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Nuclear Data Development and Shield Design for Neutrons Below 60 MeV

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of the University of California

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ABSTRACT

Nuclear Data Development and Shield Design

for Neutrons Below 60 MeV. (May 1977)

William Bradley Wilson, B.S., M.S., Texas A&M University

Co-Chairmen of Advisory Committee: Dr. J. B. Smathers
Dr. D. R. Harris

A nuclear data library has been created for medium energy neutron transport calculations. The 60-group library includes P_5 cross sections in standard LASL format for H, B, C, N, O, Si, Fe and W. The 60-group structure was chosen from a sensitivity analysis of a thick iron shield calculated with a 50 MeV deuteron-on-beryllium neutron source spectrum and a 121-group cross-section set.

The library combines processed ENDF/B-IV cross-section data below 20 MeV and higher energy cross-section parameters calculated with the intranuclear-cascade and evaporation model. A 6-group version of the library is used in the design of a shield-collimator unit for fast neutron radiotherapy. While the shield is specific for the 50 MeV d^+ -Be neutron source presently used in cancer therapy at Texas A&M University, the cross sections and methods developed are applicable to the problems of medium energy neutron shielding in general.

1 INTRODUCTION

This dissertation describes the development of a library of nuclear data describing the interaction of neutrons below 60 MeV with a variety of materials. The data library is used in the design of primary shielding for neutrons produced anisotropically by 50 MeV deuterons on beryllium. The specific application of the resulting shield design is for the Texas A&M University - M.D. Anderson Tumor Institute and Hospital fast neutron cancer therapy program at the Cyclotron Institute of Texas A&M University, College Station, Texas.¹⁾

The use of fast neutron radiation in the treatment of cancer may be traced to the early clinical trials of Stone and Larkin in the late '30s and early '40s.²⁻⁴⁾ Recent interest in fast neutron radiotherapy has been fostered by advances in understanding of the biological effects of fast neutron radiation.^{5,6)}

There exist two sources of fast neutron radiation commonly considered to have sufficient yield and energy for radiotherapy applications; these are the medium energy accelerator and the neutron generator. The neutron generator, often called the D-T machine because of its use of the ${}^3\text{H}(d,n){}^4\text{He}$ reaction, is composed of a low-energy ($\sim 200\text{keV}$) deuteron accelerator and tritium target. This machine may be used to produce nearly monoenergetic 14.-15. MeV neutrons almost isotropically, with a yield approaching 10^{13} neutrons/second for the lifetime of the target.⁷⁾ Tritium target development for higher yield and lifetime is currently receiving considerable attention.⁸⁾

A unique variation of the D-T machine, currently under development at the Los Alamos Scientific Laboratory, will use a tritium accelerator and gaseous deuterium jet target.⁹⁾ This machine, expected

This dissertation follows the format and style of Nuclear Instruments And Methods.

to yield 10^{15} neutrons/second, is to be used primarily for fusion reactor materials damage studies.

Medium energy accelerator-produced fast neutron radiation is generally described by a high intensity, anisotropic yield and continuous energy spectrum characteristic of the particular reaction and incident particle energy employed.¹⁰⁻¹⁷⁾

Preliminary to recent clinical work using both sources of fast neutron radiation, the advantages and disadvantages of each were examined in detail.¹⁸⁻²³⁾ Radiation therapy is currently in progress using both D-T machines²⁴⁾ and medium energy accelerators.²⁵⁻²⁹⁾

A research program conducted by J.T. Brennan³⁰⁾ of the University of Pennsylvania at the Texas A&M University Variable Energy Cyclotron³¹⁾ (TAMVEC) drew the interest of the M.D. Anderson Tumor Institute and Hospital of Houston in fast neutron radiotherapy of humans. Building upon the favorable results of the Hammersmith Hospital of London, using the Medical Research Council (MRC) Cyclotron in fast neutron cancer therapy, the preliminary work at the TAMVEC¹⁾ and the experience of the physics and radiotherapy groups of M.D. Anderson in conventional radiotherapy, a program leading to fast neutron cancer therapy at the TAMVEC was proposed and subsequently funded by the National Cancer Institute³²⁾ and the State of Texas.

The first year of this program involved radiation biology studies with cell cultures and small animals, with the support of the College of Veterinary Medicine of Texas A&M. During this period experiment cave 2 of the TAMVEC building (fig.1) was modified to serve as a patient treatment room. The shield-collimator unit, located in the treatment room (fig.2) was designed with experimental shielding data^{33,34)} to accommodate an isotropic source using a spectrum actually associated only with neutrons produced in the forward direction by 50 MeV deuterons on beryllium.

A future phase of the program has been considered, involving the

Figure 1. TAMVEC Building Basement Plan

Figure 2. TAMVEC Neutron Therapy Treatment Room.

addition of a therapy wing to the TAMVEC building (fig.3). The treatment room here would feature a patient positioning device to locate the prone patient in a radiation field with axis variable from vertical to horizontal. The beam transport system thus must include a bending magnet and pivoting target and shield-collimator unit. The design of the shield-collimator unit for such an application is the objective of this research.

The design criterion³⁵⁾ of the shield-collimator unit is a dose equivalent rate at the surface of the shield 0.5% of that in the collimated beam at 0° , 125 cm. from the Be target. It is now advantageous to utilize the strong anisotropy of the neutron source to achieve a shield-collimator of minimum mass and dimension required for its mobility.

In previous experimental work at the TAMVEC it was shown that tungsten and iron are preferred primary shield materials. Although tungsten was shown to be a superior attenuator of neutrons of the spectrum produced, economics precludes its use as a bulk shielding material. Tungsten blocks are currently used, however, to shape radiation fields for optimum tumor dose delivery. Although Benelex-401³⁶⁾ was shown to be a slightly superior attenuation material for collimator inserts, water extended polyester (WEP) resin³⁷⁾, combined with water, B_4C and iron filings was found to be a more desirable collimator material because of its molding and attenuating properties. Again, economics forbids the preferable combination of Tungsten with WEP.

The design of such a shield may be calculated using a number of neutron transport techniques that may be classed as discrete ordinates (Sn) and Monte Carlo methods.³⁸⁾ Monte Carlo methods are numerical procedures, based on statistical theory, producing a set of neutron histories by following individual neutrons through their various interactions and spatial migration. This method has proven useful in a number of areas, particularly in irregular geometries, and its accuracy is largely a function of the number of histories generated. It

is this requirement of a large number of histories for statistical significance in a deep penetration shielding problem and the resulting large computation time and cost that have led many experienced in shielding calculations to suggest the use of multigroup discrete ordinates calculations.³⁹⁻⁴⁴⁾ This method, selected and used for transport calculations in this design, is examined in the following section.

The solution of the neutron transport equation requires nuclear data describing the energy and angular characteristics of neutron-nucleus interactions, in the form of a cross section library. The creation of a nuclear data library may be a small task for many transport problems in fission and fusion reactor physics. A number of useful and maintained computer codes⁴⁵⁻⁴⁷⁾ are available to process into the required form point energy cross-section files^{48,49)} describing neutron-nucleus reactions below 20 MeV. In fact, a number of multigroup cross-section libraries⁵⁰⁻⁵⁴⁾ are maintained and distributed⁵⁵⁾ for particular applications, each library unique and limited by the choice of weighting flux⁵⁶⁻⁶¹⁾, number⁵⁹⁾ and structure⁶²⁻⁶⁴⁾ of energy multigroups, number of cross-section tables and materials described.

The task of generating a cross section library for medium energy neutron transport calculations is complicated by the absence of tabulated nuclear data files above 20 MeV. Examination of published information on medium energy neutron reactions above 20 MeV has produced a sparse collection of data, described in Appendix A. Medium energy neutron scattering cross-section data for this work have largely been computed, using nuclear models described in Section 2, and adjusted to agree with experimental measurements where available.

The scheme of combining processed ENDF/B^{48,49)} cross sections with those calculated using nuclear models is not new. At least two cross-section libraries have been so generated^{65,66)} and employed in neutron transport calculations.⁶⁷⁻⁷¹⁾ However, these earlier libraries were generated from ENDF/B-III data, with a maximum energy of

15 MeV, using weighting fluxes and group structures inappropriate to the problem at hand. Perhaps most significant, no attempt was made to adjust calculated medium energy cross-section data to available experimental values.

The process of evaluating effects of nuclear data and computational parameters on deep penetration transport calculations is examined in section 3. The generation of the nuclear data library, by far the largest effort of this work, is described in sections 3 and 4. The library was collapsed to a few-group set for the two-dimensional neutron transport calculations used in the shield-collimator design, as described in section 5.

2 TRANSPORT THEORY, SENSUTIVITY ANALYSIS AND CROSS-SECTION MODELS

Transport Theory

The steady-state distribution of neutrons in a medium may be expressed in terms of the angular flux $\phi(\vec{r}, E, \vec{\Omega})$,

$$\phi(\vec{r}, E, \vec{\Omega}) = vN(\vec{r}, E, \vec{\Omega}) \quad (1)$$

where $N(\vec{r}, E, \vec{\Omega})$ is the density of neutrons of energy E at $\vec{r}$ traveling in direction $\vec{\Omega}$, and v is the scalar velocity of neutrons of energy E . The angular flux is described by the time-independent Boltzmann equation⁷²):

$$\vec{\Omega} \cdot \nabla \phi + \Sigma_T \phi = S + \int_{E'} \int_{\vec{\Omega}'} \Sigma(\vec{r}, \vec{\Omega}' \rightarrow \vec{\Omega}, E' \rightarrow E) \phi d\vec{\Omega}' dE' \quad (2)$$

where Σ_T is the macroscopic total cross section, S is the local source $S(\vec{r}, \vec{\Omega}, E)$, and $\Sigma(\vec{r}, \vec{\Omega}' \rightarrow \vec{\Omega}, E' \rightarrow E)$ is the macroscopic transfer cross section.

In the course of this work transport calculations are performed in cylindrical and spherical geometries. The coordinate systems for these geometries are shown in fig.4. The multigroup discrete ordinates method is used throughout. Here the continuum of values of $\vec{\Omega}$, $\vec{r}$ and E are reduced to a small number of values or intervals of these values. Neutron reaction properties are averaged over each of NG energy groups, and energy dependences of transfer cross sections are expressed as probabilities for transfer from one energy group to another. Neutron angular distributions are evaluated at a number of discrete directions.^{73,74} The domain of $\vec{r}$ is defined by a number of contiguous spatial mesh cells. Spatial variations in neutron reaction properties due to materials or flux perturbations may be accounted for with the definition of a number of regions, each having different properties held constant throughout the region.

DTF Spherical (r, θ) coordinates TROTTRAN Cylindrical (r, z) coordinates

Figure 4. Coordinate Systems

The angular dependence of neutron transfer cross sections, describing the transfer from energy group g' and solid angle about $\vec{\Omega}'$ to energy group g and solid angle about $\vec{\Omega}$, are expressed in terms of μ , the cosine of the scattering angle θ between $\vec{\Omega}'$ and $\vec{\Omega}$. The orthogonal Legendre Polynomials are then employed to express transfer cross section in terms of μ as follows:

$$\Sigma(g' \rightarrow g, \vec{\Omega}' \rightarrow \vec{\Omega}) = \frac{1}{2\pi} \Sigma_{g',g}(\mu) = \frac{1}{2\pi} \sum_{\ell=0}^{\infty} \frac{2\ell+1}{2} \Sigma_{g',g}^{\ell} P_{\ell}(\mu) \quad (3)$$

where $P_{\ell}(\mu)$ is the Legendre Polynomial of degree ℓ . Multiplying (3) by $P_n(\mu)$ and imposing the orthogonality properties of Legendre Polynomials, the transfer cross section coefficients are easily defined as follows⁷⁵⁾:

$$\Sigma_{g',g}^{\ell} = -1 \int_{-1}^1 \Sigma_{g',g}(\mu) P_{\ell}(\mu) d(\mu) . \quad (4)$$

The transport codes DTF-IV⁷⁶⁾ and TWO'TRAN⁷⁷⁾ employed in this work accept transfer cross section coefficients described by (4).

The expansion of (3) must be truncated after some NL terms because of computer storage and cost limitations. The error incurred by truncation generally decreases as the number of terms retained increases^{78,79)} and may be strongly problem dependent. The error associated with simple truncation may be decreased with the application of transport corrections to terms retained. Transport correction methods (transport approximations) have been summarized by Harris⁸⁰⁾ and incorporated in the CINC4⁸¹⁾ program by him and this author. Throughout this work, transport calculations included no less than 4 terms (P3). Transport approximations were not used to adjust cross sections because of convergence observations examined in the following section.

The discrete ordinates for which the Boltzmann equation are solved are specified by a set of angular quadrature coefficients, consisting of a number of weighted directions. Calculation accuracy

is generally improved as the number of directions considered is increased; this dependence on quadrature increases with the scattering anisotropy characteristic of the problem. The requirements imposed upon angular quadrature sets are well documented.^{73,76,77}) Gauss-Legendre quadrature sets are used throughout this work.

The effects of the quantized representation of angular quantities discussed above are common to discrete ordinate transport calculations. One may include more terms of the Legendre Polynomial cross-section expansion and extend the discrete ordinate representation until calculated results are insensitive to further refinements. However, such an approach is not applicable to the definition of the number and value of multigroup boundaries used in the quantization of neutron energy.

The multigroup structure employed in transport calculations is generally limited to that of the nuclear data libraries available and applicable to the calculation. In the absence of available and reliable data, one has the unique opportunity of establishing a nuclear data library. The choice of the number NG of multigroup values is first limited by computer storage limitations, which is determined by a combination of the number of multigroups NG, the number of Legendre orders NL, number of discrete ordinates N, number of spatial cells NR and the compiled program storage. The choice of NG is further restricted by transport code execution time, which varies roughly as NG^2 with other variables held constant.

With the choice of the number of multigroups made, the values of the NG+1 energy boundaries of the group structure must be determined. The selection of energy boundaries has traditionally reflected cross section energy structure⁸²⁻⁸⁴), with narrow groups used to detail major cross-section variations with energy and wide groups used in the absences of such variations. In the absence of significant cross section structure, groups of equal lethargy width are commonly recommended.⁸⁵) Equal lethargy groups lead to approximately equal group integrated fluxes in many thermal reactor analysis and shielding calculations.

Sensitivity Analysis

An alternate approach for the systematic selection of multigroup boundaries has been suggested by Oblo⁶⁴⁾ and others, using sensitivity theory.⁸⁶⁾ This technique requires that the group structure of the multigroup cross section library be chosen such that some flux-integrated response be equally sensitive to cross-section data of each group. This response R is defined by⁸⁶⁾

$$R = \int_{\xi'} \Sigma_R(\xi) \phi(\xi) d\xi' = (\Sigma_R, \phi) \quad (5)$$

where ξ represents a point in phase space with variables $\vec{r}, \vec{\Omega}$ and E. $\Sigma_R(\xi)$ is the response function, and $\phi(\xi)$ is the neutron angular flux described by the Boltzmann Equation (2), which may be written in operator notation as:

$$L \phi = S \quad (6)$$

where L is the Boltzmann operator. The neutron adjoint flux $\phi^*(\vec{r}, E, \vec{\Omega})$ is described by the adjoint Boltzmann Equation:

$$-\vec{\Omega} \cdot \nabla \phi^* + \Sigma_T \phi^* = S^* + \int_{E'} \int_{\vec{\Omega}'} \Sigma(\vec{r}, \vec{\Omega} \rightarrow \vec{\Omega}', E \rightarrow E') \phi^* d\vec{\Omega}' dE' \quad (7)$$

where S^* is the adjoint source, the response function Σ_R of our calculation. In operator notation the adjoint Boltzmann equation may be written as:

$$L^* \phi^* = S^* \quad (8)$$

The adjoint operator L^* and boundary conditions on ϕ^* are defined⁸⁷⁾ by the requirement:

$$(\phi^*, L\phi) = (\phi, L^* \phi^*) \quad (9)$$

where the notation (ψ, χ) symbolizes the inner product of the two functions $\psi(\xi)$ and $\chi(\xi)$, which here is the integral

$$(\psi, \chi) = \int_{\xi'} \psi(\xi) \chi(\xi) d\xi' . \quad (10)$$

Using the inner product relationship of (9) we may combine (6) and (8) as follows:

$$(S^*, \phi) = (L^* \phi^*, \phi) = (\phi^*, L\phi) = (\phi^*, S) . \quad (11)$$

If we now perturb the problem by altering the nuclear data (and hence the operators L and L^*) while the source S and response function S^* are held constant, we may use the linear perturbation technique to write:

$$(L + \delta L) (\phi + \delta\phi) = S \quad (12)$$

and

$$(L^* + \delta L^*) (\phi^* + \delta\phi^*) = S^* . \quad (13)$$

Expanding the perturbed Boltzmann Equation (12)

$$\left[L + \delta L \right] \left[\phi + \delta\phi \right] = L\phi + \left[L\delta\phi + \delta L\{\phi + \delta\phi\} \right] = S . \quad (14)$$

Subtracting the unperturbed relationship of (6)

$$L\delta\phi + \delta L \{ \phi + \delta\phi \} = 0 . \quad (15)$$

Similarly, the perturbed adjoint Boltzmann equation yields

$$L^* \delta\phi^* + \delta L^* \{ \phi^* + \delta\phi^* \} = 0 . \quad (16)$$

We may now express the perturbation to the response R due to the perturbations imposed on L and L^* . Using

$$\delta R = \delta(S^*, \phi) = \delta(\phi^*, S) = (\phi^* + \delta\phi^*, S) - (\phi^*, S) = (\delta\phi^*, S) \quad (17)$$

or

$$\delta R = (\delta\phi^*, L\phi) = (L^* \delta\phi^*, \phi) . \quad (18)$$

From (16)

$$L^* \delta \phi^* = -\delta L^* \{ \phi^* + \delta \phi^* \} \simeq -\delta L^* \phi^* . \quad (19)$$

Then

$$\delta R \simeq (-\delta L^* \phi^*, \phi) . \quad (20)$$

We now assume that the perturbation to the operators is in the form of a small change to the group g cross sections. In particular, we assume that

$$\frac{\delta \Sigma_{Tg}}{\Sigma_{Tg}} = \frac{\delta \Sigma_{g,g'}^{\ell}}{\Sigma_{g,g'}^{\ell}} = C \quad (21)$$

for all sink groups g' and Legendre orders ℓ , where C is a small fractional constant.

The sensitivity of the result R to this perturbation of the group g cross sections is expressed by the "sensitivity"⁸⁶⁾

$$P_g = \frac{\delta R/R}{C} \quad (22)$$

which expresses the fractional change of the result due to a fractional change of the group g cross section. If the adjoint and forward neutron angular fluxes are expanded in Legendre polynomials as follows:

$$\phi_g(\vec{r}, \vec{\Omega}) = \sum_{\ell=0}^{NL} \frac{2\ell+1}{4\pi} \phi_g^{\ell}(\vec{r}) P_{\ell}(\mu) \quad (23)$$

and

$$\phi_g^*(\vec{r}, \vec{\Omega}) = \sum_{\ell=0}^{NL} \frac{2\ell+1}{4\pi} \phi_g^{*\ell}(\vec{r}) P_{\ell}(\mu) \quad (24)$$

we may use the expression for δR given by (20), referring to the adjoint operator of (7), to express the sensitivity as:

$$P_g = \frac{1}{CR} \int d\vec{r} \sum_{\ell=0}^{NL} \frac{2\ell+1}{4\pi} \phi_g^{\ell}(\vec{r}) \left[-C \Sigma_{Tg} \phi_g^{*\ell}(\vec{r}) + C \Sigma_{g,g'}^{\ell} \phi_{g'}^{*\ell}(\vec{r}) \right] \quad (25)$$

or

$$p_g = \frac{1}{R} \int d\vec{r} \sum_{\ell=0}^{NL} \frac{2\ell+1}{4\pi} \phi_g^\ell(\vec{r}) \left[-\Sigma_T \phi_g^{*\ell}(\vec{r}) + \sum_{g'} \Sigma_{g,g'}^\ell \phi_g^{*\ell}(\vec{r}) \right]. \quad (26)$$

The forward and adjoint neutron flux expansion coefficients may be produced from directed fluxes calculated in the corresponding transport calculations using the multigroup cross section data.

The graphic representation of the set of NG sensitivity values is the "sensitivity profile". The ALVIN code⁸⁸⁾ calculates the sensitivity values for any response function and geometry lending itself to one-dimensional modeling, using the directed fluxes calculated with the DTF-IV one-dimensional transport code. The sensitivity profile may be calculated with a fine group structure, containing many groups, and used to select the boundaries of a second group structure tailored to the calculation. This technique is employed in the selection of the group structure, as described in the following section.

Cross Section Models

Nuclear models are commonly used in the evaluation of nuclear data in order to⁸⁹⁾:

- 1) make sophisticated fits of consistent data
- 2) make a definite choice among inconsistent data
- 3) fill gaps among loose data
- 4) predict cross sections for which no data exists.

The Evaluated Nuclear Data File (ENDF) is composed of evaluated cross section values largely from model calculations in all of these areas. In the range of neutron energies above 20 MeV, experimental data is sparse. One must necessarily resort to model calculations in order to construct a medium energy nuclear data library.

In the course of this work, three types of nuclear models are employed. These are the optical model, phase shift model and intra-nuclear cascade plus evaporation model.

Optical Model

In his attempt at explaining the $1/v$ nature of slow neutron reaction cross sections, Bethe⁹⁰) introduced the application of wave mechanics to the description of neutron cross sections. Long range forces between a neutron and nucleus were assumed negligible, and a basic complex neutron energy potential was assumed inside the volume of the nucleus⁸⁹):

$$\begin{aligned} V(r) &= V + iW & , r \leq R_o \\ &= 0 & , r > R_o \end{aligned} \quad (27)$$

where $R_o = r_o A^{1/3}$ f.

Here only three parameters need be evaluated: V, W and r_o . The number of parameters used in the optical model potential has increased with time (and obtainable particle energies); one form recently suggested includes some 18 parameters, allowing one to reproduce the smallest details of a given set of experimental data.

