AUG 6 1962

MASTER

Copy N^o 27

APAE Memo - 315

AEC Research and
Development Report
UC-81, Reactors - Power
[Special Distribution]

primary shield optimization survey for the PL-3 reactor

Contract No. AT[30-1]-2900
with U. S. Atomic Energy Commission
New York Operations Office

ALCO PRODUCTS, INC.
NUCLEAR POWER ENGINEERING DEPARTMENT

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.

APAE Memo-315

AEC Research and
Development Report
UC-81 Reactors, Power
(Special Distribution)

PRIMARY SHIELD OPTIMIZATION
SURVEY FOR THE
PL-3 REACTOR

By:

J. F. Scoles
A. N. Crouch

Approved by:

G. E. Humphries, Project Engineer

Issued: June 28, 1962

Contract No. AT(30-1)-2900
with U. S. Atomic Energy Commission
New York Operations Office

ALCO PRODUCTS, INC.
Nuclear Power Engineering Department
Post Office Box 414
Schenectady 1, N. Y.

AEC 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, expressed 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:

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, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

ALCO LEGAL NOTICE

This report was prepared by Alco Products, Incorporated in the course of work under, or in connection with, Contract No. AT(30-1)-2900, issued by U.S. Atomic Energy Commission, NYOO; and subject only to the rights of the United States, under the provisions of this contract, Alco Products, Incorporated makes no warranty or representation, express or implied, and shall have no liability with respect to this report or any of its contents or with respect to the use thereof or with respect to whether any such use will infringe the rights of others.

DISTRIBUTION

External
Copies

1 - 2

New York Operations Office
U. S. Atomic Energy Commission
c/o Alco Products, Inc.
1 Nott Street
Schenectady 5, New York

Attention: Resident Engineer, PL-3

3 - 4

New York Operations Office
U. S. Atomic Energy Commission
376 Hudson Street
New York 14, New York

Attention: Project Engineer, PL-3
Reactor Division

5 - 9

U. S. Atomic Energy Commission
Washington 25, D.C.

Attention: Chief, Water Systems Branch
(Army Reactors)
Division of Reactor Development
Mail Station F-311

10

U. S. Atomic Energy Commission
Washington 25, D.C.

Attention: Chief, Water Reactors Branch
Civilian Power
Division of Reactor Development
Mail Station F-311

11

U. S. Atomic Energy Commission
Chief, New York Patent Group
Brookhaven National Laboratory
Upton, New York

Attention: Harmon Potter

DISTRIBUTION (CONT'D)

External
Copies

- 12 U. S. Atomic Energy Commission
 Office of the General Counsel
 Washington 25, D.C.

 Attention: Mr. Roland Anderson
- 13 U. S. Atomic Energy Commission
 Idaho Operations Office
 P.O. Box 2108
 Idaho Falls, Idaho

 Attention: Director, Division of Military Reactors
- 14 U. S. Atomic Energy Commission
 Reports and Statistics Branch
 Division of Reactor Development
 Washington 25, D.C.
- 15 - 19 Bureau of Yards and Docks
 Navy Department
 Washington 25, D.C.

 Attention: Director, Nuclear Power Division
 Code E-300
- 20 Office of the Chief of Engineers
 Department of the Army
 Building T-7
 Washington 25, D.C.

 Attention: Chief, Projects Branch
 Nuclear Power Division
- 21 Nuclear Power Field Office
 U. S. Army Engineer Reactors Group
 Fort Belvoir, Virginia

 Attention: Chief, Nuclear Power Field Office

DISTRIBUTION (CONT'D)

External
Copies

- 22 Sundance Air Force Radar Station
P.O. Box 80
Sundance, Wyoming

Attention: AEC Representative, FM-1
- 23 Nuclear Power Field Office
U. S. Army Engineer Reactors Group
Fort Belvoir, Virginia

Attention: O.I.C. SM-1
- 24 Chief, U. S. Army Reactors Group
Fort Greeley, Alaska
APO 733
Seattle, Washington

Attention: O.I.C. SM-1A
- 25 Commanding Officer
U. S. Army Polar Research and Development Center
Fort Belvoir, Virginia

