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SHIELDING OF IRRADIATED APPR-1 FUEL ELEMENTS

Contract No. AT (11-1) - 318

Issued July 6, 1956

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SHIELDING OF IRRADIATED APPR-1 FUEL ELEMENTS

The purpose of this analysis is to determine the shielding requirements for irradiated APPR-1 fuel elements both in the fuel element storage pit and in transit to the reprocessing plant. The general method of calculation and the results, for each case, are presented below.

I. Shielding for Storage Pit

Upon leaving the reactor core, used fuel elements are transferred directly to the storage pit, which is filled with water. The shielding variable in this case is the distance below the water surface at which the element is located. Calculations were made for water depths of 6, 9, 12, 15, and 18 feet.

Calculations were for one element of twice the average activity, thus allowing for variations from the average. Source strengths were calculated with the aid of charts in Reference 1. Reactor operation time was taken as infinite; time after shutdown was taken as one day. The numerous gamma groups emitted by fission products were combined into two groups. Group B contained gammas with energies from 1.10 Mev. to 1.65 Mev; the average energy for group B was taken as 1.5 Mev. Group C contained gammas from 1.80 Mev. to 2.90 Mev. in energy; the average energy for Group C was 2.5 Mev. Gammas below 1.10 Mev. in energy contribute a negligable dose, as shown in reference 2.

Gamma fluxes were calculated by using the graphs and formulas presented by Foderaro and Obenshain in Reference 3. The element was assumed to be a line source. Results were checked, and good agreement obtained, with a point source approximation. Build-up effects were included by use of the method presented in Reference 4. The following results were obtained:

a	ϕ_B	ϕ_C
6	4.35×10^5	3.52×10^5
9	1.68×10^3	3.87×10^3
12	7.58	6.12×10
15	3.71×10^{-2}	9.20×10^{-1}
18	2.00×10^{-4}	1.42×10^{-2}

where:

a = height of water above element (feet)

ϕ_B = gamma flux for group B (Mev/cm^2-sec)

ϕ_C = gamma flux for group C (Mev/cm^2-sec)

Reference 5 gives the following formulas for conversion from flux to dose rate:

$$D_B = \frac{\phi_B}{5.76 \times 10^2} \qquad D_C = \frac{\phi_C}{6.36 \times 10^2}$$

.001736   .001572

where:

D_B = dose rate for group B (mr/hour)

D_C = dose rate for group C (mr/hour)

ϕ_B and ϕ_C as defined above.

These formulas yielded the following results:

a	D_B	D_C	D_T
6	7.55×10^2	5.54×10^2	1310.
9	2.92	6.09	9.01
12	1.31×10^{-2}	9.64×10^{-2}	.109
15	6.44×10^{-5}	1.45×10^{-3}	.00151
18	3.47×10^{-7}	2.24×10^{-5}	.0000227

where: $D_T = D_B + D_C$ = total dose rate (mr/hour)

Other symbols as defined above.

