LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission to the extent that such employee or contractor prepares, handles or distributes, or provides access to, any information pursuant to his employment or contract with the Commission.

NAA-SR-113

PHYSICS AND MATHEMATICS

SHIELDING FOR THE LOW POWER RESEARCH REACTOR

Price \$ ~~0.25~~

Available from the
Office of Technical Services
Department of Commerce
Washington 25, D. C.

WRITTEN BY:
M. NEUSTADT

WORK DONE BY:
M. NEUSTADT

CLASSIFICATION CANCELLED

DATE FEB 27 1957 *GL*
For The Atomic Energy Commission

H. B. Canale
Chief, Declassification Branch

MARK MILLS - GROUP LEADER

S. SIEGEL - ASSOCIATE TECHNICAL DIRECTOR

C. STARR - TECHNICAL DIRECTOR

ATOMIC ENERGY RESEARCH DEPARTMENT
NORTH AMERICAN AVIATION, INC.
P. O. BOX 309
DOWNEY, CALIFORNIA

ISSUE DATE:
FEBRUARY 20, 1951

CONTRACT AT 11-1-GEN-8

UNCLASSIFIED

5
 6
 7
 8
 9
 10
 11
 12
 13
 14
 15
 16
 17
 18
 19
 20
 21
 22
 23
 24
 25
 26
 27
 28
 29
 30
 31
 32
 33
 34
 35
 36
 37
 38
 39
 40
 41
 42
 43
 44
 45
 46
 47
 48
 49
 50
 51
 52
 53
 54
 55
 56
 57
 58
 59
 60
 61
 62
 63
 64
 65
 66
 67
 68
 69
 70
 71
 72
 73
 74
 75
 76
 77
 78
 79
 80
 81
 82
 83
 84
 85
 86
 87
 88
 89
 90
 91
 92
 93
 94
 95
 96
 97
 98
 99
 100
 101
 102
 103
 104
 105
 106
 107
 108
 109
 110
 111
 112
 113
 114
 115
 116
 117
 118
 119
 120
 121
 122
 123
 124
 125
 126
 127
 128
 129
 130
 131
 132
 133
 134
 135
 136
 137
 138
 139
 140
 141
 142
 143
 144
 145
 146
 147
 148
 149
 150
 151
 152
 153
 154
 155
 156
 157
 158
 159
 160
 161
 162
 163
 164
 165
 166
 167
 168
 169
 170
 171
 172
 173
 174
 175
 176
 177
 178
 179
 180
 181
 182
 183
 184
 185
 186
 187
 188
 189
 190
 191
 192
 193
 194
 195
 196
 197
 198
 199
 200
 201
 202
 203
 204
 205
 206
 207
 208
 209
 210
 211
 212
 213
 214
 215
 216
 217
 218
 219
 220
 221
 222
 223
 224
 225
 226
 227
 228
 229
 230
 231
 232
 233
 234
 235
 236
 237
 238
 239
 240
 241
 242
 243
 244
 245
 246
 247
 248
 249
 250
 251
 252
 253
 254
 255
 256
 257
 258
 259
 260
 261
 262
 263
 264
 265
 266
 267
 268
 269
 270
 271
 272
 273
 274
 275
 276
 277
 278
 279
 280
 281
 282
 283
 284
 285
 286
 287
 288
 289
 290
 291
 292
 293
 294
 295
 296
 297
 298
 299
 300
 301
 302
 303
 304
 305
 306
 307
 308
 309
 310
 311
 312
 313
 314
 315
 316
 317
 318
 319
 320
 321
 322
 323
 324
 325
 326
 327
 328
 329
 330
 331
 332
 333
 334
 335
 336
 337
 338
 339
 340
 341
 342
 343
 344
 345
 346
 347
 348
 349
 350
 351
 352
 353
 354
 355
 356
 357
 358
 359
 360
 361
 362
 363
 364
 365
 366
 367
 368
 369
 370
 371
 372
 373
 374
 375
 376
 377
 378
 379
 380
 381
 382
 383
 384
 385
 386
 387
 388
 389
 390
 391
 392
 393
 394
 395
 396
 397
 398
 399
 400
 401
 402
 403
 404
 405
 406
 407
 408
 409
 410
 411
 412
 413
 414
 415
 416
 417
 418
 419
 420
 421
 422
 423
 424
 425
 426
 427
 428
 429
 430
 431
 432
 433
 434
 435
 436
 437
 438
 439
 440
 441
 442
 443
 444
 445
 446
 447
 448
 449
 450
 451
 452
 453
 454
 455
 456
 457
 458
 459
 460
 461
 462
 463
 464
 465
 466
 467
 468
 469
 470
 471
 472
 473
 474
 475
 476
 477
 478
 479
 480
 481
 482
 483
 484
 485
 486
 487
 488
 489
 490
 491
 492
 493
 494
 495
 496
 497
 498
 499
 500
 501
 502
 503
 504
 505
 506
 507
 508
 509
 510
 511
 512
 513
 514
 515
 516
 517
 518
 519
 520
 521
 522
 523
 524
 525
 526
 527
 528