Attention: Nuclear Power Officer
- 26 Commander
Air Force Special Weapons Center (AFSWC)
Albuquerque, New Mexico

Attention: SWVP
- 27 - 29 Office of Technical Information Extension
P.O. Box 62
Oak Ridge, Tennessee
- 30 Union Carbide Nuclear Corporation
Oak Ridge National Laboratory
Y-12 Building 9704-1
P.O. Box "Y"
Oak Ridge, Tennessee

Attention: Mr. L.D. Schaffer

DISTRIBUTION (CONT'D)

External
Copies

- 31 Director
 U. S. Army Cold Regions
 Research and Engineering Laboratories
 P.O. Box 282
 Hanover, New Hampshire

 Attention: Dr. R. W. Gerdel
- 32 Advanced Technology Laboratories
 Division of American-Standard
 369 Whisman Road
 Mountain View, California

 Attention: Director, Research and Development
- 33 Aerojet General Nuclenoics
 Military Products Division
 Box 77
 San Ramon, California

 Attention: Manager
- 34 The Babcock & Wilcox Company
 Atomic Energy Division
 1201 Kemper Street
 P.O. Box 1260
 Lynchburg, Virginia

 Attention: Manager
- 35 Combustion Engineering, Inc.
 Nuclear Division
 Prospect Hill Road
 Windsor, Connecticut

 Attention: Mr. J. B. Anderson
- 36 General Atomic
 Division of General Dynamics
 P.O. Box 608
 San Diego 12, California

 Attention: Assistant Laboratory Director

DISTRIBUTION (CONT'D)

External
Copies

- 37 General Electric Company
 Atomic Power Equipment
 2151 South First Street
 San Jose 12, California

 Attention: Manager, Engineering
- 38 Lockheed-Georgia Company
 Lockheed Nuclear Products
 Marietta, Georgia

 Attention: Director, Nuclear Laboratories
- 39 The Martin Company
 P.O. Box 5042
 Middle River, Maryland

 Attention: AEC Contract Document Custodian
- 40 United Nuclear Corporation
 Development Division
 5 New Street
 White Plains, N. Y.

 Attention: Library Custodian
- 41 Westinghouse Electric Corporation
 Atomic Power Department
 P.O. Box 355
 Pittsburgh 30, Pennsylvania

 Attention: Director of Projects
- 42 Internuclear Company
 7 North Brentwood Blvd.
 Clayton 5, Missouri

 Attention: Mr. O.J. Elgert

DISTRIBUTION (CONT'D)

External
Copies

43 Jackson and Moreland Company
Part Square Bldg.
Boston 16, Massachusetts

Attention: Mr. Harold Vann

Internal
Copies

44 R. Bell
45 G.E. Humphries
46 C. Klote
47 W. Enright
48 J.F. Haines
49 E.L. Cofrances
50 G. Vodopivec
51 W. Farmer
52 P.E. Bobe
53 A. Dromerick
54 H.F. Uberto
55 A.N. Crouch
56 J. Englima
57 B.E. Fried
58 F.T. Matthews
59 R.E. Neville
60 W.T. Williams
61 PL-3 File
62 - 64 Schenectady Operations File
65 Nuclear Power Division
Allis-Chalmers Mfg. Co.
Post Office Box 512
Milwaukee 1, Wisconsin

