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SHIELDING COMPUTER PROGRAM 04-0, REACTOR SHIELD ANALYSIS

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ABSTRACT OF REPORT NO. XDC 58-7-76

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ABSTRACT: Shielding Computer Program 04-0 is available for computing reactor-shield weight and fast neutron and gamma ray dose rates in and around complex reactor-shield assemblies.

Reactor and shield geometries are described by combinations of regions formed by rotating rectangles or trapezoids about the reactor axis or by translating rectangles parallel to the reactor axis. Compositions are expressed as volume fractions for each material in the reactor-shield assembly.

A cylindrical coordinate system is used to describe the location and dimensions of a volume source. Source distributions are assumed separable and may be described by cosine functions.

A modification of the Albert-Welton theory of neutron attenuation is used for fast neutron dose rate calculations. Combination of buildup factors is used with exponential attenuation to compute gamma ray dose rates. The program requires four magnetic tape units and two frames of magnetic core memory on an IBM 704. Magnetic drum memory is not required.

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1. INTRODUCTION

Shielding Computer Program 04-0 computes reactor-shield weight and fast neutron and gamma ray dose rates in and around complex reactor shield assemblies. This program is coded for use on an IBM-704 having two frames of magnetic core memory. No magnetic drum memory is required. A total of four tape units are used by the program.

Reactor and shield geometries are described by combinations of regions formed by rotating rectangles or trapezoids about the reactor axis or by translating rectangles parallel to the reactor axis. Compositions are expressed as volume fractions for each material in the reactor-shield assembly. Compositions are associated with the appropriate geometrical regions by reference to code numbers.

Source location and dimensions are described in cylindrical coordinates. Source region nodal points are located in rings formed by intersection of concentric shells and planes normal to the axis. There may be two to fifteen equally spaced planes and shells. Source region nodal points are located in two to fifteen equally spaced lines in each shell. It is necessary that each shell contain the same number.

Source distributions along the axis and radius are assumed separable and are described by cosine functions. Point-point attenuation functions (point kernels) are used for all dose rate calculations. A modification of the point kernel suggested by Albert and Welton[1,2] in their theory of neutron attenuation in hydrogenous media is used for calculating direct beam fast neutron dose rates around fission sources. Effective removal cross sections are used for elements other than hydrogen.

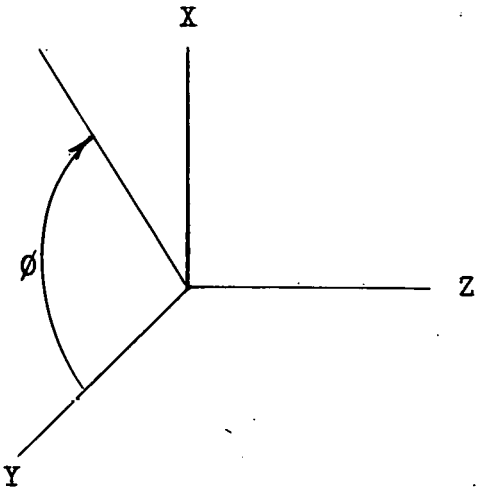
Gamma ray dose rates may be computed for positions around sources emitting photons of a maximum of twenty initial energies. An uncollided point kernel is used in conjunction with an expression providing several optional methods for combining buildup factors.

Integration over source regions is performed by the trapezoidal method.

2. PHYSICAL DESCRIPTION

2.1 Geometry Description

Reactor and shield geometries are described by combinations of regions in either a cylindrical coordinate system (r, ∅, Z) or a rectangular coordinate system (X, Y, Z). The Z-axis must be the reactor axis, and it may be positive in either direction. Orientation of the X- and Y- axes and the origin and direction of measurement of ∅ are shown in the following sketch:



Regions may be formed by rotating rectangles or trapezoids about the reactor axis or by translating rectangles parallel to the reactor axis. End surfaces of all regions must be normal to the reactor axis. Sides of the rectangular parallelopiped must be parallel or normal to the X-Z and Y-Z planes. The shield is enclosed by a cylinder of radius R_{max} and length $2(Z_{max})$. It is not necessary that all space within the enclosing cylinder be contained in regions. Space between regions is recognized as bulk material.

2.2 Material Description

Material description of the reactor-shield assembly is accomplished by associating a specified composition with each region by means of a code number. Compositions are expressed as volume fractions $\theta_{h(g)m}$ for each material contained in the assembly. The subscripts h and m indicate composition and material numbers respectively. The composition number is indicated to be a function of the region number g as discussed above. The volume fraction for each material must be specified for each composition even though some may be zero. Bulk material must be entirely material one.

3. SOURCE DESCRIPTION

Radiation sources may be defined by the function $S(\vec{r}_s, E)dE$ as the source strength per unit volume at $\vec{r}_s$ for particles or photons of energy E in range dE for continuous spatial distribution of sources with continuous energy spectra. If the energy spectrum is constant throughout the volume, the function is separable in the space coordinates and energy may be written

$$S(\vec{r}_s, E)\, dE = S(\vec{r}_s)\, B(E)dE$$

where $S(\vec{r}_s)$ may represent power density, watts/cm^3, or fission rate density, fissions/sec-cm^3, and B(E) may represent particles or photons of energy E per sec-watt or per fission. The subscript s in the above expression indicates source.

It was assumed in the coding of this program that the source density would be independent of the angle ϕ_s because of the usual symmetrical reactor loading. Hence, $S(\vec{r}_s)$ becomes $S(r_s, Z_s)$. In addition, the function was assumed separable in r_s and Z_s so that

$$S(r_s, Z_s) = S(r_s)\, S(Z_s)$$

The functions for description of axial neutron or gamma ray source distributions are

$$S(Z) = \tau_1 \cos [\tau_2 (Z - \tau_3)]$$

Functions for description of radial neutron or gamma ray source distributions are

$$S(r_s) = \epsilon_1 \cos [\epsilon_2 (r_s - \epsilon_3)]$$

The coefficients appearing in the above may be the same or different for neutrons and gamma rays except that the coefficient τ_1 for gamma rays and the coefficient ε_1 for neutrons must always equal one. Source location and dimensions are described in cylindrical coordinates.

Point-point attenuation functions used in this program for dose rate calculations are evaluated for the straight-line paths connecting a specified receiver point (r_r, ϕ_r, z_r) and specified points in the source region. These source region nodal points* are located in the rings formed by intersection of concentric shells and planes normal to the axis. There may be two to fifteen equally spaced planes and shells. There may be two to fifteen lines of source points in each shell. The numbers per shell are the same, and they are equally spaced in the shells.

* Hereafter, source region nodal points will be called simply, although somewhat incorrectly, source points.

4. WEIGHT CALCULATION

The weight of each material in the reactor-shield assembly is automatically computed by

$$W_m = \delta_m \sum_{g=1}^{G} \theta_{h(g)m} V_g \,, \quad m \geq 1 \tag{5}$$

$$W_1 = \delta_1 \left[2\pi z_{max} R_{max}^2 + \sum_{g=1}^{G} (\theta_{h(g)1} - 1) V_g \right] \tag{6}$$

where g = region number

G = number of regions in reactor-shield assembly

δ = material density

V_g = computed volume of region g

The total reactor-shield assembly is given by

$$W_T = \sum_{m=1}^{M} W_m \tag{7}$$

where M equals the number ofmaterials present. Other symbols used above have the same meaning presented previously.

5. GEOMETRY CALCULATION

The length of the straight line path between a source point and a receiver point is required in the evaluation of the fast neutron and gamma ray point-point attenuation functions for the source-receiver path. Also, the distance traversed along this path in each material is required. Actually, the fraction of the path contributed by each material is computed. Then, these path fractions θ_m are multiplied by the path length ρ to yield the desired thickness of each material.

The distance traversed in each shield region is determined by a stepping procedure which is explained in detail in Section 10. These distances are used in the following equation to compute the path fraction for $m > 1$.

$$\theta_m = \frac{1}{\rho} \sum_{g=1}^{G} \theta_{k(g)m} \rho_g + \theta_{k_t m} \rho_t \quad , m > 1 \qquad (8)$$

where ρ = total distance from source point to receiver point

ρ_g = distance traversed by source-receiver path in region g.

ρ_t = distance traversed by source-receiver path outside shield boundaries

$\theta_{k_t m}$ = volume fraction of material m in composition k_t which exists outside the shield boundaries

Other symbols were defined previously.

The path fraction for material one is given by

$$\theta_1 = 1 - \sum_{m=2}^{M} \theta_m \qquad (9)$$

If a source-receiver path encounters no material one, the machine may compute a negative θ_1 because of the way the other θ_m's are rounded off. When this happens, the machine will stop. Obviously, material one should be chosen with some discretion. Possible difficulty can frequently be avoided by specifying the material surrounding the shield as material one.

6. ATTENUATION FUNCTIONS

6.1 General

The method used in this program to determine fast neutron and gamma ray dose rates combines the use of point-point attenuation functions with integration over source regions. Since neither neutrons nor gamma rays arriving at a receiver from separate sources interact appreciably, this method can be assumed correct if the point-point attenuation function is, in general, a complicated functional which is defined for all possible paths of the particle between the source and receiver points. This functional can only be approximated in any actual situation.

