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OTHER		PROJECT SRE	DATE 10/22/57	
			PAGE 1 OF 18	
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COPIES TO: F. Anderson R. W. Dickinson L. E. Glasgow R. L. Ashley D. S. Duncan V. Keshishian R. Balent M. Dunenfeld J. E. Owens M. L. Brewer F. E. Faris W. E. Parkins H. Pearlman R. L. Tomlinson				
SUBJECT: SRE Control Rod Shielding Requirements				
CONTENTS: I STATEMENT OF PROBLEM PAGE 1 II SUMMARY OF RESULTS AND RECOMMENDATIONS PAGE 1-2 III METHOD USED, DESCRIPTION OF EQUIPMENT, SAMPLE CALCULATIONS . . . PAGE 2-13 IV REFERENCES AND APPENDICES PAGE 14 V Figures page 15-18				
I STATEMENT OF PROBLEM Analyze data taken on radiation traverse ¹⁵ of Mark I control rod (7/30/57). Predict future radiation levels for all SRE control and safety rods. Calculate both shielding necessary to ship a complete rod and shielding necessary to protect a person doing maintenance work on these rods.				
II SUMMARY OF RESULTS AND RECOMMENDATIONS A. <u>Basic Assumptions</u> In order to determine future gamma radiation levels for all control and safety rods, an analysis of the material composition of each was necessary. Based on this analysis, it is reasonably safe to assume that the Mark I control rod represents the case which will result in the highest gamma-ray dose rate due to the large amount of Co ⁶⁰ present in its composition. Proceeding on this assumption, the following shield requirements were established for a Mark I control rod which has been irradiated for 90 days and allowed to cool for 14 days. ¹⁴ B. <u>Shielding Necessary to Ship a Complete Control Rod</u> To ship a complete control rod, a maximum radial shield thickness of 7 inches of lead is required near the bottom of the control rod. The required shield thickness decreases to 2 inches of lead near the top of the rod, as is shown in Figure 3. The top shield must be at least 5 inches thick, whereas the bottom shield requires a thickness of 7 inches				

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of lead. All shield thicknesses are such that they will reduce the radiation level to 10 mrem/hr at a distance of one meter, as required by ICC shipping regulations.

C. Shielding Necessary to Protect Maintenance Personnel

The shielding necessary to reduce the radiation level for maintenance personnel to 0.75 mrem/hr at a distance of one meter from the source, corresponding to the outer surface of the shield, is shown in Figure 4. The maximum radial shield thickness is 9 inches of lead. A shield thickness of 9 inches of lead is required for the bottom shield while a shield thickness of 7 inches of lead is required for the top shield. The outer surface of the radial shield must be one meter from the center axis of the control rod, and the outer surface of the axial shields must be one meter from the surface of the rod. This will allow room for manipulators to work around the control rod,¹⁴ and will maintain the gamma-ray dose rate at a tenth tolerance level.

The unshielded gamma-ray dose rate 1 cm from the surface of the most highly activated portion of the control rod was calculated to be 5.0×10^4 r/hr 14 days after shutdown following an extended power run of 90 days duration.

D. Degree of Saturation of Extended Power Conditions

In establishing a suitable shield design, consideration should be given to the fact that much of the neutron induced activity in the Mark I control rod originates from Co^{60} , which has only reached 3% of saturation in the 90 days of irradiation that was selected to represent extended power conditions. However, the Fe^{59} activity in the control rod would have reached 60% of saturation during this same period. If longer irradiations are considered for the Mark I control rod, the additional activity associated with the Co^{60} would result in radiation levels far in excess of that for which the coffin has been designed. In considering other rods which have very little Co^{60} present in their composition, the fact that the only other major contributor (Fe^{59}) is 60% saturated indicates that the shielding specified in Figures 3 and 4 would be adequate even for longer irradiations.

III DISCUSSION OF RESULTS AND SAMPLE CALCULATIONS

A. Discussion of Method

To obtain an accurate description of the neutron induced gamma radiation emitted from the Mark I control rod due to extended 20 MW operation, it was felt desirable that a calculation of the dose rate due to recent power runs be performed first. The results of this calculation could then be compared with measured values (the Mark I control rod was removed from the reactor following the recent power runs and a radiation traverse performed) and a correction factor

introduced to account for several possible sources of error which would arise were such experimental data not available. The dose rate for extended 20 MW operation was then calculated as a function of lead shield thickness using the same decay schemes and material analysis as for the previous calculations, and the correction factor applied. The Mark I control rod was selected because it would be the rod which would produce the highest neutron induced gamma-ray dose rate due to the large amount of Co^{60} present.

