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FINAL PHYSICS REPORT FOR THE
ENGINEERING TEST REACTOR

By
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June 25, 1956

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AECU-3607

Volume II

FINAL PHYSICS REPORT
for the
ENGINEERING TEST REACTOR

June 25, 1956

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CORRECTIONS
TO THE
FINAL PHYSICS REPORT
FOR THE
ENGINEERING TEST REACTOR

Corrections to Volume II.

As mentioned in the Errata Sheet for Volume I, N^{16} sources are a factor of 2 too high. Therefore, all radiations from N^{16} sources are to be decreased by a factor of 2, giving an additional margin of safety.

A discrepancy will be noted in the 7 Mev gamma radiation values at the inner face of the biological shield as used in Sections XVII and XXII (4.2×10^{10} Mev/cm²/sec and 4.1×10^{10} Mev/cm²/sec, respectively). This was due to drafting errors in redrawing Figure 17-6 (Figure 22-3). The lower value is actually the correct one.

ERRATA SHEET
FOR THE
ETR FINAL PHYSICS REPORT

- Page 17-1 Last line of second paragraph - omit "for."
- Page 17-2 Footnote† should read: Radiation is to be attenuated to 1/10 tolerance; 0.75 mr/hr for gammas and 8 mv for fast neutrons. Due to inelastic scattering in the several inches of steel and lead in the thermal shields, at the concrete face, the above-thermal neutron flux may be conservatively considered 0.5 Mev.
- Page 17-4 Footnote †, line 5, word 10. Change "must" to "much."
- Page 17-5 C. Change $71\frac{1}{2}$ inches of concrete to 71 inches of concrete.
- Page 17-7 Line 17 should read; in the basement area at C.
- Page 18-4 First paragraph. (See Appendix X-3) should appear at the end of the paragraph.
- Figure 18-2 Ordinate is neutrons/cm² sec. ($\times 10^{13}$)
- Page 18-7 Line 13 should read: Wall and the top head shield above the vessel.
- Page 19-4 4a. Fourth sentence. Radiation from loops with unusually active coolants (sodium) is reduced to about tolerance and delayed fast neutrons from loops with fission fragments or homogeneous fuel are reduced to negligible values.
- Page 19-9 2. (1) A cylindrical source treatment was used to compute radiation in the sub-pile room, from the steel grid plate.
- Page 19-13 Line 13. Water remains in the thimble while the chamber is being removed by personnel in the control rod access room, thus eliminating excessive radiation in the sub-pile room and the control rod access room.
- Figure 19-2 Title - Gamma Radiation Streaming Into the Sub-Pile Room Through In-Shield Pipes and the Air Gap Around the Pressure Vessel.
- Page 19-26 3. (a) Third line. N^{16} gamma source formed during reactor operation at full power.
- Page 19-32 Line 18, last sentence. An analysis of scattering . . .
- Page 20-1 Last paragraph. If an active fuel element should accidentally be dropped into the canal pit, the $8\frac{1}{2}$ inches of iron below the pit will limit radiation in the sub-pile room to 580 mr/hr.

ERRATA SHEET (Continued)

- Page 20-3 Line 12. Change (2S) aluminum to (6061) aluminum.
- Page 20-5 b. (1) Last sentence. The upper sections of the walls are tapered as shown in Figure 20-1, due to the oblique path of the radiation in these areas.
- Page 20-6 (2) Second paragraph. Last word, first sentence. Cooling.
- Page 20-7 First paragraph, second sentence. Therefore, the hottest elements, could be placed . . .
- Page 20-8 d. last sentence. Appendix XII-C
- Page 21-1 B. 2. last sentence. (Forty-nine fuel elements, plus 16 control rods contributing fuel equal to eight fuel elements.)
- Page 21-3 Last paragraph, last sentence - Also by dividing the lifting operation...
- Page 22-3 C. Second sentence should begin: In Section 22 . . .
- Page 22-4 (2) Last sentence. (See Appendix XIV A)
- Page 22-7 The current equations should be written:

$$J_+ = \frac{\phi}{4} - \frac{D}{2} \frac{d\phi}{dx} = \frac{\phi}{4} - \frac{D\phi}{2} \frac{d \ln \phi}{dx}$$

$$J_+ = \frac{10^{11}}{4} - \left(\frac{3 \times 10^{11}}{2} \right) \frac{2.3}{30}$$

- Page 22-10 $\Phi_E = (4.1 \times 10^{10}) (8) (5.4 \times 10^{-10})$
- Page 22-16 Line 10. The current at the forward end of the plug is then . . .
- Line 17. and $\phi = 2 \times 10^{10} e^{-0.105x}$
- Page 22-17 Line 6. $(.866) (58.6) = 50.7\%$
- Page 22-18 Line 3. $\frac{\sum_a Fe}{\sum_a Fe + \sum_a B} = \frac{0.0483}{1.52} = 0.0318$
- Line 18. $(4.8 \times 10^8) \frac{2.9 \times 10^7}{2.5 \times 10^{10}} = 5.6 \times 10^5 \text{ Mev/sec/gm}$
- Page 22-21 Footnote *. The other ducts have smaller diameters and attenuate radiation to a greater extent.
- Page 22-22 Angle shown in sketch is θ and in line 12 we have $= 2\pi r (2.5 \sin \theta) \phi$

ERRATA SHEET (Continued)

Page 22-24 Delete last sentence.

Page A9-4 Last column of table is Linear Coefficient

Page A9-7 Change second sentence to: Using a relaxation length of 8.2 cm in the concrete, the current is reduced to 1/10 tolerance, 8 neutrons/cm²/sec,* after attenuation by 71 inches of concrete.

Change second equation to $e^{-\frac{x}{8.2}} = 2.86 \times 10^{-10} = e^{-22}$

Change third equation to $x = 180$ cm or 71 inches

Page A9-11 Last sentence. Change made conservative to more conservative.

Figure A9-1 Change first path notation from 1 and 2a to a and b(1).

Change second path from 2b to b(2).

Page A9-12 Change last sentence to: (See page 19-18 and 19-19).

Page A9-13 C. The last two sentences should be a separate paragraph.

The shielding, therefore, reduces radiation to less than 1/10 tolerance in the basement area near the canal pit. Other areas are analyzed similarly.

Page A9-17 2. First sentence. Certain experiments may employ sodium as a coolant.

Second sentence. Last word. Gamma.

Page A10-2 Second sentence. Neutrons that are captured in water produce low energy gammas...

A 1. Second sentence. , because these low energy gammas,

Page A10-5 Lines 3-5. By successive approximations, b is found to equal 8.4.

The absorption coefficients for the attenuating materials are:

$$\begin{aligned} \mu_{\text{magnetite concrete}} &= 0.10\text{cm}^{-1} \\ \mu_{\text{steel}} &= 0.225\text{cm}^{-1} \end{aligned}$$

Page A10-10 4. Grey and Black Rods*
Footnote is: *See G. E. drawing M-753

Page A10-15 Third sentence. These five sources are attenuated through the metal and water above the core and through the $4\frac{1}{2}$ inch steel top cover plate of the pressure vessel.

Replace Page A10 - 3 with the following page.

ERRATA SHEET (Continued)

2. The 7Mev Source Contributions

a) Fuel Elements

There are 49 fuel elements plus fuel element sections of approximately 8 grey rods in the core. This gives a fuel element volume of 3.0×10^5 cc, of which 40 percent is aluminum.

$$\begin{aligned} Q_v (\text{aluminum capture}) &= (nv)(\sum_a)(\text{volume}) (\text{Mev}) \\ &= (3.6 \times 10^{14})(0.011)(1.2 \times 10^5)(7) \\ &= 3.32 \times 10^{18} \text{ Mev/sec} \end{aligned}$$

$$\begin{aligned} Q_v (\text{fission}) &= (nv)(\sum_f)(\text{volume})(\text{gamma yield})(\text{Mev}) \\ &= (3.6 \times 10^{14})(0.0524)(3 \times 10^5)(0.12)(7) \\ &= 4.75 \times 10^{18} \text{ Mev/sec} \end{aligned}$$

b) Grey Rods

The grey sections of approximately 8 grey rods will be in the core. This is a volume of 4.25×10^4 cc of which 25 percent is aluminum and 22 percent is nickel.

$$\begin{aligned} Q_v (\text{capture}) &= (2.9 \times 10^{14}) \left[(0.364)(935) + (0.011)(1.1 \times 10^4) \right] (7) \\ &= 0.95 \times 10^{18} \text{ Mev/sec} \end{aligned}$$

c) Experiments

The experiments in the core contribute a 7 Mev gamma source from both metal capture and fission, of 15×10^{18} Mev/sec.*

The total 7Mev gamma source is therefore 2.4×10^{19} Mev/ sec.

* Detailed calculations are not given here due to the security classification of this data.

Page A10-16 First sentence. The following values of a^2 and μx are determined for attenuation to a point just above the pressure vessel cover plate.

Page A10-18 Line 9. . . . ($R = 0.6$ mb)

Page All-3 Line 10-13. Volume of aluminum = (57 elements) (9 in²) (37.5 in)
 (0.4) = 7.7×10^3 cu in.
 = 1.26×10^5 cc

Weight of aluminum = 3.4×10^5 gm

Line 17-18. (2.5×10^7 photons/sec/gm) ($\frac{3.6 \times 10^{14}}{10^{12}}$) (3.4×10^5 gm)
 (1.65 Mev) = 5.1×10^{15} Mev/sec

Page All-5 Line 3. Fuel section 5.1×10^{15}

Page All-7 1. Delete first sentence.

Page All-8 Line 17. +0.25
 Line 20. +0.22

Page All-9 Line 9. Total cross sectional area of air below the core (annuli and facility tubes)

Page All-13 Line 11.

$$F_2(T, t) = \frac{\lambda_1 \lambda_2}{1 - \lambda_2} \left[\frac{(1 - e^{-\lambda_2 T}) e^{-\lambda_2 t}}{\lambda_2} - \frac{(1 - e^{-\lambda_1 T}) e^{-\lambda_1 t}}{\lambda_1} \right]$$

Figure All-2 Delete $4\frac{1}{2}$ " dimension shown below grid plate.

Page All-19 Line 8. $\Psi_E = \frac{QvE}{2\mu s} B(1 - e^{-b_1(\sec\phi - 1)}) \frac{e^{-b_1}}{b_1}$

Page All-12 Line 7. The contributions from the other energy groups in the 1.65 Mev to 2.8 Mev range are determined similarly. The total radiation directly below the tube is found to be 38 mr/hr.

Page All-26 Second line from bottom. $\mu x = (0.29) (30.5) = 8.85$

Page All-27 Values given in the table are:

Gamma Energy Mev	Radiation Level mr/hr
2.8	0.7
2.5	6.6
2.1	6.8
1.65	13.2

Page All-28 Line 16. Total $\mu x = 11.1$

Page All-32 Line 8. = 2.9×10^4 Mev/cm²/sec or 32 mr/hr

Page A12-2 Line 7. 2.0×10^9 Mev/sec per operating watt

Page A12-5 Line 7. $= \sqrt{\frac{3770 \text{ cm}^2}{3.14}}$

Page A12-6 Line 2. Last term in numerator is 1.04×10^{-12}

Page A12-7 Line 13. Attenuations along the worst path are given in Table A12-I.

Page A12-9 Line 10. (Appendix X C)

Page A12-13 Line 3. Appendix IX D

Third sentence. The average above-thermal flux in the core is about 1.5×10^{15} n/cm²/sec.

Line 7. macroscopic

Page A12-17 Line 4. μx for steel is 7.56

Line 11. Appendix XII B

Page A13-3 Line 15. of 1834 mr/hr at the refueling platform.

Line 17. This gives a total dose of 330 mr.

Page A14-2 Line 16. $r = 2.875$ inches (inside radius)

Page A14-4 Line 2. $= 8.6 \times 10^9$

Line 4. 8.6×10^9 instead of 8.5×10^9

Line 5. $= 153$ mr/hr

Line 9. 8.6×10^9 instead of 8.5×10^9

Line 10. $= 0.055$ mr/hr

Page A14-9 Line 13. for 2.5 Mev gammas

Page A14-12 Last line. Appendix XI E

Page A14-16 Line 6. $r^2/2L^2 =$

Page A14-18 Line 6. $\kappa = 0.693/T^{\frac{1}{2}}$

Page A14-19 Line 4. $Q_{v2}E = Q_{v1}E e^{-\kappa t_{4-2}}$

Figure A14-2 the lower point is P_1 and is 6 1/2 feet above the console floor.

Page A14-25 $\sum_a$ for nickel is 0.364 cm^{-1} giving a V.F. $\times \sum_a$ of 0.028 cm^{-1} .

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XVI. SHIELDING PROVIDED FOR THE HEAT EXCHANGER BUILDING

- A.** This section discusses the shielding required for the heat exchanger building due to the buildup of activity in the primary coolant water.

During the early stages of the work on the ETR the tentative design for the heat exchanger building was furnished by the Kaiser Engineers. Detailed calculations on the most effective method of shielding the building were performed and were described in the Physics Progress Report of December 1, 1955. At a later stage, when construction was about to begin on the heat exchanger building, the arrangement was changed considerably. Because of the lack of time a detailed analysis of the new arrangement was not possible, and the results of the previous analysis were modified to obtain adequate shielding for the building. This section contains an analysis of the shielding provided for the heat exchanger building as it is presently being constructed.

The various wall thicknesses in the heat exchanger building were originally set to limit the radiation dose rate to less than one mr/hr on the wall adjacent to the reactor building. The dose rate in a non-operating pump pit with all other pumps operating is ten times tolerance, or 75 mr/hr. All other walls have a tolerance dose rate, or 7.5 mr/hr.

In general, the dose rates are equal to or below these levels. There are some specific points where the radiation intensity is greater than the above values, however, these areas are localized and in no case does the integrated whole-body radiation exceed the desired levels. It should be remembered, that this problem involves many difficult geometrical source arrangements and therefore is cer-

HEAT EXCHANGER BUILDING

Figure 16-1

ELEVATION OF TYPICAL BANK

Figure 16-2

It had been previously established that the important radiation during operation was the 6.5 Mev photons from the decay of N^{16} . The effect of corrosion products, recoil products, and natural contaminants after shutdown have not been considered here. Other radiations are from N^{17} , formed by the $O^{17}(n,p)N^{17}$ reaction, and O^{19} from $O^{18}(n,\gamma)O^{19}$. N^{17} emits a 4.14 Mev neutron with a 4.14 second half-life. O^{19} emits a 1.6 Mev gamma with a 29.4 second half-life. Both of these radiations are several orders of magnitude lower than the N^{16} activity. N^{16} is formed in the reaction $O^{16}(n,p)N^{16}$ and decays back to O^{16} which has a half-life of 7.35 seconds. The above reaction has a threshold of approximately 9.5 Mev. The associated gammas fall into the following energy of groups:

6.7 Mev	40 percent
6.2 Mev	40 percent
5.9 Mev	2 percent

For this problem an average energy of 6.5 Mev and a yield of 80 percent were used. The unit volume activity was calculated by two methods. The first method assumed an activation cross section of 0.014 mb averaged over the fission spectrum, and an activating fast flux of 8×10^{14} neutrons/cm²/sec to extend over a 3-1/2 foot length. The activity at the core outlet in this core is given by

$$Q_v = 0.8 N \sigma \Phi \frac{(1 - e^{-\lambda T_r})}{(1 - e^{-\lambda T_c})}$$

where

T_r = time in flux = 0.1 sec

T_c = cycle time = 69.2 sec

N = number atoms of parent isotope.

Using this method, an activity of 2.82×10^6 gammas/sec/cc is obtained.

In the above equation the total cycle time can be considerably inaccurate before introducing any significant error into the activity. The error in determining the total cycle time results from the inability to accurately determine the time in residence of the cooling water in the volume above the core. The activity of 2.82×10^6 gammas/sec/cc was rounded off to 3.0×10^6 gammas/sec/cc in early calculations. However, there is a more absolute method of determining the water activity which is discussed in detail elsewhere. In brief, the number of activations per second is

$$N_a = \int \Phi(E) \sum_a dE dv = \int N(E) \frac{\sum_a}{\sum_s} (E) dE$$

The best known reaction cross sections were used, and it was assumed that one collision with an hydrogen atom or one inelastic collision with an aluminum atom reduced the neutron energy below the 10 Mev reaction threshold. They, by numerically averaging the values over the Watt Spectrum from 10 to 20 Mev, the total active nuclei formed each second were determined. These active nuclei produce an activity at the reactor outlet of 6.6×10^6 gammas/sec/cc. This activity was used in determining the shielding requirements.

Further details of the shielding calculations can be found in Appendix VIII, which contains sample calculations.

C. Results

The following is a description of the location of points used in the shielding calculations. For each wall one of the points identified represents the point of highest radiation intensity on that wall.

<u>Point</u>	<u>Location</u>
PA1 (See Figs. A8-7 and A8-8)	Wall A, six feet above the pavement and opposite the midpoint of the heat exchangers.
PA2 (See Figs. A8-7 and A8-8)	Wall A, opposite the midpoint of the inlet headers and 3 1/2 feet above the pavement.
PA3 (See Figs. A8-7 and A8-8)	Wall A, opposite the midpoint of the inlet headers and six feet above the pavement.
PB1 PB2 PB3 (See Figs. A8-2, A8-3, A8-4, and A8-5)	Similar locations to the wall A series, except that these points are on wall B.
PD1 (See Figs. A8-9 and A8-10)	Wall D, opposite the midpoint of the first bank of heat exchanger headers and six feet above the pavement.
PD2 (See Figs. A8-9 and A8-10)	Wall D, opposite the midpoint of the first bank 18-inch header inlets and six feet above the pavement.
PC1 (See Figs. A8-11 to A8-16)	Located in the exact center of pump pit No. 1.
PC2 (See Figs. A8-13 to A8-16)	Wall C, above the pump pit roof and directly opposite the 24-inch inlet riser to the first bank.
PM1 (See Figs. A8-17 and A8-18)	Pump motor region between motor No. 1 and pump pit wall three feet above the floor.
PM2 (See Figs. A8-17 and A8-18)	Pump motor region, on floor directly above the 36-inch headers and opposite motor No. 1.
PM3 (See Figs. A8-17 and A8-18)	Pump motor region, on floor next to the base of motor No. 1 such that it receives the highest possible radiation intensity from the 36-inch header.

TABLE IV

UNIT VOLUME ACTIVITY AT VARIOUS POINTS IN THE
PRIMARY COOLANT CIRCUIT*

Point	Δt (sec)	Activity at Core Inlet	Heat Exchanger Bank† (Activity $\times 10^{-6}$ gammas/sec/cc)					
			1	2	3	2	3	4
Core Outlet			3.00					
A	8.61		1.34	1.13	0.80	1.20	1.00	0.72
B	12.07		0.968	0.82	0.58	0.88	0.73	0.52
C	14.78		0.750	0.63	0.45	0.67	0.57	0.40
D	19.21		0.490	0.42	0.30	0.45	0.38	0.27
E	28.06		0.214	0.18	0.13	0.19	0.16	0.12
F	31.05		0.161	0.14	0.097	0.15	0.13	0.089
G	33.46		0.130	0.11	0.078	0.12	0.10	0.070
H	34.69		0.115	0.097	0.068	0.10	0.087	0.062
I	37.64		0.0872	0.073	0.053	0.079	0.067	0.047
J	41.05		0.0622	0.052	0.037	0.056	0.048	0.034
Core Inlet								
1	45.55	0.0408						
2	49.25	0.0291						
3	56.55	0.0142						
2	48.01	0.0321						
3	51.71	0.0228						
4	59.01	0.0115						

* All these activities should be multiplied by $6.6/3 = 2.2$

† First three columns are for Banks 1, 2, and 3, operating, last three columns are for Banks 2, 3, and 4 operating.

Table IV shows the activities expected at the points shown in the primary circuit of Figure A8-1. These activities represent the two limiting cases for shielding. The first three columns of activities are those for banks 1, 2, and 3 operating, which is the limiting case for wall A. The last three columns are the activities for banks 2, 3, and 4 operating, which is the limiting case for wall B. These activities are based on a reactor outlet activity of 3×10^6 gammas/sec/cc. In the calculations, an outlet activity of 6.6×10^6 gammas/sec/cc was used, so the values in the table should be multiplied by 2.2. The activity at the reactor tank inlet can be obtained as follows:

$$\overline{Q}_v = \sum_1^3 \frac{Q_{v_i} V_i}{V_i}$$

For banks 1, 2, and 3

$$\overline{Q}_v = \frac{1}{3} (0.0408 + 0.0291 + 0.0142) = 2.80 \times 10^4 \text{ gammas/sec/cc}$$

$$2.80 \times 10^4 + 2.2 = 6.2 \times 10^4 \text{ gammas/sec/cc}$$

For banks 2, 3, and 4

$$\overline{Q}_v = \frac{1}{3} (0.0321 + 0.0228 + 0.0115) = 2.21 \times 10^4 \text{ gammas/sec/cc}$$

$$2.21 \times 10^4 + 2.2 = 4.9 \times 10^4 \text{ gammas/sec/cc}$$

Table V shows the dose rates to be expected at the points considered.

TABLE V

Point	Expected	Dose Rate (mr/hr)	
		Desired whole-body	
PA1	0.84	)	< 1 mr/hr
PA2	0.69	)	
PA3	1.4	)	
PB1	2.2	)	7.5 mr/hr
PB2	3.6	)	
PB3	7.3	)	
PD1	3.9	)	7.5 mr/hr
PD2	4.8	)	
PC1	24.0	)	7.5 mr/hr
PC2	11.0	)	7.5 mr/hr
PM1	1.8	)	75 mr/hr
PM2	1.4	)	
PM3	4.7	)	

D. Discussion

It can be seen in Table V that the dose rates at the points considered do not coincide with the desired intensities. For instance, the dose rate along wall A varies considerably. The main reason for this is that the flow through the heat exchangers is relatively slow compared to the 7.35 second half-life of the N^{16} activity, so the unit volume source changes considerably between the inlet and outlet. At any point on walls A or B the radiation at the six-foot level will be the highest experienced along a vertical path from floor to roof. The hot spot at PA3 and PB3 is six feet above the pavement and opposite the inlet headers, which present a large source volume. However, if the detector

were to be moved several inches in the horizontal direction, the intensity will be lower due to the increase in path length. If the detector is moved vertically downward to the 3 1/2-foot level, it can be seen that the dose rate is below the desired level, or approximately one-half that at the six-foot level. Thus, the integrated whole-body radiation for a person standing at these points would be below tolerance, and the highest intensity does not differ seriously from the desired level.

On wall C one point (PC2, located above pump pit No. 1) is higher than the desired dose rate level by a factor of 1.6. However, this is a localized area, since the greater part of this radiation comes from one 24-inch inlet riser. This is not considered serious, since the region requires little access and certainly will not require 40 hours per week access.

Table V indicates that the pump pit shielding could be decreased. However, this area is subject to more inaccuracies than other areas, since there are several sources with unusual geometry contributing to the dose rate. In previous calculations the pump pit support walls were not included in the analysis. In this problem these walls were considered.

In general, the calculations are only slightly conservative. The source strengths and geometries used matched the actual case as closely as was justified by the expected accuracy of the calculations. One factor which is known to be conservative is the buildup factor. In all cases the buildup factor for water was used and this will be higher than for concrete.

XVII. CONCRETE BIOLOGICAL SHIELD

A. Description

At the midplane of the reactor, eight feet of magnetite concrete (density 3.5 gm/cc) are required following the thermal shield to attenuate core radiation to 1/10 tolerance. With the exception of the regions penetrated by the canal and primary coolant pipes, the outer face of the concrete shield is a uniform distance from the center line of the pressure vessel.

The proposed method of fuel element transfer limits the amount of shielding between the canal and the pressure vessel. However, space is available in the canal next to the pressure vessel for 2 1/2 feet of magnetite concrete. This concrete attenuates core radiation to 1/10 tolerance in accessible areas in the region of the canal parapet. In the region penetrated by the primary coolant pipes, the concrete is 5 1/2 feet thick. The 5 1/2 feet of concrete is convenient structurally due to the proposed sequence of concrete pours and provides more than sufficient shielding for core gammas to permit access to the pipe tunnel for a few hours after shutdown.

At the console floor level, iron slabs (18 inches wide, 8 1/2 feet long, by 5 inches thick) are placed in the concrete following the 12-inch diameter vertical sleeves which penetrate the shield. These slabs compensate for the decreased thickness of the concrete shield in the region of the sleeves and provide supplementary shielding for ordinary water-cooled loops as well as particularly active experiments using sodium coolants, etc. In

almost all instances, radiation from the experimental loops will be between 1/10 tolerance and tolerance one foot from the outer face of the biological shield. See Figures 17-1 and 17-2 for construction details.

B. Method of Analysis

1. General

To simplify the design and construction of the concrete biological shield, it was decided that the outer face of the shield should be a uniform distance from the center line of the pressure vessel. On this basis, the amount of magnetite* concrete required to attenuate neutron and gamma radiation emerging radially from the thermal shield at the reactor midplane was first determined.† Approximately 12 inches of concrete were added to this amount to compensate for the concrete displaced by several 12-inch diameter sleeves vertically penetrating the shield which contain return sections of facility tubes and insulating material. The additional concrete also provides a reasonable safety factor for other sections of the shield. The total concrete thickness at the reactor midplane was rounded off to eight feet. Layout drawings showing the resulting biological shield were then analyzed to determine the sufficiency of the shield and any weak points

* Magnetite is chosen because of its desirable gamma attenuation characteristics and its availability and relatively low cost at the ETR site.

† Radiation is to be attenuated to 1/10 tolerance; 0.75 mr/hr for gammas and 8 nv for fast neutrons (due to inelastic scattering in the several inches of steel and lead in the thermal shields). At the concrete face the above-thermal neutron flux may be conservatively considered 0.5 Mev.

EAST-WEST VERTICAL SECTION OF THE REACTOR

Fig. 17-1

Fig. 17-2

HORIZONTAL SECTION THROUGH REACTOR AT CORE MIDPLANE

which existed. Radiation sources in the shield (active coolants in facility loops) were analyzed to determine the need for additional shielding.

2. Sources of Radiation

Radiation to be attenuated by the biological shield may be divided into three groups: primary gammas, secondary gammas, and neutrons.

a. Primary gammas

(1) High energy gammas produced in the core due to the fission of U^{235} and neutron capture gammas from metals such as aluminum and steel in the core and reflector.

(2) Gammas produced in the thermal shield due to the capture of thermal and fast neutrons from the core which have been thermalized in the shield.

b. Secondary gammas

Secondary gammas are produced in the concrete itself due to the capture of thermal neutrons emerging from the thermal shield and fast neutrons which have been thermalized in the concrete.

c. Neutrons

Fast and thermal neutrons emerging from the thermal shield must be attenuated to 1/10 tolerance by the concrete.

3. Determination of Concrete Thickness

A three-group, one-dimensional calculation was performed using an IBM 650 computer on the basis of the thermal shield shown in Figure 17-3. Flux distributions in the ETR core and thermal shield were obtained from this calculation for thermal, epithermal, and fast neutrons and for 1, 2, and 7 Mev

gammas at the reactor midplane. (See Figures 17-4, 17-5 and 17-6). The radiation incident on the inner face of the concrete was obtained from this data.

a. Primary gammas

At the inner concrete face the dominant gamma source in the outward direction at the reactor midplane is a 7 Mev source of 4.2×10^{10} Mev/cm²/sec. This source is treated as an infinite plane source proceeding in the outward direction. Using an absorption coefficient of 0.10 cm^{-1} for 7 Mev gammas in magnetite concrete,* a concrete thickness of 67 1/2 inches is required for attenuation to 1/10 tolerance. (See Appendix IX-B.)

b. Secondary gammas

At the inner concrete face the net neutron current in the outward direction at the reactor midplane is 2×10^9 nv thermal and 2.8×10^{10} nv fast. The thermal neutrons are quickly captured in the concrete (principally captures in the iron aggregate), but the fast neutrons must first be thermalized. In calculating the concrete required to attenuate the secondary gammas, the fast neutrons are considered thermalized and captured one age displacement into the concrete (approximately 12 inches).† It is assumed that

* See Appendix IX-A.

† This is the value given for barytes concrete in the MTR Handbook, Shielding Summary. Actually, magnetite concrete has a shorter age displacement. Since the fast neutron scattering cross section for iron is 2 1/2 times greater than for barium and the thermal absorption coefficient is a factor of 10 greater, the fast flux will be degraded and captured must faster in the magnetite concrete than in barytes concrete. If the magnetite mix (Appendix IX-A) is considered an iron-water mixture with 50 percent of the specified water lost on curing, the age is found to be about 130 cm^2 , giving an age displacement of 6 1/2 inches. This is unfortunately only an approximation, and in lieu of an experimentally verified value a 12-inch age displacement is used to be conservative.

MIDPLANE THERMAL FLUX

MIDPLANE ABOVE-THERMAL FLUX

Fig. 17-5

GAMMA FLUX EXTERNAL TO CORE AT MIDPLANE

each capture results in the emission of a 7 Mev gamma, giving a secondary gamma source of 2×10^{11} Mev/cm²/sec 12 inches inside the inner concrete face. ‡ This source is treated as an infinite plane source proceeding in the outward direction and requires 73 1/2 inches of magnetite concrete for attenuation to 1/10 tolerance. § Adding the 12-inch age displacement gives a total concrete thickness at the reactor midplane of 85 1/2 inches.

c. Neutrons

Using 8.2 cm as the relaxation length for fast neutrons in magnetite concrete, the 2.8×10^{10} nv source entering the inner face of the shield requires 71 1/2 inches of concrete for attenuation to 1/10 tolerance.

C. Discussion of Results

Secondary gammas were found to be the dominant source of radiation from the reactor core, requiring 85 1/2 inches of magnetite concrete following the external thermal shield at the reactor midplane. An additional 12 inches were added to this thickness to compensate for the concrete displaced by the 12-inch diameter sleeves vertically penetrating the shield, which in the worst instance may be air-filled. The additional concrete also provides a reasonable safety factor for other sections of the shield. The total concrete thickness was then rounded off to eight feet determining the position of the

‡ A consideration of the relative volume fractions and neutron absorption cross sections of the concrete constituents shows that essentially all of the neutron captures will occur in the iron, giving one 7 Mev gamma per capture.

§ This treatment is more conservative than considering an exponentially distributed source of capture gammas throughout the shield.

outer face of the biological shield. In addition, iron slabs (18 inches wide, 8 1/2 feet long, by 5 inches thick) were positioned directly following the sleeves* to provide a reasonable safety factor for core radiation passing through these regions and additional shielding for highly active coolants which will at some time be flowing through the facility tubes inside the sleeves. These slabs begin a few inches below the console floor level and extend to about 6 inches above the active section of the reactor core, effectively shielding personnel on the console floor in these regions. Figures 17-1 and 17-2 show vertical and horizontal sections of the shield.†

An analysis of the resulting shield indicates that the highest radiation levels are on the console floor opposite the 12-inch diameter pipes. When the pipes are air-filled, the gamma radiation from the core is reduced to 0.09 mr/hr and fast neutrons are reduced to 0.6 nv. This is 1/50 tolerance, and therefore leaves a safety factor of 5 over the design for 1/10 tolerance. In other areas on the console floor where eight feet of concrete are seen when looking at the core, gamma radiation is reduced to 0.03 mr/hr and neutrons are reduced to negligible levels. This is 1/250 tolerance. (See Appendix IX-B.)

After the experiments are installed, the 12-inch sleeves will house return sections of facility tubes containing active coolants which have decayed only several seconds after being activated in the core. The 5-inch thick iron slab

* See Figure 17-2.

† For further detail see G-E Drawings ETR-5528-MTR-642-M-596 and M-599.

following each sleeve, plus the concrete following the slab, will attenuate radiation from a typical water-cooled facility loop to 1/100 tolerance, and from a particularly active loop (sodium-cooled) to about tolerance. (See Appendix IX-D.)

The shield was analyzed to determine its adequacy in areas near the primary coolant pipes, basement areas, etc. Along such doubtful paths as A, B, and C in Figure 17-7, the material in the floors and walls compensates for the decreased attenuation in the shield itself. Along path A, for example, four feet of magnetite concrete are required in the floor above the primary coolant loops to attenuate radiation from the active water. This amount is sufficient to compensate for the decreased attenuation of core gammas by the shield in this area. Path D shows the region of the biological shield penetrated by the canal. Here it is necessary to have 2 1/2 feet of magnetite concrete following the pressure vessel to attenuate radiation in accessible areas around the top of the canal.* Path C appears to be particularly doubtful, but the concrete and water in this region are found to be more than ample to reduce radiation to 1/10 tolerance in the basement area at A. (See Appendix IX-C for the method of analysis.)

* This is covered more fully in Section XX, Canal Shielding.

EAST-WEST VERTICAL SECTION OF THE REACTOR

XVIII. TOP AREA SHIELDING

A. Description

The top head shield reduces radiation from the reactor core and active coolant water in the top head region to 1/10 tolerance during reactor operation. The shield is essentially 28 inches of magnetite concrete (density 3.5 gm/cc). Figure 18-1 shows a schematic of the shield,* as well as the supplementary shielding described in Section XXII.

The height of the biological shield has been limited to approximately eight feet below the top of the pressure vessel to permit easy access to facility loops which enter the pressure vessel below the flange level (see Figure 18-1). Because of this limitation the top head shield must cover a considerable area at a large radius, and this greatly increases its weight. The shield is suitably divided into sections which can be lifted by the 30-ton crane. To limit the weight and consequently the number of time-consuming lifts, the shield has been designed on the basis of core radiation and primary coolant decay gammas in the top head regions. (It is not intended that the top shield be sufficient to shield active coolants in facility tubes in the top head regions.)

The shielding considerations for facility loops are listed below.

1. Active coolants in the facility loops just outside the pressure vessel and below the top head shield may represent significant radiation levels,

* The actual top head shield design is being determined by Kaiser Engineers and is not firm at this time.

SCHEMATIC TOP HEAD SHIELD AND
SUPPLEMENTARY FACILITY SHIELDING

depending on the coolant and its decay time before entering the top head area. The loops will be shielded accordingly by the experimenters; i. e., steel or lead sleeves may be installed around the pipes as required.

2. The 1/8-inch insulating air annuli around the facility tubes above the reactor core contain 14 inches of steel in the form of "washers," about one inch thick, spaced throughout the lower half of the tube length. †
3. A large air-cooled facility tube above the core requires that approximately 16 inches of steel be installed for shielding. This steel may be in the form of a helix to allow the air to spiral downward with little restriction of flow. The initial turns of the helix above the core contain about 2 mm of boron to absorb thermal neutrons during operation and limit decay gammas from the steel during after-shutdown removal and facility servicing operations. †
4. Certain facility tubes penetrate the top cover plate to allow the experiment in the core to be positioned during reactor operation or to serve as an outlet for instrument leads. In the case of water-cooled facilities, such penetrations are filled with plugs duplicating the shielding displaced. For air-cooled facilities, the plugs will probably be made of ordinary concrete and come down as far as possible (to the loop side-arm). These plugs are in addition to the steel helix or similar shielding in the tube. †

† The determination of this shielding is discussed in Section XXII, Supplementary Shielding Requirements, since the amount of shielding required is actually governed by after-shutdown operations.

B. Method of Analysis

Three sources of radiation must be attenuated: N^{16} decay gammas from active water in the pressure vessel, core gammas, and secondary gammas produced by neutron capture above the core. The neutron source is not significant due to the large amount of water above the core.

