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TITLE

Comparison of Calculated and Measured Gamma-Ray Dose Rates and
Neutron Flux Distributions in the SRE Instrument Thimbles

By

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August 8, 1960

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Abstract

An attempt is made to predict the thermal neutron flux and gamma-ray dose rate in the various instrument thimbles of the SRE. The mathematical tools used are the Wanda III Code, the GRACE I Code, and neutron and gamma-ray streaming theory. Basic cross sections and other physical data used in the calculations are listed. The cases examined include the effect of cutouts in the thermal shield on the neutron flux and gamma-ray dose rate at the bottom and up through the instrument thimbles. Agreement with the results of preliminary measurements is found to be reasonable. Reasons for the differences between calculations and measurements are discussed.

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Nomenclature

S_{nv} = neutron source strength, neut/cc-sec.

$S_{\gamma i}$ = gamma-ray source strength for the i-th energy group,
Mev/cc-sec.

Σ_j = removal cross section for material in j-th shielding
region, cm^{-1} .

Σ_{ri} = removal cross section for the i-th material, cm^{-1} .

$\Sigma_{r,s}$ = removal cross section for source material, cm^{-1} .

Σ_{ai} = neutron absorption cross section for the i-th group.

$\Sigma_{s,i}$ = neutron slowing down cross section for the i-th
diffusion group, cm^{-1} .

D_i = diffusion coefficient for the i-th diffusion group, cm^2 .

ϕ_f = fast neutron flux, neut/ cm^2 -sec.

ϕ_{th} = thermal neutron flux, neut/ cm^2 -sec.

x_j = thickness of j-th shielding region, cm.

x_s = thickness of source region, cm.

P = reactor thermal power level, 2×10^7 watts.

c = a conversion factor, 3.10×10^{10} fissions/watt-sec.

ν = number of neutrons produced per thermal fission,
2.47 neut/fission.

V = volume of SRE core, $5.34 \times 10^6 \text{ cm}^3$.

$$E_2(b) = b^{-1} \int_b^{\infty} \frac{e^{-t}}{t^2} dt$$

D_γ = gamma-ray dose rate, r/hr.

a = radius of instrument thimble, in.

L = length of instrument thimble, in.

Nomenclature (Cont'd)

p = occurrence probability for a particular reaction

$\sum_{ath}^r (k)$ = thermal neutron absorption cross section for k-th material in the reflector, cm^{-1} .

$\sum_{ath}^r$ = total thermal neutron absorption cross section of reflector, cm^{-1} .

$E_{t,i}$ = total energy release in the i-th photon group of a material, Mev.

N_{th} = number of thermal neutrons captured in the material per fission.

N_f = number of fast neutrons captured in the material per fission.

BE = neutron binding energy of the material, Mev.

F = number of fissions in core per 100 neutrons born.

I Introduction

The purpose of this work was to determine the adequacy of the calculational techniques employed in determining the shielding requirements for sodium graphite reactors. Recently completed preliminary measurements (1,2,3) of the gamma-ray dose rate and thermal neutron flux in the instrument thimbles of the Sodium Reactor Experiment (SRE) permit a comparison between the predicted and the actual levels. From this comparison, the accuracy of the calculational techniques can be determined and possibly some insight can be gained into the limitations of the methods utilized.

At Atomics International, during the process of designing a reactor, the neutron flux distribution in the reactor shield is currently obtained through application of a combination of removal theory and diffusion theory. Removal theory is used to predict the fast neutron flux distribution while diffusion theory is combined with these results to yield the thermal neutron flux distribution.

Removal theory is a semi-empirical (phenomenological) theory developed to fit measurements in hydrogenous reactor shield systems, i.e., systems where the shield contains a significant amount of water.* The validity of application of this theory to non-hydrogenous systems has not been verified. The SRE is a sodium cooled and graphite moderated reactor. Its shield is composed of steel (thermal shields and tanks) and concrete. With the exception of the concrete, it is a non-hydrogenous system.

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* The minimum amount of water necessary for the theory to be valid has been the subject of much conjecture - no experimental evidence has been prepared from which the lower limit of hydrogen content can be determined.

The prediction of gamma-ray dose rates throughout the shield is theoretically a well defined problem. In general, the gamma-ray distribution in the shield is dependent on the thermal neutron flux in the shield and consequently is closely coupled to the neutron flux calculations.

In this report, neutron fluxes and gamma-ray dose rates are calculated for simplified geometries representing the SRE instrument thimbles by using existing mathematical tools. An analysis for the exact geometry appears complex and unwarranted at this time, at least considering the rather preliminary nature of the experimental data. Consequently, only a qualitative comparison between the results of the experiments and calculations is possible. However, this should be sufficient to determine the validity of existing mathematical tools when applied to reactor systems such as the SRE.

II Description of the SRE Radial Shield

An elevation view of the SRE core and its radial shield is given in Figure 1. The geometry of the core is approximately that of a right circular cylinder 6 ft high and 6 ft in diameter. It is surrounded by a reflector approximately 2 ft thick and composed of graphite canned in either zirconium or stainless steel. The reflector is surrounded by a $\frac{1}{4}$ in. thick liner of stainless steel called the core liner. The core liner and reflector are not in contact, the space between them being filled with sodium. Surrounding the core liner is a stagnant sodium blanket, $2\frac{1}{2}$ in. thick, on the outside of which is a $1\frac{1}{2}$ in. thick core tank of type 304 stainless steel. A $\frac{1}{4}$ in. thick outer tank, $12\frac{1}{2}$ ft in diameter, called the outer liner in Figure 1, contains a 2 in. helium gap and the thermal shield which is $5\frac{1}{2}$ in. thick. Both the outer liner and thermal shield are made of carbon steel. The outer liner is surrounded by about one foot of insulation blocks, consisting of calcined diatomaceous silica and asbestos, and is coated with silicate to decrease dusting. The insulation blocks are contained in the cavity liner, a $\frac{1}{4}$ in. thick carbon steel tank. Extending radially outward from the cavity liner is a final 3 ft thick shield of ordinary concrete.

