AIIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910

301/587-8202

Centimeter

Inches

[redacted] DECLASSIFIED

(CLASSIFICATION)

DOCUMENT NO.

HW-71174

SERIES AND COPY NO.

[redacted]7

DATE

September 27, 1961

GENERAL ELECTRIC

HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

[X] [redacted]RICTED [redacted]A

THIS [redacted]NT CONTA[redacted]RICTED DA[redacted] DE[redacted] IN THE A[redacted] ENERGY AC[redacted]54. [redacted]RANSMITT[redacted]THE DISCL[redacted] ITS [redacted]NTENTS [redacted]ANNER TO AN U[redacted]THORIZED PERSON [redacted]IBITED.

TITLE

NPR REACTOR SHIELD CALCULATIONS

[ ] OTHER OFFICIAL CLASSIFIED INFORMATION

THIS MATERIAL CONTAINS INFORMATION AFFECTING THE NATIONAL DEFENSE OF THE UNITED STATES WITHIN THE MEANING OF THE ESPIONAGE LAWS, TITLE 18, U. S. C., SECS. 793 AND 794, THE TRANSMISSION OR REVELATION OF WHICH IN ANY MANNER TO AN UNAUTHORIZED PERSON IS PROHIBITED BY LAW.

AUTHOR

E. G. Peterson

ISSUING FILE

THIS DOCUMENT MUST [redacted] BE LEFT UNATTENDED [redacted]WHERE AN UNAUTHORIZE[redacted]MAY HAVE ACCESS TO IT. WHEN NO[redacted], IT MUST BE STOR[redacted]PROVED LOCKED R[redacted]Y WITHIN AN APPRO[redacted] GUARDED A[redacted]WHILE IT IS YOUR [redacted]ON AND UNTIL YOU H[redacted]TAINED A SIGNED RECE[redacted] CLAS[redacted]FILES, IT IS YOU[redacted]ONSIBILITY TO KEEP IT[redacted]TS CONTENTS WITHIN T[redacted]TS OF [redacted]ROJECT AND FROM A[redacted]THORIZED PERSON. IT[redacted]SMITTAL TO, AND STOR[redacted] YOUR PLACE [redacted] RESIDENCE IS PRO[redacted]D. IT IS NOT TO BE [redacted]CATED. IF ADDITIONAL[redacted]S ARE REQUIRED, OBTAIN THEM FR[redacted] RELATED ISSUING FILE[redacted]PERSONS READING THIS DOCUMENT ARE REQUESTED TO SIGN IN THE SPACE PROVIDED BELOW.

ROUTE TO:	PAYROLL NO.	LOCATION	FILES ROUTE DATE	SIGNATURE AND DATE
[redacted]				

DECLASSIFIED

54-3000-340 (3-57) AEC-GE RICHLAND, WASH

[redacted]

(CLASSIFICATION)

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

[redacted]

DECLASSIFIED

This document consists of 13 pages. No. 11 of 25 copies. [redacted]

This document classified by

J W Finnigan
J. W. Finnigan

DISTRIBUTION

1. FW Albaugh
2. JM Batch
3. JH Brown
4. WL Bunch
5. RG Clark
6. HS Davis
7. GT Haugland
8. J DePangher
9. RL Dickeman
10. JW Finnigan
11. RM Fryar
12. GC Fullmer
13. S Goldsmith
14. CA Mansius
15. R Nilson
16. EG Peterson
17. RE Peterson
18. JJ Regimbal
19. WL Smalley
20. DE Wood
21. 300 File
22. Record Center
23-25. Extra

NPR REACTOR SHIELD CALCULATIONS

E. G. Peterson

September 27, 1961

Classification Cancelled and Changed To
DECLASSIFIED
By Authority of W A Snyder
CG-PR-2 3-2-94
By J E Savely 3-15-94
Verified By DK Hanson, 3-22-94

[redacted]

[redacted]ment contains [redacted] Its transmittal [redacted]uthorized person is prohibited.

[redacted]

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DECLASSIFIED

NPR REACTOR SHIELD CALCULATIONS

INTRODUCTION

At the request of IPD Personnel (1), calculations on neutron and gamma attenuation were made for the NPR shield. The calculations were made using a new shielding computer code developed for the IBM 7090. The calculations show the thermal neutron flux, total neutron dose rate, and gamma dose rate distribution through the entire shield assembly.