1. Shielding Required for N^{16} Gammas

The flow distributor in the ETR directs inlet primary coolant water to the top head of the pressure vessel. The flow pattern in the upper region of the vessel is rather uncertain, hence, to be safe, it was assumed that all of the inlet water goes directly to the top head region. The activity in the water is principally due to 6.5 Mev decay gammas from N^{16} formed in the core in the fast neutron reaction $O^{16}(n,p)N^{16}$. * The N^{16} decays with a 7.35-second half-life, but the relatively short decay time between the pressure vessel outlet and re-entry to the top head area (about one minute) leaves a significant N^{16} decay gamma source to be attenuated by the top head shield.

The activity of the water leaving the reactor core is 6.6×10^6 disintegrations/cc/sec† due to N^{16} activity. Since one 6.5 Mev gamma is emitted per disintegration, the gamma source at the core outlet is 4.29×10^7 Mev/cc/sec. Consideration of the flow rates and the total water in the system indicates

* Na^{24} activity is about equal to the N^{16} activity in the top head region. However, since the gamma energy is 1.4 and 2.8 Mev from Na^{24} in contrast to the 6.5 Mev from N^{16} , the N^{16} is the determining source for the top head shield. See Appendix X-E for a discussion of the Na^{24} source.

† See Appendix VIII.

that the gamma source due to N^{16} gammas at the inlet to the pressure vessel is 4.05×10^5 Mev/cc/sec.† It takes about two seconds for this water to reach the top head, reducing the source to 3.36×10^5 Mev/cc/sec. (See Appendix X-B.) This was converted to an infinite plane source at the inside surface of the pressure vessel wall, and the shielding required to reduce the radiation to 1/10 tolerance was determined.

Measuring outward from the inside surface of the pressure vessel wall, the μx required (number of mean free paths) to reduce this radiation to 1/10 tolerance is 8.4. (See Appendix X-B.) In the central region of the top head shield the top cover plate of the vessel is only 4 1/2 inches of steel, and therefore 23 inches of magnetite concrete are required for attenuation to 1/10 tolerance. In regions of the top head to the sides of the cover plate, the vessel wall is 2 1/4 inches thick. The additional shielding required for N^{16} gammas was found to be 28 inches of magnetite concrete in these areas.

2. Shielding Required for Core Gammas

a. Sources of Radiation

- (1) Fission and capture gammas from U^{235}
- (2) Capture gammas from non-fissionable core material (steel, aluminum, etc.)
- (3) Capture gammas from the aluminum reflector.

Only high energy gammas are considered (7 Mev being the major component), since lower energy gammas are attenuated to negligible intensities after

† See Appendix VIII.

penetrating the large amounts of material above the core. The U^{235} gamma contribution was obtained on the basis of the fission spectrum from the Los Alamos reactors,* 4-10 Mev gammas being considered as 7 Mev. Capture gammas were obtained on the basis of volume fractions for the metals in the core (including the initial experiments) and the flux distributions from UNIVAC calculations. The total 7 Mev source is 2.4×10^{19} Mev/sec. (See Appendix X-A.)

b. Attenuation of Core Gammas

The 36-inch active section of the core was divided into four equal point sources, each located at the center of a 9-inch section. The core is essentially 47 percent water, 37 percent aluminum, 9 percent air, and 7 percent steel, giving an effective gamma absorption coefficient of $\mu = 0.053 \text{ cm}^{-1}$. (See Appendix XI-B.) Similarly, the 3-foot section above the core (containing control rod guide tubes, facility tubes, etc.) has an effective 7 Mev gamma absorption coefficient of 0.047 cm^{-1} . (See Appendix X-C.) The material above the 3-foot section is considered water alone. (The steel facility tubes are neglected, since their wide spacing in the water does not allow the material to be considered a homogeneous water-steel mixture.)

The gamma source to be attenuated by the top head shield is found by calculating the radiation from the four point sources after attenuation up to this point. (See Appendix X-D.) The total contribution from the core at the outer

* R. Stephenson, Introduction to Nuclear Engineering, McGraw-Hill, 1954.

surface of the top cover plate is 1.45 mr/hr and is reduced to a negligible level by the 23 inches of magnetite concrete required for attenuation of the N^{16} gammas.

3. Shielding for Secondary Gammas

From IBM 650 calculations, the net neutron current in the upward direction directly above the active core was found to be 3.1×10^{12} nv thermal and 1.1×10^{14} nv fast.* The thermal flux is quickly captured in the water and metal just above the core, but the fast flux must first be thermalized. In treating the captures throughout the water and metal above the core, it was found conservative to consider the entire fast flux thermalized and captured one age displacement above the active section of the core. In the first few feet above the core, the metal-to-water ratio is 0.18, giving an age, τ , of 36 cm^2 .

The age displacement is given by τ/λ

where λ = fast neutron relation length, 9 cm

$$\tau/\lambda = 36/9 = 4 \text{ cm}$$

The capture gamma source was thus considered to originate two inches above the active section of the core.

A consideration of the relative volumes and neutron absorption cross sections of the materials above the core showed that about 64 percent of the neutron

* See Figure 18-2. Diffusion coefficients for the core are $D_f = 1.307$ and $D_s = 0.259$.

VERTICAL FLUX IN ETR

captures occur in iron, nickel, chromium, etc., giving essentially one 7 Mev gamma per capture. (See Appendix X-C.) Neglecting the relatively smaller thermal flux, the capture gamma source is $(1.1 \times 10^{14})(0.64)(7) = 4.9 \times 10^{14}$ Mev/cm²/sec. The cross sectional area of the active section of the reactor core is 5574 cm². Converting the capture gamma source to a point source gives 2.7×10^{18} Mev/sec. The source was then attenuated through the material above the core and pressure vessel and was found to equal the core gamma radiation. (See Appendix X-D.)

C. Conclusions and Method of Shield Design

From the foregoing, the following conclusions may be drawn:

1. Along any path normal to the surface of the top head, 6.5 Mev N¹⁶ gammas from active water in the top head region require a μx of 8.4 in the vessel wall and outside the top head shield.
2. 7 Mev core and secondary gammas require a μx of 23.4, measured from a point two inches above the active section of the core along the core center line.

The required top head shield may be laid out on a large scale drawing of the upper sections of the reactor core and pressure vessel. To attenuate the principal source (N¹⁶ gammas) to 1/10 tolerance, the shield was laid out corresponding to a μx of 8.4, measured from the inner wall of the top head along numerous paths normal to the top head surface. Several straight line paths were then drawn to the top head region from the point two inches above the active section of the core. The attenuation along each path was computed, and the top head shield increased as necessary to provide a total μx of 23.4

along each path. Finally, where sections of the shield were the same, or approximately the same, the thicknesses required for each source were increased as required to attenuate the total radiation to 1/10 tolerance. (If a section required the same μx for N^{16} gammas and core plus secondary gammas, the total radiation outside the shield would be 1/5 tolerance.)

As actually designed, the top head shield will be thinner than indicated here. The thickness of the major part of the shield depends on the activity of the water in the top head region. Since the actual flow pattern is not certain, all of the inlet water was assumed to flow to the uppermost regions of the pressure vessel with its flow velocity equal to that at the inlet. On the basis of this design the shield weighs approximately 180 tons and requires six lifts by the 30-ton crane to remove it. Since most of the top shield must be removed and replaced at every two-week refueling, it is highly desirable that the weight be at the bare minimum and still attenuate radiation sufficiently. Therefore, on the recommendation of Phillips Petroleum Company, the shield is being designed primarily on the basis of core radiation, with the possibility of more shielding being added as required for water activity when radiation levels are actually measured during the initial operation of the reactor.

XIX. Bottom Area Shielding

A. General

The sub-pile room contains lower sections of facility tubes which lead from the room to the heat exchangers and other equipment in the experiment cubicles outside the room. The control rod access room contains control and regulating rod drive units, leads and drive units for fission and ion chambers located below the grid plate, and leads from thermocouples and other instruments in lower sections of certain facility tubes. (See Figure 19-1.)

Radiation levels vary throughout the sub-pile room and control rod access room. These variations in radiation are caused by control rod assemblies and facility loops penetrating both the water region below the reactor core and the bottom head. For example, a person standing near the walls of the control rod access room or directly below the control rod drive mechanisms receives substantially less radiation than when standing near control rod units just below the roof of the access room.

B. Radiation Levels In the Bottom Area

1. Radiation In the Sub-pile Room

The large amount of structural steel in the grid plate (9 1/2 inches) and the bottom head (8 1/2 inches) attenuates radiation from active fuel elements and experiments in the reactor core to about 1/10 tolerance for extended access four hours after shutdown following 480 hours operation at 175 MW.

2. Radiation In the Control Rod Access Room

a. Contribution Through the Ceiling of the Control Rod Access Room

The ceiling of the access room is approximately 11 1/2 inches of magnetite concrete encased in a 7/8-inch steel top plate and a 1 1/2-inch steel bottom

CENTRAL SECTION BELOW CORE AND REFLECTOR

mounting plate. Three minutes after reactor shutdown (when access is primarily desired), radiation from the core is less than 1/10 tolerance.

During reactor operation at 175 MW the average radiation from the reactor core is about 140 mr/hr. This increases to a maximum of about 4 times this value in regions below a large water-cooled facility tube penetrating the bottom head. The radiation levels are low enough to permit short-time access (about 15 minutes) for instrument or control mechanism adjustments which may be required during full power operation.

Facility tubes just outside the control rod access room must be adequately shielded by the experimenters. Unshielded loops with active water coolants increase the maximum radiation levels to about 1800 mr/hr, limiting access time to about five minutes. However, loops with highly active coolants such as sodium require that several inches of lead be installed around the tubes to permit access to the control rod access room during operation. To permit access three minutes after shutdown, additional shielding for the loops is not necessary if the loops contain coolants which have not been activated at full power operation. In essentially all instances the concrete provides sufficient attenuation of delayed neutron sources in homogeneous fuel loops, etc., to eliminate the need for additional neutron shielding around the tubes.

b. Contribution Through the Walls of the Control Rod Access Room

The walls are 12 inches of magnetite concrete with a 1/2-inch steel inner form plate. The walls therefore provide shielding equal to the ceiling of

the room and sufficiently attenuate radiation from active coolants (except sodium) in facility loop sections nearby.

3. Areas of Increased Radiation In the Sub-pile Room and Control Rod Access Room

a. Air-Cooled Facility Loops

Air-cooled facilities require a large amount of shielding to be positioned in the facility tube below the core. This shielding may be in the form of a steel helix which allows cooling air to flow downward without appreciable restriction of flow rate. In the case of a large facility (6-inch diameter), this helix is effectively 16 inches thick and has 2 mm of boron in its first turns below the core to absorb thermal neutrons and prevent activation of the tube and helix. The shielding reduces radiation directly below the tube in the sub-pile room to 37 mr/hr four hours after shutdown. In the control rod access room the gamma and fast neutron radiation streaming down the tube is reduced to 390 mr/hr and $500 \text{ n/cm}^2/\text{sec}$ respectively. Three minutes after shutdown the radiation is reduced to less than 1/10 tolerance.

b. Insulating Air Annuli

These annuli contain steel washers providing effectively 12 inches of solid steel to attenuate gammas and scatter neutrons into the surrounding water. This shielding limits the narrow-beam radiation directly below an annulus in the sub-pile room to 3 mr/hr four hours after shutdown. In the control rod access room the gamma and fast neutron radiation streaming during full power operation is limited to 23 mr/hr and $3400 \text{ n/cm}^2/\text{sec}$ respectively.

Three minutes after shutdown the radiation is reduced to less than 1/10 tolerance.

4. Radiation In the Basement Areas

a. Contribution Through the Walls of the Sub-pile Room

The walls of the sub-pile room are four feet of magnetite concrete encased in 1/2- and 3/4-inch steel sheets. This shielding reduces core radiation in the basement areas during reactor operation to less than 1/10 tolerance. It also reduces radiation from typical water-cooled experiment loops in the sub-pile room to about 1/40 tolerance. Radiation is reduced to about tolerance from unusually active loops such as sodium-cooled loops and from loops emitting delayed fast neutrons from fission fragments in homogeneous fuel or destructive tests. To negligible values.

b. Contribution Through the Door to the Sub-pile Room

The door to the sub-pile room is 18 inches of laminated steel. To limit its thickness and weight, the door is located in a section of the room that is relatively removed from facility loops. The door attenuates radiation from active water coolants in the facility loops to 1/10 tolerance.

c. Contribution Through the Door to Control Rod Access Room

A staircase beginning on the basement floor allows entry to the control rod access room. A door at the foot of the stairs is the point of entry to the room, and the key must be obtained from the ETR Operations Office.

Radiation from the access room during operation is scattered up the staircase from the floor. However, the scattered radiation is not intense enough to require the entry door to contain any shielding.

d. Contribution from Experimental Facilities In the Cubicles

Facility loops which penetrate the walls of the sub-pile room lead into the cubicle areas and make such areas inaccessible during reactor operation unless the pipes are suitably shielded. (Suitable shielding would be lead sleeves placed around the pipes.) To preserve established radiation levels in basement areas (1 mr/hr), such active loops and associated experimental equipment are enclosed in cubicles of concrete blocks as required.

C. Method of Analysis

1. Attenuating Materials

a. Core Materials

Absorption coefficients for gammas in the core itself were obtained by homogenizing the materials in the core. The volume fractions of water, aluminum, and steel were computed for the active section of the core (taken as 30 by 30 inches). This includes fuel elements, control elements, and materials in the initial experiments.

Volume fractions for the core* are

Water	0.47
Aluminum	0.37
Air	0.09
Steel (including nickel, etc.)	0.07

b. Grid Plate

The water and steel of the grid plate were homogenized to 65 percent steel and 35 percent water (facility tube penetrations being neglected). The fast and thermal neutron fluxes below the core were attenuated by this mixture. In computing gamma attenuations below the core an attenuation coefficient for the material in the grid plate was not computed as such. Due to the large number of water holes, the percentage of non-metal in the grid plate (including facility penetrations) was computed to be 58 percent. It was assumed that this percentage of the core gamma source is attenuated in the grid plate the same as in the water and other materials below the grid plate. The attenuation of the remainder of the gamma source impinging on the steel sections of the grid plate is such that it need not be considered.

* See Appendix XI-B.

c. "Water" Between the Grid Plate and Bottom Head

The water is penetrated by steel facility tubes and aluminum sections of control rod assemblies. These materials were homogenized, because the gamma relaxation length in water is short (about 12 inches) compared to the distance between the grid plate and bottom head (about 100 inches).

Volume fractions for the various materials were calculated at the protection plate over the cross sectional area of the active section of the core. Above this level, facility tubes are slightly larger in diameter, but are considered the same diameter as the lower sections (see G-E Drawing M-606). This assumption is somewhat conservative because the larger diameter increases the percentage of steel and hence the effective absorption coefficient for the mixture.

The volume fractions for this "water"* are

Water	0.890
Steel	0.0375
Aluminum	0.028

d. Protection Plate

The steel protection plate is penetrated by a large number of holes (facility tubes and control rod assemblies). Since the plate is only two inches thick, it is neglected in computing gamma attenuations.

* See Appendix XI-B.

e. Bottom Head

The stainless steel bottom head (8 1/2 inches thick) is penetrated by control rod assemblies and facility tubes. Volume fractions were computed on the basis of a 30 by 30-inch cross sectional area (corresponding to active core cross-sectional area). Outside this area the bottom head is almost 100 per-cent steel, and attenuations are much greater.

The volume fractions for the central section* are

Steel	0.865
Water	0.0725
Aluminum	0.0196

f. Space Between Bottom Head and Roof of Control Rod Access Room

Directly below the core, the 4 1/2-foot air space is penetrated by the control rod assemblies and facility tubes. Since radiation coming through the bottom head is highly scattered, the metal in the area may be homogenized and the attenuation through the resulting medium computed. This attenuation was included when radiation was computed in the control rod access room.

The volume fractions* are

Steel	0.041
Water	0.027

g. Roof of Control Rod Access Room

This consists of 11 1/2 inches of magnetite concrete with a top plate of 7/8-inch steel and a bottom plate (the mounting plate) of 1 1/2-inch steel.

* See Appendix XI-B.

The roof is considered to be solid, since the only penetrations are for control rod units (70% air, 30% steel) and all other penetrations from the lower sections of facility tubes are plugged to duplicate the shield. Gamma absorption coefficients for each region are computed and tabulated in Appendix XI-B.

2. Determination of Radiation In the Sub-pile Room

a. Sources of radiation and areas of decreased attenuation in the shielding below the reactor core follow:

- 1) Decay gammas from the steel grid plate
- 2) Decay gammas from the reactor core
- 3) Air-cooled facility tubes penetrating the water below the core and the bottom head
- 4) Insulating air annuli around facility tubes penetrating the water below the core and the bottom head
- 5) Water-cooled facility tubes penetrating the bottom head
- 6) Control and regulating rod units penetrating the bottom head
- 7) Fission and ion chamber assemblies penetrating the bottom head
- 8) Streaming down the annulus around the pressure vessel
- 9) Streaming down in-shield pipes

(1) Decay Gammas from the Steel Grid Plate

A cylindrical source treatment was used to compute the radiation from the steel grid plate in the sub-pile room. The detector was assumed to be situated on the axis of a cylindrical source (see Appendix XI-D). The radiation level was determined to be 0.002 mr/hr, which is only 1/3000 tolerance.

(2) Decay Gammas from the Reactor Core

Using the gamma sources from Appendix XI-A and a cylindrical source treatment, a total radiation of 0.3 mr/hr was obtained. The total radiation in the sub-pile room from core gammas four hours after shutdown following 480 hours operation at 175 MW is shown below.

<u>Gamma Source (Mev)</u>	<u>Radiation Level in the Sub-pile Room (mr/hr)</u>
2.8	0.037
2.5	0.18
2.1	0.060
1.65	<u>0.035</u>
Total	= 0.31

This is about 1/20 tolerance.

(3) Air-cooled Facility Tubes Penetrating the Water Below the Core and the Bottom Head

Air-cooled facility tubes allow considerable neutron and gamma radiation to penetrate the water below the core. In all cases experimenters will be required to provide sufficient shielding for their experiments to maintain the established radiation levels throughout the reactor building. A suggested method for meeting shielding requirements for a 6-inch diameter air-cooled experiment tube is given here to show the scope of the problem. The required shielding is 16 inches of steel in the tube. This may be in the form of a helix which provides a good radiation shield and permits the downward flow of cooling air without appreciable restriction. Radiation at

the end of the tube is about 37 mr/hr, but is reduced to about 1/10 tolerance several inches to the side of the tube. The first few turns of the helix are borated. This boron has the equivalent thickness of 2 mm. During reactor operation, the boron serves to depress the thermal neutron flux in the steel. This prevents activation of the helix and lower sections of the air tube as well as prevents radiation damage to seals in the bottom head. The helix also scatters fast neutrons out of the tube and into the surrounding water. The fast neutron flux at the end of the tube is reduced to 5×10^6 n/cm²/sec. This is low enough to prevent activation of materials in sub-pile room areas when the neutrons are thermalized and captured. (See Appendix XI-E for method of analysis.)

(4) Insulating Air Annuli Around Facility Tubes Penetrating the Water Below the Core and the Bottom Head

A steel casing is installed around each facility tube to provide a 1/8-inch insulating air annulus. These annuli direct dangerous beams of gamma radiation from active elements and experiments in the core into the sub-pile room after shutdown. The required shielding is 12 inches of steel, which may be installed in the annuli in the form of washers spaced throughout the annuli. (See Figure 19-1 and Appendix XI-F.)

(5) Water-cooled Facility Tubes Penetrating the Bottom Head

These tube penetrations contribute to increased radiation levels, since the steel displaced in the bottom head is replaced by water. The radiation in these circumstances is found to be approximately 28 mr/hr just below the bottom head. However, the water-filled tube extends at least an additional

two feet below the bottom head. Access is possible to within a few inches of the sides of the tube, and here the radiation is reduced to about 1/10 tolerance, since this region is shielded by the bottom head.

(6) Control and Regulating Rod Units Penetrating the Bottom Head

The control rod units are essentially 70 percent water and 30 percent steel, the steel being in the form of several concentric tubes. Replacing the bottom head with a homogeneous mixture of this material indicates that radiation is 8 mr/hr immediately below the control rod unit penetrations. However, water in the tubes extends about two feet below the bottom head and there is essentially a three-inch steel section in the rod units immediately below the bottom head. The increased attenuation in these materials reduces the radiation considerably. Access is normally possible to within several inches of the sides of the control rod units, and radiation in such areas is reduced to about 1/10 tolerance.

The regulating rod units penetrating the bottom head are solid steel and essentially duplicate the shielding displaced. Annuli around the one-inch diameter steel rods are only a few mils in thickness. This is a negligible gap in the 8 1/2-inch thick bottom head. Since access is possible only to the sides of the rods, radiation levels are less than 1/10 tolerance.

(7) Fission and Ion Chamber Assemblies Penetrating the Bottom Head

Two fission chambers and two ion chambers are situated beneath the grid plate in the region of the aluminum reflector. These chambers are posi-

tioned in water-filled thimbles which extend down to the control rod access room. Because these thimbles are water-filled, attenuation is not decreased in these areas. Below the chambers are 20-inch long plugs of magnetite concrete which compensate for the 8 1/2 inches of steel in the bottom head that has been displaced by water. They also compensate for air gaps of up to 12 inches in the chambers. When access to sub-pile room areas is desired, the chambers are in their lower positions, which places the concrete plugs in the bottom head region. The ion chambers consist of essentially 6 inches of air and 30 inches of graphite, which attenuates after-shutdown gammas equally as well as water. A 1/2-inch diameter tube spirals through the plug and continues straight down to the control rod access room. This tube is solidly packed with instrument leads and does not detract from the over-all shielding in the chamber regions. Water remains in the thimble while the chamber is being removed from the control rod access room, thus eliminating excessive radiation in the sub-pile room and in the control rod access room below.

(8) Streaming Down the Annulus Around the Pressure Vessel

A certain amount of radiation streams into sub-pile room areas from the air annulus around the pressure vessel. The annulus is nominally a one-inch gap, except around the bottom head where the gap is reduced to 1/2 inch. The gap is terminated by about four inches of steel. The core was divided into four equal point sources, and radiation was computed for the path of least attenuation down the annulus around the bottom head. The radiation

through the gap is attenuated by about one inch of steel in the bottom head and the four inches of steel beneath the gap. (See Figure 19-2.) Radiation along paths other than that shown can only reach the sub-pile room by scattering from the walls on either side of the annulus. This scattering reduces such contributions to negligible values. The radiation streaming into the sub-pile room in this small area four hours after shutdown with a full core of active elements is 1.5 mr/hr.

(9) Radiation Streaming Down In-shield Pipes

When not in use to house a facility loop with its associated insulation, the 8-inch diameter in-shield pipes contain only air. A conservative estimate of the effect of the empty pipe was obtained by treating the core as four equal point sources and computing the radiation at the inner pipe surface after core radiation was attenuated along the weakest path. (See Figure 19-2.) The radiation at the inner pipe surface is 3.52×10^4 Mev/cm²/sec. The concrete on the opposite side of the pipe is considered to be an infinite slab scatterer, and radiation reflected into the sub-pile room is about 0.4 mr/hr.*

b. Conclusion

In addition to the water below the core, the shielding for insulating air annuli around the facility tubes, and shielding for air-cooled facility tubes, the steel grid plate and bottom head of the pressure vessel sufficiently attenuate

* The methods discussed in ORNL 1575 were used here.

GAMMA RADIATION FROM CORE THROUGH IN-SHIELD PIPE
AND AIR GAP AROUND VESSEL TO SUB-PILE ROOM

core radiation in the sub-pile room. This shielding permits extended access to the sub-pile room about 4 hours after shutdown.† It is not necessary to remove the active fuel elements and experiments from the core before access to the room is possible. However, highly active coolants in facility loops, such as sodium, have to be removed or suitably shielded by the experimenters. Whole-body radiation is about 1/10 tolerance, but depends on location of the individual in the room. The upper limit of radiation is about five times tolerance just below the water-cooled and air-cooled facility tubes penetrating the bottom head. However, this exposure is limited to instances such as the installation of facility tubes. In these cases, active fuel elements should have been removed from the core, thus reducing radiation to less than 1/10 tolerance.

The various radiation levels are listed in Table I. These are the maximum levels and are considerably reduced when active fuel elements are removed from the core after reactor shutdown.

† In outer sections of the sub-pile room, core radiation is reduced to negligible values by the increased attenuation in the lower sections of the biological shield.

TABLE I

Radiation in Sub-pile Room Four Hours after Reactor Shutdown
(Full Core of Active Fuel Elements and Experiments)

<u>Contributing Sources and Special Areas of Interest</u>	<u>Radiation Level in Sub-pile Room (mr/hr)</u>
1. Decay gammas from the reactor core	
Average in room	0.3
2. Decay gammas from grid plate	
Average in room	0.002
3. Core radiation in regions of control rod unit penetrations in the bottom head	
Directly below control rod penetrations	8.0
4. Core radiation in regions of water-cooled facility penetrations in the bottom head	
Directly below water-cooled facility penetrations	28.0
5. Radiation from 6-inch diameter air-cooled facility tube with 16 inches of steel (helix) in tube	
Directly below the air-cooled facility penetration	37.0
6. Radiation directly below air annulus around a water- cooled facility tube (with 12 inches of steel in annulus)	3.0
7. Radiation streaming down the annulus around the pressure vessel	1.5
8. Radiation streaming down an empty in-shield pipe	0.4

3. Determination of Radiation In the Control Rod Access Room

a. Radiation In Control Rod Access Room During Reactor Operation

Sources of radiation and areas of decreased attenuation in the shielding are the following:

- 1) U^{235} core gammas and capture gammas from the metal and water in the core and reflector
 - 2) Secondary gammas from neutrons being captured in the metal and water below the core
 - 3) Neutrons from the core
 - 4) Active coolants in facility loops in the sub-pile room
 - 5) N^{16} gammas in the active water below the core
 - 6) Core radiation streaming down an air-cooled facility tube
 - 7) Radiation streaming down insulating air annuli around facility tubes
 - 8) Control rod assemblies penetrating the concrete roof
 - 9) Decreased attenuation due to a water loop penetrating the bottom head
 - 10) Fission and ion chamber assemblies below the grid plate
- (1) U^{235} Core Gammas and Capture Gammas from Metal and Water In the Core and Reflector

The only gammas of interest are 7 Mev, since it is found that the low energy gammas are attenuated to negligible intensities after penetrating the large amount of attenuating material below the core. The total source of 7 Mev gammas from the core during operation is 2.4×10^{19} Mev/sec.*

The 7 Mev source actually includes 4 to 10 Mev fission gammas. The

* See Appendix X-A.

core is divided into four equal point sources, each at the center of a 9-inch section of the active core. Each source is attenuated through the material along a straight line path to the lower face of the access room roof. This produces a radiation level of 43 mr/hr in the access room. (See Appendix XI-G.)

(2) Secondary Gammas from Neutrons

It is assumed that the fast neutrons are thermalized and captured within one age displacement below the active section of the core in the grid plate. (The 3 3/4 inches of water between the ends of the active sections of the fuel elements and the top of the grid plate are neglected.) Taking the grid plate as 65 percent steel and 35 percent water, one age displacement is given by

$$\frac{\tau}{\lambda} = \frac{100 \text{ cm}^2}{9 \text{ cm}} = 11 \text{ cm}$$

A consideration of relative neutron captures in the grid plate shows that essentially all of the captures may be assumed to occur in the steel, giving one 7 Mev gamma per capture.

From IBM-650 calculations, the fast neutron current leaving the bottom of the core is given by

$$J_+ = 1.18 \times 10^{14} \text{ n/cm}^2/\text{sec}$$

where $D = 1.31 \text{ cm}$

The neutron source is assumed to be captured 11 cm below the top of the grid plate with the emission of one 7 Mev gamma per capture, producing

a source of 8.3×10^{14} Mev/cm²/sec. The gamma mean free path in a water hole is longer than the remaining thickness of the grid plate. The water holes, including experiment holes, constitute 58 percent of the grid plate. It is therefore assumed that 58 percent of the source is effective, since the remainder is greatly attenuated in the steel sections of the grid plate.

The source is converted to a point source by multiplying by the cross sectional area of the core. The source is then 2.8×10^{18} Mev/sec 11 cm below the top of the grid plate. Using a point source treatment, the radiation in the access room is 100 mr/hr.

(3) Neutrons from the Core

At the bottom of the core the above-thermal neutron current in the downward direction is 1.18×10^{14} n/cm²/sec. This is attenuated by nine feet of water penetrated by aluminum and steel in the control rod assemblies, facility tubes, etc. The effective relaxation length in this mixture is about 9 cm. This gives an attenuation of $e^{-\frac{274.5}{9}}$ or 5.7×10^{-14} . In addition, the attenuation in the roof of the room is $e^{-\frac{30.5}{8.2}} = 0.024$. The neutron source is thus attenuated to $0.2 \text{ n/cm}^2/\text{sec}$, which is less than 1/100 tolerance.

(4) Active Coolants in Facility Loops In the Sub-pile Room

a. Air-cooled Loop

In general, the air in an air-cooled loop does not become active enough to be a radiation hazard in the control rod room.

b. Water-cooled Loop

A typical loop is water-cooled. During operation, a 6-foot long section of a 3-inch diameter loop, two feet above the access room roof, produces a radiation level of about 340 mr/hr in the control rod access room.* If six of the eight through-loops are water-cooled, the radiation in the room is about 1200 mr/hr. An analysis of the attenuation lost in the roof of the access room due to control rod unit penetrations shows that directly below such areas the radiation is essentially unchanged. Although attenuation in the roof area is decreased, only a small fraction of a loop is seen through the penetration.

c. Sodium-cooled Loop

Radiation from a sodium-cooled loop is especially severe, averaging 3.5×10^5 mr/hr, which is 50,000 times tolerance.* If such unusually active loops are not drained of active coolant, access to the room is not possible unless the loops are suitably shielded. One method of shielding is half-sleeves of lead bolted together around the tube in the sub-pile room. A 5-inch lead sleeve reduces the radiation in the control rod access room to about 300 mr/hr, which permits the short-time access desired.

d. Fissionable Material in Loops

From MTR and NRX experience a delayed neutron source of up to 10^4 n/cm²/sec may exist from a facility loop employing circulating

* The method of analysis is the same as used in Appendix IX-D.

fuel. The 30.5 cm of concrete in the access room roof will attenuate this by

$$e^{-\frac{30.5}{8.2}} = 0.024$$

Radiation in the room is

$$10^4 (0.024) = 240 \text{ n/cm}^2/\text{sec}$$

The average neutron energy is about 0.75 Mev and the radiation in the most severe case is about three times tolerance.

Radiation streaming down control rod penetrations from fissionable materials is only slightly attenuated by the steel and air. Therefore, a considerably higher source is seen in some limited areas. However, short-time access is still possible.

(5) Active Water Below the Core

After passing down through the core, primary coolant water continues downward to the protection plate before it reverses flow direction. However, some of the highly active water passes through gaps in and around the sides of the protection plate and enters regions a few feet above the bottom head. The N^{16} gamma source from water emerging from the core and entering the outlet is 4.3×10^7 Mev/cc/sec. (See Section XVI.) This source strength is considered to extend to just below the protection plate, leaving three feet of water and various mechanisms* to attenuate this source. This last three feet of water is essentially stagnant. The source

* Aluminum control rod units and steel facility tubes.

is converted to a plane source by multiplying by one relaxation length for 6.5 Mev gammas in the water, which gives

$$(4.3 \times 10^7) (30) = 1.29 \times 10^9 \text{ Mev/cm}^2/\text{sec}$$

where $\lambda = 30 \text{ cm}$

Using the formula for an infinite plane source, the radiation in the access room is 0.78 mr/hr. This is about 1/10 tolerance.

(6) Core Radiation Streaming Down an Air-cooled Facility Tube

A similar treatment to that used in the sub-pile room analysis is employed here. (See Appendix XI-E.) It is assumed that there are 16 inches of steel shielding in the tube, and that the tube penetration in the roof of the access room is filled with a magnetite concrete plug. This concrete plug projects about 12 inches above the top of the roof and into the facility tube. Radiation coming down the tube sees at least six inches of the plug before being scattered out to be further attenuated in the access room roof. Gamma radiation is reduced to about 390 mr/hr. This is not significantly higher than the average radiation in the room.

As shown in Appendix XI-E, the fast neutrons in the air tube at the bottom head are $5 \times 10^6 \text{ n/cm}^2/\text{sec}$. After scattering in the steel shielding in the tube the neutron energy is reduced to about 0.5 Mev. From Oak Ridge experimental data, the relaxation length in the effective 20-inch magnetite concrete plug in the tube is about 5.5 cm.* Attenuation is $e^{-\frac{50.8}{5.5}}$, reducing the radiation to $500 \text{ n/cm}^2/\text{sec}$. This is about 7 times tolerance, thus permitting the desired access to the room.

* MTR Handbook, ORNL 963, Shielding Section.

(7) Radiation Streaming Down Insulating Air Annuli Around Facility Tubes

The 1/8-inch insulating air annuli around facility tubes contain 12 inches of steel shielding in the form of washers spaced throughout the annuli.

(See Figure 19-1.) During full power operation, gamma radiation is reduced to 23 mr/hr and fast neutrons to $3400 \text{ n/cm}^2/\text{sec}$ after attenuation in the steel washers and the roof of the access room. (See Appendix XI-F.) The neutron radiation is about 50 times tolerance, but is very narrow-beam radiation and low enough for the access desired. Radiation just below the control rod units which penetrate the access room roof is not significantly higher, since the radiation emerging from the annuli is highly scattered by the time it reaches the access room roof.

(8) Control Rod Assembly Penetrations In the Bottom Head and Access Room Roof

These penetrations in the access room roof are 2 1/2 inches in diameter and are essentially 30 percent steel and 70 percent air. The attenuation lost in the roof in these regions is a factor of 6. The geometry is such that the whole core is seen from immediately below the penetration. The core radiation in these areas is 840 mr/hr. Access to these regions normally should not be required to perform adjustments while the reactor is at full power, even though the radiation level is satisfactory for short time access.

Attenuation lost in control rod unit penetrations of both the bottom head and access room roof is compensated for by the small core area seen

with this particular geometry. Therefore, the radiation is no higher than the average intensity in the room.

(9) Water-cooled Facilities Penetrating the Bottom Head

In regions of the access room below water-cooled facilities penetrating the bottom head increased core radiation is received. The highest intensity is received below a large water-cooled facility (5-inch diameter penetration of the bottom head). Water in the tube extends below the bottom head. At least two feet of water is seen from this point, instead of the 8 1/2 inches of steel in the bottom head, thereby increasing radiation by a factor of 15. However, since no more than about one-fourth of the total core volume is seen, the over-all radiation increase is reduced to a factor of 4. This is actually the upper limit of core radiation in the access room when adjustments are being made during full power operation or three minutes after shutdown. Core radiation passing through both a large water penetration in the bottom head and a control rod unit penetration in the roof of the access room is increased by a factor of 9. However, this increased radiation is very narrow-beam radiation and is seen in only very limited areas of the access room.

(10) Fission and Ion Chambers Located Below the Grid Plate

The 20-inch magnetite concrete shielding plugs below the chambers essentially duplicate the shielding lost in the air sections of the chambers and in the water thimble penetrations of the bottom head. Between the bottom head and the inside of the access room, the water-filled thimbles

continue to the access room roof. Only the 1/2-inch diameter tube packed with instrument leads continues through the roof, and the remainder of the penetration is filled with a steel plug. The shielding is therefore sufficient in the access room regions.

b. Core Radiation In the Control Rod Access Room Three Minutes After Reactor Shutdown

This radiation is made up of decay gammas from U^{235} , fission gammas, and neutrons, since three minutes after shutdown the reactor is still operating at considerable power.