There are 8 instrument thimbles in the SRE. Each instrument thimble starts from the ground floor of the Reactor Building, curves downward and then penetrates the concrete shield and cavity liner at an angle of 30° . The bottom of each instrument thimble faces the thermal shield at the mid-height of the core and is capped by a $\frac{1}{4}$ in. thick steel plate.

Instrument thimble numbers 2 and 6 face a full cut-out ($5\frac{1}{2}$ in.) in the thermal shield, i.e., the bottom of the thimble faces a 1 foot diameter hole which has been cut completely through the thermal shield. Instrument thimble numbers 4 and 8 face a half cut-out ($2\frac{3}{4}$ in.) in the thermal shield. The other instrument thimbles face no cut-out. An elevation view of a typical instrument thimble is shown in Figure 2. The horizontal layout of all instrument thimbles, indicating their individual functions, is shown in Figure 3. Figure 3a shows the layout of the individual instrument thimbles with respect to the core loading.

Figure 1
Cut-view of SRE Radial Shield

Vertical cut-view of an SRE Instrument Thimble

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Figure 3
Horizontal Lay-out of SRE Instrument Thimbles

Figure 3a

Detail Layout of SRE Core

III Calculation of Neutron Flux Distribution

A. Fast Flux

The fast neutron flux was calculated by removal theory using the GRACE I Code.⁽⁴⁾ Since this code solves problems in truncated cone and infinite slab geometry only, each region was replaced by an infinite slab with a thickness equal to the actual thickness in the cylindrical case. The removal flux for this geometry has the form

$$\phi_f = \frac{S_{nv}}{2 \sum_{r,s}} \left[E_2 \left(\sum_i \Sigma_{ri} X_i \right) - E_2 \left(\sum_i \Sigma_{ri} X_i + \sum_{r,s} X_s \right) \right] \dots (1)$$

The microscopic removal cross sections were obtained using removal cross sections published in the shielding manual,⁽⁶⁾ known material composition and density. Part of this data is reproduced in Table I.

The neutron source strength was obtained by arbitrarily considering only neutrons born above 0.5 Mev and by assuming that the slab representing the core had the same average power density as the actual core. Under this assumption, the neutron source strength is

$$S_{nv} = 0.8531 \frac{\nu c P}{V} \dots (2)$$

where 0.8531 is the fraction of neutrons born with energy above 0.5 Mev. The fast neutron flux was calculated in this manner for two reasons:

- 1) It was assumed that, in the shield of the SRE, the attenuation of the fast flux is governed by removal theory.

- 2) To obtain fast group parameters for the Wanda III Code, (11) which was used to compute the thermal neutron flux. The procedure for obtaining these parameters is explained in the following section.

Table I

Removal Cross Sections and Operating Temperature
of the Various Regions in the SRE

Region	$\Sigma_r (\text{cm}^{-1})$	T(°F)
Core	0.0667	800
Reflector	0.0656	750
Sodium	0.0336	750
Core liner	0.168	750
Core tank	0.168	750
Thermal shield and outer liner	0.168	700
Cavity liner	0.168	700
Concrete	0.0768	150

B. Thermal Flux

The Wanda III few group diffusion code⁽¹¹⁾ with a four group breakdown was used to obtain the thermal neutron flux distribution in the radial shield. The energy breakdown used was

Group I	10 Mev - 0.5 Mev
Group II	0.5 Mev - 1.23 kev
Group III	1.23 kev - 0.414 ev
Group IV	0.414 ev - thermal

The usual procedure for obtaining group constants for the Wanda Code is to use the MUFT-IV Code⁽¹²⁾, the I-2 Code⁽¹³⁾ and the SOFOCATE Code.⁽¹⁶⁾ However, in shield regions and at distances far removed from the source regions, MUFT-IV predicts constants for Group I which result in a fast neutron flux distribution which is different from that obtained by removal theory. Consequently, the Group I constants obtained from MUFT-IV were constrained to give results similar to removal theory. The manner by which MUFT IV constants were constrained to give a fast flux identical to that obtained through the use of removal theory is discussed below.

In diffusion theory, the differential equation expressing the neutron balance condition for fast neutrons in the presence of no source is

$$D_1 \nabla^2 \phi + (\Sigma_{a1} + \Sigma_{s1}) \phi = 0 \quad \dots (3)$$

In slab geometry, the solution of equation (3) is of the form

$$\phi_1 = A \exp - \left\{ \frac{1}{D_1} (\Sigma_{a1} + \Sigma_{s1}) x \right\}^{\frac{1}{2}} \quad \dots (4)$$

Removal theory predicts a fast flux which is expressed by equation (1). For $\sum_1 \sum_1 x_1 > 3$, equation (1) can be fitted over short intervals by exponential functions, i.e.,

$$\phi = \phi_0 e^{-KX} \quad \dots (5)$$

Equation (5) satisfies equation (4) provided that

$$K^2 D_1 - \sum_{a1} = \sum_{s21} \quad \dots (6)$$

The continuity of neutron current at the interface of two shielding regions requires that

$$D_1 \nabla \phi_1 \text{ 1st region} = D_1 \nabla \phi_1 \text{ 2nd region} \quad \dots (7)$$

