.4721	50.73	3.9399	99.35	4.4443	76.44	4.9784	202.89
1.3691	.01	1.6550	.35	1.9217	.94	2.2763	3.23	2.6112	14.49	3.0322	110.11	3.4780	50.23	3.9462	96.92	4.4516	73.79	4.9857	222.25
1.3730	.04	1.6558	.42	1.9265	1.28	2.2814	2.93	2.6167	15.15	3.0381	109.03	3.4841	52.61	3.9528	100.95	4.4585	71.23	4.9931	235.94
1.3770	.07	1.6592	.45	1.9309	1.01	2.2867	2.52	2.6218	17.01	3.0441	105.07	3.4904	60.47	3.9594	104.52	4.4653	73.02	5.0004	239.31
1.3808	.07	1.6602	.32	1.9356	1.58	2.2810	3.69	2.6272	18.01	3.0494	107.51	3.4963	75.34	3.9660	113.68	4.4725	71.05	5.0082	228.16
1.3845	.05	1.6633	.43	1.9402	2.07	2.2861	4.85	2.6325	18.44	3.0553	116.58	3.5026	85.10	3.9726	118.00	4.4793	70.31	5.0154	209.80
1.3884	.05	1.6644	.53	1.9448	2.74	2.2908	6.82	2.6379	19.90	3.0608	127.87	3.5087	98.33	3.9792	125.35	4.4864	74.10	5.0225	183.94
1.3923	.04	1.6678	.38	1.9496	4.31	2.2912	9.77	2.6433	22.55	3.0670	136.27	3.5148	70.31	3.9855	129.42	4.4933	73.22	5.0298	153.32
1.3962	.03	1.6688	.42	1.9542	5.59	2.2957	12.60	2.6486	27.70	3.0729	129.71	3.5209	64.14	3.9921	137.71	4.5003	74.06	5.0374	131.42
1.4001	.05	1.6720	.43	1.9585	7.13	2.2952	13.00	2.6540	32.39	3.0785	116.45	3.5272	59.20	3.9987	149.39	4.5069	71.25	5.0447	117.50
1.4041	.04	1.6732	.51	1.9634	8.59	2.3006	13.43	2.6594	35.53	3.0842	103.87	3.5334	57.54	4.0053	161.20	4.5140	77.10	5.0522	106.84
1.4080	0.00	1.6761	.38	1.9678	7.28	2.3011	13.35	2.6647	37.83	3.0901	87.45	3.5397	58.90	4.0119	178.00	4.5211	76.77	5.0593	104.95
1.4120	.01	1.6774	.41	1.9727	7.03	2.3055	12.50	2.6701	40.45	3.0959	74.40	3.5458	60.16	4.0185	188.04	4.5270	77.37	5.0669	106.71
1.4198	.01	1.6805	.45	1.9774	6.87	2.3052	12.01	2.6756	39.57	3.1018	68.81	3.5519	63.75	4.0251	202.23	4.5348	78.24	5.0740	119.38
1.4237	.03	1.6818	.42	1.9818	6.21	2.3093	10.73	2.6810	36.45	3.1074	65.07	3.5580	69.19	4.0318	211.88	4.5421	78.62	5.0813	119.55
1.4276	.05	1.6847	.36	1.9867	5.32	2.3124	11.11	2.6863	32.29	3.1135	70.04	3.5644	73.74	4.0384	217.76	4.5492	79.91	5.0886	132.33
1.4314	.07	1.6850	.45	1.9913	4.62	2.3155	9.88	2.6914	28.62	3.1191	70.69	3.5707	76.45	4.0452	210.78	4.5558	84.36	5.0962	145.02
1.4354	.35	1.6888	.43	1.9960	4.32	2.3208	9.00	2.6971	25.56	3.1251	77.18	3.5768	85.72	4.0515	202.66	4.5628	89.30	5.1038	154.62
1.4393	.47	1.6903	.47	2.0006	4.36	2.3214	8.55	2.7027	23.38	3.1309	77.19	3.5832	96.11	4.0581	179.28	4.5697	103.94	5.1111	161.04
1.4432	.82	1.6933	.42	2.0053	4.56	2.3214	8.18	2.7081	21.22	3.1368	80.70	3.5893	107.62	4.0650	166.52	4.5767	131.49	5.1184	166.01
1.4474	.90	1.6944	.49	2.0100	4.68	2.3258	7.18	2.7138	19.24	3.1427	79.14	3.5958	119.69	4.0716	161.81	4.5841	144.41	5.1257	167.21
1.4513	.80	1.6974	.52	2.0148	6.55	2.3264	7.44	2.7188	18.24	3.1485	77.18	3.6017	127.80	4.0782	159.50	4.5907	137.20	5.1336	174.83
1.4549	.74	1.6978	.92	2.0194	7.87	2.3309	6.70	2.7244	16.80	3.1544	70.96	3.6081	134.97	4.0847	151.15	4.5977	129.73	5.1407	208.03
1.4592	.63	1.6988	.53	2.0241	9.21	2.3359	6.56	2.7298	12.78	3.1602	66.60	3.6144	136.82	4.0913	142.55	4.6051	132.31	5.1483	246.55
1.4631	.54	1.7015	.53	2.0287	8.29	2.3410	5.82	2.7352	10.92	3.1664	60.01	3.6205	135.01	4.0979	131.19	4.6119	142.13	5.1556	280.45
1.4670	.43	1.7032	.54	2.0336	7.56	2.3461	5.33	2.7409	9.87	3.1721	54.85	3.6269	137.09	4.1045	122.88	4.6193	138.24	5.1629	252.30
1.4709	.30	1.7059	.49	2.0382	6.56	2.3510	5.71	2.7462	8.61	3.1782	54.74	3.6330	136.92	4.1114	119.58	4.6262	131.38	5.1703	236.90
1.4751	.22	1.7076	.65	2.0428	5.58	2.3564	7.76	2.7516	8.45	3.1838	65.45	3.6393	139.31	4.1180	112.45	4.6332	125.47	5.1778	225.05
1.4790	.22	1.7103	.58	2.0477	5.24	2.3613	17.73	2.7572	8.07	3.1896	76.59	3.6454	135.31	4.1248	111.51	4.6403	123.11	5.1854	214.12
1.4829	.12	1.7118	.62	2.0525	5.71	2.3664	25.93	2.7626	7.51	3.1955	74.64	3.6518	131.00	4.1314	116.06	4.6474	122.20	5.1927	194.70
1.4870	.38	1.7145	.53	2.0574	5.64	2.3715	28.01	2.7680	7.01	3.2013	63.77	3.6581	119.88	4.1380	128.17	4.6545	116.35	5.2005	180.64
1.4909	.09	1.7162	.53	2.0620	5.72	2.3765	26.00	2.7736	5.95	3.2072	51.54	3.6645	114.99	4.1451	152.50	4.6618	114.60	5.2076	164.64
1.4948	.07	1.7207	.69	2.0668	6.13	2.3816	19.06	2.7792	5.53	3.2133	41.54	3.6708	110.74	4.1516	169.80	4.6686	114.43	5.2149	166.06
1.5029	.07	1.7251	.74	2.0714	6.28	2.3865	14.51	2.7846	5.42	3.2192	32.30	3.6772	103.19	4.1580	176.36	4.6760	111.55	5.2225	163.35
1.5069	.09	1.7292	.74	2.0739	6.17	2.3919	11.11	2.7899	5.70	3.2253	32.17	3.6833	102.49	4.1648	180.47	4.6833	109.48	5.2301	165.66
1.5108	.09	1.7336	.90	2.0763	6.06	2.3970	10.26	2.7957	6.88	3.2309	29.86	3.6875	112.01	4.1714	168.67	4.6901	105.91	5.2376	172.72
1.5149	.08	1.7380	.82	2.0809	6.76	2.4021	16.43	2.8010	6.24	3.2367	27.67	3.6917	121.05	4.1788	160.76	4.6972	116.56	5.2450	173.37
1.5189	.08	1.7424	1.20	2.0858	7.24	2.4070	19.94	2.8067	5.47	3.2426	26.32	3.6960	120.99	4.1852	145.38	4.7045	132.52	5.2528	189.23
1.5230	.08	1.7467	1.32	2.0905	7.57	2.4122	19.57	2.8123	5.43	3.2487	24.96	3.7023	119.87	4.1918	136.30	4.7118	150.46	5.2600	208.11
1.5272	.08	1.7512	1.40	2.0954	8.06	2.4175	17.44	2.8176	6.71	3.2546	24.30	3.7087	121.41	4.1986	122.00	4.7185	159.02	5.2680	214.07
1.5311	.30	1.7556	1.74	2.1001	8.22	2.4226	14.94	2.8235	8.50	3.2604	24.42	3.7150	128.91	4.2055	110.67	4.7255	155.11	5.2751	206.04
1.5353	.12	1.7600	2.04	2.1050	9.94	2.4278	10.49	2.8286	9.60	3.2665	22.63	3.7214	149.61	4.2120	104.89	4.7331	136.97	5.2824	188.14
1.5391	.04	1.7644	2.04	2.1095	10.08	2.4330	7.10	2.8345	10.41	3.2724	22.67	3.7277	162.36	4.2189	97.24	4.7399	120.33	5.2902	180.80
1.5433	.07	1.7688	1.85	2.1145	11.77	2.4381	5.17	2.8400	10.71	3.2785	23.02	3.7341	170.86	4.2255	94.54	4.7470	112.29	5.2978	182.54
1.5513	.07	1.7732	1.97	2.1193	11.11	2.4435	5.63	2.8456	9.60	3.2844	21.79	3.7404	183.57	4.2323	92.29	4.7541	117.13	5.3051	183.41
1.5556	.09	1.7777	2.98	2.1243	14.08	2.4488	5.01	2.8512	9.10	3.2902	21.92	3.7468	185.79	4.2389	93.20	4.7614	119.41	5.3127	194.08
1.5597	.15	1.7821	2.79	2.1290	15.21	2.4537	5.82	2.8568	12.85	3.2963	21.80	3.7531	174.08	4.2457	91.51	4.7686	122.00		
1.5639	.05	1.7864	2.71	2.1338	15.71	2.4586	6.43	2.8622	18.91	3.3024	22.79	3.7595	153.98	4.2526	90.67	4.7757	116.05		
1.5678	.05	1.7908	2.50	2.1387	16.72	2.4640	6.97	2.8678	21.63	3.3083	23.22	3.7660	139.25	4.2597	87.61	4.7828	110.31		
1.5719	.15	1.7952	2.31	2.1435	20.56	2.4691	6.76	2.8734	20.05	3.3142	24.07	3.7721	129.51	4.2660	88.37	4.7901	104.72		
1.5761	.09	1.7998	1.66	2.1484	22.98	2.4743	8.88	2.8791	20.14	3.3202	24.72	3.7787	128.44	4.2728	87.74	4.7972	101.05		
1.5802	.09	1.8042	1.29	2.1531	24.87	2.4795	10.87	2.8847	19.05	3.3263	26.01	3.7851	125.04	4.2799	88.80	4.8045	96.05		
1.5845	.16	1.8087	1.16																

CX(mb) = 0 (assumed), E(MeV) < 1.31,
= −126.76 + 89.624E − 9.0873E^2 + 0.647207E^3 − 0.0105365E^4, 5.3127≤E(MeV)≤12.3.