It is assumed in the program, that to a reasonably good approximation, the point-point attenuation function depends only on the quantity of each material encountered by the primary ray proceeding from the source point directly to the receiver point. This condition is reasonably well satisfied in many cases in which the important part of the scattered radiation is well collimated along the path of the primary ray. The point-point attenuation function can be expressed as a function of the path length ρ, the path fraction θ_m for each material, the source energy E, and the macroscopic cross section $\Sigma_m(E)$ as

$$\mathcal{F} = \mathcal{F}(\Sigma_1(E), \ldots, \Sigma_m(E), \ldots, \Sigma_n(E), \theta_1, \ldots, \theta_m, E, \rho) \qquad (10)$$

Henceforth, this function will be written

$$\mathcal{F} = \mathcal{F}(\Sigma_m(E), \theta_m, E, \rho) \qquad (11)$$

It is assumed that the radiation is emitted isotropically by a point source. It is further assumed that the function $\mathcal{F}$ can be written as a product of a

material attenuation function $\psi(\Sigma_m(E), \theta_m, E, \rho)$ and a geometric attenuation function $1/4\pi\rho^2$, i.e.,

$$\Phi(\Sigma_m(E), \theta_m, E, \rho) = \frac{\psi(\Sigma_m(E), \theta_m, E, \rho)}{4\pi\rho^2} \qquad (12)$$

More specifically, $\Phi(\Sigma_m(E), \theta_m, E, \rho)$ equals the flux at the receiver point due to particles of energy E emitted by a point source of unit strength.

6.2 Fast Neutron

A modification of the point kernel suggested by Albert and Welton in their theory of neutron attenuation in hydrogenous media is used for calculating direct beam fast neutron dose rates around fission sources. This theory combines a theoretical hydrogen cross section with an integration over an approximation to an experimentally-determined fission spectrum to obtain the uncollided flux as a function of penetration distance into the medium. The attenuation effects of materials other than hydrogen are included by assuming exponential attenuation and treating the cross sections as energy-independent adjustable parameters to be determined by best fit to experimental data. This treatment is based on the assumption that all heavy materials are followed by sufficient hydrogenous materials that the use of effective removal cross sections is valid.

The point-point material attenuation function used for fast neutrons in this program is

$$\psi_m(\Sigma_m, \theta_m, \rho) = \alpha_1 \left(\rho \sum_1^L \eta_m \theta_m\right)^{\alpha_2} \exp\left[-\alpha_3 \left(\rho \sum_1^L \eta_m \theta_m\right)^{\alpha_4}\right] \cdot \exp\left[-\rho \sum_{m=1}^M \theta_m \Sigma_m\right] \qquad (13)$$

where α_1, α_2, α_3 and α_4 are constants and η_m is the ratio of the hydrogen density in material m to that in water. The hydrogenous materials must be among the first L materials. The above equation is not energy dependent, since the method of obtaining this function results in an integration over the fission neutron energy spectrum.

6.3 Gamma Ray

The point-point material attenuation functions for computing the uncollided contribution of a point source of gamma rays having an initial energy E_j to the flux or dose rate at a receiver is

$$\psi_\gamma(\Sigma_m(E_j), \theta_m, E_j, \rho) = e^{-\rho \sum_{m=1}^{M} \theta_m \Sigma_m(E_j)} \qquad (14)$$

Photons are removed from the direct beam by several absorption and scattering processes. Photons scattered from the beam may be subsequently scattered back to the receiver, thus increasing the flux or dose rate. Photons, of course, lose energy in each scattering collision. NDA used a moments method solution of the Boltzmann transport equation to compute the scattered contributions to energy flux, energy absorption rate, and dose rate in several infinite homogeneous materials from sources of various initial energies. They did this for several material thicknesses in relaxation lengths. NDA then computed ratios of total flux to uncollided flux, etc. These ratios are known as buildup factors and are dependent on material, material thickness, and initial photon energy. Buildup factors are available in NYO-3075[3] for energy flux, energy absorption rate, and dose rate. Uncollided contributions to flux, etc., are multiplied by the appropriate buildup factor to obtain the sum of the uncollided and scattered contributions. Therefore, the point-point material attenuation function becomes

$$\psi_\gamma(\Sigma_m(E_j), \theta_m, E_j, \rho) = B(\Sigma_m(E_j), \theta_m, E_j, \rho) e^{-\rho \sum_{m=1}^{M} \theta_m \Sigma_m(E_j)} \qquad (15)$$

Actual shields can seldom be considered infinite and homogeneous. Shields usually are composed of light and heavy materials. Because of the resulting uncertainty in the proper use of the NDA infinite buildup factors, the program offers the two following choices of gamma ray kernels:

Long Form

$$\psi_\gamma(\Sigma_m(E_j), \theta_m, E_j, \rho) = \prod_{p=1}^{P} {}_pB(\Sigma_m(E_j), \theta_m, E_j, \rho) \exp\left[-\rho \sum_{m=1}^{M} \theta_m \Sigma_m(E_j)\right] \quad (16)$$

$${}_pB(\Sigma_m(E_j), \theta_m, E_j, \rho) = {}_p\beta_0(E_j) + {}_p\beta_1(E_j)\,{}_pX_j + {}_p\beta_2(E_j)\,{}_pX_j^2 + {}_p\beta_3(E_j)\,{}_pX_j^3 \quad (17)$$

$${}_pX_j = \Sigma_{m=pd}(E_j)\left[{}_p\lambda\rho + \sum_{m=1}^{M}\sum_{k=1}^{H} {}_p\gamma_{k(g)m}\theta_{k(g)m}\beta_g\right] + \sum_{m=1}^{M}\Sigma_m(E_j)\sum_{k=1}^{H} {}_p\tau_{k(g)m}\theta_{k(g)m}\beta_g \quad (18)$$

Short Form

$$\psi_\gamma(\Sigma_m(E_j), \theta_m, E_j, \rho) = B(\Sigma_m(E_j), \theta_m, E_j, \rho) \exp\left[-\rho \sum_{m=1}^{M} \theta_m \Sigma_m(E_j)\right] \quad (19)$$

$$B(\Sigma_m(E_j), \theta_m, E_j, \rho) = \beta_0(E_j) + \beta_1(E_j)X_j + \beta_2(E_j)X_j^2 + \beta_3(E_j)X_j^3 \quad (20)$$

$$X_j = \rho \sum_{m=1}^{M} \theta_m \Sigma_m(E_j) \quad (21)$$

Symbols appearing in the above which have not previously been defined have the following meaning:

X_j = thicknesses of various materials encountered by a source receiver path, relaxation lengths.

β = coefficient evaluated by fitting cubic to NDA buildup data.

λ, τ, γ = coefficient determining manner of use of buildup factor. These coefficients always have the values 0, -1, or 1.

$\Sigma(m,d)$ = macroscopic cross section for gamma rays in material d.

p = number of buildup factors.

The complexity of Equations 16 and 18 arises from the uncertainty, which existed at the time the program was coded, regarding the use of buildup factors for multi-regioned shields. Considerable versatility is achieved by writing the point kernel in this form. For example, it may be desired to estimate the buildup along the entire path of a ray by calculating the buildup for the last region encountered, g = u. If only region g = u contains the composition labeled h = t, then

$$P = 1$$

$$_1\lambda = {}_1\nu_{h(g)m} = 0 \text{ for all } h(g) \text{ and } m$$

$$_1\tau_{h(g)m} = \delta_{t,h(g)} \quad \text{for all } m$$

Equation 18 then becomes

$$_1X_j = \sum_{m=1}^{M} \Sigma_m(E_j)\,\theta_{h(u)m}\,\rho_u$$

It can be seen immediately that this result gives the total number of relaxation lengths along the path of the ray in region u.

Ambiguity can occur when several regions in a shield contain the same composition since the screening constants ν and τ are ordered according to h(g)m rather than gm. Such ambiguities can be avoided by specifying a composition for each region in the shield even though some regions have the same composition. Buildup in a multi-regioned shield is frequently approximated by buildup in a single material for a thickness equal to the sum of the thicknesses in relaxation lengths of all materials encountered by a ray. Equations

16, 17, and 18 can be used to accomplish this calculation. However, the same result can be achieved by use of Equations 19, 20, and 21 without the use of screening constants. The second method requires less inputdata and shorter computing times. A control word in input data indicates to the machine which of the methods is to be used.

6.4 Attenuation Function Print-Out

Control words in the input data may be used to request print-out of either ψ_n/ρ^2 or ψ_γ/ρ^2 or both for each source-receiver path. These are printed in the same sequence followed in the computations. This sequence will be discussed in Section 10. Knowledge of ψ_n/ρ^2 and ψ_γ/ρ^2 for each source-receiver path is frequently very valuable in iterative shield design procedures.