(1) Decay Gammas From U^{235} In the Core

Using the formula of K. Way* for U^{235} decay gammas for short times after shutdown, there is a source of 1.02×10^{19} Mev/sec three minutes after shutdown. These gammas are essentially 2.8 to 3 Mev. The core is divided into four equal point sources, each at the center of a 9-inch section of the active core and the total radiation in the access room is 0.09 mr/hr, which is 1/100 tolerance.

(2) Radiation From Core at Decreased Power Level

Fourteen percent negative reactivity is assumed in the black rods. When the rods are inserted at shutdown, the reactor power level immediately

* Taken from MTR Handbook, ORNL 963, Shielding Section.

$$Q = 6.1 F \left[t^{-0.2} - (T + t)^{-0.2} \right]$$

drops a factor of 20. At the end of three minutes, the 80-second mean lifetime (55.6 second half-life) delayed neutron group is the principal source of non-fission neutrons.* The relative yield of this group is 0.025 neutrons per 100 neutrons emitted. At three minutes after shutdown the power is reduced to

$$\frac{1}{\left(\frac{14}{0.025}\right) e^{\frac{180}{80}}} \approx \frac{1}{5000} \text{ full power}$$

Therefore, this contribution to the total radiation in the control rod access room is obtained by taking $1/5000$ of the radiation levels obtained from the core at full power.

(3) Radiation from Active Coolants in Facility Tubes

(a) Water

Radiation from a water-cooled facility is reduced to negligible values due to the rapid power decay and short (7.35 second) half-life of the N^{16} gamma source found during reactor operation at full power.

(b) Sodium

Since Na^{24} has a long half-life (15 hour), radiation from active sodium in a facility loop is the same three minutes after shutdown as during reactor operation. However, if the experiment is such that inactivated sodium can be circulated to continue cooling after shutdown, the radiation is reduced since the coolant is being activated at a low power level.

* Stephenson, Introduction to Nuclear Engineering, McGraw Hill, 1954.

(c) Fissionable Material

In the case of a loop with circulating fissionable material, the delayed neutron source three minutes after shutdown is about 1/5000 its value during reactor operation at full power. It has been shown that the average delayed neutron intensity in the access room during operation is $240 \text{ n/cm}^2/\text{sec}$. Three minutes after shutdown this is reduced to about $0.05 \text{ n/cm}^2/\text{sec}$, which is negligible.

c. Conclusion

Table II summarizes the radiation levels immediately below the control rod access room ceiling during full power operation and three minutes after shutdown.

Three minutes after shutdown, radiation from the core and from active water in unshielded facility loops in the sub-pile room is reduced to 1/10 tolerance. Highly active coolants left in facility loops and radiation streaming down air-cooled loops must be suitably shielded by the experimenters, as indicated in Table II.

During full power operation, radiation in the access room is about 1340 mr/hr from the reactor core and from active water coolants in unshielded loops (sum of items 1, 2 and 3 in Table II). Below certain large water-cooled facilities penetrating the bottom head this radiation increases to an upper limit of 1800 mr/hr (sum of items 3 and 8 in Table II). This permits access to the room for about five minutes before half weekly tolerance is

received. Therefore, even in these particularly high radiation areas, access is possible during full power operation to make minor adjustments of control drive mechanisms and certain experimental equipment in the control rod access room. The neutron radiation levels in the room shown in Table II are not excessive. The most significant neutron contribution is below an air-cooled facility tube, and this is only $500 \text{ n/cm}^2/\text{sec}$ or seven times tolerance compared to the gamma radiation levels which are about 35 times higher. The higher narrow-beam neutron radiation streaming out of the air annuli is not prohibitive (one hour exposure for weekly tolerance).

Shielding for facility loops must be provided by the experimenters. Particularly bad cases are air-cooled and sodium-cooled facilities. If it is found that access to the room during full power operation is seldom necessary, the shielding suggested may be limited considerably by the ETR Operations Office to permit access to sub-pile areas only after shutdown.

4. Determination of Radiation in Basement Areas Outside the Sub-pile Room

a. Radiation Outside Sub-pile Room Walls

(1) Contribution from the Core

During reactor operation radiation in sub-pile areas from neutron capture gammas below the core is approximately twice core gamma radiation.*

An analysis of the sub-pile room walls was performed on the basis of the

* See treatment of radiation in the control rod access room during reactor operation.

TABLE II

RADIATION IN CONTROL ROD ACCESS ROOM

(Values quoted are for a level immediately below the concrete and steel roof of the room)

Contributing Sources	Radiation Levels	
	3 minutes after shutdown Average in Room (mr/hr)	During reactor operation at 175 MW Average in Room (mr/hr)
1. Core gammas	0.10	43
2. Secondary gammas	0.02	100
3. N ¹⁶ gammas from six unshielded water-cooled facilities in sub-pile room	Negligible	1200
4. N ¹⁶ gammas from active water in lower region of pressure vessel	Negligible	0.8
5. Radiation below air-cooled 6-inch dia. facility tube, with 16-inch steel helix with boron and 3 1/2-foot magnetite concrete plug in roof penetration		
Gammas	0.1	390
Fast Neutrons	Negligible	500 n/cm ² /sec
6. Radiation below insulating air annuli around facility tubes with 12 inches of steel washers in the annuli		
Gammas	Negligible	23 mr/hr
Fast Neutrons	Negligible	3400 n/cm ² /sec

TABLE II (CONT'D)

<u>Contributing Sources</u>	<u>Radiation Levels</u>	
	<u>3 minutes after shutdown</u> <u>Average in Room</u> <u>(mr/hr)</u>	<u>During reactor operation at</u> <u>175 MW</u> <u>Average in Room</u> <u>(mr/hr)</u>
7. Core neutrons	Negligible	Negligible
8. Water-cooled facilities penetrating bottom head (core gamma radiation)	0.4	570
9. Fast neutrons from loops with large amount of fissionable material circulating (highest values)	Negligible	240 n/cm ² /sec
10. Sodium-cooled facility decay gammas with 5-inch lead sleeve	300	300

capture gamma source. The source was considered to be a point 11 cm below the top of the grid plate and to emit 2.8×10^{18} Mev/sec of 7 Mev gammas.*

Several straight-line paths were drawn on G-E Drawing M-596 between the point source and basement areas outside the sub-pile room. The amount of material along each path was measured and the attenuations computed. It was found that radiation from the core is attenuated to negligible levels of about 10^{-6} mr/hr. Radiation seen through the 18-inch steel access door is similarly reduced to considerably less than 1/10 tolerance.

(2) Contribution from Active Coolants In Facility Loops In the Sub-pile Room

The large amount of concrete in the sub-pile room walls attenuates radiation to about 1/10 tolerance in the outside basement areas, even in the case of unusually active loops. Radiation from a 10-foot length of a 3-inch diameter water-cooled facility tube one foot from the wall is reduced to 0.03 mr/hr. Radiation from a 2-inch diameter sodium-cooled loop without shielding is reduced to about 5 mr/hr.† Although the latter is an unusual case, the shielding is adequate. In most cases loops penetrate the sub-pile room walls and lead to the respective cubicles in the basement areas.

* See treatment of radiation in the control rod access room during reactor operation.

† The method of analysis is the same as used in Appendix IX-D.

b. Radiation Outside Sub-pile Room Door

Originally, a thick concrete door was strongly favored, but an 18-inch thick laminated steel door has been selected for reasons of economy. The door is relatively removed from highly active facility loops, since it is not as effective a gamma shield as the sub-pile room walls and possesses no real fast neutron attenuation characteristics.

Radiation from water-cooled loops 10 feet away from the door is reduced to about 0.7 mr/hr. Radiation from an unshielded sodium loop 10 feet away is reduced to about 140 mr/hr. Such an unusually active loop will either be located far from the door or will be shielded with lead sleeves. If this is not the case, only short-time access will be permissible near the door. Similar restrictions, or additional shielding in the form of paraffin or concrete blocks, will be required for loops containing homogeneous fuel or fission fragments from highly destructive tests which may at some time be located near the door.

c. Radiation Outside Control Rod Access Room Door

The dominant radiation sources are active coolants in facility tubes in the sub-pile room and air-cooled facility tubes. The treatment of scattering from the concrete floor of the access room indicates that the radiation scattered up the staircase is less than 1/10 tolerance.* With initial experiments in the core no shielding problem exists unless radiation levels immediately below the roof of the access room are about 10^7 tolerance. This

* ORNL 1575

is an unlikely situation, even for highly active sodium-cooled loops which are not shielded. The most formidable source would be from the improbable situation of an unshielded air-cooled facility tube penetrating both the access room roof and the bottom head of the pressure vessel. Such a situation would require about 1 1/2 feet of concrete in the access room door, but would also eliminate the desired access to the control rod room.

d. Conclusion

The walls of the sub-pile room reduce radiation from the reactor core and from active coolants in facility loops in the sub-pile room to less than 1/10 tolerance in outside basement areas. Unusually active coolants in unshielded loops in the sub-pile room increase the radiation level to about tolerance in some areas, but this is still satisfactory. The large amount of concrete in the walls attenuates delayed neutrons from unshielded loops containing fissionable material to negligible values.

The 18-inch steel sub-pile room door reduces radiation from the water-cooled experiments to 1/10 tolerance. However, due to reasons of economy the door was not made as efficient a radiation shield as the walls of the room. Therefore, if at some time highly active coolant loops are located near the door (such as loops containing active sodium and large delayed neutron sources), access to areas outside the door has to be restricted, or suitable shielding provided around the particular loops or near the door.

The radiation levels in the basement areas outside the sub-pile room are summarized in Table III.

TABLE III

Radiation in Basement Areas Outside Sub-pile
Room During Reactor Operation

<u>Contributing Sources</u>	<u>Radiation Level (mr/hr)</u>
Outside sub-pile room walls	
From water-cooled facilities	0.2
From a sodium-cooled facility	5.0
From loops with delayed fast neutrons (homogeneous fuel or destruction tests)	Negligible
Outside of 18-inch thick steel door	
Two water loops 10 feet away	0.7
Sodium loop 10 feet away	140.0
From loops with delayed fast neutrons	Up to 10^4 n/cm ² /sec

XX. CANAL SHIELDING

A. Description

The planned 20-foot water depth in the canal allows 15 1/2 feet of water above the active sections of the fuel elements. This water depth reduces radiation above the canal to 1/35 tolerance when 65 spent fuel elements are placed in the canal 2 1/2 hours after reactor shutdown. Reduction of the water depth to 19 feet still limits radiation to 1/10 tolerance.

To reduce the radial gamma radiation to 1/10 tolerance, the lower sections of the canal walls consist of six feet of portland (regular) concrete (density 2.35 gm/cc). However, the storage section walls, adjacent to experimental areas on the console floor, are seven feet thick. The upper sections of the walls are tapered to a two-foot thickness at the canal parapet, beginning seven feet above the canal floor. The seven feet of regular concrete in the canal floor attenuates radiation to 1/60 tolerance in areas below the floor.

A 2 1/2 foot thickness of magnetite concrete (density 3.5 gm/cc) between the pressure vessel and the canal water limits radiation during reactor operation to 1/10 tolerance in areas around the top of the canal parapet.

If a hot fuel element should accidentally be dropped in the canal, 8 1/2 inches of iron below the canal pit limits radiation in the sub-pile room to 580 mr/hr. This allows access to the sub-pile room for over half an hour before a weekly tolerance dose is received. If the fuel element is not quickly retrieved, the sub-pile room can be evacuated.

Water activity in the canal is not sufficient to necessitate forced water circulation.

B. Method of Analysis

The reactor is to be refueled after 20 days (480 hours) operation. Up to 65 fuel element sections can be replaced. It is estimated that the elapsed time will be 2 1/2 to 6 hours between reactor shutdown and the transfer of a significant number of the spent elements to storage racks in the canal. Calculations for the canal shielding are based on a source consisting of 65 fuel elements 2 1/2 hours after shutdown, following 480 hours operation at 175 MW. Radiation in the upward direction is to be attenuated by water, and in the radial direction by about three feet of water followed by portland concrete, or by concrete alone. In all cases, attenuation was computed for 1/10 tolerance (0.75 mr/hr) or less.

Shielding formulas, gamma absorption coefficients, and buildup factors were taken from APEX 176. However, absorption coefficients for regular concrete were obtained from NBS Handbook 55, and for magnetite concrete from calculations based on mix constituents. (The latter is discussed in Appendix IX-A.)

a. Radiation in the Canal

(1) Fuel Elements

The fuel elements are removed from the core and placed in storage racks in the canal, each rack holding 36 elements. The procedure consists of:

- (a) Shutdown of the reactor
- (b) Removal of certain experiments from the core (optional)
- (c) Lowering the water level in the pressure vessel

- (d) Removal of the top head shield
- (e) Removal of the pressure vessel head
- (f) Placement of an access platform around the top of the vessel
- (g) Lifting of each element from the core with grappling equipment (operated from canal parapet), discharging it through the transfer chute and placing it in the storage rack.

The last loading is the determining source for the canal shielding. Elements from other refuelings which have decayed from 20 to 120 days are at least a factor of 10 lower in gamma intensity.

(2) Other Sources of Gamma Radiation

Other sources of radiation in the canal are not significant. If the entire 94 (2S) aluminum reflector pieces were stored in the canal six hours after shutdown following 60 days irradiation at an average thermal flux of 7×10^{13} n/cm²/sec, they would represent only one percent of the radiation from the fuel elements considered. The four beryllium reflector pieces would contribute even less than the aluminum. * It also seems unlikely that any experimental apparatus placed in the canal will surpass the fuel elements in gamma intensity.

(3) Sources of Neutrons

The fuel element storage racks are lined with cadmium to absorb thermal neutrons and to prevent the possibility of criticality from stored fuel.

* These values are on the basis of Oak Ridge empirical data for irradiated commercial materials (MTR Handbook, Vol. I, EKF-1).

Thus the fuel elements themselves do not represent a significant neutron source. The 20 feet of water in the canal attenuates fast neutrons by a factor of at least 10^{-14} (assuming about 18 1/2 feet of water above an antimony-beryllium or similar source on the canal floor). In the radial direction there is at least 10^{-10} attenuation from the three feet of water and the six or seven feet of concrete. The canal shielding is therefore designed for neutron sources less than 10^{11} nv.

b. Gamma Source Considered

The fission product decay gamma spectrum was computed for short times after shutdown following 480 hours operation. (See Appendix XI-C.)

Fission product gammas of 1.10 to 2.80 Mev were considered and decay curves plotted (Figure A11-1).

3. Results and Discussion

a. Radiation Above the Canal

The group of 65 fuel elements was divided into four equal point sources and the contribution due to each was computed for each gamma energy at 2 1/2 hours after shutdown (see Appendix XII-A). Table I lists the contribution of each energy group to the total radiation. The contributions from Source 4 and the 1.1 - 1.4 Mev gammas were found to be negligible. The total gamma radiation above the canal from the fuel elements is 0.2 mr/hr, which is 1/35 of tolerance dose rate. This indicates that a 19-foot water depth in the canal is sufficient for shielding, since the dose rate is then 1/10 tolerance.

TABLE I

Radiation Above the Canal with a 20-foot Water Depth

<u>Gamma Energy</u> (Mev)	<u>Source 1</u> (mr/hr)	<u>Source 2</u> (mr/hr)	<u>Source 3</u> (mr/hr)
1.65	0.007	0.001	Negligible
1.8 - 2.2	0.019	0.004	0.001
2.4 - 2.6	0.069	0.015	0.003
2.8	<u>0.020</u>	<u>0.005</u>	<u>0.001</u>
Totals	0.115 mr/hr	0.025 mr/hr	0.005 mr/hr

b. Radial Gamma Radiation

(1) Radiation Outside the Canal Walls

Here, the 65 element group is approximated by a cylindrical source with a radius of 13.6 inches and a height of 36 inches. There are three feet of water between the elements and the canal wall. The source terms and gamma energies are similar to those used in computing the radiation above the canal. It is found that six feet of ordinary concrete (density 2.35 gm/cc) are required to attenuate radiation to 1/10 tolerance. The results for each energy group are given in Table II. The upper sections of the walls are tapered as shown in Figure 20-1, since the oblique path of radiation through the attenuating material is the longest path.*

* At these gamma energies, the shielding effectiveness of one foot of regular concrete is equivalent to about 2.2 feet of water.

CANAL STORAGE SECTION

TABLE II
Radiation Outside Canal Walls

<u>Gamma Energy (Mev)</u>	<u>Dose Rate (mr/hr)</u>
1.65	0.05
1.8 - 2.2	0.10
2.4 - 2.6	0.41
2.8	<u>0.10</u>
Total	0.6 mr/hr

(2) Radiation from Storage Racks Against the Canal Wall

In the spent element storage area, adjacent to experimental areas on the console floor, the canal walls are seven feet thick. Since storage space in the canal is limited, it may be necessary to store elements which have been recently removed from the core against the walls. The storage rack construction is such that there will be a minimum of three inches of water between the fuel elements and the canal wall. This water is neglected in the calculations and the element array is considered a cylinder with its periphery only about 1 1/2 inches from the wall.

Radiation just outside the seven feet of concrete is reduced to 2.5 mr/hr for 65 elements 2 1/2 hours after shutdown, and to 1.9 mr/hr for 6-hour condition. A comparison of the dose rates for the two conditions is given in Table III. Note that six hours after shutdown radiation is reduced to 1/10 tolerance five feet from the canal wall. (See Appendix XII-B.)

TABLE III

Radiation Outside the Seven-foot Thick Canal Walls

<u>Gamma Energy (Mev)</u>	<u>Dose Rate 2 1/2 Hrs After Shutdown (mr/hr)</u>	<u>Dose Rate 6 Hrs After Shutdown (mr/hr)</u>
1.65	0.26	0.26
1.8 - 2.2	0.47	0.26
2.4 - 2.6	1.43	1.25
2.8	<u>0.35</u>	<u>0.14</u>
Totals	2.5 mr/hr	1.9 mr/hr

Since each storage rack holds 36 elements, the 65 elements from one refueling do not completely fill the racks. Therefore, the hottest elements (those from the current refueling) should be placed in the racks such that they are two rows of elements from the side of the rack, or about one foot from the canal wall.

In addition, it is not likely that all the elements from a particular refueling will be grouped together, since some elements are used for irradiation of experiments in other regions of the canal. This reduction in the size of the source greatly decreases radiation levels outside the canal walls. Hence, using sensible operational procedures, radiation outside the canal walls should in no case be above 1/10 tolerance.

c. Radiation Below the Canal

The fuel elements stored in the canal have 1 1/2 feet of water below the active sections. (See Figure 20-1.) This water, in addition to the seven

feet of portland concrete in the canal floor, attenuates gamma radiation to 1/60 tolerance in areas below the canal. The contribution of each energy group is given in Table IV.

TABLE IV
Radiation Below Canal Floor

<u>Gamma Energy (Mev)</u>	<u>Dose Rate (mr/hr)</u>
1.65	0.01
1.8 - 2.2	0.02
2.4 - 2.6	0.08
2.8	<u>0.02</u>
Total	0.13 mr/hr

d. Canal Penetration of the Biological Shield

The region between the pressure vessel and the canal water must contain sufficient shielding material to attenuate radiation from the core to tolerance levels in regions around the top of the canal. Core neutrons are readily attenuated by the large amount of water in this area, but 2 1/2 feet of magnetite concrete are required to insure attenuation of gammas to 1/10 tolerance. (See Figure 20-2.) A detailed analysis of the primary and secondary gamma contribution is given in Appendix XII.

e. Canal Water Activity

Flux distributions in the ETR core and thermal shields have been determined from three-group, one-dimensional calculations on an IBM 650 computer.

ELEVATION OF REACTOR SHOWING SHIELDING IN REGION
BETWEEN CANAL AND PRESSURE VESSEL

The average fast neutron flux emerging from the thermal shield is 1.2×10^{10} n/cm²/sec. The concrete* between the thermal shield and canal water reduces the activating flux to 3.4×10^8 nv. At saturation, the gammas from the water activated in this region (250 cu ft) are reduced to negligible intensities in the accessible areas above, below, and to the sides of the canal. (See Appendix XII-D for method of analysis.) This means that forced circulation in the canal to prevent a buildup of water activity is not necessary.

f. Canal Pit Grating

A grate is positioned in the canal pit (see Figure 20-2) to allow four feet of water shielding for an accidentally dropped hot fuel element. This water, in addition to 8 1/2 inches of steel between the pit and the sub-pile room,† limits radiation in the sub-pile room to 580 mr/hr. In the event that an element is dropped, a warning is sounded by a detector in the sub-pile room, or by the operators above, to warn personnel to leave the area until the element is retrieved. (See Appendix XII-E.)

* This concrete is for structural reasons only and is not required for shielding in the canal region.

† See ETR-5528-MTR-520-M-596

XXI. WATER SHIELDING REQUIREMENTS DURING FUEL ELEMENT CHANGES

A. Description

During refueling operations, the water level in the pressure vessel will be 132 inches above the center of the discharge chute. With this amount of shielding a person on the refueling platform, four feet above the water level, receives about 120 mr while removing all 65 fuel element sections.

B. Method of Analysis

1. Fuel elements are lifted from their position in the core one at a time and moved across the top of the core to the discharge chute. At the discharge chute the elements are lifted several feet, positioned over the discharge chute opening, and lowered in. The most significant radiation doses are received by refueling personnel when an element is in the highest position above the discharge chute. To obtain the integrated radiation dose during refueling, each element is considered to be in its highest position for 10 seconds.*
2. The contemplated refueling schedule includes a two-hour reactor cooling period. Hence, for the worst case, radiation is calculated for a fuel element two hours after reactor shutdown following 480 hours operation at a total reactor power of 175 MW. During an operating cycle, 57 fuel elements contribute to the total reactor power of 175 MW. (Forty-nine fuel elements plus 16 control rods contribute fuel equal to eight fuel elements.)

* Radiation levels from fuel elements in the core are discussed in Section XXII, Supplementary Shielding Requirements.

3. Fission product decay gamma energies of 1.1 to 2.8 Mev are considered and divided into five energy groups. (Energies lower than 1.1 Mev are attenuated so much more that their contribution is negligible.)
4. The 36-inch active section of a fuel element is divided into four equal point sources, each located at the center of a 9-inch section.

C. Results and Discussion

The decay gamma energy spectrum for the five energy groups was determined using the methods and an appreciable amount of the data from APEX 134, Fission Product Decay Gamma Energy Spectrum, by John Moteff. In the calculations the presence of Rb^{88} was considered to permit extension of gamma decay information to two hours after shutdown. (Details are given in Appendix XI-C.) The

formula $\Phi = \frac{Q_0 E}{4\pi a^2} B e^{-\mu x}$ was applied to determine the radiation above the

depths of water considered. Sample results are given in Table V. (See Appendix XIII for sample calculation.)

TABLE V

Radiation Above Active Section of a Fuel Element Two Hours after Shutdown

Energy Group (Mev)	Dose Rate (6 ft water above active section) mr/hr	Dose Rate (9 ft water above active section) mr/hr
1.1 - 1.4	321	1.41
1.65	680	4.40
1.8 - 2.2	395	4.53
2.4 - 2.6	382	6.69
2.8	56	1.26
Totals	1834 mr/hr	18.3 mr/hr

The top of the active section of an ETR fuel element is 49 1/4 inches above the bottom of the element. (See G-E Drawing M-761.) Approximately three inches of working space are allowed between the discharge chute opening and the bottom of an element to provide room to move an element over the chute. Subtracting these space requirements from the 132 inches of water above the discharge chute opening shows that the minimum amount of water above a fuel element is 80 inches. This corresponds to a radiation level of 680 mr/hr four feet above the surface of the water (Figure 21-1). For 65 elements, each in its highest position for 10 seconds, the total dose is 120 mr for a person transferring the elements. This dose is less than half weekly tolerance (300 mr). Hence, as can be seen from Figure 21-1, a larger working space than three inches and a longer time than 10 seconds above the chute opening are allowable without exceeding weekly tolerance.

Radiation levels are considerably lower in areas not directly above a raised fuel element, since the refueling platform is a large distance from the fuel elements. Hence, by dividing the lifting operation for 65 elements among several men, radiation doses per man can be reduced still further and in no case need be above half weekly tolerance.

Fig. 21-1

XXII. SUPPLEMENTARY SHIELDING REQUIREMENTS

A. Radiation from the Reactor Core after Shutdown

1. General

Radiation levels are of interest during two basic operations performed after the reactor is shut down: (1) refueling the reactor and (2) servicing facility tubes in regions where the tubes penetrate the pressure vessel. During the refueling operation the water level in the pressure vessel is 17 feet above the active section of the core, and while servicing facilities the water is 10 feet 8 inches above the active core. To sufficiently reduce radiation levels during after-shutdown operations, 16 inches of steel shielding are required in a large air-cooled facility tube and 14 inches of steel in air annuli around facility tubes. In the following it was assumed there were no large fission breaks to contaminate the vessel, active coolants such as sodium were removed from the facility tubes, etc.

2. After-Shutdown Operations

a. Refueling

During refueling operations two hours after shutdown with all fuel elements in the core, the most significant radiation levels are received when transferring an element through the discharge chute.* Radiation from the core itself is insignificant. Somewhat higher narrow-beam doses are present from insulating air annuli around facility tubes (2.5 mr/hr) and from an

* See Section XXI, Water Shielding Requirements During Fuel Element Changes.

air-cooled facility tube (15 mr/hr). This latter dose rate is twice tolerance, but is only present in regions immediately above the air-cooled facility tubes. This level is satisfactory for the limited exposure received by personnel during refueling operations.

b. Servicing Facility Tubes

When servicing facility tubes in the pressure vessel with the decreased water level, the most significant radiation is 38 mr/hr from the active fuel elements in the core. Radiation streaming out of the air annuli and air-cooled facility tubes is 6.3 mr/hr and 38 mr/hr respectively.

3. Method of Analysis

a. Sources of Radiation

It was assumed that all of the fuel elements are in the core two hours after shutdown following 480 hours reactor operation at 175 MW. The significant radiation sources for both after-shutdown operations are the following:

- (1) Decay gammas from U^{235} fission products, and from the aluminum, steel, etc., in the fuel elements, experiments, and the reflector.
- (2) Radiation streaming from the insulating air annuli around facility tubes above the core. This consists of core radiation and decay gammas from the steel facility tubes and casings which are activated during reactor operation.
- (3) Radiation streaming from air-cooled facility tubes.

b. Attenuating Material Above the Core

The attenuating material above the core is divided into essentially two regions.

- (1) The first three feet above the active section of the core is considered as

a water, steel, aluminum mixture due to the large amount of metal in the control rod guide tubes and the facility tubes.

- (2) The remaining material is considered water. Although this area is penetrated by steel facility tubes, the spacing between the tubes is large. In addition, some of the facility tube sections contain air.

c. Refueling

The refueling platform is about four feet above the refueling water level in the pressure vessel. In Appendix XIII the radiation level when lifting a spent fuel element from the core and dropping it down the discharge chute has already been determined. Excluding this operation and assuming all fuel elements are in the core two hours after shutdown following 480 hours operation at 175 MW, the following radiation levels were obtained.

(1) Decay Gamma Radiation

Fission product decay gammas of 1.65 to 2.8 Mev were considered. The decay gammas from aluminum, steel, and other non-fissionable materials in the active section of the core, reflector, and filler pieces are also included in the 1.65 Mev group.* This is treated as a cylindrical source of 80 cm radius. The radiation is attenuated through the material between the source and the refueling platform, giving a 1.65 Mev contribution. The other energy groups consist of decay gammas from the fuel alone. Since there is a smaller source area for each of these energy groups, the active section of the core is divided into four equal point sources, each

* This is conservative, since the metal decay gammas are actually about 1.2 Mev.

located at the midpoint of a 9-inch section in the 36-inch long fuel section. The total radiation at the refueling platform from 1.65 to 2.8 Mev gammas is found to be 3.5×10^{-3} mr/hr, which is only 1/2000 tolerance.

(2) Radiation Streaming

Radiation in the 1/8-inch air annuli between each facility tube and its outer casing is attenuated by 14 inches of steel washers, about one-inch thick spaced throughout each annulus. The dominant source is from decay core gammas which stream up the annuli. This source is attenuated to about 2.5 mr/hr above the refueling water level. This is satisfactory, since the radiation is only narrow-beam radiation and does not represent whole-body exposure of personnel on the refueling platform. (See Appendix XIII.)

(3) Radiation from an Air-Cooled Facility

Sufficient shielding for the air-cooled facility tubes must be installed by the experimenters. The recommended shielding for a large (about 8-inch diameter) facility tube is 16 inches of steel, which may be in the form of a helix. The helix does not restrict the air flow appreciably, since the air spirals down the tube. The dominant radiation is from core gammas, and these are reduced to 15 mr/hr coming out of the tube above the water level. (See Appendix XIV-B.)

d. Servicing Facility Tubes

When servicing facility tubes in the upper sections of the pressure vessel (making or breaking side-arm connections), the water level is lowered to

88 inches below the pressure vessel flange. This is 76 inches lower than the refueling water level. The source methods used are the same as used in determining the radiation during refueling operations. However, there is less water above the core, and the radiation levels are determined at the surface of the water where personnel are working.* The radiation contributions are:

- (1) The average decay gamma radiation from the core is 38 mr/hr.
- (2) The radiation streaming out of the air annuli after attenuation by 14 inches of steel washers is 6.3 mr/hr.
- (3) Radiation above a large air-cooled facility tube after attenuation by 16 inches of steel shielding in the tube is 38 mr/hr.

* Personnel are lowered to this level in a chair arrangement by the overhead crane.

B. Shielding Required for Instruments Horizontally Penetrating the Biological Shield

1. Radiation when Chambers and Plugs are in Place

Horizontal instrument penetrations in the concrete biological shield are located one foot above the reactor midplane (about seven feet above the console floor). These penetrations, which extend through to the external thermal shield, are for three types of removable instruments. The penetrations may be divided into the following three classes due to the different shielding plugs required for each. (See Figure 22-1.)

- a. Movable gas flow ion chambers (six)
- b. Boron thermopiles (five)
- c. Thermocouples (one assembly containing 12 thermocouples)

The external thermal shield limits the forward point of each assembly. In the sections where the thermal shield is penetrated, the shield differs slightly from its general design. Sleeves which penetrate the external thermal shield will be installed when the shield is manufactured. When the shield is installed in the reactor assembly, lead plugs will be placed in these sleeves with no annulus. Thus, there are $3 \frac{3}{4}$ inches of lead in these sections of the shield in contrast to the other sections of the shield which are $3 \frac{1}{2}$ inches of lead followed by a $\frac{1}{4}$ -inch steel form. If it is found that a higher neutron flux is required in a particular instrument location, the lead plug can be removed and altered as required within limitations.

In the following analyses the significant radiation sources considered are fast neutrons and 7 Mev gammas emerging from the external thermal shield. Flux

HORIZONTAL SHIELD PLUG

A. THERMOCOUPLE

SCALE: $\frac{1}{16}$ ACTUAL SIZE

B. THERMOPILE

C. ION CHAMBER

plots for the average values of neutron and gamma radiation are presented in Figures 22-2, 22-3 and 22-4.

a. Shielding for Movable Ion Chambers

(1) Radiation Streaming

The annulus between a plug and the steel liner in the concrete is a 1/16-inch to 1/8-inch gap. However, due to irregularities, which may practically close the gap in some places, the over-all annulus is considered a 1/16-inch gap.

(a) Neutron streaming

Using Figure 22-2 for the average fast neutron flux^{*}, the neutron current entering the annulus following the external thermal shield is obtained from

$$J_+ = \frac{\phi}{4} - \frac{D}{2} \frac{d\phi}{dx} = \frac{\phi}{4} - \frac{D\phi}{2} \frac{d \ln \phi}{dx}$$

which gives

$$\begin{aligned} J_+ &= \frac{10^{11}}{4} - \frac{3(10^{11})}{2} \frac{2.3}{30} \\ &= 3.65 \times 10^{10} \text{ n/cm}^2/\text{sec} \end{aligned}$$

The path of least attenuation is considered.

- (i) Geometrical attenuation through the annulus between the 30-inch graphite section of the ion chamber and the steel liner in the concrete.

Inside radius of the annulus is about two inches. Considering a 1/16-inch gap gives an outside radius of 2 1/16 inch.

* IBM 650 calculation

FLUX IN HORIZONTAL DIRECTION

For a cosine emitter at the annulus entrance, the geometrical attenuation factor is given by

$$\frac{\phi}{\phi_o} = \frac{1}{\pi L^2} \left[(2R^2 - r^2) \cos^{-1} \frac{r}{R} - r \sqrt{(R^2 - r^2)} \right]$$

where

R = outside radius

r = inside radius

Substituting gives

$$\frac{\phi}{\phi_o} = 3.8 \times 10^{-5}$$

(ii) Attenuation in five inches of lead and 1/2-inch of aluminum in the beginning of the plug

The fast neutron scattering cross section for lead ($\sum_{s_{Pb}}$) is 0.165 cm^{-1}

The attenuation in five inches of lead is

$$e^{-\sum_s x} = e^{-(0.165)(12.7)} = 0.12$$

For aluminum, $\sum_{s_{Al}} = 0.18 \text{ cm}^{-1}$, and the attenuation in 1/2-inch of aluminum is 0.80.

(iii) Attenuation in 37 inches of magnetite concrete

The fast neutron relaxation length in the magnetite concrete is 8.2 cm. In

37 inches there is an attenuation of $e^{-\frac{94}{8.2}} = 10^{-5}$

The total attenuation is therefore

$$(0.5) (3.8 \times 10^{-5}) (0.12) (0.80) (10^{-5}) = 1.82 \times 10^{-11}$$

The neutron intensity at the outer face of the shield is then

$$(3.65 \times 10^{10}) (1.82 \times 10^{-11}) = 1 \text{ n/cm}^2/\text{sec.}$$

This is about 1/80 tolerance.

Proceeding similarly, the slow neutron source is found to be negligible.

(iv) In the air duct on either side of the movable chamber there is an additional attenuation factor of 0.5 or less.

(b) Gamma Streaming

The 7 Mev gamma source at the outer face of the external thermal shield is 4.1×10^{10} Mev/cm²/sec. (See Figure 22-3.)

The geometrical attenuation in the 30-inch annulus is 3.8×10^{-5}

The attenuation in five inches of lead and 1/2-inch aluminum is

$$\mu x (\text{aluminum}) = (0.068) (1.27) = 0.09$$

$$\mu x (\text{lead}) = (0.47) (12.7) = 5.97$$

The attenuation in 37 inches of magnetite concrete is

$$\mu x = (0.1) (94) = 9.4$$

The total material attenuation is

$$B F_0 (b) = (8) (1.2 \times 10^{-8})$$

The total attenuation is then

$$(0.5) (3.8 \times 10^{-5}) (8) (1.2 \times 10^{-8}) = 1.8 \times 10^{-12}$$

This reduces the gamma flux entering the annulus to 7.4×10^{-2} Mev/cm²/sec, which is negligible.

Fig. 22-3

(2) Direct Radiation

(a) Primary Gammas

Attenuation is through 30 inches of graphite, 1/2-inch of aluminum, five inches of lead, and 37 inches of magnetite concrete. The attenuations are

	μ	x		μx
Graphite	0.038	76.2	=	2.9
Aluminum	0.068	1.27	=	0.09
Lead	0.47	12.7	=	5.97
Concrete	0.10	94	=	9.4
Total μx =				18.4

The source is treated as a plane source and the attenuation is

$$B F_0(b) = (8) (5.4 \times 10^{-10})$$

The radiation at the outer face of the shield is then

$$\begin{aligned}\overline{\Phi}_E &= (4.06 \times 10^{10}) (8) (5.4 \times 10^{-10}) \\ &= 175 \text{ Mev/cm}^2/\text{sec or } 0.20 \text{ mr/hr.}\end{aligned}$$

The geometrical attenuation factor of 0.5 in the air duct on either side of the ion chamber reduces this radiation to 0.1 mr/hr, which is 1/70 tolerance.