TABLE VI

ADJUSTED ^{18}O (α,n) CROSS SECTION

E(MeV)	CX(mb)	E(MeV)	CX(mb)	E(MeV)	CX(mb)	E(MeV)	CX(mb)	E(MeV)	CX(mb)	E(MeV)	CX(mb)	E(MeV)	CX(mb)	E(MeV)	CX(mb)	E(MeV)	CX(mb)	E(MeV)	CX(mb)
1.1448	0.00	1.5432	.68	1.8707	4.18	2.1957	27.20	2.5100	15.32	2.9240	168.47	3.3100	119.76	3.7240	177.86	4.1944	210.40	4.6848	413.82
1.1482	.07	1.5474	.55	1.8751	1.48	2.2004	22.61	2.5140	17.46	2.9261	187.69	3.3180	135.78	3.7401	177.88	4.1984	215.74	4.6888	390.33
1.1516	.05	1.5515	.45	1.8797	.66	2.2048	10.89	2.5180	22.80	2.9281	193.04	3.3234	156.07	3.7441	184.29	4.2064	230.70	4.6928	352.90
1.1553	.07	1.5554	.38	1.8841	.43	2.2097	3.83	2.5261	29.21	2.9306	190.90	3.3287	164.62	3.7522	192.83	4.2104	246.73	4.7006	305.98
1.1590	.05	1.5596	.31	1.8885	.34	2.2143	2.38	2.5301	37.77	2.9331	168.48	3.3341	163.57	3.7602	203.53	4.2225	280.91	4.7089	271.81
1.1625	.05	1.5637	.23	1.8936	.31	2.2189	1.94	2.5341	46.32	2.9356	163.13	3.3381	157.15	3.7682	209.94	4.2292	316.16	4.7170	275.02
1.1661	.01	1.5679	.22	1.8973	.23	2.2236	1.70	2.5381	61.26	2.9381	142.84	3.3461	148.62	3.7736	223.83	4.2359	366.35	4.7250	282.50
1.1696	0.00	1.5719	.19	1.9019	.23	2.2287	1.62	2.5421	69.82	2.9406	117.22	3.3502	140.09	3.7790	239.85	4.2426	396.63	4.7296	276.54
1.1747	0.00	1.5760	.19	1.9064	.22	2.2334	1.62	2.5462	79.43	2.9431	114.02	3.3542	128.34	3.7843	251.60	4.2547	404.81	4.7371	271.84
1.1982	.13	1.5802	.16	1.9108	.18	2.2383	1.65	2.5542	124.29	2.9457	90.53	3.3622	119.81	3.7964	264.44	4.2587	412.29	4.7411	266.50
1.2019	0.00	1.5843	.16	1.9152	.19	2.2430	1.70	2.5582	215.06	2.9482	74.52	3.3662	114.48	3.8004	279.38	4.2627	425.10	4.7511	251.57
1.2056	.01	1.5885	.13	1.9196	.19	2.2478	1.80	2.5636	252.45	2.9522	54.23	3.3703	103.80	3.8044	299.69	4.2748	450.75	4.7572	249.83
1.2092	0.00	1.5926	.11	1.9242	.18	2.2525	1.86	2.5689	191.58	2.9602	46.76	3.3756	97.40	3.8125	322.11	4.2801	446.42	4.7612	235.43
1.2640	0.00	1.5965	.16	1.9286	.19	2.2574	2.55	2.5743	100.82	2.9642	36.10	3.3810	90.99	3.8205	349.88	4.2855	424.06	4.7692	223.82
1.2678	.01	1.6009	.11	1.9330	.18	2.2620	2.78	2.5823	57.04	2.9683	30.75	3.3863	81.38	3.8285	376.58	4.2908	377.08	4.7773	220.63
1.2826	.01	1.6051	.09	1.9377	.19	2.2669	2.71	2.5864	35.69	2.9723	26.49	3.3904	69.65	3.8326	399.02	4.2949	337.58	4.7813	213.15
1.2863	0.00	1.6093	.09	1.9423	.23	2.2715	2.19	2.5904	25.02	2.9763	24.35	3.3971	57.91	3.8366	415.04	4.3029	303.41	4.7895	211.03
1.3013	0.00	1.6135	.11	1.9467	.57	2.2764	1.97	2.5944	20.75	2.9843	22.23	3.4036	51.52	3.8406	416.11	4.3109	293.81	4.7974	209.97
1.3052	.01	1.6176	.09	1.9512	1.39	2.2810	2.02	2.5984	15.42	2.9924	20.10	3.4105	45.12	3.8486	400.10	4.3150	287.41	4.8014	212.11
1.3089	.01	1.6215	.08	1.9556	1.42	2.2859	1.98	2.6024	13.28	2.9964	19.05	3.4185	47.25	3.8527	373.41	4.3230	301.31	4.8054	217.46
1.3125	.03	1.6340	.08	1.9602	1.26	2.2906	2.13	2.6078	11.15	3.0044	17.98	3.4225	49.40	3.8567	353.13	4.3310	316.26	4.8175	227.07
1.3164	.16	1.6381	.07	1.9646	.57	2.2955	2.25	2.6132	9.02	3.0165	18.00	3.4265	57.55	3.8687	352.86	4.3391	349.37	4.8255	218.04
1.3201	.43	1.6423	.08	1.9693	.30	2.3002	2.34	2.6185	7.97	3.0286	16.94	3.4346	64.35	3.8728	314.70	4.3431	364.52	4.8335	218.55
1.3238	.18	1.6464	.07	1.9739	.30	2.3051	2.47	2.6268	6.90	3.0326	16.94	3.4386	69.41	3.8808	310.45	4.3511	394.23	4.8389	215.35
1.3277	.15	1.6506	.07	1.9785	.27	2.3097	2.73	2.6386	6.91	3.0446	16.96	3.4453	67.29	3.8848	308.31	4.3592	433.75	4.8445	215.35
1.3316	.07	1.6590	.08	1.9829	.26	2.3148	3.05	2.6467	6.93	3.0487	18.04	3.4520	62.42	3.8929	320.08	4.3672	482.98	4.8496	217.50
1.3353	.01	1.6631	.08	1.9876	.30	2.3195	3.20	2.6547	6.93	3.0567	19.10	3.4587	64.77	3.8969	336.08	4.3753	485.02	4.8577	249.55
1.3390	0.00	1.6675	.07	1.9921	.34	2.3244	6.70	2.6627	8.01	3.0688	20.18	3.4667	69.12	3.9009	340.36	4.3793	497.84	4.8667	325.28
1.3466	0.00	1.6841	.07	1.9967	.35	2.3290	9.46	2.6681	10.15	3.0728	21.26	3.4708	66.53	3.9130	357.45	4.3873	521.34	4.8737	438.57
1.3505	.01	1.6885	.09	2.0013	.34	2.3344	10.98	2.6735	11.22	3.0848	22.34	3.4748	75.07	3.9170	357.47	4.3994	522.42	4.8818	457.80
1.3544	0.00	1.6930	.05	2.0055	.41	2.3388	9.71	2.6788	14.43	3.0929	23.42	3.4868	86.85	3.9210	331.84	4.4034	517.04	4.8858	385.20
1.3583	0.00	1.6972	.07	2.0104	.43	2.3436	7.49	2.6869	15.51	3.0969	25.56	3.4909	99.66	3.9331	308.35	4.4074	504.28	4.8938	337.15
1.3619	.01	1.7013	.07	2.0148	.51	2.3485	6.10	2.6949	25.12	3.1009	27.70	3.4989	114.62	3.9371	273.12	4.4114	487.20	4.8979	302.98
1.3659	0.00	1.7057	.08	2.0284	.51	2.3533	4.83	2.6989	36.08	3.1049	29.84	3.5029	129.57	3.9491	235.76	4.4195	454.45	4.9059	295.78
1.3696	.01	1.7096	.09	2.0331	.49	2.3581	4.23	2.7029	64.65	3.1090	30.91	3.5086	152.01	3.9532	208.01	4.4275	429.56	4.9180	305.14
1.3811	.01	1.7140	.07	2.0377	.58	2.3630	4.05	2.7070	141.53	3.1130	30.91	3.5142	166.95	3.9612	175.97	4.4315	398.56	4.9260	325.43
1.3850	0.00	1.7228	.07	2.0424	.62	2.3679	3.89	2.7190	245.11	3.1170	26.25	3.5198	178.78	3.9652	161.03	4.4396	376.18	4.9300	343.56
1.4007	0.00	1.7270	.09	2.0470	.65	2.3728	3.58	2.7271	267.57	3.1210	41.59	3.5254	196.86	3.9733	141.82	4.4436	362.30	4.9421	363.89
1.4046	.01	1.7311	.08	2.0515	.63	2.3777	3.25	2.7311	188.55	3.1291	46.95	3.5311	210.76	3.9813	129.01	4.4516	330.26	4.9501	377.78
1.4083	0.00	1.7353	.08	2.0561	.68	2.3826	3.09	2.7351	126.62	3.1331	49.09	3.5331	220.37	3.9853	112.99	4.4597	305.72	4.9582	389.25
1.4122	0.00	1.7398	.13	2.0608	.69	2.3877	3.10	2.7391	78.57	3.1411	58.71	3.5351	237.45	3.9893	105.53	4.4637	272.62	4.9622	361.79
1.4161	.01	1.7442	.07	2.0654	.73	2.3926	3.04	2.7431	56.15	3.1492	68.32	3.5471	266.30	3.9974	101.26	4.4717	240.60	4.9702	351.11
1.4239	.01	1.7483	.08	2.0700	.74	2.3934	4.52	2.7458	43.33	3.1572	76.88	3.5512	256.70	4.0054	95.93	4.4756	200.02	4.9783	323.35
1.4281	.16	1.7527	.09	2.0747	.80	2.4014	5.46	2.7485	35.86	3.1612	84.36	3.5592	248.16	4.0094	93.86	4.4836	169.06	4.9863	306.27
1.4320	.01	1.7571	.08	2.0791	.96	2.4095	5.47	2.7512	30.52	3.1652	96.11	3.5632	242.82	4.0175	93.81	4.4918	145.57	4.9903	294.53
1.4360	.03	1.7612	.09	2.0840	.84	2.4135	5.47	2.7579	27.34	3.1693	110.00	3.5672	246.04	4.0255	94.89	4.4959	135.57	4.9984	242.22
1.4399	.15	1.7654	.08	2.0884	.94	2.4215	5.55	2.7646	21.89	3.1733	118.54	3.5753	255.65	4.0335	95.96	5.0079	141.32	5.0064	187.77
1.4438	.26	1.7700	.09	2.0930	1.04	2.4276	6.70	2.7713	14.53	3.1813	128.16	3.5833	270.62	4.0456	101.32	5.0160	168.02	5.0104	166.41
1.4477	.24	1.7787	.09	2.0976	1.07	2.4316	6.71	2.7753	10.26	3.1894	142.05	3.5913	285.57	4.0496	113.02	5.0200	238.50	5.0185	155.75
1.4516	.24	1.7828	.11	2.1023	1.15	2.4397	7.78	2.7833	10.27	3.1974	147.39	3.5994	312.28	4.0577	121.62	5.0280	317.23	5.0305	144.00
1.4555	.19	1.7870	.11	2.1069	1.36	2.4517	12.06	2.7914	10.29	3.2014	149.54	4.0038	330.44	4.0657	142.98	5.0361	362.68	5.0386	132.27
1.4594	.08	1.7914	.09	2.1117	1.47	2.4537	14.19	2.7994	11.35	3.2054	147.41	4.0074	319.76	4.0738	172.89	5.0441	419.00	5.0506	137.62
1.4633	.05	1.7958	.08	2.1160	1.62	2.4557	16.34	2.8075	12.43	3.2095	142.07	4.0155	302.68	4.0818	219.89	5.0521	452.11	5.0587	152.59
1.4790	.05	1.8004	.09	2.1207	1.97	2.4577	18.47	2.8155	13.51	3.2148	130.34	4.0195	270.65	4.0858	246.59	5.0602	476.68	5.0667	177.15
1.4832	.08	1.8046	.11	2.1256	2.17	2.4617	29.95	2.8195	15.65	3.2202	121.78	4.0275	246.11	4.0939	282.91	5.0682	455.04	5.0747	217.74
1.4871	.08	1.8087	.15	2.1300	2.52	2.4648	38.77	2.8316	16.73	3.2255	129.73	4.0316	233.29	4.0979	312.81	5.0763	413.69	5.0826	280.22
1.4910	.11	1.8131	.15	2.1348	3.01	2.4678	94.30	2.8356	19.94	3.2336	106.65	4.0396	222.61	4.1059	323.49	5.0843	426.52	5.0868	381.13
1.4952	.16	1.8177	.22	2.1392	3.51	2.4708	108.18	2.8436	23.15	3.2376	109.00	4.0436	213.02	4.1140	309.62	5.0950	432.93	5.0989	434.54
1.4991	.19	1.8220	.18	2.1441	4.06	2.4738	169.05	2.8597	37.04	3.2456	100.47	3.6517	210.88	4.1220	274.39	5.0017	445.81	5.1066	424.94
1.5031	.26	1.8266	.19	2.1488	4.47	2.4754	148.77	2.8637	65.88	3.2497	94.07	3.6597	205.55	4.1260	237.02	4.6084	443.62	5.1123	391.84
1.5072	.34	1.8308	.24	2.1534	4.67	2.4770	94.31	2.8798	81.92	3.2537	88.72	3.6637	197.02	4.1300	210.33	4.6124	436.16	5.1176	366.21
1.5111	.51	1.8352	.24	2.1580	5.09	2.4786	47.32	2.8838	95.81	3.2577	85.52	3.6718	197.02	4.1381	192.19	4.6205	439.37	5.1230	347.00
1.5150	.68	1.8396	.35	2.1624	4.87	2.4802	25.97	2.8879	104.36	3.2617	83.39	3.6798	194.90	4.1421	181.51	4.6285	441.52	5.1310	353.42
1.5192	.85	1.8440	.49	2.1673	4.58	2.4818	15.30	2.8999	126.79	3.2698	80.20	3.6878	193.83	4.1501	185.79	4.6366	441.52	5.1391	367.31
1.5231	1.15	1.8484	.80	2.1721	4.40	2.4839	15.30	2.9120	140.68	3.2738	79.14	3.6959	192.78	4.1582	194.35	4.6446	430.85	5.1471	363.04
1.5272	1.24	1.8528	2.63	2.1769	4.25	2.4859	11.03	2.9160	148.16	3.2778	82.35	3.6999	187.45	4.1622	192.21	4.6486	426.54	5.1511	332.09
1.5312	1.20	1.8574	7.21	2.1816	4.48	2.4899	9.96	2.9180	149.24	3.2858	85.56	3.7039	181.03	4.1702	192.21	4.6526	427.65		
1.5353	.92	1.8618	8.56	2.1862	6.51	2.4930	11.03	2.9200	153.51	3.2939	91.98	3.7160	183.19	4.1743	198.63	4.6567	428.75		
1.5393	.85	1.8663	7.69	2.1909	24.25	2.5019	13.18	2.9220	156.72	3.2979	103.72	3.7200	181.00	4.1823	200.77	4.6727	429.81		

CX(mb) = 0 (assumed), Threshold = $0.85 \leq E(MeV) < 1.14$,

$= -320.68 + 241.227E - 29.6216E^2 + 2.011956E^3 - 0.0511101E^4$, $5.1511 \leq E(MeV) \leq 12.3$.

TABLE VII

ALPHA DECAY SPECTRA OF ACTINIDE NUCLIDES

82-PB-210
1 ALPHA, REF B

E,MEV.	DK FRACTION
3.7198	1.70000E-08

83-BI-210
2 ALPHAS, REF B

E,MEV.	DK FRACTION
4.6479	7.80000E-07
4.6861	5.20000E-07

84-PO-210
2 ALPHAS, REF B

E,MEV.	DK FRACTION
4.5168	1.07000E-05
5.3046	9.99989E-01

83-BI-211
2 ALPHAS, REF B

E,MEV.	DK FRACTION
6.2790	1.59562E-01
6.6233	8.37698E-01

84-PO-211
3 ALPHAS, REF B

E,MEV.	DK FRACTION
6.5694	5.40000E-03
6.8914	5.50000E-03
7.4502	9.89100E-01

83-BI-212
8 ALPHAS, REF B

E,MEV.	DK FRACTION
5.3024	3.95857E-07
5.3456	5.59870E-06
5.4849	5.39805E-05
5.6069	3.95857E-03
5.6258	5.39805E-04
5.7684	6.00983E-03
6.0511	2.51589E-01
6.0902	9.78847E-02

84-PO-212
1 ALPHA, REF B

E,MEV.	DK FRACTION
8.7846	1.00000E+00

83-BI-213
2 ALPHAS, REF B

E,MEV.	DK FRACTION
5.5508	1.62963E-03
5.8687	2.03704E-02

84-PO-213
2 ALPHAS, REF B

E,MEV.	DK FRACTION
7.6123	3.00000E-05
8.3757	9.99970E-01

83-BI-214
6 ALPHAS, REF B

E,MEV.	DK FRACTION
4.9420	5.25210E-07
5.0234	4.41176E-07
5.1824	1.26050E-06
5.2638	1.21849E-05
5.4508	1.13235E-04
5.5121	8.23529E-05

84-PO-214
2 ALPHAS, REF B

E,MEV.	DK FRACTION
6.9025	1.00000E-04
7.6873	9.99900E-01

84-PO-215
3 ALPHAS, REF B

E,MEV.	DK FRACTION
6.9497	2.20000E-04
6.9559	3.40000E-04
7.3865	9.99440E-01

85-AT-215
2 ALPHAS, REF B

E,MEV.	DK FRACTION
7.6285	5.00000E-04
8.0258	9.99500E-01

84-PO-216
2 ALPHAS, REF A

E,MEV.	DK FRACTION
5.9850	2.10000E-05
6.7785	9.99979E-01

85-AT-217
4 ALPHAS, REF B

E,MEV.	DK FRACTION
6.4846	4.00100E-04
6.6113	1.00025E-04
6.8134	2.50063E-04
7.0677	9.99250E-01

86-RN-217
1 ALPHA, REF B

E,MEV.	DK FRACTION
7.7426	1.00000E+00

84-PO-218
2 ALPHAS, REF B

E,MEV.	DK FRACTION
5.1810	1.10000E-05
6.0027	9.99989E-01

85-AT-218
3 ALPHAS, REF B

E,MEV.	DK FRACTION
6.6625	6.00000E-02
6.7045	9.00000E-01
6.7567	3.00000E-02

86-RN-218
2 ALPHAS, REF B

E,MEV.	DK FRACTION
6.5349	1.00091E-03
7.1331	9.98699E-01

85-AT-219
1 ALPHA, REF B

E,MEV.	DK FRACTION
6.2733	9.70000E-01

86-RN-219
4 ALPHAS, REF B

E,MEV.	DK FRACTION
6.4250	7.49101E-02
6.5310	1.10856E-03
6.5532	1.14862E-01
6.8194	8.09029E-01

86-RN-220
2 ALPHAS, REF A

E,MEV.	DK FRACTION
5.7490	7.00000E-04
6.2883	9.99300E-01

87-FR-221
4 ALPHAS, REF B

E,MEV.	DK FRACTION
6.0752	1.36426E-03
6.1275	1.37335E-01
6.2434	1.18236E-02
6.3411	8.49477E-01

86-RN-222
2 ALPHAS, REF B

E,MEV.	DK FRACTION
4.9870	7.80172E-04
5.4898	9.99220E-01

87-FR-222
1 ALPHA, REF B

E,MEV.	DK FRACTION
5.7092	1.00000E-03

88-RA-222
2 ALPHAS, REF B

E,MEV.	DK FRACTION
6.2373	3.10000E-02
6.5557	9.69000E-01

87-FR-223
1 ALPHA, REF B

E,MEV.	DK FRACTION
5.3330	4.00000E-05

88-RA-223
12 ALPHAS, REF B

E,MEV.	DK FRACTION
5.2839	1.00000E-03
5.2885	1.30000E-03
5.3399	1.00000E-03
5.3660	1.10000E-03
5.4347	2.30000E-02
5.5025	1.00000E-02
5.5396	9.10000E-02
5.6073	2.40000E-01
5.7161	5.28900E-01
5.7474	9.10000E-02
5.8217	3.20000E-03
5.8718	8.50000E-03

88-RA-224
5 ALPHAS, REF A

E,MEV.	DK FRACTION
5.0340	3.09345E-05
5.0870	7.19873E-05
5.1610	7.29872E-05
5.4490	4.89914E-02
5.6856	9.50833E-01

89-AC-225
13 ALPHAS, REF B

E,MEV.	DK FRACTION
5.2871	2.30161E-03
5.4432	1.50105E-03
5.5539	1.00070E-03
5.5809	1.20084E-02
5.6093	1.20084E-02
5.6377	4.35305E-02
5.6824	1.25088E-02
5.7255	3.50235E-02
5.7520	3.01071E-01
5.7921	9.00630E-02
5.7939	1.80126E-01
5.8043	3.00210E-03
5.8299	5.06855E-01

88-RA-226
2 ALPHAS, REF B

E,MEV.	DK FRACTION
4.6017	5.55000E-02
4.7846	9.44500E-01

89-AC-226
1 ALPHA, REF B

E,MEV.	DK FRACTION
5.4400	6.00000E-05

90-TH-226
5 ALPHAS, REF B

E,MEV.	DK FRACTION
6.0258	2.00682E-03
6.0414	1.90648E-03
6.0997	1.27433E-02
6.2284	2.30785E-01
6.3375	7.52559E-01

89-AC-227
8 ALPHAS, REF B

E,MEV.	DK FRACTION
4.7151	4.30252E-05
4.7701	1.43308E-04
4.7967	1.12421E-04
4.8556	5.13527E-04
4.8733	8.46626E-04
4.9009	1.52670E-05
4.9416	5.55164E-03
4.9538	6.52318E-03

90-TH-227
19 ALPHAS, REF B

E,MEV.	DK FRACTION
5.5862	1.81892E-03
5.6021	1.71787E-03
5.6139	2.22312E-03
5.6686	2.12207E-02
5.6940	1.51576E-02
5.7017	3.63783E-02
5.7097	8.28618E-02
5.7142	4.95150E-02
5.7572	2.02102E-01
5.7630	2.32417E-03
5.7952	3.13258E-03
5.8076	1.31366E-02
5.8667	2.42522E-02
5.9103	1.71787E-03
5.9166	7.88197E-03
5.9599	3.03153E-02
5.9779	2.32417E-01
6.0089	2.93048E-02
6.0383	2.42522E-01

90-TH-228
5 ALPHAS, REF A

E,MEV.	DK FRACTION
5.1380	5.00050E-04
5.1770	1.80018E-03
5.2110	3.60036E-03
5.3405	2.67027E-01
5.4233	7.27073E-01

90-TH-229
14 ALPHAS, REF B

E,MEV.	DK FRACTION
4.7626	6.36107E-03
4.7987	1.28231E-02
4.8148	8.48142E-02
4.8390	4.84653E-02
4.8460	5.67847E-01
4.8622	1.81745E-03
4.9017	1.09047E-01
4.9309	1.11066E-03
4.9686	6.46204E-02
4.9795	3.23102E-02
5.0363	2.42326E-03
5.0492	5.25040E-02
5.0534	1.61555E-02
5.0783	1.00969E-04

90-TH-230
7 ALPHAS, REF A

E,MEV.	DK FRACTION
4.2440	5.00746E-04
4.2780	8.01194E-08
4.3120	1.00149E-05
4.4380	3.00448E-04
4.4800	1.20179E-03
4.6210	2.34349E-01
4.6875	7.64136E-01

91-PA-230
18 ALPHAS, REF B

E,MEV.	DK FRACTION
4.7653	6.40446E-08
4.7987	9.60672E-04
4.9343	1.28090E-07
4.9726	2.24157E-07
5.0600	1.28090E-07
5.0836	2.24157E-07
5.1190	1.92134E-07
5.1534	1.28090E-07
5.1834	1.60112E-07
5.2173	1.60112E-07
5.2683	1.12078E-06
5.2762	9.60672E-07
5.2880	9.60672E-07
5.3008	5.44381E-06
5.3126	4.16291E-06
5.3263	5.76403E-06
5.3401	4.80336E-06
5.3450	7.36516E-06

92- U-230
4 ALPHAS, REF B

E,MEV.	DK FRACTION
5.6822	2.30023E-03
5.6661	3.60036E-03
5.8176	3.19032E-01
5.8886	6.75068E-01

91-PA-231
19 ALPHAS, REF A

E,MEV.	DK FRACTION
4.5090	3.02822E-05
4.5660	8.07526E-05
4.6000	1.51411E-04
4.6330	1.00941E-03
4.6440	1.00941E-03
4.6820	1.51411E-02
4.7130	1.00941E-02
4.7360	6.47902E-02
4.7460	4.03763E-04
4.8540	1.41317E-02
4.9020	2.01882E-05
4.9340	3.02822E-02
4.9517	2.30145E-01
4.9760	4.03763E-03
4.9860	1.41317E-02
5.0141	2.56390E-01
5.0297	2.01882E-01
5.0320	2.52352E-02
5.0590	1.11035E-01

TABLE VII (cont.)