7. INTEGRATION

The response of an isotropic detector to radiation from a small element of source volume dV_s emitting particles or photons of energy E in range dE is

$$dD = \frac{S(\vec{r}_s, E)\,\psi(\Sigma_m(E), \theta_m, E, \rho)}{4\pi\rho^2}\, dV_s\, dE \qquad (22)$$

The calculated detector response will obviously be in units determined by the units of the source, material attenuation function, and spatial dimensions. The detector response from an entire source region which is emitting radiation with a continuous energy spectrum is given by

$$D = \int_{\text{source energy spectrum}} \int_{\text{source region}} \frac{S(\vec{r}_s, E)\,\psi(\Sigma_m(E), \theta_m, E, \rho)}{4\pi\rho^2}\, dV_s\, dE \qquad (23)$$

The following special version of the above equation is used in this program for computing fast neutron dose rates from cylindrical volume sources:

$$D_n = \frac{1}{2\pi}\int_{z_{sl}}^{z_{su}} S(z_s)\,dz_s \int_{r_{sl}}^{r_{su}} S(r_s), r_s\, dr_s \int_{\phi_{sl}}^{\phi_{su}} \frac{\psi_n(\Sigma_m, \theta_m, \rho)}{\rho^2}\, d\phi_s \qquad (24)$$

z_{sl} z_{su}, r_{sl} r_{su}, ϕ_{sl} and ϕ_{su} are lower and upper limits of integration respectively on the source space coordinates z_s, r_s, and ϕ_s. The equation for Dn above was written with the assumption that the integration over ϕ_s would be carried out from $-\pi/2$ to $+\pi/2$. Consequently, multiplying by two because of symmetry results in a factor of $2\pi\rho^2$ in the denominator rather than $4\pi\rho^2$. A correction is necessary if integration is not carried out between these limits. The source function, as discussed in Section 3, is assumed independent of ϕ_s and separable in z_s and r_s. Since the fast neutron point kernel is energy independent, the source function and the dose rate equation are also energy independent.

The equation coded for computing the total dose rate from a cylindrical volume source of gamma rays of J descrete energies is

$$D_\gamma = \sum_{j=1}^{J} \frac{K(E_j)B(E_j)}{2\pi} \int_{z_{s\ell}}^{z_{su}} S(z_s)\,dz_s \int_{r_{s\ell}}^{r_{su}} S(r_s)\,r_s\,dr_s \int_{\phi_{s\ell}}^{\phi_{su}} \frac{\psi_\gamma(\Sigma_m(E_j), \theta_m, E_j, \rho)}{\rho^2}\,d\phi_s \qquad (25)$$

In addition to being separable in Z_s and r_s, the source function in this equation is assumed separable in energy. A conversion factor $K(E_j)$ is included to convert from flux due to photons of initial energy E_j to any desired units. The flux or dose rate is computed and printed for each initial energy. Finally, the total dose rate is computed and printed.

Integration of these equations is accomplished by the trapezoidal method. Details of the program logic are explained in Section 10.

8. LIMITATIONS

Care should be exercised in the application of the point kernels used in this program, since they are all infinite homogeneous medium kernels. The Albert-Welton kernel, in addition, is restricted to hydrogenous media or to shields in which non-hydrogenous materials are backed with sufficient thickness of hydrogenous materials to validate the use of the kernel.

The use of an infinite medium kernel has the effect of surrounding the source and receiver points with an infinite medium of the same composition as the shielding which lies along the source-receiver path. The constants used in the kernel are those determined for such an infinite-medium case, and include the effects of scattering to the receiver from all regions in the medium. The density of each material in this medium is effectively that determined by distributing the intervening shield material uniformly over the total distance between source and receiver points. Consequently, a shield-air boundary (assuming air outside the shield) may cause the actual dose rate to be substantially different from that computed by the point kernel technique assuming an infinite homogeneous medium.

The effect of the shield-air boundary is simplest for the case of a receiver point on the shield surface. Here a point kernel calculation of the dose rate at the receiver includes scattering to the receiver from regions of the medium which are outside the shield boundary and which do not, of course, exist in the real situation. The densities of the materials in the medium will be the same in the calculation as in the real shield, obviously, since the entire source-receiver path is filled with shielding material. The shield-air boundary has the effect of removing part of the material assumed to be present in the calculation. The material removed is from regions which do not affect the direct

beam attenuation, but can only act to increase the computed dose rate through scattering. Hence, the point kernel calculations for this case will yield a dose rate which is too high.

The same effect is present for a receiver point located in the air at some distance from the shield-air boundary. Here, however, other effects occur as a consequence of the "smearing out" of the shield material into a uniform distribution along the source-receiver path, and the subsequent use of the resulting reduced densities in the infinite medium which is assumed for the calculation. The point kernel calculation will predict fewer scatterings than will actually occur in the regions filled by the actual shield; and it will predict for such scatterings an attenuation which is too high, since the calculation assumes some material in the regions actually occupied by air space. Both of these effects tend to make the calculated dose rate too low. Whether the net result of all these effects is to make the calculated dose rate too high or too low, appears impossible to answer for the general case.

If the buildup of the scattered radiation can be considered to be well collimated along the source-receiver path, the effects just described can be ignored and the infinite medium point kernel calculation can be used with some confidence. The success of the method depends on the validity of this assumption of good collimation.

The above discussion relates the effect of the finite extent of the shield to the dose rate prediction by the point kernel method. Inhomogeneous regions in the shield also may present difficulties to the method. Obviously, the greater the difference in the attenuation characteristics of the various regions of inhomogeneity, the greater the uncertainty in using homogeneous point kernels.

Thus, the application of the fast neutron kernel to inhomogeneous shields would be expected to lead to reasonably good results provided all non-hydrogenous materials were backed by sufficient hydrogenous material. On the other hand, the gamma ray kernel could conceivably lead to large errors, especially in those cases where scattering of the gamma rays around highly attenuating regions can occur (short-circuiting).

The foregoing discussion regarding the validity of the point kernel techniques is by no means exhaustive, but it is given to illustrate the need for caution in applying the technique and interpreting results of calculations based on it.

9. EQUATIONS CODED

9.1 Weight

$$W_m = \delta_m \sum_{g=1}^{G} \theta_{h(g)m} V_g \qquad , m > 1$$

$$W_1 = \delta_1 \left[2\pi Z_{max} R^2_{max} + \sum_{g=1}^{G} (\theta_{h(g)1} - 1) V_g \right]$$

$$W_T = \sum_{m=1}^{M} W_m$$

9.2 Path Fraction

$$\theta_m = \frac{1}{\rho} \sum_{g=1}^{G} \theta_{h(g)m} \rho_g + \theta_{h_t m} \rho_t, \qquad m > 1$$

$$\theta_1 = 1 - \sum_{m=2}^{M} \theta_m$$

9.3 Fast Neutron Dose Rate

$$D_m = \frac{1}{2\pi} \int_{Z_{sl}}^{Z_{su}} S(Z_s)\, dZ_s \int_{R_{sl}}^{R_{su}} S(R_s)\, dR_s \int_{\phi_{sl}}^{\phi_{su}} \frac{\psi_m(E_m, \theta_m, \rho)}{\rho^2}\, d\phi_s$$

$$\psi_m(\Sigma_m, \theta_m, \rho) = \alpha_1 \left(\rho \sum_1^L n_m \theta_m\right)^{\alpha_2} \exp\left[-\alpha_3 \left(\rho \sum_1^L n_m \theta_m\right)^{\alpha_4}\right] \exp\left[-\rho \sum_{m=1}^M \theta_m \Sigma_m\right]$$

9.4 Gamma Ray Dose Rate

$$D_\gamma = \sum_{j=1}^J \frac{K(E_j) B(E_j)}{2\pi} \int_{z_{sl}}^{z_{su}} S(z_s)\, dz_s \int_{r_{sl}}^{r_{su}} S(r_s) r_s\, dr_s \int_{\phi_{sl}}^{\phi_{su}} \frac{\psi_\gamma(\Sigma_m(E_j), \theta_m, E_j, \rho)}{\rho^5}\, d\phi_s$$

Long form

$$\psi_\gamma(\Sigma_m(E_j), \theta_m, E_j, \rho) = \prod_{p=1}^P {}_pB(\Sigma_m(E_j), \theta_m, E_j, \rho) \exp\left[-\rho \sum_{m=1}^M \theta_m \Sigma_m(E_j)\right]$$

$${}_pB(\Sigma_m(E_j), \theta_m, E_j, \rho) = {}_p\beta_0(E_j) + {}_p\beta_1(E_j) {}_pX_j + {}_p\beta_2(E_j) {}_pX_j^2 + {}_p\beta_3(E_j) {}_pX_j^3$$

$${}_pX_j = \sum_{m=p}^{} d(E_j) \left[{}_p\lambda \rho + \sum_{m=1}^M \sum_{h=1}^H {}_p\nu_{h(g)m}\, \theta_{h(g)m} \rho_g\right] + \sum_{m=1}^M \Sigma_m(E_j) \sum_{h=1}^H {}_p\zeta_{h(g)m}\, \theta_{h(g)m} \rho_g$$

Short form

$$\psi_\gamma(\Sigma_m(E_j), \theta_m, E_j, \rho) = B(\Sigma_m(E_j), \theta_m, E_j, \rho) \exp\left\{-\rho \sum_{m=1}^M \theta_m \Sigma_m(E_j)\right\}$$

$$B(\Sigma_m(E_j), \theta_m, E_j, \rho) = \beta_0(E_j) + \beta_1(E_j) X_j + \beta_2(E_j) X_j^2 + \beta_3(E_j) X_j^3$$

$$X_j = \rho \sum_{m=1}^M \theta_m \Sigma_m(E_j)$$

9.5 Source Functions

Axial

$$S(z_s) = \tau_1 \cos\left[\tau_2 (z_s - \tau_3)\right]$$

Radial

$$S(r_s) = \epsilon_1 \cos\left[\epsilon_2 (r_s - \epsilon_3)\right]$$

10. PROGRAM LOGIC

Equations and functions in this program are coded as subroutines, so that individual equations or functions may be changed without seriously altering the logic of the main program. Control words are used to indicate which calculations are to be made and which functions are to be used.

The W subroutine computes the weights of the individual materials in the reactor shield assembly and then sums these weights to obtain the total weight.

