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SUBJECT: Review of the ORR Shield for 30-Mw Operation

TO:

FROM: Walter Zobel, E. G. Silver, and T. V. Blosser

Abstract

The proposed shield for the Oak Ridge Research Reactor was reviewed for a reactor power of 30 Mw. It was concluded that the shield is more than adequate for this power providing the shield is augmented at the experimental facilities.

APPROVED BY
PATENT DEPARTMENT
7/2/58 William E. Davis
DATE SIGNATURE

DISCLAIMER

REVIEW OF THE ORR SHIELD FOR 30-Mw OPERATION

The adequacy of the proposed shield for the Oak Ridge Research Reactor (ORR) was considered for a reactor power of 30 Mw.* The points at which the dose was calculated as well as the methods used and detailed results are shown in Appendices A and B. From these results it was concluded that the reactor shield is more than adequate for a 30 Mw operation provided that the shield at the experimental facilities (see Fig. 1) is augmented locally. (The calculated dose at these positions without augmentation is 3 r/hr.) The ORR design group is fully aware of this problem and has indicated that such augmentation will be provided.

The dose at the water level above the reactor is above that recommended for long-term exposure at a permanent installation. It is not believed, however, that these criteria should be applicable since the likelihood that someone will remain there is very remote. Similar comments apply to the dose at the exclusion fence around the cooling units. In both cases, the dose is calculated to be below the maximum operational exposure level.

The dose in the demineralizer room, assuming equilibrium concentrations of all activities, will be above tolerance; however, the dose outside this room is below tolerance, and there should be no occasion for prolonged occupancy of this room while the demineralizers are in the saturated condition.

As shown in Appendix A, the dose in the room below the reactor (see Fig. 2) is practically impossible to calculate owing to the streaming down the rod openings. The ORR design group has indicated that they are willing to accept a higher dose in this area because there is again no need for personnel to remain in this room for long periods of time with the reactor at 30 Mw, and a door will be provided to shield the entrance to this room if necessary.

The shielding provided for the cooling system, discussed in Appendix B, is considered adequate as long as there is no demineralizer failure. If the demineralizer is taken out of the cooling loop the doses at the exclusion fence (50 ft from the heat exchanger) and at the pump house become excessively high, about 9.5 mr/hr and about 10 mr/hr, respectively.

It is therefore concluded that, as far as can be determined now, the shielding proposed for the ORR is adequate for normal operation at 30 Mw. Provision should be made, however, to protect personnel in the vicinity of the pump house and/or the heat exchanger exclusion fence in the event of demineralizer failure or any other contingency which would add to the activity in the cooling water.

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* This review was orginally made early in 1955 and the results were transmitted to the ORR staff by private communication at that time. Subsequently some changes were made in the shield design. The calculations have accordingly been brought up-to-date, with the exception of the calculation of the bottom plug attenuation.

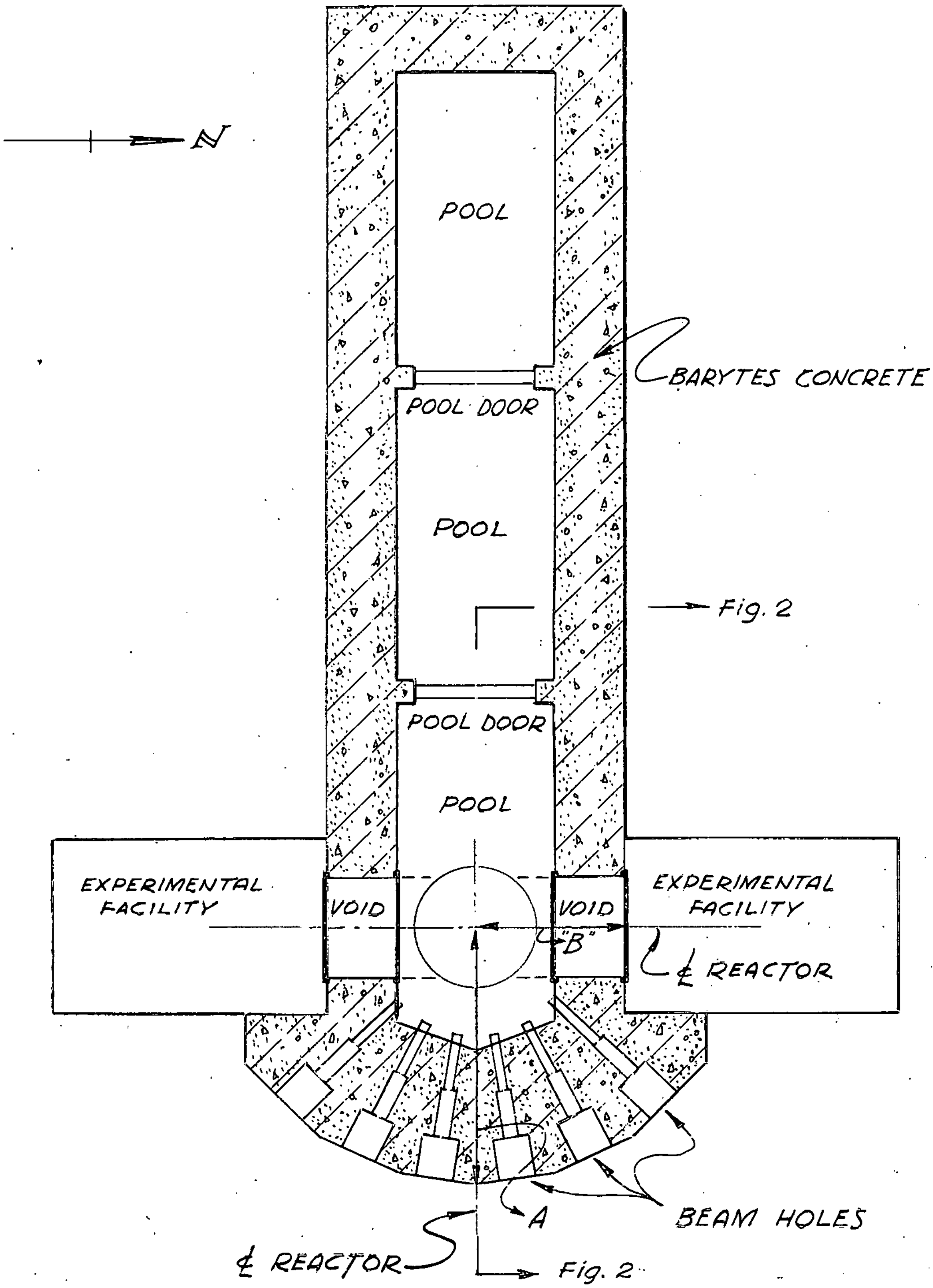
N
POOL
BARYTES CONCRETE
POOL DOOR
POOL
Fig. 2
POOL DOOR
POOL
EXPERIMENTAL FACILITY
VOID
VOID
EXPERIMENTAL FACILITY
"B"
℄ REACTOR
A
BEAM HOLES
℄ REACTOR
Fig. 2