Attention: Library

66 Allis-Chalmers Mfg. Co.
508 Kennedy St., N.W.
Washington 11, D.C.

Attention: Library

ABSTRACT

A detailed study of four reactor and shield configurations is made. Two basic reactor types, the boiling water and pressurized water reactors are considered. Shield materials of lead-water and iron-water were used with varying thicknesses for determining the optimum shield configuration for the PL-3 reactor. Also presented is a survey of available shielding codes.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT -----	ix
SUMMARY -----	xix
1.0 INTRODUCTION -----	1-1
1.1 Objectives of the Survey -----	1-1
1.2 Reactor Configurations Investigated -----	1-1
1.3 Shield Configurations Investigated -----	1-1
2.0 RADIATION LEAKAGE RESTRICTIONS -----	2-1
2.1 Shutdown Dose Rates -----	2-1
2.2 Radiation Levels in Camp and Plant Tunnels -----	2-5
2.3 Neutron Transmission to One Mile -----	2-12
3.0 MACHINE CODES USED -----	3-1
4.0 THE HIGHLY ENRICHED BWR CONCEPT -----	4-1
4.1 Description of the Configuration -----	4-1
4.2 Core Nuclear Constants -----	4-2
4.3 Primary Shield Nuclear Constants -----	4-2
4.4 Neutron Distribution -----	4-3
4.5 Gamma Dose Rate -----	4-3
4.6 Output -----	4-3
5.0 TYPE 3 CORE -----	5-1
5.1 Description of the Configuration -----	5-1
5.2 Core Nuclear Constants -----	5-2
5.3 Neutron Distribution -----	5-2
5.4 Gamma Dose Rate -----	5-5

TABLE OF CONTENTS

	<u>Page</u>
6.0 TYPE 2 CORE -----	6-1
6.1 Description of the Configuration -----	6-1
6.2 Core Nuclear Constants -----	6-7
6.3 Neutron Distribution -----	6-7
6.4 Gamma Dose Rate -----	6-7
7.0 TYPE 2 CORE - STEEL-WATER SHIELD CONFIGURATION -----	7-1
7.1 Description of Configuration -----	7-1
7.2 Gamma Dose Rate Distribution for the Type 2 Core Steel-Water Shield Configuration -----	7-2
8.0 TYPE 2 CORE - REFERENCE CONFIGURATION -----	8-1
8.1 Description of Configuration -----	8-1
8.2 Neutron Distribution -----	8-1
8.3 Gamma Dose Rate Distribution -----	8-5
9.0 CONCLUSIONS AND RECOMMENDATIONS -----	9-1
10.0 REFERENCES -----	10-1
APPENDIX A - Survey of Shielding Machine Codes -----	A-1

LIST OF ILLUSTRATIONS

<u>Figure</u>	<u>Title</u>	<u>Page</u>
2.1	Decay of Dose Rates from Fission Products	2-2
2.2	Decay of Dose Rate from Activated Stainless Steel	2-2
2.3	Operating Gamma Leakage from Primary Shield	2-6
2.4	Air Scatter Dose on Snow Surface Over Byrd Station	2-8
2.5	Dose Rates on Snow Surface Over Byrd Station Due to Neutron Captures in Air	2-11
4.1	Neutron Flux Traverse - BWR 5 x 5 Core	4-4
4.2	Gamma Dose Rate Vs Distance - BWR - 0 In. Pb, Shutdown	4-6
4.3	Gamma Dose Rate Vs Distance - BWR - 1-1/2 In. Pb, Shutdown	4-7
4.4	Gamma Dose Rate Vs Distance - BWR - 4 In. Pb, Shutdown	4-8
5.1	Neutron Flux Traverse - 5 x 5 PWR - SM-2 Type 3 Core	5-7
5.2	Gamma Dose Rate Vs Distance - 5 x 5 PWR - 0 In. Pb, Shutdown	5-7
5.3	Gamma Dose Rate Vs Distance - 5 x 5 PWR - 1-1/2 In. Pb, Shutdown	5-8
5.4	Gamma Dose Rate Vs Distance - 5 x 5 PWR - 4 In. Pb, Shutdown	5-8
6.1	Neutron Flux Traverse Type 2 Core	6-8
6.2	Gamma Dose Rate Vs Distance - 7 x 7 PWR - 0 In. Pb, Shutdown	6-10
6.3	Gamma Dose Rate Vs Distance - 7 x 7 PWR - 1-1/2 In. Pb, Shutdown	6-10
6.4	Gamma Dose Rate Vs Distance - 7 x 7 PWR - 4 In. Pb, Shutdown	6-11
6.5	Gamma Dose Rate Vs Distance - 7 x 7 PWR - 0 In. Pb, Operating	6-11