B. Calculations for Recent Power Runs

1. Description of Measurements

After completion of the SRE power runs which occurred between 7-11-57 and 7-27-57, a radiation traverse of the SRE Mark I control rod and thimble (located in hole #38) was taken on 7-30-57, three days after shutdown. The rod and thimble were raised inside the coffin while measurements were taken at 10 inch intervals using both Cutie Pie and Juno detectors located a distance of 33 inches from the center line of the control rod. During this radiation traverse, the coffin shield was raised 6 inches above floor level (See Fig. 1) and the control rod was in a "full down" position inside the thimble. A total of 22 measurements were taken. The results are shown in Fig. 2.

2. Calculation of Control Rod and Thimble Activity

The gamma-ray source intensity in $\text{Mev/cm}^3\text{-sec}$ for each element in the rod and thimble was obtained by the following expression:

$$S_v = \frac{M N_o}{V A} \sigma_a (325^\circ\text{C}) \eta E_\gamma \sum_{n=1}^6 [(1 - e^{-\lambda T_n}) e^{-\lambda t_n}] \bar{\phi}_n \quad (1)$$

where,

M = mass of isotope being considered - gms

V = volume of region being calculated - cm^3

N_o = Avogadro's number = 6.025×10^{23} atoms/gm mole

A = atomic weight of element concerned

$\sigma_a(325^\circ\text{C})$ = activation cross section of parent isotope at 325°C , corrected for a Maxwellian distribution- cm^2/atom

η = number of gammas emitted by the product isotope per decay
 E_γ = gamma energy - Mev
 λ = decay constant of the product isotope-sec⁻¹
 T_n = irradiation time - sec
 t_n = decay time - sec
 n = irradiation number
 $\bar{\phi}_n$ = thermal neutron flux in the vicinity of the control rod-n/cm²sec.

In order to proceed with the calculations it was first necessary to analyze the composition^{1,2,3,4} of elements in the regions over which the gamma-ray dose rate was measured. The decay schemes for these elements were then determined using the "Table of Isotopes," by J. M. Hollander, I. Perlman, and G. T. Seaborg⁵.

As noted in Section III, B. 1, there were 22 measurements made and hence 22 regions to be investigated, many of which contained movable parts that, because of their motion, were actually subjected to different thermal neutron flux levels during the irradiation period. Therefore, it was necessary to obtain data relative to the rod position⁶ and reactor power level as functions of time during the irradiation. For the recent power runs (See Table I) the data was reduced to six time intervals (defined here as irradiation number, n) of constant flux and constant rod position.

TABLE I

Simplified Control Rod Position History

n	Reactor Power Level (M.W.)	T_n (10 ⁴ sec)	t_n (10 ⁶ sec)	Rod Position (Inches from Full Down Position)
1	3.1	5.040	1.512	40.5
2	3.2	7.488	1.39	42.6
3	8.9	6.264	1.327	47.0
4	8.0	6.264	1.249	51.6
5	1.7	5.112	0.3555	40.5
6	3.8	8.136	0.2682	42.0

The thermal neutron flux distribution in the vicinity of the control rod during reactor operation⁸ was considered to be the same as the calculated axial distribution through the core, decreased throughout by the same ratio of peak radial flux to the flux at

the radius occupied by the rod⁷. It was assumed that, at the center of the core, the thermal neutron flux was equal to 2.858×10^{12} neutrons/cm²-sec-thermal MW.

The main contributions to the induced source intensities were found to be Co⁶⁰, which emits 1.33 Mev and 1.17 Mev gamma rays in cascade, Fe⁵⁹ which emits 1.30 Mev and 1.10 Mev gamma rays in parallel, and Cr⁵¹ which emits a 0.32 Mev gamma ray.