If the fast neutron flux is represented by equation (5), equation (7) then implies

$$D_1 K \text{ 1st region} = D_1 K \text{ 2nd region} = \text{constant} \quad \dots (8)$$

The fast group constants were then calculated as follows; K was calculated using equations (1) and (5), D_1 and $\sum_{a1}$ were calculated using the MUFT IV Code, KD_1 was then calculated for each region and turned out to be approximately the same for all materials. An average value of

$$KD_1 = 0.212 \quad \dots (9)$$

was then adopted to satisfy equation (8). Equation (9) was then used to calculate a modified diffusion coefficient leaving K for each region unchanged. Finally, $\sum_{s21}$ was calculated using equation (6)

together with the modified diffusion coefficient, Σ_{a1} , from MUFT IV, and K as obtained from equations (1) and (5). The group constants so obtained, when used in equation (3), give a fast flux that is equivalent to the one expressed by equation (1).

Below 0.5 Mev to approximately thermal energy, the neutron flux has an energy dependence of $1/E$. This is also demonstrated by the MUFT IV Code so that MUFT IV group constants were used for groups II and III. The thermal or group IV constants were obtained using microscopic thermal absorption cross sections taken from BNL-325,⁽¹⁴⁾ known material density and composition, and were averaged for a Maxwellian distribution. The four group constants so obtained are reproduced in Table II.

Having obtained the necessary cross-section data, the calculations could then be performed. Three slab geometry Wanda flux calculations were run. The first Wanda calculation was for the shield configuration shown in Figure 1. This case represents a simplified no cut-out geometry. The second Wanda calculation was for a similar shield configuration but with the thermal shield having only half its actual thickness. In view of the large diameter of the cut-out (1 foot in diameter), it was believed that this is the type of shield a thermal neutron flux detector would see when located at the bottom of an instrument thimble which faces a half cut-out in the thermal shield. The third Wanda calculation was run for a configuration with no thermal shield. Here also it was assumed that this is the type of shield a thermal neutron flux detector would see when located at the bottom of an instrument thimble which faces a full cut-out.

In these calculations, the reflector had to be broken up into three regions for the following reasons:

- 1) The inner region of the reflector is made of graphite canned in zirconium while the outer reflector region is made of graphite canned in stainless steel. On the average, the thickness of the zirconium canned reflector region is 39.1 cm while that of the steel canned reflector region is 18.0 cm.
- 2) Close to the source, equation (1) can only be represented by equation (5) over very short intervals. Consequently, the zirconium canned reflector region was broken up into two separate regions in order to afford a better fit.

The thermal fluxes obtained from the Wanda calculations are plotted in Figure 4.

Table II
Four Group Constants for the SRE and Its Radial Shield Regions

Region	Σ_{g1}	Σ_{a1}	D_1	Σ_{g2}	Σ_{a2}	D_2
Core	0.018534	0.000587	2.1806	0.00863	0.000727	0.9826
1st Reflector *	0.02300	10^{-6}	1.9400	0.00899	0.0000750	1.0117
2nd Reflector	0.0172	10^{-6}	2.6000	0.00899	0.0000750	1.0117
3rd Reflector	0.0172	5×10^{-6}	2.6000	0.00942	0.0000520	0.981
1st Sodium	0.00866	8×10^{-6}	5.1600	0.001618	0.000272	3.638
Core Liner	0.0415	0.000513	1.0660	0.001965	0.001276	0.9475
2nd Sodium	0.00864	8×10^{-6}	5.1700	0.001618	0.000272	3.638
Core Tank	0.0408	0.000513	1.0820	0.001965	0.001276	0.9475
Thermal Shield	0.0397	0.000513	1.1125	0.001965	0.001276	0.9475
Cavity Liner	0.0397	0.000513	1.1125	0.001965	0.001276	0.9475
Concrete	0.01852	0.000931	2.3000	0.022138	0.0000560	1.177

*In calculating fluxes the reflector
was broken down into three regions.

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Table II (Cont'd)

Region	Σ_{s23}	Σ_{a3}	D_3	Σ_{a4}	D_4
Core	0.005589	0.002978	0.9665	0.005981	0.89445
1st Reflector	0.007072	0.0000920	1.000	0.0011538	0.93032
2nd Reflector	0.007072	0.0000920	1.000	0.0011538	0.93032
3rd Reflector	0.007505	0.000193	0.9335	0.002610	0.8683
1st Sodium	0.000456	0.000481	4.980	0.00660	3.710
Core Liner	0.000611	0.006255	0.35136	0.1490	0.3090
2nd Sodium	0.000456	0.000481	4.980	0.0066	3.710
Core Tank	0.000611	0.006255	0.35136	0.1490	0.3090
Thermal Shield	0.000611	0.006255	0.35136	0.1282	0.3450
Cavity Liner	0.000611	0.006255	0.35136	0.1282	0.3450
Concrete	0.023966	0.000547	1.240	0.00756	0.3800

SRE Radial Flux Distribution

Power Level = 20 Mw

SEMI-LOGARITHMIC 359-S1
KEUFFEL & ESSER CO. MADE IN U.S.A.
5 CYCLES A TO DIVISIONS

Distance from Core-Reflector Interface, cm

9/10/59 DSD

SRE Radial Flux Distribution

359-91
SEMI-LOGARITHMIC
KEUFFEL & ESSER CO. MADE IN U.S.A.
5 CYCLES X 70 DIVISIONS

013

012

011

010

009

008

Distance from Core-Reflector Interface, in

9/10/59 PBD

C. Thermal Flux Distribution in the Thimble

Since thermal neutron flux distribution measurements were carried out over the entire length of the instrument thimbles, an attempt was made to correlate these results with theoretical calculations.