SUMMARY

The calculations show that the side and top primary shield design is adequate to reduce the radiation level below design tolerances. The radiation leakage through the front shield was higher than the design tolerances. Two alternate biological shield materials were studied for use on the front face. These two materials were iron serpentine concrete mixtures with densities of 245 lb/ft^3 and 265 lb/ft^3 (designated by I-S-245-P and I-S-265-P respectively). Both of these concretes reduced the radiation below design tolerances. It is recommended that the present front face biological shield be changed from I-S-220-P to I-S-245-P. With this change the NPR shield is adequate according to these calculations. The calculations reported here do not include leakage through penetration in the shield.

DISCUSSION

Methods of Calculation

The graphite reflector is treated as part of the shield so that the inner boundary conditions for the whole shield assembly are applied at the core graphite interface. The method used to calculate the thermal flux and neutron dose rate will not be fully discussed here other than to state briefly the basic concepts. A full report on the program can be found in Ref. (2). The distribution of high energy neutrons is determined by integrating over the fission sources in the reactor core using a point source kernel of the form

$$\frac{e^{-\Sigma_r(E)\cdot r}}{4\pi r^2}$$

where $\Sigma_r(E)$ is an energy dependent removal cross-section, defined for energies in the fission spectrum (1 to 18 Mev) by the formula

$$\Sigma_r(E) = \Sigma_{tot}(E) - \overline{\mu}(E)\,\Sigma_{sc}(E)$$

where $\Sigma_{sc}(E)$ is the elastic scattering cross-section, and $\Sigma_{tot}(E)$ is the total cross section, and $\overline{\mu}(E)$ is the mean cosine of the elastic scattering in the lab system. Once the distribution of penetrating neutrons is found, it is used to generate a source term for conventional multi-group diffusion theory. To illustrate the accuracy of the program, Figure V is included. The experimental results shown in Figure V were obtained in the DR shield facility. The calculated results are not normalized and boundary conditions are applied at the core-

[redacted]

DECLASSIFIED

graphite interface. The gamma dose rate was calculated using plain geometry and Taylor buildup factors. The program considers gammas generated in the core as well as those from (n, γ) reactions in the shield. Both thermal and epithermal captures are considered.

These calculations do not include radiation leakage through penetrations in the shield.

Given Conditions

The neutron fluxes incident on the NPR shield, shield thicknesses, and shield compositions were specified by W. L. Bunch and H. S. Davis. The neutron fluxes incident on the NPR shield assembly are given in Table I

TABLE I

Neutron Fluxes at the Boundary Between
The Core and Graphite Reflector

Flux	Energy	$n/cm^2/sec$
Fast	($E > 0.24$ Mev)	2.8×10^{13}
Resonance	(0.15 ev $< E < 0.24$ Mev)	5.1×10^{13}
Thermal	($E < 0.15$ ev)	2.8×10^{13}

The composition of the core and shield assembly along with shield thicknesses are given in Tables II and III.

TABLE II

Elemental Composition of the Shield Materials

			Thermal		Biological			
Element	Core	Reflector	Iron	Boron Steel	I-S-220-P	I-S-245-P	I-S-265-P	H-I-217-C
H	0.003	0.0	0.0	0.0	0.020	0.018	0.015	0.016
C	1.29	1.7	0.0	0.003	0.004	0.004	0.003	0.0
O	0.027	0.0	0.0	0.0	0.784	0.698	0.604	1.293
Mg	0.0	0.0	0.0	0.0	0.303	0.261	0.216	0.004
Al	0.0	0.0	0.0	0.0	0.023	0.022	0.020	0.013
Si	0.0	0.0	0.0	0.051	0.263	0.231	0.197	0.093
P	0.0	0.0	0.0	0.0008	0.0	0.0	0.0	0.0
S	0.0	0.0	0.0	0.0009	0.003	0.003	0.003	0.0003
B10	0.0	0.0	0.0	0.019	0.0	0.0	0.0	0.0
Ca	0.0	0.0	0.0	0.0	0.153	0.153	0.152	0.128
B11	0.0	0.0	0.0	0.082	0.0	0.0	0.0	0.0
Zr	0.28	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Mn	0.0	0.0	0.0	0.0544	0.011	0.014	0.011	0.0
Fe	0.0	0.0	7.20	7.559	1.816	2.376	2.932	2.002
U 235	0.0066	0.0	0.0	0.0	0.0	0.0	0.0	0.0
U 238	0.6934	0.0	0.0	0.0	0.0	0.0	0.0	0.0

[redacted]

[redacted]