Figure 1

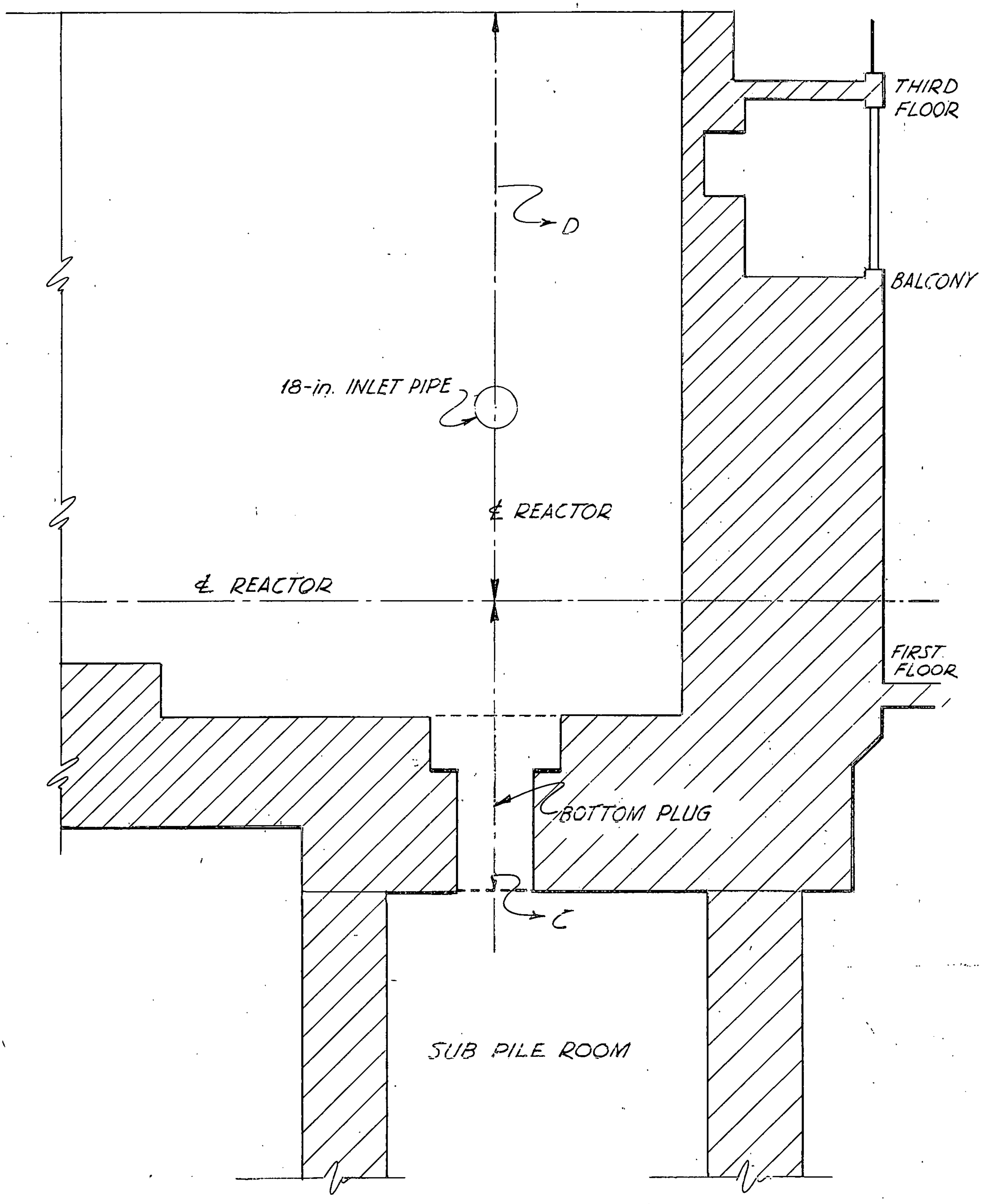
THIRD FLOOR
D
BALCONY
18-in. INLET PIPE
℄ REACTOR
℄ REACTOR
FIRST FLOOR
BOTTOM PLUG
C
SUB PILE ROOM