The CHPT subroutine computes the coordinates of source points equally spaced between integration limits. Coordinates for equal spacing are obtained by adding calculated increments to coordinates of preceding points. The size of the increment for any space variable is obtained by dividing the difference between integration limits for that variable by the specified number of equally spaced points minus one. Consequently, at least two points must be used on each variable. Extreme points are located at the integration limits. Only equal spacing on the integration variables is allowed for in Program 04-0. A modification of 04-0 permitting unequal spacing has been incorporated in Program 04-2.

The GEOM subroutine computes the total distance between a source point and a receiver point and the penetration distance in each region traversed. The total distance ρ is computed by

$$\rho = \sqrt{(X_s - X_r)^2 + (y_s - y_r)^2 + (z_s - z_r)^2}$$

where X_s, Y_s, Z_s, and X_r, Y_r, Z_r are computed rectangular coordinates of the source and receiver point respectively. The subroutine then computes the rectangular components of an input step size $\Delta\rho$ for use in a stepping proce-

dure. The NEPT subroutine proceeds by stepping along the source-receiver path. The subroutine adds the rectangular components of $\Delta\rho$ to the coordinates of the previous point on this path. It then determines whether the last step passed the receiver point or the shield boundary. If neither of these events occurred, the subroutine determines if the step crossed a boundary of the region. If the point is still in the region, another step is taken. If the step crossed a boundary of the region, the coordinates of the point of crossing are computed by one of three subroutines, depending upon the region type.

The NXI subroutine is used for regions formed by translating rectangles; the NRCO subroutines are used for regions formed by rotating trapezoids; and the NRCY subroutine is used for regions formed by rotating rectangles. The distance traversed in the region is computed and stored with the region code number. The program then searches the region description table to determine which region now contains the point, and the above process is repeated. If it is found in the NEPT subroutine that the receiver point was passed, the coordinates of the receiver point and the previous point on the path are used to compute the exact distance between the two. The subroutine then returns to the main program. When the NEPT subroutine determines that a shield boundary was crossed, the distance in the last region and the final distance to the receiver point are computed. The stepping procedure is discontinued at the shield boundaries.

If, at any time during the search of the region description tables, a point is not found in any of the described regions (a point on a boundary is not in a region unless it is on the axis in a solid cylindrical or conical region) the point is considered to be in bulk material. The NEPT subroutine searches the

complete region description table after each step from a point located in bulk material. In addition, it makes the usual tests to determine whether the receiver point or a shield boundary was reached.

The VFRAC subroutine multiplies the computed penetration distance in each region by the volume fraction of each material in the composition contained in the region. Volume fractions for the composition are located by means of the region code number stored with the computed penetration distance. Thus, the penetration distance through each material in each region is computed. All material outside the shield boundary is recognized as composition h_f, and bulk material is recognized as material one. The end result of the VFRAC subroutine is the total thickness of each material traversed by a source-receiver path.

The KERNA subroutine computes the fast neutron point-point attenuation function. Gamma ray point-point attenuation functions are computed by the KERNB subroutine. A control word determines which of the two forms of the gamma ray kernels to use.

The radial source distribution function is evaluated at source point radii by the PWRR subroutine. The PWRZ subroutine evaluates the axial source distribution function.

Integration over the source region is performed by the INTEG subroutine. Although the INTEG subroutine is coded to handle trapezoidal integration with unequal spacing of integrand points, the complete program is capable of handling only equal spacing of integrand points.

The program first computes weights of individual materials and the total reactor shield assembly weight. The program next computes and stores coordinates of the planes, shells, and lines of source points. The program then determines

what other computations are desired and proceeds to the proper initialization for the computations. The first receiver point and the source point located at the lower limit on Z_s, the lower limit on r_s, and the lower limit on ϕ_s are selected and the required computations in the GEOM, VFRAC, KERNA, and KERNB subroutines are performed. Results from the KERNA and KERNB subroutines are stored on tape until required in the INTEG subroutine. The procedure is then repeated for the source point in the same ring at the next larger value of ϕ_s. When this procedure is completed for all values of ϕ_s in the first ring of source points, the procedure is repeated for successively larger values of r_s and the same value of Z_s. When this procedure is completed for all values of r_s and ϕ_s and the lowest values of Z_s, the program selects the next larger value of Z_s and repeats the entire procedure. This procedure is continued until results from the KERNA and KERNB subroutines are stored for the paths connecting the receiver point and each source point.

The INTEG subroutine then directs the machine to integrate, separately for neutrons and gamma rays, over ϕ_s for the lowest values of r_s and Z_s. The result of the integration is stored, and the integration is repeated for successively larger values of r_s and the same values of Z_s. The integrals over ϕ_s are then multiplied by r_s and the radial source distribution function, and integration is performed over r_s.

This integration procedure is repeated successively for larger values of Z_s. The double integrals are then multiplied by the axial source distribution function, and integration is performed over Z_s.

The result of the triple integration for neutrons is multiplied by $\frac{1}{2\pi}$ to obtain the fast neutron dose rate at single receiver point. The result of the

triple integration for gamma rays of initial energy E_j is multiplied by $\frac{1}{2\pi}$, $K(E_j)$, and $B(E_j)$ to obtain the dose rate due to those gamma rays. The dose rate due to gamma rays of all initial energies is finally obtained by summation.

After computing the dose rate at a receiver point, the program selects the next receiver point and again performs all the above calculations. After all calculations are completed for the final receiver point, the machine stops.

11. INPUT INSTRUCTIONS

Control Data

KDATES — Date specified as follows:

AAAA (1) XX, XXXX

AAA. (1) XX, XXXX

AAA. (2) X, XXXX

KCHGNO — Charge number (an integer up to five digits)

MMAXCD — Code number for geometry of shield

1 = cylindrical

*FIXPC — Fixed point control integers

IDENTIFICATION - Identification number (an integer up to 10 digits)

G - Number of regions ($G \leq 100$)

h_b - Number of composition outside shield

H - Number of compositions ($M \times H \leq 400$)

M - Number of materials ($M \leq 20$)

P - Number of buildup factors in the gamma ray attenuation function ($0 \leq P \leq 5$)

J - Number of gamma ray energy levels ($J \leq 20$)

Nr_s - Number of shells of source points ($2 \leq N \leq 15$)

$N\phi_s$ - Number of lines of source points in each shell ($2 \leq N \leq 15$)

NZ_s - Number of planes of source points ($2 \leq N \leq 15$)

L - Number of values of η to be used in neutron attenuation function ($0 \leq L \leq 7$)

if $L = 0$, $\psi_\eta = \exp\left[-\rho \sum_{m=1}^{M} \theta_m \Sigma_m\right]$

Neutron = 0 - Do not compute neutron dose rate.

1 - Compute neutron dose rate

Gamma = 0 - Do not compute gamma ray dose rate.

1 - Compute gamma ray dose rate.

γKernel = 0 - Use short form for gamma ray dose rate.

1 - Use long form for gamma ray dose rate.

PRT ψ_n/ρ^2 = 0 - Do not print ψ_n/ρ^2

1 - Print ψ_n/ρ^2 for each source-receiver path.

PRT ψ_γ/ρ^2 = 0 - Do not print ψ_γ/ρ^2

1 - Print ψ_γ/ρ^2 for each source-receiver path.

NUTAU = 0 - NU's and TAU's are not included in the decimal data to follow.

1 - NU's and TAU's are included in the decimal data to follow.

Material Densities

LDENSY $\delta_1, \delta_2, \delta_3, \ldots, \delta_M$

General Data

LMAXBD Rmax - maximum shield radius

$|Z_{max}|$ - magnitude of maximum axial dimension of shield

LFLOPC r_{sl} - lower integration limit on r_s

r_{su} - upper integration limit on r_s

ϕ_{sl} - lower integration limit on ϕ_s

ϕ_{su} - upper integration limit on ϕ_s

Z_{sl} - lower integration limit on Z_s

Z_{su} - upper integration limit on Z_s

Note: Upper and lower integration limits for a variable cannot be equal.

LDELTA $\Delta\rho$ - step size

Region Description

IREDES Region descriptions. One card per region with the following information

1. Code number

   If region is rectangular 512 + h

   cylindrical 1024 + h

   conical 2048 + h

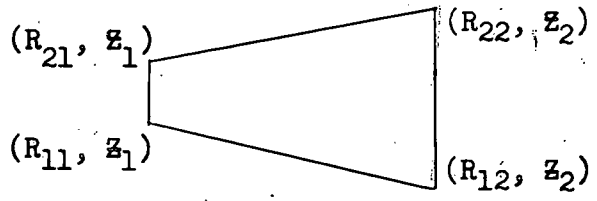
   where h is the number of the composition contained in the region. All regions which contain the same composition should have the same h.

2. Description

   Rectangular $X_1, X_2, Y_1, Y_2, Z_1, Z_2$

   Cylindrical R_1, R_2, Z_1, Z_2

   Conical $R_{11}, R_{12}, R_{21}, R_{22}, Z_1, Z_2$



Material Volume Fractions

LPFRAC Volume fraction of each material in each composition. If possible, place volume fractions for each material on a single card.