LIST OF ILLUSTRATIONS (CONT'D)

<u>Figure</u>	<u>Title</u>	<u>Page</u>
6.6	Gamma Dose Rate Vs Distance - 7 x 7 PWR - 1-1/2 In. Pb, Operating	6-12
6.7	Gamma Dose Rate Vs Distance - 7 x 7 PWR - 4 In. Pb, Operating	6-12
7.1	Gamma Dose Rate Vs Distance - 7 x 7 PWR - 2 In. Fe, 1 Iron Shield Ring, Shutdown	7-3
7.2	Gamma Dose Rate Vs Distance - 7 x 7 PWR - 6 In. Fe, 3 Iron Shield Rings, Shutdown	7-3
7.3	Gamma Dose Rate Vs Distance - 7 x 7 PWR - 8 In. Fe, 4 Iron Shield Rings, Shutdown	7-4
8.1	Neutron Flux Traverse - 7 x 7 PWR Reference Core	8-6
8.2	Gamma Dose Rate Vs Distance, Operating Condition - 7 x 7 Reference Core	8-7
8.3	Gamma Dose Rate Vs Distance, 2.4 Hr After Shutdown - 7 x 7 Reference Core	8-7
9.1	Lead Shield Thickness Vs Shield Radius - To Give 90 Mr/Hr at 2.4 Hr after Shutdown - BWR Configuration	9-2
9.2	Lead Shield Thickness Vs Shield Radius - To Give 90 Mr/Hr at 2.4 Hr after Shutdown - 5 x 5 PWR Configuration	9-2
9.3	Lead Shield Thickness Vs Shield Radius - To Give 90 Mr/Hr at 2.4 Hr after Shutdown - 7 x 7 PWR Configuration	9-3
9.4	Steel Shield Thickness Vs Shield Radius - To Give 90 Mr/Hr at 2.4 Hr after Shutdown - 7 x 7 PWR Configuration	9-3

LIST OF DRAWINGS

<u>Dwg.</u>	<u>Title</u>	<u>Page</u>
AEL-705	Reactor Vessel Concept A-1	5-3
AEL-728	Reactor Vessel Detail Concept A-3	6-3
AEL-732	Primary System Complex A-3 (PWR)	6-5
AEL-762	Primary System PWR Concept	8-3

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
2.1	R(t) Calculation	2-4
4.1	5 x 5 BWR Reactor Configuration	4-1
4.2	BWR Nuclear Constants	4-2
4.3	Shield Nuclear Constants	4-2
4.4	Output from Machine Calculations for the BWR Configuration	4-5
5.1	Type 3 Core Radial Configuration	5-1
5.2	Type 3 Core Nuclear Constants	5-2
5.3	Type 3 Core Gamma Dose Rate Results	5-6
6.1	Type 2 Core Radial Configuration	6-1
6.2	Type 2 Core Nuclear Constants	6-7
6.3	Type 2 Core Gamma Dose Rate Results	6-9
7.1	Type 2 Core Steel-Water Shield Configuration	7-1
7.2	Type 2 Core Steel-Water Gamma Dose Rate Results	7-2
8.1	Type 2 Core Reference Configuration	8-1

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

SUMMARY

In a study to optimize shield design for the PL-3 reactor, calculations were performed to obtain gamma and neutron dose rates for various thicknesses of lead, steel and water shield materials and for various configurations. The cores investigated were the BWR 5 x 5 array, PWR Type 3 (SM-2) 5 x 5 array, and PWR Type 2 (SM-1) 7 x 7 array for carbon steel and stainless pressure vessels, with the latter the reference design.

The decay dose rates outside the shielding due to the core contribution and due to saturated activation of the stainless steel are calculated.

An analysis of radiation levels in the camp and plant tunnels, mainly from radiation leakage through the primary shield and from air scatter gamma doses is included. Gamma shielding will be tailored so that radiation leakage on the primary shield surface after shutdown is uniform.