The thermal neutron flux in void ducts is determined by:

- 1) leakage of thermal neutrons from the surrounding material into the duct. (This problem is complex and no simple analytical method has been developed to evaluate its contribution.)
- 2) streaming of neutrons down the duct. (This problem has been solved in terms of simple analytical expressions. An excellent treatment can be found in Price, Horton and Spinney;⁽⁷⁾ their method is summarized in the following paragraph.)

Consider a duct with a geometry as shown in Figure 5. If the walls of the duct are non-reflecting and the source has a cosine distribution, then the ratio of the flux at P to that at the entrance is given by

$$\phi(Z)/\phi(0) = \left[1 - \left\{ 1 + \left(\frac{a}{Z} \right)^2 \right\}^{-1/2} \right] \quad \dots (10)$$

The effects of scattering from the wall of the duct may be evaluated in terms of an albedo β . If the angular distribution of the neutrons scattered back into the duct is isotropic, then the flux is given by

$$\phi_{\text{isot}} = \phi(Z) \left[\frac{\beta}{1-\beta} + 1 \right] \quad \dots (11)$$

where $\phi(Z)$ is given by equation (10). If the scattered neutrons have a distribution varying as the cosine of the angle from the normal to the duct walls, and provided z is not too small, then the flux is given by

$$\phi_{\cos} = \phi(z) \left[1 + 4 \alpha \beta / z (1 - \beta) \right] \quad \dots (12)$$

In the general case where the scattering is a combination of isotropic and cosine, the flux is then given by

$$\phi_{\text{gen}} = \beta \phi_{\text{isot}}(z) + (1 - \beta) \phi_{\cos}(z) \quad \dots (13)$$

where β is a constant determining the degree of isotropic scattering. The application of these formulas towards interpreting the experimental data is explained in Section V-A.

Figure 5
Duct Geometry for neutron Streaming.

IV Gamma-Ray Dose Rate Calculations

A. Gamma-Ray Spectra

The following sources of gamma radiation were considered in calculating the gamma-ray dose rate at a location corresponding to the bottom of the instrument thimble during operation:

- 1) prompt fission gamma-rays, fission product gamma-rays and capture gamma-rays from the core,
- 2) capture gamma-rays from the reflector, sodium regions, core liner, core tank, thermal shield, outer liner and the ordinary concrete side shield.

During operation, the intensity of decay gamma-rays from sodium, steel, U^{239} and concrete is negligible compared to that from prompt gammas and therefore was not considered. The capture gamma-ray spectra for the individual materials encountered in the SRE and its radial shield were obtained from the GRACE I Code library.⁽⁸⁾ The data in the library are based, for the most part, on the work of Bartholomew and Higgs.⁽¹⁵⁾ These spectra are reproduced in Table III.

Table III
Capture Gamma-Ray Spectra of Individual Elements Occurring in
SRE Reactor and Shield (Mev/cap)

E	U^{235} (n, f)	U^{235} (n, γ)	U^{235} f, p	U^{238} (n, γ)	304 S.S (n, γ)	Na^{23} (n, γ)	C^{12} (n, γ)	Zr (n, γ)	Carbon Steel (n, γ)	Concrete (n, γ)
0 - 1	3.060	0.495	1.960	1.90	0.17	0.80	---	0.21	0.06	0.12
1 - 2	2.490	1.512	2.770	1.60	0.34	0.39	0.40	0.50	0.42	0.58
2 - 3	1.285	2.406	1.440	0.88	0.28	1.61	---	0.50	0.28	1.27
3 - 4	0.590	1.512	0.608	0.14	0.43	2.06	1.10	2.52	0.56	0.96
4 - 5	0.258	0.495	0.229	0.28	0.53	0.30	3.46	1.71	0.67	1.51
5 - 6	0.157	---	---	---	0.80	0.39	---	1.01	0.77	0.74
6 - 7	0.061	---	---	---	0.84	1.41	---	1.15	0.73	0.74
7 - 8	---	---	---	---	3.00	---	---	0.15	3.90	0.71
8 - 9	---	---	---	---	1.40	---	---	0.11	0.06	0.04
9 - 10	---	---	---	---	0.57	---	---	---	0.35	0.03
	7.901	6.420	7.007	4.80	8.36	6.96	5.00	7.86	7.80	6.70

Table IV
Neutron Economy for the SRE Core

		<u>Absorption</u>	<u>Fission Neutrons Produced</u>
Given 100.00 fission neutrons	100.000		
Fast leakage	12.001		
Fast capture in Zr		0.554	
Fast capture in graphite		0.097	
Fast capture in stainless steel		0.780	
Fast capture in Na		1.040	
Fast (n, γ) captures in U^{235}		4.743	
Fission capture in U^{235}		11.360	
Fission neutrons produced by U^{235}			27.980
Fast capture in U^{238}		16.854	
Fission capture in U^{238}		1.787	
Fission neutrons produced by U^{238}			4.745
Total fast absorptions		37.215	
Removed to thermal	50.784		
Thermal leakage	3.247		
Thermal capture in Zr		0.623	
Thermal capture in graphite		1.740	
Thermal capture in stainless steel		1.891	
Thermal capture in sodium		4.028	
Thermal capture in U^{238}		4.809	
(n, γ) captures in U^{235}		5.263	
Fission capture in U^{235}		27.238	
Fission neutron produced by U^{235}			67.275
Thermal capture in control rods, Xe poison ring		1.946	
Total thermal absorption		47.5370	
Total fission neutrons produced			100.000

1. Core Gamma-Ray Spectrum

The core gamma-ray spectrum is composed of the following spectra:

$U^{235}(n, \gamma)$, (n, f) and (f, p) , $U^{238}(n, \gamma)$, $Na^{23}(n, \gamma)$, $C(n, \gamma)$, $Zr(n, \gamma)$ and stainless steel (n, γ) . Decay spectra of Na^{24} , U^{239} and stainless steel should also be included but since their contribution is negligible when compared to that from prompt gammas they were neglected. The spectra associated with U^{235} fission, radiative capture, and fission products were obtained from the work of Roos.⁽¹⁰⁾ These spectra, arranged in 1 Mev increments, are shown in Table III.