92- U-231
1 ALPHA , REF B

E,MEV.	DK FRACTION
5.4539	5.50000E-05

90-TH-232
3 ALPHAS, REF A

E,MEV.	DK FRACTION
3.8300	1.99601E-03
3.9530	2.29541E-01
4.0120	7.68463E-01

92- U-232
7 ALPHAS, REF A

E,MEV.	DK FRACTION
4.5090	2.39800E-07
4.9291	2.09825E-06
4.9460	1.69859E-06
4.9973	2.89759E-05
5.1373	2.79767E-03
5.2635	3.11740E-01
5.3203	6.85429E-01

92- U-233
25 ALPHAS, REF A

E,MEV.	DK FRACTION
4.3070	9.01370E-06
4.4060	8.01218E-06
4.4570	2.80426E-05
4.4650	3.00457E-05
4.4830	1.40213E-05
4.5030	1.00152E-05
4.5070	1.20183E-04
4.5130	1.80274E-04
4.5380	4.00609E-05
4.5650	2.80426E-05
4.5720	2.30350E-05
4.5900	7.01066E-05
4.6110	6.00913E-05
4.6150	4.00609E-05
4.6340	1.00152E-04
4.6410	3.00457E-05
4.6560	4.00609E-05
4.6640	4.20639E-04
4.6810	1.00152E-04
4.7010	6.00913E-04
4.7290	1.61245E-02
4.7540	1.63248E-03
4.7825	1.32201E-01
4.7960	2.80426E-03
4.8242	8.45285E-01

92- U-234
5 ALPHAS, REF A

E,MEV.	DK FRACTION
4.1200	3.29013E-07
4.2740	8.48654E-07
4.6030	2.99102E-03
4.7228	2.71177E-01
4.7730	7.22831E-01

92- U-235
12 ALPHAS, REF A

E,MEV.	DK FRACTION
4.1540	9.00000E-03
4.2170	5.70000E-02
4.2270	6.00000E-03
4.2710	4.00000E-03
4.3240	4.70000E-02
4.3640	1.70000E-01
4.3980	5.60000E-01
4.4160	2.10000E-02
4.4390	1.00000E-03
4.5020	1.70000E-02
4.5560	4.50000E-02
4.5980	5.40000E-02

93-NP-235
11 ALPHAS, REF B

E,MEV.	DK FRACTION
4.8042	1.60160E-08
4.8599	1.12112E-07
4.9227	1.84184E-06
4.9365	9.60961E-08
4.9933	9.60961E-07
5.0025	8.00801E-08
5.0033	3.84384E-06
5.0203	8.48849E-06
5.0455	2.88288E-07
5.0940	3.20320E-08
5.1030	2.40240E-07

94-PU-235
1 ALPHA , REF B

E,MEV.	DK FRACTION
5.8556	1.30000E-04

92- U-236
3 ALPHAS, REF A

E,MEV.	DK FRACTION
4.3310	2.59326E-03
4.8450	2.59326E-01
4.9940	7.38081E-01

94-PU-236
6 ALPHAS, REF A

E,MEV.	DK FRACTION
5.0880	6.00103E-06
5.2140	2.70046E-06
5.4520	2.00034E-05
5.6150	1.80031E-03
5.7210	3.09053E-01
5.7680	6.89118E-01

93-NP-237
14 ALPHAS, REF A

E,MEV.	DK FRACTION
4.5180	3.98843E-04
4.5810	3.98843E-03
4.5980	3.39017E-03
4.6391	6.16213E-02
4.6640	3.31040E-02
4.6940	4.78612E-03
4.7120	1.12673E-02
4.7659	7.97687E-02
4.7701	2.49277E-01
4.7882	6.68641E-01
4.8030	2.99133E-02
4.8170	2.49277E-02
4.8660	2.99133E-03
4.8730	2.59248E-02

94-PU-237
2 ALPHAS, REF A

E,MEV.	DK FRACTION
5.3600	2.60700E-05
5.6500	6.93000E-06

92- U-238
3 ALPHAS, REF A

E,MEV.	DK FRACTION
4.0390	2.29472E-03
4.1490	2.29472E-01
4.1960	7.68233E-01

94-PU-238
11 ALPHAS, REF A

E,MEV.	DK FRACTION
4.5670	1.10071E-09
4.5880	1.60104E-08
4.6620	3.00195E-08
4.6640	3.00195E-09
4.7030	9.00584E-07
4.7260	1.00065E-07
5.0110	7.00455E-08
5.2080	5.00325E-05
5.3590	1.30084E-03
5.4565	2.87186E-01
5.4992	7.11462E-01

94-PU-239
21 ALPHAS, REF A

E,MEV.	DK FRACTION
4.3990	2.50024E-07
4.5100	8.00078E-07
4.6300	7.00068E-06
4.6890	5.00048E-06
4.7360	4.50044E-05
4.7490	6.00058E-06
4.7690	8.00078E-06
4.7950	7.00068E-06
4.8280	2.40023E-05
4.8680	8.00078E-06
4.9110	2.00019E-06
4.9350	3.00029E-05
4.9620	3.00029E-05
4.9870	7.00068E-05
5.0080	8.00078E-05
5.0280	5.00048E-05
5.0540	2.10020E-04
5.0750	3.20031E-04
5.1046	1.15011E-01
5.1429	1.51015E-01
5.1554	7.33071E-01

94-PU-240
5 ALPHAS, REF A

E,MEV.	DK FRACTION
4.4800	2.10015E-07
4.8510	2.00014E-05
5.0140	9.10064E-04
5.1234	2.65018E-01
5.1683	7.34051E-01

95-AM-240
3 ALPHAS, REF A

E,MEV.	DK FRACTION
5.2860	2.33630E-08
5.3370	2.27932E-07
5.3780	1.64871E-06

96-CM-240
4 ALPHAS, REF B

E,MEV.	DK FRACTION
5.9909	1.39908E-04
6.1482	5.19657E-04
6.2469	2.88809E-01
6.2908	7.10531E-01

94-PU-241
11 ALPHAS, REF A

E,MEV.	DK FRACTION
4.6920	7.35662E-09
4.7320	7.35662E-09
4.7420	1.71654E-08
4.7830	4.90441E-08
4.7970	2.94265E-07
4.8535	2.96717E-06
4.8965	2.04024E-05
4.9720	3.18787E-07
4.9990	1.00540E-07
5.0420	2.50125E-07
5.0540	8.58272E-08

95-AM-241
21 ALPHAS, REF A

E,MEV.	DK FRACTION
4.8000	9.00775E-07
4.8340	1.00603E-06
5.0040	1.00086E-06
5.0680	1.40121E-06
5.0840	4.00345E-06
5.0960	4.00345E-06
5.1140	4.00345E-06
5.1560	7.00603E-06
5.1780	3.00258E-06
5.1820	9.00775E-06
5.1940	6.00517E-06
5.2230	1.30112E-05
5.2440	2.40207E-05
5.2790	5.00431E-06
5.3220	1.50129E-04
5.3690	1.33115E-02
5.4170	1.00086E-04
5.4430	1.28110E-01
5.4857	8.52734E-01
5.5130	2.00172E-03
5.5443	3.50302E-03

96-CM-241
11 ALPHAS, REF A

E,MEV.	DK FRACTION
5.6870	2.20044E-05
5.7190	8.00160E-06
5.7850	7.00140E-06
5.8630	1.40028E-05
5.8843	1.18024E-03
5.9140	1.20024E-05
5.9260	1.81036E-04
5.9386	6.89138E-03
5.9780	2.80056E-05
6.0360	1.20024E-05
6.0820	1.50030E-05

94-PU-242
4 ALPHAS, REF A

E,MEV.	DK FRACTION
4.5985	1.30001E-05
4.7543	9.80007E-04
4.8563	2.24002E-01
4.9006	7.75005E-01

95-AM-242M
7 ALPHAS, REF A

E,MEV.	DK FRACTION
5.0650	1.20056E-05
5.0860	1.53672E-05
5.1400	2.90547E-04
5.2050	4.27401E-03
5.3130	3.84181E-05
5.3650	7.20339E-05
5.4080	5.76271E-05

96-CM-242
8 ALPHAS, REF A

E,MEV.	DK FRACTION
5.1460	4.99798E-08
5.1890	2.49898E-07
5.5140	2.49898E-06
5.6090	1.99918E-07
5.8170	4.59812E-05
5.9720	3.59853E-04
6.0696	2.59894E-01
6.1129	7.39697E-01

95-AM-243
13 ALPHAS, REF A

E,MEV.	DK FRACTION
4.6950	1.60296E-05
4.9190	8.51572E-07
4.9300	1.80333E-06
4.9460	3.40629E-06
5.0080	1.60296E-05
5.0320	2.20407E-05
5.0860	4.00740E-05
5.1130	5.40999E-05
5.1810	1.10203E-02
5.2335	1.10203E-01
5.2754	8.75616E-01
5.3210	1.60222E-03
5.3492	1.80333E-03

96-CM-243
28 ALPHAS, REF A

E,MEV.	DK FRACTION
5.2260	3.86407E-06
5.2660	1.46041E-05
5.3150	9.90940E-06
5.3220	2.97282E-05
5.3310	2.97282E-05
5.5220	1.98188E-05
5.5310	5.94564E-05
5.5360	1.98188E-05
5.5670	6.93658E-05
5.5740	6.93658E-05
5.5810	8.91846E-05
5.5860	1.98188E-04
5.5920	9.90940E-05
5.6080	9.90940E-05
5.6110	3.96376E-04
5.6210	5.94564E-04
5.6380	1.38732E-03
5.6450	2.97282E-04
5.6810	1.98188E-03
5.6850	1.58550E-02
5.7415	1.13456E-01
5.7847	7.25368E-01
5.8750	6.93658E-03
5.9060	9.90940E-04
5.9920	5.64836E-02
6.0090	1.09009E-02
6.0560	4.65742E-02
6.0670	1.48641E-02

TABLE VII (cont.)

94-PU-244
2 ALPHAS, REF A

E,MEV.	DK FRACTION
4.5460	1.93757E-01
4.5890	8.04993E-01

96-CM-244
8 ALPHAS, REF A

E,MEV.	DK FRACTION
4.9200	8.99767E-07
5.0600	1.09948E-06
5.2150	1.10969E-06
5.3130	3.99896E-07
5.5130	3.99909E-05
5.6640	2.19943E-04
5.7628	2.35939E-01
5.8050	7.63802E-01

96-CM-245
6 ALPHAS, REF A

E,MEV.	DK FRACTION
5.2346	3.20256E-03
5.3038	4.97398E-02
5.3620	9.32546E-01
5.4363	4.00320E-04
5.4887	8.30665E-03
5.5292	5.80464E-03

96-CM-246
2 ALPHAS, REF A

E,MEV.	DK FRACTION
5.3430	2.09945E-01
5.3860	7.89793E-01

96-CM-247
7 ALPHAS, REF A

E,MEV.	DK FRACTION
4.8140	4.70000E-02
4.8680	7.10000E-01
4.9410	1.60000E-02
4.9830	2.00000E-02
5.1450	1.20000E-02
5.2100	5.70000E-02
5.2650	1.38000E-01

96-CM-248
2 ALPHAS, REF A

E,MEV.	DK FRACTION
5.0340	1.66049E-01
5.0780	7.51351E-01

98-CF-248
2 ALPHAS, REF B

E,MEV.	DK FRACTION
6.2281	1.80000E-01
6.2663	8.20000E-01

97-BK-249
7 ALPHAS, REF A

E,MEV.	DK FRACTION
5.0450	1.45291E-08
5.1150	3.92285E-07
5.2530	1.45291E-08
5.3510	3.77756E-07
5.3899	2.67335E-06
5.4168	1.00541E-05
5.4373	9.73447E-07

98-CF-249
16 ALPHAS, REF A

E,MEV.	DK FRACTION
5.3510	1.99410E-05
5.4310	9.97049E-05
5.5020	4.38701E-04
5.5590	1.09675E-03
5.6230	1.99410E-04
5.6940	2.99115E-03
5.7597	4.78583E-02
5.7840	2.49282E-03
5.8120	8.23562E-01
5.8495	1.39587E-02
5.9034	3.19056E-02
5.9462	3.38997E-02
6.0000	5.98229E-04
6.0720	3.98819E-03
6.1395	1.29616E-02
6.1940	2.39292E-02

98-CF-250
4 ALPHAS, REF A

E,MEV.	DK FRACTION
5.7367	9.99130E-05
5.8000	2.99739E-03
5.9891	1.61059E-01
6.0300	8.34274E-01

98-CF-251
14 ALPHAS, REF A

E,MEV.	DK FRACTION
5.5010	3.04569E-03
5.5660	1.52284E-02
5.6030	2.03046E-03
5.6320	4.56853E-02
5.6480	3.55330E-02
5.6770	3.53259E-01
5.7380	1.01523E-02
5.7620	3.85787E-02
5.7930	2.03046E-02
5.8140	4.26396E-02
5.8520	2.78173E-01
5.9430	6.09137E-03
6.0140	1.21827E-01
6.0740	2.74112E-02

98-CF-252
5 ALPHAS, REF A

E,MEV.	DK FRACTION
5.6160	5.80623E-07
5.8263	1.93541E-05
5.9766	2.32249E-03
6.0757	1.51930E-01
6.1183	8.14807E-01

98-CF-253
2 ALPHAS, REF A

E,MEV.	DK FRACTION
5.9210	1.64300E-08
5.9790	2.93570E-03

99-ES-253
28 ALPHAS, REF A

E,MEV.	DK FRACTION
5.7300	8.00043E-07
5.9100	2.70015E-07
5.9250	4.00022E-07
5.9440	1.50006E-06
6.0190	1.80010E-06
6.0370	2.90016E-06
6.0460	4.00022E-06
6.0840	2.50013E-06
6.1000	3.40018E-05
6.1220	7.80042E-06
6.1660	1.50008E-04
6.2110	3.90021E-04
6.2170	1.50008E-05
6.2300	1.20006E-06
6.2500	8.50024E-08
6.2660	8.00043E-06
6.3250	4.00022E-06
6.3540	8.20044E-05
6.4080	1.30007E-04
6.4320	2.10033E-05
6.4800	8.50046E-04
6.4980	2.60014E-03
6.5200	8.50046E-03
6.5520	7.10038E-03
6.5920	6.60036E-02
6.5940	7.00038E-03
6.6240	8.00043E-03
6.6327	8.98048E-01

98-CF-254
2 ALPHAS, REF B

E,MEV.	DK FRACTION
5.7943	5.27000E-04
5.8366	2.57300E-03

99-ES-254
8 ALPHAS, REF B

E,MEV.	DK FRACTION
6.2669	2.21953E-03
6.2758	1.61421E-03
6.3467	7.46570E-03
6.3595	2.92575E-02
6.3841	1.31154E-03
6.4155	1.71509E-02
6.4283	9.36257E-01
6.4780	2.72397E-03

99-ES-254M
7 ALPHAS, REF B

E,MEV.	DK FRACTION
6.2472	7.37056E-05
6.2797	2.78071E-04
6.3045	2.51269E-03
6.3405	6.03046E-05
6.4361	4.69036E-05
6.4791	1.94315E-04
6.5130	1.34010E-04

100-FM-254
3 ALPHAS, REF B

E,MEV.	DK FRACTION
7.0492	9.00369E-03
7.1468	1.40057E-01
7.1889	8.50349E-01

99-ES-255
3 ALPHAS, REF B

E,MEV.	DK FRACTION
6.2137	2.00000E-03
6.2609	7.84000E-03
6.2996	7.01600E-02

100-FM-255
6 ALPHAS, REF B

E,MEV.	DK FRACTION
6.8069	1.10375E-03
6.8915	6.22115E-03
6.9634	5.01706E-02
6.9829	1.30444E-03
7.0225	9.37186E-01
7.0800	4.01365E-03