In this work, however, we are interested in extending the optical model fits to experimental data performed by others. The potential to be used in following calculations is thus that potential used in such previous and applicable calculations: the combination of a Saxon real central potential, Gaussian imaginary surface potential and Thomas spin-orbit potential. The potential may be written as⁹¹)

$$V(r) = V_S \rho(r) + iV_G q(r) + V_T \left(\frac{\hbar}{\mu c} \right)^2 \frac{1}{r} \frac{d\rho(r)}{dr} \sigma \cdot \ell \quad (28)$$

where

V_S is the Saxon real central potential well depth

V_G is the Gaussian imaginary surface potential well depth

V_T is the Thomas spin-orbit potential well depth

$$\begin{aligned} \rho(r) &= 1 + \exp \left[(r - R_o)/a \right]^{-1} & , r \leq d \\ &= 0 & , r > d \end{aligned}$$

$$\begin{aligned} q(r) &= \exp \left[- (r - R_o)^2 / b^2 \right] & , r \leq d \\ &= 0 & , r > d \end{aligned}$$

a is the real central well diffuseness

b is the imaginary surface well width

d is the radial limit of the nuclear potential

beyond which only the Coulomb potential, where applicable,
is proportional to $1/r$.

σ and ℓ are the neutron spin and angular momentum, respectively.

The parameters which must be evaluated here are V_S , V_G , V_T , r_0 , a, b and d. In the following chapter the potential described above is used in optical model calculations to describe the angular distribution of medium energy elastic scattering by iron, with parameters obtained by extrapolating values used in other work. The ABACUS-2 code⁹²⁾ is used in these optical model calculations.

Phase Shift Analysis

In the above treatment a potential function is assumed from extrapolated and/or interpolated parameters and an approximate solution of the Schrödinger equation is calculated using numerical techniques. We may alternatively extrapolate from or interpolate among parameters of the functional representation of solutions to the Schrödinger equation. The wave function of elastically scattered neutrons may be expressed in terms of angular functions, such as Legendre polynomials, Legendre functions, spherical spin functions, etc.⁹³⁾ Phase shifts are included in these expressions to produce observed angular-distribution characteristics. Alternatively, the scattering matrix representation includes associated parameters, also known as phase shifts.⁹⁴⁻⁹⁷⁾ Seamon⁹⁸⁾ has tabulated phase parameters for p-p and n-p scattering, in increments of 10 MeV, over the range of 10 to 350 MeV. The n-p phase parameters have, in turn, been incorporated into the NPSCAT code by J.C. Hopkins.⁹⁹⁾ The NPSCAT code is used to describe neutron elastic scattering by hydrogen, as described in Section 4.

Intranuclear-Cascade Plus Evaporation

Medium energy neutron reaction cross sections have successfully been described by this model, which is divided into two stages.¹⁰⁰⁾ It is assumed that an incident medium-energy particle interacts with the individual nucleons of the target nucleus, since the time increment during a collision with any nucleon is much less than that between successive collisions among the target nucleons. The incident particle and energetic recoil nucleons are assumed to travel through a nucleon gas, interacting only with nucleons encountered in their paths under the restrictions of a nucleon potential and the Pauli principle. The incident particle may thus pass completely through a nucleus without interaction, or produce an avalanche of nucleon cascades in a period of 10^{-22} - 10^{-23} seconds.

Some of these cascade nucleons may escape the nucleus, while others are absorbed by the nucleus with their energies distributed among the remaining nuclei. The residual nucleus, at the end of this first cascade stage, is left strongly excited because of its retention of part of the incident particle's energy. The de-excitation of the residual nucleus is accomplished through the evaporation or "boiling off" of gamma rays, nucleons and/or clusters of nucleons as massive as ^{10}Be . In the evaporation stage, which is much slower (10^{-13} - 10^{-18} s.), the residual nucleus is considered as a system of Fermi particles undergoing strong reactions with each other, with nucleons or clusters of nucleons spontaneously becoming energetic enough to escape the nucleus.

Intranuclear cascade plus evaporation cross section calculations for this work were performed with the CROIX-II¹⁰¹⁾ Monte Carlo code, which combines modified versions of the MECC-3¹⁰²⁾ code for intranuclear cascade calculations and the EVAP-3¹⁰³⁻¹⁰⁵⁾ code for calculating the evaporation of particles from excited compound nuclei. The nucleus is modeled as three concentric spherical shells, with the nucleon density constant in each shell; this is a step function

approximation to the momentum distribution of a degenerate Fermi gas¹⁰⁶⁾, which in turn approximates the nuclear charge distribution of Hofstadter.¹⁰⁷⁾ A similar model¹⁰⁶⁾ uses seven nucleon regions to better represent this nucleon distribution, giving better detail to the very diffuse outer edge of the nucleus. However, quantities calculated with these models are remarkably similar.¹⁰⁸⁾

Cross-section calculations for incident particles of energy E consist of a large number N ($\geq 10^4$) of individual Monte Carlo calculations, the number determined by the statistical accuracy desired. Each calculation corresponds to a particle of energy E , incident on the geometrical cross section σ_{geom} of the target, with a randomly chosen impact parameter $b \leq R_0$. A history of the type, energy and angular orientation of cascade particles is retained. The type and energy of evaporation particles, which are assumed to be emitted isotropically, are added to this history record. Cross section values for these non-elastic reactions may be determined from the file of Monte Carlo histories thus constructed. If, for example, of the N incident particles N_T find the nucleus transparent, the total nonelastic cross section is given by

$$\sigma_{\text{ne}}(E) = \frac{N - N_T}{N} \sigma_{\text{geom}}. \quad (29)$$

The number of each type of secondary particle and their energy and angular distributions may be determined by an appropriate survey of the reaction histories.

There is strong evidence that the IC+E model is not valid below a "few dozen" MeV.¹⁰⁰⁾ This is particularly true if the model is used to calculate individual reaction cross sections, e.g., $(n, 2n)$, (n, p) , etc. However, the measured energy distribution of secondary neutrons produced by 15-18 MeV protons on a variety of targets have been computed with reasonable accuracy.¹⁰⁹⁾ Bertini¹¹⁰⁾ has shown that computed total nonelastic cross sections for 30-60 MeV protons on a number of targets agree well with data for light nuclei, but are low for heavy nuclei.

In this work the IC+E model is used to calculate the number of secondary neutrons and their energy and angular distributions. Cross-section values will, however, be adjusted as described in the following section.

3 EXAMINATION OF CROSS SECTION LIBRARY AND CALCULATIONAL PARAMETERS

One-Dimensional Model

In order to examine the effect of the choice of the parameters of the data library and transport calculation on the shield design, trial calculations were performed in one-dimensional geometry with the DTF-IV code.⁷⁶⁾ The shield depicted in fig.2 (p.4) was represented as a thick spherical shell (I.D.=8 cm, O.D.=140 cm.). The neutron source used was isotropic, with the spectrum described below, and was uniformly distributed within the central void. All one-dimensional calculations were based on a total source rate of one neutron/second. The effectiveness of the shield was expressed in terms of the tissue dose equivalent rate at the shield extremity.

Neutron Fluence-To-Dose Equivalent Response Function (NFDE)

In order to determine the neutron dose equivalent rate (NDER) on the shield surface, a neutron fluence-to-dose equivalent response function is needed. A survey of published data¹¹¹⁾ shows that there is general agreement among data below ~1 MeV. However, since the peak dose in tissue from neutrons above a few MeV does not occur at the surface of the tissue but at a depth that increases with energy, one must judiciously interpret the data. In order to establish a response function for use with a continuous neutron energy spectrum, the NDER associated with neutrons above 10 MeV should be an average over the first centimeter in tissue. The response function is shown in fig.5 with referenced values. The response function passes through values of Rossi, et al.¹¹²⁾ below 5 keV; above this energy it passes through values of Stevenson, et al.¹¹³⁾, which provides the spatial average noted above. The function is listed in table 1 as a set of points, between which log-log interpolation is assumed.

Figure 5. Neutron Fluence-to-Dose Equivalent Response Function

TABLE 1

NEUTRON FLUENCE-TO-DOSE EQUIVALENT RESPONSE FUNCTION

ENERGY (EV.)	VALUE (REM/N-CM**2)	ENERGY (EV.)	VALUE (REM/N-CM**2)
1.000E-05	1.021E-09	2.000E+04	2.500E-09
1.000E-01	1.021E-09	5.000E+04	5.000E-09
2.500E-01	1.090E-09	1.000E+05	8.000E-09
5.000E-01	1.190E-09	2.000E+05	1.300E-08
1.000E+00	1.240E-09	5.000E+05	2.500E-08
1.000E+01	1.240E-09	1.000E+06	3.000E-08
2.500E+01	1.230E-09	2.000E+06	3.500E-08
5.000E+01	1.215E-09	5.000E+06	4.000E-08
1.000E+02	1.197E-09	2.000E+08	4.000E-08
2.500E+02	1.130E-09	5.000E+08	5.000E-08
5.000E+02	1.075E-09	1.000E+09	6.000E-08
1.000E+03	1.021E-09	2.000E+09	9.000E-08
2.500E+03	1.050E-09	5.000E+09	1.400E-07
5.000E+03	1.200E-09	1.000E+10	2.000E-07
1.000E+04	1.700E-09		

LOG-LOG INTERPOLATION IS USED BETWEEN VALUES

The FLUSE code¹¹⁴⁾ is used to process flux-weighted multigroup NFDE values, using the response function of table 1 and a description of the flux profile. The DOSE1 code¹¹⁴⁾ multiplies multigroup scalar flux values from DTF-IV and multigroup NFDE values from FLUSE and, by accumulating these multigroup contributions, calculates the radial values of the NDER due to the central source of 1 n/sec. These values have true significance only on the shield surface, indicating the dose-equivalent rate in tissue at that point. However, the plot of these values as a function of radius is used to indicate the effect of parameters.

Neutron Source Spectrum

The spectrum used in the one-dimensional calculations is that produced at 0° by 50 MeV deuterons on a thick beryllium target. This neutron spectrum has been measured by a number of experimenters, using neutron time-of-flight techniques, as shown in fig.6. It should be noted that, because of the agreement of the Hertel and Smathers¹¹⁵⁾ data with that of Northcliffe,¹¹⁶⁾ the data of Meulders¹¹⁷⁾ appears to be inconsistent below 25 MeV. This is particularly obvious relative to the neutron spectrum measured by Schweimer¹¹⁸⁾ for 53.8 MeV deuterons on beryllium. The spectrum of Hertel and Smathers was thus assumed correct, and multigroup source values were constructed from this spectrum.

The time-of-flight data of Hertel and Smathers does not extend below 4 MeV and is considered unreliable below 6 MeV.¹¹⁹⁾ As a result, the neutron source at lower energies is assumed to be zero in the following calculations. The consequences of this assumption are examined in section 4.

Calculations With The LASL 41-Group Library

Prior to this work, the Los Alamos Scientific Laboratory Nuclear Data Group T-2 prepared a multigroup cross-section library for medium

Figure 6. Measured Neutron Spectrum at 0° Produced by 50.-53.8 MeV Deuterons on Beryllium

energy neutron transport calculations. This library combined ENDF/B-III cross sections, processed into the LASL-30 group structure, with medium energy cross sections processed by an early version of CROIX, producing a 41-group library extending to 800 MeV with the energy structure given in Table 2. The library included H, C, O, Si, Al, Fe, Ca, Mo, W and Pb. All cross sections were expressed as P_0 - P_3 Legendre coefficients except for iron, which extended to P_8 .

In the formation of this library, unadjusted nonelastic cross sections were used. The group-to-group cross-section values from source groups above 15 MeV were taken as the average of values from bordering point-energy calculations. This data library was used in the preliminary design of shielding for neutrons produced by 800-MeV protons.¹²⁰⁾

Of the 41 energy groups of the library, only 34 energy groups are included in the range of the neutron source spectrum of fig.6 (p.26). Multigroup source values were constructed as described above.

The 41-group library was used to determine the necessary detail in angular quadrature, Legendre expansion and spatial definition. It should be emphasized that the minimum detail needed to accurately calculate the dose on the surface of a spherical shield due to an isotropic central source is not necessarily sufficient for calculating a shield in cylindrical geometry with a strongly anisotropic source. There is little experience, even among contributors¹²¹⁾ to the TWOTRAN code, in two-dimensional S_n anisotropic source problems.

One might suppose that the spatial definition considered sufficient for the spherical shield would also be adequate for the anisotropic source problem, but that one or two increments in angular refinement (e.g., from $P_n S_{2n}$ to $P_{n+1} S_{2n+2}$) would be required. This is the conclusion of the author.

The spherical shield problem was calculated with discrete ordinate quadrature sets ranging from S_4 to S_{16} , cross-section expansions ranging from P_2 to P_8 , and radial mesh sizes in iron ranging from 1 cm. to 3 cm. One centimeter intervals are maintained in the source

TABLE 2

LASL 41-GROUP STRUCTURE BOUNDARIES

GROUP NO.	UPPER BOUND (EV.)	GROUP NO.	UPPER BOUND (EV.)
1	8.000E+08	22	1.738E+06
2	7.000E+08	23	1.353E+06
3	6.000E+08	24	8.230E+05
4	5.000E+08	25	5.000E+05
5	4.000E+08	26	3.030E+05
6	3.000E+08	27	1.840E+05
7	2.000E+08	28	6.760E+04
8	1.000E+08	29	2.480E+04
9	5.000E+07	30	9.120E+03
10	2.500E+07	31	3.350E+03
11	2.000E+07	32	1.235E+03
12	1.700E+07	33	4.540E+02
13	1.500E+07	34	1.670E+02
14	1.350E+07	35	6.140E+01
15	1.200E+07	36	2.260E+01
16	1.000E+07	37	8.320E+00
17	7.790E+06	38	3.060E+00
18	6.070E+06	39	1.130E+00
19	3.680E+06	40	4.140E-01
20	2.865E+06	41	1.520E-01
21	2.232E+06	LOWER BOUND	1.390E-04

and external void regions in all calculations. The results of these calculations, in terms of the surface dose equivalent rate due to the 1 n/sec. source, are given in table 3. The range of the calculated NDER values throughout the sphere are shown in the dose equivalent rate profile of fig.7.

Examination of table 3 reveals that one may accurately reproduce within ~8% the most exacting spherical calculation (i.e., $S_{16} P_8$ with 1 cm. intervals in iron) with an $S_4 P_2$ calculation, using a $2\frac{1}{2}$ cm. spatial mesh in the shielding material. The anisotropic source calculations are thus performed with S_3 quadrature, P_3 Legendre cross-section expansions and $2\frac{1}{2}$ cm. spatial mesh dimensions.

In an attempt to estimate the significance of errors associated with the above use of medium energy multigroup cross-section values with no adjustment to available experimental values, a library of cross sections incorporating adjustments to group values above 15 MeV was constructed, using the CINX4 code.⁸¹⁾ This adjustment consisted of 16%-24% estimated increases in source group quantities; the external dose equivalent rate calculated in an $S_{16} P_5$ calculation with this library was approximately 10% less than that calculated with the corresponding unadjusted library.

In the course of this work D.R. Harris and this author wrote the ALVIN code⁸⁸⁾ for sensitivity analysis and data adjustment. The initial sensitivity analyses performed were for the spherical calculation described above, using the adjusted iron cross-section library. Although these calculations with the 41-group library were not sufficient to define a group structure for future transport calculations, the general sensitivity of the external NDER to the data was produced, as seen in fig.8. In order to verify the sensitivity profile calculated, a number of perturbed cross-section libraries were constructed from the adjusted iron cross-section library by the PERTCX code.¹¹⁴⁾ Each perturbed library consisted of the adjusted iron cross-section library with the values of a single group altered by some fractional

Figure 7. Radial NDER Values.

TABLE 3
PARAMETER SENSITIVITY STUDY

Calculated Surface NDER Values, (Rem/second)				
Calculation	Spatial Increments in iron			
	1.0cm	2.0cm	2.5cm	3.0cm
S16 P8	.776982E-13	.759678E-13	.751584E-13	.735488E-13
S16 P5	.777380E-13	.760066E-13	.751967E-13	.735869E-13
S8 P5	.775545E-13	.756983E-13	.743497E-13	.728399E-13
S8 P4	.775492E-13	.756930E-13	.743447E-13	.728351E-13
S8 P3	.773390E-13	.754943E-13	.741048E-13	.726397E-13
S6 P3	.769621E-13	.748853E-13	.735226E-13	.719426E-13
S6 P2	.766326E-13	.744473E-13	.728409E-13	-----
S4 P2	.750758E-13	.729179E-13	.713208E-13	-----

Figure 8. 41-Group Sensitivity Profile

increment, ranging from 0.1% to 10%. The perturbed libraries were each used to calculate the surface NDER, and group sensitivities were calculated (i.e., the ratio of the fractional change in the calculated NDER from the unperturbed cross-section calculation result to the fractional cross section perturbation). These individual sensitivity values are superimposed upon the 41-group sensitivity profile of fig.8. Within the limitations of linear perturbation techniques, the agreement between the group sensitivities calculated in the two independent ways is quite good.

There was concern that the gamma rays emitted from neutron-induced reactions may contribute significantly to the surface dose equivalent rate. In order to evaluate this effect, the GAMISOQ code¹¹⁴⁾ was written to produce isotropic gamma-ray source rates throughout the spherical shield from the 41-group neutron scalar fluxes, using the gamma-ray production rates of LaBauve, et al.⁵³⁾ This data extends only to 15MeV.; the gamma-ray sources originating from neutron reactions above 15 MeV were thus not included in the determination of the total gamma-ray source rate.

The gamma ray source was represented in a 12-group structure⁵³⁾ extending to 30 MeV; the multigroup gamma ray fluxes at the shield surface were calculated with the DTF-IV code,⁷⁶⁾ using the gamma ray transport cross sections of LaBauve, et.al.⁵³⁾ Twelve-group gamma-ray fluence-to-dose equivalent response values were calculated by FLUSE,¹¹⁴⁾ using the data of Claiborne and Trubey.¹¹¹⁾ The gamma-ray dose equivalent rate at the shield surface calculated by the DOSE1 code is 0.262729×10^{-15} Rem/sec, or about 0.3% of the surface NDER values of table 3. The surface dose equivalent rate due to gamma rays was thus considered negligible.

The 121-Group Library

A P_5 121-group cross section library was processed for iron in order to evaluate the importance of elastic scattering above 20 MeV,

provide a weighting flux for future processing, and to establish an accurate and detailed sensitivity profile. The group structure, described in table 4, was chosen to represent the detail of the iron cross-section shape. Elastic scattering is not included in the intranuclear-cascade and evaporation model, and it is suggested³⁹⁾ that elastic scattering above 20 MeV is of no consequence to neutron transport calculations because of the associated small energy losses and small angular deflections. This suggestion, however, is offered without proof.

Importance Of Elastic Scattering

In order to determine the magnitude of the iron elastic scattering cross section above 20 MeV, a comprehensive literature search, which comprises part of appendix A, was conducted. The measured values of the total cross section in the energy range from 20 MeV to 1 BeV are shown in fig.9. This data is expanded in fig.10, revealing the giant resonances¹²²⁾ due to elastic scattering. If the estimated nonelastic cross section, shown in fig.11, is subtracted from the total cross section of fig.9, the remainder is the elastic scattering cross section shown in fig.12.

With a good estimate of the magnitude of the elastic scattering cross section, it is now necessary to obtain the angular distribution of the elastically scattered neutrons. The optical model was used in determining these angular distributions. A literature search of optical model fits to experimental data produced a sparse collection of information. The potential expression almost unanimously selected by others is that given by (28), (p.17). The parameters for the potential well depths V_S , V_G and V_T , the real central well diffuseness a , the imaginary surface well width b , and the nuclear radius constant r_0 were determined by extrapolating the values of Bjorklund and Fernbach⁹¹⁾ at 4.1 MeV, 7 MeV and 14 MeV and those of Stuart, Anderson and Wong¹²³⁾ at 24.8 MeV. These values

TABLE 4

121-GROUP STRUCTURE BOUNDARIES

GROUP NO.	UPPER BOUND (EV.)	GROUP NO.	UPPER BOUND (EV.)	GROUP NO.	UPPER BOUND (EV.)
1	6.00000E+07	42	1.49570E+06	83	8.17500E+04
2	5.80000E+07	43	1.35340E+06	84	8.10000E+04
3	5.60000E+07	44	1.28730E+06	85	7.63510E+04
4	5.40000E+07	45	1.22460E+06	86	7.42500E+04
5	5.20000E+07	46	1.19430E+06	87	7.30000E+04
6	5.00000E+07	47	1.16480E+06	88	7.20000E+04
7	4.80000E+07	48	1.10800E+06	89	7.10000E+04
8	4.60000E+07	49	1.05400E+06	90	6.73790E+04
9	4.40000E+07	50	1.00260E+06	91	5.65620E+04
10	4.20000E+07	51	9.77830E+05	92	4.08680E+04
11	4.00000E+07	52	9.53690E+05	93	3.51750E+04
12	3.80000E+07	53	9.07180E+05	94	3.18280E+04
13	3.60000E+07	54	7.80820E+05	95	2.80880E+04
14	3.40000E+07	55	7.06510E+05	96	2.60580E+04
15	3.20000E+07	56	6.72060E+05	97	2.47880E+04
16	3.00000E+07	57	6.39280E+05	98	2.41760E+04
17	2.75000E+07	58	6.08100E+05	99	2.35790E+04
18	2.50000E+07	59	4.28520E+05	100	2.18750E+04
19	2.25000E+07	60	3.17460E+05	101	2.13350E+04
20	2.00000E+07	61	1.92550E+05	102	1.93050E+04
21	1.86820E+07	62	1.85500E+05	103	1.03330E+04
22	1.69050E+07	63	1.84250E+05	104	9.11880E+03
23	1.49180E+07	64	1.83160E+05	105	8.04730E+03
24	1.34990E+07	65	1.74220E+05	106	7.10170E+03
25	1.19120E+07	66	1.69920E+05	107	6.26730E+03
26	1.00000E+07	67	1.67500E+05	108	5.53080E+03
27	8.82500E+06	68	1.66500E+05	109	5.00450E+03
28	7.59570E+06	69	1.65730E+05	110	4.09730E+03
29	6.06530E+06	70	1.61630E+05	111	3.03540E+03
30	4.84320E+06	71	1.49960E+05	112	1.23410E+03
31	3.86740E+06	72	1.42640E+05	113	4.54000E+02
32	2.86500E+06	73	1.38000E+05	114	1.67020E+02
33	2.46600E+06	74	1.36500E+05	115	6.14420E+01
34	2.42510E+06	75	1.35690E+05	116	2.26030E+01
35	2.38530E+06	76	1.29070E+05	117	8.31530E+00
36	2.36530E+06	77	1.27000E+05	118	3.05900E+00
37	2.34570E+06	78	1.22720E+05	119	1.12540E+00
38	2.30690E+06	79	9.80370E+04	120	4.13990E-01
39	2.26890E+06	80	8.65170E+04	121	1.52000E-01
40	2.23130E+06	81	8.40000E+04	LOWER	
41	1.87310E+06	82	8.25000E+04	BOUND	1.39000E-04

Figure 9. Iron Total Cross Section, 10 MeV to 1 BeV.