$\theta_{11}, \theta_{21}, \theta_{31}, \ldots \theta_{h1}$
$\theta_{12}, \theta_{22}, \theta_{32}, \ldots \theta_{h2}$
$\vdots$
$\theta_{1m}, \theta_{2m}, \theta_{3m}, \ldots \theta_{hm}$

Neutron Data

LCONSA Constants for neutron attenuation function

$\alpha_1, \alpha_2, \alpha_3, \alpha_4, \kappa_1, \kappa_2, \kappa_3, \ldots, \kappa_L$
Neutron Removal Cross Sections

LCSECT Neutron removal cross sections for m materials

$\Sigma_1, \Sigma_2, \Sigma_3, \ldots, \Sigma_M$
Neutron Axial Source Function Data

LCONZA Constants τ_1, τ_2, τ_3 for cosine axial neutron source function

Neutron Radial Source Function Data

LCONRA Constants $\epsilon_1, \epsilon_2, \epsilon_3$ for cosine radial neutron source function
ϵ_1 must equal 1.

Gamma Ray Conversion Factors

LCONKJ Gamma ray conversion factors for J energy levels
$K(E_1), K(E_2), K(E_3), \ldots, K(E_J)$

Gamma Ray Source Data

LCONBJ Gamma ray source strengths for J energy levels
$B(E_1), B(E_2), B(E_3), \ldots, B(E_J)$

Gamma Ray Axial Source Function Data

LCONZB Constants τ_1, τ_2, τ_3 for cosine axial gamma ray source function

τ_1 must equal 1.

Gamma Ray Radial Source Function Data

LCONRB Constants $\epsilon_1, \epsilon_2, \epsilon_3$ for cosine radial gamma ray source function

Gamma Ray Linear Absorption Coefficients

LSIGEJ Gamma ray linear absorption coefficients for M materials and J energy levels.

$\Sigma_{11}, \Sigma_{21}, \Sigma_{31}, \ldots, \Sigma_{M1}$

$\Sigma_{12}, \Sigma_{22}, \Sigma_{32}, \ldots, \Sigma_{M2}$

$\Sigma_{1J}, \Sigma_{2J}, \Sigma_{3J}, \ldots, \Sigma_{MJ}$

Gamma Ray Buildup Coefficients

LBETAS Buildup coefficients (4 x P x J values)

$1\beta_{01}, 1\beta_{11}, 1\beta_{21}, 1\beta_{31},$

$2\beta_{01}, 2\beta_{11}, 2\beta_{21}, 2\beta_{31},$

$P\beta_{01}, P\beta_{11}, P\beta_{21}, P\beta_{31},$

$1\beta_{02}, 1\beta_{12}, 1\beta_{22}, \beta_{32},$

$P\beta_{0J}, P\beta_{1J}, P\beta_{2J}, P\beta_{3J}.$

Gamma Ray Buildup Materials

LMDJGG Number of material used in computation of buildup factor p.

$1^d, 2^d, 3^d, 4^d, 5^d$

Gamma Ray Buildup Lambdas

LLAMDA Constant used in computation of buildup factor p.

$1\lambda, 2\lambda, 3\lambda, 4\lambda, 5\lambda.$

Gamma Ray Screening Constants

LTAFRA Gamma ray screening constants $p\nu_{h(g)m}$ and $p\tau_{h(g)m}$ entered for each p in the following order. Each group begins with initial address LTAFRA.

LTAFRA,	LTAFRA,	LTAFRA,
$1\nu_{11}$	$1\tau_{11}$	$2\nu_{11}$
$1\nu_{21}$	$1\tau_{21}$	$2\nu_{21}$
$1\nu_{31}$	$1\tau_{31}$	$2\nu_{31}$
$\downarrow$	$\downarrow$	$\downarrow$
$1\nu_{H1}$	$1\tau_{H1}$	$2\nu_{H1}$
$1\nu_{12}$	$1\tau_{12}$	$2\nu_{12}$
$\downarrow$	$\downarrow$	$\downarrow$
$1\nu_{HM}$	$1\tau_{HM}$	$2\nu_{HM}$

Receiver Point Coordinates

#RECPT r_r, ϕ_r, z_r of receiver point.

Note: Input data for this program must be entered as floating point data except where specified as fixed point data in these instructions. Floating point data may be written in the following manner:

3.726/-2 or .03726

The limitation is eight digits and that representation which uses the least number of characters (numbers, period, slash, and minus signs are considered characters) is preferable. Do not include insignificant zeros such as 0.0354 or .03540.

Each piece of information must be separated by a comma and the last one must be followed by a comma. Sixty-five characters, including commas, are permitted on a line for data.

12. INPUT INTERNAL CONSISTENCY CHECKS

1. Region descriptions must be entered for the number of regions entered at G.

2. The integer entered at h_b cannot exceed the integer entered at H, but it may be any integer 1 to H.

3. Material volume fractions must be entered for each material for the number of compositions entered at H.

4. The total number of volume fractions entered must equal M x H.

5. The sum of the material volume fractions for any composition h must not exceed 1.

6. Material densities must be entered for the number of materials entered at M.

7. If a zero is entered under NEUTRON, the input under CONSA, CSECT, CONRA, and CONZA may be omitted (except that ϵ_1 must always be a one). If a one appears under NEUTRON, the above data must be included.

8. The number of values for ν entered in the CONSA table must equal the integer entered in L.

9. If a zero is entered under GAMMA, the input under BETAS, MDJGG, LAMDA, TAFRA, CONRB, CONZB, CONBJ, CONKJ, SIGEJ may be omitted (except that ϵ_1 must always be a one). If a one appears under GAMMA, portions of the above must be included.

10. The integer entered under J must equal the number of gamma ray conversion factors entered under CØNKJ and the number of gamma ray source strengths entered under CØNBJ.

11. The number of gamma ray absorption coefficients entered in the SIGEJ table must equal M x J.

12. If a zero appears under γ-KERNEL, the input under MDJGG, TAFRA, may be omitted. If a one appears under γKERNEL, portions of the above data must be included.

13. The number of buildup coefficients in the BETAS table must equal four times the integer under J times the integer under P.

13. PROGRAM OPERATING INSTRUCTIONS

13.1 Make-up of Binary Deck (168 Cards)

1. BILD 1 and 2, (2 cards)
2. GEPRO deck, (71 cards)
3. Program deck.
4. Transition card, (3100_8)

13.2 Make-up of Decimal Deck (which is placed directly behind binary deck).

1. If ν's and τ's are included in data, they are first in decimal deck. Each group is followed by an end of file card.
2. Date and charge number. (2 cards)
3. Fixed point constants with end of record punch. (1 card)
4. All other data (excepting receiver points). Groups of input data may be in any order, but within a group cards must follow order given on data sheet.
5. End of file card.
6. All receiver points, one card for each, each card with an end of record punch.
7. End of file card.
8. If there is more than one shield, each is set up as in (1) - (7) and placed directly behind the preceding shield.

13.3 Console and Tape Units

1. Sense switch 2 down.
2. Tapes 3, 5, and 6 ready. Tape 4 ready if ν's and τ's used.
3. Ready cards in reader and depress load card button to start program.

4. Program writes answers on tape 3 if sense switch 3 up. If sense switch 3 down, answers on direct line printer.

13.4 General Description of Program as Seen by Machine Operator

1. Read binary deck.
2. If tape 4 requested will read groups of cards and write on tape 4.
3. Reads 3 cards.
4. Writes small record on tape 5 and 6.
5. Reads cards up to end of file card.
6. Computes and writes small record on tape 3.
7. Reads receiver point.
8. Alternates computing and writing on both tapes 5 and 6 from 8 - 125 times, depending on input data
   (a) will sometimes write on tape 3 after each period of computing.
   (b) will also read tape 4 if it is used.
9. Rewinds tape 5, computes, and writes on tape 3 from 1 - 21 times.
10. Follows (7) - (9) for each receiver point.

14. PROGRAM STOPS AND INSTRUCTIONS

STOP (octal)	REASON	INSTRUCTIONS
3107	Error in data card	Pull deck
3142	Tape error on 4	See A.
3166	Error in data card	Pull deck
3423	Tape error on 6	See A.
3464	Tape error on 5	See A.
3646	Tape error on 4	See A.
4141	Tape error on 5	See A.
4143	Tape error on 5	See A.
4467	Tape error on 5	See A.
4471	Tape error on 5	See A.
4661	Error in receiver point data card	Push start
4665	Shield Completed	If another shield, push Start; if not, pull deck.
11616	Error in GEOM SRT	Pull deck
11674	Error in VFRAC SRT	Pull deck
11754	Error in KERN SRT	Pull deck
12025	Error in PWR SRT	Pull deck
Any other	Record; Pull; Do not get PM	

A. Pull any receiver points that have already run.

Switch Tape Units. Clear and reload deck.

15. APPENDIX

15.1 Sample Problem

The following sample problem was actually run on the IBM-704. The output masters were printed by the IBM printer.

Regions of all types permitted by Program 04-0 were used in this problem. Cross sectional views of the fictitious reactor shield assembly are shown in Figure 1.

Problem 1040 involves integration over the cylindrical core (region 1) for calculation of fast neutron and gamma ray dose rates.

Materials assumed for this problem were H_2O, Al, Be, Pb, U, and void. All compositions except that of the core (composition 1) were composed of single materials. Energy levels assumed for the gamma ray source were 2.5, 3.5, 4.5, 5.5, and 6.5 Mev.

The axial source function for Problem 1040 was

$$S(Z_s) = 150 \cos \left[.0349 \, (Z_s - 0) \right]$$

The radial source function was

$$S(r_s) = 1 \cos \left[.0261 \, (r_s - 0) \right]$$

The above source functions were applied for both neutrons and gamma rays.

The above functions are presented with the hope that they may make understanding of the input data a little easier. It is hoped that the remaining input data will be understood without such detailed presentation.