The Alco ROC primary shielding codes, which determine dose rates both for reactor operating conditions and at shutdown, were programmed and run on an IBM-650 computer. For the ROC code a normalized neutron flux input was provided by the VALPROD modified two-group multi-region code, which consists of fast and thermal core constants, axial buckling and geometry values. A series of plots are developed for gamma dose vs distance from core centerline.

Conclusions indicated the shield radius most compatible with other design considerations is 7.5 ft consisting of 5.8 in. of steel or 3.8 in. of lead for the Type 2 7 x 7 core. This would reduce shutdown dose to 90 mr/hr at 2.4 hr after shutdown and meet design criteria of 100 mr/hr at 2 hr after shutdown from full power run.

Lead is recommended rather than steel as the gamma shield material since steel incurs a weight penalty due to the added material required, has higher operating gamma doses than lead-water and has problems associated with activation of the steel. A lead shield would permit weight saving during air shipment, and result in a more compact design, as indicated by curves of lead shield thickness vs shield radius for the 90 mr/hr dose rate requirement.

1.0 INTRODUCTION

1.1 OBJECTIVES OF THE SURVEY

The study was initiated to gain a basis for the PL-3 primary shield design by comparison of various shielding configurations, and to evaluate heavy materials such as lead and steel for use as primary gamma shielding.

Gamma dose rates were determined for various combinations of neutron and gamma primary shield materials. Machine calculations made it possible to obtain the dose rate contributed by the reactor core and the dose rate contributed considering the shield as a source. These are reported separately and totalled.

For the purposes of this study, the shield is defined as that portion of the reactor configuration beyond the reactor core, which includes such components as the pressure vessel and support ring as parts of the shielding. Lead and steel were materials studied. All dose rates were calculated on the horizontal midplane through the reactor core.

Shielding codes which could be used for further design analysis studies were surveyed and are reported in the Appendix.

1.2 REACTOR CONFIGURATIONS INVESTIGATED

Comparing the effectiveness of different shielding combinations, the present survey considered two reactor types.- the boiling water and the pressurized water reactor. The boiling water reactor investigated employed a highly enriched plate type core. One pressurized water reactor consisted of a Type 3 core (1) with a 5 x 5 fuel element array. The other PWR type configurations has a 7 x 7 array of Type 2 elements (SM-1 type core). The PL-3 reference configuration is based upon the 7 x 7 core in a stainless steel pressure vessel 2-1/2 in. thick.

This report hereafter will refer to these various reactor configurations as the BWR, the Type 3, the Type 2, and the reference configuration.

1.3 SHIELD CONFIGURATIONS INVESTIGATED

Gamma shield materials investigated were lead (density = 11.4 gm/cm^3) and iron (density = 7.8 gm/cm^3). Both materials were immersed in water and placed as close as possible to the reactor pressure vessel.

The thickness of lead varied up to 4 in. Cases were also run with the lead removed from the shield tank, leaving an all-water shield. For the cases using steel for shielding, one, three, and four 2-in. rings immersed in water were considered. The water annulus between successive rings was taken as 1 in. wide. The steel-water configuration is the same type as in the primary shield of the SM-1 reactor at Fort Belvoir, Va.

The reference shield selected was lead, 2-1/2 in. thick. This was used with the reference configuration to obtain gamma dose rate values.

2.0 RADIATION LEAKAGE RESTRICTIONS

2.1 SHUTDOWN DOSE RATES

The radiation leakage from the primary shield 2 hr after shutdown from an extended full power run is restricted by the technical provisions of the PL-3 Contract to 100 mrem/hr. Dose rate contributions from sources external to the primary shield are additive and not included in the 100 mrem/hr limitation.

Extensive files are available for the ROC⁽⁶⁾ code for 0.1 day shutdown time, therefore, for the survey reported here, the design point was taken at 2.4 hr shutdown time. In the reference design configuration the dose outside the primary shield at 2.4 hr after shutdown is calculated in Section 8 to be composed of 80 percent from core contributions and 20 percent from shield activation contributions if an infinite operation time at full power is assumed.

The decay dose rate outside the shielding due to the core contribution is given in Fig. 2.1. The dose rate from the core at relatively short shutdown times is independent of operating time and is due entirely to the fission product activities in the core. The analysis from which this curve is derived is presented in Ref. (2).