Figure 10. Iron Total Cross Section, 20. MeV to 110 Mev.

Figure 11. Iron Nonelastic Cross Section, 10 MeV to 1 BeV.

Figure 12. Iron Elastic Scattering Cross Section, 10 MeV to 1 BeV.

are given in table 5. All references assumed the radial limit $d = r_o A^{1/3}$.

Table 5
Optical Model Parameters

	Neutron Energy, MeV								
	4.10	7.00	14.00	20.00	24.80	30.00	40.00	50.00	60.00
V_S (MeV)	50.00	45.50	44.00	41.76	40.00	38.07	34.37	30.66	29.96
V_G (MeV)	7.0	7.50	11.00	11.00	11.00	11.00	11.00	11.00	11.00
V_T (MeV)	9.5	8.60	8.30	7.80	7.40	6.97	6.14	5.30	4.47
a (f)	0.65	0.65	0.65	0.70	0.70	0.70	0.70	0.70	0.70
b (f)	0.98	0.98	0.98	1.00	1.00	1.00	1.00	1.00	1.00
r_o (f)	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25	1.25

Optical model calculations were performed at 14 , 20 , 30 , 40 , 50 and 60 MeV, using the parameters of table 5 and the ABACUS code.⁹²) The calculated angular distributions are shown in figs. 13-14. The angular distributions above 20 MeV were combined with elastic scattering cross sections obtained from the numerics associated with fig.12 to produce a special ENDF/B format file. This file was then processed by MINX⁴⁵), producing multigroup P_5 elastic scattering cross sections.

There was some concern for the accuracy of medium energy cross sections processed by MINX⁴⁵) or NJOY⁴⁶), since these codes do not incorporate relativistic kinematics. These concerns are negated in Appendix B.

The multigroup elastic scattering cross sections were added to the 121-group library; the resulting library was used to calculate ($P_5 S_{16}, \delta r = 1. \text{cm}$) the spherical problem and the surface NDER. The surface NDER calculated with the 121-group library with elastic scattering above 20 MeV was 0.33% less than that calculated with the 121-group library with no medium-energy elastic scattering treatment.

Figure 13. Calculated Angular Distributions for 14, 20, and 30 MeV Neutrons Elastically Scattered by Iron.

Figure 14. Calculated Angular Distributions for 40, 50, and 60 MeV Neutrons Elastically Scattered by Iron.

As a result, elastic scattering above 20 MeV was ignored for all nuclides other than H in following calculations.

Flux Weighting Function

Multigroup neutron cross-section values are computed as the flux-weighted-average cross section over the energy range of each group. It is necessary to supply a flux weighting function representative of the general flux shape throughout the shield for use in cross-section processing. The multigroup scalar flux calculated at a radius of 20.5 cm. in the spherical shield was selected as a representative flux shape in the shield. The 121-group flux was approximated by a set of 26 log-log segments, which are superimposed upon the multigroup values in fig.15. The log-log flux approximation to the flux at $r=20.5$ cm. is used as a flux weighting function in all following cross-section processing.

Cross-Section Averaging

The formation of the 121-group library required the development of a new system for combining the processed ENDF/B-IV data and the results of the CROIX IC+E model calculations. The 41-group library was computed by trapazoidal integration of cross-section data¹²⁴⁾ using the available point energy cross sections calculated at the multigroup boundaries. For example, the cross section for transfer from group 9 (25-50 MeV) to group 15 (10-12 MeV) was found as the average of the nonelastic scattering cross sections from 25 MeV to group 15 and from 50 MeV to group 15. This treatment is not inappropriate to the energy distribution of secondary neutrons at medium energies, which is composed of a slowly-varying medium energy distribution of cascade neutrons reflecting the incident particle's energy and an evaporation neutron spectrum that is less dependent on the incident particle's energy. However, it is desirable to use data for more than two source energy points, particularly in computing

Figure 15. 121-Group Flux at 20.5 cm., With Log-
Log Flux Weighting Function Approximation.

TABLE 6

CROSS SECTION FLUX-WEIGHTING FUNCTION

ENERGY (EV.)	FLUX VALUE (N/CM**2 -SEC-EV)	ENERGY (EV.)	FLUX VALUE (N/CM**2 -SEC-EV)
1.00E-05	3.00E-03	9.00E+04	3.30E-03
2.50E-01	1.60E+00	1.50E+05	2.35E-03
4.00E-01	1.90E+00	4.25E+05	1.00E-03
1.00E+00	1.80E+00	7.00E+05	5.30E-04
2.00E+00	1.55E+00	6.00E+06	2.80E-06
5.00E+00	1.00E+00	9.00E+06	1.80E-06
2.00E+01	4.35E-01	1.10E+07	1.65E-06
9.00E+02	2.00E-02	1.50E+07	1.70E-06
1.60E+03	1.60E-02	2.00E+07	2.00E-06
3.50E+03	1.25E-02	2.20E+07	1.95E-06
5.00E+03	1.00E-02	3.20E+07	5.60E-07
8.00E+03	7.00E-03	3.50E+07	3.20E-07
1.50E+04	4.80E-03	6.00E+07	1.00E-08
5.00E+04	4.00E-03		

LOG-LOG INTERPOLATION IS USED BETWEEN VALUES

the in-group scattering cross section.

The DANA5 code¹¹⁴) was constructed to allow a better description of medium energy multigroup cross sections. The code divides each medium energy source group G into J energy increments. The cross section for scattering from the midpoint $\bar{E}_j$ of each energy increment to a secondary energy group g above 1 MeV is determined in terms of the neutron yield of $\bar{E}_j$, the nonelastic cross section at $\bar{E}_j$ and the fraction of the normalized secondary neutron distribution, expressed in the Legendre polynomial expansion, associated with group g .

The determination of this fraction of the normalized secondary neutron distribution is accomplished by first determining the energy limits of group g , expressed as fractions of $\bar{E}_j$. If group g has upper and lower bounds of E_g and E_{g+1} , respectively, then this range is $E_{g+1}/\bar{E}_j$ to $E_g/\bar{E}_j$. If $E_U > \bar{E}_j$ is the energy at which a CROIX calculation has been performed, the fraction of the normalized secondary neutron distribution associated with secondary neutrons between $E_U E_{g+1}/\bar{E}_j$ and $E_U E_g/\bar{E}_j$ is determined from the secondary neutron histories, which are preprocessed into a very fine secondary energy structure with Legendre polynomial expansions. A similar calculation is performed for the CROIX calculation at $E_L < \bar{E}_j$ for secondary neutrons with energies between $E_L E_{g+1}/\bar{E}_j$ and $E_L E_g/\bar{E}_j$. The fraction of the normalized secondary neutron distribution at $\bar{E}_j$ to group g is determined by interpolating between the values of E_L and E_U .

The yield and nonelastic cross section at $\bar{E}_j$ is determined by interpolating between tabulated evaluated values, and the transfer cross section from $\bar{E}_j$ to group g is then calculated as the product of the yield, nonelastic cross section and the distribution function. The group G to g cross section is determined from a histogram integration of these values.

Special treatment is given the source group with a lower limit of 20 MeV, the interface of the medium energy model-calculated cross sections and ENDF/B-IV processed multigroup values. The

ENDF/B-IV processed data includes a very narrow group at 20 MeV. The elastic scattering contribution to the transfer cross sections from this narrow group are removed, and the code uses the secondary neutron energy-angle distribution in the same manner as higher energy point energy calculations. This treatment prevents the interpolation between a secondary neutron angle-energy distribution which includes elastic scattering and one that does not.

Sensitivity Analysis

The directed fluxes at each interval of the shield geometry and the directed adjoint fluxes relative to the adjoint source (i.e., the 121-group histogram NFDE response function) at the shield surface were calculated with the DTF-IV code⁷⁶), which was modified to produce an interface file for the ALVIN code⁸⁸) The results of the sensitivity calculation are shown in the sensitivity profile of fig.16. It is significant to recall that the 121-group structure was chosen to reflect the structure of the iron total cross section below 20 MeV. The iron total cross section is replete with resonance structure in the range from 5 keV to 6 MeV , with deep minima throughout. The importance of these windows to the surface NDFR is obvious in fig.16.

The ALVIN code selects a new group structure of NG groups, each corresponding to an equal area under the sensitivity profile, with the 20 MeV interface energy forced as one boundry. The 60-group structure determined by ALVIN, with small changes to allow for compatibility with cross section features and other group structures⁶²), is listed in table 7. The production of the 60-group library is described in the following section.

Figure 15. 121-Group Sensitivity Profile

TABLE 7

60-GROUP STRUCTURE BOUNDARIES

GROUP NO.	UPPER BOUND (EV.)	GROUP NO.	UPPER BOUND (EV.)
1	6.0000E+07	32	3.9328E+05
2	3.3018E+07	33	3.4446E+05
3	2.8459E+07	34	3.0225E+05
4	2.5817E+07	35	2.6599E+05
5	2.3839E+07	36	2.3407E+05
6	2.2023E+07	37	2.0599E+05
7	2.0000E+07	38	1.8486E+05
8	1.7252E+07	39	1.8365E+05
9	1.3351E+07	40	1.7957E+05
10	6.9088E+06	41	1.6740E+05
11	3.1591E+06	42	1.6538E+05
12	2.0739E+06	43	1.5539E+05
13	1.5671E+06	44	1.5000E+05
14	1.2236E+06	45	1.4000E+05
15	1.0727E+06	46	1.3636E+05
16	9.4796E+05	47	1.3300E+05
17	9.0274E+05	48	1.2725E+05
18	8.4000E+05	49	1.1880E+05
19	7.3630E+05	50	1.0290E+05
20	6.7558E+05	51	8.2213E+04
21	6.5444E+05	52	8.1251E+04
22	6.3730E+05	53	7.0322E+04
23	6.2634E+05	54	5.9905E+04
24	6.1556E+05	55	2.6968E+04
25	5.9866E+05	56	2.4589E+04
26	5.8300E+05	57	2.3946E+04
27	5.6300E+05	58	2.2853E+04
28	5.3800E+05	59	2.1369E+04
29	5.0800E+05	60	1.5780E+04
30	4.7200E+05	LOWER	
31	4.3655E+05	BOUND	1.3900E-04

4 THE 60-GROUP LIBRARY

All work previously described has concerned pure iron shield calculations. The shield-collimator unit presently in use, illustrated in fig.2 (p.4), is constructed of gray cast iron and WEP with boron and iron. These are the materials to be used in the shield design at hand, and the cross section library must include all elements contained in these materials, as well as materials of future interest. This combination of materials, described in table 8, requires that the library include H,B,C,N,O,Si,Fe and W.

All cross section sets include ENDF/B-IV data processed into the lower 54 groups of the 60-group structure, using the weighting flux of table 6. In fact the data is processed into 55 groups, including a very narrow group at 20 MeV as described in the preceeding section. The 55-group data is combined with medium energy data by the DANA5 code¹¹⁴) in a manner previously described.

Iron

The iron cross-section set is described initially since it is the principle shielding material constituent. Some additional one-dimensional spherical shield calculations were performed to investigate additional uncertainties. Natural iron is described as a single material (MAT 1192) in the ENDF/B-IV data library.

Cross-Section Self Shielding

Because of the pronounced resonance structure of the iron cross section, it is necessary to investigate the effects of self-shielding on the calculation of multigroup cross-section values (i.e., the

TABLE 8

MATERIALS ANALYSIS

POLYETHYLENE DENSITY=0.9 GM/CM**3

ELEMENT	WEIGHT FRACTION	ATOM DENSITY (ATOMS/CM**3)	ATOM FRACTION ACTUAL	USED
H	.14371403	7.73109562E+22	.6666667	.6666667
C	.85628597	3.86554781E+22	.3333333	.3333333
TOTAL	1.	1.15966434E+23	1.	1.

BENELEX-401 DENSITY=1.4 GM/CM**3

ELEMENT	WEIGHT FRACTION	ATOM DENSITY (ATOMS/CM**3)	ATOM FRACTION ACTUAL	USED
H	.065	5.43927161E+22	.48512077	.48512077
C	.482	3.38473590E+22	.30187971	.30187971
O	.453	2.38819344E+22	.21299952	.21299952
TOTAL	1.	1.12122010E+23	1.	1.

WEP WITH BORON DENSITY= 1.0785 GM/CM**3

ELEMENT	WEIGHT FRACTION	ATOM DENSITY (ATOMS/CM**3)	ATOM FRACTION ACTUAL	USED
H	.08408865	5.42071885E+22	.55923484	.559235
B	.00743488	4.46863832E+20	.00461012	.004610
C	.42307360	2.28868377E+22	.23611475	.236115
N	.00065797	3.05237139E+19	.00031491	.000315
O	.46641292	1.89423556E+22	.19542104	.195421
SI	.01776270	4.10947364E+20	.00423959	.004239
FE	.0	.0	.0	.000065
CO	.00056928	6.27672371E+18	.00006475	.0
TOTAL	1.	9.69309934E+22	1.	1.

(CONTINUED NEXT PAGE)

WEP WITH BORON AND IRON DENSITY=2.1770 GM/CM**3

ELEMENT	WEIGHT FRACTION	ATOM DENSITY (ATOMS/CM**3)	ATOM FRACTION	
			ACTUAL	USED
H	.02901858	3.77601698E+22	.45547363	.455474
B	.00258264	3.13331021E+20	.00377949	.003779
C	.14599345	1.59419527E+22	.19229625	.192296
N	.00022705	2.12613613E+19	.00025646	.000257
O	.16095838	1.31951790E+22	.15916390	.159164
SI	.00612346	2.85964671E+20	.00344938	.003449
FE	.65490000	1.53808569E+22	.18552815	.185581
CO	.00019644	4.37195105E+18	.00005274	.0
TOTAL	1.	8.29030874E+22	1.	1.

WEP WITH BORON AND TUNGSTEN DENSITY= 2.5339 GM/CM**3

ELEMENT	WEIGHT FRACTION	ATOM DENSITY (ATOMS/CM**3)	ATOM FRACTION	
			ACTUAL	USED
H	.02901858	4.39506175E+22	.52303244	.523032
B	.00258264	3.65070339E+20	.00434450	.004345
C	.14599345	1.85554955E+22	.22081888	.220819
N	.00022705	2.47469745E+19	.00029450	.000294
O	.16095838	1.53584125E+22	.18277213	.182772
SI	.00612346	3.32846063E+20	.00396102	.003961
FE	.0	.0	.0	.000061
CO	.00019644	5.08869397E+18	.00006056	.0
W	.6549	5.43810749E+21	.06471597	.064716
TOTAL	1.	8.40303851E+22	1.	1.

GRAY CAST IRON DENSITY= 7.2032 GM/CM**3

ELEMENT	WEIGHT FRACTION	ATOM DENSITY (ATOMS/CM**3)	ATOM FRACTION	
			ACTUAL	USED
C	.0355	1.28280000E+22	.1430414	.1430414
SI	.0215	3.32520000E+21	.0370784	.0432469
P	.0025	3.50210000E+20	.0039051	.0
S	.0015	2.02990000E+20	.0022634	.0
MN	.0070	5.52940000E+20	.0061657	.0
FE	.9320	7.24210000E+22	.8075460	.8137117
TOTAL	1.	8.96803400E+22	1.	1.

effects of flux suppression due to the cross-section resonances and flux peaking due to neighboring cross-section minima). This effect is obvious in the 121-group flux diagram of fig.15 (p.45).

The MINX⁴⁵) and NJOY⁴⁶) neutron cross-section processing codes use the Bondarenko¹²⁵) scheme to describe the effect of flux perturbation on multigroup cross section values. Here the flux weighting function is described by:

$$\phi(E, \sigma_0) = \frac{C(E)}{\sigma_t(E) + \sigma_0} \quad (31)$$

where $C(E)$ represents the broad energy behavior of the energy-dependent flux (e.g., the flux weighting function of table 6), $\sigma_t(E)$ is the energy-dependent total cross section and σ_0 is the background group constant. The flux weighting function is thus strongly perturbed if a very small value of σ_0 (e.g., $\sigma_0 = 0.01\text{b}$) is used, forcing the flux to reflect a $1/\sigma_t(E)$ behavior. Large values of σ_0 result in essentially no flux perturbation due to the energy dependence of the cross section.

Sixty-group cross-section sets for iron were processed at five values of σ_0 , ranging from 0.01b . to 10^{10}b . Each cross section set was used to calculate the spherical iron shield ($P_5S_{16}, \delta R = 1\text{cm.}$); surface NDER values calculated are compared in table 9.

Table 9

Calculated Surface NDER Values
Corresponding to Different σ_0 Values

σ_0 (barns)	Surface NDER
0.01	1.97228E-13
0.4	-----
1.0	1.60817E-13
5.0	-----
1.E10	1.91145E-13

The range of NDER values calculated indicate that flux peaking at the iron cross section minima may strongly affect multigroup cross section values, with neutrons streaming from the system through these cross section windows. The above range of values, differing by as much as 95% from the infinite dilution ($\sigma_0 \gg \sigma_t(E)$) calculation, demand careful attention to the self shielding problem.

The above calculation describes a pure iron shield. The principle shielding material, gray cast iron, is approximately 81% (at.) iron, with the balance made up of carbon and silicon. One might expect these other constituents to diminish flux peaking by providing a background cross section σ_0 of considerable magnitude. Smith and Soran⁵⁶⁾ suggest that, in a multi-isotope material, the background cross section appropriate for iron at energy E should be

$$\sigma_0(E) = \frac{1}{N_{Fe}} \{N_c \sigma_t^c(E) + N_{Si} \sigma_t^{Si}(E)\} \quad (32)$$

where N_{Fe} , N_c and N_{Si} are the constituent atom number densities and $\sigma_t^c(E)$ and $\sigma_t^{Si}(E)$ are the energy-dependent total cross sections of carbon and silicon, respectively. Values of $\sigma_0(E)$ calculated from (32) are listed in table 10 corresponding to the energies of a number of iron cross section minima.

Table 10

Values of $\sigma_0(E)$ for Iron in Gray Cast Iron

Energy (eV)	$\sigma_0(E)$ (barns)
2.50E+04	0.92
7.10E+04	0.95
1.38E+05	0.79
1.84E+05	0.73
6.40E+05	0.70
2.20E+06	0.42

MacFarlane and Becker¹²⁶) have modeled an iron transport problem with both multigroup S_N and Monte Carlo calculations. Monte Carlo histories were binned to determine reaction rates, effective group cross sections and effective σ_0 values. In addition to the variation of σ_0 with energy, appropriate σ_0 values were seen to decrease as the distance from the source increased.

It would thus seem appropriate to form a cross section set using different σ_0 values for each group. The spatial variation of σ_0 could then be represented with the use of a number of spatial regions with different associated cross-section sets, each calculated with a unique set of group σ_0 values. Such an elaborate treatment will not be pursued in this work; the iron cross section set used in describing gray cast iron were processed with $\sigma_0=0.4$ barns, the minimum value of table 10.

Low-Energy Neutron Source

The neutron source spectrum of Hertel and Smathers¹¹⁵), described in section 3, extends to a lower energy of 4.11 MeV. The spectrum of neutrons below this energy is not known, and previous calculations have not included neutrons in the low-energy groups. It is necessary to investigate the significance of low-energy source neutrons in calculating surface NDER values.

The source was alternately assumed to step to zero below 4.11 MeV, decrease linearly to zero below this energy, or remain constant at low energies. These low-energy source treatments are reflected in fig.17. The spherical iron shield was calculated with the 60-group infinite dilution iron cross section set for each of the three source descriptions. The source spectrum above 4.11 MeV remained constant; the neutron source rate associated with source spectra extending below 4.11 MeV was thus greater than 1 neutron/second. The source rates and calculated surface NDER values are listed in table 11. Because of the complexities of the anisotropic

Figure 17. Low-Energy Neutron Source Approximation

Table 11

Low-Energy Neutron Source Treatments

<u>Low-Energy Source Treatment</u>	<u>Neutron Source Rate (n/s)</u>	<u>Surface NDER</u>
Zero	1.000000	1.01145E-13
Linear	1.020660	1.02593E-13
Constant	1.078538	1.06173E-13

source description of section 6, the low-energy source neutrons are ignored in future calculations and the associated error accepted.

Iron Cross Sections

The iron medium-energy neutron cross section data of Appendix A is shown in fig.18 along with ENDF/B-IV data, calculated values and the medium energy evaluation of this work. Values calculated for ^{56}Fe with the IC+E model are considerably below ENDF/B-IV data and measured nonelastic cross section values. Values calculated with the optical model, as described in section 3, appear to be reasonable. The evaluated and IC+E model values for the iron non-elastic cross section are compared in table 12. The evaluation extends to 1 BeV because of the existence of measured values¹²⁷⁾ above 379. MeV.

Figure 18. Total and Nonelastic Cross Sections of Iron

Table 12

Iron Nonelastic Cross-Section Values

Energy (MeV)	Evaluated σ_{ne} (E) (barns)	IC+E Model σ_{ne} (E) (barns)
15.	-----	1.08332
20.	1.3248	1.03796
25.	1.25	0.99026
33.	1.155	0.93431
42.	1.075	0.89172
50.	1.02	0.85867
60.	0.97	0.82242
80.	0.899	-----
100.	0.849	0.73451
150.	0.780	-----
200.	0.752	0.67217
300.	0.742	0.65816
400.	0.767	0.67761
500.	0.792	0.70634
600.	0.824	0.73038
700.	0.852	0.74309
800.	0.874	0.74691
900.	0.893	-----
1000.	0.903	-----

Hydrogen

The total cross section of hydrogen, shown in fig.19, is composed almost entirely of the elastic scattering cross section. The only other contribution to the total cross section is from radiative capture, which is 26.04 $\mu\text{b.}$ at 20 MeV. Medium energy elastic-scattering cross sections and angular distributions were calculated for ^1H with the phase shift model code NPSCAT⁹⁹), using the phase shift data of Seamon⁹⁸) as described in section 2. These cross section values and angular distributions were combined with the data of the ENDF/B-IV data file for ^1H (MAT1269) to form a special ENDF/B-IV formatted data file (MAT269) for hydrogen. This special file was processed by NJOY⁴⁶) to produce the infinitely dilute 60-group cross-section set. Errors associated with the nonrelativistic kinematic calculations of NJOY are examined in Appendix B and are shown to be negligible. The 0.015% abundant ^2H was ignored.