TABLE III

Shield Assemblies for NPR

	Materials	Thickness (Inches)	Density (g/cm^3)
Front			
Reflector	Graphite	20	1.7
Thermal	Iron	7-7/8	7.20
Biological		40	
Case I	Concrete (I-S-220-P)		3.38
Case II	Concrete (I-S-245-P)		3.78
Case III	Concrete (I-S-265-P)		4.15
Side			
Reflector	Graphite	60	1.7
Thermal	Boron-steel	1-1/8	7.77
Biological	Concrete (H-I-217-C)	43	3.55
Top			
Reflector	Graphite	46-3/4	1.7
Thermal	Boron steel	1-1/8	7.77
Biological	Concrete (H-I-217-C)	43	3.55

The top biological shield assembly as given here is somewhat different from the actual conditions. The biological shield is actually 66 inches thick and has a density of about 3.0 g/cm^3. The exact composition will be determined later to utilize the existing stocks of heavy aggregate. The actual biological shield will make a better shield than the case stated above and the results obtained using the above conditions will be conservative.

Tolerances

"When the reactor is operating at maximum design power, radiation levels at the outside surface of the primary reactor shield shall be less than 12 rem/hr, including 2 rem/hr of neutrons and 10 r/hr of gamma radiation.

Neutron levels external to the primary shield during reactor operation shall be sufficiently low so that activation acquired by construction materials and piping will be less than 10 mr/hr". (3).

To estimate the activation of construction materials, the following assumptions were made:

1. A thermal neutron flux incident on an infinite slab of stainless steel one cm. thick. (Infinite-slab source)
2. The incident thermal flux is 2×10^7 $n/cm^2/sec$

[redacted]

The gamma flux at the surface of the plane is given by

$$\phi = \frac{S_v}{2U_s}\left[1 - E_2(b)\right]$$

where:

ϕ = gamma flux, photons/cm^2-sec^2
S_v = source strength per unit volume
U_s = attenuation coefficient of steel
X = thickness of the infinite plane
b = $X\ U_s$
E_2 = $\exp(-b) - b\ E_1(b)$
E_1 = $\int_b^{\infty} \frac{e^{-t}}{t}\,dt$

The results of the calculation are given in Table IV.

[redacted]

[redacted]

TABLE IV

ACTIVITIES IN STAINLESS STEEL AS SATURATION IN A THERMAL NEUTRON FLUX OF 2×10^{7} n/cm^{2}/sec

Element	% Disintegration	Gamma Energy Mev E	d/sec/cm^3 Sv	γ/cm^2/sec ϕ	Mev/cm^2/sec E x ϕ	Conv. factors $\frac{\text{Mr/hr}}{\text{Mev/cm}^2\text{/sec}}$	Dose Rate mr/hr	Total Dose Rate mr/hr	Percentage Weight of Element in Steel	Half-life
Iron	50 50	1.1 1.3	1.52×10^{3} 1.52×10^{3}	1.07×10^{3} 1.14×10^{3}	1.18×10^{3} 1.48×10^{3}	$.17\times10^{-3}$ $.17\times10^{-3}$	.20 .25	.45	65.3	46 d
Chromium	3	.32	4.56×10^{2}	2.23×10^{2}	7.15×10^{1}	$.20\times10^{-3}$	.02	.02	18.0	27.8 d
Nickel	29 43 29	.37 1.12 1.49	1.88×10^{3} 2.79×10^{3} 1.88×10^{3}	1.02×10^{3} 2.09×10^{3} 1.49×10^{3}	3.77×10^{2} 2.34×10^{3} 2.22×10^{3}	$.20\times10^{-3}$ $.177\times10^{-3}$ $.177\times10^{-3}$	.75 .41 .39	1.55	14	2.57 hr
Manganese	100 30 20	.822 1.77 2.06	4.52×10^{5} 1.36×10^{5} 9.04×10^{4}	2.99×10^{5} 1.16×10^{5} 7.72×10^{4}	2.46×10^{5} 2.05×10^{5} 1.59×10^{5}	$.20\times10^{-3}$ $.177\times10^{-3}$ $.177\times10^{-3}$	49.2 36.3 28.1	113.6	2	2.58 hr
Cobalt	100 100	1.17 1.33	1.48×10^{4} 1.48×10^{4}	1.11×10^{4} 1.11×10^{4}	1.29×10^{4} 1.47×10^{4}	$.177\times10^{-3}$ $.177\times10^{-3}$	2.28 2.60	5.88	0.026	5.28 yr
								121.5		

[redacted]

By changing the incident thermal neutron flux, the dose rates would change proportionately. Thus using stainless steel as a base, the thermal neutron tolerance is about 2 x 10^6 n/cm^2/sec to generate an activation dose rate of 10 mr/hr.