The decay dose rate outside of the shielding due to saturated activation of the stainless steel is presented in Fig. 2.2. The decay is related to a unit dose at 2.0 hr after shutdown. Details of the assumptions and analysis are presented in the following paragraphs.

The decay of the dose rate due to stainless steel activation was evaluated on the basis of data reported in Ref. (3). Assumptions for these calculations were:

1. The activation of stainless steel is the same as that which would occur in Type 304 stainless steel reported in Ref. (3).
2. The proportionality of the activation cross sections for the competing activation reactions remains constant when the activating neutron spectrum is changed from that in the X-10 reactor to that in the PL-3 vessel wall and shielding.
3. The 1.15, 1.17, 1.2, and 1.33 Mev gammas produced as a result of the activation are attenuated equally.
4. The shielding outside the activated steel regions is sufficiently thick that the 0.32 Mev gammas may be neglected.

Figure 2.1. Decay of Dose Rates from Fission Products Figure 2.2. Decay of Dose Rate from Activated Stainless Steel

The third assumption above implies that the dose rate is proportional to the source strength for the four gammas identified. Therefore,

$$D_i \propto S_i = K \Sigma_i e^{-\lambda_i t}$$

Where,

D_i = Dose rate due to the i-th induced activity

S_i = Source of the i-th induced activity

Σ_i = Macroscopic cross section for productions of the i-th induced activity

λ_i = Decay constant for the i-th activity

t = Reactor shutdown time

K = Proportionality constant independent of i or t.

Therefore,

$$D_T \propto K \Sigma_1 e^{-\lambda_1 t} + \Sigma_2 e^{-\lambda_2 t} + \Sigma_3 e^{-\lambda_3 t}$$

Where,

D_T = Dose rate from induced activities 1, 2, and 3

Since the decay of the stainless steel activation dose rate is the object of interest, the decay factor, $R(t)$, is defined as follows,

$$R(t) = \frac{D_T(t)}{D_T(2 \text{ hr})}$$

The calculation of $R(t)$ is detailed in Table 2-1.

The decay from 2.0 to 2.4 hr for stainless steel activity is 0.95. The corresponding decay for fission products is 0.91. The composite dose rate decay for the reference configuration is then $0.8 \times 0.91 \times 0.2 \times 0.95 = 0.92$. A dose rate of 90 mrem/hr was selected as the target dose rate for 2.4 hr decay time for these survey calculations.

TABLE 2-1
R(t) CALCULATION

<u>t_(hr)</u>	<u>Isotope</u>	<u>$\lambda(\text{hr}^{-1})$</u>	<u>λt</u>	<u>$e^{-\lambda t}$</u>	<u>Σ</u>	<u>$\Sigma e^{-\lambda t}$</u>	<u>R</u>
2	1	0.277	0.554	0.575	8.0 (-4)	4.60 (-4)	1.0
	2	6.93 (-4)	1.39 (-3)	0.999	3.6 (-5)	3.60 (-5)	
	3	1.49 (-5)	2.98 (-5)	1.0	5.0 (-4)	5.0 (-4)	
2.4	1	0.277	0.664	0.515	8.0 (-4)	4.12 (-4)	0.95
	2	6.93 (-4)	1.67 (-3)	0.998	3.6 (-5)	3.59 (-5)	
	3	1.49 (-5)	3.57 (-5)	1.0	5.0 (-4)	5.0 (-4)	
4	1	0.277	1.108	0.318	8.0 (-4)	2.54 (-4)	0.79
	2	6.93 (-4)	2.77 (-3)	0.997	3.6 (-5)	3.59 (-5)	
	3	1.49 (-5)	5.96 (-5)	1.0	5.0 (-4)	5.0 (-4)	
8	1	0.277	2.22	0.109	8.0 (-4)	8.69 (-5)	0.62
	2	6.93 (-4)	5.54 (-3)	0.994	3.6 (-5)	3.58 (-5)	
	3	1.49 (-5)	1.19 (-4)	1.0	5.0 (-4)	5.0 (-4)	
16	1	0.277	4.43	0.012	8.0 (-4)	9.52 (-6)	0.55
	2	6.93 (-4)	1.11 (-2)	0.988	3.6 (-5)	3.56 (-5)	
	3	1.49 (-5)	2.38 (-4)	1.0	5.0 (-4)	5.0 (-4)	
24	2	6.93 (-4)	1.66 (-2)	0.983	3.6 (-5)	3.54 (-5)	0.54
	3	1.49 (-5)	3.58 (-4)	1.0	5.0 (-4)	5.0 (-4)	
48	2	6.93 (-4)	3.33 (-2)	0.968	3.6 (-5)	3.49 (-5)	0.54
	3	1.49 (-5)	7.15 (-4)	0.999	5.0 (-4)	5.0 (-4)	
96	2	6.93 (-4)	6.65 (-2)	0.936	3.6 (-5)	3.37 (-5)	0.54
	3	1.49 (-5)	1.43 (-3)	0.999	5.0 (-4)	5.0 (-4)	
192	2	6.93 (-4)	1.33 (-1)	0.876	3.6 (-5)	3.16 (-5)	0.53
	3	1.49 (-5)	2.86 (-3)	0.997	5.0 (-4)	4.9 (-4)	