Boron

Boron appears as a minor constituent in WEP mixtures because of the high thermal (n, α) cross section of ^{10}B . There is no ENDF/B-IV evaluation for elemental B; the two constituents, 19.8% ^{10}B (MAT1273) and 80.2% ^{11}B (MAT1160) were processed separately into infinitely dilute 55 group values by NJOY⁴⁶) and combined, according to their isotopic abundances, by the SIGMIX2 code¹¹⁴).

The WEP mixtures consist of less than 0.5% (at.) boron. The small contribution to the WEP mixture group macroscopic cross sections above 20 MeV from boron does not justify IC+E model calculations. The medium energy nonelastic cross sections of boron were thus estimated, using the ENDF/B-IV values above 10 MeV and the carbon evaluation as a guide. The energy and angular distributions of carbon IC+E model calculations were combined by DANA5 with the

Figure 19. Total Cross Section of Hydrogen

estimated cross section magnitude and 55-group values produced by SIGMIX2 to produce the 60-group boron cross-section set.

Carbon

Carbon is described in the ENDF/B-IV library only by its 98.89% abundant ^{12}C isotope. The ENDF/B-IV evaluation (MAT1274) is a particularly poor updating of the ENDF/B-III ^{12}C evaluation. The ENDF/B-III library evaluations extend only to 15 MeV; the ENDF/B-IV evaluations largely complimented the earlier version with recent data and model calculations, extending the cross section descriptions to 20. MeV. This extension for the ^{12}C evaluation consisted of adding a single energy point at 20. MeV with the cross section value of the 15. MeV point, as seen in fig.20. The total cross-section structure indicated by measured values in the 15.-20. MeV range suggest structure in cross sections contributing to neutron transfer. However, this variation in the measured values is on the order of +2% to +10% from the simple treatment of the MAT1274 evaluation; one might thus expect errors of this magnitude in the determination of multigroup values in the 15.-20. MeV range. MAT1274 was processed, without modification, by the NJOY code into infinitely dilute 55-group values. The contributions of ^{13}C and ^{14}C to the carbon cross-section values were not considered.

The evaluated medium energy nonelastic cross-section values are compared to those calculated with CROIX in table 13. As was the case for iron, the evaluation extends to 1 BeV because of the existance of measured values¹²⁷⁾ above 379 MeV. DANA5 was used to produce the 60-group values.

Nitrogen

Of the two naturally present isotopes of nitrogen, only the principle 99.64% (at.) ^{14}N is described in ENDF/B-IV. The ^{14}N

Figure 20. Total and Nonelastic Cross Sections of Carbon

Table 13

Carbon Nonelastic Cross-Section Values

Energy (MeV)	Evaluated σ_{ne} (E) (barns)	IC+E Model σ_{ne} (E) (barns)
20.	0.5289	0.48358
25.	0.467	-----
30.	0.415	-----
33.	0.382	0.40891
40.	0.330	-----
42.	0.321	0.37072
50.	0.285	-----
55.	0.276	-----
60.	0.264	0.32839
70.	0.252	-----
80.	0.242	-----
90.	0.233	-----
100.	0.230	0.26707
125.	-----	0.25048
140.	0.218	-----
200.	0.221	-----
300.	0.230	-----
1000.	0.267	-----

evaluation, MAT1275, was processed by NJOY to produce the 55-group infinite distribution values; these values were combined with the secondary neutron distributions of the ^{14}N CROIX calculations by DANA5. The ^{14}N evaluated medium energy nonelastic cross sections, shown in fig.21, are compared to the IC+E model values in table 14. Cross section values of ^{15}N are not included in the nitrogen cross section set.

Figure 21. Total and Nonelastic Cross Sections of Nitrogen

Table 14

Nitrogen Nonelastic Cross-Section Values

Energy (MeV)	Evaluated $\sigma_{ne}^{(E)}$ (barns)	IC+E Model $\sigma_{ne}^{(E)}$ (barns)
20.	0.5655	0.52230
25.	0.524	0.48794
30.	0.491	-----
33.	0.478	0.44499
42.	0.447	0.40796
50.	0.424	0.38195
60.	0.400	0.35690
100.	-----	0.29932
125.	-----	0.27969

Oxygen

Oxygen, like C and N, is described only by its principle isotope, 99.756% (at.) ^{16}O , in ENDF/B-IV. The ^{16}O evaluation (MAT1276) was processed by NJOY, producing 55-group values that were combined by DANA5 with the secondary neutron distributions from CROIX and the nonelastic cross section evaluated values shown in fig.22. It should be noted that, of the six elements modeled by the IC+E model, oxygen is the only nuclide well described by the model as to the nonelastic cross-section magnitude. The evaluated and calculated nonelastic cross section values are compared in table 15.

Figure 22. Total and Nonelastic Cross Sections of Oxygen

Table 15

Oxygen Nonelastic Cross-Section Values

Energy (MeV)	Evaluated $\sigma_{ne}^{(E)}$ (barns)	IC+E Model $\sigma_{ne}^{(E)}$ (barns)
20.	0.596	0.56344
25.	0.545	0.51969
30.	0.498	-----
33.	0.4825	0.4825
42.	0.44415	0.44415
50.	0.41628	0.41628
60.	0.39148	0.39148
70.	0.376	-----
100.	-----	0.32720
125.	-----	0.30634
200.	-----	0.27947
300.	-----	0.27061
400.	-----	0.27773
500.	-----	0.29723
600.	-----	0.30894
700.	-----	0.31711
800.	-----	0.31973

Silicon

ENDF/B-IV includes an evaluation (MAT1194) for elemental silicon, which was processed by NJOY to produce 55-group infinite dilution values. These were combined in the above fashion with the IC+E model secondary neutron distributions by DANA5, using the evaluated nonelastic cross-section values shown in fig.23. The evaluated values are compared to those calculated by CROIX in table 16.

Figure 23. Total and Nonelastic Cross Sections of Silicon

Table 16

Silicon Nonelastic Cross-Section Values

Energy (MeV)	Evaluated σ_{ne} (E) (barns)	IC+E Model σ_{ne} (E) (barns)
20.	0.95974	0.74392
25.	0.872	0.70626
30.	0.807	-----
33.	0.784	0.65349
40.	0.729	-----
42.	0.725	0.61691
50.	0.683	0.58096
60.	0.647	0.51712
100.	-----	0.47944
200.	-----	0.41878
300.	-----	0.39381
400.	-----	0.42311
500.	-----	0.44387
600.	-----	0.46093
700.	-----	0.47250
800.	-----	0.47315

Tungsten

Natural tungsten is composed of five isotopes. Only the four most abundant are described in the ENDF/B-IV library, and these four are combined to form the tungsten data set as described by table 17. Each tungsten evaluation was processed by NJOY, producing four 55-group infinite dilution cross section sets. These were combined by SIGMX2¹¹⁴), according to the abundances described in table 17., to produce a 55-group set for natural tungsten.

CROIX calculations were performed for ¹⁸⁴W, an average A value

Table 17

Isotopic Abundance of Tungsten Isotopes

Isotope	ENDF/B-IV MAT NO.	Natural Abundance	Abundance Used
^{180}W	N/A	0.1%	0.
^{182}W	1128	26.3%	26.4%
^{183}W	1129	14.3%	14.3%
^{184}W	1130	30.7%	30.7%
^{186}W	1131	28.6%	28.6%

representative of the isotopic mixture; trial CROIX calculations with greater than 23,000 interactions of neutrons on ^{183}W were within statistical agreement of similar calculations with ^{184}W . The secondary neutron distributions were combined with the 55-group cross-section set for natural tungsten by DANA5, using the evaluated tungsten nonelastic cross section values shown in fig.24. The evaluated values are compared to those calculated by CROIX in table 18.

Cross-Section Mixtures

The DTF-IV and TWOTRAN transport codes allow the input of NL cross section tables for each element used to form materials in the calculation. Each table of the 60-group library requires 3780 words of storage; each element requires 4 such blocks of storage. Since two cross-section sets for iron, corresponding to different σ_0 values, are used in constructing cross sections for the gray cast iron and WEP with boron and iron mixtures of table 8 (p.52), the total storage required to input, mix and store cross-section tables would be 120,960 words. The storage required may be reduced to 30,240 words by mixing the cross section tables, value-by-value,

Figure 24. Total and Nonelastic Cross Sections of Tungsten

prior to transport calculations.

Material cross-section tables have been prepared for all of the materials listed in table 8 (p.51). The total number density values of table 8 are then appropriate for the material cross section tables thus produced.

Table 18

Tungsten Nonelastic Cross-Section Values

Energy (MeV)	Evaluated σ_{ne} (E) (Barns)	IC+E Model σ_{ne} (E) (barns)
20.	2.5623	1.89688
25.	2.537	1.84428
30.	2.472	-----
33.	2.426	1.78168
40.	2.315	-----
42.	2.278	1.73295
50.	2.185	1.68452
60.	2.065	1.63686
70.	1.981	-----
100.	-----	1.38810
200.	-----	1.45163
300.	-----	1.44818
400.	-----	1.46931
500.	-----	1.52349
600.	-----	1.56322
700.	-----	1.55667
800.	-----	1.56651

5 SHIELD DESIGN

In the previous sections some of the parameters and data necessary to perform the final shield design calculations were resolved. The shield design criterion was presented in section 1. In section 3 the neutron source spectrum and response function were examined, and the minimum detail in angular quadrature, cross section expansion and spatial mesh dimensions were investigated. Uncertainties associated with the use of these minimum values and the failure to incorporate reaction gamma rays and medium energy neutron elastic scattering were evaluated in section 3. The development of the cross-section library, including considerations of the flux weighting function, resonance self shielding, group structure, model calculations, cross-section processing, and materials was described in sections 3 and 4.

Anisotropic Neutron Source

In section 2 the measured spectra of neutrons produced by 50 - 53.8 MeV neutrons on beryllium, shown in fig.6(p.26), were examined. The data of Hertel and Smathers¹¹⁵) was used to construct multigroup source values for one-dimensional transport calculations. This data, however, describes the neutron spectrum in the forward direction only; no measurements were made of the angular distribution of neutrons. In order to construct an anisotropic neutron source for design calculations one must estimate the angular distribution of neutrons in each energy group.

The data of Meulders, et al.¹¹⁾ at 0° was rejected because of its disagreement with the data of others. This data, however, included angular distributions for neutrons in 1 MeV energy bins at $0^\circ, 5^\circ, 10^\circ, 15^\circ, 25^\circ, 40^\circ$, and 64° . It may be assumed that the

time-of-flight data of Meulders differs from that of Hertel and Smathers and of Northcliffe because of errors incurred in the Meulders experiment in estimating the energy-dependent detector efficiency and/or the effective solid angle subtended by the detector. These errors are common to all angular measurements for each energy bin. The relative angular variation of the yield of neutrons with an energy bin may therefore then be considered to be well described by these angular distribution measurements. The group source values at the above angles, adjusted to agree with the 0° spectrum of Hertel and Smathers, were formed by first computing group source values $Q_g^M(\psi)$ from the data of Meulders, et al. at each of the above seven angles. The source values at these angles were thus described by

$$Q_g(\psi) = \{Q_g^{HS}(0^\circ) / Q_g^M(0^\circ)\} Q_g^M(\psi) \quad (33)$$

where $Q_g^{HS}(0^\circ)$ are the forward direction group source values of Hertel and Smathers. Multigroup source angular distributions over the range $0^\circ \leq \psi \leq 180^\circ$ were constructed by graphical extrapolation of a semi-log plot of the $Q_g(\psi)$ values. The resulting neutron source group values are evaluated in table 19 at several values of η , the cosine of the angle ψ of figure 4 (p.10).

Two source input options for the TWOTRAN code were considered. The distributed anisotropic source may be input to TWOTRAN in the form of spherical harmonic expansion coefficients. Alternately sources at a surface maybe described as values of $Q_g(\psi_j)$, where the ψ_j values are discrete ordinate values. This latter option was the source input selected for the two-dimensional calculations; the $Q_g(\psi_j)$ values are among the source values listed in table 19.

TABLE 19
DIRECTED NEUTRON SOURCE VALUES

ETA	NEUTRONS/ DEUTERON-SH.									
	GROUP 1	GROUP 2	GROUP 3	GROUP 4	GROUP 5	GROUP 6	GROUP 7	GROUP 8	GROUP 9	GROUP 10
1.00000	4.5868E-03	4.8527E-03	4.9642E-03	5.2619E-03	5.9560E-03	7.0992E-03	9.1769E-03	1.0693E-02	1.1938E-02	6.6566E-03
.99510	3.9946E-03	3.8258E-03	3.6793E-03	3.5329E-03	4.1932E-03	5.0087E-03	6.6375E-03	8.2964E-03	1.0213E-02	6.3716E-03
.98481	2.9311E-03	2.3292E-03	2.0522E-03	1.9328E-03	2.2035E-03	2.6572E-03	3.7048E-03	5.1202E-03	7.7742E-03	6.1206E-03
.98203	2.6270E-03	2.0697E-03	1.8309E-03	1.7891E-03	2.1016E-03	2.4678E-03	3.4708E-03	4.7285E-03	7.3555E-03	5.9863E-03
.96593	1.9360E-03	1.4342E-03	1.2332E-03	1.1427E-03	1.3116E-03	1.5827E-03	2.3133E-03	3.4023E-03	5.8382E-03	5.5341E-03
.96230	1.7195E-03	1.2737E-03	1.1145E-03	1.0655E-03	1.2737E-03	1.5443E-03	2.2440E-03	3.2538E-03	5.7316E-03	5.3813E-03
.94538	1.3374E-03	9.7118E-04	8.7565E-04	8.5973E-04	1.0030E-03	1.2259E-03	1.8309E-03	2.5474E-03	4.9355E-03	5.1743E-03
.93500	1.1622E-03	8.7381E-04	7.8013E-04	7.6421E-04	9.0739E-04	1.1145E-03	1.7035E-03	2.4678E-03	4.6967E-03	5.0151E-03
.91058	8.9954E-04	6.6868E-04	6.1296E-04	5.9704E-04	7.3237E-04	8.9954E-04	1.4249E-03	2.1971E-03	4.1872E-03	4.7124E-03
.90631	8.6292E-04	6.4623E-04	5.8557E-04	5.8701E-04	7.1501E-04	8.9317E-04	1.3905E-03	2.1907E-03	4.0614E-03	4.5375E-03
.85000	6.3684E-04	4.2191E-04	3.7414E-04	3.9006E-04	4.7763E-04	6.4480E-04	1.0110E-03	1.7513E-03	3.5026E-03	4.0917E-03
.83302	5.8589E-04	3.7414E-04	3.3434E-04	3.5185E-04	4.3305E-04	5.9704E-04	9.3456E-04	1.6239E-03	3.3434E-03	3.9494E-03
.79332	5.0947E-04	2.9613E-04	2.6110E-04	2.7862E-04	3.5026E-04	4.7763E-04	7.8172E-04	1.3979E-03	3.0250E-03	3.6931E-03
.76604	4.7289E-04	2.6079E-04	2.2608E-04	2.4104E-04	2.9486E-04	4.1824E-04	7.0928E-04	1.2871E-03	2.8130E-03	3.5376E-03
.74746	4.4897E-04	2.2767E-04	2.0379E-04	2.1653E-04	2.6270E-04	3.7733E-04	6.4162E-04	1.1829E-03	2.6906E-03	3.4705E-03
.68813	3.8520E-04	1.7832E-04	1.5603E-04	1.5921E-04	1.8628E-04	2.7862E-04	4.7763E-04	1.0508E-03	2.3035E-03	3.1842E-03
.65075	3.5345E-04	1.6080E-04	1.3421E-04	1.3333E-04	1.5841E-04	2.3563E-04	4.0599E-04	8.4361E-04	2.1493E-03	3.0253E-03
.60000	3.1842E-04	1.4010E-04	1.0906E-04	1.0885E-04	1.2578E-04	1.8628E-04	3.3434E-04	7.0052E-04	1.8946E-03	2.8021E-03
.57735	3.0091E-04	1.3326E-04	9.9506E-05	1.0030E-04	1.1415E-04	1.6876E-04	3.0409E-04	6.4480E-04	1.7632E-03	2.7066E-03
.53639	2.7703E-04	1.2227E-04	8.5177E-05	8.6769E-05	9.7596E-05	1.4170E-04	2.6270E-04	5.5405E-04	1.6239E-03	2.5474E-03
.43937	2.2465E-04	1.0204E-04	5.3902E-05	6.6948E-05	7.4287E-05	1.0175E-04	1.9471E-04	4.1204E-04	1.2923E-03	2.1716E-03
.39119	2.0857E-04	9.6322E-05	5.5723E-05	6.1296E-05	6.7664E-05	9.1546E-05	1.7991E-04	3.4708E-04	1.1941E-03	2.0220E-03
.23009	1.6319E-04	7.9605E-05	4.1713E-05	4.9355E-05	5.3335E-05	7.1644E-05	1.4170E-04	2.1971E-04	8.5973E-04	1.5921E-03
.19233	1.5603E-04	7.5943E-05	3.8802E-05	4.7763E-05	5.0947E-05	6.8460E-05	1.3435E-04	1.9931E-04	7.8491E-04	1.5284E-03
.13344	1.4450E-04	7.2122E-05	3.6618E-05	4.4897E-05	4.7763E-05	6.3684E-05	1.2578E-04	1.7195E-04	6.9255E-04	1.4170E-03
0.00000	1.2418E-04	6.4430E-05	3.1842E-05	4.0121E-05	4.2191E-05	5.6520E-05	1.0826E-04	1.3453E-04	5.2539E-04	1.2100E-03
-.60000	6.7664E-05	4.6171E-05	1.9424E-05	2.9931E-05	2.8658E-05	4.0121E-05	6.3654E-05	7.8013E-05	1.7672E-04	6.6072E-04
-.80000	5.7316E-05	4.2509E-05	1.7035E-05	2.8180E-05	2.6110E-05	3.9006E-05	5.7316E-05	6.9566E-05	1.4010E-04	5.8433E-04
-.90000	5.4131E-05	4.1395E-05	1.6080E-05	2.7384E-05	2.5633E-05	3.8051E-05	5.5723E-05	6.6669E-05	1.3055E-04	5.5943E-04
-1.00000	5.2539E-05	4.1235E-05	1.5921E-05	2.7066E-05	2.5474E-05	3.8051E-05	5.5405E-05	6.5276E-05	1.2737E-04	5.5405E-04

Design Considerations

The 6-Group Library

In order to perform preliminary TWOTRAN⁷⁷) calculations at reasonable expense the 60-group library was collapsed to a 6-group structure. Each group of this coarse group structure corresponds to ten groups of the 60-group structure described in table 7 (p.50). The anisotropic neutron source, described in table 19 for the ten highest energy groups of the 60-group structure are thus contained in the highest energy group of the 6-group structure. The sixty multigroup flux values from DTF-IV were used in collapsing the 60-group cross sections and NFDE values. The 6-group library was validated by calculating the spherical shield problem for gray cast iron, using the 6- and 60-group libraries and NFDE response values. Surface NDER values calculated with the two libraries were in agreement within 30%.

The 6-group library was intended for preliminary calculations only. However, the initial 6-group TWOTRAN calculation required 25 minutes of CDC-7600 CPU time. It immediately became apparent that an equivalent 60-group calculation would be prohibitively expensive. As a result, all TWOTRAN calculations were performed with the 6-group library.

Shield Geometry

The geometry of a radiotherapy shield-collimator unit rigorously satisfying the design criterion of section 1 would be described by a smooth, continuous outer surface along which the NDER is 0.5% of that in the collimated beam at 125 cm. from the beryllium target assembly. The resolution of such a geometry would require many iterative Monte Carlo neutron transport calculations, with successive trial geometries developed from the previous

results. This would be a costly approach to the design, resulting in a geometry that does not lend itself to efficient casting techniques.

The approach previously selected, using the discrete ordinates technique, requires that the shield geometry be represented in cylindrical geometry. The contour of the shield surface is thus approximated by a series of adjacent cylindrical discs, with the inside diameter describing the interface of the shield and collimator and the outside diameter approximating the continuous surface satisfying the design criterion.

Collimator inserts currently in use at the Texas A&M facility are cast of the WEP mixture previously described. The outside dimension of each collimator conforms to the shield's inside dimensions, described by three collinear rectangular parallelepipeds. The inside void of each is described by a truncated rectangular pyramid, projecting a rectangular irradiation field. The minimum lateral dimensions of the inner and outer surfaces of the collimator correspond to the inside (source) position of the shield.

The transport calculations for the shield design represent spatial variations in all quantities with 2.5 cm. mesh cells. This requires that the collimator's outer surface be described in increments of this dimension. The collimator used in the TWOTRAN calculations is compared to those currently in use in table 20.

Table 20

Outer Dimensions of Collimator Inserts

<u>Collimator</u>	<u>Inner Section</u>		<u>Central Section</u>		<u>Outer Section</u>	
	<u>length</u> <u>(cm.)</u>	<u>lateral</u> <u>(cm.)</u>	<u>length</u> <u>(cm.)</u>	<u>lateral</u> <u>(cm.)</u>	<u>length</u> <u>(cm.)</u>	<u>lateral</u> <u>(cm.)</u>
Now in use	17.78	8.73	35.56	14.92	17.15	20.00
This design	17.50	10.00	35.00	15.00	to face	20.00

Collimated Beam

Initial calculations were performed for a 70 cm. long, 85 cm. diameter cylindrical gray cast iron shield. The WEP collimator insert contained a 2 cm. diameter central beam hole. The neutron source of 1 neutron per second was confined to the R axis ($z=0$) and was uniformly distributed between $r=0$ and $r=1$ cm. Six group fluxes were calculated (S_8P_3) by TWOTRAN and converted to NDER values by DOSE2¹¹⁴⁾. Isodose curve points were calculated by DOSE2 by interpolating among the cell NDER values. These isodose curves, corresponding to half-decade multiples of the 0^0 face NDER, are shown in fig.25.