In this portion of the analysis, it was necessary to keep the contributions from the control rod and thimble separate, since the shielding calculations which were to be performed for operation at full power were only to be concerned with the control rod. (Under normal maintenance conditions, it would not be necessary to remove both the control rod and the thimble.) In this manner, some of the calculated values could be used for both cases.

3. Calculation of Gamma-Ray Dose Rate and Comparison with Measurements

The gamma-ray dose rate from the control rod was calculated using the following volume equivalent cylindrical source approximation¹¹.

$$\dot{\phi} = \frac{B S_V R_o^2}{2(a+z)k} F(\theta, b_2) - \text{mr/hr} \quad (2)$$

where,

B = buildup factor ≈ 1

S_V = source intensity - Mev/cm³-sec

R_o = radius of source - cm

a = distance from surface of source to the detector - cm

z = effective self-attenuation distance - cm

k = conversion factor between Mev/cm²-sec and mr/hr

$\theta = \tan^{-1} \frac{h}{2(a+z)}$

h = height of cylinder-cm

$b_2 = \mu_1 t_1 + \mu_s z$

μ_1 = mass absorption coefficient¹⁰ for shield region - cm⁻¹

μ_s = mass absorption coefficient¹⁰ for source materials - cm⁻¹

t_1 = thickness of shield region = 0 cm (in this case)

$$F(\theta, b_2) = \int_0^\theta e^{-b_2 \sec \phi} d\phi$$

The thimble was also calculated using this equation with $z = 0$ since it is a cylindrical shell with very little self absorption.

The results of these calculations are compared with the measured values in Figure 24. Correction factors were calculated for each region of the control rod in order that they may be applied to the results of the calculations for extended full power operation.

4. Sources of Error

As was mentioned in Section III A. above, the possibility of the existence of several large sources of error made it desirable to perform a calculation of gamma-ray dose rate from the recent power runs in order to compare these results with measurements. Some of these sources of error are as follows.

- a. The calculation of the thermal neutron flux distribution in the vicinity of the control rod, discussed in Section III B. 2. above, introduces the largest potential source of error. Contributing to this are the effects of the rod on flux depression and blackening, both of which are difficult to evaluate in the case considered,
- b. The method for obtaining the assumed rod position⁶ and reactor power level.
- c. Analysis of percent composition of materials and decay schemes.

5. Neutron Streaming

The question of neutron streaming through the control rod and in the voids between the control rod and thimble and the resultant expected increase in the gamma-ray dose rate from portions of the rod at considerable distances from the core was resolved in the following manner. The method used to calculate the gamma-ray dose rate from both the rod and thimble completely neglects neutron streaming, i.e., the thermal neutron fluxes that were used did not consider the effects of streaming. Any significant difference in the slope of the calculated and measured dose rates would be an indication of the amount of neutron streaming that occurs since the measured values would necessarily include this effect. Such a comparison, as shown in Figure 1, indicates that both the calculated and measured values have nearly the same slope. Therefore, it can be stated that neutron streaming does not occur with a magnitude sufficient to perturb the flux distribution or effect the gamma-ray dose rate from the control rod and thimble.

C. Calculations for Extended Full Power Operation

1. Calculation of Control Rod Activity

The neutron induced activity was recalculated for each region of the control rod using Eq. (1). The calculation considered an irradiation time of 90 days at full power (20 MW) and a decay time of 14 days. It was necessary to make the assumption that the control rod position was the same as during the recent power runs. This is a conservative assumption since the rod is likely to be positioned further out from the core during extended power operation in order to override the effects of fission product poisoning.

2. Unshielded Gamma-Ray Dose Rate

In order to assess the feasibility of using various organic materials in the maintenance operations, an evaluation of the unshielded gamma-ray dose rate was undertaken 1 cm from the surface of the control rod using the cylindrical source approximation of Eq. (2). A maximum value of 5.0×10^4 r/hr was obtained for the most highly activated part of the control rod.

3. Shielding Necessary to Ship a Control Rod

Using the source intensities obtained in Section III. C. 1. above, the gamma-ray dose rate was calculated as a function of shield thickness for extended operation (90 days).

a. Tolerance Specifications

Three restrictions as prescribed by ICC Shipping Regulations are pertinent to the case at hand.