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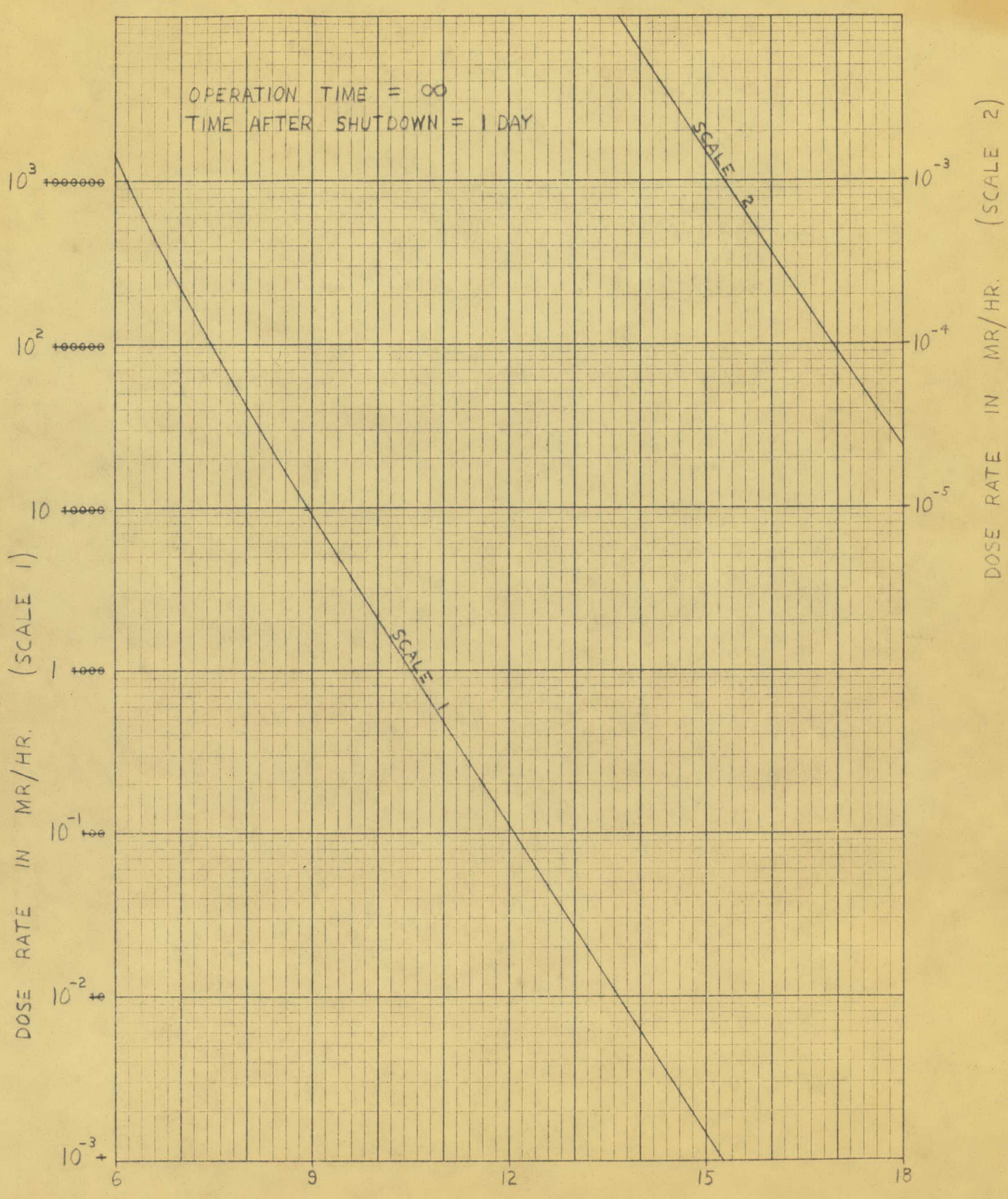
These results are plotted on the graph included below. Note that the results, as presented in both chart and graph, are for a single fuel element of twice the average activity. The distance indicated is from the surface of the water to the nearest point of the active core section.

WATER SHIELDING OF IRRADIATED ELEMENT

DOSE RATE v. WATER SHIELDING

MODEL DATE



II. Shielding in Transit

The calculations presented here are simply a check upon the existing design for the fuel element shipping container (layout no. R-9-48-1001). The maximum dose rate was calculated at the surface of the container and at a point one meter from the surface, for both the radial and axial directions. Calculations were for six fuel elements (the capacity of the container), each having 1.5 times the average activity, to allow for variations in activity.

The method of calculation was similar to that employed in the fuel pit calculations above. Source strengths were obtained from Reference 1, using various lengths of operation time and time after shutdown. The gammas were again divided into two energy groups.

Gamma fluxes were determined by use of References 3 and 4. A cylinder approximation was used for the radial direction, and a line source approximation for the axial direction. The formulas for conversion from flux to dose rate were the same as those presented in the previous section.

Dose rates for 1000 hours operation and for infinite operation, both at 100 days after shutdown, were desired. Unfortunately, however, Reference 1 does not give data on infinite operation at 100 days after shutdown. Since data was given for both 1000 hour and infinite operation at 10 days after shutdown, dose rate for the infinite operation at 100 days after shutdown case was calculated by assuming the ratio of dose rate at 10 days after shutdown to that at 100 days after shutdown is a constant, regardless of operation time.

The calculated dose rates are presented in the chart below. They lie within the established ICC limits of 200 mr/hour at the surface and 10 mr/hour one meter from the surface.

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Time after shutdown = 100 days

Operation Time	Direction	Dose Rate in mr/hour At Surface	1 m. from surface	
1000 hours	radial	10.2	2.83	.2774
	axial	9.7	4.07	.42
infinite	radial	20.8	5.77	.277
	axial	19.7	8.27	.42

$$\frac{X}{100+X} = .42$$

$= 28.5''$

$42 = .58X$

25.4
1.27

REFERENCES

1. Moteff, Fission Product Decay Gamma Energy Spectrum, APEX 134.

2. Meem and Fairbanks, Shielding Requirements for the Army Package Power Reactor, APAE No. 3, May, 1956.

3. Foderaro and Obenshain, Fluxes from Regular Geometric Sources, WAPD-TN-508, June 1955.

4. Taylor, Application of Gamma Ray Build-up Data to Shield Design, WAPD-RM-217, January, 1954.

5. Reactor Handbook, volume 1, RH-1, page 946.