The gamma-ray spectra for the other reactions were obtained from the GRACE Code Library and are also reproduced in Table III of this report.

The core gamma-ray spectrum was obtained by using the results of a neutron economy calculation, shown in Table IV.⁽⁹⁾ In order to account for the additional energy associated with fast neutron capture (as opposed to thermal neutron capture), it was assumed that the average fast neutron energy at capture was 1.0 Mev; this energy was then added to the binding energy. Thus, the energy released in a given photon group of a certain material is given by

$$E_{t,i} = N_{th} E_i + N_f E_i \left(\frac{1 + BE}{BE} \right) \quad \dots (14)$$

In the use of this equation it is implicitly assumed that fast and thermal neutron absorption reactions yield the same capture gamma-ray spectra. The resulting core gamma-ray spectrum in units of Mev/neutron born is shown in Table V. Finally, the core gamma-ray source strength

was obtained by multiplying the spectrum as given in Table V by the average number of neutrons born per unit core volume, or

$$S_i = \frac{100 \text{ c P}}{F V} E_i \quad \dots (15)$$

Table VCore Gamma-Ray Spectrum of the SRE

Energy Range (Mev)	E_{γ} (Mev/neutron born)
0 - 1	2.60
1 - 2	2.73
2 - 3	1.68
3 - 4	.85
4 - 5	.44
5 - 6	.12
6 - 7	.14
7 - 8	.09
8 - 9	.04
9 - 10	.02
Total	8.71 (=21.6 Mev/fission)

2. Capture Gamma-Ray Spectra from the Reflector

The capture gamma-ray spectra of the reflector regions is composed of the spectra from the following reactions: $C(n, \gamma)$, $Na(n, \gamma)$, $Zr(n, \gamma)$ and stainless steel (n, γ) . All the spectra for these reactions were found in the GRACE Code Library and are reproduced in Table III. Decay spectra for Na^{24} and stainless steel should also be included but, as in the case for the core, they were neglected since their contribution is negligible compared to that from capture gamma-rays.

There are two spectra associated with the reflector, since, in the model used, the reflector is divided into two distinct source regions. The inner region is assumed to consist of graphite canned in zirconium and to be 39.1 cm thick, whereas the outer region is assumed to consist of graphite canned in stainless steel and to be 18 cm thick. In addition, both regions contain sodium coolant.

The gamma-ray spectrum for either reflector region was then obtained in a manner similar to the procedure used in determining the core spectrum, i.e., a summation over each reaction spectrum multiplied by its occurrence probability. Since a neutron economy study was not available, the occurrence probability for a particular reaction was determined by

$$P = \frac{\sum_{ath}^r (k)}{\sum_{ath}^r} \quad \dots (16)$$

The pertinent data used in evaluating equation (16) is given in Table VI.

Table VIThermal Neutron Absorption Cross Section for the Reflector Regions

Material	$\sum_{ath}^r (k)$ for Inner Reflector	$\sum_{ath}^r (k)$ for Outer Reflector
C	0.000209	0.000224
Na	0.000696	0.000310
Zr	0.000073	----
S.S	<u>0.000172</u>	<u>0.002110</u>
Total	0.001150	0.002644

The gamma-ray spectra obtained for the two reflector regions are in units of Mev/neutron captured, and are reproduced in Table VII. The volume gamma-ray source strength for either region is obtained by multiplying the spectra in Table VII by the neutron capture rate, or

$$S_{\gamma} = \sum_{ath}^r \phi_{lh} E_{\gamma i} \quad \dots (17)$$

Table VII
Capture Gamma-Ray Spectra for the Reflector Regions

Energy Range Mev	Inner Reflector Mev/cap	Outer Reflector Mev/cap
0 - 1	0.52	0.23
1 - 2	0.38	0.35
2 - 3	1.05	0.40
3 - 4	1.67	0.68
4 - 5	1.00	0.75
5 - 6	0.41	0.68
6 - 7	1.05	0.84
7 - 8	0.46	2.39
8 - 9	0.22	1.12
9 - 10	0.08	0.46
	Total	Total
	6.84	7.90

3. Capture Gamma-Ray Spectra from Sodium Regions, Core Liner, Core Tank, Thermal Shield, Cavity Liner and Concrete Shield

Unlike the core and reflector regions, which are composites, these regions are made up of materials for which capture gamma-ray spectra are well known and are also in the GRACE Code Library. The spectra for these regions are reproduced in Table III. It should be noted that the core liner and core tank are made of stainless steel while the thermal shield and cavity liner are made of carbon steel. The gamma-ray source strength for a particular region is obtained by multiplying the gamma-ray spectrum for the material encountered in that region by the neutron capture rate, or

$$S_{\gamma} = \sum_{ath} \phi_{th} E_{\gamma i} \quad \dots (18)$$

B. Dose Rate Calculation

The dose rate during reactor operation at a position corresponding to the bottom of an instrument thimble was obtained using the GRACE I Code.⁽⁴⁾ As mentioned earlier, this code is written for slab and truncated cone geometry; its description has been published elsewhere and therefore will not be discussed in this report.