Figure 2

Appendix A

DOSE CALCULATIONS FOR THE BIOLOGICAL SHIELD AT 30 Mw OPERATION

In order to determine the biological shielding needs for the ORR at a power of 30 Mw a survey of the adequacy of the present shielding design was made at the obviously weaker points of the shield, near the service facility, experimental holes and activated cooling water lines.

In making these determinations, account must be taken of the fact that a 7 x 9 lattice reactor is rather flexible. The loading of the reactor as well as the beryllium reflector may be shifted to satisfy a given experimental condition. Thus for dosage estimations a reasonable assumption for the leakage from the ORR reactor would be the same as for a H_2O-reflected BSF reactor operating at the same power level.

Throughout these determinations, unless otherwise specified, the attenuation[1-3] of barytes concrete for gamma rays (8 Mev or lower) will be taken as a 10-cm relaxation length, and for neutrons as a 8.3-cm relaxation length.

The ORR designers have assumed that the experimenter at a given facility will be required to shield his facility to less than a tolerance dose.

Health physics recommends that the maximum operational exposure level for each type of radiation for a 40-hr week not exceed the following:

Fast neutrons:	0.5 mrep/hr or 50 neutrons/cm^2·sec at 2 Mev
Thermal neutrons:	1750 neutrons/cm^2·sec
Gamma rays:	7.5 mr/hr

Each position of the following estimate is shown in Figs. 1 and 2 and the data for each position is summarized in Table 1.

A. Estimate of the Dose on the East Face of the Reactor Shield

It is assumed that the beam holes are filled with shielding equivalent to that of the surrounding reactor shield and that the attenuation of the aluminum ducting in the water portion of the shield is negligible.

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1. N. F. Lansing, Letter and attachment re: Barytes Cement by M. Greenhill to C. F. Knesel, CF-50-9-54 (Sept. 15, 1950).
2. E. P. Blizard, "Reactor Shielding with an Iron Aggregate Concrete, Comparison with Ordinary Concrete," ORNL-209 (Mar. 16, 1949).
3. S. Podgor, "Neutron Relaxation Length in Barytes Concrete," ORNL-CF-50-4-54 (April 13, 1950).

Table 1. Expected Radiation Doses

Position	Gamma-Ray Dose (r/hr)	Fast-Neutron Dose (rep/hr)	Thermal-Neutron Flux (neutrons/cm^2·sec)
A. East Face of shield	3.6×10^{-7}	9.2×10^{-15}	3.6×10^{-10}
B. Experimental facility	3.0	1.6×10^{-5}	0.6
C. Subpile Room			
Through bottom plug	5.6×10^{-13}	1.03×10^{-9}	3×10^{-4}
Through concrete	1.2×10^{-2}	Negligible	Negligible
D. H_2O level above reactor	3.3×10^{-3}	Negligible	Negligible

Gamma-Ray Dose:

186.5 cm = thickness of water shield (distance from surface of reactor to inner surface of concrete shield)

1.5×10^{-3} r/hr·watt = gamma-ray dose in H_2O 186.5 cm from BSF reactor

1.5×10^{-3} r/hr·watt $\times 3.0 \times 10^{7}$ watts = 4.5×10^{4} r/hr impinging on interface of concrete shield and water

The attenuation in 8 ft. 5 in. (256.5 cm) of barytes concrete would be:

$$e^{-256.5/10} = 8.2 \times 10^{-12}$$

Therefore, the dose on the outer surface of the shield would be:

$$3.6 \times 10^{-7} \text{ r/hr}$$

Thermal-Neutron Flux:

186.5 cm = thickness of water shield (distance from surface of reactor to inner surface of concrete shield)

3.5×10^{-4} neutrons/cm^2·sec·watt = thermal-neutron flux in H_2O 186.5 cm from BSF reactor

$3.5 \times 10^{-4} \times 3.0 \times 10^{7}$ watts = 1.05×10^{4} neutrons/cm^2·sec on inner surface of concrete shield