Examination of NDER values within the central void shows that the collimated beam is not well described by the TWOTRAN calculations. This observation is common to S_n calculations involving streaming, and significant improvement in the resolution of the collimated beam may only be accomplished through the major refinements of spatial mesh and quadrature.^{121,132)}

The collimated beam may be well described, however, by the measured neutron source spectrum at 0^0 . This value, 0.0711853 neutrons/deuteron-Sr., must be normalized to the 1 neutron/second source rate. This normalization was accomplished by first integrating the directed source values of table 19 over the forward 2π solid angle. The integration, performed by assuming an exponential source dependence between values of η , yielded a value of 0.051322 neutrons/deuteron for $0 \leq \eta \leq 1$. The neutron source rate of 1 neutron/second thus corresponds to an incident deuteron current of 19.485 deuterons per second. The normalized 0^0 directed source rate $\bar{Q}(0^0)$ is thus

$$\bar{Q}(0^0) = 19.485 \text{ d/s} \times 0.0711853 \text{ n/d-sr.} = 1.38705 \text{ n/sr.-s.} \quad (34)$$

Since the neutron source is confined to group 1 (3.1591 - 60 MeV) of the 6-group structure, the reference 0^0 NDER value D at 125 cm.

Figure 25. Isodose Profile for 70 cm. Cylinder. Isodose Curves Are Shown for Multiples of the 0^0 Surface NDER.

may be calculated using the group 1 NFDE response value of 3.97×10^8 Rem/n/cm² as follows:

$$\begin{aligned} D &= 1.38705 \text{ n/sr.-s} \times 3.97 \times 10^8 \text{ Rem/n/cm}^2 \times 1 \text{ sr./}(125 \text{ cm.})^2 \quad (35) \\ &= 3.5242 \times 10^{12} \text{ Rem/s} \quad . \end{aligned}$$

Shield Design Calculations

The geometry of the shield-collimator unit was obtained by first calculating a 1 meter long, 1.3 meter diameter cylindrical gray cast iron shield with WEP collimator inserts. All spatial increments within the shield were no larger than 2.5 cm., and those in the external void were no larger than 5 cm. The anisotropic neutron source was confined to the R axis (z=0), distributed between r=0 and r=1 cm. The DOSE2¹¹⁴) code was used to transform the 6-group scalar fluxes calculated by TWOTRAN to NDER values, as previously described, and to determine NDER isodose values throughout the volume of the shield-collimator unit and external void regions. These isodose values, corresponding to half-decade multiples of the NDER of the design criterion (0.5% D) are shown in fig.26.

This 1 meter long cylindrical unit satisfies or exceeds the design criterion. A more efficient geometry is obtained by approximating the contour of the isodose curve with stacked annuli, each having a thickness of some multiple of 2.5 cm. Such a geometry was constructed and is described in table 21. This geometry was calculated by TWOTRAN, and 6-group fluxes were used to produce the isodose curves of fig.27.

The NDER values at z=125 cm. are shown in fig.28 as a function of r, the distance from the collimated beam axis. Here the great disparity between NDER values calculated with TWOTRAN within the collimated beam and that calculated from the uncollided 0^0 directed

Figure 26. Isodose Profile for 1 meter Cylinder

Figure 27. Isodose Profile for Shield D sign

Table 21

Annular Components of Shield Unit

<u>Component</u>	<u>Thickness (cm.)</u>	<u>Inside radius (cm.)</u>	<u>Outside radius (cm.)</u>
1	17.5	5.0	65.0
2	17.5	7.5	65.0
3	17.5	7.5	65.0
4	5.0	10.0	65.0
5	15.0	10.0	60.0
6	5.0	10.0	55.0
7	5.0	10.0	50.0
8	5.0	10.0	45.0
9	2.5	10.0	40.0
10	2.5	10.0	32.5
11	2.5	10.0	25.0
12	2.5	10.0	15.0

neutron source are obvious. It is the conclusion of the author that the true NDER at $z=125$ cm. is well described at $r=0$ by the uncollided 0^0 directed source calculation, and that the NDER at $r \gtrsim 15$ cm. is well described by the results of the TWOTRAN calculation. The confluence of these two results is strongly effected by edge scattering near the orifice of the collimator and may be described only by Monte Carlo calculations of source neutrons in the forward cone about the beam axis.

The NDER values at $z=125$ cm. and $r > 15$ cm. are less than 0.1% of that calculated at $r=0$. Parnell and Kelsey³⁵) have concluded that, based on leukemia and carcinoma induction, a total body dose of 1.0% of that within the collimated beam is an acceptable goal in the design of neutron therapy machines. The acceptance of this higher NDER outside of the collimated beam would, of course, permit the design of smaller and less massive shield-collimator unit. The

geometry described in table 21, however, further decreases the patients' risk of radiation-induced complications.

The shield, as described, has a total mass of 8.4 tons, less than 60% of that of the shield currently in use. This geometry must be complimented with mechanical and side shielding additions as required by the mechanical design and casting features.

Figure 28. Radial NDER Values at $z = 125$ cm., Design Geometry

6 CONCLUSIONS

In the development of the nuclear data library a number of pertinent considerations have been examined relative to their effects on shield penetration. These include the multigroup energy structure, neutron flux weighting function, resonance self-shielding, medium energy cross-section values, Legendre Polynomial series truncation, quadrature and spatial representation. Of these, the treatment of resonance self-shielding was found to be most critical relative to shield penetration.

The shield geometry determined by the design calculations strongly reflects the anisotropy of the neutron source. The geometry satisfies the design criterion of a shield surface NDER equal to 0.5% of that in the collimated beam at $z=125$ cm., the patient total body NDER is far less than that considered acceptable by Parnell and Kelsey.³⁵⁾ The shield of this design might thus be considered to be too massive relative to the acceptable total body NDER, particularly if one considers that the background total body dose equivalent rate due to neutrons and charged particles scattered from the treatment field may reach 1% of that in the treatment field.¹¹⁹⁾ However, an additional benefit is realized from a massive shield - that of activation gamma-ray shielding.

In the course of the past few years of radiotherapy treatments, physicians and technicians have received considerable occupational doses from gamma-ray radiation emitted from the activated cast iron shield. With the collimator hole filled, the gamma-ray dose rate at $z=140$ cm. has reached a level as high as 0.02 roentgen per hour.¹¹⁹⁾ The massive shield of this design will contribute to the minimization of this problem.

The design of a shield-collimator unit to protect patients and personnel from gamma radiation due to shield activation requires

the accurate description of the various reaction cross sections (e.g., $(n,2n)$, (n,p) , etc.) throughout the energy range of the neutron spectrum. With group values of these cross sections one may determine, from the flux history, concentrations of radioactive nuclides throughout the shield geometry. These nuclide concentrations may then be transformed into gamma-ray group source rates, using nuclide decay data. Surface gamma-ray dose rates may then be determined from gamma-ray transport calculations.

At present, the IC+E model code CROIX is not now capable of accurately calculating individual reaction cross-section values. Improvements to the model are now planned and, if such improvements allow the model to accurately reproduce experimentally measured values, the above approach would be tractable.

REFERENCES

- 1) W.A.McFarlin and A.D.Suttle, Jr., IEEE Trans.Nucl. Sci.NS-18,3(1971)780
- 2) J.C.Larkin and R.S.Stone, Radiology 39(1942)608
- 3) R.S.Stone, Amer. J. Roent. 59(1948)771
- 4) P.C.Aebersold, Phys. Rev. 56(1939)714
- 5) J.F.Fowler, Radiology 108(1973)139
- 6) D.K.Bewley, Radiology 86(1966)251
- 7) D.L.Alger and R.Steinberg, IEEE Trans.Nucl. Sci. NS-20,1(1973)420
- 8) J.C.Davis, Trans. Am. Nucl. Soc. 23(1976)2
- 9) C.R.Emigh, J.E.Brolley, Jr. and M.C.Cline, LASL proposal LA-5191-P(1973)
- 10) L.C.Northcliffe, et al., in USAEC report ORO-3398-45(1971)
- 11) J.P.Meulders, P.Leleux, P.C.Macq and C.Pirart, Phys. Med. Biol. 20(1975)235
- 12) C.J.Parnell, G.D.Oliver, Jr., P.R.Almond and J.B.Smathers, Phys. Med. Biol. 17(1972)429
- 13) L.C.Northcliffe, private communication
- 14) K.Geobel and A.J.Miller, Nucl. Inst. Meth. 96(1971)581
- 15) K.A.Weaver, J.D.Anderson, H.H.Barschall and J.C.Davis, Phys. Med. Biol. 18(1973)64
- 16) G.W.Schweimer, Nuclear Physics A100(1967)537
- 17) N.Hertel, M.S.Thesis, Texas A&M University(1975)
- 18) Emphatic and conflicting views and data on neutron radiotherapy, particularly during its revival in England during the early '60s, found a convenient, inexpensive and fluid forum in Nature via informal letters.
- 19) P.D.Lomer and D.Green, Nature 198(1963)260
- 20) R.J.Berry, R.Oliver and E.H.Porter, Nature 199(1963)923
- 21) P.D.Lomer and D.Green, Nature 199(1963)923
- 22) J.F.Fowler, Nature 199(1963)923
- 23) D.K.Bewley, Nature 200(1963)70
- 24) D.Green and R.L.Thomas, Br. J. Radiol. 41(1968)455

- 25) D.K.Bewley and C.J.Parnell, Br.J.Radiol.42(1969)281
- 26) R.B.Theus, in LASL conf.report LA-5180-C(1973)
- 27) P.R.Almond, Trans.Am.Nucl.Soc.17(1973)116
- 28) D.Young, private communication
- 29) R.Faverty, Time, Dec.20, 1976, p.58
- 30) J.T.Brennan, Rad.Clin.N.Amer.7(1969)365
- 31) W.A.McFarlin and D.J.Goertz, Jr., IEEE Trans.Nucl.Sci. NS-13,4(1966),401
- 32) National Cancer Institute Grant CA-12542-01
- 33) J.B.Smathers, G.D.Oliver, Jr., W.B.Wilson and E.B.Hrinitsky, Proc., Third Intn'l. Conf. on Medical Physics, Including Medical Engineering, Goteborg, Sweden(1972)
- 34) J.B.Smathers, W.B.Wilson, V.A.Otte, P.R.Almond and W.Grant, Trans.Am.Nucl.Soc.23(1976)634
- 35) C.J.Parnell and C.A.Kelsey, Health Phys.23(1972)576
- 36) Benelex is a product of the Masonite Corp.
- 37) WEP is a product of the Ashland Chemical Corp., which cautions users that WEP shielding properties may change with time because of water loss.
- 38) G.I.Bell and S.Glasstone, Nuclear Reactor Theory, Van Nostrand Reinhold Co., New York(1970)
- 39) R.G.Alsmiller, Jr., private communication
- 40) D.R.Harris, private communication
- 41) R.J.LaBauve, private communication
- 42) K.D.Lathrop, private communication
- 43) K.O'Brien, private communication
- 44) R.Santoro, private communication
- 45) C.R.Weisbin, P.D.Soran, R.E.MacFarlane, D.R.Harris, R.J.LaBauve, J.S.Hendricks, J.E.White and R.B.Kidman, LASL report LA-6486-MS (1976)
- 46) R.E.MacFarlane and R.M.Boicourt, Trans.Am.Nucl.Soc.22(1975)720
- 47) R.E.Schenter, J.L.Baker and R.B.Kidman, BNWL report BNWL-1002(1969)
- 48) O.Ozer, Trans.Am.Nucl.Soc.17(1973)483
- 49) M.K.Drake, BNL report BNL-50274(1970)
- 50) R.B.Kidman and R.E.MacFarlane, Trans.Am.Nucl.Soc.23(1976)506
- 51) C.R.Weisbin, R.W.Roussin, J.E.White and R.Q.White, ORNL report ORNL-TM-5142(ENDF-224)(1975)

- 52) R.W.Roussin, C.R.Weisbin, J.E.White, N.M.Greene and R.H.Johnson, Trans.Am.Nucl.Soc.22(1975)820
- 53) R.J.LaBauve, R.E.Seamon and N.L.Whittemore, LASL report LA-5043-MS(1972)
- 54) H.A.Sandmeier, G.E.Hansen, R.E.Seamon, T.J.Hirons and A.H.Marshall, LASL report LA-5137(1974)
- 55) A variety of data libraries and computer codes are available from the Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
- 56) L.L.Smith and P.D.Soran, Trans.Am.Nucl.Soc.23(1976)528
- 57) M.G.Stamatelatos R.J.LaBauve and M.J.Becker, Trans.Am.Nucl.Soc.23(1976)529
- 58) C.R.Weisbin, J.S.Hendricks, D.E.Cullen, E.M.Oblow and P.D.Soran, Trans.Am.Nucl.Soc.17(1973)483
- 59) P.D.Soran, R.J.LaBauve, C.R.Weisbin and J.S.Hendricks, Trans.Am.Nucl.Soc.17(1973)484
- 60) D.E.Cullen and C.R.Weisbin, Trans.Am.Nucl.Soc.17(1973)488
- 61) R.J.LaBauve, W.B.Wilson and T.R.England, LASL report LA-6472-PR(1976)p.47
- 62) C.R.Weisbin and R.J.LaBauve, LASL report LA-5277-MS(1973)
- 63) R.J.LaBauve and W.B.Wilson, LASL report LA-6240-P(1976)
- 64) E.Oblow, J.Ching and J.Drischler, Trans.Am.Nucl.Soc.17(1973)547
- 65) D.R.Harris, et.al., Trans.Am.Nucl.Soc.15(1972)962
- 66) R.G.Alsmitter, Jr., et al., unpublished
- 67) R.G.Fluharty, P.A.Seeger, D.R.Harris, J.J.Koelling and O.L.Deutch, LASL preprint LA-UR-73-200(1973)
- 68) R.W.Roussin, R.G.Alsmitter, Jr. and J.Barish, ORNL report ORNL-TM-3659(1971)
- 69) R.G.Alsmitter, Jr., F.R.Mynatt, J.Barish and W.W.Engle, Jr., Nucl.Instr.Meth.72(1969)213
- 70) R.G.Alsmitter, Jr., F.R.Mynatt, J.Barish and W.W.Engle, Jr., Nucl.Sci.Eng.36(1969)251
- 71) R.G.Alsmitter, Jr. and J.Barish, ORNL report ORNL-TM-4060(1973)
- 72) G.I.Bell and S.Glasstone, Nuclear Reactor Theory, VanNostrand Reinhold Co., New York(1970)
- 73) K.D.Lathrop and B.G.Carison, LASL report LA-3186(1965)
- 74) B.G.Carison and K.D.Lathrop, LASL report LA-3251(1965)
- 75) The expansion of transfer cross sections in Legendre polynomials is common to most discrete ordinate transport codes.

However, the forms of the expansion of (3) and the resulting coefficients of (4) are not universally employed. In particular, discrete ordinate transport codes developed at the Oak Ridge National Laboratory do not accept cross section coefficients described by (4).

- 76) K.D.Lathrop, LASL report LA-3373(1965)
- 77) K.D.Lathrop and F.W.Brinkley, LASL report LA-4848-MS(1973)
- 78) K.D.Lathrop, Nucl.Sci.Eng.21(1965)498
- 79) K.D.Lathrop and A.Leonard, Nucl.Sci.Eng.22(1965)115
- 80) D.R.Harris, Trans.Am.Nucl.Soc.15(1972)958
- 81) D.R.Harris and W.B.Wilson, unpublished codes for the CDC-7600
- 82) M.Klein, C.R.Weisbin, R.J.LaBauve and O.Ozer, Trans.Am.Nucl.Soc.17(1973)489
- 83) D.E.Barine, F.R.Mynatt, R.W.Roussin, J.R.Knight and R.Q.Wright, Trans.Am.Nucl.Soc.19(1974)472
- 84) J.E.White, R.Q.Wright, L.R.Williams and C.R.Weisbin, Trans.Am.Nucl.Soc.23(1976)507
- 85) Reference 72), p.181
- 86) D.E.Bartine, F.R.Mynatt and E.M.Oblow, ORNL report ORNL-TM-4110(1973)
- 87) Reference 72), p.254
- 88) D.R.Harris, W.A.Reupke and W.B.Wilson, LASL report LA-5987(1975)
- 89) V.Benzi, F.Fabbri and G.Reffo, JAERI report JAERI-M-5984(1975)
- 90) H.A.Bethe, Phys.Rev.47(1935)747
- 91) F.Bjorklund and S.Fernbach, Phys.Rev.109(1958)1295
- 92) E.H.Auerbach, BNL report BNL-6562(1962)
- 93) I.Ulehla, L.Gomolcak and Z.Pluchar, Optical Model Of The Atomic Nucleus, Publishing House of the Czechoslovak Academy of Science, Prague(1964)
- 94) L.Wolfenstein, Ann.Rev.Nucl.Sci.6(1956)43
- 95) E.H.S.Burhop, ed., High Energy Physics, Vol.1, Academic Press, New York(1967), p.21
- 96) G.Breit, Rev.Mod.Phys.39(1967)560
- 97) R.E.Seamon, private communication
- 98) R.E.Seamon, K.A.Friedman, G.Breit, R.D.Haracz, J.M.Holt and A.Prakash, Phys.Rev.165(1968)1579
- 99) J.C.Hopkins, private communication to L.Stewart

- 100) V.A.Konshin and E.S.Matusevich, Atomic Energy Rev.6(1968)3
- 101) D.G.Foster, Jr., unpublished code for the CDC-7600
- 102) H.W.Bertini, M.P.Guthrie and O.W.Hermann, ORNL report ORNL-4564(1971)
- 103) L.Dresner, ORNL report ORNL-TM-196(1962)
- 104) M.P.Guthrie, ORNL report ORNL-4379(1969)
- 105) M.P.Guthrie, ORNL report ORNL-TM-3119(1970)
- 106) K.Chen, Z.Fraenkel, G.Friedlander, J.R.Grover, J.M.Miller and Y.Shimamoto, Phys.Rev.116(1968)949
- 107) R.Hofstadter, Ann.Rev.Nucl.Sci.7(1957)295
- 108) V.S.Barashenkov, H.W.Bertini, K.Chen, G.Friedlander, G.D.Harp, A.S.Iljinov, J.M.Miller and V.D.Toneev, Nuclear Physics A187(1972)531
- 109) R.G.Alsmiller, Jr. and O.W.Hermann, Nucl.Sci.Eng.40(1970)254
- 110) H.W.Bertini, Phys.Rev.C5(1972)2118
- 111) The interpretation of neutron fluence-to-dose equivalent data is summarized by this author and others in LA-5655-PR(1974), p.17. The gamma-ray fluence-to-dose equivalent values used in this work were taken from H.C.Claiborne and D.K.Trubey, Nucl.Appl.Technol.8(1970)450.
- 112) H.H.Rossi, et.al., NCRP report No.38(1971)
- 113) G.R.Stevensen, M.Hofert, J.Neufeld, A.Rindi, J.T.Roulti and S.B.Pretre, IAEA report IAEA/SM-167121(1972)
- 114) W.B.Wilson, unpublished codes for the CDC-7600
- 115) N.Hertel, V.A.Otte, J.B.Smathers, A.R.Smith and P.R.Almond. Med.Phys.2(1975)152
- 116) L.Northcliffe, private communication
- 117) J.P.Meulders, P.Leleux, P.C.Macq and C.Pirart, Phys.Med. Biol.20(1975)235
- 118) G.W.Schweimer, Nuclear Physics A100(1967)537
- 119) J.B.Smathers, private communication
- 120) Shielding calculations for the Weapons Neutron Research Facility (WNR) at LASL were performed with DTF-V, using the 41-group library, and with the NMTC Monte Carlo code. See reference 67).
- 121) F.W.Brinkley and R.D.Odell, private communication.
- 122) P.B.Jones, The Optical Model in Nuclear and Particle Physics, Interscience Publishers, New York(1963)
- 123) T.P.Stuart, J.D.Anderson and C.Wong, Phys.Rev.125(1962)276

- 124) Multigroup cross sections incorporate the neutron yield in the group-to-group scattering terms. Medium energy neutron reactions may yield, on the average, more than two secondary neutrons. For example, a CROIX calculation for 24,400 interactions of 60 MeV neutrons with ^{56}Fe produced 18,956 cascade neutrons and 38,735 evaporation neutrons; the yield, then, is $Y(60)=2.36\dots$
- 125) I.I. Bondarenko, (ed.), Group Constants for Nuclear Reactor Calculations, Consultants Bureau, New York (1964). See also ⁴⁵, p. 11
- 126) R.E. MacFarlane and M. Becker, Trans. Am. Nucl. Soc. 22(1975)668
- 127) Reference S₆ of Appendix A lists measured values of the total cross section and deduced nonelastic cross-section values, which are incorrectly referred to as inelastic cross-section values.
- 128) A.M. Baldin, V.I. Goldanskii and I.L. Rozental, Kinematics of Nuclear Reactions, Oxford University Press, Amen House, London (1961)
- 129) H. Arzelies, Relativistic Point Dynamics, Pergamon Press Ltd., Headington Hill Hall, Oxford, England (1972)
- 130) J.D. Jackson, Classical Electrodynamics, John Wiley and Sons, Inc., New York (1965)
- 131) R.D. Evans, The Atomic Nucleus, McGraw-Hill Book Co., Inc., N.Y. (1955)
- 132) M.E. Battat, private communication

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APPENDIX A MEDIUM ENERGY CROSS SECTIONS