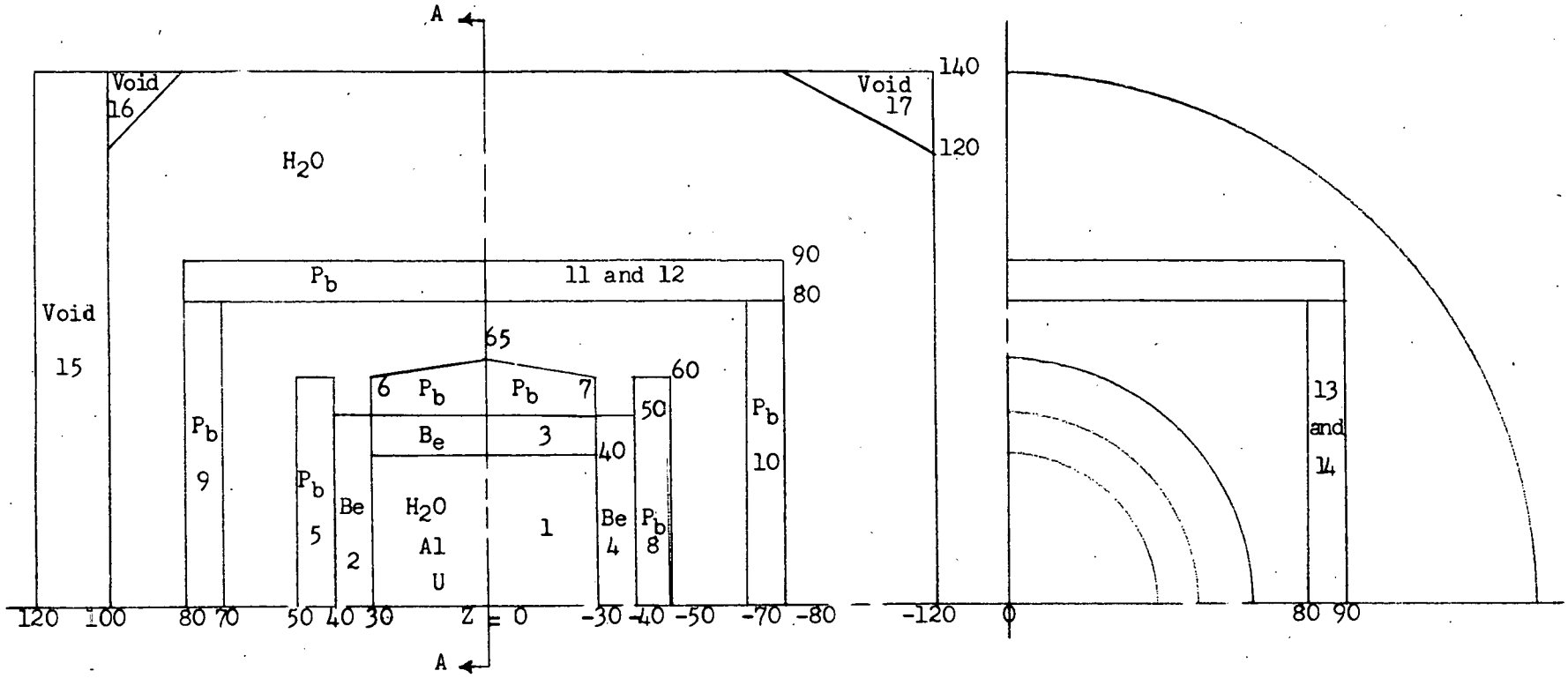


FIGURE 1

SAMPLE PROBLEM CONFIGURATION

15.2 Sample Problem Input

SHIELDING COMPUTER PROGRAM 04-0

DATA INPUT SHEETS

Problem Originator _____

Ext. _____

Date _____

Charge Number _____

☐ Must be Verified ☐ Verify if Time Allows

PROGRAM 04-0

Control Data

Col. 1	Col. 2-7		Col. 73

K	DATES,	2,	Date: Feb. 25, 1958

K	CHGNO,	1,	Charge Number: 70050

M	MAXCD,	35,	1,

*	FIXPC,	35,	Identification	G	n_b	H	M	P	J	Nr_s	$N\Phi_s$	
			1040 ,	17 ,	4 ,	4 ,	6 ,	1 ,	5 ,	4 ,	4 ,	FPC1
		NZ_s	L	Neutron	Gamma	γ Kernel	Prt.Ψ_n/ρ^2	Prt.Ψ_γ/ρ^2	NUTAU			
		4 ,	1 ,	1 ,	1 ,	0 ,	1 ,	1 ,	0 ,			FPC2

Material Densities

L	DENSY,	δ_1	δ_2	δ_3	δ_4	δ_5		DEL1
		1 ,	2.7 ,	18.7 ,	1.83 ,	11.4 ,		
		δ_6	δ_7	δ_8	δ_9	δ_{10}		DEL2
		0 ,	,	,	,	,		
		δ_{11}	δ_{12}	δ_{13}	δ_{14}	δ_{15}		DEL3
		,	,	,	,	,		
		δ_{16}	δ_{17}	δ_{18}	δ_{19}	δ_{20}		DEL4
		,	,	,	,	,		

870 076

PROGRAM 04-O

GENERAL DATA

Col 1	Col. 2-7	Col. 8	Col. 73

GAMMA RAY BUILDUP MATERIALS

		1^d	2^d	3^d	4^d	5^d
L	MDJGG,	,	,	,	,	,

GAMMA RAY BUILDUP LAMBDAS

		1^λ	2^λ	3^λ	4^λ	5^λ
L	LAMDA,	,	,	,	,	,

PROGRAM 04-0

GENERAL DATA

Col 1	Col.2-7	Col. 8		Col. 73
		R_{max}	$\|Z_{max}\|$	
L	MAXBD,	140 ,	120 ,	

		r_{sl}	r_{su}	φ_{sl}	φ_{su}	Z_{sl}	Z_{su}
L	FLØPC,	0 ,	40 ,	-1.5708 ,	1.5708 ,	-30 ,	30 ,

		$\Delta\rho$
L	DELTA,	9.5 ,

050 040

PROGRAM 04-0

Col. 1	Col. 2-7	Col. 8							Col. 73
		Code No.	Region Descriptions						
L	REDES,	1025	0	40	-30	30			R01
		1026	0	50	30	40			R02
		1026	40	50	-30	30			R03
		1026	0	50	-40	-30			R04
		1027	0	60	40	50			R05
		2051	50	50	65	60	0	30	R06
		2051	50	50	60	65	-30	0	R07
		1027	0	60	-50	-40			R08
		515	-80	80	-80	80	70	80	R09
		515	-80	80	-80	80	-80	-70	R10
		515	80	90	-90	90	-80	80	R11
		515	-90	-80	-90	90	-80	80	R12
		515	-80	80	80	90	-80	80	R13
		515	-80	80	-90	-80	-80	80	R14
		1028	0	140	100	120			R15
		2052	140	120	140	140	80	100	R16
		2052	120	140	140	140	-120	-80	R17
									R18
									R19
									R20
									R21
									R22
									R23
									R24
									R25

320 051

PROGRAM 04-0

Col. 1	Col. 2-7	Col. 8							Col. 73
		Code No.	Region Descriptions						
		,	,	,	,	,	,	,	
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340 052

PROGRAM 04-0

MATERIAL VOLUME FRACTIONS

Col. 1	Col. 2-7	Col. 8	Col. 73
		$\theta_{11} \rightarrow \theta_{H1}$	
L	PFRAC,	.583, 0, 0, 0,	M01
		$\theta_{12} \rightarrow \theta_{H2}$	
		.415, 0, 0, 0,	M02
		$\theta_{13} \rightarrow \theta_{H3}$	
		.002, 0, 0, 0,	M03
		$\theta_{14} \rightarrow \theta_{H4}$	
		0, 1, 0, 0,	M04
		$\theta_{15} \rightarrow \theta_{H5}$	
		0, 0, 1, 0,	M05
		$\theta_{16} \rightarrow \theta_{H6}$	
		0, 0, 0, 1,	M06
		$\theta_{17} \rightarrow \theta_{H7}$	
			M07
		$\theta_{18} \rightarrow \theta_{H8}$	
			M08
		$\theta_{19} \rightarrow \theta_{H9}$	
			M09
		$\theta_{110} \rightarrow \theta_{H10}$	
			M10

320 053

PROGRAM 04-0

MATERIAL VOLUME FRACTIONS

Col. 1	Col. 2-7	Col. 8	Col. 73
		$\theta_{111} \rightarrow \theta_{H11}$	M11
		$\theta_{112} \rightarrow \theta_{H12}$	M12
		$\theta_{113} \rightarrow \theta_{H13}$	M13
		$\theta_{114} \rightarrow \theta_{H14}$	M14
		$\theta_{115} \rightarrow \theta_{H15}$	M15
		$\theta_{116} \rightarrow \theta_{H16}$	M16
		$\theta_{117} \rightarrow \theta_{H17}$	M17
		$\theta_{118} \rightarrow \theta_{H18}$	M18
		$\theta_{119} \rightarrow \theta_{H19}$	M19
		$\theta_{120} \rightarrow \theta_{H20}$	M20

PROGRAM 04-0

NEUTRON DATA

Col. 1	Col. 2-7	Col. 8						Col. 73
		α_1	α_2	α_3	α_4	η_1		
L	CONSA,	7.29/9 ,	.29 ,	.83 ,	.58 ,	1 ,		CONS1
		η_2	η_3	η_4	η_5	η_6		
		,	,	,	,	,		CONS2
		η_7						
		,						CONS3