Front Primary Shield

The thickness of the front primary shield is governed by (1) the amount of neutron attenuation required to prevent activation of the piping on the front face and (2) the front primary shield has to reduce two gamma dose rate below the dose rate generated by gammas in the recirculating water. Figure I shows three different types of biological shields, I-S-220-P, I-S-245-P and I-S-265-P. Table V shows a comparison of the leakage through the different types of shields.

TABLE V

Comparison of Biological Shields on the Front Face of NPR

Type of Leakage	Design Tolerance	Type of Biological Shield I-S-220-P	I-S-245-P	I-S-265-P
Thermal Flux, n/cm^2/sec	2x10^6	3x10^3	1.3x10^3	8.0x10^2
Gamma Dose Rate, r/hr	10	14	3.5	1.3
Neutron Dose Rate, rem/hr	2	.5	0.4	0.3

The I-S-220-P will not meet the design tolerances and from an economic standpoint there is not too much incentive in going to the I-S-265-P. Thus the change from I-S-220-P to I-S-245-P is recommended on the bases of these calculations.

NPR Primary Shielding

Using the I-245-P for the front biological shield, a summary of the results is shown in Figures II, III and IV and in Table VI.

TABLE VI

Radiation Leakage Through the NPR Shield

Type of Leakage	Design Tolerance	Front	Side	Top
Thermal Flux, n/cm^2/sec	2x10^6	1.3x10^3	1.6x10^1	1.7x10^2
Gamma Dose Rate, r/hr	10	3.5	0.65	1.5
Neutron Dose Rate, rem/hr	2	0.4	0.004	0.04

[redacted]

Conclusions

From the results obtained using the new shielding code, the NPR shield meets the design tolerances if the biological shield on the front face is changed from I-S-220-P to I-S-245-P.

References

1. W. L. Bunch, Private Communication
2. E. G. Peterson, Computer Code For Design of Reactor Shields, HW-71173
3. H. S. Davis, Project CAI-816 105-N Design Criteria Primary Reactor Shield and Inlet Barrier Wall, HW-57010 RD

E G Peterson, Physicist

Reactor Technology Development Operation
HANFORD LABORATORIES OPERATION

EG Peterson:mw

[redacted]