C

2

٤٤٤

1

✓

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

TABLE OF CONTENTS

	Page
Abstract.	4
I. Introduction	5
II. Discussion.	5
III. Computations.	6
A. Neutrons.	6
B. Gamma Rays	8
IV. Conclusions	11
References.	12

ABSTRACT

The radiation shield required for the Low Power Research Reactor is analyzed. Neutrons, primary gamma rays, secondary gamma rays, as well as the effect of self-shielding, have been considered.

It is shown that a 34-inch MO concrete shield is adequate. This is less than the shield thickness given in NAA-SR-34.

This report is based upon studies conducted for the Atomic Energy Commission under Contract AT-11-1-GEN-8.

DECLASSIFIED

I. INTRODUCTION

One seeks as accurate a calculation of the shield as possible. This is desirable (a) from the point of view of personnel safety and (b) from the point of view of minimum shield cost. In order to get an accurate estimate of the shield attenuation needed, a study of the self-shielding by the core and reflector has been made. This was done for a spherical reactor with graphite moderator and reflector. The results have general application and are therefore presented in a separate report.⁽¹⁾ The results include (a) the leakage of neutrons from the reflector surface per neutron produced, and (b) the leakage of gamma rays from the reflector surface per gamma ray produced in the core. A knowledge of the reactor power specifies the neutron and gamma ray production rate. One can then get the current of neutrons and gamma rays from the reflector. This, together with the current desired at the outer shield surface, enables one to specify the attenuation needed in the shield. From this and the neutron and gamma ray attenuation lengths in MO concrete, the shield thickness can be calculated.

II. DISCUSSION

The reactor considered is a spherical reactor with graphite moderator and reflector. The power level of the reactor is taken at 160 kilowatts. The dosage rates desired at the outer shield surface are 1/100 of the permissible daily dosage rates. The permissible daily dosage rate⁽²⁾ is 0.05 r/8 hours. This corresponds to 1650 2-Mev gamma rays/cm²/sec or to 20 2-20-Mev neutrons⁽³⁾/cm²/sec. The currents desired at the outer shield surface are therefore 16.5 2-Mev gamma rays/cm²/sec and 0.20 fast neutrons/cm²/sec. The neutrons and gammas below 2-Mev energy are not considered owing to the fact that their attenuation lengths are much smaller and their tolerance levels much larger than in the case of 2-Mev neutrons and gamma rays.

The report on reactor self-shielding⁽¹⁾ gives $\Psi_{\infty}(E)$ for neutrons. Here $\Psi_{\infty}(E)$ is the number of neutrons leaking out of the reflector with energies greater than E per neutron produced in the core. The particular case of interest here is $E = 2.0$ Mev. Integrations have been carried out over the U-235 fission neutron spectrum. Here $\Psi_{\infty}(2.0)$ includes neutrons which have made zero collisions, one collision, two collisions, three collisions, etc., but still have energy greater than 2.0 Mev when they escape from the reflector. The results apply to the case $\mu R = 7.32$, $t/R = 1.25$, where μ is the macroscopic total neutron cross section in graphite; R is the core radius; t is the reflector thickness. The quantity $\mu = N\sigma$,

DECLASSIFIED

where $N = 0.0802 \times 10^{24}$ graphite atoms per cubic centimeter and σ is the total cross section of carbon for neutrons. The calculation was made for the case $R = 60.8$ cm, $t = 76.2$ cm (Reactor A). In the range $0.5 < E < 2.0$ Mev, $\sigma = 1.5$ barns was used as an average cross section. Corresponding to this $\mu = 0.1204$ cm⁻¹. For this case $\Psi_{\infty}(2.0) = 1.684 \times 10^{-4}$ neutrons leaking from the reflector per neutron produced.