NEUTRON REMOVAL CROSS SECTIONS

		Σ_1	Σ_2	Σ_3	Σ_4	Σ_5		
L	CSECT,	.033 ,	.079 ,	.17 ,	.132 ,	.117 ,		SECT1
		Σ_6	Σ_7	Σ_8	Σ_9	Σ_{10}		
		0 ,	,	,	,	,		SECT2
		Σ_{11}	Σ_{12}	Σ_{13}	Σ_{14}	Σ_{15}		
		,	,	,	,	,		SECT3
		Σ_{16}	Σ_{17}	Σ_{18}	Σ_{19}	Σ_{20}		
		,	,	,	,	,		SECT4

NEUTRON AXIAL SOURCE FUNCTION DATA

		τ_1	τ_2	τ_3		
L	CONZA,	150 ,	.0349 ,	0 ,		

NEUTRON RADIAL SOURCE FUNCTION DATA

		ϵ_1	ϵ_2	ϵ_3		
L	CONRA,	1 ,	.0261 ,	0 ,		

PROGRAM 04-0

GAMMA RAY CONVERSION FACTORS

Col. 1	Col. 2-7	Col. 8						Col. 73
		$K(E_1)$	$K(E_2)$	$K(E_3)$	$K(E_4)$	$K(E_5)$		
L	CONKJ,	1.47 /-6 ,	1.35 /-6 ,	1.25 /-6 ,	1.18 /-6 ,	1.11 /-6 ,		KJ1
		$K(E_6)$	$K(E_7)$	$K(E_8)$	$K(E_9)$	$K(E_{10})$		
		,	,	,	,	,		KJ2
		$K(E_{11})$	$K(E_{12})$	$K(E_{13})$	$K(E_{14})$	$K(E_{15})$		
		,	,	,	,	,		KJ3
		$K(E_{16})$	$K(E_{17})$	$K(E_{18})$	$K(E_{19})$	$K(E_{20})$		
		,	,	,	,	,		KJ4

GAMMA RAY SOURCE DATA

		$B(E_1)$	$B(E_2)$	$B(E_3)$	$B(E_4)$	$B(E_5)$		
L	CONBJ,	7.7 /10 ,	3.4 /10 ,	1.5 /10 ,	6.4 /9 ,	2.8 /9 ,		BJ1
		$B(E_6)$	$B(E_7)$	$B(E_8)$	$B(E_9)$	$B(E_{10})$		
		,	,	,	,	,		BJ2
		$B(E_{11})$	$B(E_{12})$	$B(E_{13})$	$B(E_{14})$	$B(E_{15})$		
		,	,	,	,	,		BJ3
		$B(E_{16})$	$B(E_{17})$	$B(E_{18})$	$B(E_{19})$	$B(E_{20})$		
		,	,	,	,	,		BJ4

GAMMA RAY AXIAL SOURCE FUNCTION DATA

		τ_1	τ_2	τ_3		
L	CONZB,	1 ,	.0349 ,	0 ,		

GAMMA RAY RADIAL SOURCE FUNCTION DATA

		ϵ_1	ϵ_2	ϵ_3		
L	CONRB,	150 ,	.0261 ,	0 ,		

PROGRAM 04-O

GAMMA RAY BUILDUP COEFFICIENTS

Col. 1	Col. 2-7	Col. 8				Col. 73
		$p^{\beta_{0j}}$	$p^{\beta_{1j}}$	$p^{\beta_{2j}}$	$p^{\beta_{3j}}$	
L	BETAS,	1,	.746,	.016,	-.00049,	BET11
		1,	.633,	.0048,	-.00005,	
		1,	.544,	-.0007,	.00002,	
		1,	.487,	-.005,	.00011,	
		1,	.438,	-.0066,	.00013,	
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PROGRAM 04-0

GAMMA RAY LINEAR ABSORPTION COEFFICIENTS

Col. 1	Col. 2-7	Col. 8	Col. 73
		$\Sigma_{11} \cdots \Sigma_{M1} \cdots \Sigma_{1J} \cdots \Sigma_{MJ}$	
L	SIGEJ,	.0435, .103, .842, .0642, .477, 0,	S01
		.0361, .0886, .804, .0525, .465, 0,	S02
		.0319, .0788, .823, .0456, .477, 0,	S03
		.029, .0734, .86, .0408, .499, 0,	S04
		.0268, .0699, .898, .0371, .516, 0,	S05
			S06
			S07
			S08
			S09
			S10
			S11
			S12
			S13
			S14
			S15
			S16
			S17
			S18
			S19
			S20

320 058

PROGRAM 04-0

GAMMA RAY SCREENING CONSTANTS $_{p}V_{h(g)m}$

Col. 1	Col. 2-7	Col. 8	Col. 73
L	TAFRA,		NU1
			NU2
			NU3
			NU4
			NU5
			NU6
			NU7
			NU8
			NU9
			NU10
			NU11
			NU12
			NU13
			NU14
			NU15
			NU16
			NU17
			NU18
			NU19
			NU20

650 059

PROGRAM 04-0

GAMMA RAY SCREENING CONSTANTS $p^{\tau}h(g)m$

Col. 1	Col. 2-7	Col. 8	Col. 73
L	TAFRA,		TAU1
			TAU2
			TAU3
			TAU4
			TAU5
			TAU6
			TAU7
			TAU8
			TAU9
			TAU10
			TAU11
			TAU12
			TAU13
			TAU14
			TAU15
			TAU16
			TAU17
			TAU18
			TAU19
			TAU20

PROGRAM 04-0

RECEIVER POINT COORDINATES

Col. 1	Col. 2-7	Col. 8 r_r	φ_r	z_r	
$	RECPT,	0 ,	1.5708 ,	1524 ,	
$	RECPT,	1524 ,	1.5708 ,	0 ,	
$	RECPT,	0 ,	1.5708 ,	−1524 ,	
$	RECPT,	,	,	,	
$	RECPT,	,	,	,	
$	RECPT,,	,	,	,	
$	RECPT,	,	,	,	
$	RECPT,	,	,	,	
$	RECPT,	,	,	,	
$	RECPT,	,	,	,	
$	RECPT,	,	,	,	
$	RECPT,	,	,	,	
$	RECPT,	,	,	,	
$	RECPT,	,	,	,	
$	RECPT,	,	,	,	
$	RECPT,	,	,	,	
$	RECPT,	,	,	,	

```
KDATES,2,FEB.  5,1958
KCHGNO,1,70050
*FIXPC,35,1040,17,4,4,6,1,5,4,4,4,1,1,1,0,1,1,0,                    FPC
LFLOPC,0,40,-1.5708,1.5708,-30,30,                                  FLPC1
LDELTA,9.5,
MMAXCD,35,1,
LMAXBD,140,120,
LDENSY,1,2.7,18.7,1.83,11.4,0,                                      DEL1
LCONSA,7.29/9,.29,.83,.58,1,                                        CONS1
LCSECT,.033,.079,.17,.132,.117,0,                                   SECT1
LCONRA,1,.0261,0,
LCONZA,150,.0349,0,
LCONBJ,7.7/10,3.4/10,1.5/10,6.4/9,2.8/9,                            BJ1
LCONKJ,1.47/-6,1.35/-6,1.25/-6,1.18/-6,1.11/-6,                     KJ1
LCONRB,150,.0261,0,
LCONZB,1,.0349,0,
LBETAS,1,.746,.016,-.00049,                                         BET11
       1,.633,.0048,-.00005,                                        BET12
       1,.544,-.0007,.00002,                                        BET13
       1,.487,-.005,.00011,                                         BET14
       1,.438,-.0066,.00013,                                        BET15
LSIGEJ,.0435,.103,.842,.0642,.477,0,                                S01
       .0361,.0886,.804,.0525,.465,0,                               S02
       .0319,.0788,.823,.0456,.477,0,                               S03
       .029,.0734,.86,.0408,.499,0,                                 S04
       .0268,.0699,.898,.0371,.516,0,                               S05
LREDES,1025,0,40,-30,30,                                            R01
       1026,0,50,30,40,                                             R02
       1026,40,50,-30,30,                                           R03
       1026,0,50,-40,-30,                                           R04
       1027,0,60,40,50,                                             R05
       2051,50,50,65,60,0,30,                                       R06
       2051,50,50,60,65,-30,0,                                      R07
       1027,0,60,-50,-40,                                           R08
       515,-80,80,-80,80,70,80,                                     R09
       515,-80,80,-80,80,-80,-70,                                   R10
       515,80,90,-90,90,-80,80,                                     R11
       515,-90,-80,-90,90,-80,80,                                   R12
       515,-80,80,80,90,-80,80,                                     R13
       515,-80,80,-90,-80,-80,80,                                   R14
       1028,0,140,100,120,                                          R15
       2052,140,120,140,140,80,100,                                 R16
       2052,120,140,140,140,-120,-80,                               R17
LPFRAC,.583,0,0,0,                                                  M01
       .415,0,0,0,                                                  M02
       .002,0,0,0,                                                  M03
       0,1,0,0,                                                     M04
       0,0,1,0,                                                     M05
       0,0,0,1,                                                     M06
$RECPT,0,1.5708,1524,
$RECPT,1524,1.5700,0,
$RECPT,0,1.5708,-1524,
```