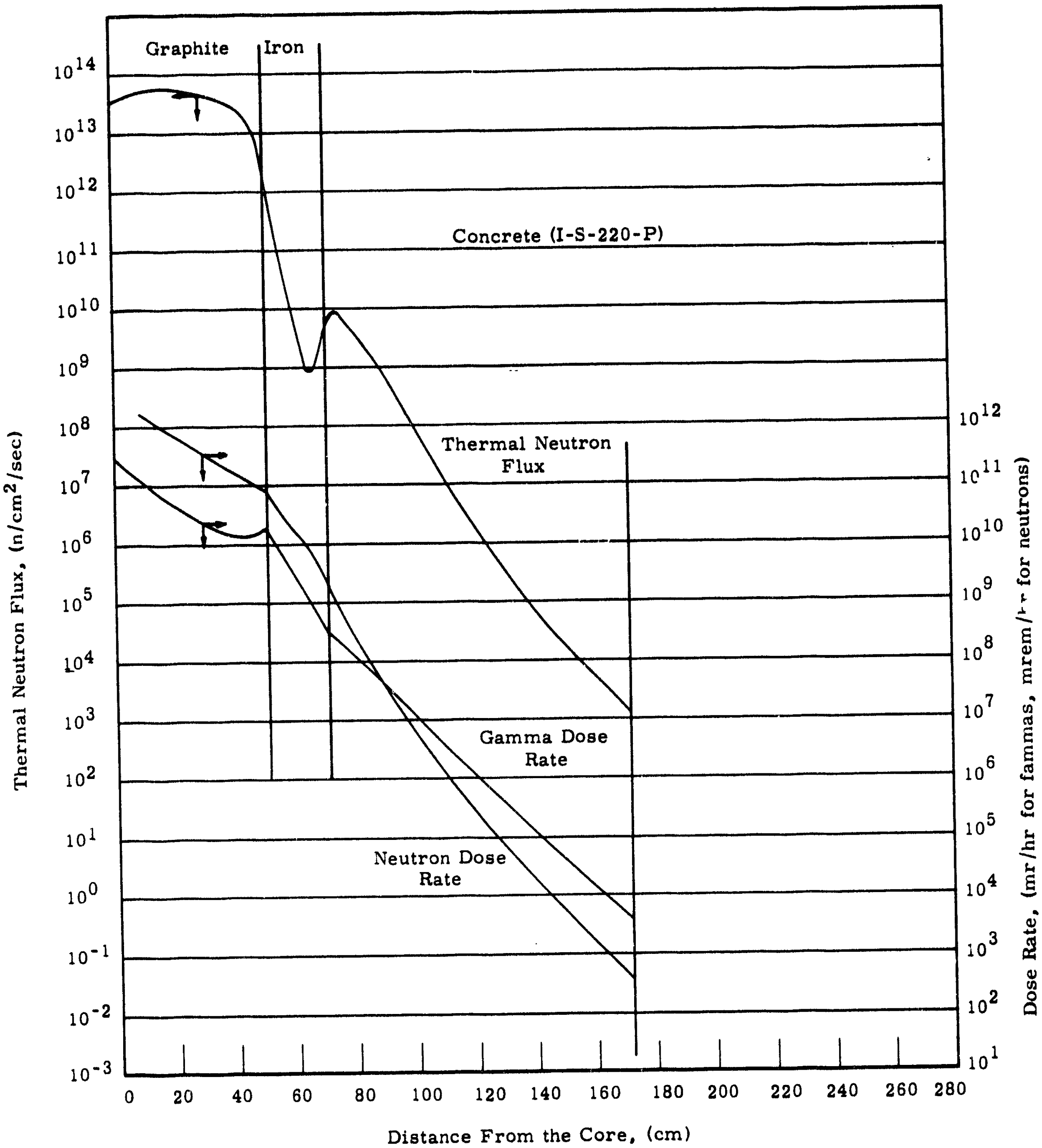

FIGURE 1

NPR Front Shield Calculations

[redacted]