(b) Secondary Gammas

The fast neutrons entering the ion chamber assembly are assumed thermalized and captured one age displacement inside the graphite of the

ion chamber. This is $\gamma/h = \frac{350}{9} = 39$ cm. Each neutron is assumed captured with the emission of one 5 Mev gamma per capture. The source is therefore

$$(3.65 \times 10^{10}) (5) = 1.8 \times 10^{11} \text{ Mev/cm}^2/\text{sec.}$$

This is considered a plane source and is attenuated through the remaining material. The attenuations are

	<u>μ</u>	<u>x</u>	<u>μx</u>
Graphite	0.045	38	= 1.71
Aluminum	0.076	1.27	= .10
Lead	0.47	12.7	= 5.97
Concrete	0.11	94	= 10.3
		Total μx	= 18.1

The radiation at the outer face of the shield is then

$$\begin{aligned} \overline{\Phi}_E &= (1.8 \times 10^{11}) B F_0 (18.1) \\ &= 1.8 \times 10^{11} (8) (7.5 \times 10^{-10}) \\ &= 1080 \text{ Mev/cm}^2/\text{sec or } 1.2 \text{ mr/hr.} \end{aligned}$$

The geometrical attenuation factor of 0.5 in the air duct on either side of the ion chamber reduces this radiation to 0.6 mr/hr, which is 1/10 tolerance.

Assuming all of the captures to occur in the graphite gives a high estimate of the secondary gamma radiation. In reality, the captures are distributed throughout the chamber and plug, and the gamma radiation from captures

in the borated concrete is less than the radiation from captures in the graphite.

(c) Neutrons

The fast neutrons entering the ion chamber are 3.65×10^{10} n/cm²/sec.

These have a relaxation length of 9 cm in the 76 cm of graphite and 8.2 cm in the magnetite concrete. At the outer face of the shield the neutron current is reduced to

$$(3.65 \times 10^{10}) (e^{-\frac{76}{9}}) (e^{-\frac{94}{8.2}}) = 83 \text{ n/cm}^2/\text{sec}$$

The additional 0.5 attenuation factor in the air duct on either side of the chamber reduces the fast neutrons to 42 n/cm²/sec, which is about 1/2 tolerance. This is actually a very conservative value, since scattering in the five inches of lead in the plug and the steel outside the plug was neglected. This reduces the direct radiation to about 1/10 tolerance.

(3) Radiation Streaming down the Steel Drive Shaft and Annulus around the Shaft.

Direct radiation is amply attenuated in the steel. The path of least attenuation is through the annulus (1/16-inch maximum) around the one-inch shaft. The whole length of the shaft is threaded, but the sleeve in which it rides is threaded for only about one inch. Therefore, there is little attenuation of radiation which streams down the annulus.

(a) Fast Neutrons

The neutron beam emerging from the graphite is

$$(3.65 \times 10^{10}) (e^{-\frac{76}{9}}) = 7.8 \times 10^6 \text{ n/cm}^2/\text{sec}$$

This value is reduced to $3.9 \times 10^6 \text{ n/cm}^2/\text{sec}$ by the geometrical attenuation factor of 0.5 in the air duct on either side of the chamber. The geometrical attenuation in the 49-inch long annulus is 7.03×10^{-6} , which reduces the fast neutrons to $27 \text{ n/cm}^2/\text{sec}$. The annulus is interrupted by approximately one inch of steel in the section where the threaded shaft engages the sleeve threads. This gives

$$\frac{e^{-\sum_s x}}{e} = e^{-(0.254)(2.54)} = 0.53$$

which reduces the over-all radiation to about $14 \text{ n/cm}^2/\text{sec}$. This radiation is about 1/6 tolerance, but is satisfactory since it is very narrow-beam radiation and above head height.

(b) Gamma Radiation

The 7 Mev gamma source entering the ion chamber is $4.1 \times 10^{10} \text{ Mev/cm}^2/\text{sec}$. This radiation is reduced to $2 \times 10^{10} \text{ Mev/cm}^2/\text{sec}$ by the geometrical attenuation factor of 0.5 in the air duct. Using a plane source treatment and a total μx of 3.5, the attenuation in the graphite and one-inch threaded section of the steel is 1.54×10^{-2} . The annulus attenuation is 7.03×10^{-6} , giving a total radiation level at the shield surface of

$$(2 \times 10^{10}) (1.54 \times 10^{-2}) (7.03 \times 10^{-6}) = 2110 \text{ Mev/cm}^2/\text{sec}$$

This is about 2.3 mr/hr or just under tolerance, but is considered satisfactory since it is narrow-beam radiation and above head height.

b. Boron Thermopile Plug

(1) Radiation Streaming

The step placement in the thermopile plug is considerably better than that used in the ion chamber plugs. Hence, in comparing the two plugs it is found that radiation from annular streaming is reduced below 1/10 tolerance.

(2) Direct Radiation

It was shown in Section XVII, Concrete Biological Shield, that about 86 inches of magnetite concrete were required at the reactor midplane to attenuate core radiation to 1/10 tolerance. Therefore, the 7 1/2 feet of concrete in the thermopile plug is sufficient to reduce direct radiation to 1/10 tolerance.

c. Thermocouple Plug

The thermocouple plug duplicates the shielding it displaces. The plug is not borated since it is never removed. Another reason the plug is not borated is that it must be essentially the same material as the biological shield if accurate temperature readings are to be made. When the plug is finally fitted into place, there is almost no clearance between the forward section of the plug and the shield. The outer section of the plug has a 1/16-inch annulus.

(1) Radiation Streaming

(a) Neutron Streaming

The fast neutron current entering the inner face of the concrete is 3.65×10^{10} n/cm²/sec. The relaxation length for fast neutrons in the magnetite concrete is 8.2 cm, giving an attenuation in 44 inches of

$$e^{-\frac{112}{8.2}} = 1.17 \times 10^{-6}$$

The neutron source at the annulus is therefore 4.3×10^4 n/cm²/sec.

For a length of 52 inches, an inside radius of 2 1/2 inches, and an outside radius of 2 9/16 inches, the geometrical attenuation in the annulus is 1.3×10^{-5} . The fast neutrons emerging from the annulus are therefore reduced to

$$(4.3 \times 10^4) (1.3 \times 10^{-5}) = 0.56 \text{ n/cm}^2/\text{sec}$$

This is less than 1/100 tolerance.

(b) Gamma Streaming

The geometrical attenuation factor in the annulus for gamma radiation is 1.3×10^{-5} . This is equivalent to attenuation in 40 inches of magnetite concrete. Adding the 44 inches of solid concrete in front of the annulus, there is essentially 84 inches of concrete. This is more than enough to reduce direct gamma radiation to 1/10 tolerance.

(2) Direct Radiation

Since the plug essentially duplicates the biological shield, direct radiation is reduced to less than 1/10 tolerance.

2. Radiation From a Plug Assembly After Removal

a. Radiation from an Activated Thermopile Plug

(1) Activity in the Concrete

The activity in the first two feet of the plug is considered. The boral sheet in the front of the plug captures the thermal neutrons entering directly. The net fast flux entering the plug is

$$-D\nabla\phi = 2.9 \times 10^{10} \text{ n/cm}^2/\text{sec}$$

These neutrons are thermalized and captured in the concrete. It is assumed that all the thermal neutrons entering the plug are captured by the iron or boron in the concrete. The current at the end of the plug is then

$$J_+ = \frac{\phi}{4} - \frac{D\phi}{2} \frac{d \ln \phi}{dx} = 5.4 \times 10^9 \text{ n/cm}^2/\text{sec}$$

Across the sides of the plug the current is $J = \frac{\phi}{4}$ The number of neutrons

entering the plug per second is determined from

$$\int J \cdot ds = \pi r^2 \left[(2.9 \times 10^{10}) + (5.4 \times 10^9) \right] + \int_0^{\infty} \frac{\phi}{4} 2\pi r dx$$

where $r = 1.5 \text{ inches} = 3.81 \text{ cm}$

$$= 2 \times 10^{10} e^{-0.105x}$$

The total number of neutrons entering the concrete plug is therefore

$$\pi (3.81)^2 (3.4 \times 10^{10}) + 10^{10} \pi (3.81) \left(\frac{1}{0.105} \right) = 2.7 \times 10^{12} \text{ neutrons/sec}$$

The concrete used contains one percent boron and 86.6 percent magnetite by weight. The magnetite contains 81 percent Fe_3O_4 . Then the percent of iron in the magnetite aggregate is

$$0.81 \cdot \frac{3 \times 55.8}{(3 \times 55.8) + (4 \times 16)} = 58.6\% \text{ Fe}$$

The percent by weight of iron in the concrete is

$$(0.866) (0.586) = 50.7\%$$

Since the density of the concrete is 3.5 gm/cc,

$$\begin{array}{l} \text{density of iron} \\ \text{in the concrete} \end{array} = (3.5) (0.507) = 1.77 \text{ gm/cc}$$

$$\begin{array}{l} \text{density of boron} \\ \text{in the concrete} \end{array} = (3.5) (0.01) = 0.035 \text{ gm/cc}$$

The macroscopic capture cross sections were obtained from the formula

$$\sum_a = \frac{\rho \sigma N_0}{A}$$

where

$$\sum_a = \text{macroscopic capture cross section}$$

$$\rho = \text{density of the material}$$

$$\sigma = \text{microscopic capture cross section}$$

$$N_0 = \text{Avagadro's number}$$

Then for iron

$$\begin{aligned} \sum_{a_{\text{Fe}}} &= \frac{(1.77) (2.53 \times 10^{-24}) (6.02 \times 10^{23})}{55.8} \\ &= 0.0483 \text{ cm}^{-1} \end{aligned}$$

and for boron

$$\begin{aligned} \sum_{a_{\text{B}}} &= \frac{(0.035) (755 \times 10^{-24}) (6.02 \times 10^{23})}{10.82} \\ &= 1.47 \text{ cm}^{-1} \end{aligned}$$

Practically all of the captures in the concrete occur in the iron and boron, therefore the fraction of total captures occurring in the iron is

$$\frac{\sum a_{\text{Fe}}}{\sum a_{\text{Fe}} + \sum a_{\text{B}}} = \frac{0.0483}{1.52} = 0.0318$$

Thus about three percent of the total neutron captures occur in the iron.

This is

$$(2.7 \times 10^{12}) (0.03) = 8.1 \times 10^{10} \text{ captures/second}$$

All of the neutrons are assumed captured in the first two feet of the plug.

The number of captures/cc/sec in the iron is therefore 2.9×10^7 n/cc/sec.

The activity of Brookhaven concrete (similar to magnetite since it is 58 % limonite, 48 % iron) is 4.8×10^8 Mev/sec/gm four hours after 700 hours irradiation at 10^{12} n/cm²/sec thermal. *

In the Brookhaven concrete (density 3.5 gm/cc) the density of the iron is 0.97 gm/cc and $\sum a = 0.025$ cm⁻¹. A flux of 10^{12} n/cm²/sec produces 2.5×10^{10} captures/cc/sec.

In the borated magnetite plug the total captures in the iron were found to be 2.9×10^7 captures/cc/sec. The activity of the Brookhaven concrete was adjusted to correspond to the magnetite mix, giving an activity for the magnetite concrete of

$$(4.8 \times 10^8) \frac{2.9 \times 10^7}{2.5 \times 10^{10}} = 5.6 \times 10 \text{ Mev/sec/gm}$$

$$\text{or } 2 \times 10^6 \text{ Mev/cc/sec}$$

(2) Activity in the Aluminum Concrete Casing

The casing is actually Type 6061 aluminum, but the activation may con-

*MTR Handbook, ORNL 963, Shielding Summary

servatively be considered the same as for Type 2S aluminum which emits decay gammas of 3×10^7 Mev/sec/gm four hours after extended activation at 10^{12} n/cm²/sec. *

Due to the presence of boron in the thermopile plug, the thermal flux in the plug is somewhat less than the flux shown in Figure 22-4. However, it may conservatively be assumed that the flux at the edge of the plug is the same as the value shown in Figure 22-4. This may then be represented by

$$\phi = (2 \times 10^{10}) e^{-0.105 x}$$

The average flux over the two-foot end section of the plug is

$$\phi = \frac{1}{61} \int_0^{61} \phi \, dx = 3.2 \times 10^9 \text{ n/cm}^2/\text{sec}$$

The activity from the aluminum is then

$$\begin{aligned} & (3.0 \times 10^7 \text{ Mev/sec/gm}) (2.7 \text{ gm/cc}) \left(\frac{3.2 \times 10^9}{10^{12}} \right) \\ & = 2.6 \times 10^5 \text{ Mev/cc/sec} \end{aligned}$$

(3) Radiation from the Plug

The activity of the concrete is greater than that of the aluminum casing by approximately a factor of 10. The two-foot section of the plug was

* MTR Handbook, ORNL 963, Shielding Summary

FLUX IN HORIZONTAL DIRECTION

treated as a cylindrical source, and the radiation one foot from the source determined to be 1.23 r/hr. Weekly tolerance is therefore received in about 15 minutes. This radiation is not considered excessive, and a coffin should not be required to remove the plug. During the time the plug is being removed, only personnel necessary to perform the operation should be in the area. After the plug is removed, the ion chamber or thermopile is replaced and the plug reinserted in the biological shield. Thus the time elapsed while the plug is out of the shield should not be more than about 10 minutes. (This operation is normally performed only once or twice a year.)

The chambers themselves are not as active as the plugs, since the aluminum activity in the plugs was found to be 1/10 the concrete activity. Although the chambers are in front of the plugs and therefore see a higher activating flux, their casing is made of magnesium which has a smaller activation cross section than the aluminum.

b. Radiation from an Activated Ion Chamber Assembly

Radiation from an activated ion chamber plug is less than from a thermopile plug, since the plug is shielded by the 2 1/2 feet of graphite and the boral plate in the chamber. Neutron flux that has diffused through the concrete into the sides of the plug is also small, because the plug is about 4 1/2 feet from the external thermal shield. Activity in the steel shaft does not present a problem, since in the highest radiation area (the forward section of the shaft following the chamber) the direct activating flux is reduced by the chamber shielding.

c. Radiation from an Activated Thermocouple Plug

The thermocouple plug is not borated because it must be essentially the same material as the biological shield to give accurate temperature readings. The plug is welded into place, since it is not intended to be removable. However, if at some time the plug should be removed, radiation from it will be high (about 40 r/hr), and suitable precautions will have to be taken.

3. Radiation Streaming Out of an Air Duct in the Shield When a Plug Has Been Removed*

a. Core Radiation

The core is considered a cylindrical source. The radiation from 1.65 to 2.8 Mev gammas four hours after shutdown (Appendix XI-C) is attenuated by the reflector, water, steel, and lead between the core and the beginning of the air duct.

Since the source is some distance from the duct, the geometrical attenuation in the duct is calculated on the basis of a Fermi source distribution at the duct entrance. The attenuation† is

$$\frac{\phi}{\phi_0} = 1.268 \frac{r^2}{L^2} = 8.4 \times 10^{-4}$$

where

$$r \cong 2.5 \text{ inches}$$

$$L = 96 \text{ inches}$$

* The ion chamber duct is treated here. The other ducts have smaller diameters and produce less radiation.

† TID 7004.

The total core radiation emerging from the duct is found to be 4.2 mr/hr.

b. Decay Gammas from the Steel Liner

The lining of the duct consists of steel which becomes activated during reactor operation. The secondary radiation is estimated by considering radiation from a steel thickness corresponding to the relaxation length of decay gammas with an energy of 1.2 Mev. This length is 2.5 cm. A two-foot length following the thermal shield is analyzed. (See diagram below.)

The volume of interest is

$$\begin{aligned}
 V &= 2\pi r (dr) L \\
 &= 2\pi r (2.5 \sin \quad) (61) \\
 &= 136 \text{ cc}
 \end{aligned}$$

where

$$\sin \Theta = \frac{2}{72} = 0.028$$

Assuming an average neutron flux of 5×10^9 n/cm²/sec, the activity is given by

$$\begin{aligned} Q_v &= (7.8 \text{ gm/cc}) (3.8 \times 10^8 \text{ Mev/sec/gm}) \left(\frac{5 \times 10^9}{10^{12}} \right) \\ &= 1.48 \times 10^7 \text{ Mev/cc/sec} \end{aligned}$$

The total activity amounts to 2.0×10^9 Mev/sec. If this is considered a point source located 72 inches from the outside of the concrete, the radiation outside the duct is

$$\begin{aligned} \overline{\Phi}_E &= \frac{2.4 \times 10^9}{4\pi(183)^2} \\ &= 4.75 \times 10^3 \text{ Mev/cm}^2/\text{sec or about } 8.3 \text{ mr/hr} \end{aligned}$$

When a plug is removed the total gamma radiation at the outer end of the duct is about 12.5 mr/hr. This is about twice tolerance, but is satisfactory since the plug is out of the shield only a short time while an ion chamber or thermopile is replaced.

C. Radiation in Accessible Areas Near the Pipe Tunnel

The two three-foot diameter primary coolant pipes running between the pressure vessel and the heat exchanger building represent a formidable source of gamma radiation during reactor operation. To limit this radiation to 1/10 tolerance, the two pipes are enclosed in a "tunnel" with four-foot thick walls of magnetite concrete.

The activity in the water leaving the reactor core is principally 6.5 Mev gammas from the decay of N^{16} formed in the fast neutron reaction $O^{16}(n,p)N^{16}$. As the water leaves the core the source is 6.6×10^6 photons/cc/sec.* The water flows to the heat exchanger building with a velocity of 14.6 ft/sec, and the activity decays with its characteristic 7.35 second half-life. There is a delay of several seconds in passing through the heat exchangers, after which the water returns to the pressure vessel through the inlet pipe.

During reactor operation access to the tunnel region is normally possible in two general areas: the console floor at the sides of the lower section of the tunnel, and the first floor at the top of the tunnel. (See Figures A14-1 and A14-2.) For the most severe case, radiation on the console floor (at the beginning of the pipe tunnel) is 0.1 mr/hr. Similarly on the first floor the radiation level is a maximum of 0.3 mr/hr. (See Appendix XIV-C.) This is a desirable safety factor of 10 which should assure that the radiation levels around the pipe tunnel are no higher than 1/10 tolerance.

* See Section XVI, Shielding Provided for the Heat Exchanger Building.

D. Fuel Element Transfer Cask

As constructed, the transfer cask incorporates the shielding shown in Figure 22-5.

The storage area of the cask contains active fuel element sections immersed in water. The decay gamma sources were obtained from APEX 176, and the required shielding was calculated on the basis of a cylindrical source for gamma energies of 1.65 to 2.8 Mev. *

Radiation from nine ETR fuel elements 120 days after shutdown, following 1000 hours[†] operation in the reactor, is limited to 6 mr/hr one meter to the sides of the cask and 2 mr/hr one meter above the cask.

Radiation from one seven-day decayed fuel element is limited to 18 mr/hr one meter to the sides of the cask and about 1 mr/hr one meter above the cask.

* See Appendix XII-B for a typical cylindrical source analysis.

† An actual operating cycle is about 480 hours, but using a source after 1000 hours operation compensates for elements which have operated at a higher than average power.

FUEL ELEMENT TRANSFER CASK

Fig. 22-5

E. Removal of Regulating Rods, Guide Tubes, and Similar Activated Materials From the Reactor

The high activity of long metal tubes which have been in the reactor core during operation require special techniques to remove them from the pressure vessel.

The discharge chute is used to transfer spent fuel elements from the reactor to the canal. Control rods are similarly transferred, since they are constructed in three separate sections which are latched together. Each of the three sections may be unlatched under water, and then handled with grapples in the same manner as a fuel element. During these operations a fuel element or control rod section has a minimum of six feet of water above it. This minimum depth occurs when the element or section is raised to its maximum height before being dropped down the discharge chute opening.

The regulating rods, aluminum guide tubes for control rods, and facility tubes are removed in one section which is about 15 feet long. The guide tubes and facility tubes are too long to be accommodated by the discharge chute. The regulating rods could be transferred through the chute, but when the rods are lifted above the chute opening the most active section (the effective stainless steel portion operating in the reflector) is above the water level in the pressure vessel. This would create hazardous radiation levels in the first floor area.

Due to the intensity of the decay gamma radiation from these aluminum and steel tubes, it is suggested that the tubes be removed during a night shift or other period when there are few personnel in the building and surrounding areas.

The following removal procedure is suggested:

1. The tube to be removed is connected to the auxiliary overhead crane hook.
2. The tube is raised a few feet from its in-core position to insure that removal may proceed normally.
3. All personnel then proceed to areas below the first floor level.
4. The crane is operated remotely from a position on the console floor opposite to the canal. (See Figure A14-3.) The crane operator presses the "up" control and the crane lifts the tube out of the reactor core and pressure vessel. The upward motion stops when a limit switch is reached. (Limit switches are installed on the crane before beginning this operation.) At this point the bottom of the tube is about one foot above the vessel flange. The operator then presses the "horizontal travel" control, and the tube is moved to a position above the canal water where it reaches the second limit switch. When the "down" control is pressed, the tube is lowered to the bottom of the canal until the third limit switch is reached.

The first floor area may then be reoccupied, the crane hook disconnected, and the tube handled with grapples in the canal to move it to the desired storage position.

Radiation levels on the console floor are about 300 mr/hr when an aluminum guide tube or regulating rod is above the vessel. (See Appendix XIV-D.) A guide tube will probably be in this position for no more than five minutes, producing an integrated gamma dose of 25 mr (1/12 weekly tolerance).

A stainless steel facility tube is 10 times more active than an aluminum guide tube. The steel tube produces a dose rate of one r/hr and a five-minute integrated dose of $1/4$ weekly tolerance. As can be seen from Figure A14-3, persons standing near the biological shield are exposed to negligible radiation because of the attenuation in the biological shield along a straight-line path from the tube above the vessel.

The stainless steel grid plate and other highly active materials may be removed from the core in a similar manner.

APPENDIX VIII

SHIELDING FOR THE HEAT EXCHANGER BUILDING

This appendix contains a description of the procedure and methods of calculation, including several sample calculations, used to determine the shielding requirements for the heat exchanger building. *

A. Determination of Flow Time

Pertinent Data

Total flow	44,400 gpm
Degasser flow	300 gpm
H. E. flow per bank	14,700 gpm
Area 36-inch headers	6.78 sq ft
Area 24-inch headers	3.01 sq ft
H. E. header length	3.00 ft
H. E. header area	7.50 sq ft
H. E. header wall thickness	11/16-inch
H. E. tubes	5/8-inch, 18 gage
Total tube area	2.57 sq ft
H. E. tube length	20.00 ft
Pump volume (water)	24.00 cu ft
Area 18-inch pipe	1.58 sq ft

Reference Drawings

ETR-5528-MTR-644-P5
-P13
-P14
-P17
-P19

* Reference used throughout was APEX 176, Miscellaneous Data for Shielding Calculations, J. Moteff, O. T. S., Dept. of Commerce, 1954.

1. Delay Time Between the Bottom of the Core and the Outlet Header

It was assumed that the total water volume below the core and above the protection plate changes continuously. This leads to a delay time of 1.67 seconds. From a study of flow through the thermal shields it was established that the average velocity was 6.29 ft/sec with a flow area of 15.77 sq ft, which gives an average delay time of 0.991 seconds in the thermal shields. From a study of flow in the top annulus the flow volume per 30-degree segment, the flow velocity, and the flow time to travel to the outlet were determined. The average flow time in the annulus was found to be 0.605 seconds and was determined from

$$\overline{T} = \frac{\sum_1^6 V_i T_i}{\sum_1^6 V_i}$$

2. Flow Time Around the Primary Loop to the Tank Inlet

Table A-VI shows the results of the study of flow time around the primary loop and through the first heat exchanger bank. The location of the lettered points can be seen by referring to Figure A8-1. The flow times between points CD and EF are the average times between the heat exchanger header inlets and take into consideration the difference in time required to reach each heat exchanger in a bank. The flow time required from outlet to inlet for the first heat exchanger bank is 42.3 seconds. (See Table A-VI.) The average time is determined as follows.

second bank

$$\text{add } 36 \text{ ft at } 14.6 \times \frac{2}{3} \text{ ft/sec} = 9.8 \text{ ft/sec}$$

$$\Delta t = \frac{36}{9.8} = 3.7 \text{ sec}$$

Figure A8-1

TABLE A

FLOW TIME AROUND PRIMARY LOOP AND THROUGH HEAT EXCHANGER BANK

Point	Distance (feet)	Flow (gpm)	Flow (cu ft/sec)	Pipe Diam. (inches)	Actual Pipe Area (sq ft)	Velocity (ft/sec)	Δt (sec)
out - A	78	44,400	98.9	36	6.78	14.6	5.34
AB	37.3	14,700	32.7	2	3.01	10.8	3.46
BC	29.3	14,700	32.7	2	3.01	10.8	2.71
Average CD							4.43
Header)	3	4,900	10.9		7.5	1.45	2.07
)							
Tubes)-DE	20	4,900	10.9		2.57	4.25	4.71
)							
Header)	3	4,900	10.9		7.5	1.45	2.07
Average EF							2.99
FG	26	14,700	32.7	24	3.01	10.8	2.41
GH	13.3	14,700	32.7	24	3.01	10.8	1.23
Pump							0.73
HI	24	14,700	32.7	24	3.01	10.8	2.22
LJ	36.8	14,700	32.7	24	3.01	10.8	3.41
J-in	67	44,400	98.9	36	6.78	14.6	4.58
							<u>42.36</u>

Total flow time = 42.4 sec

second bank (cont'd)

flow time = 46.1 sec

third bank

add 36 ft at $14.6 \times \frac{1}{3}$ ft/sec = 4.9 ft/sec

$$\Delta t = \frac{36}{4.9} = 7.3 \text{ sec}$$

flow time = 53.4 sec

Therefore, average flow time from outlet to inlet = 47.3 seconds.

3. Time in Residence Above Core

The residence time above the core is actually unimportant for these calculations, but is estimated to be 18.5 seconds by considering the water volume below the reactor tank head.

Total Cycle Time

	Time (sec)
Average from outlet to inlet	47.3
Time in volume below core	1.67
Time in thermal shield	0.99
Time in top annulus	0.61
Approximate time in core	0.10
Time in volume above core	<u>18.5</u>
Total cycle time	69.2

The time spent in the degasser cycle has been ignored. This is a flow of about 300 gpm.

B. Water Activity

The method of determining the water activity has been discussed in Section XVI. Reference is made to Table IV on page 16-6. The delay times in the table include the time under the core, in the thermal shield, and in the top annulus.

C. Useful Constants

The coefficient of self absorption for the heat exchangers was determined by finding the volume fraction of the tubes.

$$\overline{\mu}_s = V. F. H_2O \times \mu_s H_2O + V. F. Fe \times \mu_s Fe$$

$$\overline{\mu}_s = (0.85) (0.027) + (0.15) (0.225)$$

$$\overline{\mu}_s = 0.0568; \text{ taken as } 0.06 \text{ cm}^{-1}$$

The volume fraction of the primary coolant is 0.363, so that the specific activity should be reduced by this fraction.

The heat exchanger building is shielded with ordinary concrete having an assumed density of 2.3 g/cc. The coefficient of absorption, μ , was taken as 0.06 cm^{-1} for 6.5 Mev photons. A photon intensity of 1.4×10^2 gammas/cm²/sec will produce a dose rate of one mr/hr.

D. Shielding Provided for Wall B

Since Wall B represents a typical shielding problem, the methods of calculation have been discussed here in detail. The methods of calculation for other points of interest are similar to those for Wall B and have therefore been discussed in less detail.

1. Point PB1 (Wall B, six feet above the pavement and opposite the midpoint of the exchangers).

The shielding for Wall B was determined by assuming the case where exchanger banks 2, 3, and 4 were operating and bank 1 was shutdown.

a. Radiation contribution of fourth bank heat exchangers.

Figure A8-2 shows the general geometry of Wall B. The headers were considered separately and the tube section of the exchanger was treated as a cylinder 20 feet long. Since an average time to point D had been determined, it was assumed that each exchanger had the same specific activity. It will be seen later that the angle Φ formed by the intersection of a line drawn from the detector to the midpoint of the exchanger and a line drawn to the end of the tube section is about 40 degrees. Now in the equation for flux from a cylindrical source, the term $F(\Phi, b_2)$ appears where

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

$$\text{and } b_2 = \mu_1 t_1 + \mu_2 t_2 + \dots + \mu_s z.$$

For fairly large b_2 , $F(30^\circ, b_2) \approx F(\pi/2, b_2)$. Thus, for these angles the cylinder appears to the detector to be infinite in length, in fact $F(20^\circ, b_2) \approx 0.8 F(\pi/2, b_2)$. Essentially, since $\Phi \approx 40^\circ$, it was assumed that the detector sees only the center section of the exchanger and not the end section. Therefore, instead of dividing the exchanger into several cylinders,

SHIELDING DIAGRAM WALL B

Figure A8-2

the activity at the midpoint was taken as the activity throughout the cylinder length. However, the N^{16} half-life is short, and the activity changes along the tube path. It was assumed that increases and decreases in activity about the midpoint of the exchanger would compensate for each other, and thus the midpoint represents a good generalization for the reactivity in the heat exchanger.

The activity at the midpoint of the exchanger was determined by calculating the header residence time and the delay time through the tubes to the midpoint. It was found that $Q_{vm} = 0.66 Q_v^{D4}$.

The photon flux is given by

$$\psi = \frac{B Q_v c^2 F(\Phi_1 b_2)}{2(a+z)} \quad 0.363$$

See data for point PB1 (fourth bank exchangers) on page A-9.

It was assumed that the thickness, t , of the heat exchanger wall is 1/2-inch (steel).

$$\mu t = (1.25) (0.225) = 0.28$$

$$\mu_s c = (0.06) (45) = 2.70$$

$$\mu_s z = 1.3, \quad z = 22.3 \text{ cm}$$

$$Q_v = (0.66) (2.7 \times 10^5) (2.2) = 3.9 \times 10^5 \text{ gammas/cc/sec}$$

By inspection of Figure A8-2 it can be seen that the top inlet header, d , can be neglected.

**DATA USED TO DETERMINE GAMMA FLUX CONTRIBUTION OF
FOURTH BANK HEAT EXCHANGERS TO THE DOSE RATE AT POINT PB1**

	$\tan \Theta$	$\sec \Theta$	x (cm)	x $\sec \Theta$ (cm)	a + z (cm)	$\tan \Theta$	$\bar{\Phi}$
a ₄	0.565	1.15	122	140	380	0.755	37°
b ₄	0.130	1.01	122	123	340	0.870	41°
c ₄	0.304	1.05	122	128	350	0.831	40°

	x $\sec \Theta$	μt	μ_{sz}	b ₂	B	c ² (cm ²)	F($\bar{\Phi}$, b ₂)	Q _v (γ /cc/sec)
a ₄	8.4	0.3	1.3	10.0	4.8	2 x 10 ³	1.6 x 10 ⁻⁵	3.9 x 10 ⁵
b ₄	7.4	0.3	1.3	9.0	4.4	2 x 10 ³	4.8 x 10 ⁻⁵	3.9 x 10 ⁵
c ₄	7.7	0.3	1.3	9.3	4.6	2 x 10 ³	3.5 x 10 ⁻⁵	3.9 x 10 ⁵

	ψ (γ /cm ² sec)
a ₄	0.78 x 10 ²
b ₄	2.42 x 10 ²
c ₄	1.80 x 10 ²

TOTAL 5.00 x 10² (0.363) = 1.8 x 10² gammas/cm²/sec

b. Radiation contribution of fourth bank headers.

The headers are treated as spherical sources with point source geometry, where the fraction of radiation seen by a detector is given by

$$\frac{I}{I_0} = F_p(\mu a)$$

where a = radius in cm

μ = linear absorption coefficient

$p = 3/2$ for a sphere

and I and I_0 are the intensities from large sources seen by a detector with and without self absorption respectively. The function $F_p(\mu a)$ can be found in APEX 176.

It was found that the radius of a sphere with a volume equal to that of a header was 53.5 cm.

Then

$$\mu a = (0.027) (53.5) = 1.44$$

$$F_p(\mu a) = \frac{I}{I_0} = 0.42$$

and with a wall thickness of 11/64 inches

$$\mu t = (0.225) (1.75) = 0.4$$

and the volume, $V = 6.1 \times 10^5$ cc

From Figure A8-3, d in one plane was obtained.

THIRD BANK HEADERS

Figure A8-3

THIRD BANK HEAT EXCHANGERS

Figure A8-4

Since $\tan \theta_1 = 1.03$, $\sec \theta_1 = 1.43$

$$d_1 = (1.43) (12) (11.2) 2.54 = 487 \text{ cm}$$

$$x_1 = (122) 1.43 = 174 \text{ cm.}$$

Since we have identical geometry, let

$$Q' = (Q_V^{E_4} + Q_V^{D_4}) V$$

$$Q' = (1.2 + 2.7)(2.2 \times 10^5)(6.1 \times 10^5) = 5.2 \times 10^{11} \text{ gammas/sec}$$

$$\psi = \frac{I}{I_0} \frac{BQ e^{-b_1}}{4\pi d^2}$$

See data for point PB1 (fourth bank headers) on page A8-12.

**DATA USED TO DETERMINE GAMMA FLUX CONTRIBUTION OF
FOURTH BANK HEADERS TO THE DOSE RATE AT POINT PB1**

	sec Θ_2	x_1 (cm)	x_1 sec Θ_2 (cm)	d_1 (cm)	d_2 (cm)	$4\pi d_2^2$ (cm ²)	x_1 sec Θ_2	μt
a_4	1.15	174	200	487	560	3.9×10^6	12.0	0.4
b_4	1.01	174	176	487	492	3.0×10^6	10.6	0.4
c_4	1.05	174	183	487	513	3.3×10^6	11.0	0.4

	b_1	e^{-b_1}	$Q' (\gamma/\text{sec})$	B	$(\gamma/\text{cm}^2/\text{sec})$
a_4	12.4	4.4×10^{-6}	5.2×10^{11}	5.8	0.03×10^2
b_4	11.0	1.6×10^{-5}	5.2×10^{11}	5.1	0.14×10^2
c_4	11.4	1.1×10^{-5}	5.2×10^{11}	5.5	0.10×10^2

TOTAL $0.27 \times 10^2 \times 0.42 = 0.11 \times 10^2$ gammas/cm²/sec

c. Radiation contribution of third bank headers.

This calculation is similar to the preceeding one, except that only contribution of c_3 was determined. It was then multiplied by a factor of 1.5 to allow for some contribution from a_3 and b_3 . See Figure A8-2 for the shadow shielding effect.