100-FM-256
1 ALPHA , REF B

E,MEV.	DK FRACTION
6.9152	1.00000E+00

100-FM-257
5 ALPHAS, REF B

E,MEV.	DK FRACTION
6.3467	3.01811E-03
6.4410	2.01207E-02
6.5199	9.35614E-01
6.6965	3.52113E-02
6.7572	6.03622E-03

REFERENCE A= ENDFB/B-V
REFERENCE B= TABLE OF ISOTOPES, SEVENTH EDITION

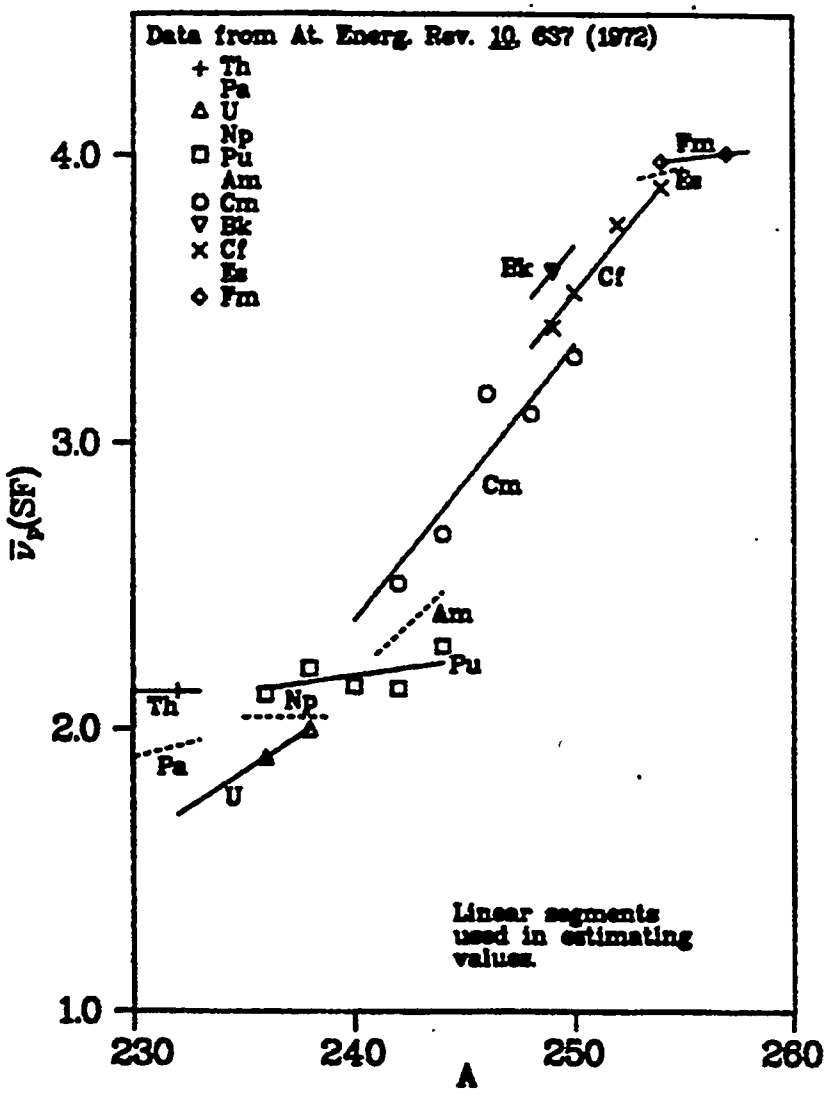


Fig. 4.
Values of $\bar{\nu}_p$(SF).

TABLE VIII

SPONTANEOUS-FISSION NEUTRON PRODUCTION BY ACTINDE DECAY

NUCLIDE	NU-BAR VALUES PROMPT	DELAYED	TOTAL	SPONTANEOUS FISSION BRANCHING	NEUTRONS PER NUCLIDE DECAY
90-TH-230	2.13	.01	2.14	5.330-13 A	1.14 -12
91-PA-231	1.92	.01	1.93	2.980-12 A	5.75 -12
90-TH-232	2.130+.200 B	.01	2.14	1.410-11 A	3.02 -11
92- U-232	1.70	.01	1.71	9.000-13 C	1.54 -12
92- U-233	1.75	.01	1.76	1.300-12 C	2.29 -12
92- U-234	1.80	.01	1.81	1.200-11 C	2.17 -11
92- U-235	1.85	.01	1.86	2.011-09 A	3.74 -09
92- U-236	1.900+.050 B	.01	1.91	1.200-09 C	2.29 -09
94-PU-236	2.120+.130 B	.01	2.13	8.100-10 C	1.73 -09
93-NP-237	2.04	.01	2.05	2.140-12 A	4.39 -12
92- U-238	2.000+.030 B	.01	2.01	5.450-07 C	1.095-06
94-PU-238	2.210+.130 B	.01	2.22	1.840-09 C	4.08 -09
94-PU-239	2.15	.01	2.16	4.400-12 C	9.37 -12
94-PU-240	2.151+.006 B	.01	2.16	5.000-08 C	1.08 -07
96-CM-240	2.38	.01	2.39	3.860-08 A	9.23 -08
95-AM-241	2.26	.01	2.27	4.100-12 C	9.31 -12
94-PU-242	2.141+.190 B	.01	2.15	5.500-06 C	1.18 -05
95-AM-242M	2.33	.01	2.34	1.600-10 C	3.74 -10
96-CM-242	2.510+.060 B	.01	2.52	6.800-08 C	1.71 -07
95-AM-243	2.41	.01	2.42	2.200-10 C	5.32 -10
94-PU-244	2.290+.190 B	.01	2.30	1.250-03 C	2.88 -03
96-CM-244	2.681+.011 B	.01	2.69	1.347-06 C	3.62 -06
96-CM-246	3.170+.220 B	.01	3.18	2.614-04 C	8.31 -04
96-CM-248	3.100+.090 B	.01	3.11	8.260-02 C	2.569-01
98-CF-248	3.33	.01	3.34	2.850-05 A	9.52 -05
97-BK-249	3.590+.160 B	.01	3.60	4.600-10 C	1.66 -09
98-CF-249	3.400+.400 B	.01	3.41	5.020-09 A	1.71 -08
96-CM-250	3.300+.080 B	.01	3.31	7.000-01 D	2.32 +00
98-CF-250	3.520+.090 B	.01	3.53	3.092-02 C	2.72 -03
98-CF-252	3.756+.012 B	.009 B	3.765+.010 B	3.092-02 C	1.164-01
99-ES-253	3.92	.01	3.93	8.700-08 C	3.42 -07
98-CF-254	3.890+.050 B	.01	3.890+.050 E	9.969-01 A	3.88 +00
99-ES-254	3.94	.01	3.95	3.020-08 A	1.19 -07
99-ES-254M	3.94	.01	3.95	4.500-08 A	1.78 -07
100-FM-254	3.980+.140 B	.01	3.96 +.14 F	5.900-04 A	2.34 -03
99-ES-255	3.96	.01	3.97	4.000-05 A	1.59 -04
100-FM-255	3.99	.01	3.73 +.18 F	2.290-07 A	8.54 -07
100-FM-256	4.00	.01	4.01	9.190-01 A	3.69 +00
100-FM-257	4.010+.130 B	.01	3.85 +.05 G	2.100-03 A	8.09 -03
100-FM-258	4.02	.01	4.03	1.000+00 A	4.03 +00

DATA REFEPENCES USED
A=TABLE OF ISOTOPES, SEVENTH EDITION
B=MANERO AND KONSHIN, ATOMIC ENERGY REV. 10,637-756(1972)
C=ENDF/B-V
D=A.TOBIAS,U.K.,PRIVATE COMMUNICATION
E=C.J.ORTH,NUCL.SCI.ENG.43,54(1971)
F=Y.A.LAZAPEV,ATOMIC ENERGY REV.15,75(1977)
G=D.C.HOFFMAN ET AL.,PHYS.REV.C21,637(1980)
ADDITIONAL REFERENCES SURVEYED
J.W.BOLDEMAN,IN NEUTRON STD. REF.DATA, I.A.E.A. VIENNA (1974)
J.P.BALAGNA ET AL.,PHYS.REV.LETT.26,145(1971)

PPOMPT NU-BAR VALUES GIVEN WITHOUT REFERENCE HAVE BEEN ESTIMATED FROM THE VALUES OF PEFERENCE B. DELAYED NU-BAR VALUES GIVEN WITHOUT REFERENCE HAVE BEEN ARBITRARILY ASSUMED.

IV. CALCULATION OF THE THICK-TARGET NEUTRON-PRODUCTION FUNCTION $P_i(E_\alpha)$

The neutron-production function $P_i(E_\alpha)$ defined by Eqs. (6) and (9) gives the contribution from reactions with nuclude i to the probability of neutron production by a decay alpha particle of energy E_α emitted within the material. The POFEAL code calculates values of P_i OF E-ALPHA using the algorithm

$$P(J) = 1.E + 6 * \frac{N_i}{N} \sum_{j=2}^{J} \frac{[\sigma_i(j-1)+\sigma_i(j)]/2}{[\varepsilon(j-1)+\varepsilon(j)]/2} [E(j)-E(j-1)] \quad , \tag{25}$$

where

N_i is the atom density of nuclide i (atoms/cm^3),

N is the total atom density (atoms/cm^3),

E_j is the jth regular energy point at or above the cross-section threshold (MeV),

$\sigma_i(j)$ is the value of the (α,n) cross section of nuclide i at E_j (mb),

$\varepsilon(j)$ is the value of the stopping cross section (eV/10^{15} atoms/cm^2),

and the leading quantity of 1 x 10^6 is required because of the units of σ, ε, and E.

The ^{17}O and ^{18}O contributions to the (α,n) neutron-production rate are given in Tables IX-XII for each of the four fuel compositions given in Table I. Values for the four compositions at any energy differ by less than 4%. The ^{17}O and ^{18}O contributions to (α,n) neutron production in spent UO_2 fuel are shown in Fig. 5.

V. RESULTS

The half-lives, average decay energies, and spent UO_2 fuel neutron-production values $R_k(\alpha,n)$, $R_k(SF)$, and R_k for each of the actinide nuclides k are given in Table XIII. Values of $R_k(SF)$ are repeated from Table VIII. Values of $R_k(\alpha,n)$ were obtained using the alpha spectra data of Table VII and $P(E_\alpha)$ values given in Table XI for $^{17,18}O(\alpha,n)$ in spent UO_2 fuel.

TABLE IX

$^{17,18}O(\alpha,n)$ NEUTRONS PRODUCTION IN CLEAN ThO_2 FUEL BY ALPHA PARTICLES BELOW 10 MeV

E,MEV	**NEUTRONS-PER-ALPHA* O-17	O-18	TOTAL
0.000	0.	0.	0.
1.139	0.	0.	0.
1.161	0.	0.	0.
1.161	0.0	1.5-	1.52-
1.186	0.0	2.6-	2.61-
1.211	0.0	2.6-	2.61-
1.236	0.0	2.6-	2.61-
1.261	0.0	2.6-	2.61-
1.286	0.0	3.0-	3.03-
1.304	0.0	3.2-	3.19-
1.311	1.0	3.3-	3.29-
1.319	1.0	5.7-	5.86-
1.324	2.1	8.0-	8.26-
1.349	4.1	1.0-	1.02-
1.374	6.2	1.0-	1.07-
1.399	1.0	1.1-	1.17-
1.424	1.1	1.2-	1.29-
1.449	1.1	1.2-	1.31-
1.474	1.1	2.1-	2.01-
1.499	1.1	2.2-	2.30-
1.524	1.1	2.2-	2.33-
1.549	1.1	1.0-	1.06-
1.574	1.1	1.0-	1.17-
1.599	1.1	1.1-	1.27-
1.624	1.1	1.2-	1.31-
1.649	1.1	1.2-	1.36-
1.674	2.1	1.2-	1.41-
1.699	1.1	1.2-	1.32-
1.724	3.1	1.1-	1.50-
1.749	4.1	1.2-	1.73-
1.774	5.1	1.2-	1.94-
1.799	5.1	1.2-	2.20-
1.824	6.1	1.2-	2.36-
1.849	6.1	1.2-	2.56-
1.874	7.1	1.2-	2.69-
1.899	7.1	1.2-	2.80-
1.924	8.1	1.2-	3.05-
1.949	8.1	1.2-	3.20-
1.974	9.1	1.2-	3.39-
2.024	1.0	1.2-	3.75-
2.049	1.0	1.2-	3.95-
2.074	1.0	1.2-	4.05-
2.124	1.0	1.2-	4.18-
2.149	1.0	1.2-	4.26-
2.174	1.0	1.2-	4.36-
2.199	1.0	1.2-	4.46-
2.224	1.0	1.2-	4.56-
2.249	1.0	1.2-	4.66-
2.274	1.0	1.2-	4.76-
2.299	1.0	1.2-	4.86-
2.324	1.0	1.2-	4.96-
2.349	1.0	1.2-	5.06-
2.374	1.0	1.2-	5.16-
2.399	1.0	1.2-	5.26-
2.424	1.0	1.2-	5.36-
2.449	1.0	1.2-	5.46-
2.474	1.0	1.2-	5.56-
2.499	1.0	1.2-	5.66-
2.524	1.0	1.2-	5.76-
2.549	1.0	1.2-	5.86-
2.574	1.0	1.2-	5.96-
2.599	1.0	1.2-	6.06-
2.624	1.0	1.2-	6.16-
2.649	1.0	1.2-	6.26-
2.674	1.0	1.2-	6.36-
2.699	1.0	1.2-	6.46-
2.724	1.0	1.2-	6.56-
2.749	1.0	1.2-	6.66-
2.774	1.0	1.2-	6.76-
2.799	1.0	1.2-	6.86-
2.824	1.0	1.2-	6.96-
2.849	1.0	1.2-	7.06-
2.874	1.0	1.2-	7.16-
2.899	1.0	1.2-	7.26-
2.924	1.0	1.2-	7.36-
2.949	1.0	1.2-	7.46-
2.974	1.0	1.2-	7.56-
3.024	1.0	1.2-	7.76-
3.049	1.0	1.2-	7.86-
3.074	1.0	1.2-	7.96-
3.124	1.0	1.2-	8.16-
3.149	1.0	1.2-	8.26-
3.174	1.0	1.2-	8.36-
3.199	1.0	1.2-	8.46-
3.224	1.0	1.2-	8.56-
3.249	1.0	1.2-	8.66-
3.274	1.0	1.2-	8.76-
3.299	1.0	1.2-	8.86-
3.324	1.0	1.2-	8.96-
3.349	1.0	1.2-	9.06-
3.374	1.0	1.2-	9.16-
3.399	1.0	1.2-	9.26-
3.424	1.0	1.2-	9.36-
3.449	1.0	1.2-	9.46-
3.474	1.0	1.2-	9.56-
3.499	1.0	1.2-	9.66-
3.524	1.0	1.2-	9.76-
3.549	1.0	1.2-	9.86-
3.574	1.0	1.2-	9.96-
3.599	1.0	1.2-	1.01-
3.624	1.0	1.2-	1.02-
3.649	1.0	1.2-	1.03-
3.674	1.0	1.2-	1.04-
3.699	1.0	1.2-	1.05-
3.724	1.0	1.2-	1.06-
3.749	1.0	1.2-	1.07-
3.774	1.0	1.2-	1.08-
3.799	1.0	1.2-	1.09-
3.824	1.0	1.2-	1.10-
3.849	1.0	1.2-	1.11-
3.874	1.0	1.2-	1.12-
3.899	1.0	1.2-	1.13-
3.924	1.0	1.2-	1.14-
3.949	1.0	1.2-	1.15-
3.974	1.0	1.2-	1.16-
4.024	1.0	1.2-	1.18-
4.049	1.0	1.2-	1.19-
4.074	1.0	1.2-	1.20-
4.124	1.0	1.2-	1.22-
4.149	1.0	1.2-	1.23-
4.174	1.0	1.2-	1.24-
4.199	1.0	1.2-	1.25-
4.224	1.0	1.2-	1.26-
4.249	1.0	1.2-	1.27-
4.274	1.0	1.2-	1.28-
4.299	1.0	1.2-	1.29-
4.324	1.0	1.2-	1.30-
4.349	1.0	1.2-	1.31-
4.374	1.0	1.2-	1.32-
4.399	1.0	1.2-	1.33-
4.424	1.0	1.2-	1.34-
4.449	1.0	1.2-	1.35-
4.474	1.0	1.2-	1.36-
4.499	1.0	1.2-	1.37-
4.524	1.0	1.2-	1.38-
4.549	1.0	1.2-	1.39-
4.574	1.0	1.2-	1.40-
4.599	1.0	1.2-	1.41-
4.624	1.0	1.2-	1.42-
4.649	1.0	1.2-	1.43-
4.674	1.0	1.2-	1.44-
4.699	1.0	1.2-	1.45-
4.724	1.0	1.2-	1.46-
4.749	1.0	1.2-	1.47-
4.774	1.0	1.2-	1.48-
4.799	1.0	1.2-	1.49-
4.824	1.0	1.2-	1.50-
4.849	1.0	1.2-	1.51-
4.874	1.0	1.2-	1.52-
4.899	1.0	1.2-	1.53-
4.924	1.0	1.2-	1.54-
4.949	1.0	1.2-	1.55-
4.974	1.0	1.2-	1.56-
5.024	1.0	1.2-	1.57-
5.049	1.0	1.2-	1.58-
5.074	1.0	1.2-	1.59-
5.099	1.0	1.2-	1.60-
5.124	1.0	1.2-	1.61-
5.149	1.0	1.2-	1.62-
5.174	1.0	1.2-	1.63-
5.199	1.0	1.2-	1.64-
5.224	1.0	1.2-	1.65-
5.249	1.0	1.2-	1.66-
5.274	1.0	1.2-	1.67-
5.299	1.0	1.2-	1.68-
5.324	1.0	1.2-	1.69-
5.349	1.0	1.2-	1.70-
5.374	1.0	1.2-	1.71-
5.399	1.0	1.2-	1.72-
5.424	1.0	1.2-	1.73-
5.449	1.0	1.2-	1.74-
5.474	1.0	1.2-	1.75-
5.499	1.0	1.2-	1.76-
5.524	1.0	1.2-	1.77-
5.549	1.0	1.2-	1.78-
5.574	1.0	1.2-	1.79-
5.599	1.0	1.2-	1.80-
5.624	1.0	1.2-	1.81-
5.649	1.0	1.2-	1.82-
5.674	1.0	1.2-	1.83-
5.699	1.0	1.2-	1.84-
5.724	1.0	1.2-	1.85-
5.749	1.0	1.2-	1.86-
5.774	1.0	1.2-	1.87-
5.799	1.0	1.2-	1.88-
5.824	1.0	1.2-	1.89-
5.849	1.0	1.2-	1.90-
5.874	1.0	1.2-	1.91-
5.899	1.0	1.2-	1.92-
5.924	1.0	1.2-	1.93-
5.949	1.0	1.2-	1.94-
5.974	1.0	1.2-	1.95-
6.024	1.0	1.2-	1.97-
6.049	1.0	1.2-	1.98-
6.074	1.0	1.2-	1.99-
6.124	1.0	1.2-	2.01-
6.149	1.0	1.2-	2.02-
6.174	1.0	1.2-	2.03-
6.199	1.0	1.2-	2.04-
6.224	1.0	1.2-	2.05-
6.249	1.0	1.2-	2.06-
6.274	1.0	1.2-	2.07-
6.299	1.0	1.2-	2.08-
6.324	1.0	1.2-	2.09-
6.349	1.0	1.2-	2.10-
6.374	1.0	1.2-	2.11-
6.399	1.0	1.2-	2.12-
6.424	1.0	1.2-	2.13-
6.449	1.0	1.2-	2.14-
6.474	1.0	1.2-	2.15-
6.499	1.0	1.2-	2.16-
6.524	1.0	1.2-	2.17-
6.549	1.0	1.2-	2.18-
6.574	1.0	1.2-	2.19-
6.599	1.0	1.2-	2.20-
6.624	1.0	1.2-	2.21-
6.649	1.0	1.2-	2.22-
6.674	1.0	1.2-	2.23-
6.699	1.0	1.2-	2.24-
6.724	1.0	1.2-	2.25-
6.749	1.0	1.2-	2.26-
6.774	1.0	1.2-	2.27-
6.799	1.0	1.2-	2.28-
6.824	1.0	1.2-	2.29-
6.849	1.0	1.2-	2.30-
6.874	1.0	1.2-	2.31-
6.899	1.0	1.2-	2.32-
6.924	1.0	1.2-	2.33-
6.949	1.0	1.2-	2.34-
6.974	1.0	1.2-	2.35-
7.024	1.0	1.2-	2.37-
7.049	1.0	1.2-	2.38-
7.074	1.0	1.2-	2.39-
7.099	1.0	1.2-	2.40-
7.124	1.0	1.2-	2.41-
7.149	1.0	1.2-	2.42-
7.174	1.0	1.2-	2.43-
7.199	1.0	1.2-	2.44-
7.224	1.0	1.2-	2.45-
7.249	1.0	1.2-	2.46-
7.274	1.0	1.2-	2.47-
7.299	1.0	1.2-	2.48-
7.324	1.0	1.2-	2.49-
7.349	1.0	1.2-	2.50-
7.374	1.0	1.2-	2.51-
7.399	1.0	1.2-	2.52-
7.424	1.0	1.2-	2.53-
7.449	1.0	1.2-	2.54-
7.474	1.0	1.2-	2.55-
7.499	1.0	1.2-	2.56-
7.524	1.0	1.2-	2.57-
7.549	1.0	1.2-	2.58-
7.574	1.0	1.2-	2.59-
7.599	1.0	1.2-	2.60-
7.624	1.0	1.2-	2.61-
7.649	1.0	1.2-	2.62-
7.674	1.0	1.2-	2.63-
7.699	1.0	1.2-	2.64-
7.724	1.0	1.2-	2.65-
7.749	1.0	1.2-	2.66-
7.774	1.0	1.2-	2.67-
7.799	1.0	1.2-	2.68-
7.824	1.0	1.2-	2.69-
7.849	1.0	1.2-	2.70-
7.874	1.0	1.2-	2.71-
7.899	1.0	1.2-	2.72-
7.924	1.0	1.2-	2.73-
7.949	1.0	1.2-	2.74-
7.974	1.0	1.2-	2.75-
7.999	1.0	1.2-	2.76-