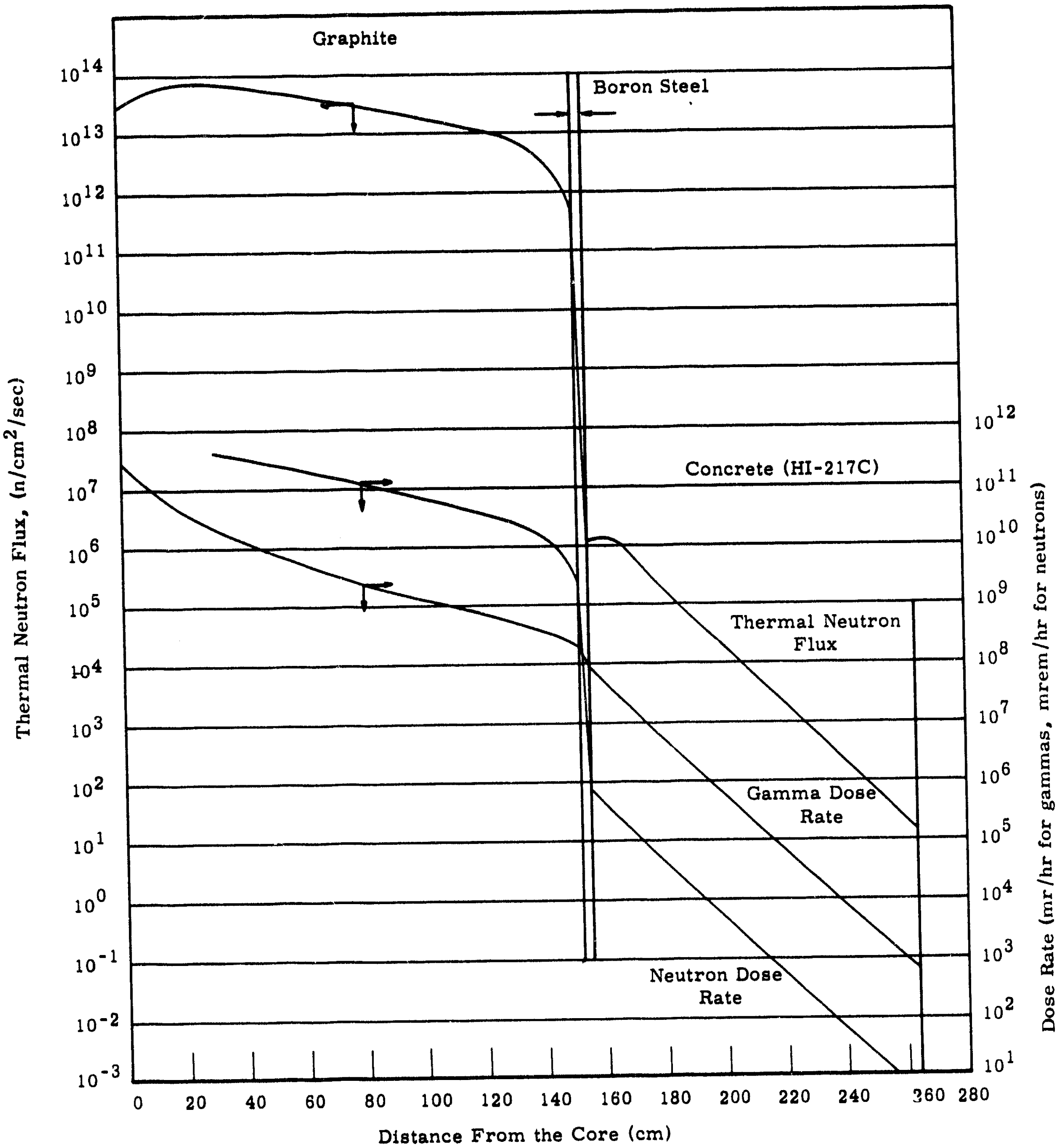

FIGURE 2

NPR Side Shield Calculations

[redacted]

[redacted]

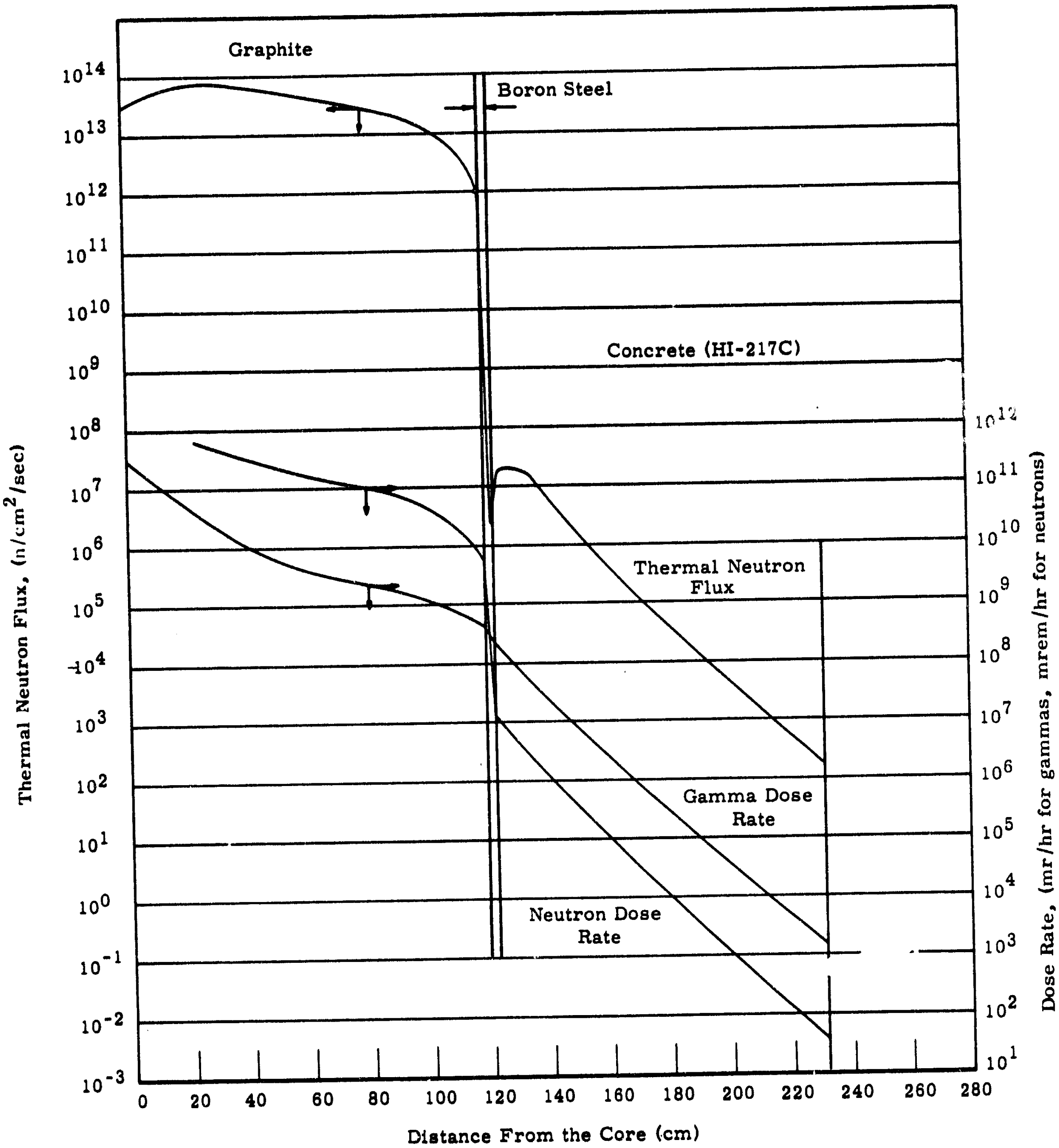

FIGURE 3

NPR Top Shield Calculations

[redacted]