Here we let

$$Q' = (Q_v^{D_3} + Q_v^{E_3}) V$$

$$Q' = (3.8 + 1.6) (2.2 \times 10^5) (6.1 \times 10^5) = 7.3 \times 10^{11} \text{ gammas/sec}$$

$$\psi_\gamma = (0.78 \times 10^2) (0.42) (1.5) = 0.51 \times 10^2 \text{ gammas/cm}^2/\text{sec}$$

d. Radiation contribution of third bank heat exchangers

As is shown in Figure A8-4 the angle Φ for this case is only 18 degrees, so the argument of Section D-1a is not adaptable to this case. When the delay time in the header was considered, it was found that $0.81 Q_v^{D_3}$ is the activity at the inlet to the tube section. Here it was first assumed that $0.81 Q_v^{D_3}$ was the activity from D to m, and $Q_v^1 = 0.59 Q_v^D$ the activity from E to m. Then the expression for the intensity from the sections El and nD is

$$\psi_{\gamma_{\text{El nD}}} = \frac{0.363}{2} \frac{B}{2(a+z)} (Q_v^1 + 0.81 Q_v^D) c^2 \left[F(18^\circ, b_2) - F(10^\circ, b_2) \right] \quad (20)$$

It was assumed that $Q_{vm} = 0.66 Q_v^D$ was the activity from 1 to n, so

$$\psi_Y^{\ln} = 0.363 \frac{B}{2(a+z)} (0.66 Q_v^{D_3}) c^2 F(10^0, b_2)$$

or

$$= \frac{0.363}{2} \frac{Bc^2}{2(a+z)} \left\{ 1.4 Q_v^{D_3} \left[F(18^0, b_2) - F(10^0, b_2) \right] + 1.3 Q_v^{D_3} F(10^0, b_2) \right\}$$

This expression was used to find the radiation contribution of c_3 and was multiplied 1.5 to allow for a_3 and b_3 , which are shadow shielded. The flux from d_3 was neglected entirely.

The reduction in activity for the various points can be determined by referring to Table A-VI which lists the velocities for the various section.

DATA USED TO DETERMINE GAMMA FLUX CONTRIBUTION OF THIRD BANK HEAT EXCHANGERS TO THE DOSE RATE AT POINT PB1

μx	μt	$\mu_s z$	b_2	B	$c^2(\text{cm}^2)$	$a + z(\text{cm})$
7.3	0.3	1.3	8.9	4.5	2×10^3	8.8×10^2
			$F(10^0, b_2)$	$F(18^0, b_2)$	ΔF	
			2.2×10^{-5}	3.8×10^{-5}	1.6×10^{-5}	

$$\text{and } Q_v^{D_3} = (3.8 \times 10^5) (2.2) = 8.35 \times 10^5 \text{ gammas/cc/sec}$$

The contribution of the third bank exchangers is

$$\psi_{\gamma} = 1.5 (0.40 \times 10^2) = 0.60 \times 10^2 \text{ gammas/cm}^2/\text{sec}$$

The total dose rate at PB1 is given below.

Dose Rate at PB1

<u>Source</u>	<u>Gamma flux ($\gamma/\text{cm}^2/\text{sec} \times 10^{-2}$)</u>
Fourth bank heat exchangers	1.82
Third bank heat exchangers	0.60
Fourth bank headers	0.11
Third bank headers	0.51
Total	3.04

$$\text{Dose Rate} = \frac{3.04 \times 10^2}{1.4 \times 10^2} = 2.2 \text{ mr/hr}$$

2. PB3 (Wall B, six feet above the pavement and opposite the midpoint of the inlet headers).

a. Radiation contribution of fourth bank downcomer

Referring to Figure A8-5, since Φ is 40 degrees, a cylinder 10 feet long was considered.

$$\psi_{\gamma} = \frac{B Q_v c^2 F(\Phi_1 b_2)}{2(a+z)}$$

where

$$\begin{array}{llll} x = 122 \text{ cm}, & \mu_x = 7.3 & b_2 = 8.1 & a + z = 170 \text{ cm} \\ \mu_t = 0.3 & B = 4.2 & F(\Phi_1 b_2) = 1.1 \times 10^{-4} & \\ \mu_s = 0.5 & z = 18 \text{ cm} & & \end{array}$$

and

$$Q_v = Q_v^{D4} = (2.7 \times 10^5) (2.2) \text{ gammas/cc/sec}$$

$$\psi_\gamma = 7.3 \times 10^2 \text{ gammas/cm}^2/\text{sec}$$

b. Radiation contribution of fourth bank headers

Here again the headers were treated as spherical sources with I/I_0 equal to 0.42, and point source geometry was used. Figures A8-2 and A8-6 show the header locations.

$$\psi_\gamma = I/I_0 \frac{B Q e^{-b_1}}{4 \pi d^2}$$

$$\psi_{\gamma a, b, c} = 3.5 \times 10^2 \text{ gammas/cm}^2/\text{sec}$$

However, the 18-inch horizontal inlet pipes completely shield 20 percent of the header, and it was estimated that the 24-inch downcomer would shield one-half the remainder. Therefore,

$$\psi_{\gamma a, b, c} = (3.5 \times 10^2) (0.4) = 1.4 \times 10^2 \text{ gammas/cm}^2/\text{sec}$$

c. Radiation contribution of fourth bank heat exchangers.

Referring to Figure A8-6, it was determined that Φ_1 is greater than 40 degrees for all the exchangers. Therefore, the approximation for an infinite cylinder was used, where

$$\psi_{\gamma mo} = \frac{1}{2} \frac{B Q_v c^2 F(\pi/2, b_2)}{2(a+z)}$$

Here $Q_v = Q_v^{D4} = 5.95 \times 10^5$ gammas/cc/sec and the total dose rate was determined by comparing it to PB1, where

$$\psi_{\gamma mo} = 1.82 \times 10^2 \text{ gammas/cm}^2/\text{sec}$$

$$\text{and } Q_v = 3.9 \times 10^5 \text{ gammas/cc/sec}$$

thus

$$\psi_{\gamma mo} = \frac{1}{2} \frac{(1.82 \times 10^2) (5.95 \times 10^5)}{(3.9 \times 10^5)} = 1.39 \times 10^2 \text{ gammas/cm}^2/\text{sec}$$

However, this expression includes the length mn as if it were a part of the heat exchanger. Comparing this contribution to that of the fourth bank headers (Section 2b) and adjusting for both the specific activity and a new I/I_0 (since the μ_s changed) yields

$$\mu_s a = (0.06) 45 = 2.7$$

$$I/I_0 = 0.26$$

$$\begin{aligned} \psi_{\gamma m, n} &= (3.5 \times 10^2) \left(\frac{0.26}{0.42} \right) (0.363) (1/2) \\ &= 0.39 \times 10^2 \text{ gammas/cm}^2/\text{sec} \end{aligned}$$

$$\text{So that } \psi_{\gamma} = \psi_{\gamma mo} - \psi_{\gamma mn} = 1.0 \times 10^2 \text{ gammas/cm}^2/\text{sec}$$

d. Radiation contribution of third bank heat exchangers.

The contribution of the third bank heat exchangers was calculated in a manner similar to that used for the fourth bank exchangers, except that $\overline{\Phi}_1$ and $\overline{\Phi}_2$ were sufficiently small that the heat exchanger was divided. (See Figure A8-6.) In this case $\overline{\Phi}_1 = 21$ degrees and $\overline{\Phi}_2 = 36$ degrees. It can then be shown that

$$Q_v^{mo} = Q_v^{D3} - Q_{vm} = 0.34 Q_v^{D3}$$

$$Q_v^{mp} = Q_{vm} = 0.66 Q_v^{D3}$$

The contribution of the header section, mn, was subtracted as before. Then,

$$\psi_r^{np} = \frac{0.363}{2} \frac{Bc^2}{2(a+z)} Q_v^{D3} \left[0.66 F(\overline{\Phi}_2 b_2) + 0.34 F(\overline{\Phi}_1 b_2) \right] - \psi_r^{mn}$$

$$\psi_r^{np} = 0.32 \times 10^2 \text{ gammas/cm}^2/\text{sec}$$

e. Radiation contribution of third bank inlet downcomers.

These downcomers are partially shielded by the fourth bank heat exchangers. The exposed sections were treated as short cylinders, which are also shielded by the fourth bank inlet downcomer. The contribution of the downcomers was found to be

$$\psi_r = 0.22 \times 10^2 \text{ gammas/cm}^2/\text{sec}$$

f. Total dose rate at point PB3.

<u>Source</u>	<u>gamma flux ($\gamma/\text{cm}^2/\text{sec} \times 10^{-2}$)</u>
Fourth bank exchangers	1.0
Fourth bank headers	1.4
Fourth bank downcomers	7.3
Third bank exchangers	0.32
Third bank downcomers	0.22
Total	10.2

$$\text{Dose Rate} = \frac{10.2 \times 10^2}{1.4 \times 10^2} = 7.3 \text{ mr/hr}$$

E. Shielding Provided for Walls A and D

The calculations for the determination of the dose rate at points on Wall A are very similar to the calculations for Wall B except that the 24-inch horizontal pipes are more important and the inlet downcomers less important. Figures A8-7 and A8-8 show the geometries used in the calculations for Wall A.

Figures A8-9 and A8-10 show the geometries used in the calculations for Wall D. It was found that PD2 has the highest dose rate on Wall D. In this calculation the first bank headers were treated with point source geometry, and I/I_0 was determined for a large spherical source. The 18-inch diameter inlet pipes were treated as four-foot long cylinders with the detector opposite their midpoint. The 24-inch diameter downcomer was treated as follows:

WALL A SHIELDING SECOND BANK EXCHANGERS

Figure A8-8

Figure A8-9

WALL D SHIELDING

Figure A8-10

$$\psi_{\gamma} = \frac{1}{2} \frac{B Q_v c^2}{2(a+z)} \left[F(\overline{\Phi}_1 b_2) + F(\overline{\Phi}_2 b_2) \right]$$

It can be seen that the third bank contribution to this point is negligible. It was also determined that PD2 is actually in a greater flux region than a point located between the first and third heat exchanger banks.

F. Shielding Provided for the Pump Pits

The pump pit shielding calculations are based on the assumption that access to a pump pit would be required during reactor operation. The point which will be in the highest flux region is located in pump pit No. 2 when heat exchanger banks 1 and 3 are in operation. There are several sources which contribute to the dose rate at this point. A point in the exact center of the pump pit was taken for the location of a representative detector. Figures A8-11 and A8-12 show the location of the point. As will be shown later, a large amount of the flux originates in the heat exchangers, the first bank inlet riser, and the third bank downcomer. Thus, by referring to Figures A8-13, A8-14, and A8-15, it can be seen that the point chosen is a good representation of the highest flux region. In general, the geometries used in these calculations are similar to those used in previous calculations. The heat exchangers were assumed to be line sources with $I/I_0 = 0.26$, and were divided into two line sources as shown in Figure A8-16. In general the flux for this geometry is given by

$$\psi_{\gamma} = \frac{B Q_L}{4 \pi a} \left[F(\overline{\Phi}_1 b_1) - F(\overline{\Phi}_2 b_1) \right]$$

LOCATION OF DETECTOR IN PUMP PIT

Figure A8-11

LOCATION OF DETECTOR IN PUMP PIT

Figure A8-13

PUMP PIT SHIELDING

PUMP PIT SHIELDING

Figure A8-14

PUMP PIT SHIELDING

HEAT EXCHANGER CONTRIBUTION TO PUMP PITS

In this case we assumed that Q_L^D was the activity from D to m, and Q_L^m was the activity from m to E. This assumption is conservative. Then,

$$Q_L^m = 0.66 Q_L^D$$

and the flux from one heat exchanger at PC1 is

$$\psi_\gamma = \frac{F_p(\mu, c) Q_L}{4\pi a} \left\{ 0.66 B_{1,2} \left[F(\overline{\Phi}_1 b_{1,2}) - F(\overline{\Phi}_2 b_{1,2}) \right] \right. \\ \left. + B_{2,3} \left[F(\overline{\Phi}_2 b_{2,3}) - F(\overline{\Phi}_3 b_{2,3}) \right] \right\}$$

where $F_p(\mu, c) = I_{I_0}$, and $B_{1,2}$, $B_{2,3}$, $b_{1,2}$, and $b_{2,3}$ account for the effective thickness of Wall C due to the difference in elevation between the source and detector.

Now $\Delta F(\overline{\Phi}_1, b)$ is small, so in order to determine its value accurately the following numerical integration was performed:

$$F(\overline{\Phi}_1 b_1) - F(\overline{\Phi}_2 b_1) = \int_{\phi_2}^{\phi_1} e^{-b_1 \sec \phi} d\phi \\ \approx \sum_i e^{-b_i \sec \phi_i} \Delta \phi$$

where $\Delta \phi = 0.01$ radians. Since the above calculation included the header,

an adjustment was made in the header calculation to account for this contribution.

Dose Rate at PC1	
<u>Source</u>	<u>Gamma flux ($\gamma/\text{cm}^2/\text{sec} \times 10^{-2}$)</u>
Headers	1.6
Heat exchangers	6.6
Top inlet, first bank	4.8
Inlet riser, first bank	8.4
Downcomer, third bank	1.8
Outlet, third pump (H. E. room)	0.2
Outlet, from third bank to downcomer	0.96
Outlet, first pump	1.6
First pump	1.8
Outlet, third pump	2.9
24-inch elbow, third pump	2.1
Third pump	1.3
24-inch return first bank to 36-inch pipes	0
Vertical inlet to first pump	0
24-inch horizontal return from first and third banks	0
36-inch pipes	<u>0</u>
Total	34.06

$$\text{Dose rate} = \frac{34.06 \times 10^2}{1.4 \times 10^2} = 24 \text{ mr/hr}$$

G. Shielding Provided for Wall C and Pump Motor Region

In addition to a point within the pump pit, another point on Wall C was considered. This point, PC2, is located outside and above the first pump pit and opposite the first bank inlet riser. The location of PC2 is shown on Figures A8-13, A8-14 and A8-15. This point was considered to be the hottest point above the pump pit roof. In this and in previous calculations the 24 cu ft of water which the pumps contain was assumed to occupy an outer annulus surrounded by one inch of steel.

Dose Rate at PC2

<u>Source</u>	<u>Gamma flux ($\gamma/\text{cm}^2/\text{sec} \times 10^{-2}$)</u>
Riser, first bank	5.8
Headers, first bank	0.4
Downcomer, second bank	0.2
Pump outlet, first bank	1.6
Pump, first bank	<u>7.5</u>
Total	15.5

$$\text{Dose rate} = \frac{15.5 \times 10^2}{1.4 \times 10^2} = 11 \text{ mr/hr}$$

The points used in the pump motor region are shown in Figures A8-17 and A8-18. It was found that the highest flux region near the motors will be at point PM3, which is located at the floor level and offset from the 36-inch outlet pipe such that the maximum contribution of the inlet header will be received. The flux from the 36-inch pipe at a point three feet above the

SHIELDING OF PUMP MOTOR REGION

Figure A8-17

SHIELDING OF PUMP MOTOR REGION

floor will be approximately the same. In addition, there will be some flux contribution from the pump and its piping.

Dose Rate Three Feet Above PM3

<u>Source</u>	<u>Gamma flux ($\gamma/\text{cm}^2/\text{sec} \times 10^{-2}$)</u>
36-inch inlet pipe	3.6
36-inch outlet pipe	0
24-inch inlet pipe	1.1
Pump	0.7
Pump elbow	1.2
	<hr/>
Total	6.6

$$\text{Dose rate} = \frac{6.6 \times 10^2}{1.4 \times 10^2} = 4.7 \text{ mr/hr}$$

APPENDIX IX

CALCULATIONS FOR CONCRETE BIOLOGICAL SHIELD

A. GAMMA ABSORPTION COEFFICIENTS FOR MAGNETITE CONCRETE

The Kaiser Engineers have specified the following constituents for the magnetite concrete mix:

<u>Material</u>	<u>Mix 1</u>	<u>Mix 2</u>
Cement (lb/cu yd)	462	542
Water (lb/cu yd)	296	291
Magnetite aggregate (lb/cu yd)		
1. Fine	2210	2380
2. Medium	1650	1740
3. Coarse	1650	1220
Density (lb/cu ft)	233	228

Composition of the magnetite by weight is 81 percent Fe_3O_4 , with the remainder made up of SiO_2 and CaO .

Composition of the cement by weight is

CaO	64.31%
SiO_2	23.40
Al_2O_3	3.71
FeO_3	2.53
MgO	2.09
SO_3	1.96
Insoluble	0.11
Loss on ignition	2.01

The absorption coefficients are calculated for Mix 2, since this is the lighter of the two mixes and should give a more conservative value.

The density of the concrete is taken as 3.5 gm/cc which is the minimum specified.

1. Magnetite aggregate

Fraction of iron in the concrete = (magnetite in concrete) x (Fe_3O_4 in magnetite) x (iron in Fe_3O_4)

$$= (0.866) (0.81) (0.725) = 0.508$$

Density of iron in the concrete = $(0.508) (3.5 \text{ gm/cc}) = 1.78 \text{ gm/cc}$

Similarly, the fraction of oxygen in the concrete is

$$(0.866) (0.81) (0.275) (3.5) = 0.675 \text{ gm/cc}$$

The remaining 19 percent of the magnetite is considered SiO_2 . As before

$$(0.866) (0.19) (3.5) = 0.575 \text{ gm/cc}$$

2. Cement

The cement is considered entirely SiO_2 .

Fraction of cement in the concrete is

$$\frac{\text{Cement weight/cu yd}}{\text{Total weight/cu yd}} = \frac{542 \text{ lbs}}{6173 \text{ lbs}} = 0.0878$$

Density of SiO_2 in the concrete = $(0.0878) (0.98) (3.5) = 0.301 \text{ gm/cc}$

3. Water

Fraction of water in the concrete is

$$\frac{291 \text{ lbs}}{6173 \text{ lbs}} = 0.0471$$

Density of the water in the concrete = $(0.0471) (3.5) = 0.165 \text{ gm/cc}$

7 Mev Gammas

3 Mev Gammas

Material	Density in the Concrete (gm/cc)	Mass Absorption Coefficient* (cm ² /gm)	Linear Coefficient (Mass Coeff. x density) (cm ⁻¹)	Mass Absorption Coefficient (cm ² /gm)	Coefficient (Mass Coeff. x density) (cm ⁻¹)
Magnetite					
Iron	1.78	0.030	0.0534	0.036	0.0640
Oxygen	0.675	0.024	0.0162	0.036	0.0243
SiO ₂	0.575	0.0328	0.0189	0.041	0.0236
Cement					
SiO ₂	0.301	0.0328	0.0099	0.041	0.0123
Water	0.165	0.026	0.0043	0.039	0.0064
			0.1027 cm ⁻¹		0.1306 cm ⁻¹

Assuming a 50 percent water loss on curing of the concrete, the linear absorption coefficients are

for 7 Mev gammas $\mu = 0.100 \text{ cm}^{-1}$

for 3 Mev gammas $\mu = 0.127 \text{ cm}^{-1}$

* Mass absorption coefficients are from APEX 176, Miscellaneous Data for Shielding Calculations.

B. DETERMINATION OF CONCRETE BIOLOGICAL SHIELD THICKNESS AND SUFFICIENCY OF THE SHIELD AS DESIGNED

1. Concrete Required to Attenuate Radiation to 1/10 Tolerance

a. Attenuation of 7 Mev Primary Gammas to 1/10 Tolerance

At the concrete face there is a 7 Mev gamma source of 4.2×10^{10} Mev/cm²/sec from core radiation and neutron captures in the thermal shield (see Figure 17-6). This is a plane source proceeding in the outward direction and is treated by the formula $\overline{\Phi}_E = Q_A E B F_O(b)$. To limit radiation outside the shield to 0.75 mr/hr

$$B F_O(b) = \frac{0.75 (900)}{Q_A E}$$

$$\begin{aligned} B F_O(b) &= \frac{0.75 (900)}{4.2 \times 10^{10}} \\ &= 1.61 \times 10^{-8} \end{aligned}$$

where 900 is the conversion factor Mev/cm²/sec to mr/hr.

By successive approximations, $B = 7$ and $b = 17.1$. Since $b = \mu x = 17.1$,

$$x = \frac{17.1}{0.1} = 171 \text{ cm.}$$

Therefore, 171 cm (67 1/2 inches) of magnetite concrete are required to attenuate radiation from 7 Mev primary gammas to 1/10 tolerance.

b. Attenuation of Secondary Gammas to 1/10 Tolerance

The fast neutron current entering the concrete is given by

$$J_+ = \frac{\Phi}{4} - \frac{D}{2} \frac{d\Phi}{dx}$$

The back scattered component is

$$J_- = \frac{\phi}{4} + \frac{D}{2} \frac{d\phi}{dx}$$

Using a diffusion coefficient of 3.0 cm in the external thermal shield and neutron fluxes from IBM 650 calculations (see Figure 17-5), the net current proceeding in the outward direction into the concrete face at the reactor midplane is

$$\begin{aligned} J &= D \nabla \phi = D \phi \frac{d(\ln \phi)}{dx} \\ &= 3 (1.3 \times 10^{11}) (0.072) \\ &= 2.8 \times 10^{10} \text{ neutrons/cm}^2/\text{sec} \end{aligned}$$

This source is assumed thermalized and captured 12 inches into the concrete, giving one 7 Mev gamma per capture. This is a source of 2.0×10^{11} Mev/cm²/sec. Proceeding as in a. above, for 1/10 tolerance

$$\begin{aligned} B F_o(b) &= \frac{(0.75)(900)}{2 \times 10^{11}} \\ &= 3.38 \times 10^{-9} \end{aligned}$$

By successive approximations, $B \approx 8$ and $b = 18.7$.

Therefore, 187 cm (73 1/2 inches) of magnetite concrete are required to attenuate the source to 1/10 tolerance. Adding the 12-inch age displacement gives a total concrete thickness of 85 1/2 inches at the reactor midplane.

c. Attenuation of Fast Neutrons to 1/10 Tolerance

The fast neutron current entering the concrete is 2.8×10^{10} neutrons/cm²/sec (from b. above). Using a relaxation length of 8.2 cm in the concrete, the current is reduced to 1/10 tolerance for 8 neutrons/cm²/sec* attenuated by 71 1/2 inches of concrete.

$$(2.8 \times 10^{10}) e^{-\frac{x}{8.2}} = 8$$

$$e^{-\frac{x}{8.2}} = 2.85 \times 10^{-10} = e^{-22.1}$$

$$x = 181 \text{ cm or } 71 \frac{1}{2} \text{ inches}$$

* 8 nv is 1/10 tolerance for 0.5 Mev neutrons. The majority of the neutrons will be of this energy or less after inelastic scattering in the thermal shields.

2. Radiation Outside of the Shield as Designed

a. Path of Least Attenuation (through the facility tube housings in the shield)

Secondary Gamma Analysis

The path of minimum attenuation through the biological shield, as designed, is shown by the diagram.

Attenuating material along path A

Steel	pipe	1/2 in.
	slab	5 in.
	form	3/4 in.

Total Steel = 6 1/4 inches

Total concrete in path = 96 inches - (12-inch diameter in-shield pipe + 5-inch iron slab) = 79 inches. Subtracting the 12-inch age displacement distance for the secondary gamma source, the effective concrete along the path becomes 79 inches - 12 inches = 67 inches.

The attenuating materials are

	μ (cm ⁻¹)	x (cm)	μx
Concrete	0.10	170	17.0
Steel	0.225	15.9	3.58

$$\text{Total } \mu x = 20.58$$

Using a treatment similar to the preceding sections, radiation outside the shield is

$$\begin{aligned} \bar{\Phi}_E &= \frac{Q_A E B F_o (b)}{900} \\ &= \frac{2 \times 10^{11} (8) (5.2 \times 10^{-11})}{900} \\ &= 0.093 \text{ mr/hr} \end{aligned}$$

This is 1/80 tolerance.

The second most formidable source is fast neutrons which are attenuated by 79 inches of concrete. After attenuation in the concrete the level outside the shield is $(2.8 \times 10^{10}) e^{-\frac{201}{8.2}} = 0.6 \text{ neutrons/cm}^2/\text{sec}$. This is about 1/130 tolerance.

The total radiation outside the shield is therefore about 1/50 tolerance in the worst case giving a safety factor of 5 over the design basis of 1/10 tolerance.

b. Radiation Along an Average Path Through the Shield

Along paths through the shield which do not pass through the vertical sleeves, there are 96 inches of concrete plus the 3/4-inch steel form plate following the external thermal shield. Subtracting the 12-inch age displacement for secondary gammas gives 84 inches of concrete.

The attenuating materials are

	$\mu \text{ (cm}^{-1}\text{)}$	$x \text{ (cm)}$	μx
Concrete	0.10	213	21.3
Steel	0.225	1.9	0.43

$$\text{Total } \mu x = 21.73$$

Radiation outside the shield is

$$\Phi_E = \frac{2 \times 10^{11} (8.5) 1.6 \times 10^{-11}}{900}$$

$$= 0.03 \text{ mr/hr}$$

This is 1/250 tolerance.

Similarly, the fast neutrons are reduced to

$$(2.8 \times 10^{10}) e^{-\frac{244}{8.2}} = 0.0037 \text{ neutrons/cm}^2/\text{sec.}$$

This is 1/21,000 tolerance.

Therefore, along paths through the shield which do not pass through the vertical sleeves there is a safety factor of 25 over the design for 1/10 tolerance.

C. METHOD OF BIOLOGICAL SHIELD ANALYSIS

1. Radiation in Basement Area in Region of Canal Pit

The biological shield is analyzed to determine its adequacy along paths which appear doubtful. Consider the basement area in the region of the canal pit. See Figure A9-1.

Three sources are considered:

- a. Core radiation, consisting of gammas from U^{235} fission and from neutron capture in the aluminum, steel, etc. in the core
- b. 7 Mev gammas from neutron captures in
 - (1) the steel thermal shields
 - (2) the steel grid plate below the core
- c. Neutrons from the reactor core.

2. Method of Calculation

- a. Contributions a and b(1) are obtained from IBM 650 calculations for gamma fluxes in a horizontal direction from the reactor core. See Figure 17-6.

The 7 Mev gamma source at the concrete face is considered an infinite plane source and is attenuated along a path from the concrete face at the core center line to the basement area (see Figure A9-1). Actually the path might be considered looking at the center of the core, but the method used is actually made conservative in this case since less attenuating material is seen.

RADIATION IN BASEMENT NEAR CANAL PIT

From Drawing M-596

The attenuating materials are

$$\text{Concrete} \quad \mu x = (0.10) (194.5) = 19.45$$

$$\text{Water} \quad \mu x = (0.026) (221) = \underline{5.75}$$

$$\text{Total } \mu x = 25.2$$

We use $\bar{\Phi}_E = Q_A E B F_0(b)$, where $b = \text{total } \mu x$.

From Figure 17-6, $Q_A E = 4.2 \times 10^{10} \text{ Mev/cm}^2/\text{sec}$.

Using the buildup factor for water at the total μx ($B = 9.2$) and substituting

$$\begin{aligned} \bar{\Phi}_E &= 4.2 \times 10^{10} (9.2) (3.4 \times 10^{-13}) \\ &= 0.13 \text{ Mev/cm}^2/\text{sec} \end{aligned}$$

$$\text{This is } \frac{0.13}{900} = 0.00015 \text{ mr/hr.}$$

- b. The grid plate is considered to be 65 percent steel and 35 percent water.

The fast flux from the core is considered thermalized and captured in the grid plate one age displacement below the active section of the core.

This is a distance of 11 cm into the grid plate, and each capture is considered to result in the emission of one 7 Mev gamma. The source is

$8.3 \times 10^{14} \text{ Mev/cm}^2/\text{sec}$ which is converted to a point source of $2.8 \times 10^{18} \text{ Mev/sec}$ by multiplying the source by the cross sectional area of the core.

(See Appendix XI-G, Radiation in the Control Rod Access Room During Reactor Operation.)

The attenuating materials are (from Drawing M-596)

Grid plate (65% steel, 35% water)	$\mu x = (0.155) (22.8) = 3.54$
Water	$\mu x = (0.026) (232.6) = 6.05$
Steel	$\mu x = (0.225) (4) = 0.90$
Lead	$\mu x = (0.50) (8.9) = 4.45$
Concrete	$\mu x = (0.10) (198.2) = 19.82$

Total $\mu x = 34.76$

Using a point source treatment

$$\begin{aligned} \bar{\Phi}_E &= \frac{Q_0 E Be^{-\mu x}}{4 \pi a^2} \\ &= \frac{(2.8 \times 10^{18}) (12) (8 \times 10^{-16})}{4 \pi (467)^2} = 9.8 \times 10^{-3} \text{ Mev/cm}^2/\text{sec} \end{aligned}$$

This is 10^{-5} mr/hr

- c. Fast neutrons are readily attenuated, since in this area the water is just as effective as the concrete in attenuating the neutrons. Beyond the external thermal shield there is considerably more than the six feet of concrete required to reduce fast neutrons to 1/10 tolerance (see Appendix IX-B). The shield was found to reduce radiation to less than 1/10 tolerance in the basement area near the canal pit.

Other areas are analyzed similarly.

D. SHIELDING FOR IN-SHIELD FACILITY LOOPS

The experimental loops passing vertically through the concrete biological shield will contain active water, homogeneous fuel, sodium, air, etc., depending on the particular experiment.

1. Active Water

The activity in the water leaving the reactor core is due mainly to N^{16} formed in the reaction $O^{16} (n, p) N^{16}$. In Appendix VII it is shown that 2.4×10^{14} atoms of N^{16} are formed per second in the ETR pressure vessel. If all of these are formed in the core region, there is an average formation of

$$\frac{2.4 \times 10^{14}}{5.1 \times 10^5} = 4.7 \times 10^8 \text{ } N^{16} \text{ atoms/sec/cc}$$

The ETR fuel section is only 60 percent water, so in a pure water region the number of active nuclei which would be expected to form is

$$\frac{4.7}{0.6} = 7.8 \times 10^8 \text{ } N^{16} \text{ atoms/sec/cc}$$

Eighty percent of these active nuclei produce a 6.5 Mev gamma ray. Thus, the number of active nuclei formed which give off 6.5 Mev gamma rays is

$$7.8 \times 10^8 \times 0.8 = 6 \times 10^8 \text{ } N^{16} \text{ atoms/sec/cc}$$

The activity of the water leaving the reactor core is given by

$$Q_v = \frac{AE (1 - e^{-\lambda t_r})}{(1 - e^{-\lambda T_c})}$$

where

$$Q_v = \text{Mev/cc/sec}$$

$$A = \text{the number of active nuclei formed in the core} = 6 \times 10^8 \text{ N}^{16} \text{ atoms/cc/sec}$$

$$\lambda = \text{N}^{16} \text{ decay constant} = 0.094 \text{ sec}^{-1}$$

$$t_r = \text{exposure time in core (sec)}$$

$$T_c = \text{total cycle time (sec)}$$

$$E = \text{energy of N}^{16} \text{ gamma} = 6.5 \text{ Mev}$$

A six-inch diameter experiment loop has a flow rate of about 1000 gpm (2.23 cu ft/sec). This results in an activation time of 0.27 sec and a total cycle time of 6.7 sec for a 75-foot loop.

Substituting gives

$$Q_v = (6 \times 10^8) (6.5) \frac{(1 - e^{-(0.094)(0.27)})}{(1 - e^{-(0.094)(6.7)})}$$

$$Q_v = 2.1 \times 10^8 \text{ Mev/cc/sec}$$

In the region of interest in the biological shield, the water has decayed about 2.5 seconds after leaving the core. The water activity is therefore reduced to

$$(2.1 \times 10^8) e^{-(0.094)(2.5)} = 1.66 \times 10^8 \text{ Mev/cc/sec}$$

The contributing length of the facility tube is taken as 10 feet at a distance of five feet from the tube (one foot outside the shield on the console floor).

This gives an effective angle of $\phi = 90^\circ$. The radiation outside the biological shield is then

$$\overline{\Phi}_E = \frac{Q_v E C^2}{2(a+z)} BF(\phi, b)$$

where

$$\overline{\Phi}_E = \text{Mev/cm}^2/\text{sec}$$

$$Q_v E = \text{source in Mev/cc/sec}$$

$$C = \text{source radius (cm)}$$

$$a = \text{distance from source to detector (cm)}$$

$$z = \text{effective self-absorption distance (cm)}$$

$$B = \text{buildup factor}$$

$$b = \mu x + \mu_s z$$

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

Following the tube there is a 5-inch iron slab, 43 inches of concrete, and a 3/4-inch steel outer concrete form.

The attenuating materials are

$$\text{Concrete} \quad \mu x = (0.1) (108) = 10.8$$

$$\text{Iron} \quad \mu x = (0.225) (14.6) = 3.3$$

$$\mu_s z = (0.1) = 0.1$$

$$\text{Total } b = 14.2$$

$$\frac{\Phi}{E} = \frac{1.66 \times 10^8}{313} (58) (6.5) (2.1 \times 10^{-7}) = 42 \text{ Mev/cm}^2/\text{sec or} \\ 0.05 \text{ mr/hr.}$$

This is less than 1/100 tolerance.

2. Active Sodium

Certain experiments employ sodium as a coolant (in about two-inch diameter tubes at 100 gpm). Activity is due to Na^{24} , which has a 15-hour half-life and decays by emission of a 2.8 Mev and 1.4 Mev gammas. The 1.4 Mev gammas are attenuated in the concrete and iron shielding much more rapidly than the 2.8 Mev gammas. Thus the 2.8 Mev gammas constitute the main source.

Proceeding as in the case of water activity

$$Q_v = nv \sum_a E \frac{(1 - e^{-\lambda t_r})}{(1 - e^{-\lambda T_c})}$$

where

$$\text{activating thermal flux} = 4 \times 10^{14} \text{ n/cm}^2/\text{sec}$$

$$\sum_a = \text{activation cross section} = 0.0127 \text{ cm}^{-1}$$

Substituting gives

$$Q_v = (4 \times 10^{14}) (0.0127) (2.8) \frac{1 - e^{-(1.29 \times 10^{-5}) (0.30)}}{1 - e^{-(1.29 \times 10^{-5}) (7.4)}} \\ = (1.42 \times 10^{13}) \left(\frac{0.30}{7.4} \right) \\ = 5.75 \times 10^{11} \text{ Mev/cc/sec}$$

This source is not decreased significantly by the 2 1/2-second decay time, since the Na^{24} has a 15-hour half-life.

The attenuating materials are

Concrete	$\mu_x = (0.127) (108) = 13.7$
Iron	$\mu_x = (0.27) (14.6) = \underline{3.95}$
	μ_s^z is negligible

$$\text{Total } b = 17.7$$

$$\begin{aligned} \bar{\Phi}_E &= \frac{5.75 \times 10^{11}}{313} (6.45) (15) (5.7 \times 10^{-9}) \\ &= 10^3 \text{ Mev/cm}^2/\text{sec or } 1.7 \text{ mr/hr} \end{aligned}$$

This is about 1/4 tolerance. Hence the shielding provided is quite good, even in the case of an unusually active loop.

3. Neutrons

Experience at the MTR and NRX indicates that delayed neutrons from experimental loops where destructive tests are run will be 10^4 nv or less. Using 8.2 cm as the relaxation length for fast neutrons in magnetite concrete, only a two-foot thickness is required to reduce the neutron dose to 1/10 tolerance (5 nv).

4. Conclusion

Five inches of steel following the experimental pipes, 43 inches of magnetite concrete, and the 3/4-inch steel outer form plate adequately shield the experimental loops. In general, radiation is reduced to 1/10 tolerance, but may be near tolerance in particularly bad instances. In these latter cases, additional shielding external to the biological shield can be added if necessary and/or personnel can be excluded from the area of high radiation.

APPENDIX X

CALCULATIONS FOR TOP AREA SHIELDING

A. SOURCES OF GAMMA RADIATION DURING REACTOR OPERATION

During reactor operation there are two significant sources of gamma radiation:

(1) fission gammas, and (2) gammas from neutron capture in metal. Neutrons captured in water produce soft gammas which are strongly attenuated and can be neglected in comparison to the other sources. Each neutron capture in metal is considered to result in the release of a gamma in the 7 Mev range.