15.3 Sample Problem Output

PROBLEM NUMBER 1040 W=3.5292, 7 FEB. 5,1958

1.0500, 7 3.3793, 5 1.1280, 4 5.9791, 5 2.3845, 7 0.0000, 0

FEB. 5,1958

OUR	PROBLEM NO.	R	PHI	Z	D
1.2229,-4	1.2229,-4	1.2229,-4	1.2229,-4		
1.2222,-4	1.2222,-4	1.2222,-4	1.2222,-4		
1.2200,-4	1.2200,-4	1.2200,-4	1.2200,-4		
1.2165,-4	1.2165,-4	1.2165,-4	1.2165,-4		
8.7877,-4	8.7877,-4	8.7877,-4	8.7877,-4		
8.7831,-4	8.7831,-4	8.7831,-4	8.7831,-4		
8.7690,-4	8.7690,-4	8.7690,-4	8.7690,-4		
8.7456,-4	8.7456,-4	8.7456,-4	8.7456,-4		
6.7534,-3	6.7534,-3	6.7534,-3	6.7534,-3		
6.7501,-3	6.7501,-3	6.7501,-3	6.7501,-3		
6.7403,-3	6.7403,-3	6.7403,-3	6.7403,-3		
6.7240,-3	6.7240,-3	6.7240,-3	6.7240,-3		
5.6685,-2	5.6685,-2	5.6685,-2	5.6685,-2		
5.6661,-2	5.6661,-2	5.6661,-2	5.6661,-2		
5.6588,-2	5.6588,-2	5.6588,-2	5.6588,-2		
5.6467,-2	5.6467,-2	5.6467,-2	5.6467,-2		
PRODUCT IS	1040	0.000	1.5708	1524.000	1.8373, 4
6.3165-13	6.3165-13	6.3165-13	6.3165-13		
6.3125-13	6.3125-13	6.3125-13	6.3125-13		
6.3005-13	6.3005-13	6.3005-13	6.3005-13		
6.2807-13	6.2807-13	6.2807-13	6.2807-13		
2.4027-12	2.4027-12	2.4027-12	2.4027-12		
2.4012-12	2.4012-12	2.4012-12	2.4012-12		
2.3969-12	2.3969-12	2.3969-12	2.3969-12		
2.3898-12	2.3898-12	2.3898-12	2.3898-12		
9.0533-12	9.0533-12	9.0533-12	9.0533-12		
9.0483-12	9.0483-12	9.0483-12	9.0483-12		
9.0332-12	9.0332-12	9.0332-12	9.0332-12		
9.0081-12	9.0081-12	9.0081-12	9.0081-12		
3.3741-11	3.3741-11	3.3741-11	3.3741-11		
3.3724-11	3.3724-11	3.3724-11	3.3724-11		
3.3672-11	3.3672-11	3.3672-11	3.3672-11		
3.3585-11	3.3585-11	3.3585-11	3.3585-11		
PROGRESS 1	1040	0.000	1.5708	1524.000	1.8827, 0
1.6867-12	1.6867-12	1.6867-12	1.6867-12		
1.6857-12	1.6857-12	1.6857-12	1.6857-12		
1.6828-12	1.6828-12	1.6828-12	1.6828-12		
1.6778-12	1.6778-12	1.6778-12	1.6778-12		
5.2387-12	5.2387-12	5.2387-12	5.2387-12		
5.2358-12	5.2358-12	5.2358-12	5.2358-12		
5.2271-12	5.2271-12	5.2271-12	5.2271-12		
5.2126-12	5.2126-12	5.2126-12	5.2126-12		
1.6174-11	1.6174-11	1.6174-11	1.6174-11		
1.6165-11	1.6165-11	1.6165-11	1.6165-11		
1.6140-11	1.6140-11	1.6140-11	1.6140-11		
1.6098-11	1.6098-11	1.6098-11	1.6098-11		

FEB. 5,1958

OUR	PROBLEM NO.	R	PHI	Z	D
4.9581-11	4.9581-11	4.9581-11	4.9581-11		
4.9557-11	4.9557-11	4.9557-11	4.9557-11		
4.9484-11	4.9484-11	4.9484-11	4.9484-11		
4.9363-11	4.9363-11	4.9363-11	4.9363-11		
PROGRESS 2	1040	0.000	1.5708	1524.000	1.2991,0
1.9289-12	1.9289-12	1.9289-12	1.9289-12		
1.9278-12	1.9278-12	1.9278-12	1.9278-12		
1.9244-12	1.9244-12	1.9244-12	1.9244-12		
1.9189-12	1.9189-12	1.9189-12	1.9189-12		
5.3380-12	5.3380-12	5.3380-12	5.3380-12		
5.3351-12	5.3351-12	5.3351-12	5.3351-12		
5.3264-12	5.3264-12	5.3264-12	5.3264-12		
5.3118-12	5.3118-12	5.3118-12	5.3118-12		
1.4708-11	1.4708-11	1.4708-11	1.4708-11		
1.4700-11	1.4700-11	1.4700-11	1.4700-11		
1.4677-11	1.4677-11	1.4677-11	1.4677-11		
1.4639-11	1.4639-11	1.4639-11	1.4639-11		
4.0314-11	4.0314-11	4.0314-11	4.0314-11		
4.0294-11	4.0294-11	4.0294-11	4.0294-11		
4.0234-11	4.0234-11	4.0234-11	4.0234-11		
4.0135-11	4.0135-11	4.0135-11	4.0135-11		
PROGRESS 3	1040	0.000	1.5708	1524.000	4.7364,-1
1.5439-12	1.5439-12	1.5439-12	1.5439-12		
1.5430-12	1.5430-12	1.5430-12	1.5430-12		
1.5403-12	1.5403-12	1.5403-12	1.5403-12		
1.5359-12	1.5359-12	1.5359-12	1.5359-12		
3.9962-12	3.9962-12	3.9962-12	3.9962-12		
3.9940-12	3.9940-12	3.9940-12	3.9940-12		
3.9873-12	3.9873-12	3.9873-12	3.9873-12		
3.9762-12	3.9762-12	3.9762-12	3.9762-12		
1.0311-11	1.0311-11	1.0311-11	1.0311-11		
1.0305-11	1.0305-11	1.0305-11	1.0305-11		
1.0289-11	1.0289-11	1.0289-11	1.0289-11		
1.0261-11	1.0261-11	1.0261-11	1.0261-11		
2.6502-11	2.6502-11	2.6502-11	2.6502-11		
2.6489-11	2.6489-11	2.6489-11	2.6489-11		
2.6448-11	2.6448-11	2.6448-11	2.6448-11		
2.6381-11	2.6381-11	2.6381-11	2.6381-11		
PROGRESS 4	1040	0.000	1.5708	1524.000	1.3273,-1
1.2682-12	1.2682-12	1.2682-12	1.2682-12		
1.2675-12	1.2675-12	1.2675-12	1.2675-12		
1.2653-12	1.2653-12	1.2653-12	1.2653-12		
1.2616-12	1.2616-12	1.2616-12	1.2616-12		
3.1359-12	3.1359-12	3.1359-12	3.1359-12		
3.1341-12	3.1341-12	3.1341-12	3.1341-12		
3.1288-12	3.1288-12	3.1288-12	3.1288-12		
3.1201-12	3.1201-12	3.1201-12	3.1201-12		
7.7340-12	7.7340-12	7.7340-12	7.7340-12		
7.7299-12	7.7299-12	7.7299-12	7.7299-12		
7.7173-12	7.7173-12	7.7173-12	7.7173-12		

FEB. 5,1958

OUR	PROBLEM NO.	R	PHI	Z	D
7.6965-12	7.6965-12	7.6965-12	7.6965-12		
1.9017-11	1.9017-11	1.9017-11	1.9017-11		
1.9007-11	1.9007-11	1.9007-11	1.9007-11		
1.8977-11	1.8977-11	1.8977-11	1.8977-11		
1.8928-11	1.8928-11	1.8928-11	1.8928-11		
PROGRESS 5	1040	0.000	1.5708	1524.000	4.0818,-2
PROGRESS T	1040	0.000	1.5708	1524.000	3.8290, 0

PRODUCT IS	2040	1524.000	1.5708	0.000	3.1621, 2
PROGRESS 1	2040	1524.000	1.5708	0.000	1.2327, -1
PROGRESS 2	2040	1524.000	1.5708	0.000	1.0953, -1
PROGRESS 3	2040	1524.000	1.5708	0.000	4.3579, -2
PROGRESS 4	2040	1524.000	1.5708	0.000	1.2583, -2
PROGRESS 5	2040	1524.000	1.5708	0.000	3.9602, -3
PROGRESS T	2040	1524.000	1.5708	0.000	2.9292, -1
PRODUCT IS	2040	0.000	1.5708	-1524.000	1.7369, 3
PROGRESS 1	2040	0.000	1.5708	-1524.000	8.3999, -1
PROGRESS 2	2040	0.000	1.5708	-1524.000	6.6668, -1
PROGRESS 3	2040	0.000	1.5708	-1524.000	2.6157, -1
PROGRESS 4	2040	0.000	1.5708	-1524.000	7.7062, -2
PROGRESS 5	2040	0.000	1.5708	-1524.000	2.4624, -2
PROGRESS T	2040	0.000	1.5708	-1524.000	1.8699, 0

15.4 Bibliography

1. WAPD-15, "A Simplified Theory of Neutron Attenuation and Its Application to Reactor Shield Design", R. D. Albert and T. A. Welton, November 30, 1950, Declassified

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