- i. The maximum strength of any source to be transported by public carrier may not exceed 300 curies of Co^{60} and 2.7 curies of Fe^{59} . (Actually sources in excess of these amounts can be transported provided the container meets with the approval of the Bureau of Explosives.)
- ii. The radiation level must be less than 10 mrem/hr at 1 meter from the shielded source.
- iii. The radiation level must be less than 200 mrem/hr at the surface of the package.

b. Source Strength

To satisfy the first of these, the source strength was calculated from the relation:

$$C = \sum_i \frac{L_i S_{Li}}{E K} \quad (3)$$

where,

L_i = length of the i-th region of the control rod - cm

S_{Li} = source intensity of Co^{60} or Fe^{59} for the i-th section of the control rod - Mev/cm-sec.

E = gamma-ray energy per disintegration = 2.50 Mev/disintegration for Co^{60} and 1.20 Mev/disintegration for Fe^{59} .

$K = 3.7 \times 10^{10}$ disintegration/sec. -curie.

Evaluation of Eq. 3 yields 85.70 curies of Co^{60} and 5.90 curies of Fe^{59} . In as much as the Co^{60} has reached only 3% of saturation in the 90 day exposure considered, and since the source strength of Fe^{59} is in excess of the amount prescribed by the source strength restrictions, the container used for shipping the control rod must be approved by the Bureau of Explosives. Although other rods have less cobalt, the amount of iron will be essentially the same as in the Mark I rod, so that Bureau of Explosives' approval will be required for transportation of any of the control or safety rods.

c. Shielding

1. Radial Shield

To satisfy the second restriction, the radiation level at a distance of 1 meter from the source after penetration of several thicknesses of lead was calculated using the following equation¹¹, in which the control rod is represented by an equivalent line source:

$$\phi_{P,\theta} = \frac{A S_L}{4 \pi a k} \left\{ F[\theta_2, b_1 (1-\alpha)] - F[\theta_1, b_1 (1-\alpha)] \right\} \quad (4)$$

where,

$$b_1 = \lambda_1 t_1$$

and A and α are constants associated with an exponential approximation to the dose buildup factor, i.e., $B = Ae^{\alpha \lambda t}$, where λt is the number of relaxation lengths under consideration. The remaining symbols are as defined previously. In the calculations, the radial lead shield thickness was adjusted to reduce the gamma-ray dose rate to 10 mrem/hr at 1 meter from the source.

At wide angles from the normal to the source, the control rod may be represented by a point source. Therefore the calculation of the penetration of gamma rays from the source through lead shielding at wide angles was accomplished with the following equation¹¹:

$$\textcircled{a} \quad P_2, \theta = \frac{B S_0 e^{-b_1(\sec \theta)}}{4\pi (a \sec \theta)^2 k} \quad (5)$$

where,

S_0 = point source intensity-Mev/sec

B = buildup factor for thickness, t_1 (sec θ), of lead
 $\theta > 45^\circ$

and the remaining symbols are as defined previously.

ii. Top and Bottom Shield

The bottom shield thickness was determined by calculating the gamma-ray dose rate one meter from the source with the control rod represented as an equivalent end cylindrical source

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using the following equation¹¹:

$$\phi = \frac{A S_V}{2 H_S k} \left\{ E_2 \left[(1-\alpha) b_1 \right] - \frac{E_2 \left[(1-\alpha) b_1 \sec \theta \right]}{\sec \theta} \right\} \quad (6)$$

where,

$$\theta = \tan^{-1} \frac{d}{a}$$

d = diameter of cylinder - cm

and the remaining symbols are as defined previously.

In evaluating the top shield thickness, it should be noted that the gamma radiation penetrating the top shield originates primarily from the lower sections of the control rod. Because of the relatively large distance from these regions of the control rod to the top shield surface, an end line source representation was used to determine the top shield thickness. In these calculations, the gamma-ray dose rate was evaluated at a point one meter from the top of the control rod. This calculation was performed using the following equation¹³:

$$\phi = \frac{A S_L e^{-(1-\alpha)} (\mu_S a - \mu_i t_1)}{4 \pi k} \left\{ \frac{E_2 [\mu_S a (1-\alpha)]}{a} - \frac{E_2 [\mu_S (a+l) (1-\alpha)]}{a+l} \right\} \quad (7)$$

where,

l = length of line source - cm

and the remaining symbols are as defined previously.