1. The Gamma Spectrum per fission is shown in Table A10I taken from "Introduction to Nuclear Engineering" by Stephenson.

TABLE A10I

Gamma Spectrum Per Fission

<u>Energy Interval (Mev)</u>	<u>Average Photon Energy (Mev)</u>	<u>Gammas Per Fission</u>	<u>Energy (Mev/fission)</u>
0-2	1	9.3	9.3
2-4	3	0.75	2.25
4-6	5	0.10	0.5
6-8	7	0.015	0.105
8-10	9	0.003	0.027

The contribution from the 0-4 Mev range need not be considered, because these soft gammas, as in the water capture, are attenuated very strongly in the large amount of water and other material. The 4-10 Mev range has a total of 0.12 gammas released per fission, and these are considered to have an average energy of 7 Mev.

2. The 7 Mev Source Contributions

a. Fuel element (8 grey rods considered in core).

There are 57 fuel elements with a total volume of 3.0×10^5 cc, of which 40 percent is aluminum.

$$\begin{aligned}Q_v (\text{capture}) &= (nv) \sum_a (\text{volume}) (\text{Mev}) \\&= (3.6 \times 10^{14}) (0.011) (1.2 \times 10^5) (7) \\&= 3.5 \times 10^{18} \text{ Mev/sec} \\Q_v (\text{fission}) &= (nv) (\sum_f) (\text{volume}) (\text{gamma yield}) (\text{Mev}) \\&= (3.6 \times 10^{14}) (0.05236) (1.2 \times 10^5) (0.12) (7) \\&= 1.9 \times 10^{18} \text{ Mev/sec} \\ \text{Total} &= 5.4 \times 10^{18} \text{ Mev/sec}\end{aligned}$$

b. Grey Rods

There are 8 grey rods with a total volume of 4.25×10^4 cc, of which 25 percent is aluminum and 22 percent is nickel.

$$\begin{aligned}Q_v (\text{capture}) &= (2.9 \times 10^{14}) \left[(0.364) (9.7 \times 10^2) + (0.011) (1.1 \times 10^4) \right] (7) \\&= 7.1 \times 10^{18} \text{ Mev/sec}\end{aligned}$$

Contribution from 7 Mev gamma captures in the initial experiments is 8.8×10^{18} Mev/sec.*

Total in core = 23.9×10^{18} Mev/sec = 2.4×10^{19} Mev/sec.

* Detailed calculations are not given here due to the security classification of this data.

B. SHIELDING REQUIRED IN THE TOP HEAD SHIELD TO REDUCE N^{16} GAMMAS TO 1/10 TOLERANCE

The N^{16} source is 4.05×10^5 Mev/cc/sec* at the inlet to the pressure vessel. It takes approximately two seconds for this water to reach the top head areas reducing the activity by $e^{-0.094(2)}$, which gives 3.36×10^5 Mev/cc/sec in the top head region. This is converted to a plane source of 6.5 Mev gammas by multiplying by one relaxation length for the gammas in water. Thus $(3.36 \times 10^5) (38.5) = 1.29 \times 10^7$ Mev/cm²/sec. To determine the required attenuation an infinite plane source treatment is used.

$$\bar{\Phi}_E = \frac{Q_A E B F_0(b)}{2 (9 \times 10^2)}$$

where

$$\bar{\Phi}_E = 0.75 \text{ mr/hr}$$

$$Q_A E = 1.29 \times 10^7 \text{ Mev/cm}^2/\text{sec}$$

$$9 \times 10^2 = \text{conversion factor Mev/cm}^2/\text{sec to mr/hr}$$

B = buildup factor

$$F_0(b) = \int_0^{\infty} \frac{e^{-y}}{y} dy$$

b = μx for the attenuating material

x = thickness in cm.

* See Appendix VIII.

Substituting

$$F_o(b) = \frac{(0.75)(2)(9 \times 10^2)}{B(1.29 \times 10^7)} = \frac{1.045 \times 10^{-4}}{B}$$

By successive approximations, b is found to = 8.4. Then

$$\mu_{\text{magnetite concrete}} = 0.10 \text{ cm}^{-1}$$

$$\mu_{\text{steel}} = 0.225 \text{ cm}^{-1}$$

In regions to the sides of the top cover plate, the vessel wall is 2 1/4 inches of steel. The μx for the steel is therefore $(0.225)(5.71) = 1.3$. For the magnetite concrete shield the μx required is then $8.4 - 1.3 = 7.1$. This corresponds to a thickness of $\frac{7.1}{0.1} = 71 \text{ cm}$, or 28 inches of concrete.

Similarly, in the central section of the top head, the top cover plate is 4 1/2 inches of steel and the concrete requirement in this central section is 23 inches.

C. VOLUME FRACTIONS AND ABSORPTION COEFFICIENTS FOR MATERIALS ABOVE THE REACTOR CORE

1. Steel and Water in Facility Tubes and Casings Region

Volume fraction of metal and water above the core. Metal represents the casings and tubes. The water given is the amount in the tubes. (See G-E Drawing ETR-5528-MTR-642-M-606.)

a. Facility tubes

C33L10

C33F10

C33J10

C33H10

(1) Tubes

$$\text{OD} = (\text{ID of casing} - 2 (1/8\text{-in.}) \text{ air gap})$$

$$= 2.78 - 0.250 = 2.53 \text{ inches}$$

$$\text{ID} = 2.53 - (Q - I)$$

where Q and I are taken from G-E Drawing M-606

$$\text{ID} = 2.53 - (2.03 - 1.57)$$

$$= 2.53 - 0.460 = 2.07 \text{ inches}$$

$$A = \frac{\pi}{4} \left[(\text{OD})^2 - (\text{ID})^2 \right]$$

$$A = \frac{\pi}{4} \left[(2.53)^2 - (2.07)^2 \right] = 1.65 \text{ sq. inches}$$

(2) Casings

Thickness of casing = 0.11 inches

$$\text{OD} = 3 \text{ inches}$$

$$\text{ID} = 3 - 2 (0.11) = 2.78 \text{ inches}$$

$$A = \frac{\pi}{4} \left[(9) - (2.78)^2 \right]$$

$$A = 0.998 \text{ sq. inches}$$

(3) Water in the tubes

$$A = \pi r^2 = 3.375 \text{ sq. inches (each tube)}$$

Three of the facilities are water cooled.

b. Facility tubes

C66M13

C66J13

C66G13

C66G7

(1) Tubes

$$\text{OD} = 5.728 - 0.250 = 5.478 \text{ inches}$$

$$\text{ID} = 5.478 - (Q - I)$$

$$\text{ID} = 5.478 - 0.685 = 4.793 \text{ inches}$$

$$A = \frac{\pi}{4} \left[(5.478)^2 - (4.793)^2 \right]$$

$$A = 5.525 \text{ sq. inches}$$

(2) Casings

Thickness of casing = 0.136 inches

$$\text{OD} = 6 \text{ inches}$$

$$\text{ID} = 6 - 0.272 = 5.728 \text{ inches}$$

$$A = \frac{\pi}{4} \left[(6)^2 - (5.728)^2 \right]$$

$$A = 2.505 \text{ sq. inches}$$

(3) Water in the tubes

$$A = \pi r^2 = 18.04 \text{ sq. inches}$$

Three of the facilities are water cooled.

c. Facility tube

C99M7

(1) Tubes

$$OD = 8.656 - 0.250 = 8.406 \text{ inches}$$

$$ID = 8.406 - 0.484 = 7.922 \text{ inches}$$

$$A = \frac{\pi}{4} \left[(8.406)^2 - (7.922)^2 \right]$$

$$A = 6.20 \text{ sq. inches}$$

(2) Casing

Thickness of casing = 0.171 inches

$$OD = 9 \text{ inches}$$

$$ID = 9 - 0.342 = 8.658$$

$$A = \frac{\pi}{4} \left[(9)^2 - (8.658)^2 \right]$$

$$A = 4.768 \text{ sq. inches}$$

(3) Air in the tube

$$A = \pi r^2 = 49.26 \text{ sq. inches}$$

d. Water between casing OD and square limit of facility

$$4 \left(9 - \frac{\pi}{4} \cdot 9 \right) = 7.73 \text{ sq. in.}$$

$$3 \left(36 - \frac{\pi}{4} \cdot 36 \right) = 23.18$$

$$1 \left(54 - \frac{\pi}{4} \cdot 36 \right) = 25.72$$

$$1 \left(81 - \frac{\pi}{4} \cdot 81 \right) = \underline{17.38}$$

$$\text{Total} = 74.01 \text{ sq. in.}$$

2. Volume Fractions of Steel and Water in Facility Tube Regions

a. Volume fractions of steel in pipes and casings

$$4 \text{ (3 x 3) facilities} \quad 4 (0.998 + 1.65) = 10.60 \text{ sq. in.}$$

$$4 \text{ (6 x 6) facilities} \quad 4 (2.505 + 5.525) = 32.12$$

$$1 \text{ (9 x 9) facility} \quad 4.77 + 6.20 = \underline{10.97}$$

$$\text{Total} = 53.69 \text{ sq. in.}$$

$$\text{Total cross sectional area of active section of core} = 864 \text{ sq. in.}$$

$$\text{Volume fraction of steel} = \frac{53.69}{864} = 0.062$$

b. Volume fraction of water

(1) Water in facilities

$$3 (3 \times 3) \text{ facilities} \quad 3 (3.375) \quad = \quad 10.13 \text{ sq. in.}$$

$$3 (6 \times 6) \text{ facilities} \quad 3 (18.04) \quad = \quad \underline{54.12}$$

$$\text{Total} \quad = \quad 64.25 \text{ sq. in.}$$

$$\text{Volume fraction of water in facilities} = \frac{64.25}{864} = 0.074$$

(2) Water between facility tube casings and the square limit of the facility

$$\text{Total water in these regions} = 74.01 \text{ sq. in.}$$

$$\text{Volume fraction} = \frac{74.01}{864} = 0.086$$

3. Volume Fraction of Water Above Fuel Element Region

$$\frac{49}{96} \frac{(3 \times 3) \text{ squares}}{(3 \times 3) \text{ squares}} = 0.51$$

4. Grey and Black Control Rods

a. Four black rods

(1) Cross sectional area of aluminum

$$4 \left[3 (0.203) - (0.203)^2 \right] = 4 (0.609 - 0.0412) \\ = 2.271 \text{ sq. in.}$$

$$\text{Volume fraction of aluminum for 9-sq. in. rod} = \frac{2.271}{9.000} = 0.252$$

(2) Cross sectional area of steel

$$(2.5)^2 - (2.0625)^2$$

$$= 1.996 \text{ sq. in.}$$

$$\text{Volume fraction of steel} = \frac{1.996}{9.000} = 0.222$$

(3) Cross sectional area of water

$$\text{Total rod cross sectional area} = 9.000 \text{ sq. in.}$$

$$\text{Area of metal} = 4.267 \text{ sq. in.}$$

$$\text{Area of water} = 4.733 \text{ sq. in.}$$

$$\text{Volume fraction of water} = \frac{4.733}{9.000} = 0.526$$

b. Twelve grey rods

The cross sectional areas of the grey rods are the same as for the black rods except the steel in the black rods is replaced by nickel.

c. Volume fractions for metal and water of control rods over the total cross sectional area of the core

(1) Volume fraction of nickel

$$\frac{12}{96} (0.222) = 0.028$$

(2) Volume fraction of steel

$$\frac{4}{96} (0.222) = 0.009$$

(3) Volume fraction of aluminum

$$\frac{16}{96} (0.252) = 0.042$$

(4) Volume fraction of water

$$\frac{16}{96} (0.526) = 0.088$$

5. Summary of Volume Fractions

a. Steel

black rods 0.009

facility tubes and casings 0.062

Total steel = 0.071

b. Aluminum

black and grey rods 0.042

c. Nickel

grey rods 0.028

d. Water

black and grey rods 0.088

water in facilities 0.074

between facility tube casings and
square limit of the facilities 0.086

water above fuel element region 0.510

Total water = 0.758

6. Fraction of Neutron Captures in the Metal in the First Three Feet Above
the Active Section of the Core

a. Aluminum captures

$$\begin{aligned} (\sum_a) (\text{volume fraction of aluminum}) &= (0.0112) (0.042) \\ &= 4.7 \times 10^{-4} \end{aligned}$$

b. Steel captures

$$(\sum_a) \text{ (volume fraction of steel)} = (0.225 \times 0.071) = 1.60 \times 10^{-2}$$

c. Nickel captures

$$(\sum_a) \text{ (volume fraction of nickel)} = (0.364) (0.028) = 1.0 \times 10^{-2}$$

d. Water captures

$$(\sum_a) \text{ (volume fraction of water)} = (0.0197) (0.758) = 1.49 \times 10^{-2}$$

Fraction of total neutron captures that occur in the metal, giving 7 Mev gammas, is:

$$\frac{2.65 \times 10^{-2}}{4.14 \times 10^{-2}} = 0.64$$

7. Effective 7 Mev Gamma Absorption Coefficient for the First Three Feet
Above the Active Section of the Core

a Aluminum

$$(\mu) \text{ (volume fraction)} = (0.068) (0.042) = 2.9 \times 10^{-3} \text{ cm}^{-1}$$

b. Steel

$$(\mu) \text{ (volume fraction)} = (0.225) (0.0713) = 1.6 \times 10^{-2} \text{ cm}^{-1}$$

c. Nickel

$$(\mu) \text{ (volume fraction)} = (0.285) (0.0276) = 7.9 \times 10^{-3} \text{ cm}^{-1}$$

d. Water

$$(\mu) \text{ (volume fraction)} = (0.026) (0.758) = 2.0 \times 10^{-2} \text{ cm}^{-1}$$

$$\text{Effective } \mu = 0.047 \text{ cm}^{-1}$$

8. Effective 7 Mev Gamma Absorption Coefficient for Region Above the First Three Feet Above the Active Section of the Core

In this region the materials are water plus steel in the facility tubes and casings.

a. Steel

$$(\mu) \text{ (volume fraction)} = (0.225) (0.062) = 0.0139 \text{ cm}^{-1}$$

b. Water

$$\text{Steel in tubes and casings} = 0.062$$

$$\text{Air in tubes and casings} = 0.101$$

$$\text{Volume fraction of water} = (1.000 - 0.163) = 0.837$$

$$(\mu) \text{ (volume fraction)} = (0.026) (0.837) = 0.0218 \text{ cm}^{-1}$$

$$\text{Effective } \mu = 0.036 \text{ cm}^{-1}$$

D. ATTENUATION OF CORE AND SECONDARY GAMMAS

Geometry

		<u>inches</u>	<u>cm</u>	<u>$\mu(\text{cm}^{-1})$</u>
H	magnetite concrete	-----	-----	0.1
G	<u>steel</u>	4 1/2	12.7	0.225
F	water	228	579	0.026
E	water and metal	36	91.4	0.047
D	↑ 2" ↓ 4.5"			
C	.			
B	core	36	91.4	0.053
A	.			

The core gamma source is considered as four equal point sources, A through D.

The total source is 2.4×10^{19} Mev/sec, so that each point source is 6×10^{18}

Mev/sec. The secondary source, E, is 2.7×10^{18} Mev/sec. These five sources are attenuated through the 4 1/2 inches of steel above the water. The point source formula is used.

$$\Phi_E = \frac{B \phi_o E e^{-\mu x}}{4 \pi a^2 (900)}$$

where

B = buildup factor

ϕ_0 = source (Mev/sec)

μ = absorption cross section

x = shield thickness

a = total path length

$\bar{\Phi}_E$ = intensity (mr/hr)

900 = conversion factor Mev/cm²/sec to mr/hr

For attenuation through the stainless steel, the following values of a^2 and μx are determined from the geometry for the five sources.

<u>Source</u>	<u>a^2 (cm²)</u>	<u>μx (total)</u>
A	5.88×10^5	27.2
B	5.52×10^5	25.8
C	5.17×10^5	24.4
D	4.83×10^5	23.0
E	4.60×10^5	22.0

The buildup factor ≈ 10 for all cases. Substituting for Case A,

$$\begin{aligned}\bar{\Phi}_A &= \frac{(10) (6 \times 10^{18})}{4 \pi (5.88 \times 10^5) (900) (6.5 \times 10^{11})} \\ &= 0.014 \text{ mr/hr}\end{aligned}$$

The radiation intensities from the other four sources are calculated similarly.

<u>Source</u>	<u>Intensity above steel (mr/hr)</u>
A	0.014
B	0.06
C	0.257
D	<u>1.12</u>
Total from core	1.45
E	<u>1.48</u>
Total from core and capture	2.93 mr/hr

This represents the core and capture gamma dose entering the concrete above the cover plate. The attenuation up to this level has been $\mu x = 22.0$ (measured from Source E). The additional attenuation required to reduce this radiation to 1/10 tolerance is $\mu x = 1.36$, equivalent to 5.4 inches of magnetite concrete. Thus, the total attenuation required for core and capture gammas along any path through the top shield will be $\mu x = 22.0 + 1.36 \approx 23.4$, measured from two inches above the active section of the core.

E. Na^{24} RECOIL ACTIVITY IN THE ETR

During reactor operation, Al^{27} nuclei near the surface of the fuel plates capture fast neutrons and recoil into the coolant water. These nuclei, with the instantaneous emission of alpha particles, become active Na^{24} atoms. Na^{24} decays with a 14.9-hour half-life to Mg^{24} in two steps, emitting 2.8 and 1.4 Mev gammas.

The following data was used in the determination of Na^{24} activity:

R = recoil range (2.4×10^{-4} cm)*

$\sum_R$ = fast neutron cross section of Al^{27} ($R = 0.6$ mb)

Φ_f = fast flux (1.5×10^{15} neutrons/cm²/sec)

A = exposed aluminum area (1.72×10^6 cm²)

V = coolant volume (2.1×10^8 cc)

If it is assumed that one quarter of the interactions within the recoil range of the surface eject a Na^{24} nucleus into the water and that at saturation (since the cycle time is much shorter than the Na^{24} half-life) the rate of decay is equal to the rate of production, then

$$\begin{aligned} \text{the number of } \text{Na}^{24} \text{ atoms} \\ \text{entering the coolant/sec} &= \frac{1}{4} \sum_R \Phi_f AR \\ &= 5.6 \times 10^{12} \end{aligned}$$

$$\text{and saturated activity} = \frac{5.6 \times 10^{12}}{V} = 2.7 \times 10^4 \text{ disintegrations/cc/sec.}$$

* M. T. R. Handbook, ORNL 963, Shielding Section

APPENDIX XI

CALCULATIONS FOR BOTTOM AREA SHIELDING

A. SOURCES OF GAMMA RADIATION IN SUB-PILE ROOM FOUR HOURS AFTER SHUTDOWN

1. U^{235} Decay Gamma Sources Four Hours After Shutdown Following 480 Hours Reactor Operation* are given below

<u>Gamma Energy (Mev)</u>	<u>Source (Mev/sec/watt)</u>	<u>For operation at 175 MW (Mev/sec)</u>
1.65	2.0×10^9	3.5×10^{17}
2.1	3.3×10^8	5.8×10^{16}
2.5	1.9×10^8	3.3×10^{16}
2.8	1.2×10^7	2.1×10^{15}

2. Decay Gammas from Materials Irradiated in the Reactor during Operation

Decay gamma intensities from aluminum, steel and beryllium in the reactor core were calculated on the basis of Oak Ridge experimental data† for irradiation of commercial materials at a thermal flux of 10^{12} n/cm²/sec. The experimental values for aluminum, steel and beryllium listed below were used in the analyses that follow.

Type 2S aluminum

Four hours decay following irradiation for 15,000 hours = 2.5×10^7 photons/sec/gm.

Type 304 stainless steel

Four hours decay following irradiation for 1,000 hours = 3.2×10^8 photons/sec/gm.

* The methods of APEX 134 are used. See Appendix XI-C.

† MTR Handbook, ORNL 963, Shielding Section.

Beryllium

Four hours decay following irradiation for 12,000 hours = 6.9×10^6
photons/sec/gm.

The decay gamma energy is 1.2 Mev, but to be conservative it is considered to be 1.65 Mev and this contribution is added to the 1.65 Mev U^{235} decay gammas from the fuel.

a. Decay Gammas from Aluminum

(1) Decay Gammas from Aluminum in the Fuel Elements

The fuel elements are 40 percent aluminum.

$$\begin{aligned}\text{Volume of Al} &= (58 \text{ elements}) (9 \text{ in.}^2) (37.5 \text{ in.}) (0.4) \\ &= 7.83 \times 10^{13} \text{ cu. in.}\end{aligned}$$

$$= 1.28 \times 10^5 \text{ cc}$$

$$\text{Weight of Al} = 3.46 \times 10^5 \text{ gm}$$

From UNIVAC calculations, the average thermal flux in the fuel elements with no grey rods in the core is 3.1×10^{14} n/cm²/sec. With eight grey rods inserted, the average thermal flux is about 3.6×10^{14} n/cm²/sec.

$$\begin{aligned}\text{Source} &= (2.5 \times 10^7 \text{ photons/sec/gm}) \frac{(3.6 \times 10^{14})}{10^{12}} (3.46 \times 10^5 \text{ gm})(1.65 \text{ Mev}) \\ &= 5.18 \times 10^{15} \text{ Mev/sec}\end{aligned}$$

The other contributions are similarly determined.

(2) Decay Gammas from Aluminum in the Grey Rods

There are eight grey rods in the core during an average operating cycle.

$$\text{Weight of Al} = 3.01 \times 10^4 \text{ gm}$$

The activating flux is 80 percent of the average flux in the fuel section, producing a source of 3.57×10^{14} Mev/sec.

(3) Decay Gammas from the Aluminum Reflector and Filler Pieces

$$\text{Volume of Al (Reflector)} = 4.72 \times 10^5 \text{ cc}$$

$$\text{Volume of Al (Filler)} = 2.98 \times 10^5 \text{ cc}$$

$$\text{Total weight of Al} = 2.08 \times 10^6 \text{ gm}$$

From IBM 650 calculations, the average activating thermal flux is 7.0×10^{13} n/cm²/sec.

$$\text{Source} = 6.00 \times 10^{15} \text{ Mev/sec}$$

(4) Decay Gammas from Aluminum in the Initial Experiments

A typical initial experiment is considered.

$$\text{Volume fraction of aluminum} = 0.487$$

$$\text{Weight of aluminum} = 9.77 \times 10^4 \text{ gm}$$

From UNIVAC calculations, the average thermal flux = 2.6×10^{14} n/cm²/sec.

$$\text{Source} = 8.70 \times 10^{14} \text{ Mev/sec}$$

The other experiments are analyzed similarly. *

* Details of the experiment source calculations are not included here due to the classified nature of the constituents.

(5) Summary of Aluminum Decay Gamma Sources

<u>Material</u>	<u>Source (Mev/sec)</u>
Fuel section	5.8×10^{15}
Grey rods	0.36×10^{15}
Reflector and filler pieces	6.00×10^{15}
Experiments	1.39×10^{15}
Total = 1.29×10^{16} Mev/sec	

b. Decay Gammas from Steel

(1) Decay Gammas from Steel in the Grey Rods

In the analysis the nickel was treated as steel.

$$\text{Weight of steel} = 7.68 \times 10^4 \text{ gm}$$

$$\text{Source} = 1.17 \times 10^{16} \text{ Mev/sec}$$

(2) Decay Gammas from Steel in the Initial Experiments

A typical initial experiment is considered.

Volume fraction of the steel is 0.14.

$$\text{Weight of steel} = 8.16 \times 10^4 \text{ gm}$$

$$\text{Source} = 9.3 \times 10^{15} \text{ Mev/sec}$$

(3) Summary of Steel Decay Gamma Sources

<u>Material</u>	<u>Source (Mev/sec)</u>
Grey Rods	1.17×10^{16}
Experiments	1.72×10^{16}
Total = 2.89×10^{16} Mev/sec	

c. Decay Gammas from the Beryllium Reflector

Reflector volume = 3.67×10^5 cc

Reflector weight = 6.75×10^5 gm

The average thermal flux in the reflector is 2.3×10^{14} n/cm²/sec.

Source = 1.77×10^{15} Mev/sec

3. Summary of 1.65 Mev Decay Gamma Sources

<u>Material</u>	<u>Source Mev/sec</u>
U ²³⁵	3.50×10^{17}
Aluminum	0.13×10^{17}
Steel	0.29×10^{17}
Beryllium	0.02×10^{17}
Total	$= 3.9 \times 10^{17}$ Mev/sec

B. VOLUME FRACTIONS AND GAMMA ABSORPTION COEFFICIENTS FOR THE REACTOR CORE AND THE MATERIAL BELOW THE CORE

1. Materials in the Reactor Core

The relative volumes of materials in the fuel elements, control rods, and experiments given in Appendix XI-C were used. The volume fractions and gamma absorption coefficients for the core were computed and are summarized below.

Gamma Absorption Coefficients for Reactor Core

<u>Material</u>	<u>V.F.</u>	<u>7 Mev</u>		<u>2.8 Mev</u>	
		<u>μ</u>	<u>μ V.F.</u>	<u>μ</u>	<u>μ V.F.</u>
Water	0.47	0.026	0.0122	0.04	0.019
Aluminum	0.37	0.068	0.0252	0.094	0.035
Steel	0.07	0.225	<u>0.0158</u>	0.27	<u>0.019</u>
Effective μ_s			0.053 cm^{-1}		0.073 cm^{-1}

<u>Material</u>	<u>2.5 Mev</u>		<u>2.1 Mev</u>		<u>1.65 Mev</u>	
	<u>μ</u>	<u>μ V.F.</u>	<u>μ</u>	<u>μ V.F.</u>	<u>μ</u>	<u>μ V.F.</u>
Water	0.043	0.020	0.048	0.023	0.055	0.026
Aluminum	0.1	0.037	0.11	0.041	0.13	0.048
Steel	0.29	<u>0.020</u>	0.31	<u>0.022</u>	0.35	<u>0.025</u>
Effective μ_s		0.077 cm^{-1}		0.086 cm^{-1}		0.099 cm^{-1}

2. Region Between Grid Plate and Bottom Head

a. Aluminum Housings and Shafts for Control Rod Units

Aluminum cross section = 3 in. OD and 2 5/8 in. ID

Cross sectional area of Al
in one control rod unit = $\pi \left[(1.5)^2 - (1.312)^2 \right] = 0.53 \pi \text{ sq. in.}$

in 16 control rod units = 26.6 sq. in.

b. Steel Facility Tubes

Facility tube dimensions are taken at the protection plate level. See G-E Drawing M-606.

(1) Tube wall

For example, consider Tube L10. There are four tubes of this size.

Total cross sectional area = $4 \pi \left[\frac{(2.03)^2 - (1.573)^2}{4} \right] = 5.5 \text{ sq. in.}$

(2) Tube casing

Insulating air annulus thickness = 0.125 in.

Tube casing thickness = 0.11 in.

Casing ID = 2.03
- .25
2.28 in.

OD = 2.28
- .22
2.50 in.

Total casing area = $4 \pi \left[\frac{(2.5)^2 - (2.28)^2}{4} \right] = 3.3 \text{ sq. in.}$

Proceeding similarly for the other facilities gives a total cross sectional steel area of 33.73 sq. in.

c. Insulating Air Annuli and Air-cooled Facilities

(1) Air Annuli

Air in the insulating annuli is found to be 11.24 sq. in.

(2) Air-cooled facilities

It is assumed one "3 by 3", one "6 by 6", and one "9 by 9" facility are air-cooled. The true dimensions are obtained from G-E Drawing M-606.

Total cross sectional area
of the air = 28.98 sq. in.

Total cross sectional area
above the core (annuli
and tubes) = 40.22 sq. in.

d. Water

The cross sectional area of the active section of the core was conservatively taken as 900 sq. in.

Total cross sectional area
of water = (cross sectional area of the
active core)
- (total cross sectional area
of the aluminum, steel and
air below the grid plate)
= 800 sq. in.

The volume fractions and gamma absorption coefficients for the region between the grid plate and bottom head were computed and are summarized on the next page.

Gamma Absorption Coefficients for "Water" Between
Grid Plate and Bottom Head

<u>Material</u>	<u>V. F.</u>	<u>7 Mev</u>		<u>2.8 Mev</u>	
		<u>μ</u>	<u>μ V.F.</u>	<u>μ</u>	<u>μ V.F.</u>
Water	0.89	0.026	0.0232	0.04	0.0356
Steel	0.0375	0.225	0.0084	0.27	0.0101
Aluminum	0.028	0.068	<u>0.0019</u>	0.094	<u>0.0026</u>
Effective μ		0.0335 cm ⁻¹		0.0483 cm ⁻¹	

<u>Material</u>	<u>2.5 Mev</u>		<u>2.1 Mev</u>		<u>1.65 Mev</u>	
	<u>μ</u>	<u>μ V.F.</u>	<u>μ</u>	<u>μ V.F.</u>	<u>μ</u>	<u>μ V.F.</u>
Water	0.043	0.0382	0.048	0.0428	0.055	0.0490
Steel	0.29	0.0109	0.31	0.0116	0.35	0.0131
Aluminum	0.10	<u>0.0028</u>	0.11	<u>0.0031</u>	0.13	<u>0.0036</u>
Effective μ	0.0519 cm ⁻¹		0.0575 cm ⁻¹		0.0657 cm ⁻¹	

3. Bottom Head

Gamma absorption coefficients for the bottom head were obtained similarly to the preceeding sections and are summarized below.

Gamma Absorption Coefficients for the Bottom Head

<u>Material</u>	<u>V. F.</u>	<u>7 Mev</u>		<u>2.8 Mev</u>	
		<u>μ</u>	<u>μ V.F.</u>	<u>μ</u>	<u>μ V.F.</u>
Steel	0.865	0.225	0.195	0.27	0.233
Water	0.0725	0.026	0.0019	0.04	0.0029
Aluminum	0.0196	0.068	<u>0.0013</u>	0.094	<u>0.0019</u>
Effective μ		0.198 cm ⁻¹		0.238 cm ⁻¹	

Gamma Absorption Coefficients for the Bottom Head (Cont'd)

<u>Material</u>	<u>2.5 Mev</u>		<u>2.1 Mev</u>		<u>1.65 Mev</u>	
	<u>μ</u>	<u>μ V.F.</u>	<u>μ</u>	<u>μ V.F.</u>	<u>μ</u>	<u>μ V.F.</u>
Steel	0.29	0.250	0.31	0.268	0.35	0.302
Water	0.043	0.0031	0.048	0.0035	0.055	0.0040
Aluminum	0.10	<u>0.0020</u>	0.11	<u>0.0022</u>	0.13	<u>0.0026</u>
Effective μ		0.255 cm ⁻¹		0.274 cm ⁻¹		0.309 cm ⁻¹

4. Region Between Bottom Head and Roof of Control Rod Access Room

This area is penetrated by control rod units, facility tubes, and other mechanisms. Only 7 Mev and 2.8 Mev gammas are of interest here.

Gamma Absorption Coefficients for Region Between Bottom Head and Roof of Control Rod Access Room

<u>Material</u>	<u>V.F.</u>	<u>7 Mev</u>		<u>2.8 Mev</u>	
		<u>μ</u>	<u>μ V.F.</u>	<u>μ</u>	<u>μ V.F.</u>
Steel	0.0412	0.225	0.0093	0.27	0.0111
Water	0.027	0.026	<u>0.0007</u>	0.04	<u>0.0011</u>
Effective μ			0.010 cm ⁻¹		0.012 cm ⁻¹

C. FISSION PRODUCT DECAY GAMMA ENERGY SPECTRUM

A fission product decay gamma energy spectrum was calculated for 480 hours reactor operation for 2, 4 and 6 hours after shutdown. The nuclides considered were those with a significant half-life, fairly high gamma energy, and a relatively high fission yield. The essential data are tabulated in Table A11-I.

TABLE A11-I*

Mass Number	Nu- clide	Half-life	Fission Yield (nuclide/ fission) Y	% Disinte- grations (gammas/ nuclide) g	Mev/ Gamma E	Mev/sec/operating watt N**
88	Rb	17.8 min	0.031	0.21 0.01	1.86 2.80	4.09×10^8 2.93×10^8
91	Sr	9.7 hr	0.050	0.40	1.30	8.79×10^8
132	I	2.4 hr	0.0360	0.50 0.027	1.40 2.00	8.52×10^8 6.57×10^7
133	I	22.4 hr	0.0460	0.01	1.40	2.18×10^7
135	I	6.68 hr	0.0560	0.0195 0.0400	2.40 1.80	8.86×10^7 1.36×10^8
140	La	40 hr	0.061	0.04 0.96	2.55 1.65	2.10×10^8 3.27×10^9
144	Pr	17.5 min	0.053	0.02 0.02 0.05	2.60 2.20 1.10	9.32×10^7 7.88×10^7 9.85×10^7
156	Eu	15.4 days	0.00013	0.60	2.00	5.27×10^6

* This is essentially Table I given in APEX 134.

** As mentioned in APEX 134, N includes a factor of 1.04 to correct the fission yield to a new standard of 6.37 percent. In Table A11-I the values given for N also correct those in Table I, APEX 134, which had several inconsistencies.

In Table A11-I, $N = YfgE$

where

$$y = \frac{\text{nuclides}}{\text{fission}} \quad g = \frac{\text{gammas}}{\text{nuclide}} \quad E = \frac{\text{Mev}}{\text{gamma}}$$

$$f = \frac{1 \text{ fission}}{192 \text{ Mev}} \times \frac{1 \text{ Mev}}{1.6 \times 10^{-6} \text{ ergs}} \times \frac{10^7 \text{ ergs}}{\text{watt-sec}} = 3.25 \times 10^{10} \text{ fissions/sec/watt}$$

Then

$$N = \left(\frac{\text{nuclides}}{\text{fission}} \right) \left(\frac{\text{gammas}}{\text{nuclide}} \right) \left(\frac{\text{Mev}}{\text{gamma}} \right) (3.25 \times 10^{10} \text{ fissions/sec/watt})$$

where

$$N = \text{Mev/sec/operating watt}$$

Fractions of saturation, * dependent only upon the time variables and half-lives of the involved nuclides, were calculated using

$$F_1(T, t) = (1 - e^{-\lambda_1 T}) e^{-\lambda_1 t}$$

$$F_2(T, t) = \frac{\lambda_1 \lambda_2}{\lambda_1 - \lambda_2} \frac{(1 - e^{-\lambda_2 T}) e^{-\lambda_2 t}}{\lambda_2} - \frac{(1 - e^{-\lambda_1 T}) e^{-\lambda_1 t}}{\lambda_1}$$

where

$$\lambda = \frac{0.693}{\text{half-life (sec)}}$$

λ_1 refers to the nuclide and λ_2 to the daughter

$F_1(T, t)$ is used for the case where the precursor nuclide has a half-life of the order of several minutes, while $F_2(T, t)$ is applicable to long half-life nuclides. T refers to time of operation, and t is the time after shutdown in seconds.

* Table A11-I is on the basis of 100 percent saturation or 5 Mev/fission decay gamma energy release of fission products.

TABLE A11-II

Fractions of Saturation after Shutdown Following 480 Hours Operation

Nuclide	Hours after Shutdown		
	2 hrs	4 hrs	6 hrs
Rb ⁸⁸	0.68	0.41	0.25
Sr ⁹¹	0.87	0.75	0.65
I ¹³²	0.98	0.97	0.96
I ¹³³	0.97	0.93	0.87
I ¹³⁵	0.81	0.66	0.54
La ¹⁴⁰	0.61	0.62	0.62
Pr ¹⁴⁴	0.05	0.05	0.05
Eu ¹⁵⁶	0.58	0.58	0.58

Values of N (Mev/sec/operating watt) at 100 percent saturation (Table A11-I) are multiplied by the appropriate fractions of saturation from Table A11-II and summed for each energy group in Table A11-III.