In the no cut-out case, the dose rate was obtained using infinite slab geometry, i.e., each region was represented by an infinite slab of thickness equal to the actual thickness in the cylindrical case. The thermal neutron fluxes used were those obtained for slab geometry (shown earlier in Figure 4).

For the half cut-out case, the dose rate was calculated by a scheme illustrated in Figure 6. Briefly the scheme involves the following operations:

- (a) Calculate the dose rate for the no cut-out geometry (D_a).
- (b) Representing each region by a truncated cone as shown in Figure 6b, calculate the dose rate due to this arrangement (D_b) using the calculated thermal neutron flux distribution obtained for a no cut-out geometry.
- (c) Calculate the dose rate for an arrangement as shown in Figure 6c (D_c), again using the calculated thermal neutron flux distribution as obtained for a half cut-out geometry.

The dose rate for the cut-out geometry was then obtained by

$$D_{\gamma \text{ cut-out}} = D_{\gamma a} - D_{\gamma b} + D_{\gamma c} \quad \dots (19)$$

In the calculations, the buildup factor for iron was used for all source regions inside the cavity liner. The buildup factor for ordinary concrete was used only for capture gamma-rays originating in the concrete shield.

Since neutron flux measurements indicated that the cut-out has little effect on the magnitude of the thermal flux in the concrete (see Table IX), the no cut-out thermal neutron fluxes were used in all cases for calculating the contribution to the dose rate due to capture gammas produced in the concrete. In doing this, it was assumed that the majority of the concrete capture gammas arriving at the detector originate from regions outside the cut-out area. The results of the calculations are summarized in Table VIII.

Scheme for Calculating the Dose Rate of
a half Cut-out Geometry

Table VIIIGamma-Ray Dose Rate at Cavity Liner During Reactor Operation at 20 Mw

Source Region	Gamma-Ray Dose Rate, r/hr	
	No Cut-Out Case	Half Cut-Out Case
Core	950	4680
Reflector	3360	18110
1st Na Region	400	2770
Core Liner	1900	11570
2nd Na Region	460	3440
Core Tank	4200	29560
Thermal Shield and Cavity Liner	19870	46920
Concrete Shield	<u>9250</u>	<u>9250</u>
Total	40390	126300

C. Gamma-Ray Dose Rate Inside the Thimble

As in the case with the thermal neutron fluxes, gamma-ray dose rate measurements were also performed along the instrument thimbles. An attempt was also made to correlate these measurements with the results of theoretical calculations.

The gamma-ray dose rate inside the thimble is made up of two components.

- 1) Leakage of gamma radiation from the surrounding shield into the thimble. The contribution from the first component should approximately equal the dose rate in the concrete since, over relatively short distances, the propagation of gamma-rays is governed by ray theory rather than diffusion theory. Consequently, the gamma-ray dose rate inside the concrete region for a no cut-out configuration was calculated and its magnitude was assumed to be equal to the dose rate in the thimble due to leakage from the surrounding shield. The results of this calculation are plotted in Figures 9 and 10.

- 2) Streaming of gamma-rays along the thimble.

An analysis of this component can be found in the Shielding Manual.⁽⁶⁾ Briefly, assuming a cosine distribution for the source, this component varies as

$$D_{\gamma} = D_{\gamma_0} \frac{a^2}{L^2}, \text{ for } a \ll L \quad \dots (20)$$

Equation (20) was evaluated using values of 3.12×10^4 and 1.17×10^5 for D_0 . These values are used, rather than the totals as given in Table VIII, so as not to include the contribution from the concrete shield, which is actually accounted for in the direct beam (GRACE I) calculation. The results are plotted in Figures 9 and 10.

V Comparison of Calculations and Measurements

A. Neutron Fluxes

Values of the thermal neutron flux as measured at the bottom of some of the instrument thimbles, and as calculated to exist at the cavity liner-concrete interface by the method outlined in this report, are listed in Table IX.

Comparing the two sets of values, it can be seen that measured and calculated data differ by a factor of 6 for the no cut-out configuration and by a factor of 10 for the half cut-out configuration. It might be well to mention that thermal neutron fluxes calculated by diffusion theory generally peak higher in hydrogenous media than is actually the case. The large discrepancy between the two sets of data is believed to be due, in addition to the above, to the following reasons:

- (1) The measurements were carried out in a void channel where streaming effects are important. Also, due to the angle which the thimble makes, the measuring device cannot be considered as completely backed up by concrete. Consequently, it is expected that the neutron flux as seen by the measuring device when located at the bottom of a channel is not dependent on back scattering from the concrete. The calculations were performed for a detector positioned at the core liner-concrete interface where there is no void channel and hence backscattering from the concrete would be important. Also, it can be seen in Figure 4 that the calculated thermal neutron flux has a peak in the concrete region. Measurements carried out within the thimble, and shown in Figures 7 and 8, do not show this peak. This offers some support for the contention that the back-scattering flux into the channel constitutes a negligible source of neutrons.

Table IX

Comparison of Measured and Calculated Thermal Neutron
Fluxes at the Bottom of Instrument Thimbles in the SRE

Location	Distance from Bottom of Thimble (inches)	Thermal Neutron Flux, n/cm ² -sec	
		Measured	Calculated
No Cut-Out	0	---	8.60×10^9
Thimble #7	1.5	$1.49 \times 10^9(1)$	---
	4.5	$1.13 \times 10^9(1)$	---
Thimble #5	0	$1.19 \times 10^9(3)$	---
Half Cut-Out	0	---	1.85×10^{10}
Thimble #8	1.5	$9.80 \times 10^8(1)$	---
	4.5	$8.20 \times 10^8(1)$	---
Thimble #8	0	$2.20 \times 10^9(2)$	---
	6	$1.30 \times 10^9(2)$	---

- (2) In calculating the thermal neutron flux for the half cut-out case, it was assumed that the detector faces a cut-out of infinite diameter. Actually, the cut-out is only 12 inches in diameter.
- (3) Removal theory, which was used to obtain the fast neutron distribution in the shield regions, does not apply at the detector location due to the complete absence of hydrogen between the neutron source and the bottom of the instrument thimble.