TABLE X

$^{17,18}O(\alpha,n)$ NEUTRON PRODUCTION IN CLEAN UO_2 FUEL BY ALPHA PARTICLES BELOW 10 MeV

E,MEV	**NEUTRONS-PER-ALPHA* O-17	O-18	TOTAL
0.000	0.	0.	0.
1.139	0.	0.	0.
1.141	0.	0.	0.
1.161	0.0	1.51-14	1.51-14
1.186	0.0	1.78-14	1.78-14
1.211	0.0	2.63-14	2.63-14
1.236	0.0	2.63-14	2.63-14
1.261	0.0	2.63-14	2.63-14
1.286	0.0	3.04-14	3.04-14
1.309	0.0	3.21-14	3.21-14
1.311	1.0-16	3.21-14	3.21-14
1.319	1.5-15	3.67-14	3.67-14
1.326	2.4-15	8.30-14	8.30-14
1.349	4.2-15	1.02-13	1.02-13
1.374	6.2-15	1.07-13	1.07-13
1.399	1.0-14	1.11-13	1.11-13
1.424	1.2-14	1.16-13	1.16-13
1.449	1.4-14	1.21-13	1.21-13
1.474	1.6-14	1.26-13	1.26-13
1.499	1.8-14	1.31-13	1.31-13
1.524	2.0-14	6.31-13	6.31-13
1.549	2.2-14	1.04-12	1.04-12
1.574	2.4-14	1.17-12	1.17-12
1.599	2.6-14	1.30-12	1.30-12
1.624	2.8-14	1.43-12	1.43-12
1.649	3.0-14	1.56-12	1.56-12
1.674	3.2-14	1.69-12	1.69-12
1.699	3.4-14	1.82-12	1.82-12
1.724	3.6-14	1.95-12	1.95-12
1.749	3.8-14	2.07-12	2.07-12
1.774	4.0-14	2.20-12	2.20-12
1.799	4.2-14	2.33-12	2.33-12
1.824	4.4-14	2.46-12	2.46-12
1.849	4.6-14	2.59-12	2.59-12
1.874	4.8-14	2.72-12	2.72-12
1.899	5.0-14	2.85-12	2.85-12
1.924	5.2-14	2.98-12	2.98-12
1.949	5.4-14	3.11-12	3.11-12
1.974	5.6-14	3.24-12	3.24-12
1.999	5.8-14	3.37-12	3.37-12
2.024	6.0-14	3.50-12	3.50-12
2.049	6.2-14	3.63-12	3.63-12
2.074	6.4-14	3.76-12	3.76-12
2.099	6.6-14	3.89-12	3.89-12
2.124	6.8-14	4.02-12	4.02-12
2.149	7.0-14	4.15-12	4.15-12
2.174	7.2-14	4.28-12	4.28-12
2.199	7.4-14	4.41-12	4.41-12
2.224	7.6-14	4.54-12	4.54-12
2.249	7.8-14	4.67-12	4.67-12
2.274	8.0-14	4.80-12	4.80-12
2.299	8.2-14	4.93-12	4.93-12
2.324	8.4-14	5.06-12	5.06-12
2.349	8.6-14	5.19-12	5.19-12
2.374	8.8-14	5.32-12	5.32-12
2.399	9.0-14	5.45-12	5.45-12
2.424	9.2-14	5.58-12	5.58-12
2.449	9.4-14	5.71-12	5.71-12
2.474	9.6-14	5.84-12	5.84-12
2.499	9.8-14	5.97-12	5.97-12
2.524	1.0-13	6.10-12	6.10-12
2.549	1.0-13	6.23-12	6.23-12
2.574	1.0-13	6.36-12	6.36-12
2.599	1.0-13	6.49-12	6.49-12
2.624	1.0-13	6.62-12	6.62-12
2.649	1.0-13	6.75-12	6.75-12
2.674	1.0-13	6.88-12	6.88-12
2.699	1.0-13	7.01-12	7.01-12
2.724	1.0-13	7.14-12	7.14-12
2.749	1.0-13	7.27-12	7.27-12
2.774	1.0-13	7.40-12	7.40-12
2.799	1.0-13	7.53-12	7.53-12
2.824	1.0-13	7.66-12	7.66-12
2.849	1.0-13	7.79-12	7.79-12
2.874	1.0-13	7.92-12	7.92-12
2.899	1.0-13	8.05-12	8.05-12
2.924	1.0-13	8.18-12	8.18-12
2.949	1.0-13	8.31-12	8.31-12
2.974	1.0-13	8.44-12	8.44-12
2.999	1.0-13	8.57-12	8.57-12

E,MEV	**NEUTRONS-PER-ALPHA* O-17	O-18	TOTAL
2.824	4.7-11	4.7-10	5.17-10
2.849	4.8-11	4.8-10	5.30-10
2.874	4.9-11	5.1-10	5.63-10
2.899	5.2-11	5.7-10	6.27-10
2.924	5.7-11	6.6-10	7.14-10
2.949	6.5-11	7.4-10	8.04-10
2.974	7.0-11	7.8-10	8.33-10
2.999	7.4-11	7.8-10	8.50-10
3.024	8.2-11	7.9-10	8.69-10
3.049	9.4-11	8.0-10	8.42-10
3.074	1.1-10	8.1-10	9.17-10
3.099	1.2-10	8.2-10	9.42-10
3.124	1.3-10	8.4-10	9.71-10
3.149	1.4-10	8.8-10	1.013-9
3.174	1.5-10	9.3-10	1.076-9
3.199	1.6-10	1.02-9	1.167-9
3.224	1.8-10	1.10-9	1.256-9
3.249	1.9-10	1.17-9	1.362-9
3.274	2.1-10	1.22-9	1.430-9
3.299	2.2-10	1.27-9	1.493-9
3.324	2.3-10	1.35-9	1.521-9
3.349	2.4-10	1.45-9	1.622-9
3.374	2.5-10	1.53-9	1.704-9
3.399	2.6-10	1.61-9	1.760-9
3.424	2.7-10	1.68-9	1.811-9
3.449	2.8-10	1.78-9	1.884-9
3.474	2.9-10	1.88-9	1.966-9
3.499	3.0-10	1.98-9	2.072-9
3.524	3.1-10	2.09-9	2.191-9
3.549	3.2-10	2.19-9	2.295-9
3.574	3.3-10	2.29-9	2.402-9
3.599	3.4-10	2.39-9	2.509-9
3.624	3.5-10	2.49-9	2.602-9
3.649	3.6-10	2.59-9	2.695-9
3.674	3.7-10	2.69-9	2.802-9
3.699	3.8-10	2.79-9	2.909-9
3.724	3.9-10	2.89-9	3.015-9
3.749	4.0-10	2.99-9	3.122-9
3.774	4.1-10	3.09-9	3.229-9
3.799	4.2-10	3.19-9	3.336-9
3.824	4.3-10	3.29-9	3.443-9
3.849	4.4-10	3.39-9	3.550-9
3.874	4.5-10	3.49-9	3.657-9
3.899	4.6-10	3.59-9	3.764-9
3.924	4.7-10	3.69-9	3.871-9
3.949	4.8-10	3.79-9	3.978-9
3.974	4.9-10	3.89-9	4.085-9
3.999	5.0-10	3.99-9	4.192-9
4.024	5.1-10	4.09-9	4.299-9
4.049	5.2-10	4.19-9	4.406-9
4.074	5.3-10	4.29-9	4.513-9
4.099	5.4-10	4.39-9	4.620-9
4.124	5.5-10	4.49-9	4.727-9
4.149	5.6-10	4.59-9	4.834-9
4.174	5.7-10	4.69-9	4.941-9
4.199	5.8-10	4.79-9	5.048-9
4.224	5.9-10	4.89-9	5.155-9
4.249	6.0-10	4.99-9	5.262-9
4.274	6.1-10	5.09-9	5.369-9
4.299	6.2-10	5.19-9	5.476-9
4.324	6.3-10	5.29-9	5.583-9
4.349	6.4-10	5.39-9	5.690-9
4.374	6.5-10	5.49-9	5.797-9
4.399	6.6-10	5.59-9	5.904-9
4.424	6.7-10	5.69-9	6.011-9
4.449	6.8-10	5.79-9	6.118-9
4.474	6.9-10	5.89-9	6.225-9
4.499	7.0-10	5.99-9	6.332-9
4.524	7.1-10	6.09-9	6.439-9
4.549	7.2-10	6.19-9	6.546-9
4.574	7.3-10	6.29-9	6.653-9
4.599	7.4-10	6.39-9	6.760-9

E,MEV	**NEUTRONS-PER-ALPHA* O-17	O-18	TOTAL
4.624	8.9-10	1.07-8	1.158-8
4.649	9.1-10	1.10-8	1.192-8
4.674	9.2-10	1.13-8	1.226-8
4.699	9.4-10	1.16-8	1.257-8
4.724	9.6-10	1.18-8	1.281-8
4.749	9.8-10	1.21-8	1.303-8
4.774	1.0-9	1.23-8	1.327-8
4.799	1.01-9	1.24-8	1.341-8
4.824	1.02-9	1.26-8	1.355-8
4.849	1.04-9	1.27-8	1.377-8
4.874	1.06-9	1.30-8	1.403-8
4.899	1.07-9	1.33-8	1.434-8
4.924	1.09-9	1.35-8	1.459-8
4.949	1.10-9	1.36-8	1.486-8
4.974	1.12-9	1.38-8	1.519-8
4.999	1.13-9	1.40-8	1.540-8
5.024	1.19-9	1.46-8	1.602-8
5.049	1.21-9	1.48-8	1.632-8
5.074	1.22-9	1.50-8	1.650-8
5.099	1.24-9	1.52-8	1.677-8
5.124	1.27-9	1.55-8	1.705-8
5.149	1.29-9	1.57-8	1.729-8
5.174	1.32-9	1.60-8	1.760-8
5.199	1.34-9	1.63-8	1.796-8
5.224	1.36-9	1.66-8	1.829-8
5.249	1.38-9	1.68-8	1.856-8
5.274	1.40-9	1.71-8	1.881-8
5.299	1.42-9	1.74-8	1.916-8
5.324	1.45-9	1.77-8	1.952-8
5.349	1.47-9	1.80-8	1.956-8
5.374	1.49-9	1.83-8	2.001-8
5.399	1.51-9	1.86-8	2.025-8
5.424	1.53-9	1.90-8	2.061-8
5.449	1.55-9	1.93-8	2.096-8
5.474	1.57-9	1.97-8	2.132-8
5.499	1.59-9	2.00-8	2.168-8
5.524	1.73-9	2.03-8	2.204-8
5.549	1.76-9	2.06-8	2.240-8
5.574	1.79-9	2.10-8	2.277-8
5.599	1.82-9	2.13-8	2.314-8
5.624	1.85-9	2.17-8	2.351-8
5.649	1.88-9	2.20-8	2.388-8
5.674	1.91-9	2.23-8	2.426-8
5.699	1.94-9	2.27-8	2.464-8
5.724	1.98-9	2.30-8	2.502-8
5.749	2.01-9	2.34-8	2.540-8
5.774	2.04-9	2.37-8	2.578-8
5.799	2.07-9	2.41-8	2.617-8
5.824	2.10-9	2.45-8	2.656-8
5.849	2.14-9	2.48-8	2.696-8
5.874	2.17-9	2.52-8	2.735-8
5.899	2.20-9	2.56-8	2.775-8
5.924	2.24-9	2.59-8	2.815-8
5.949	2.27-9	2.63-8	2.855-8
5.974	2.30-9	2.67-8	2.896-8
5.999	2.34-9	2.70-8	2.937-8
6.024	2.37-9	2.74-8	2.978-8
6.049	2.41-9	2.78-8	3.019-8
6.074	2.44-9	2.82-8	3.060-8
6.099	2.48-9	2.86-8	3.102-8
6.124	2.51-9	2.89-8	3.144-8
6.149	2.55-9	2.93-8	3.186-8
6.174	2.58-9	2.97-8	3.229-8
6.199	2.62-9	3.01-8	3.271-8
6.224	2.66-9	3.05-8	3.314-8
6.249	2.69-9	3.09-8	3.358-8
6.274	2.73-9	3.14-8	3.401-8
6.299	2.77-9	3.18-8	3.445-8
6.324	2.81-9	3.22-8	3.489-8
6.349	2.85-9	3.26-8	3.533-8
6.374	2.89-9	3.30-8	3.577-8
6.399	2.92-9	3.35-8	3.622-8