The attenuation in 265.5 cm of barytes for thermal neutrons would be:

$$e^{-256.5/8.3} = 3.4 \times 10^{-14}$$

Therefore, the flux at the outside of barytes concrete shield would be:

$$3.6 \times 10^{-10} \text{ neutrons/cm}^2\text{·sec}$$

Fast-Neutron Dose:

186.5 = thickness of water shield (from surface of reactor to inner surface of concrete shield)

9×10^{-9} rep/hr·watt = fast-neutron dose in H_2O 186.5 cm from BSF reactor

9×10^{-9} rep/hr·watt x 3.0×10^{7} watts = 0.27 rep/hr at inner surface of concrete

The attenuation of fast neutrons by 256.5 cm of barytes concrete would be:

$$e^{-256.5/8.3} = 3.4 \times 10^{-14}$$

Therefore, the dose on the outer surface of the concrete shield would be:

$$9.2 \times 10^{-15} \text{ rep/hr}$$

B. Estimate of Dose at Void and Experimental Facility Boundary

When the experimental facility is not in use, a tank covered with 1.5 in. of lead and filled with water is attached to a 4-ft barytes concrete plug and inserted into the void, thus forming shielding equivalent to the surrounding area.

Gamma-Ray Dose:

7.6×10^{1} r/hr·watt = gamma-ray dose extrapolated to the surface of BSF reactor

7.6×10^{1} r/hr·watt x 3.0×10^{7} watt = 2.3×10^{9} r/hr dose on surface of ORR reactor

The attenuation of 1.5 in. (3.8 cm) lead would be

$$e^{-3.8/3.8} = 0.37$$

where the relaxation length for lead is 3.8 cm (LTSF Data). The dose entering the water shield would be 8.4×10^{8} r/hr. The 115.6-cm-thick water shield will reduce the dose by a factor of 1.52×10^{3}, so there would be a dose of 5.6×10^{5} r/hr at the inner surface of the concrete plug. The attenuation of 4 ft (122 cm) of concrete would be

$$e^{-122/10} = 5.2 \times 10^{-6}$$

Therefore, the dose on the outer surface of the concrete plug would be

$$3.0 \text{ r/hr}$$

Fast-Neutron Dose:

It is assumed that the 1.5-in. lead covering on the tank has a relaxation length for neutrons the same as H_2O.

119.4 cm = thickness of lead and water shield

1.3×10^{-6} rep/hr·watt = fast-neutron dose in H_2O 119.4 cm from the BSF reactor

1.3×10^{-6} rep/hr·watt $\times 3.0 \times 10^{7}$ watts = 3.9×10^{1} rep/hr on inner surface of concrete plug

The attenuation of 4 ft (122 cm) of concrete would be

$$e^{-122/8.3} = 4.1 \times 10^{-7}$$

Therefore, the dose on the outer surface of the concrete plug would be

$$1.6 \times 10^{-5} \text{ rep/hr}$$

Thermal-Neutron Flux:

It is assumed that the thermal-neutron flux at large distances in the water shield is the result of slowed-down fast neutrons in that region.

119.4 cm = thickness of water shield

5.0×10^{-1} neutrons/cm^2·sec = flux in H_2O 119.4 cm from BSF reactor

5.0×10^{-1} neutrons/cm^2·sec·watt $\times 3.0 \times 10^{7}$ watts
= 1.5×10^{6} neutrons/cm^2·sec on surface of concrete plug

The attenuation of 122 cm concrete would be

$$e^{-122/8.3} = 4.1 \times 10^{-7}$$

Therefore, the flux at the outer surface of the concrete plug would be

$$0.6 \text{ neutrons/cm}^2\cdot\text{sec}$$

Estimate of Capture Gamma Dose in Concrete Plug

Since experimental data[1-3] are available for both the thermal-neutron flux and the gamma-ray distributions in barytes concrete, the emergent gamma-ray flux on the outer surface of the concrete may be determined from the thermal-neutron flux at the water-concrete interface. Thus the source distribution of the capture gamma-ray flux within the concrete would have the smae relaxation length as the thermal-neutron flux but the gamma-ray flux would be attenuated by the concrete as the gamma rays diffuse from their "birth" location to the outer surface of the shield. Assuming that the concrete consists of a number of 22-cm-thick infinite slab sources and the contribution of each is concentrated at its surface, a slab x cm from the outer surface of the concrete would contribute to the dose an amount

$$\Gamma = \phi/2\ F_o(b) \text{ photons/cm}^2\cdot\text{sec}$$

where

ϕ = neutrons/cm$^2\cdot$sec at x,

$$F_o(b) = \int_b^\infty \frac{e^{-y}}{y}\,dy,$$

x = distance from the outer surface,

$b = x/\ell$,

ℓ = 10-cm relaxation length.