The assumption of replacing the cylindrical geometry of the core and shield by a configuration of infinite slabs should also account for part of the difference. Furthermore, Figure 3a indicates that each instrument thimble faces a different fuel configuration and reflector thickness. This could well account for some of the lack of difference between the measured cut-out and no cut-out fluxes since, as can be seen from Table IX, fluxes measured by Reference 1 in the no cut-out configuration are higher than those in the half cut-out configuration. According to the calculations, the half cut-out fluxes should be twice as large as the no cut-out fluxes.

The thermal neutron fluxes as measured along the instrument thimble are shown in Figures 7 and 8. Also shown, for comparison purposes, is the streaming flux as expressed by equation 12 using an albedo value of 0.55⁽⁷⁾ and a $\phi(0)$ value of 8.6×10^9 and 1.85×10^{10} for the no cut-out and half cut-out cases, respectively. The measurements were also compared with the flux distributions as given by equations 11 and 13, but these were found to give results which were not significantly different from the fluxes as given by equation 12 and, therefore, were omitted.

Figures 7 and 8 indicate that the slopes of the calculated and measured fluxes do not agree at distances up to about 30 inches from the bottom of the thimble. Over these distances, it can therefore be assumed that leakage from the concrete into the channel is important. This in part substantiates the magnitude of the calculated thermal neutron fluxes in the concrete which are higher than those calculated by use of streaming theory. At distances beyond about 40 inches from the bottom of the channel, the slope of the results of Reference 2 agrees quite well with that given by equation (12). The results of Reference 1 can also be brought into agreement with equation (12) provided the curve representing the latter is shifted to the right and downward. This is equivalent to the assumption that the flux in the region of the thimble is made up of neutrons that first leak from the concrete into the duct and then stream along the duct.

Of the two sets of measurements, Reference 1 should be considered more reliable. The data was obtained by gold foil activation techniques. The data of Reference 2 was obtained with a compensated ion chamber.

ϕ_{th} , neut/cm²-sec

Thermal neutron flux in Thimble #7
 Reactor Power 20 megawatt.

Measurements obtained by
 Reference 1

Solid line represents ϕ_{th}
 as expressed by equation by
 equation 12 assuming $\phi_{th} = 6.2 \times 10^9$

Measurements performed on March 26, 1959

Measurements performed
 March 27, 1959

Distance from Bottom of Thimble, inches.

Figure 8

Thermal Neutron Flux in Thimble #8 (half cut out)

Reactor Power 30 Megawatt

x ————— Measurements obtained by Reference 1.

o ————— Measurement obtained by Reference 2.

Solid line represents flux expressed by equation 12
assuming $\phi_0 = 1.85 \times 10^{10}$

10¹⁰
9
8
7
6
5
4
3
2
10⁹
9
8
7
6
5
4
3
2
10⁸
9
8
7
6
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4
3
2
10⁷
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10⁶
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10⁴
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10³
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4
3
2
10²
9
8
7
6
5
4
3
2
10¹
9
8
7
6
5
4
3
2
10⁰

Measurements performed on March 30, 1959

Distance from Bottom of Thimble, inches

022

050

B. Gamma-Ray Dose Rates

A comparison of the measured and calculated gamma-ray dose rates at the cavity liner-concrete interface is shown in Table X. The two sets of data cannot be compared directly since the measurements were carried out with instruments calibrated in a gamma-ray facility only, whereas the measured dose rates actually include a contributing component due to neutrons, the magnitude of which is unknown and difficult to analytically estimate.

Calculated gamma-ray dose rates should therefore be lower than the measured dose rates. The results in Table X support this argument in the case of a no cut-out configuration but not in the case of a half cut-out configuration. This is as might be expected, however, since the ratio of calculated to measured thermal flux at the cavity liner-concrete interface is much higher in the half cut-out case than in the no cut-out case. Reference to Table VIII indicates a substantial contribution from the iron regions, hence the behavior is not surprising. It must be pointed out here that, as in the case of the neutron fluxes, the measured dose rates fail to show any appreciable difference between the no cut-out and half cut-out configurations. This could be due to the fact that diffusion theory generally does not give good agreement with experiment in hydrogenous media, (e.g., in the concrete) as mentioned earlier, or in strong absorbers (e.g., the iron regions). As a result, errors in the thermal flux distribution in the iron regions would be carried over into the gamma-ray dose rate calculations (See Table VIII).

The gamma-ray dose rates measured along the instrument thimble are shown in Figures 9 and 10. Also shown is the calculated dose rate in the concrete and the dose rate due to gamma-ray streaming as expressed by equation (20). It can be seen that the calculated total gamma-ray dose rate is almost always lower than the measured dose rates as is expected since the measured dose rates also include a neutron component of unknown magnitude. It is interesting to note that, up to about 40 inches from the bottom of the thimble, the slopes of the calculated and measured dose rates agree; the magnitude of the difference between the two is about a factor of 3. However, at distances above 40 inches from the bottom of the thimble, the slope of the measured dose rate does not agree with that of the calculated dose rate and, with the exception of the measurements obtained with a Radacon Ion Chamber in instrument thimble #8 (see Figure 10), the magnitude of the difference is greater than a factor of 10. Over these greater distances, however, the slope of the measurements, with the exception of those made with the Radacon Ion Chamber, agrees with that obtained from the neutron flux measurements (see Figures 7 and 8). This might indicate that the measurements obtained with ferrous sulfate solutions and the Jordan Meter are more sensitive to neutron fluxes than those with the Radacon Ion Chamber.