iii. Scattered Radiation

Calculations were also performed to determine the gamma-ray scattering off the top shield and the extent to which these gamma rays would penetrate the upper shield walls. The calculations were performed using the following equation:¹²

$$I = I_0 A a^{-2} b^{-2} R(E_1, \theta_1, \theta_2) \quad (8)$$

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where,

I = scattered gamma-ray intensity at the inner surface of the shield wall - r/hr

 I_0 = intensity 1 cm from a point source - r/hrA = projected area of the scatterer (top shield) in the direction of the incident beam - cm^2

a = distance from the source to the scatterer - cm

b = distance from the scatterer to the point of detection inside of shield walls - cm.

 $R(E_1, \theta_1, \theta_2)$ = reflection coefficient for the lead scatterer

$$= \frac{.024 K (E_1, \theta)}{\frac{\mu_1}{\rho} + \frac{\mu_2 \cos \theta_1}{\rho \cos \theta_2}}$$

 θ_1 = angle between the incident beam and the normal to the surface of the scatterer. θ_2 = angle between the emergent beam and the normal to the surface of the scatterer. E_1 = initial gamma-ray energy - Mev. θ = supplement to the angle between the incident and emergent beam of radiation. $\frac{\mu_1}{\rho}, \frac{\mu_2}{\rho}$ = absorption coefficients of the scatterer for the incident and emergent radiation, respectively - cm^2/gm . $K(E_1, \theta)$ = Klien-Nishina Equation
 $= \frac{1}{2} (P^2 - P^3 \sin^2 \theta + P^4)$

$$P = \frac{E_2}{E_1} \left[1 + (E_1/mc^2) (1 - \cos \theta) \right]^{-1}$$

 E_2 = energy of the scattered photons - Mev mc^2 = the self energy of the electron = 0.5107 Mev.

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Using this method, the scattered gamma-ray intensity, I , at the inner surface of the shield wall was calculated to be 3.78 mr/hr for the most active section of the control rod, the average energy being 0.306 Mev. The total scattered dose rate will be, at most, a factor of 10 higher than this value, because of the additional contribution from the less active sections of the control rod. The results of this scattering calculation show that the shielding necessary to reduce direct radiation from the control rod to the required level is more than adequate for reducing scattered radiation to the required level.

iv. Surface Dose Rate

Additional calculations were performed to determine the dose rate at the surface of the shield using Eq. (2) with the distance, a , represented by the shield thickness at that point. The results of this calculation show that the dose rate at the surface of the shield would be well within the ICC shipping requirement of 200 mrem/hr.

d. Weight Saving

The final shield dimensions required for the shipping of a control rod, as obtained by these calculational methods, are shown in Figure 3. It should be noted that a considerable saving in shield weight can be effected by shipping the control rod in 2 sections, since much of the lead in the upper portion of the shield is required merely to attenuate the higher gamma dose rates from the bottom half of the control rod.

4. Shielding Necessary to Protect Maintenance Personnel

The shielding necessary to protect maintenance personnel was calculated by the same methods that were used in establishing the shielding requirements for shipping. Using these methods, lead shield thicknesses were selected which would reduce the gamma-ray dose rate to a tenth tolerance level of 0.75 mr/hr. In this calculation, it was considered necessary to place the outer surface of the shield a distance of at least one meter from the center of the control rod, in order to allow room for manipulators to work around the control rod.

It should be noted that the various shield thicknesses for both the shipping and the maintenance calculations were obtained for the case where the control rod is in a full down position. If it is desirable to move the control rod up and down during such maintenance or shipping, or if the control rod is jammed in an up

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position either during operation or maintenance the following provisions must be made:

- a. The design must allow the shield to move up and down simultaneously with the rod, so that a given portion of the control rod will always be shielded by a corresponding lead thickness.
- b. The actual radial shield configuration must assume the thickness of the thickest shield region (at least to the extent where motion is possible).

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Figure 1.

Location of Detectors During Gamma Radiation
Traverse Measurements of Mark I Control Rod

NOTE: Coffin Shield in raised
position as shown

Figure 2

Gamma Ray Dose Rate Due To Retent
SRE Power Runs

Figure 3
Shielding Required To Ship Control Rod

Figure 4

Shielding Required To Protect Maintenance Personnel