For various values of x the capture gamma-ray doses would be:

x (cm)	ϕ	b	Γ
22	5.0×10^1	2.2	1.85
42	4.0×10^2	4.2	1.2
62	4.8×10^3	6.2	1.34
82	6.0×10^4	8.2	1.92
102	6.3×10^5	10.2	2.14
122	1.5×10^6	12.2	0.57
		Total	9.02

Total/2 = 4.51 photons/cm$^2\cdot$sec

2.2×10^5 = conversion factor from photons/cm$^2\cdot$sec (for assumed energy of 3 Mev) to r/hr

$4.51 \div 2.2 \times 10^5 = 2.0 \times 10^{-5}$ r/hr at outer surface of concrete plug

C. Estimate of Dose in Subpile Room

Gamma-Ray Dose:

181.5 cm = thickness of water shield from lower surface of reactor (reactor fuel) to top of bottom plug

1.7×10^{-3} r/hr·watt = dose at 181.5 cm from BSF reactor

1.7×10^{-3} r/hr·watt $\times 3.0 \times 10^{7} = 5.1 \times 10^{4}$ r/hr at top of bottom plug hole

The attenuation offered by the bottom plug, which is assumed to be filled with 1/2 in. of lead shot and the free volume filled with paraffin,* is calculated as follows:

The ratio of the volume occupied by shot to the free volume is 5:3. This would appear as a reduced density effect.

The following absorption coefficients are based on a gamma-ray energy of 5 Mev (capture from Al, Be)

Absorber	(cm^{-1})
Water	0.03
Pb (density = 11.3)	0.47
Pb shot (reduced density = 6.8)	0.28

The attenuation in 152.4 cm of lead shot-paraffin bottom plug would be

$$(152.4 \times 0.28)e^{-152.4 \times 0.28} = 1.1 \times 10^{-17}$$

Therefore, the gamma-ray dose in the subpile room through bottom plug would be

$$5.6 \times 10^{-13} \text{ r/hr}$$

Obviously the bottom plug is over-shielded, but since the reactor control rods pass through the bottom plug the largest contribution to the dose in the subpile room will be from the streaming around the control rods and through the adjacent concrete.

*Since these calculations were performed, the composition of the plug was changed to an iron shot and barytes concrete mixture.

The thickness of concrete adjacent to the bottom plug hole is 152.4 cm. It will attenuate the gamma-ray dose by

$$e^{-152.4/10} = 2.5 \times 10^{-7}$$

Therefore, the dose in the subpile room through the concrete would be

$$1.2 \times 10^{-2} \text{ r/hr}$$

Fast-Neutron Dose:

181.5 cm = thickness of water shield from lower surface of reactor (reactor fuel) to top of bottom plug

2.3×10^{-9} rep/hr·watt = fast-neutron dose in H_2O 181.5 cm from BSF reactor

3×10^{7} watt $\times$ 2.3×10^{-9} rep/hr·watt = 6.9×10^{-2} rep/hr

A reasonable assumption for the attenuation of fast neutrons by a lead shot paraffin medium would be a 10-cm relaxation length, which is comparable to the relaxation length of H_2O at 150 cm from the BSF reactor. The attenuation of 152.4 cm of lead shot-paraffin medium would be

$$e^{-152.4/10} = 1.5 \times 10^{-8}$$

Therefore, the dose on the outer surface of the bottom plug would be

$$1.03 \times 10^{-9} \text{ rep/hr}$$

Thermal-Neutron Flux:

181.5 cm = thickness of water shield

6.5×10^{-4} neutrons/cm^2·sec·watt = flux 181.5 cm in H_2O from BSF reactor

6.5×10^{-4} neutrons/cm^2·sec·watt $\times$ 3.0×10^{7} watts

= 1.95×10^{4} neutrons/cm^2·sec at water and bottom plug interface

This flux is approximately a factor of 10 above the tolerance level. In order to obtain the dose below the plug, it was assumed, from age displacement, that the thermal-neutron flux has the same relaxation length as the fast-neutron flux. The flux level thus obtained is 3×10^{-4} neutrons/cm^2·sec.

D. Dose at H_2O Level Above Reactor

On the pool side of the reactor there is a 7-in. water tank through which water is circulated, thus reducing the dispersion of activated water through the pool. This tank is essential as shown by experiments carried out at the BSF.

Gamma-Ray Dose:

731.5 cm = thickness of water shield from reactor (reactor fuel) to H_2O level

1.1×10^{-10} r/hr·watt = dose 731.5 cm in H_2O from BSF reactor

1.1×10^{-10} r/hr·watt $\times$ 3.0×10^{7} watt = 3.3×10^{-3} r/hr on surface of H_2O

Neutron Dose:

Since the neutron dose in water at 181.5 cm is only about 70 mrep/hr, it is obvious that the dose at 731.5 cm is insignificant.

Appendix B

DOSE CALCULATIONS FOR COOLING SYSTEM AT 30-Mw OPERATION

The calculation of radiation[1] from the ORR cooling water assumes water conditions equal to those at the LITR. An approximate equation for the equilibrium concentration of a given nuclide at the core outlet is:

$$a_i = \frac{\phi \Sigma t_r}{1 - e^{-(\alpha + \lambda t)}}$$

where

ϕ = flux in the pile (average),

Σ = activation cross section for production of a given nuclide,

λ = decay constant of the nuclide,

t_r = residence time in the core,

$\alpha = \frac{\text{demineralizer flow rate}}{\text{total flow rate}}$,

t = "loop" time.