The case $R = 50$ cm, $t = 62.5$ cm (Reactor B) is also of interest. The same value of $\Psi_{\infty}(2.0) = 1.684 \times 10^{-4}$ neutrons leaking from the reflector per neutron produced may be used here, since this result applies to any reactor for which $\mu R = 7.32$, $t/R = 1.25$. In this case $\mu = 7.32/50 = 0.1464$ cm⁻¹ and $\sigma = 1.8$ barns. Since σ is only an average cross section and $\sigma = 1.8$ barns is not significantly different from $\sigma = 1.5$ barns (see total cross section vs energy curve for carbon in range $0.5 < E < 2.0$ Mev), the value of $\Psi_{\infty}(2.0)$ may be regarded as insensitive to this change in total cross section.

The report on reactor self-shielding⁽¹⁾ gives Ψ for gamma rays. Here Ψ is the number of gamma rays leaking out of the reflector per gamma ray produced in the core. The quantity Ψ is given as a function of $\mu_{\gamma}R$ from $\mu_{\gamma}R = 0$ to $\mu_{\gamma}R = 20$ for values of t/R from $t/R = 0$ to $t/R = 2.00$ at intervals of 0.25. Here μ_{γ} = gamma ray absorption coefficient in graphite for monochromatic gamma rays of energy E . Taking $E = 2$ Mev as the gamma ray energy, $\mu_{\gamma} = 0.077$ cm⁻¹ in graphite.⁽⁴⁾ For the case $R = 60.8$ cm, $t = 76.2$, and $\mu_{\gamma} = 0.077$ cm⁻¹, one has $\mu_{\gamma}R = 4.68$ and $t/R = 1.25$. One therefore has⁽¹⁾ $\Psi = 2.21 \times 10^{-4}$ 2-Mev gamma rays leaking out of the reflector per gamma ray produced in the core. For the case $R = 50.0$ cm, $t = 62.5$ cm, $\mu_{\gamma} = 0.077$ cm⁻¹, one has $\mu_{\gamma}R = 3.85$ and $t/R = 1.25$. One therefore has⁽¹⁾ $\Psi = 8.2 \times 10^{-4}$ 2-Mev gamma rays leaking out of the reflector per gamma ray produced in the core. The shield thickness is determined in the present report for the two cases: (1) reactor A where $R = 60.8$ cm and $t = 76.2$ cm and (2) reactor B where $R = 50.0$ cm and $t = 62.5$ cm. In each of these cases between the graphite reflector and the MO concrete there is a 2-inch air channel for circulating air to cool the reflector and also a 6-inch iron thermal shield. The shield thickness calculation is based on the neutron, the primary gamma ray and the secondary gamma ray requirements.

III. COMPUTATIONS

A. NEUTRONS

Corresponding to a power level of 160 kilowatts, there are

DECLASSIFIED

4.99×10^{15} fissions/sec. Integration over the U-235 fission neutron spectrum⁽¹⁾ gives 0.98 neutrons above 2-Mev energy per fission. Consequently, there are 4.89×10^{15} neutrons/sec liberated above 2-Mev energy. From the reactor self-shielding study,⁽¹⁾ the straight ahead calculation, which includes integration over the U-235 fission neutron spectrum, gives $\Psi_{\infty}(2.0) = 1.684 \times 10^{-4}$ neutrons leaking from the reflector with energies greater than 2.0 Mev per neutron produced. This is valid as already mentioned for the case $R = 60.8$ cm, $t = 76.2$ cm, and for the case $R = 50.0$ cm, $t = 62.5$ cm. The number of neutrons leaking out in all directions through the reflector surface is therefore $(4.89 \times 10^{15})(1.684 \times 10^{-4}) = 8.23 \times 10^{11}$ neutrons/sec above 2.0 Mev. It will be conservative to assume that all of these neutrons emerge moving radially, that is, normal to the reflector surface. There is a 2-inch cooling air channel adjacent to the reflector surface, followed by a 6-inch iron thermal shield between the reflector and the concrete.⁽⁵⁾ This makes a total of 8 inches or 20.3 centimeters between the reflector and the concrete shield.

1. Reactor A - In this case, the core radius R is 60.8 cm, the reflector thickness t is 76.2 cm and the reflector radius A is 137.0 cm. The inner radius of the shield is $137.0 + 20.3 = 157.3$ cm. The area of the shield is $4\pi(157.3 + \chi)^2$ where χ is the thickness of the shield. The 6 inches of iron introduces^(6, 7) a neutron attenuation factor of 5. This is conservative.⁽⁷⁾ This attenuation is due primarily to inelastic scattering in the iron. The leakage flux before attenuation in the concrete is

$$\frac{8.23 \times 10^{11}}{20 \pi (157.3 + \chi)^2} \text{ neutrons/cm}^2/\text{sec.}$$

The permissible daily dosage rate⁽²⁾ is 0.05 r/8 hours. This corresponds to 20 neutrons⁽³⁾ (2-20 Mev)/cm²/sec. The flux level desired at the outer shield surface is 1/100 of the permissible daily dosage rate. The neutron flux desired at the outer shield surface is therefore 0.20 fast neutrons/cm²/sec.