DECLASSIFIED

[redacted]

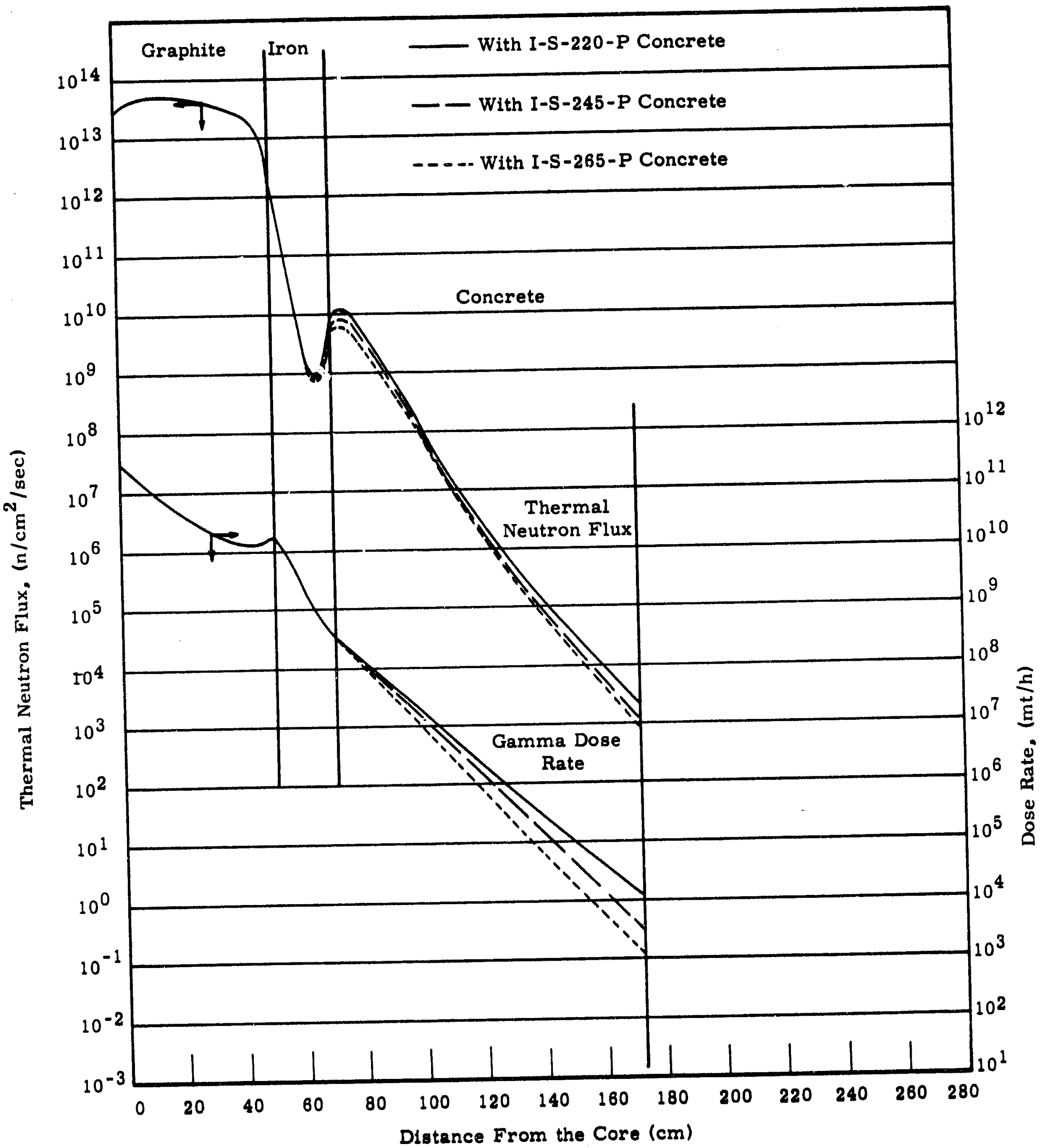

FIGURE 4

NPR Front Shield Calculations Using Various Types of Concretes

[redacted]