This formula assumes (1) complete demineralizer efficiency, (2) that $\alpha \ll 1$, and (3) that $t_r \ll 1/\lambda$. Then:

$$\frac{a_i(0)}{a_i(L)} = \frac{V_0 V_L}{V_L V_0} \; \frac{P_0}{P_L} \; \frac{Q_L}{Q_0} \cdot \frac{1 - e^{-(\alpha + \lambda t)_L}}{1 - e^{-(\alpha + \lambda t)_0}}$$

where the subscripts L and O refer to two different reactors, which have the same fuel density and water density in their cores, Q is the pumping rate, P is the reactor power, and V is the coolant volume in the core. If the subscript O refers to the ORR and L to the LITR, then:

$$\frac{a_i(0)}{a_i(L)} = \frac{P_0}{P_L} \; \frac{Q_L}{Q_0} \; E_i = 0.67 E_i$$

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1. N. F. Lansing, private communication.

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where

$$E_i = \frac{1 - e^{-(\alpha + \lambda t)_L}}{1 - e^{-(\alpha + \lambda t)_0}}$$

and assuming the system conditions shown below.

It is possible to compute E_i for each of the activities of interest. The calculated activity depends strongly on the system conditions at the ORR and LITR assumed for the calculation. The values used were furnished by private communication from F. Binford[2] and are as follows:

	LITR	ORR
Pump capacity	1200 gpm	18,000 gpm
Demineralizer fraction, α	0.0025	0.0044
Loop time, t	8 min.	3 min.
Reactor Power	3 Mw	30 Mw

The results are:

$$E(Na) = 1.30$$

$$E(Al) = 1.52$$

$$E(Mg) = 2.14$$

Therefore:

$$\frac{a^{(0)}}{a^{(L)}} (Na) = 0.866$$

$$\frac{a^{(0)}}{a^{(L)}} (Al) = 1.013$$

$$\frac{a^{(0)}}{a^{(L)}} (Mg) = 1.43$$

The measurements at the LITR[3] give the activity for each of the nuclides of interest at the LITR core outlet. We can therefore find the activity

2. May 16, 1958. These are different from the ones orginally used, and the calculations have been changed accordingly.

3. W. S. Lyon and S. A. Reynolds, "Radioactivities in Cooling Water from the ORNL Low Intensity Test Reactor," CF-54-9-99 (Sept. 16, 1954).

of the ith component at the ORR cooling unit by the following equation:

$$(N_i)_{OC} = \left(\frac{a^{(0)}}{a^{(L)}}\right)_i e^{-\lambda_i t_c} (N_i)_L$$

where

$(N_i)_{OC}$ = activity of the ith nuclide at the ORR cooling unit in curies/ft^3,

$(N_i)_L$ = activity at LITR core outlet as measured by Lyon and Reynolds in curies/ft^3,

$\left(\frac{a^{(0)}}{a^{(L)}}\right)_i$ = ratio of activities as defined above,

t_c = time of flow from core outlet to cooling unit of ORR = 101 sec.

A. Dose from Cooling Unit

The volume above ground at the cooling unit is found to be 64.1 ft^3 per section. There are eight sections and they form a row 160 ft long. Therefore, the source strength due to the ith activity, D_i, considering the cooling unit to be a line source, is:

$$D_i = 3.7 \times 10^{10} \ \frac{64.1 \times 8}{160} (N_i)_{OC} = 11.9 \times 10^{10} \ (N_i)_{OC} \ \frac{\text{disintegrations}}{\text{ft}\cdot\text{sec}}$$

1. The gamma-ray flux at a point 50 ft from the midpoint of this line is then:

$$\phi_i(\text{photons/cm}^2\cdot\text{sec}) = \int_{-L}^{L} \frac{D_i\,dx}{4\pi\,(x^2 + b^2)} = \frac{D_i}{2\pi b} \tan^{-1} \frac{L}{b}$$

$$\phi_i = 1.26 \times 10^8 \ (N_i)_{OC} \ \text{photons/cm}^2\cdot\text{sec}$$

where

$b = 50$ ft,

$L = 1/2 \times 160$ ft $= 80$ ft.

The flux is then converted to dose rate and the following results are obtained.

Na	3.62 mr/hr
Al	4.30 mr/hr
Mg	0.85 mr/hr
Total	8.77 mr/hr

However, allowance must be made for air attenuation, self shielding, and pipe shielding, and this was done by assuming (1) only 20% contribution from tube banks, and (2) attenuation by 3 in. of H_2O, 1/4 in. of Fe, and 50 ft of air. This yields a total dose rate of 4.7 mr/hr.