The neutron attenuation length in M-1 concrete⁽⁷⁾ (of density 4.7 gm/cm³) is 5.9 cm, and in MO concrete⁽⁷⁾ (of density 5.6 gm/cm³) is 6.1 cm. Both M-1 and MO concretes contain iron as well as hydrogen. Let us use the 6.1 cm attenuation length of MO concrete, to be conservative. The attenuation is conservatively given by

$$\frac{1}{0.20} \frac{8.23 \times 10^{11}}{20 \pi (157.3 + \chi)^2} = e^{\frac{\chi}{6.1}} \quad \dots (1)$$

Solution of this for χ yields $\chi = 85.0$ cm. This corresponds to 33.5 inches.

DECLASSIFIED

The attenuation by the concrete is $e^{X/6.1} = 1.12 \times 10^6$. The attenuation by the iron thermal shield and by the concrete is 5.60×10^6 .

2. Reactor B - In this case the core radius R is 50.0 cm., the reflector thickness t is 62.5 cm, and the reflector radius A is 112.5 cm. The inner radius of the shield is $112.5 + 20.3 = 132.8$ cm. Assuming again, conservatively, a neutron attenuation factor of 5 for the 6-inch iron thermal shield, the leakage flux before attenuation in the concrete is

$\frac{8.23 \times 10^{11} \text{ neutrons}}{20 \pi (132.8 + X)^2 \text{ cm}^2\text{-sec}}$. Using the same flux at the surface of the shield, 0.20 fast neutrons/cm²/sec and the same attenuation length 6.1 cm as in case 1, one has

$$\frac{1}{0.20} \frac{8.23 \times 10^{11}}{20 \pi (132.8 + X)^2} = e^{\frac{X}{6.1}} \quad \dots (2)$$

Solution of this for X yields $X = 86.3$ cm. This corresponds to 34.0 inches. The attenuation by the concrete is $e^{X/6.1} = 1.37 \times 10^6$. The attenuation by the iron thermal shield and by the concrete is 6.85×10^6 .

B. GAMMA RAYS

1. Primary Gamma Rays - Corresponding to a power of 160 kilowatts, there are 4.99×10^{15} fissions per second. Taking 5 Mev of prompt gamma rays and 5 Mev of fission product gamma rays, there are 10 Mev of gamma rays released per fission. Assuming, conservatively, that they are all 2-Mev gamma rays, one has 5 gamma rays per fission. Corresponding to the power of 160 kilowatts there are 2.50×10^{16} quanta/sec produced in the reactor core. Graphite has the absorption coefficient 0.077 cm^{-1} for 2-Mev gamma rays.

Reactor A (core radius $R = 60.8$ cm and reflector thickness $t = 76.2$ cm) Since $\mu = 0.077 \text{ cm}^{-1}$ and $R = 60.8$ cm, $\mu R = 4.68$. Also $t/R = 1.25$. Corresponding to this value of μR and t/R the gamma ray leakage from the reflector per gamma ray produced in the core⁽¹⁾ is $\Psi = 2.21 \times 10^{-4}$. The number of gamma rays leaking out of the reflector per second is $(2.50 \times 10^{16})(2.21 \times 10^{-4}) = 5.52 \times 10^{12}$ quanta/sec. Attenuation measurements⁽⁸⁾ show that both MO and M-1 concrete have nearly equal shield performance in terms of relaxation weight $\rho\lambda$ for gamma rays, where ρ is the density of the concrete and λ is the attenuation length. Here⁽⁸⁾ $\rho\lambda = 42 \text{ gm/cm}^2$, $\rho = 5.8 \text{ gm/cm}^3$ and $\lambda = 7.24$ cm, in MO concrete.