E,MEV	**NEUTRONS-PER-ALPHA* O-17	O-18	TOTAL
6.424	2.96-9	3.39-8	3.667-8
6.449	3.00-9	3.44-8	3.714-8
6.474	3.03-9	3.48-8	3.757-8
6.499	3.07-9	3.53-8	3.803-8
6.524	3.11-9	3.57-8	3.849-8
6.549	3.15-9	3.62-8	3.895-8
6.574	3.19-9	3.66-8	3.941-8
6.599	3.23-9	3.71-8	3.988-8
6.624	3.27-9	3.75-8	4.035-8
6.649	3.31-9	3.80-8	4.082-8
6.674	3.35-9	3.84-8	4.130-8
6.699	3.40-9	3.89-8	4.177-8
6.724	3.44-9	3.94-8	4.225-8
6.749	3.48-9	3.98-8	4.273-8
6.774	3.52-9	4.03-8	4.322-8
6.799	3.56-9	4.07-8	4.370-8
6.824	3.61-9	4.12-8	4.419-8
6.849	3.65-9	4.17-8	4.468-8
6.874	3.69-9	4.21-8	4.517-8
6.899	3.74-9	4.26-8	4.567-8
6.924	3.78-9	4.31-8	4.617-8
6.949	3.82-9	4.36-8	4.667-8
6.974	3.87-9	4.40-8	4.717-8
6.999	3.91-9	4.45-8	4.768-8
7.024	3.96-9	4.50-8	4.819-8
7.049	4.00-9	4.55-8	4.870-8
7.074	4.05-9	4.60-8	4.921-8
7.099	4.09-9	4.65-8	4.973-8
7.124	4.14-9	4.70-8	5.025-8
7.149	4.19-9	4.75-8	5.077-8
7.174	4.23-9	4.80-8	5.129-8
7.199	4.28-9	4.85-8	5.182-8
7.224	4.33-9	4.90-8	5.234-8
7.249	4.37-9	4.95-8	5.288-8
7.274	4.42-9	5.00-8	5.341-8
7.299	4.47-9	5.05-8	5.395-8
7.324	4.52-9	5.10-8	5.448-8
7.349	4.57-9	5.15-8	5.502-8
7.374	4.62-9	5.20-8	5.557-8
7.399	4.67-9	5.25-8	5.611-8
7.424	4.72-9	5.30-8	5.666-8
7.449	4.77-9	5.35-8	5.721-8
7.474	4.82-9	5.40-8	5.777-8
7.499	4.87-9	5.45-8	5.832-8
7.524	4.92-9	5.50-8	5.888-8
7.549	4.97-9	5.55-8	5.944-8
7.574	5.02-9	5.60-8	6.001-8
7.599	5.07-9	5.65-8	6.057-8
7.624	5.13-9	5.71-8	6.114-8
7.649	5.18-9	5.76-8	6.171-8
7.674	5.23-9	5.81-8	6.229-8
7.699	5.29-9	5.86-8	6.286-8
7.724	5.34-9	5.92-8	6.344-8
7.749	5.39-9	5.97-8	6.403-8
7.774	5.45-9	6.03-8	6.461-8
7.799	5.50-9	6.08-8	6.520-8
7.824	5.56-9	6.14-8	6.579-8
7.849	5.61-9	6.19-8	6.638-8
7.874	5.67-9	6.25-8	6.697-8
7.899	5.72-9	6.31-8	6.757-8
7.924	5.78-9	6.36-8	6.817-8
7.949	5.84-9	6.42-8	6.877-8
7.974	5.89-9	6.48-8	6.938-8
7.999	5.95-9	6.54-8	6.998-8
8.024	6.01-9	6.59-8	7.059-8
8.049	6.07-9	6.65-8	7.120-8
8.074	6.12-9	6.71-8	7.182-8
8.099	6.18-9	6.77-8	7.244-8
8.124	6.24-9	6.83-8	7.306-8
8.149	6.30-9	6.89-8	7.368-8
8.174	6.36-9	6.95-8	7.431-8
8.199	6.42-9	7.01-8	7.494-8

E,MEV	**NEUTRONS-PER-ALPHA* O-17	O-18	TOTAL
8.224	6.48-9	7.07-8	7.557-8
8.249	6.54-9	7.13-8	7.620-8
8.274	6.60-9	7.19-8	7.684-8
8.299	6.66-9	7.25-8	7.748-8
8.324	6.72-9	7.31-8	7.812-8
8.349	6.78-9	7.37-8	7.876-8
8.374	6.84-9	7.43-8	7.941-8
8.399	6.90-9	7.50-8	8.006-8
8.424	6.97-9	7.57-8	8.071-8
8.449	7.03-9	7.63-8	8.137-8
8.474	7.09-9	7.70-8	8.203-8
8.499	7.15-9	7.76-8	8.269-8
8.524	7.22-9	7.83-8	8.335-8
8.549	7.28-9	7.89-8	8.401-8
8.574	7.34-9	7.96-8	8.468-8
8.599	7.41-9	8.03-8	8.535-8
8.624	7.47-9	8.09-8	8.602-8
8.649	7.53-9	8.16-8	8.670-8
8.674	7.60-9	8.23-8	8.737-8
8.699	7.66-9	8.30-8	8.805-8
8.724	7.73-9	8.36-8	8.873-8
8.749	7.79-9	8.43-8	8.941-8
8.774	7.86-9	8.50-8	9.009-8
8.799	7.92-9	8.57-8	9.078-8
8.824	7.99-9	8.64-8	9.146-8
8.849	8.06-9	8.71-8	9.215-8
8.874	8.12-9	8.78-8	9.284-8
8.899	8.19-9	8.85-8	9.353-8
8.924	8.26-9	8.92-8	9.423-8
8.949	8.32-9	8.99-8	9.492-8
8.974	8.39-9	9.06-8	9.562-8
8.999	8.46-9	9.13-8	9.632-8
9.024	8.52-9	9.20-8	9.702-8
9.049	8.59-9	9.27-8	9.772-8
9.074	8.66-9	9.34-8	9.843-8
9.099	8.73-9	9.41-8	9.913-8
9.124	8.80-9	9.48-8	9.984-8
9.149	8.87-9	9.55-8	1.005-7
9.174	8.93-9	9.62-8	1.012-7
9.199	9.00-9	9.69-8	1.019-7
9.224	9.07-9	9.76-8	1.026-7
9.249	9.14-9	9.83-8	1.033-7
9.274	9.21-9	9.90-8	1.040-7
9.299	9.28-9	9.97-8	1.047-7
9.324	9.35-9	1.00-7	1.054-7
9.349	9.42-9	1.01-7	1.061-7
9.374	9.49-9	1.02-7	1.068-7
9.399	9.56-9	1.03-7	1.075-7
9.424	9.63-9	1.04-7	1.082-7
9.449	9.70-9	1.05-7	1.089-7
9.474	9.77-9	1.06-7	1.096-7
9.499	9.84-9	1.07-7	1.103-7
9.524	9.91-9	1.08-7	1.110-7
9.549	9.98-9	1.09-7	1.117-7
9.574	1.01-8	1.10-7	1.124-7
9.599	1.02-8	1.11-7	1.131-7
9.624	1.03-8	1.12-7	1.138-7
9.649	1.04-8	1.13-7	1.145-7
9.674	1.05-8	1.14-7	1.152-7
9.699	1.06-8	1.15-7	1.159-7
9.724	1.07-8	1.16-7	1.166-7
9.749	1.08-8	1.17-7	1.173-7
9.774	1.09-8	1.18-7	1.180-7
9.799	1.10-8	1.19-7	1.187-7
9.824	1.11-8	1.20-7	1.194-7
9.849	1.12-8	1.21-7	1.201-7
9.874	1.13-8	1.22-7	1.208-7
9.899	1.14-8	1.23-7	1.215-7
9.924	1.15-8	1.24-7	1.222-7
9.949	1.16-8	1.25-7	1.229-7
9.974	1.17-8	1.26-7	1.236-7
9.999	1.18-8	1.27-7	1.243-7