2. In case the demineralizer fails one must set $\alpha_0 = 0$ in the equation for E_i and then calculate again. This yields, without accounting for attenuation, as mentioned above:

Na	= 10.45 mr/hr
Al	= 4.30 mr/hr
Mg	= 0.87 mr/hr
Total	15.62 mr/hr

and again taking the attenuation into account:

9.47 mr/hr

3. There will be a contribution to the dose rate at Bldg. 3010 due to the heat exchanger. This is calculated with all factors the same except for the integral representation of the geometry. The integral becomes:

$$\phi = \frac{D_i}{4\pi} \int_N^M \frac{dx}{x^2 + b^2} = \frac{D_i}{4\pi b} \tan^{-1} \frac{M}{b} - \tan^{-1} \frac{N}{b}$$

where the geometry is shown in Fig. 3 and

$$b = 133\ \text{ft},$$
$$M = 60\ \text{ft},$$
$$N = 249\ \text{ft},$$
$$M/b = 0.52$$
$$N/b = 1.93.$$

This yields:

0.072 mr/hr under normal conditions with self attenuation,

0.149 mr/hr with self attenuation and demineralizer failure.

B. Dose from Pump House

It was thought that the dose rate due to the pump house would affect the largest number of people at Bldg. 3010. Calculations were therefore made at that location and extrapolated back to the pump house wall where necessary, as shown below.

1. The dose rate due to the pump house varies, depending on the location in Bldg. 3010 (BSF) since part of the second floor of this building is not shielded by the 8-in. barytes concrete wall of the pump house. This situation arises because the shielding wall is only 8 ft high, and the pump house is so much lower than Bldg. 3010 that some radiation streaming over this wall can reach the second floor of Bldg. 3010 (see Fig. 4). Allowing for shielding through 8 in. of barytes and a distance of 100 ft, the dose from the pump house is 0.105 mr/hr. On the second floor, assuming 1/3 of the pump house liquid to be exposed over the wall, the dose rate due to the pump house is 0.34 mr/hr.

At the exterior pump house wall, assumed to be 10 ft from the pumps, considered as point sources, one gets a dose $100^2/10^2 = 100$ times as great as that computed at Bldg. 3010 (100 ft away) or 10.5 mr/hr. Since, however, the radiation is largely self-shielded and is not a point source, a factor of 1/2 is considered conservative, yielding 5.2 mr/hr.

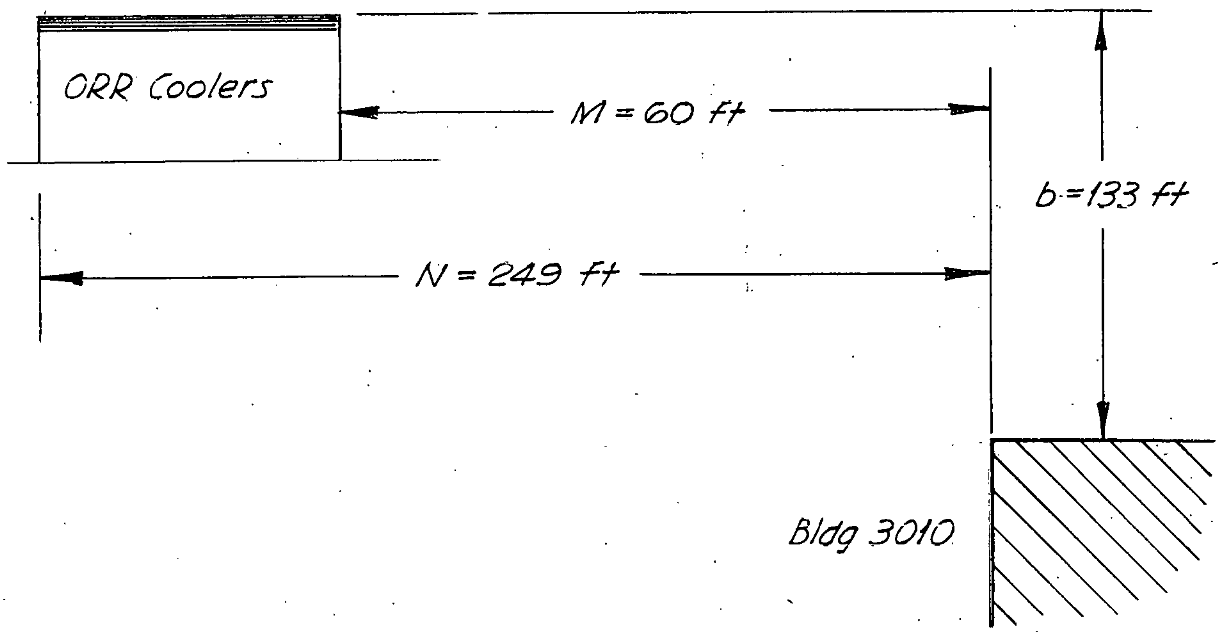

Fig. 3. Relative Locations of ORR Coolers and Bldg. 3010.