In the 6-inch or 15.24-centimeter iron thermal shield, 2-Mev gamma

DECLASSIFIED

rays⁽⁴⁾ have the attenuation length $L = 2.9$ cm. They are therefore attenuated by the factor $e^{-\frac{15.24}{2.9}} = e^{-5.25} = 0.00524$. The inner radius of the shield is $137.0 + 20.3 = 157.3$ cm. The permissible dosage rate⁽²⁾ is 0.05 r/8 hours. This corresponds to 1650 2-Mev gamma rays/cm²/sec. The flux level desired at the outer shield surface is $1/100$ of the permissible daily dosage rate. The gamma ray flux desired at the outer shield surface is therefore 16.5 2-Mev gamma rays/cm²/sec. The attenuation in the concrete is

$$\frac{1}{16.5} (0.00524) \frac{5.52 \times 10^{12}}{4\pi(157.3 + \chi)^2} = e^{-\frac{\chi}{7.24}} \quad \dots (3)$$

Solution of this for χ yields $\chi = 58.0$ cm. This corresponds to 22.9 inches. The gamma ray attenuation in the MO concrete is 3.00×10^3 . The gamma ray attenuation in the iron thermal shield and the MO concrete is 5.73×10^5 .

Reactor B (core radius $R = 50.0$ cm and reflector thickness $t = 62.5$ cm) Since $\mu = 0.077$ cm⁻¹ and $R = 50.0$ cm, $\mu R = 3.85$. Also $t/R = 1.25$. Corresponding to this value of μR and t/R , the gamma ray leakage from the reflector per gamma ray produced in the core⁽¹⁾ is $\Psi = 8.2 \times 10^{-4}$. The number of gamma rays leaking out of the reflector per second is $(2.50 \times 10^{16})(8.2 \times 10^{-4}) = 2.05 \times 10^{13}$ quanta/sec. Assuming again, conservatively, an attenuation factor of 0.00524 for the 2-Mev gamma rays in the 6-inch iron thermal shield; 16.5 2-Mev gamma rays/cm²/sec at the surface of the shield; an attenuation length of 7.24 cm in the MO shield; the inner radius of the shield to be $112.5 + 20.3 = 132.8$ cm, one has

$$\frac{1}{16.5} (0.00524) \frac{2.05 \times 10^{13}}{4\pi(132.8 + \chi)^2} = e^{-\frac{\chi}{7.24}} \quad \dots (4)$$

Solution of this for χ yields $\chi = 68.6$ cm. This corresponds to 27.0 inches. The gamma ray attenuation in the MO concrete is 1.27×10^4 . The gamma ray attenuation in the iron thermal shield and in the MO concrete is 2.43×10^6 . Results for both Reactor A and Reactor B are conservative because the attenuation length 7.24 cm corresponds more to secondary than to primary gamma rays.

2. Secondary Gamma Rays - When a neutron is captured, highly penetrating gamma radiation is emitted. As before, the number of neutrons leaking out of the reflector is 8.23×10^{11} neutrons/sec. It is conservative to assume that all of these neutrons emerge moving radially, that is, normal to the reflector surface. As before, the iron is assumed to introduce a neutron attenuation factor of 5 . There are thus

DECLASSIFIED

1.647×10^{11} neutrons/sec entering the shield. This is conservative.

Reactor A - The inner radius of the shield is $A_1 = 137.0 + 20.3 = 157.3$ cm. The neutron current entering the shield is therefore $I_0 = \frac{1.647 \times 10^{11}}{4\pi A_1^2} = 5.29 \times 10^5$ neutrons/cm²/sec. As a first approximation the shield may be considered to be an infinite slab of thickness $\chi = 85.0$ cm. The neutron current at distance χ' in the shield is given by

$$I_n = I_0 e^{-\mu_n \chi'} \quad \dots (5)$$

where $\mu_n = \frac{1}{6.1} = 0.1640$ cm⁻¹ in MO concrete. The fraction of the neutrons absorbed between χ' and $\chi' + d\chi'$ is $\mu_n d\chi'$. Let $C_{n\gamma}$ equal one secondary gamma ray emitted per neutron absorbed. Assume on the average half of the gamma rays to be emitted in the $+\chi'$ direction and half of them in the $-\chi'$ direction. The gamma rays reach the surface of the shield at $\chi' = \chi$ after being attenuated by the factor $e^{-\mu_\gamma (\chi - \chi')}$. The intensity of the gamma rays at the surface of the shield is given by

$$I_\gamma = \frac{C_{n\gamma}}{2} I_n e^{-\mu_\gamma (\chi - \chi')} \mu_n d\chi' \quad \dots (6)$$

where $\mu_\gamma = \frac{1}{7.24} = 0.1382$ cm⁻¹ for MO concrete.