TABLE A11-III

U²³⁵ Decay Gamma Source Following 480 hours Operation

Energy Group (Mev)	Hours after Shutdown (Mev/sec/watt)		
	2 hrs	4 hrs	6 hrs
1.1-1.4	1.6×10^9	1.5×10^9	1.4×10^9
1.65	2.0×10^9	2.0×10^9	2.0×10^9
1.8-2.2	4.6×10^8	3.3×10^8	2.5×10^8
2.4-2.6	2.1×10^8	1.9×10^8	1.8×10^8
2.8	2.0×10^7	1.2×10^7	7.3×10^6

These values are plotted in Figure A11-1.

ENERGY GROUP DISTRIBUTION OF THE GAMMA DECAY RATES FOLLOWING 480 HOURS REACTOR OPERATION

References

Decay of Fission Product Gammas, Clark, NDA-27-39.

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

Table of Isotopes, Hollander, Review of Modern Physics, April 1953.

U^{235} Fission Product Decay Chains, Faller, ANL-4807.

D. RADIATION IN THE SUB-PILE ROOM FROM THE DECAY GAMMA SOURCE IN THE STEEL GRID PLATE

The grid plate is Type 347 stainless steel, 62 1/2 inches in diameter by 9 1/2 inches thick. The plate is penetrated by fuel element and reflector end sections, facility tubes, and control element assemblies. To be conservative, the large facility tube penetrations were neglected in calculating the amount of steel activated in the plate, resulting in a 65 percent steel and 35 percent water mixture.

The significant decay gamma source (1.2 Mev) from Type 347 stainless steel activated at a thermal flux of 10^{12} n/cm²/sec for 1000 hours is 1.02×10^9 Mev/sec/gm* after four hours decay time. The volume of steel in the grid plate is

$$\begin{aligned} V &= \pi (31)^2 (9.5) (0.65) = 1.86 \times 10^4 \text{ cu in.} \\ &= 3.05 \times 10^5 \text{ cc} \end{aligned}$$

From IBM 650 calculations, the average thermal flux in the grid plate is 3.0×10^{13} n/cm²/sec. Correcting the decay gamma source for the higher flux in the grid plate gives a source of

$$\begin{aligned} &(1.02 \times 10^9) \frac{3 \times 10^{13}}{10^{12}} \\ &= 3.06 \times 10^{10} \text{ Mev/sec/gm} \end{aligned}$$

Multiplying by the steel density (taken as 7.6 gm/cc) gives 2.3×10^{11} Mev/cc/sec. The geometry is shown in Figure A11-2.

* MTR Handbook, ORNL 963, Shielding Section.

GEOMETRY USED IN DETERMINING RADIATION
FROM THE GRID PLATE

Fig. All-2

Radiation opposite the end of a cylindrical source* is given by

$$\bar{\Phi}_E = \frac{Q_v E B}{2\mu_s} \left[F_1(b_1) - F_1(b_3) + \frac{F_1(b_3 \sec \phi)}{\sec \phi} - \frac{F_1(b_1 \sec \phi)}{\sec \phi} \right] \quad (1)$$

where

$$\bar{\Phi}_E = \text{Mev/cm}^2/\text{sec}$$

$$Q_v E = \text{source (Mev/cc/sec)}$$

$$\mu_s = \text{self-absorption coefficient for the source (cm}^{-1}\text{)}$$

$$b_1 = \text{the absorption coefficient multiplied by the thickness in cm for each attenuating material } (\mu_1 x_1 + \mu_2 x_2 + \dots)$$

$$b_3 = b_1 + \mu_s h \text{ (h = cylinder height in cm)}$$

$$F_1(b) = b \int_b^{\infty} \frac{e^{-y}}{y^2} dy$$

The effective 1.2 Mev gamma absorption coefficients are the following.

- 1) Region between grid plate and bottom head (0.89 water, 0.0375 steel, 0.028 aluminum)

$$\begin{aligned} \text{Effective } \mu &= (\mu_{\text{water}}) (\text{V. F. water}) + (\mu_{\text{steel}}) (\text{V. F. steel}) \\ &\quad + (\mu_{\text{aluminum}}) (\text{V. F. aluminum}) \\ &= (0.064) (0.89) + (0.42) (0.0375) + (0.15) (0.028) \\ &= 0.0773 \text{ cm}^{-1} \end{aligned}$$

* See WAPD, TN-508.

2) Bottom head (0.865 steel, 0.0725 water, 0.0196 aluminum)

$$\text{Effective } \mu = 0.3716 \text{ cm}^{-1}$$

3) Grid plate (0.65 steel, 0.35 water)

$$\mu_s = 0.296 \text{ cm}^{-1}$$

For the large $\mu_s h$ in the grid plate, the $F_1(b_3)$ term is much smaller than $F_1(b_1)$ in equation (1) and may be neglected. Equation (1) may be rewritten

$$\Phi_E = \frac{Q_v E B}{2 \mu_s} \left[b_1 \int_{b_1}^{b_1 \sec \phi} \frac{e^{-y}}{y^2} dy \right] \quad (2)$$

Let $z = y - b_1$

Then

$$y = z + b_1$$

$$y^2 = (z + b_1)^2$$

$$dy = dz$$

The b_1 term becomes

$$b_1 \int_0^{b_1(\sec \phi - 1)} \frac{e^{-(z + b_1)}}{(z + b_1)^2} dz \quad (3)$$

Since $b_1 = 27.6$ and $\sec \phi = 1.04$, then z varies from 0 to 1.10. Thus in expression (3) z can be neglected in the denominator without appreciable error.

Therefore

$$b_1 \int_0^{b_1(\sec \phi - 1)} \frac{e^{-(z + b_1)}}{(z + b_1)^2} dz = \frac{e^{-b_1}}{b_1} \int_0^{1.10} e^{-z} dz$$

$$= \frac{e^{-b_1}}{b_1} (1 - e^{-1.1}) = 0.667 \frac{e^{-b_1}}{b_1}$$

Substituting into equation (2) gives

$$\frac{\bar{\phi}}{E} = \frac{Q_v E}{2 \mu_s} B (1 - e^{-b_1(\sec \phi - 1)}) \frac{e^{-b_1}}{b_1}$$

$$= \frac{2.3 \times 10^{11}}{2(0.296)} (120) (0.667) \frac{(1.03 \times 10^{-12})}{27.6}$$

$$= 1.16 \text{ Mev/cm}^2/\text{sec}$$

$$\frac{1.16}{570} \cong 0.002 \text{ mr/hr}$$

This is about 1/3000 tolerance, which indicates radiation in the sub-pile room from the grid plate is negligible.

E. SHIELDING FOR AN AIR-COOLED FACILITY TUBE BELOW THE CORE

1. Gamma Attenuation

An experiment was performed at the RMF in the MTR canal using an air-filled tube (4 1/2-inch diameter) surrounded by water. One end of the tube was positioned above a configuration of highly active MTR fuel elements which had decayed for about 36 hours. Steel and concrete shielding plugs were placed in various locations in the tube, and the radiation distribution above the water was measured. The following calculation techniques compare well with the experimental data obtained.

Assume that 12 elements on the periphery of the tube act as three elements in the center of the tube and that the experiment in the tube is heavily loaded with U²³⁵ equal to 10 ETR elements. The total source at the center is thus 13 ETR elements.

Then

$$\frac{13 \text{ elements}}{65 \text{ elements in core}} = \frac{1}{5} \text{ total core source}$$

$$\frac{1}{5} (175 \text{ MW}) = 35 \text{ MW power during operation}$$

Therefore, the 2.5 Mev gamma source four hours after shutdown following 480 hours operation* is

$$(1.9 \times 10^8 \text{ Mev/sec/watt}) (3.5 \times 10^7 \text{ watts}) = 6.65 \times 10^{15} \text{ Mev/sec}$$

Below the core the pipe has a maximum diameter of about six inches.

* See Appendix XI-C.

The cross sectional area of the facility tube is

$$\pi r^2 = \pi (7.62)^2 = 182 \text{ cm}^2$$

The source at the inlet is then

$$\frac{6.65 \times 10^{15}}{182} = 3.65 \times 10^{13} \text{ Mev/cm}^2/\text{sec}$$

This source is considered to consist of four equal contributions in the fuel elements.

In the experiment the elements are considered to have an active section of 36 inches and are divided into four equal plane sources. Each source is located at the center of a 9-inch section. The elements are considered 20 percent steel, 40 percent hydrogenous material, or graphite, and 40 percent air.

The effective μ for 2.5 Mev gammas is

$$\text{Steel} = \mu V.F. = (0.29) (0.20) = 0.058 \text{ cm}^{-1}$$

$$\text{"Water"} = \mu V.F. = (0.043) (0.40) = 0.017 \text{ cm}^{-1}$$

$$\text{Effective } \mu = 0.075 \text{ cm}^{-1}$$

Each source is attenuated through the material below.

$$\text{For point source 1} \quad \mu x = (0.075) (11.4) = 0.86$$

$$\begin{array}{l} \text{For 16 inches of steel} \\ \text{in helix in the tube} \end{array} \quad \mu x = (0.29) (40.6) = 11.8$$

$$\text{Total } \mu x = 12.7$$

The geometrical attenuation factor is $\frac{r^2}{L^2}$, where $L = 10$ feet.

It is assumed that 1/2 the source radiation proceeds downward. Then the 2.5 Mev gamma radiation from the tube for source 1 is

$$\Phi_E = \frac{1}{2} \left[\frac{3.65 \times 10^{13}}{4} \cdot \frac{(3)^2}{(120)^2} \right] B F_O (12.7)$$

$$= (2.85 \times 10^9) (11) (2.2 \times 10^{-7})$$

$$= 6.9 \times 10^3 \text{ Mev/cm}^2/\text{sec} \text{ or } 10 \text{ mr/hr}$$

The other plane source radiations were similarly determined, giving a total 2.5 Mev contribution of 12 mr/hr. The other energy groups in the 1.65 Mev to 2.8 Mev range produce a total of 38 mr/hr directly below the tube. About 12 inches to the sides of the tube the radiation is reduced by a factor of about 100, giving 0.4 mr/hr which is 1/20 tolerance.

2. Neutron Attenuation

a. Thermal neutrons

If a helix is used, the first turns below the core will contain a layer of boron. The boron depresses the thermal flux in the steel and thereby prevents high activation and accompanying high gamma radiation from the steel when the helix is removed from the tube. The boron also prevents excessive activation of the facility tube and prevents radiation damage to seals in the bottom head.

The boron thickness required in the helix is determined as follows.

The corrected $\sum_a$ for boron is 72.9 cm^{-1} .

The thermal flux down the tube is about $10^{14} \text{ n/cm}^2/\text{sec}$.

A 2mm layer of boron reduces the flux by a factor of

$$e^{-(72.9)(0.2)} = 4.65 \times 10^{-7}$$

which reduces the flux to

$$(4.65 \times 10^{-7})(10^{14}) = 4.65 \times 10^7 \text{ n/cm}^2/\text{sec}$$

This value is low enough to prevent activation problems in the helix, seals, etc.

b. Fast Neutrons

Fast neutrons are scattered out of the tube by the steel in the helix. The scattering cross section for steel is about 0.254 cm^{-1} . Then for 16 inches of steel

$$\sum_s x = (0.254)(40.6) = 10.3$$

Streaming down the tube is $5 \times 10^{14} \text{ n/cm}^2/\text{sec}$.

The geometrical attenuation factor = $\frac{r^2}{2L}$, and the flux after attenuation is

given by

$$\left[(5 \times 10^{14}) \frac{(3)^2}{2(120)^2} \right] e^{-10.3} = 5 \times 10^6 \text{ n/cm}^2/\text{sec}$$

This flux is low enough so that when thermalized and captured in the sub-pile areas no serious activation of materials will result.

The addition of significantly more steel to further attenuate the neutrons does not appear possible in the limited area available below the core.

F. SHIELDING FOR INSULATING AIR ANNULI AROUND FACILITY TUBES BELOW THE CORE

1. Geometrical Attenuation

A typical facility tube in the lower regions of the pressure vessel has an OD of four inches and a casing ID of 4 1/4 inches separated by an insulating air annulus.

The distance between the reactor core and the bottom of the bottom head is 10 feet. The geometrical attenuation* is given by

$$\frac{1}{\pi L^2} \left[(2 R^2 - r^2) \cos^{-1} \frac{r}{R} - r \sqrt{R^2 - r^2} \right]$$

where

$$r = 2 \text{ inches}$$

$$R = 2.125 \text{ inches}$$

$$= \frac{1}{\pi (120)^2} \left[(9 - 4) (\cos^{-1} 0.94) - 2 \sqrt{0.52} \right]$$

$$= (0.221 \times 10^{-4}) (0.3)$$

$$= 6.63 \times 10^{-6}$$

2. Gamma Radiation in the Sub-pile Room Four Hours after Shutdown

From Appendix XI-C the 2.5 Mev gamma source for the whole core is

3.3×10^{16} Mev/sec. The total annuli cross sectional area is 11.24 sq. in.

* TID 7004, Reactor Shielding Design Manual, T. Rockwell.

The active section of the core is 864 sq. in. The fraction of the source streaming down the annuli is $\frac{11.24}{864}$ or 0.013. This is multiplied by 2 to account for heavy fuel loadings in the experiments. The source is divided by 2 to approximate the radiation streaming down the annuli. Then radiation streaming down the annuli is

$$\frac{3.3 \times 10^{16}}{2} (0.026) = 4.3 \times 10^{14} \text{ Mev/sec}$$

Converting this to a plane source

$$\begin{aligned} S_A &= \frac{4.3 \times 10^{14} \text{ Mev/sec}}{72.2 \text{ cm}^2} \\ &= 5.95 \times 10^{12} \text{ Mev/cm}^2/\text{sec.} \end{aligned}$$

Multiplying by the geometrical attenuation factor gives

$$S_A = (5.95 \times 10^{12}) (6.63 \times 10^{-6}) = 3.94 \times 10^7 \text{ Mev/cm}^2/\text{sec.}$$

Steel washers are spaced throughout the central sections of these annuli between the core and bottom head and fill the gap between the outer casing and the facility tube with a total of 12 inches of steel. (See Figure 19-1.)

The attenuation in the steel for 2.5 Mev gammas is determined below.

$$\mu x = (0.29) (30.5) = (1.12 \times 10^{-4})$$

$$B F_0 (8.85) = 7.5 (1.5 \times 10^{-5}) = (1.12 \times 10^{-4})$$

Therefore, radiation emerging from the annuli is

$$\begin{aligned}\overline{\Phi}_E &= (3.94 \times 10^7) (1.12 \times 10^{-4}) \\ &= 4.41 \times 10^3 \text{ Mev/cm}^2/\text{sec} \text{ or } 6.6 \text{ mr/hr}\end{aligned}$$

The radiation levels for the other energy groups were similarly determined and are summarized below.

<u>Gamma Energy (Mev)</u>	<u>Radiation Level (mr/hr)</u>
2.8	0.5
2.5	5.6
2.1	6.9
1.65	29.8

The total radiation directly below the annuli is 27 mr/hr. The annuli are obstructed by a steel washer at the protection plate. This washer reduces the radiation below the annuli by at least a factor of 10^* , or to 0.6 tolerance. The radiation is further reduced to 1/10 tolerance several inches to the side of an annulus where people will normally be standing when in the sub-pile room.

3. Radiation in the Control Rod Access Room

a. Core Gamma Radiation during Operation

During reactor operation the significant gamma radiation has an energy of 7 Mev and a source strength of 2.4×10^{19} Mev/sec. Using the same

* It should be noted that the annuli are offset at the protection plate.

method as in F-2, the radiation down the annuli is

$$S = \frac{2.4 \times 10^{19}}{2} (0.026)$$
$$= 3.1 \times 10^{17} \text{ Mev/sec.}$$

Converting this to a plane source

$$S_A = \frac{3.1 \times 10^{17}}{72.2} = 4.3 \times 10^{15} \text{ Mev/cm}^2/\text{sec}$$

Multiplying by the geometrical attenuation factor gives

$$\frac{\Phi}{E} = (4.3 \times 10^{15}) (6.63 \times 10^{-6})$$
$$= 2.9 \times 10^{10} \text{ Mev/cm}^2/\text{sec.}$$

Attenuation in the steel washers and the roof of the room is determined as follows

Steel washers

$$\mu x = (0.225) (30.5) = 6.87$$

Access room roof

$$\text{Concrete } \mu x = (0.1) (30.5) = 3.05$$

$$\text{Steel } \mu x = (0.225) (5.08) = 1.14$$

$$\text{Total } x = 11.1$$

Then the total material attenuation is

$$B F_o (11.1) = 6 (1.2 \times 10^{-6})$$
$$= 7.2 \times 10^{-6}$$

Therefore, the radiation level immediately below the access room roof is

$$\Phi_E = (2.9 \times 10^{10}) (7.2 \times 10^{-6})$$

$$= 2.1 \times 10^5 \text{ Mev/cm}^2/\text{sec} \text{ or } 233 \text{ mr/hr}$$

Each annulus is obstructed by a steel washer at the protection plate. This washer reduces the radiation level below the access room roof by a factor of 10, or to 23 mr/hr.

b. Fast neutron radiation

The fast neutron source in the region of the annuli is about $10^{15} \text{ n/cm}^2/\text{sec}$. Then the streaming down the annuli is conservatively taken as half this value or $5 \times 10^{14} \text{ n/cm}^2/\text{sec}$. Multiplying by the geometrical attenuation factor reduces the current to

$$\begin{aligned} \Phi_f &= (5 \times 10^{14}) (6.63 \times 10^{-6}) \\ &= 3.3 \times 10^9 \text{ n/cm}^2/\text{sec}. \end{aligned}$$

The fast neutron scattering cross section for steel is 0.254 cm^{-1} . The attenuation by 12 inches of steel is then

$$e^{-7.75} = 4.3 \times 10^{-4}.$$

Using a mean free path (λ) of 8.2 cm in concrete gives an attenuation in the 12-inch thick roof of

$$\begin{aligned} e^{-x/\lambda} &= e^{-\frac{30.5}{8.2}} \\ &= 0.024. \end{aligned}$$

Therefore, the neutron radiation level in the access room is

$$\bar{\Phi}_f = (3.3 \times 10^9) (4.3 \times 10^{-4}) (0.024)$$

$$= 3.4 \times 10^4 \text{ n/cm}^2/\text{sec}$$

As mentioned previously, each annulus is obstructed by a washer at the protection plate. This washer reduces the fast flux by a factor of approximately 10, which reduces the radiation in the access room to $3.4 \times 10^3 \text{ n/cm}^2/\text{sec}$.

G. RADIATION IN THE CONTROL ROD ACCESS ROOM FROM CORE GAMMAS DURING REACTOR OPERATION

From Appendix X-A, the gamma source is 2.4×10^{19} Mev/sec. As in the treatment of radiation in the sub-pile room, it is assumed that 58 percent of the source penetrates the water and facility tube regions of the grid plate. In computing attenuations it is assumed the water and metal mixture below the grid plate extends to the bottom of the active section of the core. The core is divided into four equal point sources, each located at the midpoint of a 9-inch active section. The radiation from each point source is attenuated by self-absorption in the core and absorption in the other attenuating material along a straight-line path to the control rod access room.

The attenuation factors for the various materials are shown below. The gamma absorption coefficients are obtained from Appendix XI-B.

<u>Attenuating Material</u>	<u>μ</u>	<u>x</u>	<u>μx</u>
Core			
Source 1	0.053	11.4	0.60
Source 2	0.053	34.2	1.81
Source 3	0.053	57	3.02
"Water" below core	0.0335	274.5	9.2
Bottom head	0.198	21.6	4.28
Bottom head to access room roof	0.01	137	1.37
Roof of access room*			
Concrete	0.10	30.5	3.05
Steel	0.225	5.08	1.14

* The roof of the access room is actually 11 1/2 inches of magnetite concrete plus 2 3/8 inches of steel, but is tapered slightly to allow drainage. The effective thickness of concrete and steel used are approximately correct in the over-all analysis.

The gamma source at point 1 is

$$S = \frac{2.4 \times 10^{19}}{4} = 6 \times 10^{18} \text{ Mev/sec}$$

The amount of radiation going through the grid plate is

$$S = (6 \times 10^{18}) (0.58) = 3.5 \times 10^{18} \text{ Mev/sec}$$

Using the point source formula

$$\begin{aligned} \Phi_E &= \frac{Q_0 E B e^{-\mu x}}{4 \pi a^2} \\ &= \frac{3.5 \times 10^{18}}{4 \pi (480)^2} (8) e^{-19.64} \end{aligned}$$

$$= 2.9 \times 10^5 \text{ Mev/cm}^2/\text{sec} \text{ or } 32 \text{ mr/hr}$$

Proceeding similarly for the other two point sources (source 4 contribution is negligible) gives total of 43 mr/hr.

APPENDIX XII

CALCULATIONS FOR THE CANAL SHIELDING

A. DEPTH OF WATER REQUIRED IN THE CANAL

Consider the contribution from 1.65 Mev gammas after attenuation by 15 1/2 feet of water above the active section of a fuel element. During operation at 175 MW there are about 65 fuel element sections in the reactor core (49 fuel elements and 16 control rods containing fuel element sections). From Appendix XI-C, Figure A11-1, radiation from 1.65 Mev decay gammas 2 1/2 hours after shutdown is 2.05×10^9 Mev/sec per operating watt. For operation at 175 MW, the gamma source is thus 3.5×10^{17} Mev/sec.

Consider the 65 fuel element sections in the canal. The 36-inch high active section was divided into four equal point sources, each located at the midpoint of a 9-inch section. Each point source is thus $\frac{3.5 \times 10^{17}}{4} = 0.875 \times 10^{17}$ Mev/sec.

$$\bar{\Phi}_E = \frac{Q_0 E B e^{-\mu_1 x_1} e^{-\mu_2 x_2}}{4 \pi a^2}$$

where

$$\bar{\Phi}_E = \text{Mev/cm}^2/\text{sec}$$

$$Q_0 E = \text{Mev/sec}$$

a = distance from source to detector (cm)

x = thickness of attenuating material (cm)

μ = absorption coefficient of the material (cm^{-1})

B = buildup factor

For point source 1:

$$a = x_1 \text{ water} + x_2 \text{ fuel element}$$

$$a = 473 \text{ cm} + 11.4 \text{ cm} = 484.4 \text{ cm}$$

$$\mu_1 = \text{absorption coefficient for water} = 0.055 \text{ cm}^{-1}$$

$$\mu_2 = \text{absorption coefficient for fuel elements} = 0.085 \text{ cm}^{-1}$$

B = effective buildup factor, taken as the buildup factor for water
at the total μx . B = 70

Then

$$\bar{\Phi}_E = \frac{(0.875 \times 10^{17}) (70) (e^{-(0.055)(473)} e^{-(0.085)(11.4)})}{4\pi (473 + 11.4)^2}$$

$$= 4.39 \text{ Mev/cm}^2/\text{sec or } 0.007 \text{ mr/hr}$$

Proceeding similarly for the other fuel element sections, a contribution of 0.008 mr/hr was obtained for the 1.65 Mev gammas after attenuation by 15 1/2 feet of water in the canal.

The contributions of the other energy groups were obtained, giving a total of 0.2 mr/hr for the gamma radiation.

B. RADIATION OUTSIDE THE SEVEN-FOOT PORTLAND CONCRETE CANAL WALL

The following calculation illustrates the method used to determine radiation levels outside the six- and seven-foot thick canal walls.

Consider the contribution from 1.65 Mev gammas after attenuation by seven feet of concrete adjacent to the 65 hot fuel elements in the canal storage racks. From Appendix XI-C, Figure A11-1, radiation from 1.65 Mev decay gammas 2 1/2 hours after shutdown is 2.0×10^9 Mev/sec per operating watt. For operation at 175 MW, the gamma source is thus 3.5×10^{17} Mev/sec.

$$\bar{\Phi}_E = \frac{B Q_v E c^2 F(\phi, b_2)}{2(a + z)}$$

where

$$\bar{\Phi}_E = \text{Mev/cm}^2/\text{sec}$$

$$Q_v E = \text{Mev/cc/sec}$$

a = distance from source to detector (cm)

B = buildup factor

C = radius of disk or cylinder (cm)

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

z = effective self-absorption distance

ϕ = angle in degrees subtended by length of cylinder, as observed from
a point at distance $a + z$ from the cylinder

Then, for 1.65 Mev gammas

$$Q_v E = \frac{\text{gamma radiation intensity (Mev/sec)}}{\text{volume of 65 fuel elements (cc)}} = \frac{3.5 \times 10^{17}}{3.46 \times 10^5}$$

$$= 1.011 \times 10^{12} \text{ Mev/cc/sec}$$

$$C = \sqrt{\frac{1}{\pi} \cdot \text{cross sectional area of 65 fuel elements (cm}^2\text{)}}$$

$$= \sqrt{\frac{3.770 \text{ cm}^2}{3.14}} = 34.63 \text{ cm, } c^2 = 1198 \text{ cm}^2$$

$$b_1 = \mu x (\text{concrete}) = (0.116 \text{ cm}^{-1}) (213.36 \text{ cm}) = 24.65$$

B = effective buildup factor, taken as the buildup factor for water
at the total μx . B = 57

$$\phi = 2 \tan^{-1} \frac{1.5}{7.0} = 24.16^\circ$$

$$\mu_s = \mu_{\text{H}_2\text{O}} (V.F. \text{H}_2\text{O}) + \mu_{\text{Al}} (V.F. \text{Al})$$

$$= (0.055) (0.6) + (0.13) (0.4) = 0.085 \text{ cm}^{-1}$$

$$b_2 = b_1 + \mu_s z = 24.65 + 1.45 = 26.10$$

$$F(\phi, b_2) = 1.04 \times 10^{-12}$$

$$2(a + z) = 2(213.36 + 17.04) = 460.8 \text{ cm}$$

Substituting gives

$$\overline{\Phi}_E = \frac{57 (1.01 \times 10^{12} \text{ Mev/cc/sec}) (1.20 \times 10^3 \text{ cm}^2) (1.04 \times 10^{12})}{461 \text{ cm}}$$

$$= 156 \text{ Mev/cm}^2/\text{sec} \text{ or } 0.26 \text{ mr/hr}$$

The contributions of the other energy groups were similarly obtained, giving a total of 2.5 mr/hr for the gamma radiation.

C. ATTENUATION OF RADIATION IN THE REGION BETWEEN THE PRESSURE VESSEL AND CANAL

1. Primary Gammas

The 7 Mev core gamma source consists of U^{235} fission gammas, capture gammas from aluminum in the fuel elements and reflector, and capture gammas from metals in the experiments. This source is 2.4×10^{19} Mev/sec. (See Appendix X-A.)

The core is divided into four equal point sources, the attenuation for each computed and the sum taken. Using G-E Drawing M-596, lines are drawn from each point source to the top of the canal. With 2 1/2 feet of magnetite concrete following the 2 1/4 inches of steel in the pressure vessel wall, the amount of core material, water, steel, and concrete penetrated by each path is measured and the attenuation computed. Attenuations along an average path are given in Table A12-I.

TABLE A12-I

Attenuation of 7 Mev Core Gammas

<u>Material</u>	<u>x (cm)</u>	<u>μ (cm⁻¹)</u>	<u>μx</u>
Water	473	0.026	12.30
Magnetite Concrete	137	0.10	13.7
Steel	5.72	0.225	1.29
Core			
Source 1-	13.2	0.06	0.79
Source 2	39.0	0.06	2.34
Source 3	63.8	0.06	3.83
Source 4	Negligible	----	-----

To obtain the radiation above the canal

$$\bar{\Phi}_E = \frac{Q_0 E B e^{-\mu x}}{4 \pi a^2}$$

where

$$\bar{\Phi}_E = \text{Mev/cm}^2/\text{sec}$$

$$Q_0 E = \frac{2.4 \times 10^{19}}{4} \text{ Mev/sec, for each point source (from Appendix X-A)}$$

a = distance from source to detector (cm)

$e^{-\mu x}$ = total attenuation (from Table A12-I)

B = effective buildup factor, taken as the buildup factor for water at the total μx (from Table A12-I)

For point source 1:

$$\begin{aligned} \bar{\Phi}_E &= \frac{(0.6 \times 10^{19}) (9.9) (6.06 \times 10^{-13})}{4 \pi (629)^2} \\ &= 7.25 \text{ Mev/cm}^2/\text{sec} \text{ or } 0.008 \text{ mr/hr} \end{aligned}$$

The contributions of the other point sources were similarly obtained, giving a total of 0.01 mr/hr for the gamma radiation above the canal during reactor operation.

2. Secondary Gammas

The fast neutrons emerging from the top of the active section of the core are assumed to be thermalized and captured at a distance above the core equal to

their age displacement, τ/λ

where

τ = the age of neutrons in the mixture

λ = the relaxation length for fast neutrons.

A consideration of the volume fractions and neutron absorption cross section of metal and water above the reactor core shows that 64 percent of the captures occur in the metal, giving one 7 Mev gamma per capture. (The remaining captures in water give 2 Mev gammas which are attenuated much more rapidly.) The steel-to-water ratio in the first three feet above the core is about 0.18 (Appendix X-C and X-D). This corresponds to an age of about 36 cm^2 .

The λ for the mixture is about 9 cm

$$\text{Therefore } \frac{\tau}{\lambda} = \frac{36 \text{ cm}^2}{9 \text{ cm}} = 4 \text{ cm}$$

Hence, the capture gamma source originates about two inches above the active section of the reactor core. From IBM 650 calculations, the net fast current in the upward direction at the top of the reactor core is given by

$$\begin{aligned} D \nabla \Phi &= \frac{1.307 (15.8 \times 10^{14} - 7.4 \times 10^{14})}{10} \\ &= 1.307 (8.4 \times 10^{13}) \\ &= 1.1 \times 10^{14} \text{ n/cm}^2/\text{sec} \end{aligned}$$

At two inches above the active section of the core, the 7 Mev capture gamma source is

$$(1.1 \times 10^{14}) (7) (0.64) = 4.9 \times 10^{14} \text{ Mev/cm}^2/\text{sec}$$

The horizontal cross sectional area of the core is about $900 \text{ in.}^2 = 5,780 \text{ cm}^2$.

Converting the area source to a point source gives

$$(4.9 \times 10^{14}) (0.578 \times 10^4) = 2.84 \times 10^{18} \text{ Mev/sec}$$

Table A12-II gives the attenuations for radiation above the water in the canal originating from capture gammas above the reactor core.

TABLE A12-II

Attenuation of 7 Mev Capture Gammas Above
the Reactor Core

<u>Material</u>	<u>x (cm)</u>	<u>μ (cm⁻¹)</u>	<u>μx</u>
Water	473	0.026	12.3
Magnetite Concrete	145	0.10	14.5
Steel	<u>5.72</u>	0.225	<u>1.3</u>

$$a = 624 \text{ cm}$$

$$\text{total } \mu x = 28.1$$

To obtain the radiation above the canal

$$\bar{\Phi}_E = \frac{Q_0 E B e^{-\mu x}}{4 \pi a^2}$$

where

$$\bar{\Phi}_E = \text{Mev/cm}^2/\text{sec}$$

$$Q_0 E = 2.84 \times 10^{18} \text{ Mev/sec}$$

$$a^2 = 3.89 \times 10^5 \text{ cm}^2$$

$$e^{-\mu x} = e^{-28.1} = 6.06 \times 10^{-13}$$

$$B = 9.9$$

Substituting gives

$$\begin{aligned} \frac{\Phi}{E} &= \frac{(2.84 \times 10^{18}) (9.9) (6.06 \times 10^{-13})}{12.56 (3.89 \times 10^5)} \\ &= 3.49 \text{ Mev/cm}^2/\text{sec} \text{ or } 0.004 \text{ mr/hr} \end{aligned}$$

3. Neutrons

The fast neutron current of 1.1×10^{14} n/cm²/sec is to be attenuated by 473 cm of water and 137 cm of magnetite concrete. Using relaxation lengths of about 18 cm (water not preceded by steel) and 8.2 cm respectively, the attenuations are

$$\text{For water} \quad e^{-\frac{473}{18}} = 4 \times 10^{-12}$$

$$\text{For magnetite concrete} \quad e^{-\frac{137}{8.2}} = 5.5 \times 10^{-8}$$

This gives a total attenuation of 2.2×10^{-19} , which reduces the neutron flux to 0.5×10^{-6} tolerance.

4. Active Water in the Pressure Vessel

In Appendix X, Calculations for the Top Area Shielding, it is shown that 28 inches of magnetite concrete are required outside the pressure vessel to attenuate N^{16} gammas from the active water in the vessel to $\frac{1}{10}$ tolerance.

5. Conclusion

- Two and one-half feet of magnetite concrete between the pressure vessel and canal water will reduce radiation to 1/10 tolerance in accessible regions around the top of the canal.

D. ACTIVITY OF THE CANAL WATER NEAR PRESSURE VESSEL

The water activity is primarily due to N^{16} from the reaction $O^{16} (n,p) N^{16}$.

In Appendix XI-I, Radiation from Active Water Loops, it is shown that

$6 \times 10^8 N^{16}$ nuclei per cc per second are formed in the water of the reactor

core and that these decay by means of 6.5 Mev gamma rays. In Section II,

Thermal Flux in the ETR, it is shown that the average above-thermal flux in

the core is about 1.5×10^{15} nv. An effective macroscopic activation cross

section for water can be defined as

$$\sum_a = \frac{6 \times 10^8}{1.5 \times 10^{15}} = 4 \times 10^{-7} \text{ cm}^{-1}$$

At saturation the activity of the water is given by

$$Q_v E = nv \sum_a \text{ Mev/cc/sec}$$

where

nv = activating above-thermal flux

$$\sum_a = 4 \times 10^{-7} \text{ cm}^2/\text{cc H}_2\text{O}$$

$E = 6.5 \text{ Mev}$ (average N^{16} gamma)

From one-dimensional IBM 650 calculations, the average above-thermal neutron current into the concrete is $1.7 \times 10^{10} \text{ n/cm}^2/\text{sec}$. This flux is attenuated by about 12 inches of magnetite concrete in the region between the pressure vessel and canal water. Using 8.2 cm as the relaxation length for fast neutrons, the flux is reduced to

$$(1.7 \times 10^{10}) e^{-\frac{30.5}{8.2}} = 4.1 \times 10^8 \text{ n/cm}^2/\text{sec.}$$

Thus,

$$\begin{aligned} Q_v E &= (4.1 \times 10^8) (4 \times 10^{-7}) (6.5) \\ &= 1.1 \times 10^3 \text{ Mev/cc/sec.} \end{aligned}$$

Consider a cylinder of active water, 8 feet in diameter and 5 feet high (region near thermal shields in the canal). This represents a volume of 7.1×10^6 cc and a source of $(7.1 \times 10^6) (1100) = 8 \times 10^9$ Mev/sec. Treating this as a point source

$$\Phi_E = \frac{Q_o E B e^{-\mu x}}{4\pi a^2}$$

where

$$\Phi_E = \text{Mev/cm}^2/\text{sec}$$

$Q_o E$ = source strength (Mev/sec)

a = distance from source to detector (cm)

μ = absorption coefficient for the attenuating material (cm^{-1})

x = thickness of material (cm)

B = buildup factor

The radiation above the canal (after attenuation by 15 feet of water) is

$$\Phi_E = \frac{(8 \times 10^9) (6) (e^{-12.3})}{4\pi (457)^2}$$

$$= 0.08 \text{ Mev/cm}^2/\text{sec or } 1 \times 10^{-4} \text{ mr/hr}$$

This can be neglected.

The radiation to the sides of the canal is attenuated by 6 feet of portland concrete.