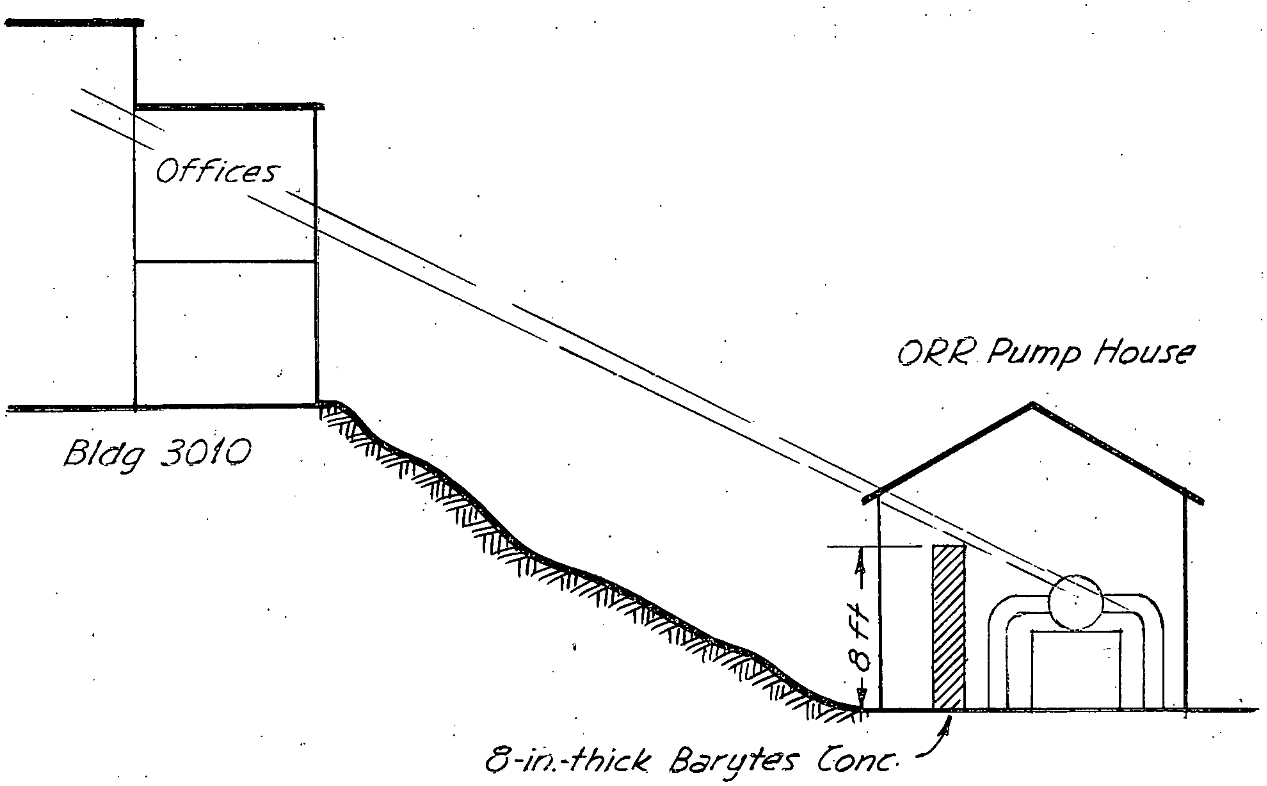

Fig. 4. Relative Vertical Positions of ORR Pump House, Pump House Shield, and Bldg. 3010.

2. In the event of demineralizer failure the dose will vary in the same ratio as at the point 50 ft from the heat exchanger considered in A-2, i.e. 9.47/4.7 = 2.02. Multiplication by this factor results in a dose rate at the pump house wall of 2.02 x 5.2 = 10.5 mr/hr.

It should be noted that, even under normal conditions, it would be hazardous to work on the roof of the pump house or to enter the pump cells for extended periods.

C. Dose at Building 3010.

1. The dose rate to be expected at Bldg. 3010 is then the sum of the dose rate due to the heat exchanger and due to the pump house, i.e., 0.105 + 0.072 = 0.177 mr/hr below the second floor, and 0.34 + 0.072 = 0.412 mr/hr on the second floor under normal conditions.

2. In the event of demineralizer failure one again multiplies by 2.02 to get 0.356 mr/hr and 0.825 mr/hr on the first and second floors, respectively, of Bldg. 3010.

The results are summarized in Table 2.

Table 2. Dose Rates Due to ORR Cooling System at 30 Mw

Location	Dose Rate (mr/hr)	
	Normal	No Demineralization
At pump house wall	5.2	10.5
At Bldg. 3010, first floor	0.18	0.36
At Bldg. 3010, second floor	0.41	0.82
50 ft from coolers	4.7	9.5

Distribution

1-3	F. T. Binford
4	E. P. Blizard
5	T. V. Blosser
6	C. E. Clifford
7	T. E. Cole
8	J. A. Cox
9	F. L. Keller
10	G. deSaussure
11-12	F. C. Maienschein
13	E. G. Silver
14	W. H. Tabor
15	W. Zobel
16-17	Laboratory Records
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