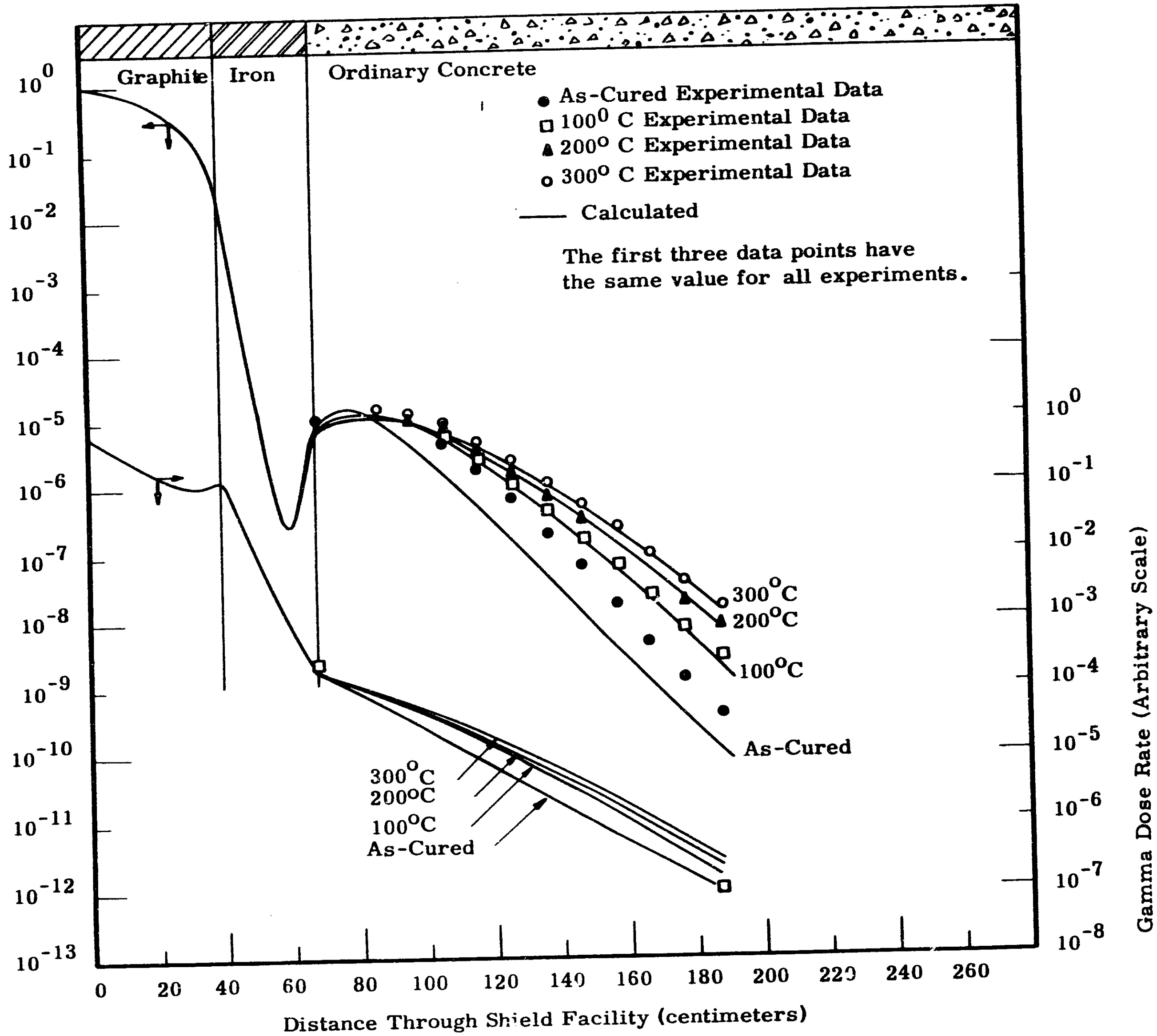
Graphite
Iron
Ordinary Concrete
As-Cured Experimental Data
100° C Experimental Data
200° C Experimental Data
300° C Experimental Data
Calculated
The first three data points have the same value for all experiments.
300°C
200°C
100°C
As-Cured
300°C
200°C
100°C
As-Cured
Thermal Neutron Flux (Arbitrary Scale)
Gamma Dose Rate (Arbitrary Scale)
Distance Through Shield Facility (centimeters)
0
20
40
60
80
100
120
140
160
180
200
220
240
260

FIGURE 5