NUCLIDE	NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNS)	X-SECTION ERROR (BARNS)	REACTION	REF
*****	*****	+--+--	*****	+--+--+--+	*****	***
H	14.020	.070	.7200000	.0100000	TOTAL	B 2
H	14.100		.6900000	.0120000	TOTAL	C 2
H	14.100	.050	.6890000	.0050000	TOTAL	P 1
H	14.700		.0260000	.0030000	(N,2N)	B12
H	15.000		.6620000	.0100000	TOTAL	C 2
H	15.500		.6480000	.0120000	TOTAL	C 2
H	15.800	.330	.5370000	.0300000	TOTAL	B 1
H	16.000		.5820000	.0300000	TOTAL	C 2
H	16.130	.330	.6340000	.0340000	TOTAL	B 1
H	16.460	.340	.5800000	.0330000	TOTAL	B 1
H	16.500		.6090000	.0150000	TOTAL	C 2
H	16.800	.350	.5970000	.0320000	TOTAL	B 1
H	17.000		.6120000	.0300000	TOTAL	C 2
H	17.150	.360	.5610000	.0290000	TOTAL	B 1
H	17.500		.5870000	.0200000	TOTAL	C 2
H	17.520	.370	.5440000	.0280000	TOTAL	B 1
H	17.800	.180	.5600000	.0090000	TOTAL	P 2
H	17.900	.390	.5330000	.0270000	TOTAL	B 1
H	18.000		.5390000	.0150000	TOTAL	C 2
H	18.290	.400	.5170000	.0270000	TOTAL	B 1
H	18.690	.420	.5230000	.0280000	TOTAL	B 1
H	19.110	.430	.4980000	.0400000	TOTAL	B 1
H	19.550	.440	.5220000	.0260000	TOTAL	B 1
H	19.665	.026	.4940000	.0025000	TOTAL	D 2
H	19.930	.025	.5040000	.0100000	TOTAL	D 1
H	20.000	.460	.4790000	.0240000	TOTAL	B 1
H	20.460	.470	.4790000	.0220000	TOTAL	B 1
H	20.600	.210	.4700000	.0100000	TOTAL	P 2
H	20.930	.480	.4490000	.0210000	TOTAL	B 1
H	21.410	.500	.4470000	.0200000	TOTAL	B 1
H	21.910	.510	.4200000	.0200000	TOTAL	B 1
H	22.430	.520	.4080000	.0200000	TOTAL	B 1
H	22.960	.540	.4330000	.0190000	TOTAL	B 1
H	23.510	.570	.3997000	.0160000	TOTAL	B 1
H	24.090	.590	.3930000	.0150000	TOTAL	B 1
H	24.690	.610	.3935000	.0140000	TOTAL	B 1
H	25.300	.250	.3880000	.0090000	TOTAL	P 2
H	25.310	.630	.3629000	.0135000	TOTAL	B 1
H	25.940	.650	.3776000	.0127000	TOTAL	B 1
H	26.600	.680	.3457000	.0123000	TOTAL	P 1
H	27.290	.720	.3354000	.0115000	TOTAL	B 1
H	28.030	.760	.3215000	.0110000	TOTAL	P 1

NUCLIDE	NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNS)	X-SECTION ERROR (BARNS)	REACTION	REF
*****	*****	+-+--+-	*****	+-+--+-+--+	*****	***
H	28.300	.280	.3400000	.0100000	TOTAL	P 2
H	28.800	.780	.3122000	.0105000	TOTAL	B 1
H	29.000	.290	.3310000	.0080000	TOTAL	P 2
H	29.590	.800	.3091000	.0095000	TOTAL	B 1
H	30.400	.830	.2817000	.0088000	TOTAL	B 1
H	31.240	.860	.2864000	.0083000	TOTAL	B 1
H	32.120	.900	.2887000	.0074000	TOTAL	B 1
H	33.050	.960	.2760000	.0068000	TOTAL	B 1
H	34.030	1.010	.2601000	.0072000	TOTAL	B 1
H	35.080	1.070	.2458000	.0067000	TOTAL	B 1
H	36.200	1.070	.2200000	.0099000	TOTAL	B 1
H	37.320	1.150	.2268000	.0063000	TOTAL	B 1
H	38.000	1.000	.2230000	.0076000	TOTAL	T 1
H	38.480	1.210	.2235000	.0060000	TOTAL	B 1
H	39.000	1.000	.2230000	.0076000	TOTAL	T 2
H	39.750	1.290	.2191000	.0057000	TOTAL	B 1
H	41.100	1.380	.2046000	.0054000	TOTAL	B 1
H	42.000		.2030000	.0070000	TOTAL	H 1
H	42.530	1.450	.1960000	.0052000	TOTAL	B 1
H	44.000	1.500	.1891000	.0049000	TOTAL	B 1
H	45.500	1.600	.1873000	.0046000	TOTAL	B 1
H	47.100	1.700	.1703000	.0040000	TOTAL	B 1
H	48.800	1.800	.1668000	.0041000	TOTAL	B 1
H	50.600	1.900	.1520000	.0038000	TOTAL	B 1
H	52.500	2.000	.1520000	.0036000	TOTAL	B 1
H	54.500	2.100	.1448000	.0033000	TOTAL	B 1
H	56.600	2.200	.1415000	.0031000	TOTAL	B 1
H	58.800	2.300	.1358000	.0029000	TOTAL	B 1
H	61.100	2.400	.1243000	.0028000	TOTAL	B 1
H	63.000	1.600	.1260000	.0030000	TOTAL	T 1
H	63.500	2.500	.1168000	.0026000	TOTAL	B 1
H	64.500	1.600	.1260000	.0030000	TOTAL	T 2
H	66.100	2.700	.1149000	.0025000	TOTAL	B 1
H	93.400	.500	.0770000	.0050000	TOTAL	C 1
H	95.000		.0730000	.0015000	TOTAL	D 5
H	95.000	1.500	.0739000	.0030000	TOTAL	T 1
H	97.200	1.000	.0760000	.0030000	TOTAL	C 1
H	101.100	.500	.0800000	.0070000	TOTAL	C 1
H	106.800	.200	.0590000	.0160000	TOTAL	C 1
H	126.000	2.000	.0569000	.0018000	TOTAL	T 1
H	153.000	3.000	.0454000	.0012000	TOTAL	T 1
H	156.000	3.000	.0464000	.0012000	TOTAL	T 7
H	270.000		.0380000	.0015000	TOTAL	D 6
H	280.000		.0330000	.0030000	TOTAL	F 5
H	351.500	2.000	.0356000	.0007000	TOTAL	A 8
H	410.000		.0337000	.0013000	TOTAL	N 2
H- 1	14.400	.200	.0000316	.0000031	(N,GAMMA)	C 4

NUCLIDE		NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNS)	X-SECTION ERROR (BARNS)	REACTION	REF
*****	*****	+	-	+	-	*****	***
H-	1	27.200		.0320000		ELASTIC	B11
H-	1	34.500	7.500	.2650000		ELASTIC	B 3
H-	1	34.500	7.500	.0001260	.0000800	(N,GAMMA)	B 3
H-	1	40.000		.1770000	.0100000	ELASTIC	H 3
H-	1	40.000		.1540000	.0150000	ELASTIC	H 3
H-	1	47.000	7.000	.1700000		ELASTIC	B 3
H-	1	47.000	7.000	.0001070	.0000500	(N,GAMMA)	B 3
H-	1	50.000	10.000	.0001100	.0000600	(N,GAMMA)	B 3
H-	1	60.500	6.500	.1300000		ELASTIC	B 3
H-	1	60.500	6.500	.0000420	.0000900	(N,GAMMA)	B 3
H-	2	14.400	.200	.0000294	.0000058	(N,GAMMA)	C 4
H-	2	156.000	3.000	.0707000	.0020000	TOTAL	T 7
B		14.100		.9000000	.0500000	ELASTIC	T 3
B		14.500		1.3800000		TOTAL	F 1
B		15.000		1.4000000		TOTAL	F 1
B		15.300		1.3500000		TOTAL	F 1
B		42.000		.9850000	.0200000	TOTAL	H 1
B-	10	14.100		1.4500000		TOTAL	F 1
B-	10	14.100		1.4500000	.0200000	TOTAL	C 2
B-	10	14.120		1.4700000	.0300000	TOTAL	C 3
B-	10	14.900		1.5000000		TOTAL	F 1
B-	10	15.000		1.4800000	.0300000	TOTAL	C 2
B-	10	15.300		1.4700000		TOTAL	F 1
B-	10	15.500		1.4600000	.0200000	TOTAL	C 2
B-	10	16.000		1.4600000	.0200000	TOTAL	C 2
B-	10	16.500		1.4400000	.0200000	TOTAL	C 2
B-	10	17.000		1.4200000	.0200000	TOTAL	C 2
B-	10	17.500		1.4800000	.0400000	TOTAL	C 2
B-	10	18.000		1.4200000	.0300000	TOTAL	C 2
B-	11	14.100		1.4300000	.0200000	TOTAL	C 2
B-	11	14.100		.0350000	.0070000	(N,ALPHA)	S 1
B-	11	14.120	.040	1.4000000	.0300000	TOTAL	C 3
B-	11	14.700		.0033000	.0006000	(N,P)	K 1
B-	11	14.700		.0310000	.0060000	(N,ALPHA)	A 3
B-	11	15.000		1.4200000	.0300000	TOTAL	C 2
B-	11	15.500		1.4400000	.0200000	TOTAL	C 2
B-	11	16.000		1.4500000	.0200000	TOTAL	C 2
B-	11	16.500		1.4100000	.0300000	TOTAL	C 2
B-	11	17.000		1.4100000	.0300000	TOTAL	C 2
B-	11	17.500		1.3400000	.0300000	TOTAL	C 2
B-	11	18.000		1.3000000	.0300000	TOTAL	C 2
C		14.010	.070	1.3100000	.0100000	TOTAL	B 2
C		14.060	.300	1.2700000	.0200000	TOTAL	M 2
C		14.100	.050	1.2790000	.0040000	TOTAL	P 1
C		14.100		1.2900000	.0200000	TOTAL	C 2
C		14.100		.7750000	.0400000	ELASTIC	T 3
C		14.120	.040	1.3200000	.0200000	TOTAL	C 3

NUCLIDE	NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNS)	X-SECTION ERROR (BARNS)	REACTION	REF
*****	*****	+-+--	*****	+-+--+-+	*****	***
C	14.150	.030	1.2700000	.0200000	TOTAL	M 2
C	14.250	.030	1.3000000	.0200000	TOTAL	M 2
C	14.340	.030	1.3100000	.0200000	TOTAL	M 2
C	14.370	.070	1.3100000	.0200000	TOTAL	B 2
C	14.700		1.5000000		TOTAL	F 1
C	15.000		1.4200000		TOTAL	F 1
C	15.000		1.3500000	.0200000	TOTAL	C 2
C	15.300		1.5000000		TOTAL	F 1
C	15.500		1.3800000	.0200000	TOTAL	C 2
C	15.800	.330	1.4600000	.0320000	TOTAL	B 1
C	15.900		1.5300000		TOTAL	F 1
C	16.000		1.4000000	.0200000	TOTAL	C 2
C	16.130	.330	1.4800000	.0320000	TOTAL	B 1
C	16.460	.340	1.4500000	.0320000	TOTAL	B 1
C	16.500		1.3700000	.0200000	TOTAL	C 2
C	16.800	.350	1.4500000	.0310000	TOTAL	B 1
C	17.000		1.3300000	.0200000	TOTAL	C 2
C	17.150	.360	1.4700000	.0310000	TOTAL	B 1
C	17.200	.025	1.3900000	.0400000	TOTAL	D 1
C	17.500		1.3300000	.0200000	TOTAL	C 2
C	17.520	.370	1.3900000	.0310000	TOTAL	B 1
C	17.800	.200	1.4200000	.0100000	TOTAL	P 2
C	17.900	.390	1.5200000	.0300000	TOTAL	B 1
C	18.000		1.3400000	.0200000	TOTAL	C 2
C	18.290	.400	1.4700000	.0300000	TOTAL	B 1
C	18.690	.420	1.4800000	.0300000	TOTAL	B 1
C	19.000	.025	1.4500000	.0400000	TOTAL	D 1
C	19.110	.430	1.4500000	.0290000	TOTAL	B 1
C	19.550	.440	1.4800000	.0450000	TOTAL	B 1
C	20.000	.460	1.4500000	.0280000	TOTAL	B 1
C	20.070	.025	1.5200000	.0300000	TOTAL	D 1
C	20.460	.470	1.5000000	.0280000	TOTAL	B 1
C	20.600	.300	1.4600000	.0100000	TOTAL	P 2
C	20.930	.480	1.4600000	.0270000	TOTAL	B 1
C	21.000		1.4460000	.0050000	TOTAL	B14
C	21.000		.4900000	.0400000	NONELASTIC	M 9
C	21.150		1.4560000	.0070000	TOTAL	B14
C	21.410	.500	1.4100000	.0250000	TOTAL	B 1
C	22.430	.520	1.3960000	.0230000	TOTAL	B 1
C	22.900		1.4070000	.0130000	TOTAL	B14
C	22.960	.540	1.4390000	.0220000	TOTAL	B 1
C	23.510	.570	1.3670000	.0220000	TOTAL	B 1
C	24.090	.590	1.3750000	.0210000	TOTAL	B 1
C	24.690	.610	1.3930000	.0210000	TOTAL	B 1
C	25.300	.300	1.3800000	.0100000	TOTAL	P 2
C	25.310	.630	1.3400000	.0200000	TOTAL	B 1
C	25.500		.4400000	.0400000	NONELASTIC	M 9

NUCLIDE	NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNs)	X-SECTION ERROR (BARNs)	REACTION	REF
*****	*****	+-+--	*****	+-+--	*****	***
C	25.940	.650	1.3350000	.0200000	TOTAL	B 1
C	26.600	.680	1.3490000	.0190000	TOTAL	B 1
C	26.800		1.3810000	.0070000	TOTAL	B14
C	27.290	.720	1.3150000	.0180000	TOTAL	B 1
C	27.550		1.3690000	.0030000	TOTAL	B14
C	28.030	.760	1.2980000	.0180000	TOTAL	B 1
C	28.150		1.3480000	.0100000	TOTAL	B14
C	28.300	.300	1.3200000	.0200000	TOTAL	P 2
C	28.800	.780	1.2910000	.0170000	TOTAL	B 1
C	29.100		1.3300000	.0130000	TOTAL	B14
C	29.100	.300	1.3200000	.0100000	TOTAL	P 2
C	29.200		.4500000	.0400000	NONELASTIC	M 9
C	29.590	.800	1.2580000	.0160000	TOTAL	B 1
C	30.400	.830	1.2500000	.0160000	TOTAL	B 1
C	31.240	.860	1.2370000	.0150000	TOTAL	B 1
C	32.120	.900	1.2200000	.0150000	TOTAL	B 1
C	33.050	.960	1.2310000	.0200000	TOTAL	B 1
C	34.030	1.010	1.2030000	.0200000	TOTAL	B 1
C	35.080	1.070	1.1800000	.0140000	TOTAL	B 1
C	35.100		1.2450000	.0050000	TOTAL	B14
C	35.200		1.2470000	.0080000	TOTAL	B14
C	36.200	1.120	1.1910000	.0130000	TOTAL	B 1
C	36.300		1.2430000	.0050000	TOTAL	B14
C	37.000	1.000	1.1500000	.0200000	TOTAL	T 1
C	37.320	1.150	1.1430000	.0130000	TOTAL	B 1
C	38.000	1.000	1.1000000	.0200000	TOTAL	T 1
C	38.480	1.210	1.1230000	.0120000	TOTAL	B 1
C	38.600		1.1630000	.0120000	TOTAL	B14
C	39.000	1.000	1.1000000	.0200000	TOTAL	T 2
C	39.100	.800	.8800000	.1300000	TOTAL	R 1
C	39.600	3.100	1.0600000	.0710000	TOTAL	R 1
C	39.750	1.290	1.0960000	.0120000	TOTAL	B 1
C	40.000		.9300000		TOTAL	H 3
C	40.150		1.1620000	.0030000	TOTAL	B14
C	40.800	.900	1.0230000	.1240000	TOTAL	R 1
C	41.100	1.380	1.0810000	.0110000	TOTAL	B 1
C	41.200		1.1440000	.0080000	TOTAL	B14
C	42.000		1.0890000	.0110000	TOTAL	H 1
C	42.000		1.1120000	.0090000	TOTAL	B14
C	42.530	1.450	1.0680000	.0110000	TOTAL	B 1
C	42.600	.900	1.1420000	.1860000	TOTAL	R 1
C	44.000	1.500	1.0520000	.0110000	TOTAL	B 1
C	44.000		1.1110000	.0070000	TOTAL	B14
C	44.000		1.0870000	.0030000	TOTAL	B14
C	44.150		1.0630000	.0080000	TOTAL	B14
C	44.400	1.000	.9030000	.0890000	TOTAL	R 1
C	45.000	3.800	.9430000	.0460000	TOTAL	L 3

NUCLIDE	NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNs)	X-SECTION ERROR (BARNs)	REACTION	REF
*****	*****	+--+--	*****	+--+--	*****	***
C	45.500	1.600	1.0120000	.0100000	TOTAL	B 1
C	46.400	1.100	.7480000	.0540000	TOTAL	R 1
C	47.100	1.700	.9670000	.0090000	TOTAL	B 1
C	48.600	1.200	.8580000	.0520000	TOTAL	R 1
C	48.800	1.800	.9690000	.0090000	TOTAL	B 1
C	49.500	4.400	.7710000	.0210000	TOTAL	R 1
C	50.600	1.900	.9300000	.0090000	TOTAL	B 1
C	50.900	1.200	.8280000	.0530000	TOTAL	R 1
C	52.000	1.500	.8920000	.0220000	TOTAL	T 1
C	52.500	2.000	.9120000	.0090000	TOTAL	B 1
C	53.000	3.000	.8250000	.0350000	TOTAL	L 3
C	53.400	1.300	.7170000	.0310000	TOTAL	R 1
C	54.500	2.100	.8780000	.0090000	TOTAL	B 1
C	55.000		.2760000		INELASTIC	V 1
C	56.000	1.400	.7170000	.0310000	TOTAL	R 1
C	56.600	2.200	.8440000	.0080000	TOTAL	B 1
C	58.800	2.300	.8080000	.0080000	TOTAL	B 1
C	58.900	1.500	.7240000	.0280000	TOTAL	R 1
C	59.300	3.600	.7040000	.0160000	TOTAL	R 1
C	60.000	1.500	.7890000	.0120000	TOTAL	T 1
C	61.000	.500	.6740000	.0620000	TOTAL	C 1
C	61.100	2.400	.7820000	.0070000	TOTAL	B 1
C	62.000	3.800	.7420000	.0220000	TOTAL	L 3
C	62.100	1.700	.7080000	.0230000	TOTAL	R 1
C	63.000	1.600	.7840000	.0050000	TOTAL	T 1
C	63.500	2.500	.7750000	.0070000	TOTAL	B 1
C	64.500	1.600	.7840000	.0050000	TOTAL	T 2
C	65.400	1.800	.7040000	.0240000	TOTAL	R 1
C	66.100	1.000	.6710000	.0420000	TOTAL	C 1
C	66.100	2.700	.7400000	.0070000	TOTAL	B 1
C	71.200	.500	.6010000	.0410000	TOTAL	C 1
C	76.700	.500	.6140000	.0310000	TOTAL	C 1
C	81.000	2.000	.2020000	.0210000	INELASTIC	V 1
C	81.200	1.000	.5850000	.0170000	TOTAL	C 1
C	84.000	1.500	.5710000	.0060000	TOTAL	T 1
C	85.500	.500	.6020000	.0190000	TOTAL	C 1
C	92.300	7.000	.5200000	.0800000	TOTAL	C 1
C	93.400	.500	.5180000	.0060000	TOTAL	C 1
C	95.000	1.500	.5080000	.0050000	TOTAL	T 1
C	95.000		.4980000	.0030000	TOTAL	D 5
C	97.200	1.000	.4940000	.0040000	TOTAL	C 1
C	99.400	4.000	.5180000	.0070000	TOTAL	C 1
C	101.100	.500	.4660000	.0070000	TOTAL	C 1
C	105.000	3.000	.2340000	.0080000	INELASTIC	V 1
C	106.800	2.000	.5080000	.0180000	TOTAL	C 1
C	117.000	2.000	.4080000	.0040000	TOTAL	T 1
C	126.000	2.000	.3900000	.0030000	TOTAL	T 1

NUCLIDE	NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNS)	X-SECTION ERROR (BARNS)	REACTION	REF
*****	*****	+--+--	*****	+--+--+--	*****	***
C	136.000	4.000	.1590000	.0100000	ELASTIC	V 2
C	139.000	3.000	.3550000	.0030000	TOTAL	T 1
C	140.000	4.000	.2210000	.0100000	INELASTIC	V 1
C	153.000	3.000	.3300000	.0030000	TOTAL	T 1
C	155.000	5.000	.3270000	.0020000	TOTAL	H 6
C	156.000	3.000	.3300000	.0030000	TOTAL	T 7
C	270.000		.2880000	.0030000	TOTAL	D 6
C	280.000		.2790000	.0040000	TOTAL	F 5
C	351.500	2.000	.2853000	.0016000	TOTAL	A 8
C	379.000		.2990000	.0080000	TOTAL	S 6
C	379.000		.2680000	.0450000	NONELASTIC	S 6
C	410.000		.2790000	.0030000	TOTAL	N 2
C	414.000		.2870000	.0080000	TOTAL	S 6
C	414.000		.2610000	.0380000	NONELASTIC	S 6
C	451.000		.3170000	.0080000	TOTAL	S 6
C	451.000		.2420000	.0320000	NONELASTIC	S 6
C	488.000		.3170000	.0080000	TOTAL	S 6
C	488.000		.2330000	.0280000	NONELASTIC	S 6
C	545.000		.3330000	.0060000	TOTAL	S 6
C	545.000		.2100000	.0180000	NONELASTIC	S 6
C	624.000		.3280000	.0050000	TOTAL	S 6
C	624.000		.2480000	.0150000	NONELASTIC	S 6
C	705.000		.3500000	.0060000	TOTAL	S 6
C	705.000		.2600000	.0130000	NONELASTIC	S 6
C	788.000		.3530000	.0060000	TOTAL	S 6
C	788.000		.2420000	.1100000	NONELASTIC	S 6
C	873.000		.3510000	.0070000	TOTAL	S 6
C	873.000		.2590000	.0100000	NONELASTIC	S 6
C	959.000		.3580000	.0070000	TOTAL	S 6
C	959.000		.2610000	.0100000	NONELASTIC	S 6
C- 12	15.310		.5500000	.0300000	NONELASTIC	H 4
O	14.000		1.5000000		TOTAL	F 1
O	14.100		1.6500000	.0500000	TOTAL	C 2
O	14.100	.050	1.6400000	.0400000	TOTAL	P 1
O	14.120	.040	1.5600000	.0400000	TOTAL	C 3
O	14.300		1.4200000		TOTAL	F 1
O	14.500		1.3700000		TOTAL	F 1
O	14.500		.8500000	.0300000	NONELASTIC	F 2
O	14.700		1.5000000		TOTAL	F 1
O	14.800		1.6000000		TOTAL	F 1
O	14.920		1.3500000		ELASTIC	B 4
O	15.000		1.6700000	.0500000	TOTAL	C 2
O	15.200		1.5000000		TOTAL	F 1
O	15.400		1.4000000		TOTAL	F 1
O	15.500		1.6300000	.0500000	TOTAL	C 2
O	15.500		1.5000000		TOTAL	F 1
O	15.800		1.5000000		TOTAL	F 1