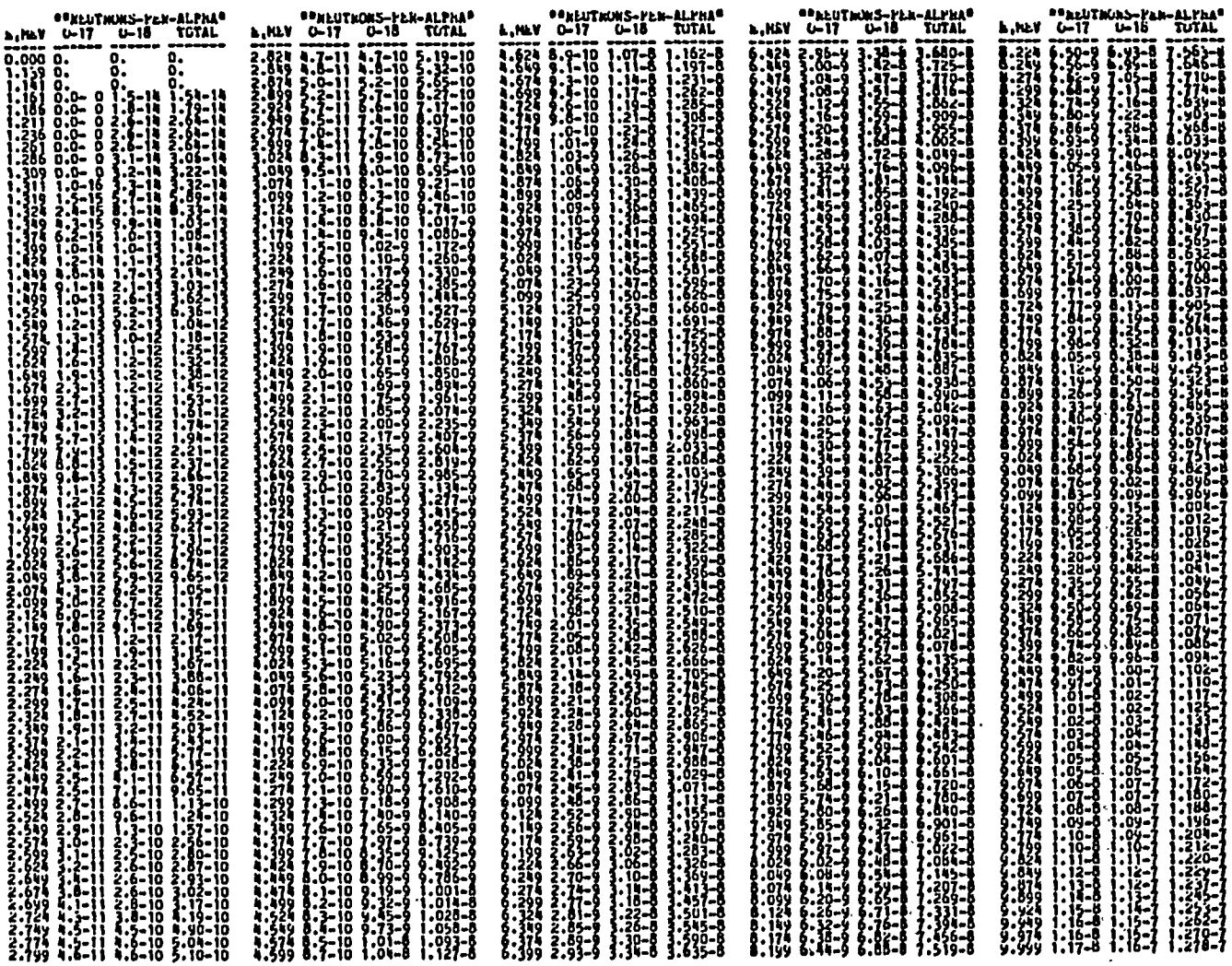
TABLE XI

$^{17,18}O(\alpha,n)$ NEUTRON PRODUCTION IN SPENT UO_2 FUEL BY ALPHA PARTICLES BELOW 10 MeV

E, MeV	**NEUTRONS-PER-ALPHA** O-17	O-18	TOTAL
0.000	0.	0.	0.
1.139	0.	0.	0.
1.161	0.	0.	0.
1.161	0.0	0	1.52-##
1.186	0.0	0	2.59-##
1.211	0.0	0	2.59-##
1.236	0.0	0	2.59-##
1.261	0.0	0	2.59-##
1.286	0.0	0	3.01-##
1.309	0.0	0	3.17-##
1.311	1.0	15	3.67-##
1.319	1.0	15	3.60-##
1.324	2.2	15	0.20-##
1.349	4.2	15	1.01-13
1.374	6.1	15	1.06-13
1.399	1.0	14	1.12-13
1.424	1.2	14	1.18-13
1.449	8.7	13	2.11-13
1.474	9.0	13	2.99-13
1.499	1.0	13	3.57-13
1.524	1.1	13	6.26-13
1.549	1.2	13	1.03-12
1.574	1.3	13	1.16-12
1.599	1.4	13	1.23-12
1.624	1.6	13	1.30-12
1.649	1.9	13	1.36-12
1.674	2.2	13	1.43-12
1.699	2.5	13	1.50-12
1.724	3.2	13	1.50-12
1.749	4.4	13	1.71-12
1.774	5.4	13	1.91-12
1.799	7.1	13	2.17-12
1.824	9.6	13	2.31-12
1.849	1.3	12	2.62-12
1.874	1.7	12	2.91-12
1.899	2.2	12	3.31-12
1.924	2.9	12	3.62-12
1.949	3.8	12	4.03-12
1.974	4.9	12	4.50-12
1.999	6.4	12	5.06-12
2.024	8.3	12	5.62-12
2.049	1.1	11	6.50-12
2.074	1.4	11	7.24-12
2.099	1.8	11	8.06-12
2.124	2.3	11	8.96-12
2.149	2.9	11	9.96-12
2.174	3.7	11	1.11-11
2.199	4.7	11	1.23-11
2.224	6.0	11	1.37-11
2.249	7.6	11	1.52-11
2.274	9.7	11	1.69-11
2.299	1.2	10	1.88-11
2.324	1.5	10	2.09-11
2.349	1.9	10	2.32-11
2.374	2.4	10	2.58-11
2.399	3.0	10	2.87-11
2.424	3.8	10	3.19-11
2.449	4.8	10	3.55-11
2.474	6.0	10	3.95-11
2.499	7.5	10	4.40-11
2.524	9.4	10	4.90-11
2.549	1.2	9	5.46-11
2.574	1.5	9	6.07-11
2.599	1.8	9	6.76-11
2.624	2.3	9	7.52-11
2.649	2.8	9	8.37-11
2.674	3.5	9	9.31-11
2.699	4.4	9	1.04-10
2.724	5.4	9	1.15-10
2.749	6.8	9	1.28-10
2.774	8.4	9	1.42-10
2.799	1.0	8	1.58-10
2.824	1.3	8	1.75-10
2.849	1.6	8	1.95-10
2.874	2.0	8	2.17-10
2.899	2.5	8	2.41-10
2.924	3.1	8	2.68-10
2.949	3.8	8	2.98-10
2.974	4.7	8	3.31-10
2.999	5.8	8	3.68-10
3.024	7.2	8	4.09-10
3.049	8.9	8	4.54-10
3.074	1.1	7	5.04-10
3.099	1.4	7	5.59-10
3.124	1.7	7	6.20-10
3.149	2.1	7	6.87-10
3.174	2.6	7	7.62-10
3.199	3.2	7	8.45-10
3.224	3.9	7	9.36-10
3.249	4.8	7	1.03-9
3.274	5.9	7	1.14-9
3.299	7.2	7	1.26-9
3.324	8.8	7	1.39-9
3.349	1.1	6	1.53-9
3.374	1.3	6	1.69-9
3.399	1.6	6	1.86-9
3.424	1.9	6	2.05-9
3.449	2.3	6	2.26-9
3.474	2.8	6	2.49-9
3.499	3.4	6	2.74-9
3.524	4.1	6	3.02-9
3.549	5.0	6	3.32-9
3.574	6.1	6	3.65-9
3.599	7.4	6	4.02-9
3.624	9.0	6	4.42-9
3.649	1.1	5	4.86-9
3.674	1.3	5	5.34-9
3.699	1.6	5	5.87-9
3.724	1.9	5	6.45-9
3.749	2.3	5	7.09-9
3.774	2.8	5	7.79-9
3.799	3.4	5	8.56-9
3.824	4.1	5	9.41-9
3.849	5.0	5	1.03-8
3.874	6.1	5	1.13-8
3.899	7.4	5	1.24-8
3.924	9.0	5	1.36-8
3.949	1.1	4	1.49-8
3.974	1.3	4	1.64-8
3.999	1.6	4	1.80-8
4.024	1.9	4	1.98-8
4.049	2.3	4	2.17-8
4.074	2.8	4	2.39-8
4.099	3.4	4	2.62-8
4.124	4.1	4	2.88-8
4.149	5.0	4	3.16-8
4.174	6.1	4	3.47-8
4.199	7.4	4	3.81-8
4.224	9.0	4	4.18-8
4.249	1.1	3	4.59-8
4.274	1.3	3	5.04-8
4.299	1.6	3	5.53-8
4.324	1.9	3	6.07-8
4.349	2.3	3	6.66-8
4.374	2.8	3	7.31-8
4.399	3.4	3	8.02-8
4.424	4.1	3	8.80-8
4.449	5.0	3	9.66-8
4.474	6.1	3	1.06-7
4.499	7.4	3	1.16-7
4.524	9.0	3	1.28-7
4.549	1.1	2	1.40-7
4.574	1.3	2	1.54-7
4.599	1.6	2	1.69-7
4.624	1.9	2	1.86-7
4.649	2.3	2	2.04-7
4.674	2.8	2	2.24-7
4.699	3.4	2	2.46-7
4.724	4.1	2	2.70-7
4.749	5.0	2	2.97-7
4.774	6.1	2	3.26-7
4.799	7.4	2	3.58-7
4.824	9.0	2	3.93-7
4.849	1.1	1	4.32-7
4.874	1.3	1	4.74-7
4.899	1.6	1	5.21-7
4.924	1.9	1	5.72-7
4.949	2.3	1	6.28-7
4.974	2.8	1	6.90-7
4.999	3.4	1	7.58-7
5.024	4.1	1	8.33-7
5.049	5.0	1	9.15-7
5.074	6.1	1	1.01-6
5.099	7.4	1	1.10-6
5.124	9.0	1	1.21-6
5.149	1.1	0	1.32-6
5.174	1.3	0	1.45-6
5.199	1.6	0	1.59-6
5.224	1.9	0	1.74-6
5.249	2.3	0	1.91-6
5.274	2.8	0	2.10-6
5.299	3.4	0	2.30-6
5.324	4.1	0	2.53-6
5.349	5.0	0	2.78-6
5.374	6.1	0	3.05-6
5.399	7.4	0	3.36-6
5.424	9.0	0	3.69-6
5.449	1.1	-1	4.06-6
5.474	1.3	-1	4.46-6
5.499	1.6	-1	4.90-6
5.524	1.9	-1	5.38-6
5.549	2.3	-1	5.91-6
5.574	2.8	-1	6.49-6
5.599	3.4	-1	7.13-6
5.624	4.1	-1	7.83-6
5.649	5.0	-1	8.60-6
5.674	6.1	-1	9.45-6
5.699	7.4	-1	1.04-5
5.724	9.0	-1	1.14-5
5.749	1.1	-2	1.25-5
5.774	1.3	-2	1.38-5
5.799	1.6	-2	1.52-5
5.824	1.9	-2	1.67-5
5.849	2.3	-2	1.84-5
5.874	2.8	-2	2.03-5
5.899	3.4	-2	2.23-5
5.924	4.1	-2	2.46-5
5.949	5.0	-2	2.71-5
5.974	6.1	-2	2.99-5
5.999	7.4	-2	3.29-5
6.024	9.0	-2	3.63-5
6.049	1.1	-3	4.00-5
6.074	1.3	-3	4.41-5
6.099	1.6	-3	4.86-5
6.124	1.9	-3	5.36-5
6.149	2.3	-3	5.91-5
6.174	2.8	-3	6.52-5
6.199	3.4	-3	7.19-5
6.224	4.1	-3	7.93-5
6.249	5.0	-3	8.75-5
6.274	6.1	-3	9.65-5
6.299	7.4	-3	1.06-4
6.324	9.0	-3	1.17-4
6.349	1.1	-4	1.29-4
6.374	1.3	-4	1.42-4
6.399	1.6	-4	1.57-4
6.424	1.9	-4	1.73-4
6.449	2.3	-4	1.91-4
6.474	2.8	-4	2.11-4
6.499	3.4	-4	2.33-4
6.524	4.1	-4	2.57-4
6.549	5.0	-4	2.84-4
6.574	6.1	-4	3.13-4
6.599	7.4	-4	3.46-4
6.624	9.0	-4	3.82-4
6.649	1.1	-5	4.22-4
6.674	1.3	-5	4.66-4
6.699	1.6	-5	5.15-4
6.724	1.9	-5	5.69-4
6.749	2.3	-5	6.28-4
6.774	2.8	-5	6.94-4
6.799	3.4	-5	7.66-4
6.824	4.1	-5	8.46-4
6.849	5.0	-5	9.34-4
6.874	6.1	-5	1.03-3
6.899	7.4	-5	1.14-3
6.924	9.0	-5	1.26-3
6.949	1.1	-6	1.39-3
6.974	1.3	-6	1.53-3
6.999	1.6	-6	1.69-3
7.024	1.9	-6	1.87-3
7.049	2.3	-6	2.06-3
7.074	2.8	-6	2.28-3
7.099	3.4	-6	2.52-3
7.124	4.1	-6	2.78-3
7.149	5.0	-6	3.07-3
7.174	6.1	-6	3.39-3
7.199	7.4	-6	3.74-3
7.224	9.0	-6	4.13-3
7.249	1.1	-7	4.56-3
7.274	1.3	-7	5.03-3
7.299	1.6	-7	5.55-3
7.324	1.9	-7	6.12-3
7.349	2.3	-7	6.75-3
7.374	2.8	-7	7.45-3
7.399	3.4	-7	8.22-3
7.424	4.1	-7	9.07-3
7.449	5.0	-7	1.00-2
7.474	6.1	-7	1.10-2
7.499	7.4	-7	1.22-2
7.524	9.0	-7	1.34-2
7.549	1.1	-8	1.48-2
7.574	1.3	-8	1.63-2
7.599	1.6	-8	1.80-2
7.624	1.9	-8	1.98-2
7.649	2.3	-8	2.19-2
7.674	2.8	-8	2.41-2
7.699	3.4	-8	2.66-2
7.724	4.1	-8	2.93-2
7.749	5.0	-8	3.24-2
7.774	6.1	-8	3.57-2
7.799	7.4	-8	3.94-2
7.824	9.0	-8	4.35-2
7.849	1.1	-9	4.80-2
7.874	1.3	-9	5.29-2
7.899	1.6	-9	5.84-2
7.924	1.9	-9	6.44-2
7.949	2.3	-9	7.11-2
7.974	2.8	-9	7.84-2
7.999	3.4	-9	8.65-2
8.024	4.1	-9	9.54-2
8.049	5.0	-9	1.05-1
8.074	6.1	-9	1.16-1
8.099	7.4	-9	1.28-1
8.124	9.0	-9	1.41-1
8.149	1.1	-10	1.56-1
8.174	1.3	-10	1.72-1
8.199	1.6	-10	1.90-1
8.224	1.9	-10	2.10-1
8.249	2.3	-10	2.32-1
8.274	2.8	-10	2.56-1
8.299	3.4	-10	2.83-1
8.324	4.1	-10	3.12-1
8.349	5.0	-10	3.45-1
8.374	6.1	-10	3.81-1
8.399	7.4	-10	4.21-1
8.424	9.0	-10	4.65-1
8.449	1.1	-11	5.14-1
8.474	1.3	-11	5.68-1
8.499	1.6	-11	6.27-1
8.524	1.9	-11	6.93-1
8.549	2.3	-11	7.65-1
8.574	2.8	-11	8.45-1
8.599	3.4	-11	9.33-1
8.624	4.1	-11	1.03-0
8.649	5.0	-11	1.14-0
8.674	6.1	-11	1.26-0
8.699	7.4	-11	1.39-0
8.724	9.0	-11	1.53-0
8.749	1.1	-12	1.69-0
8.774	1.3	-12	1.87-0
8.799	1.6	-12	2.06-0
8.824	1.9	-12	2.27-0
8.849	2.3	-12	2.51-0
8.874	2.8	-12	2.77-0
8.899	3.4	-12	3.06-0
8.924	4.1	-12	3.37-0
8.949	5.0	-12	3.72-0
8.974	6.1	-12	4.10-0
8.999	7.4	-12	4.52-0
9.024	9.0	-12	4.98-0
9.049	1.1	-13	5.49-0
9.074	1.3	-13	6.05-0
9.099	1.6	-13	6.67-0
9.124	1.9	-13	7.35-0
9.149	2.3	-13	8.10-0
9.174	2.8	-13	8.93-0
9.199	3.4	-13	9.84-0
9.224	4.1	-13	1.08-1
9.249	5.0	-13	1.19-1
9.274	6.1	-13	1.31-1
9.299	7.4	-13	1.44-1
9.324	9.0	-13	1.59-1
9.349	1.1	-14	1.75-1
9.374	1.3	-14	1.93-1
9.399	1.6	-14	2.13-1
9.424	1.9	-14	2.35-1
9.449	2.3	-14	2.59-1
9.474	2.8	-14	2.86-1
9.499	3.4	-14	3.15-1
9.524	4.1	-14	3.48-1
9.549	5.0	-14	3.84-1
9.574	6.1	-14	4.24-1
9.599	7.4	-14	4.68-1
9.624	9.0	-14	5.17-1
9.649	1.1	-15	5.71-1
9.674	1.3	-15	6.31-1
9.699	1.6	-15	6.97-1
9.724	1.9	-15	7.70-1
9.749	2.3	-15	8.51-1
9.774	2.8	-15	9.40-1
9.799	3.4	-15	1.04-0
9.824	4.1	-15	1.15-0
9.849	5.0	-15	1.27-0
9.874	6.1	-15	1.40-0
9.899	7.4	-15	1.55-0
9.924	9.0	-15	1.71-0
9.949	1.1	-16	1.89-0
9.974	1.3	-16	2.09-0
9.999	1.6	-16	2.31-0

TABLE XII

$^{17,18}O(\alpha,n)$ NEUTRON PRODUCTION IN CLEAN $(U,Pu)O_2$ FUEL BY ALPHA PARTICLES BELOW 10 MeV



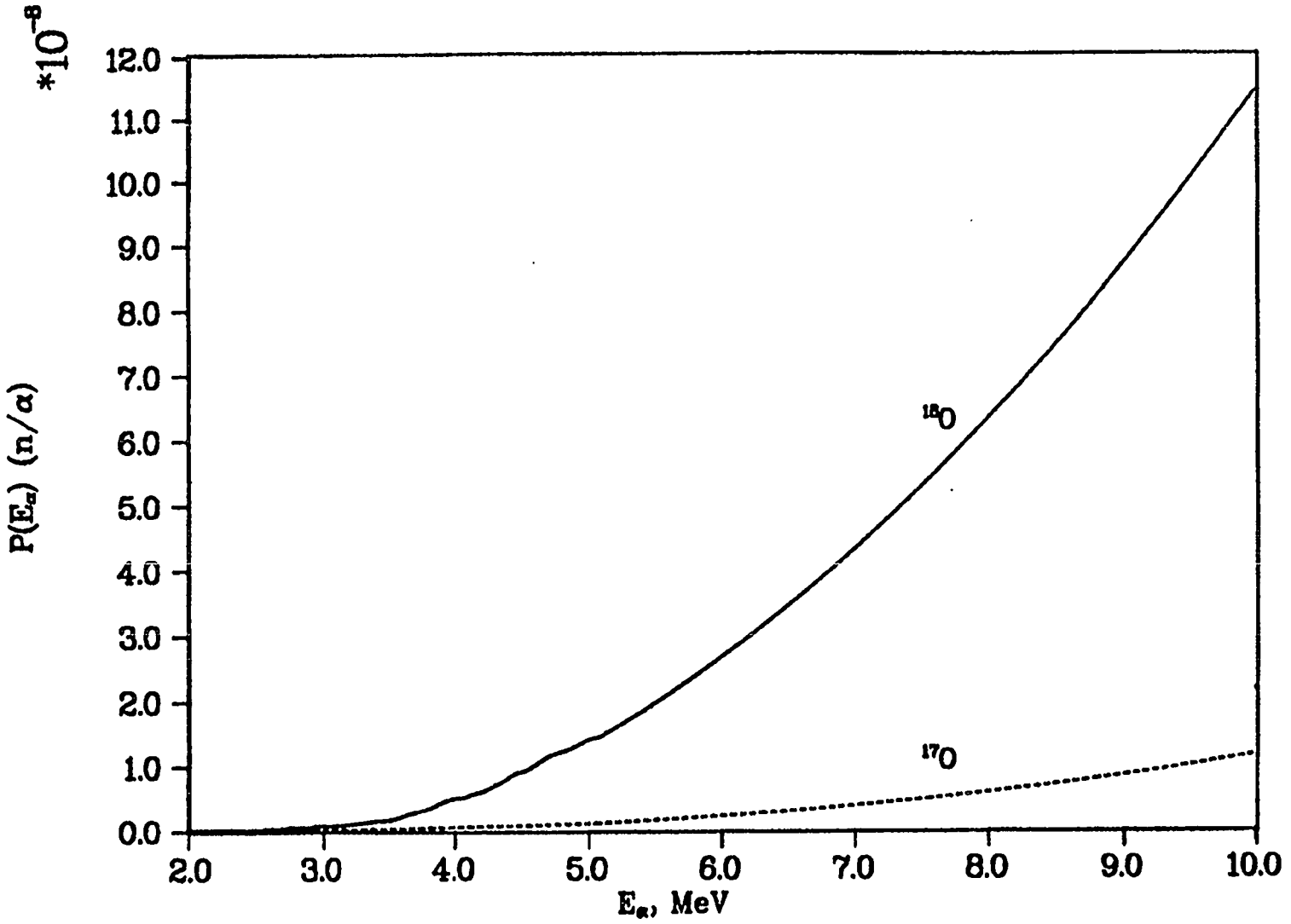


Fig. 5.
$^{17,18}O(\alpha,n)$ neutron production by decay alphas in LWR irradiated UO_2 fuel.

TABLE XIII

NEUTRON PRODUCTION FROM ACTINIDE DECAY IN UO_2 FUEL

NUCLIDE	HALF-LIFE (SECONDS)	DECAY ENERGY (MEV)	DE-CAY REF	◆◆◆◆NEUTRONS PER DECAY◆◆◆◆ ALPHA,N IN UO2	SPONT. FISSION	TOTAL
80-HG-206	4.89000+2	0.5274	A	0.	0.	0.
81-TL-206	2.50980+2	0.5402	A	0.	0.	0.
82-PB-206	STABLE	0.	-	0.	0.	0.
81-TL-207	2.87400+2	0.5194	A	0.	0.	0.
82-PB-207	STABLE	0.	-	0.	0.	0.
81-TL-208	1.84200+2	3.9702	B	0.	0.	0.
82-PB-208	STABLE	0.	-	0.	0.	0.
81-TL-209	1.32000+2	2.8315	A	0.	0.	0.
82-PB-209	1.17108+4	0.2234	A	0.	0.	0.
83-BI-209	6.3115+25	0.	A	0.	0.	0.
81-TL-210	7.80000+1	4.2765	A	0.	0.	7.00 -05
82-PB-210	7.02472+8	0.0441	A	5.68 -17	0.	5.68 -17
83-BI-210	4.33123+5	0.3899	A	1.56 -14	0.	1.16 -14
84-PO-210	1.19557+7	5.4076	A	1.87 -08	0.	1.87 -08
82-PB-211	2.16600+3	0.5353	A	0.	0.	0.
83-BI-211	1.29000+2	6.7881	A	3.88 -08	0.	3.88 -08
84-PO-211	0.5160000	7.5942	A	5.64 -08	0.	5.64 -08
82-PB-212	3.83040+4	0.3180	B	0.	0.	0.
83-BI-212	3.63600+3	2.9030	A	1.076-08	0.	1.076-08
84-PO-212	2.96000-7	8.9536	A	8.94 -08	0.	8.94 -08
83-BI-213	2.73540+3	0.7172	A	5.85 -10	0.	5.85 -10
84-PO-213	4.20000-6	8.5360	A	7.86 -08	0.	7.86 -08
82-PB-214	1.60800+3	0.5389	A	0.	0.	0.
83-BI-214	1.18200+3	2.1923	A	4.39 -12	0.	4.39 -12
84-PO-214	1.63700-4	7.8337	A	6.19 -08	0.	6.19 -08
83-BI-215	4.44000+2	0.8445	A	0.	0.	0.
84-PO-215	1.77800-3	7.5265	A	5.52 -08	0.	5.52 -08
85-AT-215	1.00000-4	8.1780	A	6.98 -08	0.	6.98 -08
84-PO-216	0.1500000	6.9064	B	4.28 -08	0.	4.28 -08
85-AT-217	0.0323000	7.2004	A	4.85 -08	0.	4.85 -08
86-RN-217	5.40000-4	7.8880	A	6.32 -08	0.	6.32 -08
84-PO-218	1.83000+2	6.1149	A	2.909-08	0.	2.909-08
85-AT-218	1.7500000	6.8830	A	4.14 -08	0.	4.14 -08
86-RN-218	0.0350000	7.2664	A	4.99 -08	0.	4.99 -08
85-AT-219	5.40000+1	6.2165	A	3.26 -08	0.	3.26 -08
86-RN-219	3.9600000	6.9463	A	4.25 -08	0.	4.25 -08
86-RN-220	5.56000+1	6.4048	B	3.39 -08	0.	3.39 -08
87-FR-221	2.88000+2	6.4580	A	3.45 -08	0.	3.45 -08
86-RN-222	3.30351+5	5.5905	A	2.129-08	0.	2.129-08
87-FR-222	8.64000+2	0.7450	A	2.45 -11	0.	2.45 -11
88-RA-222	3.80000+1	6.6760	A	3.846-08	0.	3.846-08
87-FR-223	1.30800+3	0.4559	A	7.65 -13	0.	7.65 -13
88-RA-223	9.87949+5	------	A	2.39 -08	0.	2.39 -08
88-RA-224	3.16224+5	5.7903	B	2.40 -08	0.	2.40 -08
88-RA-225	1.27872+6	0.1433	A	0.	0.	0.
89-AC-225	8.64000+5	5.9354	A	2.57 -08	0.	2.57 -08
88-RA-226	5.0461+10	4.8708	A	1.304-08	0.	1.304-08
89-AC-226	1.04400+5	0.4099	A	1.24 -12	0.	1.24 -12

TABLE XIII (cont.)