Figure 9

Measured Dose Rate in Thimble #7 (No cut-out)
 Corrected to 20 MWt
 o Reference 1, Ferrous Sulfate Solutions
 + Reference 2, Jordan Meter

— Dose Rate calculated with Grace Code
 No cut-out Configuration

--- Equation 20, $D_0 = 3.12 \times 10^4$ r/hr

— Sum of Grace Code Results and equation 20

Dose Rate, r/hr

10^3

9

8

7

6

5

4

3

2

10^2

9

8

7

6

5

4

3

2

Distance from bottom of Thimble, inches

Figure 10

Measured Dose Rate in Thimble #8 (half cut out)
 Corrected to 20 mwt
 o Reference 1, Ferrrous Sulfate Solutions
 + Reference 1, Radosan Ion Chamber
 + Reference 2, Jordan Meter

— Dose Rates Calculated with Grace Code
 No cut-out configuration

--- Equation 20, $D_0 = 1.1 \times 10^{-5}$ r/hr

--- Sum of Grace Code Results and equation 20

Table X

Comparison of Measured and Calculated Gamma Dose Rates at the Bottom
of Instrument Thimbles in the SRE

<u>Location</u>	<u>Distance from Bottom of Thimble, inches</u>	<u>Gamma-Ray Dose Rate, r/hr Measured</u>	<u>Calculated</u>
No Cut-Out (Thimble #7)	0	$1.3 \times 10^5(3)$	4.04×10^4
		$1.2 \times 10^5(2)$	
	1.5	$5.31 \times 10^4(1)$	
	6	$9.8 \times 10^4(2)$	
	8	$3.88 \times 10^4(1)$	
Half Cut-Out (Thimble #8)	0	$1.0 \times 10^5(7)$	1.26×10^5
	2	$5.12 \times 10^4(1)$	
	6	$8.30 \times 10^4(2)$	
	8	$3.60 \times 10^4(1)$	

Conclusions

From the comparisons which have been made in the previous section, it is evident that, with the methods used in this study, the agreement between calculation and experiment is fairly reasonable. The major factor influencing the adequacy of the calculational method is undoubtedly the lack of a good technique for handling the perturbation of the fluxes caused by the cut-outs and voids in the shields. This probably accounts for the greatest single contribution to the disagreement between calculation and experiment. Several other features of the problem which could also significantly affect the results are discussed below.

The thermal neutron flux distributions were obtained using diffusion theory, which from past experience is known to over-estimate the thermal flux at the start of hydrogenous regions and to under-estimate the flux in more strongly absorbing media. This effect has been observed here also. In addition, it might be pointed out that, although the thermal neutron flux distributions were obtained using a diffusion theory calculation in which the fast group was constrained to agree with what would be predicted by removal theory, application of this constraint should not significantly affect the results. This is due to the fact that the thermal flux in those regions which were examined is dependent on either the initial thermal flux or the intermediate flux and not on the fast component. This relative independence from the fast flux is due to the fact that in the shield regions investigated either K_{th} or $K_{int} < 1/\Sigma_R$. It should be mentioned that in this study the fast flux was assumed to behave as removal theory would predict, even in those shield regions for which there is no hydrogen between them and the neutron source.

There are several other less important aspects of the analysis which could also influence the results. For example, the SRE core and reflector are not completely symmetrical with relation to the different instrument thimbles and, as a result, will yield different fluxes for the same apparent position (either in the thimble or in the bulk shield). In addition, the geometric approximations which have been made, e.g., cylindrical geometry for the neutron flux calculations and slab or truncated cone geometry for the dose rate calculations, would undoubtedly contribute somewhat to the over-all error, but it is not expected that the magnitude would be too significant.

Another point which should be mentioned insofar as the analytical model is concerned is the manner in which the buildup factors were handled. In this work these factors were approximated by assuming a single material, iron, as being representative of the buildup factor for all the materials. This assumption markedly simplifies the analysis and, at the same time, should not significantly affect the results. This is due to the fact that there is little difference between buildup factors for materials with atomic numbers up to and somewhat beyond that of iron (except in the extreme low energy range, i.e., less than ~ 1.5 Mev).

Lastly, some word should be mentioned about the measurements on which the comparisons in this study have been based. The measurements which were obtained were only of a preliminary nature and although more detailed, and perhaps accurate data might have been desirable at the time, it was thought worthwhile to try to use the preliminary data before launching an extensive measurements program. As can be seen from the results of the measurements, several inconsistencies are apparent. For example, there

appears to be no obvious reason for the relatively flat shape in thimble #7 (beyond about 50") of the measured gamma dose rate using the ferrous sulfate solution detector. In addition, there is no way at present of rectifying the difference between the results obtained using the Jordan meter and either the ferrous sulfate or Radacon Ion Chamber, insofar as instrument thimble #8 is concerned.

Considering the above, it would appear that before a complete evaluation of the adequacy of the methods used in this study can be accomplished, more detailed and carefully controlled experiments would be in order. With such data, it would then be possible to more accurately assess the validity of the mathematical models used in this report. However, as mentioned earlier, the analytical results obtained can be considered as reasonably acceptable for handling the type of situations which were investigated in this report.

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