NUCLIDE	NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNS)	X-SECTION ERROR (BARNS)	REACTION	REF
*****	*****	+-+--	*****	+-+--	*****	***
0	15.830		1.4400000		ELASTIC	B 4
0	15.850		1.4200000		TOTAL	F 1
0	16.000		1.5100000	.0700000	TOTAL	C 2
0	16.500		1.5900000	.0500000	TOTAL	C 2
0	17.000		1.5300000	.0600000	TOTAL	C 2
0	17.500		1.5900000	.0600000	TOTAL	C 2
0	18.000		1.6000000	.0500000	TOTAL	C 2
0	18.000	.200	1.6000000	.0200000	TOTAL	P 2
0	20.800	.200	1.6900000	.0300000	TOTAL	P 2
0	24.000	.200	1.6500000	.0300000	TOTAL	P 2
0	27.600	.300	1.6100000	.0200000	TOTAL	P 2
0	42.000		1.3580000	.0120000	TOTAL	H 1
0	93.400	.500	.7210000	.0130000	TOTAL	C 1
0	95.000		.6630000	.0070000	TOTAL	D 5
0	97.200	1.000	.6750000	.0090000	TOTAL	C 1
0	101.100	.500	.6490000	.0140000	TOTAL	C 1
0	106.800	2.000	.6680000	.0420000	TOTAL	C 1
0	156.000	3.000	.4300000	.0040000	TOTAL	T 7
0	270.000		.3720000	.0070000	TOTAL	D 6
0	280.000		.3800000	.0080000	TOTAL	F 5
0	351.500	2.000	.3660000	.0030000	TOTAL	A 8
0	410.000		.3780000	.0050000	TOTAL	N 2
0-	16 14.000	1.000	.0860000		(N,P)	M 4
0-	16 14.150		.2950000	.0350000	(N,ALPHA)	B 6
0-	16 14.220	.190	.0418200	.0023800	(N,P)	D 3
0-	16 14.300	.150	.0420000	.0030200	(N,P)	D 3
0-	16 14.500		.0490000	.0250000	(N,P)	P 3
0-	16 14.700	1.000	.0790000		(N,P)	M 4
0-	16 14.700		.0382000	.0050000	(N,P)	K 1
0-	16 14.760	.160	.0332900	.0024000	(N,P)	D 3
0-	16 14.800		.3300000	.0030000	(N,ALPHA)	B 6
0-	16 14.910		.0341000	.0018100	(N,P)	D 3
0-	16 15.220	.160	.0408500	.0029400	(N,P)	D 3
0-	16 15.400	1.000	.0670000		(N,P)	M 4
0-	16 15.440	.180	.0291200	.0015400	(N,P)	D 3
0-	16 15.610	.170	.0303100	.0022700	(N,P)	D 3
0-	16 16.000	.210	.0298400	.0015800	(N,P)	D 3
0-	16 16.000		.3600000	.0300000	(N,ALPHA)	B 6
0-	16 16.130	.170	.0304400	.0022200	(N,P)	D 3
0-	16 16.200		.0570000		(N,P)	M 4
0-	16 16.950		.3450000	.0300000	(N,ALPHA)	B 6
0-	16 17.000		.0500000		(N,P)	M 4
0-	16 17.080	.160	.0336100	.0019500	(N,P)	D 3
0-	16 17.210	.170	.0323200	.0023500	(N,P)	D 3
0-	16 17.500		.0460000		(N,P)	M 4
0-	16 17.610	.110	.0281400	.0020500	(N,P)	D 3
0-	16 17.700		.0001000		(N,2N)	B 5

NUCLIDE	NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNS)	X-SECTION ERROR (BARNS)	REACTION	REF
*****	*****	+-+--	*****	+-+--+-+--	*****	***
O- 16	17.740	.110	.0288500	.0016400	(N,P)	D 3
O- 16	17.800		.0420000		(N,P)	M 4
O- 16	18.000	.200	.0430000		(N,P)	M 4
O- 16	18.000		.2400000	.0280000	(N,ALPHA)	B 6
O- 16	18.280	.120	.0263800	.0014700	(N,P)	D 3
O- 16	18.790	.110	.0248400	.0018600	(N,P)	D 3
O- 16	19.040		.0259000	.0015000	(N,P)	D 3
O- 16	19.550		.2150000	.0260000	(N,ALPHA)	B 6
O- 16	19.700		.0020000	.0005000	(N,2N)	B 5
O- 16	21.300		.0030000	.0007000	(N,2N)	B 5
O- 16	22.000		.0058000	.0015000	(N,2N)	B 5
O- 16	25.000		.0118000	.0025000	(N,2N)	B 5
O- 16	27.600		.0149000	.0013000	(N,2N)	B 5
O- 16	30.400		.0158000	.0010000	(N,2N)	B 5
O- 16	34.000		.0187000	.0008500	(N,2N)	B 5
O- 16	37.500		.0095000	.0020000	(N,2N)	B 5
SI	14.000		1.9600000		TOTAL	T 5
SI	14.120	.040	1.8300000	.0400000	TOTAL	C 3
SI	14.120	.040	1.9000000	.1300000	TOTAL	C 3
SI	14.400		.1600000	.0160000	(N,P)	H 2
SI	14.500		1.0200000	.0600000	NONELASTIC	F 2
SI	15.000		1.8800000		TOTAL	T 5
SI	16.000		2.0200000		TOTAL	T 5
SI	17.000		1.8800000		TOTAL	T 5
SI	18.000	.180	1.8400000	.0200000	TOTAL	P 2
SI	19.000	.025	1.9400000	.0600000	TOTAL	D 1
SI	20.800	.200	1.9400000	.0300000	TOTAL	P 2
SI	24.000	.240	1.8900000	.0200000	TOTAL	P 2
SI	27.600	.300	2.0000000	.0300000	TOTAL	P 2
SI- 28	14.000		.4400000	.0500000	(N,P)	K 2
SI- 28	14.250		.3700000	.0300000	(N,P)	K 2
SI- 28	14.500		.2200000	.0230000	(N,P)	P 3
SI- 28	14.700		.3400000	.0400000	(N,P)	K 2
SI- 28	14.900	.200	.1810000	.0200000	(N,P)	J 1
SI- 28	15.000		.3000000	.0300000	(N,P)	K 2
SI- 28	15.400		.3000000	.0300000	(N,P)	K 2
SI- 28	15.700		.3150000	.0350000	(N,P)	K 2
SI- 28	16.000		.2480000	.0200000	(N,P)	K 2
SI- 28	16.200		.2300000	.0300000	(N,P)	K 2
SI- 28	16.300		.2900000	.0400000	(N,P)	K 2
SI- 28	16.500	.200	.1250000	.0140000	(N,P)	J 1
SI- 28	16.700		.2700000	.0300000	(N,P)	K 2
SI- 28	16.800		.1550000	.0200000	(N,P)	J 1
SI- 28	17.000		.2150000	.0200000	(N,P)	K 2
SI- 28	17.500		.1800000	.0350000	(N,P)	J 1
SI- 28	17.600	10.000	.2150000	.0200000	(N,P)	K 2
SI- 28	18.000		.1000000		(N,P)	C 5

NUCLIDE	NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNS)	X-SECTION ERROR (BARNS)	REACTION	REF
*****	*****	+-+--	*****	+-+--+-+	*****	***
SI- 28	18.150	8.000	.1600000	.0250000	(N,P)	K 2
SI- 28	18.150	.200	.1400000	.0160000	(N,P)	J 1
SI- 28	19.600	.200	.0750000	.0100000	(N,P)	J 1
SI- 28	20.600	.200	.0880000	.0100000	(N,P)	J 1
SI- 28	21.000	.200	.0790000	.0100000	(N,P)	J 1
SI- 29	14.500		.1010000	.0300000	(N,P)	P 3
SI- 30	14.000		.1800000	.0600000	(N,P)	P 4
SI- 30	14.500		.0459000	.0252000	(N,ALPHA)	P 3
P- 31	14.000		.1140000	.0140000	(N,P)	A 1
P- 31	14.000		.1630000	.0140000	(N,N)P	A 1
P- 31	14.100		.0910000	.0090000	(N,P)	F 4
P- 31	14.120		1.9700000	.0400000	TOTAL	C 3
P- 31	14.400		.1550000	.0160000	(N,P)	H 2
P- 31	14.400		.0700000	.0140000	(N,N)P	H 2
P- 31	14.400		.1170000	.0140000	(N,ALPHA)	G 1
P- 31	14.500		1.1300000	.0300000	NONELASTIC	F 2
P- 31	14.500		.1460000	.0290000	(N,ALPHA)	P 3
P- 31	14.500		.0642000	.0080000	(N,P)	P 3
P- 31	14.700		.0089000	.0012000	(N,2N)	K 1
P- 31	14.700		.1530000	.0200000	(N,ALPHA)	K 1
P- 31	14.700		.0820000	.0100000	(N,P)	K 1
P- 31	14.700		.0107000	.0009600	(N,2N)	B12
P- 31	14.740	.270	.0087000	.0027000	(N,2N)	F 3
P- 31	14.900		.1140000	.0170000	(N,ALPHA)	G 1
P- 31	15.200		.0290000	.0020000	(N,2N)	B 6
P- 31	15.500		.1100000	.0090000	(N,ALPHA)	G 1
P- 31	16.000		.0350000	.0035000	(N,2N)	B 6
P- 31	16.600		.1060000	.0130000	(N,ALPHA)	G 1
P- 31	16.960	.340	.0436000	.0052000	(N,2N)	F 3
P- 31	17.200		.0474000	.0040000	(N,2N)	B 6
P- 31	17.980	.240	.0740000	.0074000	(N,2N)	F 3
P- 31	18.800		.0580000	.0060000	(N,2N)	B 6
P- 31	19.700		.0610000	.0080000	(N,2N)	B 6
S	14.000		.1050000	.0250000	(N,N)P	A 1
S	14.000		.9100000	.0500000	ELASTIC	S 2
S	14.000		1.1000000		NONELASTIC	S 2
S	14.000		.8200000	.0500000	ELASTIC	T 3
S	14.100		2.0000000	.0500000	TOTAL	C 2
S	14.120	.040	1.9200000	.0400000	TOTAL	C 3
S	14.200		1.1400000	.0400000	NONELASTIC	M 5
S	14.400		.2060000	.0210000	(N,P)	H 2
S	14.400		.0730000	.0150000	(N,N)P	H 2
S	14.500		1.1500000	.0300000	NONELASTIC	F 2
S	16.000		1.9000000	.0500000	TOTAL	C 2
S	18.000		1.8100000	.0500000	TOTAL	C 2
S	28.400		2.1100000	.0700000	TOTAL	D 4
S	42.000		1.9740000	.0300000	TOTAL	H 1

NUCLIDE	NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNs)	X-SECTION ERROR (BARNs)	REACTION	REF
*****	*****	+-+--	*****	+-+--	*****	***
S- 32	14.100	.040	.2480000	.0130000	(N,P)	A 2
S- 32	14.240	.080	.2350000	.0050000	(N,P)	S 5
S- 32	14.500		.3690000	.0440000	(N,P)	P 3
S- 32	14.500	.060	.2260000		(N,P)	S 5
S- 32	14.530	.260	.2250000	.0120000	(N,P)	A 2
S- 32	14.600	.400	.0000072	.0000016	(N,T)	Q 1
S- 32	14.600		.0000040	.0000010	(N,T)	B 7
S- 32	14.680	.060	.2190000	.0050000	(N,P)	S 5
S- 32	14.700		.0000200	.0000050	(N,T)	W 1
S- 32	14.700	.290	.2200000	.0110000	(N,P)	A 2
S- 32	14.760	.060	.2160000	.0050000	(N,P)	S 5
S- 32	14.800		.1090000	.0160000	(N,ALPHA)	K 3
S- 32	14.830	.400	.2140000	.0110000	(N,P)	A 2
S- 32	14.930	.450	.2130000	.0110000	(N,P)	A 2
S- 32	14.950	.470	.2140000	.0110000	(N,P)	A 2
S- 32	16.600	.600	.1300000	.0060000	(N,P)	S 5
S- 32	17.500	1.000	.1230000	.0060000	(N,P)	S 5
S- 32	18.500	.500	.0950000	.0040000	(N,P)	S 5
S- 32	19.200	.800	.0880000	.0070000	(N,P)	S 5
S- 32	19.800	.300	.0870000	.0050000	(N,P)	S 5
S- 32	20.300	.300	.0680000	.0050000	(N,P)	S 5
S- 34	14.100		.1260000	.0070000	(N,ALPHA)	A 2
S- 34	14.500		.0852000	.0380000	(N,P)	P 3
S- 34	14.500		.1380000	.0350000	(N,ALPHA)	P 3
MN- 55	14.000		.1100000	.0150000	(N,P)	A 5
MN- 55	14.000	.300	.0326000	.0017000	(N,ALPHA)	B 9
MN- 55	14.000		.0570000	.0070000	(N,ALPHA)	K 4
MN- 55	14.000		.0450000	.0070000	(N,P)	A 1
MN- 55	14.100		.1075000	.0620000	(N,P)	L 1
MN- 55	14.120	.040	2.5400000	.0500000	TOTAL	C 3
MN- 55	14.300	.100	.0310000	.0060000	(N,ALPHA)	G 1
MN- 55	14.500		.0525000	.0080000	(N,ALPHA)	P 3
MN- 55	14.500		.7600000	.0800000	(N,GAMMA)	P 5
MN- 55	14.650		.0330000	.0040000	(N,ALPHA)	G 1
MN- 55	14.700		.8370000	.0570000	(N,2N)	B12
MN- 55	14.800		.0394000	.0080000	(N,ALPHA)	K 3
MN- 55	14.850		.0340000	.0040000	(N,ALPHA)	G 1
MN- 55	15.000	.300	.0354000	.0026000	(N,ALPHA)	B 9
MN- 55	15.300		.0310000	.0060000	(N,ALPHA)	G 1
MN- 55	15.800		.0344000	.0025000	(N,ALPHA)	B 9
MN- 55	15.900		.0270000	.0050000	(N,ALPHA)	G 1
MN- 55	16.250		.0230000	.0030000	(N,ALPHA)	G 1
MN- 55	16.500		.0300000	.0040000	(N,ALPHA)	G 1
MN- 55	16.700	.300	.0337000	.0024000	(N,ALPHA)	B 9
MN- 55	16.750		.0240000	.0040000	(N,ALPHA)	G 1
MN- 55	17.250		.0220000	.0040000	(N,ALPHA)	G 1
MN- 55	17.500	.300	.0291000	.0021000	(N,ALPHA)	B 9

NUCLIDE	NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNS)	X-SECTION ERROR (BARNS)	REACTION	REF
*****	*****	+-+--	*****	+-+--+-+--	*****	***
MN- 55	17.700		.0110000	.0020000	(N,ALPHA)	G 1
MN- 55	18.100	.300	.0278000	.0020000	(N,ALPHA)	B 9
MN- 55	18.800	.200	.0247000	.0018000	(N,ALPHA)	B 9
MN- 55	18.880	.100	.0227000	.0016000	(N,ALPHA)	B 9
MN- 55	28.400		2.3600000	.1200000	TOTAL	D 4
FE	14.000	.030	2.4700000	.0500000	TOTAL	M 2
FE	14.100	.030	2.5300000	.0500000	TOTAL	M 2
FE	14.100		.5000000	.0400000	(N,2N)	A 4
FE	14.120	.040	2.6000000	.0500000	TOTAL	C 3
FE	14.190	.030	2.5400000	.0500000	TOTAL	M 2
FE	14.280	.030	2.5400000	.0500000	TOTAL	M 2
FE	14.500		1.1400000	.6000000	TOTAL	C 9
FE	14.500	.070	2.5400000	.0300000	TOTAL	B 2
FE	14.700		.4740000	.0350000	(N,2N)	B12
FE	15.100		2.6000000		TOTAL	M 1
FE	15.600		2.6000000		TOTAL	M 1
FE	15.700	.700	.2860000	.0290000	(N,P)	C11
FE	16.400		2.5000000		TOTAL	M 1
FE	17.100		2.4500000		TOTAL	M 1
FE	17.300	.200	2.3500000	.0300000	TOTAL	P 2
FE	17.400	.700	.2570000	.0260000	(N,P)	C11
FE	17.700		2.4500000		TOTAL	M 1
FE	18.100		2.4500000		TOTAL	M 1
FE	19.000	.025	2.2300000	.0700000	TOTAL	D 1
FE	19.200		2.4000000		TOTAL	M 1
FE	19.950		2.3500000		TOTAL	M 1
FE	20.750		2.4100000		TOTAL	M 1
FE	21.400	.200	2.2300000	.0200000	TOTAL	P 2
FE	24.000	DIFFERENTIAL ELASTIC SCATTERING DATA				S 7
FE	25.000		2.2600000		TOTAL	U 1
FE	25.100	.250	2.2600000	.0300000	TOTAL	P 2
FE	25.500		1.2100000	.0700000	NONELASTIC	M 9
FE	27.800	.300	2.3000000	.0200000	TOTAL	P 2
FE	28.400		2.3900000	.0900000	TOTAL	D 4
FE	34.800	.700	1.4910000	.6800000	TOTAL	R 1
FE	36.100	.700	2.9480000	.2870000	TOTAL	R 1
FE	37.600	.800	2.7530000	.2340000	TOTAL	R 1
FE	38.500		2.5000000		TOTAL	U 1
FE	38.600	3.000	2.5030000	.0980000	TOTAL	R 1
FE	39.100	.800	2.3320000	.1990000	TOTAL	R 1
FE	40.800	.900	2.4770000	.1710000	TOTAL	R 1
FE	42.000		2.4410000	.0210000	TOTAL	H 1
FE	42.600	.900	2.4770000	.1710000	TOTAL	R 1
FE	44.400	1.000	2.3240000	.1620000	TOTAL	R 1
FE	44.500	4.400	2.2870000	.0900000	TOTAL	R 1
FE	44.700		2.3200000		TOTAL	U 1
FE	46.400	1.100	2.2360000	.1420000	TOTAL	R 1

NUCLIDE	NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNS)	X-SECTION ERROR (BARNS)	REACTION	REF
*****	*****	+-+--	*****	+-+--+-+--	*****	***
FE	48.600	1.200	2.3610000	.1080000	TOTAL	R 1
FE	50.900	1.200	2.4720000	.1230000	TOTAL	R 1
FE	51.100		2.3500000		TOTAL	U 1
FE	51.100	2.800	2.3380000	.0610000	TOTAL	R 1
FE	53.400	1.300	2.3490000	.0910000	TOTAL	R 1
FE	56.000	1.400	2.1890000	.0850000	TOTAL	R 1
FE	58.900	1.500	2.2330000	.0790000	TOTAL	R 1
FE	59.200		2.1900000		TOTAL	U 1
FE	59.300	3.600	2.1790000	.0440000	TOTAL	R 1
FE	62.100	1.700	2.2420000	.0690000	TOTAL	R 1
FE	65.400	1.800	2.2650000	.0710000	TOTAL	R 1
FE	70.100		2.1500000		TOTAL	U 1
FE	79.500		2.0200000		TOTAL	U 1
FE	90.000		1.7800000		TOTAL	U 1
FE	93.400	.500	1.9470000	.0480000	TOTAL	C 1
FE	97.200	1.000	1.8740000	.0370000	TOTAL	C 1
FE	101.100	.500	1.7350000	.0800000	TOTAL	C 1
FE	102.000		1.7200000		TOTAL	U 1
FE	106.800	2.000	1.9400000	.1850000	TOTAL	C 1
FE	156.000	3.000	1.2380000	.0160000	TOTAL	T 7
FE	379.000		1.0240000	.0290000	TOTAL	S 6
FE	379.000		.7570000	.0650000	NONELASTIC	S 6
FE	410.000		1.0730000	.0120000	TOTAL	N 2
FE	414.000		.9870000	.0290000	TOTAL	S 6
FE	414.000		.8140000	.0590000	NONELASTIC	S 6
FE	451.000		1.0350000	.0290000	TOTAL	S 6
FE	451.000		.8340000	.0530000	NONELASTIC	S 6
FE	488.000		1.0670000	.0300000	TOTAL	S 6
FE	488.000		.7720000	.0510000	NONELASTIC	S 6
FE	545.000		1.1070000	.0250000	TOTAL	S 6
FE	545.000		.8160000	.0350000	NONELASTIC	S 6
FE	624.000		1.0730000	.0270000	TOTAL	S 6
FE	624.000		.8560000	.0320000	NONELASTIC	S 6
FE	705.000		1.1260000	.0260000	TOTAL	S 6
FE	705.000		.8960000	.0290000	NONELASTIC	S 6
FE	788.000		1.1140000	.0270000	TOTAL	S 6
FE	788.000		.8180000	.0280000	NONELASTIC	S 6
FE	873.000		1.1420000	.0290000	TOTAL	S 6
FE	873.000		.8250000	.0280000	NONELASTIC	S 6
FE	959.000		1.1810000	.0310000	TOTAL	S 6
FE	959.000		.8840000	.0300000	NONELASTIC	S 6
FE- 54	14.000		.0100000	.0040000	(N,2N)	A 6
FE- 54	14.000		.4600000		(N,P)	A 6
FE- 54	14.000		.2200000		(N,N)P	A 6
FE- 54	14.000		.1900000		(N,P)	A 6
FE- 54	14.050		.0916000	.0371000	(N,ALPHA)	S 3
FE- 54	14.050		.3680000		(N,P)	S 3