Summing up the gamma rays produced throughout the slab one gets

$$I_\gamma = \frac{\mu_n C_{n\gamma}}{2} I_0 \int_0^\chi e^{-\mu_n \chi'} e^{-\mu_\gamma (\chi - \chi')} d\chi' \quad \dots (7)$$

Solution of Eq. (7) leads to

$$I_\gamma = I_0 \frac{C_{n\gamma}}{2} \frac{\mu_n}{\mu_n - \mu_\gamma} (e^{-\mu_\gamma \chi} - e^{-\mu_n \chi}) \quad \dots (8)$$

Using the values of I_0 , $C_{n\gamma}$, μ_n , μ_γ and χ given above, one gets $I_\gamma = 11.7$ gamma rays/cm²/sec. This is permissible. For a spherical shield even less than this will emerge.

Reactor B - The inner radius of the shield is $A_2 = 112.5 + 20.3 = 132.8$ cm. The neutron current entering the shield is therefore $I_0 = \frac{1.646 \times 10^{11}}{4\pi A_2^2} = 7.45 \times 10^5$ neutrons/cm²/sec. As a first approximation the shield may be considered to be an infinite slab of thickness $\chi = 86.3$ cm. Using the same values of $C_{n\gamma}$, μ_n and μ_γ as with Reactor A, one gets, from Eq. (8) $I_\gamma = 14.12$ gamma rays/cm²/sec emerging. This is permissible. For a spherical shield even less than this will emerge.

IV. CONCLUSIONS

Based on a reactor power of 160 kilowatts the shield thickness is calculated for the Low Power Research Reactor. This shield will allow 0.0005 r/8 hours at the shield surface or 1/100 of the permissible daily dosage rate. This corresponds to 16.5 2-Mev gamma rays/cm²/sec or to 0.20 fast neutrons/cm²/sec. There is a 6-inch iron thermal shield between the reflector and the concrete. The shield consists of MO type concrete. Two cases are considered: (A) a reactor with core radius $R_1 = 60.8$ cm and reflector thickness $t_1 = 76.2$ cm, and (B) a reactor with core radius $R_2 = 50.0$ cm and reflector thickness $t_2 = 62.5$ cm.

In Reactor A the neutron requirement is 85.0 cm or 33.5 inches of concrete. The primary gamma ray requirement is 58.0 cm or 22.9 inches of concrete. The secondary gamma ray current from the shield is less than 12 gamma rays/cm²/sec which is allowable.

In Reactor B the neutron requirement is 86.3 cm or 23.4 inches of concrete. The primary gamma ray requirement is 68.6 cm or 27.0 inches of concrete. The secondary gamma ray current from the shield is less than 15 gamma rays/cm²/sec which is allowable.

As a result, a shield thickness $\chi = 86.3$ cm or 34.0 inches is good for either Reactors A or B and is satisfactory for both neutrons and gamma rays.

The shield attenuations and shield thicknesses given in the preliminary report NAA-SR-34, pages 9, 17 and 27 are very conservative approximations. Instead of a neutron attenuation $\sim 10^9$ and a gamma ray attenuation $\sim 10^8$ from the surface of the reflector, they should be: a neutron attenuation $\sim 10^7$ and gamma ray attenuation $\sim 10^6$ from the surface of the reflector for 0.0005 r/8 hours. Instead of a shield thickness of 52 inches, the shield thickness should conservatively be 34 inches for the reactors considered here.

DECLASSIFIED

REFERENCES

1. Neustadt, M., E. R. Cohen, "Reactor Self-Shielding," NAA-SR-112, to be published.
2. "Safe Handling of Radioactive Isotopes," National Bureau of Standards, Handbook 42, September, 1949, page 6.
3. Mitchell, J. S., "British Journal of Radiology" 20, 177 (1947).
4. Gillette, P. R., "Activity of Fission Products", HW-17415, April 14, 1950, page 18.
5. Thompson, A. S. and T. Fahrner, "Design Study of a Low Power Research Reactor," NAA-SR-34, October 14, 1949, Fig. 3.
6. "Hanford Technical Manual," HW-10475, rev. August 1946, page 818.
7. Sleeper, H. P., Jr., "A Critical Review of ORNL Shield Measurements: Neutron Attenuation," ORNL-436, December 21, 1949, pp. 16, 18, 38.
8. Sleeper, H. P., Jr., C. E. Clifford, "Attenuation Measurements in Heavy Aggregate M-1 Concrete," ORNL-438, September 26, 1949, pp. 8, 14.