Using $\mu = 0.06 \text{ cm}^{-1}$

$$\begin{aligned}\overline{\Phi}_E &= \frac{(8 \times 10^9) (5.5) e^{-11}}{4\pi (183)^2} \\ &= 1.7 \text{ Mev/cm}^2/\text{sec} \text{ or } 0.002 \text{ mr/hr}\end{aligned}$$

This can be neglected. (10^{-4} tolerance)

This treatment of canal water activity shows that radiation in accessible areas is not high enough to necessitate forced circulation of the canal water.

E. RADIATION IN SUB-PILE ROOM FROM SINGLE FUEL ELEMENT ON GRATING IN CANAL PIT

In the most severe case, if an element is dropped on the grating, there will be four feet of water and 8 1/2 inches of steel between the element and a person in the sub-pile room. (See diagram below.)

Using a cylindrical source treatment, the radiation directly below the element in the sub-pile room is given by

$$\bar{\Phi}_E = \frac{BQ_v EC^2 F(\phi, b_2)}{2(a+z)}$$

For 1.65 Mev gammas, the attenuating materials are

<u>Material</u>	<u>x(cm)</u>	<u>μ (cm⁻¹)</u>	<u>μx</u>
Water	121.9	0.055	6.71
Steel	21.6	0.35	7.65
Fuel Element			$\mu_s z = 0.22$
			$b_2 = 14.49$

and $B = 24$

For one fuel element (3 x 3 inches)

$$\pi C^2 = (9 \text{ in.}^2) (2.54 \text{ cm/in.})^2$$

$$C^2 = 18.42 \text{ cm}^2$$

From Appendix XII-D, $Q_v E = 1.01 \times 10^{12} \text{ Mev/cc/sec}$

Substituting gives

$$\begin{aligned} \frac{\Phi}{E} &= \frac{(24) (1.01 \times 10^{12}) (18.42) (1.4 \times 10^{-7})}{474} \\ &= 1.32 \times 10^5 \text{ Mev/cm}^2/\text{sec} \end{aligned}$$

$$\frac{1.32 \times 10^5}{600} = 220 \text{ mr/hr}$$

Proceeding similarly for the other energy groups gives the following contributions:

<u>Gamma Energy (Mev)</u>	<u>Dose Rate (mr/hr)</u>
1.65	220
2.1	158
2.5	173
2.8	28
Total = 579 mr/hr	

APPENDIX XIII

SAMPLE CALCULATION FOR DEPTH OF WATER REQUIRED DURING REFUELING

Consider the contribution from 1.1 to 1.4 Mev gammas after attenuation by six feet of water above the active section of a fuel element.

If we assume that the 36-inch active section of a fuel element is divided into four equal point sources, each located at the midpoint of a 3 by 3 by 9-inch section, and that there are 57 fuel elements effectively operating at a total power of 175 MW, then

$$\frac{175 \times 10^6 \text{ watts}}{57 \text{ elements}} \approx 3.1 \times 10^6 \text{ watts/element}$$

From Appendix XI-C, Figure A11-1, for two hours after shutdown following 480 hours operation radiation from 1.10 to 1.40 Mev gammas is 1.65×10^9 Mev/sec/watt. At 3.1×10^6 watts there is a gamma source of $(3.1 \times 10^6)(1.60 \times 10^9) = 5.0 \times 10^{15}$ Mev/sec/element, or 1.25×10^{15} Mev/sec for each 1/4-element section.

Then

$$\bar{\Phi} = \frac{Q_0 E B e^{-\mu_1 x_1} e^{-\mu_2 x_2}}{4 \pi a^2}$$

where

$$\bar{\Phi} = \text{Mev/cm}^2/\text{sec}$$

$$Q_0 E = \text{Mev/sec}$$

a = distance from source to detector on refueling platform (cm)

x = thickness of attenuating material (cm)

μ = absorption coefficient of the material (cm^{-1})

B = buildup factor

The refueling platform is four feet above the water level in the pressure vessel.

For Source 1:

$$a = x_1 (\text{water}) + x_2 (\text{fuel element}) + x_3 (\text{air})$$

$$a = 182.9 \text{ cm} + 11.4 \text{ cm} + 121.9 \text{ cm} = 316.2 \text{ cm}$$

$$\mu_1 = \text{absorption coefficient for water} = 0.06 \text{ cm}^{-1}$$

$$\mu_2 = \text{absorption coefficient for fuel element} = 0.092 \text{ cm}^{-1}$$

B = effective buildup factor, taken as the buildup factor for

water at the total μx . B = 26.4

$$\text{Hence, } \Phi = \frac{(1.25 \times 10^{15}) (26.4) e^{-(0.06 \times 182.9)} e^{-(0.092 \times 11.4)}}{4 (316)^2}$$

$$= 1.61 \times 10^5 \text{ Mev/cm}^2/\text{sec} \text{ or } 278 \text{ mr/hr.}$$

Proceeding similarly for the other fuel element sections, 321 mr/hr is obtained for the contribution of 1.1 to 1.4 Mev gammas after attenuation by six feet of water.

The contributions of the other energy groups were also obtained, giving a total of 1834 mr/hr at the six-foot water depth. Each element is in its highest position for about 10 seconds. During a complete refueling (65 fuel elements), this gives a total dose of 330 mr at the six-foot water depth.

APPENDIX XIV

CALCULATIONS FOR SUPPLEMENTARY SHIELDING

A. SHIELDING FOR INSULATING AIR ANNULI AROUND FACILITY TUBES ABOVE THE CORE

There is a 1/8-inch insulating air gap around each facility tube between the tube and its outside casing. Shielding is required in each annulus to attenuate radiation streaming into accessible areas above the reactor core during (1) reactor operation, and (2) after-shutdown operations (refueling the reactor and servicing facility tubes). The necessary shielding is 14 inches of steel installed in the form of washers about one-inch thick spaced throughout the central section of each air annulus. (See Figure 18-1.)

1. Radiation During Reactor Operation

a. Core Gammas

A typical facility insulating air annulus has an OD of six inches and an ID of 5 3/4 inches. (See G-E Drawing M-606.) The length of the annulus section above the reactor core is about 20 feet. Then for

$$R = 3 \text{ inches (outside radius)}$$

$$r = 2.975 \text{ inches (inside radius)}$$

The geometrical attenuation factor is given by

$$\begin{aligned} & \frac{1}{\pi L} \left[\cos^{-1} \frac{r}{R} (2R^2 - r^2) - r \sqrt{R^2 - r^2} \right] * \\ &= \frac{1}{\pi (240)^2} \left[(\cos^{-1} 0.958) (18 - 8.266) - 2.88 \sqrt{0.734} \right] \\ &= \frac{1}{\pi (5.76 \times 10^4)} \left[(0.290) (9.734) - (2.464) \right] \\ &= 2.0 \times 10^{-6} \end{aligned}$$

* TID-7004, Rockwell, T., Reactor Shielding Design Manual.

From Appendix X-C, the air annuli area is 0.018 of the active section area of the core. This area is the fraction of the core source that the annuli see. The fraction is doubled to take into account heavy fuel loadings in the facilities.

From Appendix X-A, the core gamma source during reactor operation is

$$2.4 \times 10^{19} \text{ Mev/sec of 7 Mev photons}$$

The amount of the source seen by the annuli is

$$(2.4 \times 10^{19}) (0.036) \text{ or } 8.64 \times 10^{17} \text{ Mev/sec}$$

Dividing by the total annular area gives

$$\frac{8.64 \times 10^{17} \text{ Mev/sec}}{101 \text{ cm}^2} = 8.6 \times 10^{15} \text{ Mev/cm}^2/\text{sec}$$

About half of this source proceeds up the annuli. The annuli terminate some two feet below the top cover plate, leaving two feet of water, the 4 1/2-inch steel cover plate, and about 23 inches of magnetite concrete to attenuate the radiation before it reaches areas accessible during operation.

The attenuating materials are

	$\mu \text{ (cm}^{-1}\text{)}$	$x(\text{cm})$	μx
Water	0.026	61	1.59
Steel	0.225	11.4	2.57
Concrete	0.10	58.4	5.85
Total $\mu x =$			10.0

The source at the inlet times the annuli geometrical attenuation gives

$$(4.3 \times 10^{15}) (2.0 \times 10^{-6}) = 8.5 \times 10^9 \text{ Mev/cm}^2/\text{sec at the outlet.}$$

Using a plane source treatment, the dose rate above the concrete is

$$\begin{aligned} \bar{\Phi}_E &= \frac{8.5 \times 10^9}{900} (5) (3.2 \times 10^{-6}) \\ &= 150 \text{ mr/hr} \end{aligned}$$

With 14 inches of steel in the annuli

$$\mu x_{\text{steel}} = (0.225) (35.6) = 8.0$$

Total μx is now 18.0

$$\begin{aligned} \bar{\Phi}_E &= \frac{8.5 \times 10^9}{900} (7) (8.2 \times 10^{-10}) \\ &= 0.054 \text{ mr/hr} \end{aligned}$$

This is 1/50 tolerance.

In the top shield being designed by the Kaiser Engineers, the concrete above the top cover plate will probably be initially about 12 inches. This will increase the radiation level determined above to 1/8 tolerance.

b. Secondary Gammas

Consider the 7 Mev gammas from neutrons captured in the steel tube and casing. $\sum_a \text{iron} = 0.19 \text{ cm}^{-1}$ for thermal neutrons. For a thermal source of 5×10^{14} , assuming all the thermal neutrons are captured in the

steel tube section in the core, there is a 7 Mev gamma source of 3.5×10^{15} Mev/sec. Dividing by an annulus area gives

$$\frac{3.5 \times 10^{15}}{15.7} = 2.2 \times 10^{14} \text{ Mev/cm}^2/\text{sec.}$$

The geometrical attenuation is 1.97×10^{-6} . Treating this as an area source and assuming that half the source radiation proceeds upwards, gives a 7 Mev gamma intensity emerging from an annulus of

$$\bar{\Phi}_E = \frac{(1.1 \times 10^{14}) (1.97 \times 10^{-6})}{(900)} B F_O(b) = 2.4 \times 10^5 B F_O(b) \text{ mr/hr}$$

Considering only 12 inches of steel washers attenuating the source, the total $\mu x = b$ is 16.85. Then

$$\begin{aligned} \bar{\Phi}_E &= (2.4 \times 10^5) (7) (2.7 \times 10^{-9}) \\ &= 0.0045 \text{ mr/hr} \end{aligned}$$

This is $\frac{1}{1700}$ tolerance.

If only 12 inches of concrete are in the top shield this dose is increased to 1/80 tolerance.

By assuming all of the thermal neutrons are captured at the first or second washer, the source is approximately half way up the annulus. Consequently, the geometric attenuation is less, but the neutron flux is also reduced. Thus, the value obtained above is not significantly changed.

c. Fast Neutrons

The fast neutron source in annuli regions is about 10^{15} n/cm²/sec, and the flux proceeding upward is about 5×10^{14} n/cm²/sec. The geometrical attenuation factor is 1.97×10^{-6} .

The attenuating materials are given below.

14 inches steel in annulus

$$x = 35.6 \text{ cm}$$

$$\sum_s = 0.254 \text{ cm}^{-1}$$

$$\sum_s x = 9.05$$

where $\sum_s$ is the fast scattering cross section of steel.

The steel in the annulus scatters neutrons into the water where they are rapidly attenuated.

Water

$$x = 61 \text{ cm}$$

$$\sum_s = 9 \text{ cm}^{-1}$$

$$\frac{x}{\sum_s} = 6.67$$

To be conservative scattering in the steel cover plate is neglected.

Concrete

$$\frac{x}{\sum_s} = \frac{58.5}{8.2} = 7.15$$

$$\text{Material attenuation} = e^{-22.9} = 1.1 \times 10^{-10}$$

$$\begin{aligned} \text{Neutron flux above the top shield} &= (5 \times 10^{14}) (1.97 \times 10^{-6}) (1.1 \times 10^{-10}) \\ &= 0.11 \text{ n/cm}^2/\text{sec} \end{aligned}$$

This is 1/700 tolerance.

Reducing the concrete in the top shield to 12 inches will increase this to 1/30 tolerance.

2. Radiation After Reactor Shutdown

a. Core Radiation at the Refueling Water Level (17 feet above the core)

Two hours after shutdown the 2.5 Mev core gamma source is 3.7×10^{16} Mev/sec. (See Appendix XI-C.) Using a similar treatment as before, the source seen by the annuli is

$$S = (3.7 \times 10^{16}) (0.036) = 1.33 \times 10^{15} \text{ Mev/sec.}$$

Dividing by the annuli area gives

$$S_A = \frac{1.33 \times 10^{15}}{101} = 1.32 \times 10^{13} \text{ Mev/cm}^2/\text{sec.}$$

Assuming that half the source radiation proceeds upward

$$\frac{S_A}{2} = S'_A = 6.6 \times 10^{12} \text{ Mev/cm}^2/\text{sec.}$$

For geometrical attenuation L now equals 204 inches, and the attenuation is 2.73×10^{-6} . Multiplying by the geometrical attenuation gives

$$\Phi_E = (6.6 \times 10^{12}) (2.73 \times 10^{-6}) = 1.8 \times 10^7 \text{ Mev/cm}^2/\text{sec}$$

emerging from the annuli.

With 14 inches of steel (washers) in the annulus

$$\mu x = (0.29) (35.6) = 10.3$$

Radiation after steel attenuation is

$$\begin{aligned}\bar{\Phi}_E &= \frac{1.8 \times 10^7}{680} (8.5) (3 \times 10^{-6}) \\ &= 0.7 \text{ mr/hr}\end{aligned}$$

(1) For 1.65 Mev gammas

The source is 3.5×10^{17} Mev/sec.

The source radiation proceeding up the annuli is

$$S'_A = \frac{(3.5 \times 10^{17}) (0.036)}{2 (101)} = 6.25 \times 10^{13} \text{ Mev/cm}^2/\text{sec.}$$

Multiplying by the geometrical attenuation gives

$$\bar{\Phi}_E = (6.25 \times 10^{13}) (2.73 \times 10^{-6}) = 1.7 \times 10^8 \text{ Mev/cm}^2/\text{sec}$$

Considering the 14 inches of steel, where $\mu x = (0.35) (35.6) = 12.5$

$$\begin{aligned}\bar{\Phi}_E &= \frac{1.7 \times 10^8}{600} (13) (2.8 \times 10^{-7}) \\ &= 1.0 \text{ mr/hr}\end{aligned}$$

(2) For 2.1 Mev gammas

The source is 7.9×10^{16} Mev/sec.

The source radiation proceeding up the annuli is

$$S'_A = \frac{(7.9 \times 10^{16}) (0.036)}{2 (101)} = 1.41 \times 10^{13} \text{ Mev/cm}^2/\text{sec}$$

Multiplying by the geometrical attenuation gives

$$\bar{\Phi}_E = (1.4 \times 10^{13}) (2.73 \times 10^{-6}) = 3.8 \times 10^7 \text{ Mev/cm}^2/\text{sec}$$

Considering the 14 inches of steel, where $\mu x = (0.31) (35.6) = 11.0$

$$\begin{aligned} \bar{\Phi}_E &= \frac{3.8 \times 10^7}{650} (10) (1.4 \times 10^{-6}) \\ &= 0.8 \text{ mr/hr} \end{aligned}$$

The total radiation from the annuli at the water level is therefore about 2.5 mr/hr or 1/3 tolerance.

b. Radiation Received while Servicing Facility Tubes in the Pressure Vessel

The water level is 88 inches below the pressure vessel flange. When L is 128 inches, the geometrical attenuation is 6.93×10^{-6} .

For 2.5 Mev gammas

Using the previous calculations, the radiation from annuli at the water level is

$$(0.7 \text{ mr/hr}) \left(\frac{6.93}{2.73} \right) = 1.8 \text{ mr/hr}$$

$$\text{Total} = (2.5 \text{ mr/hr}) \frac{6.93}{2.73} = 6.3 \text{ mr/hr}$$

c. Streaming Through the Annuli of Decay Gammas from Activated Steel in Tubes and Casings

The average thermal flux activating the steel pipes and casings in the active section of the core is about 5×10^{14} nv. From the MTR Handbook for

Type 304 stainless steel activated at a thermal flux of 10^{12} nv, the source four hours after shutdown is 3.8×10^8 Mev/sec/gm. Correcting this for a flux of 5×10^{14} , the source becomes

$$(3.8 \times 10^8) (500) = 1.9 \times 10^{11} \text{ Mev/sec/gm}$$

The radiation from the steel casing in the core is conservatively estimated by considering the thickness of the material to be one mean free path and neglecting self absorption. In the calculation the effective thickness is rounded off to 0.01 cm, which corresponds to about 2 1/2 relaxation lengths.

Considering the three-foot casing section in the reactor core with this effective thickness gives a volume for the casing of

$$V = \pi (R^2 - r^2) L = \pi \left[(7.62)^2 - (7.61)^2 \right] 92 = 44 \text{ cc}$$

The weight of the steel in the casing is given by

$$W = V\rho = (44) (7.8) = 340 \text{ gm}$$

The source for one annulus is therefore

$$\begin{aligned} S &= (1.9 \times 10^{11} \text{ Mev/sec/gm}) (340 \text{ gm}) \\ &= 6.45 \times 10^{13} \text{ Mev/sec} \end{aligned}$$

Considering a 1/8-inch annulus on a 3-inch radius, the annulus cross sectional area is

$$\pi \left[(7.62)^2 - (7.30)^2 \right] = 15.7 \text{ cm}^2$$

Converting to an area source gives

$$S_A = \frac{6.45 \times 10^{13}}{15.7} = 4.1 \times 10^{12} \text{ Mev/cm}^2/\text{sec}$$

The number of relaxation lengths in 14 inches of steel in the annulus is

$$b = \mu x = (0.42) (35.6) = 15.0$$

and with a geometrical attenuation factor of 6.9×10^{-6}

$$\begin{aligned} \overline{\Phi}_E &= \frac{Q_A E}{2} B F_0 (b) \\ &= \frac{(4.1 \times 10^{12}) (6.9 \times 10^{-6})}{2 (900)} (18) (1.9 \times 10^{-8}) \\ &= 0.005 \text{ mr/hr.} \end{aligned}$$

This is 1/1500 tolerance and shows that this particular contribution to the total radiation at the 88-inch water level is negligible.

B. SHIELDING FOR AIR-COOLED FACILITY TUBE ABOVE THE CORE*

Consider a large facility eight inches in diameter.

1. Radiation during After-shutdown Operations

a. At the Refueling Platform

The refueling platform is four feet above the refueling water level.

The 2.5 Mev gamma source is 3.7×10^{16} Mev/sec.

Assuming a source similar in material to normal fuel elements and producing 35 MW during operation, the source strength is given by*

$$\frac{35 \text{ MW}}{175 \text{ MW}} (3.7 \times 10^{16}) = 7.4 \times 10^{15} \text{ Mev/sec}$$

The tube cross sectional area is 324 cm^2 , so

$$S_A = \frac{7.4 \times 10^{15}}{324} = 2.3 \times 10^{13} \text{ Mev/cm}^2/\text{sec}$$

Assuming that half the source radiation proceeds upward

$$\frac{S_A}{2} = 1.2 \times 10^{13} \text{ Mev/cm}^2/\text{sec}$$

This source is then divided into four equal plane sources, each located at the midplane of a 9-inch active fuel element section.

The final value of radiation at the water level in this case is that given in Appendix XI-F adjusted for the increased source at two hours after shut-

* See Appendix XI-E for a more complete explanation of the method used.

down, the longer tube cross section, and the greater geometrical attenuation. Then the dose rate is

$$\begin{aligned}
 12 \text{ mr/hr} & \left[\left(\frac{2.1 \times 10^8 \text{ Mev/sec/watt}}{1.9 \times 10^8 \text{ Mev/sec/watt}} \right) \left(\frac{182 \text{ cm}^2}{324 \text{ cm}^2} \right) \left(\frac{3.86 \times 10^{-4}}{6.25 \times 10^{-4}} \right) \right] \\
 & = 12 (0.385) \\
 & = 4.6 \text{ mr/hr}
 \end{aligned}$$

The total radiation from all energy groups is 15 mr/hr.

To decrease the radiation hazard when handling the removable air-cooled facility shielding, it is recommended that if a helix is used the initial turns above the core contain essentially a 2 mm layer of boron to depress the flux in the steel during operation. This greatly reduces activity in the steel.

b. At the 88-inch water level

The decrease in geometrical attenuation is a factor of 2.52.

Thus, at the 88-inch water level the radiation from the active fuel elements is 38 mr/hr.

c. Radiation from Activated Steel in the Helix and Tube Walls

From the analysis of the insulating air annuli around the tubes (Appendix XIV-A), it is seen that this contribution is negligible in comparison with decay gammas from the fuel.

2. Radiation During Reactor Operation

a. Core Gammas

The 7 Mev core source is 2.4×10^{19} Mev/sec.

As before, the source for the facility tube is

$$S = \frac{35 \text{ MW}}{175 \text{ MW}} (2.4 \times 10^{19}) = 4.8 \times 10^{18} \text{ Mev/sec}$$

Dividing by the tube cross sectional area gives

$$S_A = \frac{4.8 \times 10^{18}}{324} = 1.48 \times 10^{16} \text{ Mev/cm}^2/\text{sec.}$$

Assuming that half the source radiation proceeds upward

$$\frac{S_A}{2} = 7.4 \times 10^{15} \text{ Mev/cm}^2/\text{sec}$$

The air-cooled facility tube has a concrete plug about 10 feet long in its uppermost section. Therefore, the geometrical attenuation is along a 12-foot section of tube between the top of the core and the plug.

The geometrical attenuation is

$$\frac{r^2}{L^2} = \frac{(4)^2}{(144)^2} = 7.8 \times 10^{-4}$$

The attenuation in 16 inches of steel is

$$\mu x = (0.225) (40.6) = 9.15$$

The source is divided into four plane sources as before.

$$\mu_s = (0.2) (0.225) + (0.4) (0.026) = 0.055$$

$$\mu_s x \text{ for source 1} = (0.055) (11.4) = 0.63$$

$$\text{Total } \mu x = 9.8$$

For Source 1

$$\begin{aligned} \Phi_E &= \frac{(7.4 \times 10^{15}) (7.8 \times 10^{-4})}{(4) (900)} (4.5) (5.1 \times 10^{-6}) \\ &= 3.7 \times 10^4 \text{ mr/hr} \end{aligned}$$

The path of least attenuation from the beginning of the plug to access areas above the top head is through 10 feet of water, 4 1/2 inches of steel in the cover plate, and about 2 feet of magnetite concrete in the top head shield.

The attenuating materials are

	$\mu \text{ (cm}^{-1}\text{)}$	$x \text{ (cm)}$	μx
Water	0.026	305	7.9
Steel	0.225	11.4	2.57
Concrete	0.10	61	6.1

$$\text{Total } \mu x \text{ above the beginning of the plug} = 16.57$$

Adding the μx in the steel helix gives 26.3 Then,

$$\begin{aligned} \Phi_E &= \frac{(7.4 \times 10^{15}) (7.8 \times 10^{-4})}{(4) (900)} (10) (1.4 \times 10^{-13}) \\ &= 0.002 \text{ mr/hr} \end{aligned}$$

This is 1/3000 tolerance If the concrete in the top head shield is decreased to 12 inches, the radiation is increased to 1/100 tolerance.

b. Fast Neutrons

The fast neutron current proceeding up the tube is about 5×10^{14} n/cm²/sec.

The geometrical attenuation is

$$\frac{r^2}{L^2} = \frac{(4)^2}{2(144)^2} = 3.9 \times 10^{-4}$$

$\sum_s x$ in the 16-inch steel helix is

$$(0.254) (40.6) = 10.3$$

If all radiation reaching the concrete plug is then scattered out into the water, the path of least attenuation would be through 10 feet of water plus the 2 feet of concrete above the vessel head. Using $\lambda = 9$ cm for water and 8.2 cm for magnetite concrete, the number of relaxation lengths is

$$\text{water} \quad \frac{x}{\lambda} = \frac{305}{9} = 34$$

$$\text{concrete} \quad \frac{x}{\lambda} = \frac{61}{8.2} = 7.4$$

Then the radiation above the top shield is

$$(5 \times 10^{14}) (3.9 \times 10^{-4}) (e^{-52})$$

This is negligible.

C. RADIATION FROM ACTIVITY IN THE INLET AND OUTLET COOLANT PIPES

1. Activity in the Coolant Pipes

The following calculations determine the activity ($Q_v E$) at Points 1, 3, 5, 2, 4, 6 in the outlet and inlet pipes as shown in Figure A14-1. *

Point 1:

Given:

(1) Water gamma activity in core

$$Q_v = 6.6 \times 10^6 \text{ photons/cc/sec}$$

$$Q_v E = 4.3 \times 10^7 \text{ Mev/cc/sec}$$

(2) Time for water to travel from core to Point 1

$$t = 3.27 \text{ sec}$$

(3) Water gamma activity at Point 2

$$Q_v = 6.24 \times 10^4 \text{ photons/cc/sec}$$

$$Q_v E = 4.1 \times 10^5 \text{ Mev/cc/sec}$$

(4) $T^{\frac{1}{2}}$ (half-life) of N^{16} gamma activity

$$T^{\frac{1}{2}} = 7.35 \text{ sec}$$

(5) Water velocity in the pipes

$$v = 14.6 \text{ ft/sec}$$

* Refer to Appendix VIII, Calculations for the Heat Exchanger Building.

ACTIVITY IN THE COOLANT PIPES

Fig. A14-1

$$\begin{aligned}
Q_{v1} E &= Q_{v \text{ core}} E e^{-\lambda t} \\
&= (4.3 \times 10^7) e^{-(0.0943) (3.27)} \\
&= (4.3 \times 10^7) (0.735) \\
&= 3.2 \times 10^7 \text{ Mev/cc/sec}
\end{aligned}$$

where

$$T^{\frac{1}{2}} = 0.693$$

$$\lambda = \frac{0.693}{7.35 \text{ sec}}$$

$$\lambda = 0.0943 \text{ sec}^{-1}$$

Point 3:

$$\begin{aligned}
Q_{v3} E &= Q_{v1} E e^{-\lambda t_{1-3}} \\
&= (3.2 \times 10^7) e^{-(0.0943) (1.6)} \\
&= (3.2 \times 10^7) e^{-(0.150)} = (3.2 \times 10^7) (0.861) \\
&= 2.8 \times 10^7 \text{ Mev/cc/sec}
\end{aligned}$$

$$\begin{aligned}
\text{where } t_{1-3} &= \frac{23.5 \text{ ft}}{14.6 \text{ ft/sec}} \\
&= 1.6 \text{ sec}
\end{aligned}$$

Point 5:

$$\begin{aligned}
Q_{v5} E &= Q_{v3} E e^{-\lambda t_{3-5}} \\
&= (2.8 \times 10^7) (0.861) \\
&= 2.4 \times 10^7 \text{ Mev/cc/sec}
\end{aligned}$$

$$\text{where } t_{3-5} = 1.6 \text{ sec}$$

Point 2:

$$Q_{v2} E = 4.1 \times 10^5 \text{ Mev/cc/sec}$$

Point 4:

$$Q_{v2} E = Q_{v4} E e^{\lambda t_{4-2}}$$

$$\text{where } t_{4-2} = 1.6 \text{ sec}$$

$$Q_{v4} E = Q_{v2} E e^{\lambda t_{4-2}}$$

$$= (4.1 \times 10^5) e^{(0.0943)(1.60)}$$

$$= (4.1 \times 10^5) e^{(0.150)}$$

$$= (4.1 \times 10^5) (1.16)$$

$$= 4.8 \times 10^5 \text{ Mev/cc/sec}$$

Point 6:

$$Q_{v6} E = Q_{v4} E e^{\lambda t_{6-4}}$$

$$\text{where } t_{6-4} = 1.6 \text{ sec}$$

$$= (4.8 \times 10^5) (1.16)$$

$$= 5.6 \times 10^5 \text{ Mev/cc/sec}$$

2. Radiation Outside Pipe Tunnel

Radiation from the inlet pipe is assumed insignificant compared to radiation from the outlet pipe. Calculations are for 6.5 Mev gammas.

$$\overline{\Phi}_E = \frac{B Q_v E C^2 F(\phi, b_2)}{2(a+z)}$$

a. At Point P_1 : (Located 6 1/2 feet above the console floor and 6 inches from the tunnel wall. See Figure A14-2.)

$$\mu_s \text{ (water)} = 0.026 \text{ cm}^{-1}$$

$$\mu \text{ (magnetite concrete)} = 0.10 \text{ cm}^{-1}$$

$$\mu \text{ (steel)} = 0.225 \text{ cm}^{-1}$$

$$C^2 = r^2 = (44.7 \text{ cm})^2 = 19.98 \times 10^2 \text{ cm}^2$$

$$b_1 = \mu_1 x_1 + \mu_2 x_2 = (0.225)(1.016) + (0.10)(137.2)$$

$$= 0.229 + 13.7 = 14.0$$

$$\mu_s C = (0.026)(44.7) = 1.16$$

$$\mu_s z = 0.67; \quad z = 25.8 \text{ cm}$$

$$b_2 = b_1 + \mu_s z = 14.0 + 0.67 = 14.7$$

$$F(90^\circ, b_2) = 1.40 \times 10^{-7}$$

$$2(a+z) = 2(228.6 + 25.8) = 508.8 \text{ cm}$$

$$B = 6.6$$

CROSS SECTION OF COOLANT PIPE TUNNEL

Fig. A14-2

$$\bar{\Phi}_E = \frac{(6.6) (3.2 \times 10^7) (1.998 \times 10^3) (1.40 \times 10^{-7})}{5.088 \times 10^2}$$

$$= 1.16 \times 10^2 \text{ Mev/cm}^2/\text{sec} \text{ or } \frac{1.16 \times 10^2}{900} = 0.13 \text{ mr/hr}$$

b. At Point P₂: (Located above the tunnel adjacent to the pressure vessel. See Figure A14-2.)

$$b_1 = (0.225 \times 1.016) + (0.10 \times 121.92) = 12.42$$

$$b_2 = b_1 + \mu_s z = 13.1$$

$$F(90^\circ, b_2) = 6.6 \times 10^{-7}$$

$$2(a + z) = 882 \text{ cm}$$

$$B = 5.8$$

$$\bar{\Phi}_E = \frac{(5.8) (3.2 \times 10^7) (1.998 \times 10^3) (6.6 \times 10^{-7})}{8.82 \times 10^2}$$

$$= 2.77 \times 10^2 \text{ Mev/cm}^2/\text{sec} \text{ or } \frac{2.77 \times 10^2}{900} = 0.31 \text{ mr/hr}$$

D. REMOVAL OF AN ALUMINUM GUIDE TUBE AND REGULATING ROD

1. Aluminum Guide Tube

The aluminum guide tubes in the core are 3 inches square, about 15 feet 3 inches long, and 0.203 inches thick. The top of the guide tube is three feet above the active section of the core. The length of the guide tube in the core is three feet. The average vertical flux in the core is used in the calculation.

$$\begin{aligned}\text{Volume of 3-foot section} &= (36) \left[9 - (2.594)^2 \right] = 82.1 \text{ cu. in.} \\ &= 1350 \text{ cc}\end{aligned}$$

$$\text{Weight of 3-foot section} = 3642 \text{ gm}$$

For Type 2S aluminum activated at a thermal flux of 10^{12} n/cm²/sec for 15,000 hours, the decay gammas are about 2.5×10^7 photons/sec/gm four hours after exposure.* With an average gamma energy of 1.2 Mev, this is 3.0×10^7 Mev/sec/gm. The average flux at the center tube is 5.3×10^{14} n/cm²/sec.

Therefore, the source strength is

$$\begin{aligned}Q &= (3.0 \times 10^7 \text{ Mev/sec/gm}) \left(\frac{5.3 \times 10^{14}}{10^{12}} \right) (3.642 \times 10^3 \text{ gm}) \\ &= 5.8 \times 10^{13} \text{ Mev/sec}\end{aligned}$$

With this source strength the operation of the crane for removal of an aluminum guide tube cannot be accomplished from the first floor. The dose rate is determined for an operator located on the floor below where the crane can

* MTR Handbook, ORNL 963, Shielding Section.

be operated remotely. (See Figure A14-3.) This places the operator about 44 feet from the source, with about 14 inches of ordinary concrete in the attenuation path.

The μ of concrete is 0.135 cm^{-1} and neglecting air absorption:

$$b_1 = (0.135) (35.6) = 4.80$$

$$a = (44) (30.5) = 1342 \text{ cm}$$

Using a point source analysis

$$\begin{aligned} \frac{\Phi}{E} &= \frac{(5.6 \times 10^{13}) (8) e^{-(4.80)}}{4 (1342)^2} \\ &= 1.6 \times 10^5 \text{ Mev/cm}^2/\text{sec} \text{ or } \frac{16 \times 10^4}{5.6 \times 10^2} = 290 \text{ mr/hr.} \end{aligned}$$

2. Regulating Rod

The regulating rod is approximately 14 feet long. In transferring the rod to the canal it can be either dropped down the discharge chute or removed through the top head. In either case, at least part of the rod has to come above the water (two feet when dropped down the chute, all when removed through the top head). Since removing the rod through the top head is probably the faster of the two methods, it is the one considered here.

SCHEMATIC SHOWING REMOVAL OF ALUMINUM GUIDE TUBE
AND/OR REGULATING ROD

SCALE: $\frac{1}{8}" = 1'-0"$

A schematic of the regulating rod is shown in the diagram. The three-foot steel section at the top of the rod moves in and out of the beryllium. The aluminum and other stainless steel section extend below the core. The activation of these sections is not considered here, since it is small compared to the section in the beryllium (i. e. the aluminum and the other stainless steel section are in a low flux area).

To be conservative, it is assumed that the entire three-foot stainless steel section is activated by the average thermal flux in the beryllium. The stainless steel contains about one percent boron by weight, which depresses the activating thermal flux in the stainless steel. However, since the boron burns up, it is assumed only 1/5 percent effective.

$$\begin{aligned}\text{Volume of stainless steel} &= (36) (\pi) \left[\frac{(1.673)^2}{2} - \frac{(1.3125)^2}{2} \right] \\ &= 30.6 \text{ cu. in.} \\ &= 503 \text{ cc}\end{aligned}$$

$$\text{Weight of stainless steel} = 3.93 \times 10^3 \text{ gm}$$

For Type 304 stainless steel activated at a thermal flux of 10^{12} n/cm²/sec for 1,000 hours, the decay gammas are about 3.2×10^8 photons/sec/gm four hours after exposure.* These have an energy of about 1.2 Mev. Therefore, the source strength is

$$\begin{aligned}Q &= (3.2 \times 10^8) (1.2) \frac{(1.5 \times 10^{14})}{10^{12}} (3.9 \times 10^3) \\ &= 2.3 \times 10^{14} \text{ Mev/sec}\end{aligned}$$

* MTR Handbook, ORNL 963, Shielding Section.

where the flux of 1.5×10^{14} was taken from UNIVAC calculations.

Consider Type 304 stainless steel (9% Ni, 19% Cr, 71% Fe, 1% Mn, 0.5% Co by weight) with 0.2 percent boron by weight added.

	<u>V. F.</u>	<u>$\sum_a$</u>	<u>V. F. x $\sum_a$</u>
Ni	0.0755	0.372	0.028
Cr	0.2050	0.206	0.042
Fe	0.6754	0.190	0.128
Mn	0.0104	0.0923	0.001
B	0.0059	93.4	0.551
Co	0.0043	2.92	<u>0.013</u>
			0.763

A comparison of the boron captures and the total captures shows that the flux in the steel of the regulating rod is depressed to about 30 percent of the flux value originally used. Hence, the source strength may be decreased to

$$Q = (2.3 \times 10^{14}) (0.3)$$

$$= 6.9 \times 10^{13} \text{ Mev/sec}$$

This source strength is only slightly greater than that for the aluminum guide tube.