NUCLIDE	NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNs)	X-SECTION ERROR (BARNs)	REACTION	REF
*****	*****	+-+--	*****	+-+--+-+	*****	***
FE- 54	14.050		.0028300	.0005000	(N,2N)	S 3
FE- 54	14.600	.400	.0001050	.0000350	(N,T)	Q 1
FE- 54	14.700		.0143000	.0070000	(N,2N)	B12
FE- 54	14.800		.2700000	.1350000	(N,ALPHA)	C 7
FE- 54	14.800		.0006000	.0001000	(N,T)	C 7
FE- 54	16.750		.0962000	.0157000	(N,ALPHA)	S 3
FE- 54	16.750		.2040000		(N,P)	S 3
FE- 54	16.750		.0504000	.0050000	(N,2N)	S 3
FE- 56	14.000		.1440000	.0190000	(N,P)	Y 1
FE- 56	14.000		.0350000	.0070000	(N,N)P	A 1
FE- 56	14.000		.4400000	.0900000	(N,2N)	W 2
FE- 56	14.100		.1240000	.0120000	(N,P)	F 4
FE- 56	14.280	.080	.1100000	.0040000	(N,P)	S 4
FE- 56	14.400		.1080000	.0100000	(N,P)	G 1
FE- 56	14.500	.060	.1100000	.0040000	(N,P)	S 4
FE- 56	14.500		.0967000	.0120000	(N,P)	P 3
FE- 56	14.680	.060	.1080000	.0040000	(N,P)	S 4
FE- 56	14.700		.4900000	.0360000	(N,2N)	B12
FE- 56	14.710	.440	.1050000	.0020000	(N,P)	T 4
FE- 56	14.760	.060	.1070000	.0040000	(N,P)	S 4
FE- 56	14.780	.100	.1098000	.0029000	(N,P)	R 2
FE- 56	15.480	.440	.0960000	.0020000	(N,P)	T 4
FE- 56	16.100	.600	.0824000	.0045000	(N,P)	S 4
FE- 56	16.230	.410	.0840000	.0020000	(N,P)	T 4
FE- 56	16.600	.600	.0762000	.0043000	(N,P)	S 4
FE- 56	16.890	.320	.0740000	.0020000	(N,P)	T 4
FE- 56	16.930	.470	.0695000	.0044000	(N,P)	L 2
FE- 56	17.270	.460	.0651000	.0041000	(N,P)	L 2
FE- 56	17.500	.500	.0674000	.0040000	(N,P)	S 4
FE- 56	17.500	.500	.0564000	.0031000	(N,P)	S 4
FE- 56	17.590	.440	.0610000	.0038000	(N,P)	L 2
FE- 56	17.890	.080	.0620000	.0020000	(N,P)	T 4
FE- 56	17.900	.420	.0584000	.0035000	(N,P)	L 2
FE- 56	18.190	.390	.0566000	.0034000	(N,P)	L 2
FE- 56	18.470	.360	.0547000	.0033000	(N,P)	L 2
FE- 56	18.710	.330	.0520000	.0032000	(N,P)	L 2
FE- 56	18.940	.290	.0510000	.0030000	(N,P)	L 2
FE- 56	19.100	.800	.0529000	.0030000	(N,P)	S 4
FE- 56	19.290	.200	.0492000	.0028000	(N,P)	L 2
FE- 56	19.580	.190	.0467000	.0026000	(N,P)	L 2
FE- 56	19.800	.300	.0483000	.0026000	(N,P)	S 4
FE- 56	20.300	.300	.0446000	.0025000	(N,P)	S 4
FE- 57	14.100		.0500000	.0080000	(N,P)	P 6
FE- 57	14.500		.0750000	.0080000	(N,P)	C10
FE- 57	14.800		.0061000	.0026000	(N,N)P	C 7
FE- 57	14.800		.0710000	.0070000	(N,P)	C 7
FE- 58	14.800		.0215000	.0020000	(N,ALPHA)	C 7

NUCLIDE	NEUTRON ENERGY (MEV)	ENERGY ERROR (MEV)	X-SECTION VALUE (BARNs)	X-SECTION ERROR (BARNs)	REACTION	REF
*****	*****	+--+--	*****	+--+--+--	*****	***
W	14.120	.040	5.3000000	.1100000	TOTAL	C 3
W	14.250	.070	5.4000000	.0600000	TOTAL	B 2
W	14.700		2.2400000	.1640000	(N,2N)	B12
W	17.300	.200	5.4200000	.0700000	TOTAL	P 2
W	21.300	.200	5.2400000	.0600000	TOTAL	P 2
W	25.200	.300	4.9600000	.0600000	TOTAL	P 2
W	28.800	.300	4.6100000	.0500000	TOTAL	P 2
W	42.000		4.3100000	.0600000	TOTAL	H 1
W	270.000		2.6100000	.0500000	TOTAL	D 6
W-183	14.700		2.2320000	.1000000	(N,2N)	B12
W-184	14.500		.0049000		(N,P)	M 3
W-184	14.500		.0047500	.0009500	(N,P)	C 8
W-184	14.800		.0140000	.0040000	(N,P)	P 7
W-186	14.500	.300	.0014000	.0003000	(N,P)	B10
W-186	14.500	.300	.0001100	.0000500	(N,N)P	B10
W-186	14.500		.0040000	.0005000	INELASTIC	P 5
W-186	14.500		.0008500		(N,ALPHA)	M 3
W-186	14.700		2.2900000	.2300000	(N,2N)	B12
W-186	14.800		.0025000	.0010000	(N,ALPHA)	P 7
W-186	14.850	.150	.0028000	.0005000	(N,P)	B10
W-186	14.850	.150	.0003000	.0001500	(N,N)P	B10
W-186	18.000	1.000	.0047000	.0015000	(N,N)P	B10
W-186	18.000	1.000	.0096000	.0020000	(N,P)	B10
W-186	19.700	1.300	.0145000	.0030000	(N,P)	B10
W-186	19.700	1.300	.0146000	.0030000	(N,N)P	B10
W-186	21.200	1.000	.0136000	.0027000	(N,P)	B10
W-186	21.200	1.000	.0171000	.0030000	(N,N)P	B10

REFERENCES

- A 1 D.L.ALLEN, NUCLEAR PHYSICS 24(1961)274
- A 2 L.ALLEN,JR.,W.A.BIGGERS,R.J.PRESTWOOD AND R.K.SMITH,
PHYS.REV.107(1957)1363
- A 3 A.H.ARMSTRONG AND G.M.FYRE,JR., PHYS.REV.103(1956)335
- A 4 V.J.ASHBY,H.C.CATRON,L.L.NEWKIRK AND C.J.TAYLOR,
PHYS.REV.111(1958)616
- A 5 P.AVIGNON AND L.ROSIER, COMPT.REND.247(1958)1849
- A 6 D.L.ALLEN, PROC.PHYS.SOC.70(1957)195
- A 7 B.ADAMS,R.BATCHELOR AND T.S.GREEN,
J.NUCLEAR ENERGY 14A/B(1961)85
- A 8 A.ASHMORE,R.G.JARVIS,D.S.MATHER AND S.K.SEN,
PROC.PHYS.SOC.(LONDON) A70(1957)745
- B 1 P.H.BOWEN,J.P.SCANLON,G.H.STAFFORD,J.J.THRESHER AND
P.E.HODGSON, NUCLEAR PHYSICS 22(1961)640

- B 2 A. BRATENAH, J. M. PETERSON AND J. P. STÖERING,
PHYS. REV. 110(1958)927
- B 3 N. C. BARFORD, R. B. PALMER AND B. TALLINI,
PROC. PHYS. SOC. 78(1961)912
- B 4 N. A. BOSTROM, I. L. MORGAN, J. T. PRUD'HOMME AND A. R. SATTAR,
WADC-TR-57-446(1957)
- B 5 O. D. BRILL, N. A. VLASOV, S. P. KALININ AND L. S. SOKOLOV,
SOVIET PHYSICS-DOKLADY 6(1961)24
- B 6 M. BORMANN, S. CIERJACKS, E. FRETWURST, K. -J. GIESECKE,
H. NEUERT AND H. POLLEHN, Z. PHYSIC 174(1963)1
- B 7 A. P. BAERG AND G. C. BOWES, CAN. J. CHEM. 39(1961)684
- B 8 E. A. BIVETSKY AND J. M. FRANK,
PROC. INTN'L CONF. ON NUCLEAR PHYSICS, PARIS(1964)817
- B 9 M. BORMANN, E. FRETWURST, P. SCHEHKA AND G. WREGE,
NUCLEAR PHYSICS 63(1965)438
- B10 J. F. BARRY, R. F. COLEMAN, B. E. HAWKER AND J. L. PERKIN,
PROC. PHYS. SOC. 74(1959)632
- B11 J. E. BROLLEY, JR., J. H. COON AND J. L. FOWLER,
PHYS. REV. 82(1951)190
- B12 Z. T. BODY AND J. CSIKAI, ATOMIC ENERGY REV. 11(1973)153
- B13 A. BRATENAH, S. FERNBACH, R. H. HILDEBRAND, C. E. LEITH AND
B. J. MOYER., PHYS. REV. 77(1950)597
- B14 I. F. BUBB, ET. AL., CAN. J. PHYS. 52(1974)648
- B15 W. P. BALL, UCRL-1938(1952)
- C 1 V. CULLER AND R. W. WANIEK, PHYS. REV. 99(1955)740
- C 2 C. F. COOK AND T. W. BONNER, PHYS. REV. 94(1954)651
- C 3 J. H. COON, E. R. GRAVES AND H. H. BARSCHALL
PHYS. REV. 88(1952)562
- C 4 M. CERINEO, K. ILAKOVAC, I. SLAUS AND P. TOMAS
PHYS. REV. 124(1961)1947
- C 5 A. V. COHEN AND P. H. WHITE, NUCLEAR PHYSICS 1(1956)73
- C 6 A. CHATTERJEE, NUCLEAR PHYSICS 47(1963)511
- C 7 D. M. CHITTENDEN II, D. G. GARDNER AND R. W. FINK,
PHYS. REV. 122(1961)860
- C 8 R. F. COLEMAN, B. E. HAWKER, L. P. O'CONNOR AND J. L. PERKIN,
PROC. PHYS. SOC. 73A(1959)215
- C 9 J. H. COON, R. W. DAVIS, H. E. FELTHAUSER AND B. D. NICODEMUS,
PHYS. REV. 111(1958)250
- C10 W. G. CROSS, AS SHOWN IN BNL-325
- C11 E. E. CARROLL, JR. AND G. G. SMITH, NUCL. SCI. ENG. 22(1965)411
- D 1 R. B. DAY AND R. L. HENKEL, PHYS. REV. 92(1953)358
- D 2 R. B. DAY, R. L. MILLS, J. E. PERRY, JR. AND F. SCHERB,
PHYS. REV. 114(1959)209
- D 3 J. A. DE JUREN AND R. W. STOOKSBERRY, PHYS. REV. 127(1962)1229
- D 4 G. DECONNINCK, A. GONZE, P. MACQ AND J. P. MEULDER,
J. PHYS. RADIUM 22(1961)652
- D 5 J. DE JUREN AND N. KNABLE, PHYS. REV. 77, (1950)606
- D 6 J. DE JUREN, PHYS. REV. 80(1950)20
- F 1 D. B. FOSSAN, R. L. WALTER, W. E. WILSON AND H. H. BARSCHALL,
PHYS. REV. 123(1961)209
- F 2 N. N. FLEROV AND V. M. TALYZIN, J. NUCLEAR ENERGY 4(1957)529

- F 3 J.M.FERGUSON AND W.E.THOMPSON, PHYS.REV.118(1960)228
 F 4 S.G.FORBES, PHYS.REV.88(1952)1309
 F 5 R.FOX,C.LEITH,L.WOUTERS AND K.R.MAC KENZIE,
 PHYS.REV.80(1950)23
 G 1 F.GABBARD AND B.D.KERN, PHYS.REV.128(1962)1276
 G 2 A.K.GANGULY AND H.BAKHRU,
 PROC.SYMP.ON NUCLEAR PHYSICS,BOMBAY(1963)
 H 1 R.H.HILDEBRAND AND C.E.LEITH, PHYS.REV.80(1950)842
 H 2 F.L.HASSLER AND R.A.PECK,JR., PHYS.REV.125(1962)1011
 H 3 J.HADLEY,E.KELLY,C.LEITH,E.SEGRE',C.WIEGAND AND H.YORK
 PHYS.REV.75(1949)351
 H 4 L.F.HANSEN,J.D.ANDERSON,M.L.STELTS AND C.WONG,
 PROC.CONF.ON NEUTRON CROSS SEC.AND TECH.(1968)225
 H 5 R.H.HILDEBRAND, UCRL-1159(1951)
 H 6 R.S.HARDING, PHYS.REV.111(1958)1164
 J 1 J.M.F.JERONYMO,G.S.MANI,J.OLKOWSKY,A.SADEGHI AND
 C.F.WILLIAMSON, NUCLEAR PHYSICS 47(1963)157
 K 1 J.KANTELE AND D.G.GARDNER, NUCLEAR PHYSICS 35(1962)353
 K 2 B.D.KERN,W.E.THOMPSON AND J.M.FERGUSON,
 NUCLEAR PHYSICS 10(1959)226
 K 3 I.KUMABE, J.PHYS.SOC.JAPAN 13(1958)325
 K 4 C.S.KHURANA AND H.S.HANS, PROC.SYMP.ON LOW ENERGY
 NUCLEAR PHYSICS,WALT AIR(1960)297
 K 5 O.N.KAUL, NUCLEAR PHYSICS 39(1962)325
 K 6 I.KUMABE,E.TAKEKOSHI,H.OGATA,Y.TSUNEOKA AND S.OKI,
 J.PHYS.SOC.JAPAN 13(1958)129
 L 1 J.LANGMANN, AS SHOWN IN BNL-325
 L 2 H.LISKIN AND A.PAULSEN, J.NUCLEAR ENERGY 19A/B(1965)73
 L 3 W.I.LINLOR, UCRL-2303(1953)
 M 1 G.J.MC CALLUM,G.S.MANI AND A.T.G.FERGUSON,
 NUCLEAR PHYSICS 16(1960)313
 M 2 F.MANERO,B.H.ARMITAGE AND J.M.FREEMAN,
 NUCLEAR PHYSICS 59(1964)583
 M 3 S.K.MUKHERJEE AND H.BHAKROO
 PROC.SYMP.LOW ENERGY NUCLEAR PHYSICS,BOMBAY(1963)
 M 4 H.C.MARTIN, PHYS.REV.93(1954)498 (AS ADJUSTED BY K.SURE
 IN WAPD-BT-25(1962)27
 M 5 W.H.MAC GREGOR,W.P.BALL AND R.BOOTH,
 PHYS.REV.108(1957)726
 M 6 M.H.MAC GREGOR, UCRL-5229(1958)
 M 7 P.W.MARTIN AND D.T.STEWART,J.NUCLEAR ENERGY 19(1965)447
 M 8 M.H.MAC GREGOR
 PROC.INTN L.CONG.NUCLEAR PHYSICS,PARIS(1958)609
 M 9 M.H.MAC GREGOR,W.P.BALL AND R.BOOTH
 PHYS.REV.111(1958)1155
 N 1 V.V.NEFEDOV,V.I.POPOV AND YJ.S.YAZVITSKII,
 S VIET PROGRESS IN NUCLEAR PHYSICS(1961)241
 N 2 V.A.NEDZEL, PHYS.REV.94(1954)194
 P 1 H.L.POSS,E.O.SALANT,G.A.SNOW AND LUKE C.L.YUAN,
 PHYS.REV.87(1952)11

- P 2 J.M.PETERSON,A.BRATENAHLE AND J.P.STOERING,
PHYS.REV.120(1960)521
- P 3 E.B.PAUL AND R.L.CLARKE, CAN.J.PHYS.31(1953)267
- P 4 E.PEETERS, PHYS.LETTERS 7(1963)142
- P 5 J.L.PERKIN,L.P.O'CONNOR AND R.F.COLEMAN,
PROC.PHYS.SOC.72(1958)505
- P 6 H.POLLEHN AND H.NEUERT, Z.NATURFORSCH 16A(1961)227
- P 7 A.POULARIKAS,R.W.FINK AND D.G.GARDNER,
ANN.PROG.RPT.NUC.CHEM,U.ARKANSAS(1960)
- P 8 V.M.PANKRATOV,N.A.VLASOV AND B.V.RYBAKOV,
J.NUCLEAR ENERGY 16A/B(1962)494
- Q 1 S.M.QAIM AND G.L.STOCKLIN, J.INORG.NUCL.CHEM.35(1973)19
- R 1 E.RAGENT, UCRL-2337(1953)
- R 2 J.C.ROBERTSON,B.N.AUDRIC AND P.TOLKOWSKI,
J.NUCLEAR ENERGY 27(1973)139
- S 1 M.SAKISAKE, J.PHYS.SOC.JAPAN 14(1959)554
- S 2 C.ST.PIERRE,M.K.MACHWE AND P.LORRAIN,
PHYS.REV.115(1959)999
- S 3 S.R.SALISBURY AND R.A.CHALMERS,PHYS.REV.140(1965)B305
- S 4 D.C.SANTRY AND J.P.BUTLER, CAN.J.PHYS.42(1964)1030
- S 5 D.C.SANTRY AND J.P.BUTLER, CAN.J.CHEM.41(1963)123
- S 6 W.SCHIMMERLING,T.J.DEVLIN,W.W.JOHNSON,K.G.VOSBURGH AND
R.E.MISCHKE, PHYS.REV.C7(1973)248
- S 7 T.P.STEWART,J.D.ANDERSON AND C.WONG,
PHYS.REV.125(1962)276
- T 1 A.E.TAYLOR AND E.WOOD,PHIL.MAG.44(1953)95
- T 2 A.E.TAYLOR,T.G.PICKAVANCE,J.M.CASSELS AND T.C.RANDLE,
PHIL.MAG.42(1951)328
- T 3 K.TESCH, NUCLEAR PHYSICS 37(1962)412
- T 4 J.TERRELL AND D.M.HOLM, PHYS.REV.109(1958)2031
- T 5 K.TSUKADA AND Y.C.HSU,AS SHOWN IN BNL-325
- T 6 A.E.TAYLOR,T.G.PICKAVANCE,J.M.CASSELS AND T.C.RANDLE,
PHIL.MAG.42(1951)20
- T 7 A.E.TAYLOR,T.G.PICKAVANCE,J.M.CASSELS AND T.C.RANDLE,
PHIL.MAG.42(1951)751
- U 1 UCRL UNPUBLISHED, AS SHOWN IN BNL-325
- V 1 R.G.P.VOSS AND R.WILSON, PROC.PHYS.SOC.A236(1957)41
- V 2 C.P.VAN ZYL,R.G.P.VOSS AND R.WILSON
PHIL.MAG.1(1956)1003
- W 1 E.WIEGOLD AND R.N.GLOVER,NUCLEAR PHYSICS 32(1962)106
- W 2 R.WENUSCH AND H.VONACH,
AKAD.WISS.MATH-NATURW.KL.,ANZ.1(1962)1
- W 3 P.H.WHITE, J.NUCLEAR ENERGY 16A/B(1962)261
- Y 1 S.YASUMI, J.PHYS.SOC.JAPAN 12(1957)443

APPENDIX B

RELATIVISTIC KINEMATICS

Medium energy (20. - 60. Mev) cross section calculations have been performed for hydrogen and iron by the MINX⁴⁵⁾ and NJOY⁴⁶⁾ cross-section processing codes. These programs employ non-relativistic kinematic relationships, and it is necessary to evaluate the errors associated with these calculations.

In this work only elastic scattering calculations, performed in the center-of-mass system, were conducted. The results of these calculations are transformed to the laboratory system, in which the transport calculations are conducted. It is thus necessary to investigate the scattering angles in each system as determined by the incident and scattered particle energies. These angular values are examined below. The threshold energy values corresponding to some nonelastic reactions are also computed.

If E and E' are the laboratory energies of the incident and scattered neutron, M is the neutron rest mass energy, R is the ratio of the neutron mass to the mass of the target, Q is the Q -value of the reaction, E_{th} is the reaction threshold energy and θ_L and θ_{CM} are the laboratory and center-of-mass system scattering energies, then the classical expressions describing neutron scattering are as follows:

$$\begin{aligned} E_{th} &= -Q (1+R) / R \quad , \quad Q < 0 \\ &= 0 \quad , \quad Q \geq 0 \end{aligned}$$

$$\cos(\theta_{CM}) = 1 - \frac{(E-E') / E}{2R / (1+R)^2}$$

$$\cos(\theta_L) = \frac{E+E'-R(E-E'+Q)}{2(E E')^{1/2}}$$

The relativistic expressions are:¹²⁸⁻¹³¹⁾

$$E_{th} = Q^2/2RM - Q(1+R)/R, \quad Q < 0$$

$$= 0, \quad Q > 0$$

$$\cos(\theta_{CM}) = \frac{E + M - T_1 \{E + (1+R)M\}/2}{T_2 T_3/2}$$

$$\tan(\theta_L) = \frac{\{(1+R)^2 M^2 + 2ERM\}^{1/2} \sin(\theta_{CM})}{\{E + (1+R)M\} \{\cos(\theta_{CM}) + T_1 T_2/T_3\} E + (1+R)M}$$

where

$$T_1 = 1 + \frac{(1-R^2)M^2 + 2RMQ - Q^2}{(1+R^2)M^2 + 2ERM}$$

$$T_2 = \{E(E+2M)\}^{1/2}$$

$$T_3 = 1 - \frac{\{(1+R)M-Q\}^2}{\{(1+R)^2 M^2 + 2ERM\}} \quad 1 - \frac{\{(1-R)M+Q\}^2}{\{(1+R)^2 M^2 + 2ERM\}}$$

Quantities calculated with the relativistic and nonrelativistic expressions are compared below.

CLASSICAL AND RELATIVISTIC REACTION VALUES

Nuclide	Reaction	E (MeV.)	E (MeV.)	Q (MeV.)	E _{th} (MeV.)	Classical		E _{th}	Relativistic	
						cos(θ _{CM})	cos(θ _L)		cos(θ _{cm})	cos(θ _L)
¹ H	elastic	20.	15.	0.	0.	0.500000	0.866145	0.	.500000	∴68420
	elastic	60.	30.	0.	0.	-1.72E-7	0.707400	0.	-1.67E-7	.712928
¹⁰ B	(n,γ)	-	-	-2.2246	4.4510	-	-	4.4537	-	-
	(n,n')	-	-	-0.717	1.10074	-	-	1.10076	-	-
	(n,n')	-	-	-18.0	19.8133	-	-	19.8296	-	-
¹² C	(n,n')	-	-	-4.433	4.8056	-	-	4.8065	-	-
	(n,n')	-	-	-7.274	7.8854	-	-	7.8878	-	-
¹⁴ N	(n,2n)	-	-	-10.553	11.3159	-	-	11.3232	-	-
	(n,n')	-	-	-2.313	2.4802	-	-	2.4804	-	-
	(n,n')	-	-	-18.25	19.5693	-	-	19.5821	-	-
¹⁶ O	(n,n')	-	-	-6.052	6.4336	-	-	6.4349	-	-
	(n,n')	-	-	-18.55	19.7198	-	-	19.7313	-	-