NUCLIDE	HALF-LIFE (SECONDS)	DECAY ENERGY (MEV)	DE-CAY REF	◆◆◆◆NEUTRONS PER DECAY◆◆◆◆ ALPHA,N IN UO2	SPONT. FISSION	TOTAL
90-TH-226	1.85400+3	6.4517	A	3.42 -08	0.	3.42 -08
89-AC-227	6.87097+8	0.0878	A	2.01 -10	0.	2.01 -10
90-TH-227	1.61720+6	6.1466	A	2.72 -08	0.	2.72 -08
88-RA-228	1.82087+8	0.0146	A	0.	0.	0.
89-AC-228	2.20680+4	1.3696	A	0.	0.	0.
90-TH-228	6.03725+7	5.5176	B	2.004-08	0.	2.004-08
90-TH-229	2.3163+11	5.1686	A	1.391-08	0.	1.391-08
90-TH-230	2.4299+12	4.7609	B	1.207-08	1.14 -12	1.21 -08
91-PA-230	1.52928+6	0.6577	A	6.03 -13	0.	6.03 -13
92- U-230	1.79712+6	5.9928	A	2.69 -08	0.	2.69 -08
90-TH-231	9.18720+4	0.1537	B	0.	0.	0.
91-PA-231	1.0338+12	5.0601	B	1.478-08	5.75 -12	1.48 -08
92- U-231	3.62880+5	0.1017	A	1.14 -12	0.	1.14 -12
90-TH-232	4.4337+17	4.0862	B	5.52 -09	3.02 -11	5.55 -09
91-PA-232	1.13184+5	1.098	B	0.	0.	0.
92- U-232	2.26263+9	5.4145	B	1.871-08	1.54 -12	1.87 -08
90-TH-233	1.33800+3	0.4422	B	0.	0.	0.
91-PA-233	2.33280+6	0.4080	B	0.	0.	0.
92- U-233	5.0232+12	4.8978	B	1.336-08	2.29 -12	1.34 -08
90-TH-234	2.08233+6	0.1473	A	0.	0.	0.
91-PA-234	2.43000+4	2.2453	A	0.	0.	0.
91-PA-234M	7.05000+1	0.8141	A	0.	0.	0.
92- U-234	7.7188+12	4.8685	B	1.299-08	2.17 -11	1.301-08
90-TH-235	4.14000+2	------	A	0.	0.	0.
91-PA-235	1.45200+3	------	A	0.	0.	0.
92- U-235	2.2210+16	4.6651	B	8.89 -09	3.74 -09	1.26 -08
92- U-235M	1.48080+3	0.0001	A	0.	0.	0.
93-NP-235	3.42230+7	0.0810	A	2.44 -13	0.	2.44 -13
94-PU-235	1.53600+3	5.8675	A	3.48 -12	0.	3.48 -12
92- U-236	7.3890+14	4.5809	B	9.89 -09	2.29 -09	1.218-08
93-NP-236	3.6290+12	0.3390	B	0.	0.	0.
93-NP-236M	8.10000+4	0.1353	B	0.	0.	0.
94-PU-236	8.99688+7	5.8634	B	2.517-08	1.73 -09	2.69 -08
92- U-237	5.83200+5	0.3103	B	0.	0.	0.
93-NP-237	6.7532+13	4.9470	B	1.303-08	4.39 -12	1.303-08
94-PU-237	3.94243+6	0.0628	B	6.72 -13	0.	6.72 -13
92- U-238	1.4100+17	4.2755	B	6.64 -09	1.095-06	1.102-06
93-NP-238	1.82908+5	0.7916	B	0.	0.	0.
94-PU-238	2.76912+9	5.5871	B	2.124-08	4.08 -09	2.532-08
92- U-239	1.41000+3	0.4650	B	0.	0.	0.
93-NP-239	2.03385+5	0.4180	B	0.	0.	0.
94-PU-239	7.6084+11	5.2396	B	1.664-08	9.37 -12	1.665-08
92- U-240	5.07600+4	0.1755	A	0.	0.	0.
93-NP-240	4.02000+3	1.5755	A	0.	0.	0.
93-NP-240M	4.50000+2	1.0407	A	0.	0.	0.
94-PU-240	2.0670+11	5.3274	B	1.676-08	1.08 -07	1.25 -07
95-AM-240	1.82880+5	1.0920	B	3.74 -14	0.	3.74 -14
96-CM-240	2.31552+6	6.3844	A	3.37 -08	9.23 -08	1.26 -07
94-PU-241	4.63886+8	0.0054	B	3.39 -13	0.	3.39 -13
95-AM-241	1.3639+10	5.6131	B	2.115-08	9.31 -12	2.116-08
96-CM-241	2.83392+6	1.1100	B	2.79 -10	0.	2.79 -10
94-PU-242	1.1875+13	4.9812	B	1.406-08	1.18 -05	1.18 -05
95-AM-242	5.76360+4	0.1944	B	0.	0.	0.
95-AM-242M	4.79665+9	0.0631	B	9.22 -11	3.74 -10	4.56 -10

TABLE XIII (cont.)

NUCLIDE	HALF-LIFE (SECONDS)	DECAY ENERGY (MEV)	DE-CAY REF	◆◆◆◆NEUTRONS PER DECAY◆◆◆◆ ALPHA,N IN UO2	SPONT. FISSION	TOTAL
96-CM-242	1.40745+7	6.2169	B	3.07 -08	1.714-07	2.02 -07
94-PU-243	1.78452+4	0.1957	B	0.	0.	0.
95-AM-243	2.3289+11	5.4224	B	1.82 -08	5.32 -10	1.87 -08
96-CM-243	8.99372+8	6.1598	B	2.62 -08	0.	2.62 -08
94-PU-244	2.5877+15	4.6510	B	1.083-08	2.875-03	2.88 -03
95-AM-244	3.63600+4	1.1177	B	0.	0.	0.
95-AM-244M	1.56000+3	0.5088	B	0.	0.	0.
96-CM-244	5.71495+8	5.9010	B	2.582-08	3.623-06	3.65 -06
94-PU-245	3.78280+4	0.8103	A	0.	0.	0.
95-AM-245	7.38000+3	0.3199	A	0.	0.	0.
96-CM-245	2.6744+11	5.5881	B	1.948-08	0.	1.95 -08
94-PU-246	9.37440+5	0.2514	A	0.	0.	0.
95-AM-246M	1.50000+3	1.4433	A	0.	0.	0.
96-CM-246	1.4926+11	5.4714	B	1.971-08	8.313-04	8.31 -04
96-CM-247	4.9229+14	5.3522	B	1.466-08	0.	1.47 -08
96-CM-248	1.0720+13	4.7270	B	1.441-08	2.569-01	2.57 -01
97-BK-248	2.84018+8	------	A	--------	--------	--------
97-BK-248M	8.46000+4	0.1684	A	0.	0.	0.
98-CF-248	2.88144+7	6.3613	A	3.336-08	9.519-05	9.52 -05
96-CM-249	3.84900+3	0.2932	B	0.	0.	0.
97-BK-249	2.76480+7	0.0331	B	2.906-13	1.656-09	1.66 -09
98-CF-249	1.1064+10	6.2903	B	2.646-08	1.712-08	4.36 -08
96-CM-250	3.5660+11	------	C	--------	2.32 +00	2.32 +00
97-BK-250	1.15812+4	1.1829	B	0.	0.	0.
98-CF-250	4.12764+8	6.1227	B	2.941-08	2.718-03	2.72 -03
96-CM-251	1.00800+3	0.5925	A	0.	0.	0.
97-BK-251	3.33600+3	0.4988	A	0.	0.	0.
98-CF-251	2.8338+10	6.0260	B	2.532-08	0.	2.53 -08
98-CF-252	8.32471+7	6.0317	B	2.996-08	1.164-01	1.164-01
98-CF-253	1.53878+6	0.0980	B	8.89 -11	0.	8.89 -11
99-ES-253	1.76860+6	6.7367	B	3.995-08	3.419-07	3.82 -07
98-CF-254	5.22720+6	0.0184	A	8.167-11	3.88 +00	3.88 +00
99-ES-254	2.38205+7	6.6172	A	3.627-08	1.193-07	1.56 -07
99-ES-254M	1.41480+5	0.7351	A	1.138-10	1.778-07	1.78 -07
100-FM-254	1.16640+4	7.2996	A	5.08 -08	2.34 -03	2.34 -03
98-CF-255	6.84000+3	------	A	0.	0.	0.
99-ES-255	3.30912+6	0.5956	A	2.72 -09	1.59 -04	1.59 -04
100-FM-255	7.22520+4	7.2407	A	4.75 -08	8.54 -07	9.02 -07
99-ES-256	1.32000+3	0.6169	A	0.	0.	0.
100-FM-256	9.45720+3	7.0250	A	4.55 -08	3.69 +00	3.69 +00
100-FM-257	8.68320+6	6.8640	A	3.81 -08	8.09 -03	8.09 -03
100-FM-258	3.80000-4	------	A	0.	4.03 +00	4.03 +00

DECAY DATA REFERENCES
- A=TABLE OF ISOTOPES
- B=ENDF/B-V
- C=A.TOBIAS,U.K.,PRIVATE COMMUNICATION

ADDITIONAL NOTES

MISSING DATA NOTED AS ------

81-TL-210, NEUTRONS FROM DELAYED NEUTRON EMISSION FROM 82-PB-210 LEVELS PRODUCED IN BETA DECAY.

92- U-235, SPONTANEOUS FISSION BRANCHING IN ENDF/B-V IS ZERO BY OMISSION. S.F. BRANCHING(2.011-9) TAKEN FROM REFERENCE A.

97-BK-248 DECAY CHARACTERISTICS UNKNOWN.

These values of R_k may be used with detailed calculated activity inventory to determine total neutron production within oxide fuel, using Eq. (22).

ACKNOWLEDGMENTS

The authors gratefully acknowledge the assistance of P. A. Ombrellaro of Hanford Engineering Development Laboratory in providing information and helpful discussions about the methodology and data used; he, with D. L. Johnson and R. E. Schenter, provided low-energy $^{17,18}O(\alpha,n)$ cross-section values used in earlier calculations. We also appreciate conversations with J. K. Bair and J. Gomez del Campo of Oak Ridge National Laboratory and L. F. Hansen of Lawrence Livermore National Laboratory concerning their $O(\alpha,n)$ cross-section measurements and related observations.

At Los Alamos, M. E. Battat, R. J. LaBauve, and T. R. England worked on the accumulation of alpha-decay and spontaneous-fission decay data for the actinide nuclides. R. W. Hardie posed the problem and furnished fast-reactor fuel information. D. M. McClellan painstakingly took cross-section values from graphical form, and N. L. Whittemore prepared tables and figures. We sincerely appreciate their contributions.

REFERENCES

1. P. A. Ombrellaro and D. L. Johnson, "Subcritical Reactivity Monitoring: Neutron Yields from Spontaneous (α,n) Reactions in FFTF Fuel," Hanford Engineering Development Laboratory report HEDL TME 78-39 (June 1978). Information in this document was supplemented and updated by personal communications with the authors.

2. W. B. Wilson, T. R. England, R. J. LaBauve, M. E. Battat, D. E. Wessol, and R. T. Perry, "Status of CINDER and ENDF/B-V Based Libraries for Transmutation Calculations," Proc. Int. Conf. on Nuclear Waste Transmutation, Austin, Texas, July 22-24, 1980.

3. R. T. Perry and W. B. Wilson, "The (α,n) Neutron Production by Alpha Particles in PuO_2, UO_2, and ThO_2 Fuels," in "Applied Nuclear Data Research and Development, April 1-June 30, 1980," Los Alamos Scientific Laboratory report LA-8524-PR (1980), p. 20.

4. R. T. Perry and W. B. Wilson, "Neutron Production from (α,n) Reactions in PuO_2, UO_2, and ThO_2 Fuels," Trans. Am. Nucl. Soc. 35, 549 (1980).

5. W. B. Wilson, R. T. Perry, T. R. England, R. J. LaBauve, M. E. Battat, and N. L. Whittemore, "Neutron Production from Actinide Decay in Oxide Fuels," in "Applied Nuclear Data Research and Development, July 1-September 30, 1980," Los Alamos Scientific Laboratory report LA-8630-PR (1980), p. 23.

6. H. J. Hirsch and Hj. Matzke, "Stopping Power and Range of α-Particles in (U,Pu)C and UC and Application to Self-Diffusion Measurements Using Alpha Spectroscopy," J. of Nucl. Mater. 45, 29 (1972).

7. V. Nitzki and Hj. Matzke, "Stopping Power of 1-9 MeV He^{++} Ions in UO_2, $(U,Pu)O_2$, and ThO_2," Phys. Rev. B8, 1894 (1973).

8. L. C. Northcliffe and R. F. Schilling, "Range and Stopping Power Tables for Heavy Ions," Nucl. Data Tables A7, 233 (1970).

9. J. F. Ziegler and W. K. Chu, "Stopping Cross Sections and Backscattering Factors for ^4He Ions in Matter," Atomic Data and Nucl. Data Tables 13, 463 (1974).

10. W. H. Bragg and R. Kleeman, "On the Alpha Particles of Radium and Their Loss of Range in Passing Through Various Atoms and Molecules," Phil. Mag. 10, 318 (1905).

11. H. H. Anderson, Bibliography and Index of Experimental Range and Stopping Power Data, Vol. 2 of The Stopping and Ranges of Ions in Matter Series (Pergammon Press, New York, 1977).

12. H. Bichsel, "A Critical Review of Experimental Stopping Power and Range Data," in "Studies in Penetration of Charged Particles in Matter," National Academy of Sciences - National Research Council publication 1133 (1964).

13. J. F. Ziegler, Helium Stopping Powers and Ranges in All Elemental Matter, Vol. 4 of The Stopping and Ranges of Ions in Matter Series (Pergammon Press, New York, 1977).

14. J. K. Bair and H. B. Willard, "Level Structure in Ne^{22} and Si^{30} from the Reactions $O^{18}(\alpha,n)Ne^{21}$ and $Mg^{26}(\alpha,n)Si^{29}$," Phys. Rev. 128, 299 (1962).

15. J. K. Bair and F. X. Haas, "Total Neutron Yield from the Reactions $^{13}C(\alpha,n)^{16}O$ and $^{17,18}O(\alpha,n)^{20,21}Ne$," Phys. Rev. C7, 1356 (1973).

16. J. K. Bair and J. Gomez del Campo, "Neutron Yields from Alpha-Particle Bombardment," Nucl. Sci. Eng. 71, 18 (1979).

17. L. F. Hansen, J. D. Anderson, J. W. McClure, B. A. Pohl, M. L. Stelts, J. J. Wesolowski, and C. Wong, "The (α,n) Cross Sections on ^{17}O and ^{18}O Between 5 and 12.5 MeV," Nucl. Phys. A98, 25 (1967).

18. J. K. Bair (retired), Oak Ridge National Laboratory, personal communication (June 1980).

19. J. Gomez del Campo, Oak Ridge National Laboratory, personal communication (June 1980).

20. L. F. Hansen, Lawrence Livermore National Laboratory, personal communication (June 1980).

21. D. L. Johnson, P. A. Ombrellaro, and R. E. Schenter, Hanford Engineering Development Laboratory, personal communication (June 1980).

22. W. B. Wilson, T. R. England, R. J. LaBauve, M. Battat, and N. L. Whittemore, "CINDER Actinide Decay Library Development," in "Applied Nuclear Data Research and Development, January 1-March 31, 1980," Los Alamos Scientific Laboratory report LA-8418-PR (1980), p. 28.

23. C. M. Lederer and V. S. Shirley, Eds., Table of Isotopes, Seventh Edition (John Wiley and Sons, Inc., New York, 1978).

24. F. W. Walker, G. J. Kirouac, and F. M. Rourke, Chart of the Nuclides, Twelfth Edition (General Electric Company, Knolls Atomic Power Laboratory, Schenectady, New York, 1977).

25. M. E. Battat, W. B. Wilson, R. J. LaBauve, and T. R. England, "Actinide Decay Data," in "Applied Nuclear Data Research and Development, April 1-June 30, 1980," Los Alamos Scientific Laboratory report LA-8524-MS (1980), p. 18.

26. F. Manero and V. A. Konshin, "Status of the Energy-Dependent $\overline{\nu}$-Values for the Heavy Isotopes (Z>90) from Thermal to 15 MeV and of $\overline{\nu}$-Values for Spontaneous Fission," At. Energy Rev. 10, 637 (1972).