Numbers in parenthesis indicate multiplicative powers of 10.

2.2 RADIATION LEVELS IN CAMP AND PLANT TUNNELS

Leakage from the primary shield contributes a significant portion to the radiation levels in the plant and camp. Direct penetration of radiation through the snow tunnel walls is a negligible contributor in the nearby tunnels. Gammas from secondary interactions in air account for a significant dose rate at the site of the plant tunnels. Important components of the air scatter gamma dose are scattering of gamma radiation, gammas resulting from air capture of neutrons, and gammas from inelastic scattering of fast neutrons. An experimental and analytical study by Convair (Ref. (4)) provides a basis for simple analysis of these effects.

The configuration of the PL-3 tunnels is not well established at present. However, it is desired that no snow cover over the reactor tunnel be required for shielding purposes. To permit calculations to be made, it will be assumed that the reactor tunnel is of rectangular cross section, 27 ft wide, 58 ft long, and has the reactor center located at a depth of 40 ft from the finished snow surface. The reactor is located in the center of this tunnel, surrounded by a 15 ft diam primary shield. Gamma shielding will be tailored such that radiation leakage after shutdown is uniform on the primary shield surface to a height at which lead shielding is not required. Radiation is scattered through the top surface of the open tunnel and will be assumed to have escaped into air for scattering capture.

If we assume the operating gamma leakage from the shield is the same form as the shutdown leakage, it will be of the form illustrated in Fig. 2.3. The operating dose through a shield is that which will meet the shutdown requirement mentioned earlier. About one-third the leakage which escapes from the side of the primary shield is in a radial direction from the core at greater than 124 in. above the core midplane, shield region 1, and will escape from the tunnel unscattered. All of that which escaped from the top of the primary shield tanks, shield region 2, will escape from the tunnels unscattered. The preponderance of the gamma leakage is from the side of the primary shield, shield region 3, and must scatter at least once from the snow tunnel walls to escape. The integrated gamma escapes from the reactor tunnel due to gamma shield penetration in each of the regions is indicated below:

Shield Region 1:

$$\begin{aligned}
 D_{sg}(1) &= \frac{3.4 \times 10^5 \times 15\pi}{3} \int_{10.3}^{15} e^{-1.075h} dh \text{ rem-ft}^2/\text{hr} \\
 &= \frac{3.4 \times 10^5 \times 15\pi}{3 \times 1.075} \left[1.54 \times 10^{-5} - 1.03 \times 10^{-7} \right] \\
 &= 50 \text{ rem-ft}^2/\text{hr}
 \end{aligned}$$

Figure 2.3. Operating Gamma Leakage from Primary Shield

Shield Region 2:

$$\begin{aligned} D_{sg}(2) &= 0.32 \times \pi \int_0^{7.5} r e^{-1.26r} dr \\ &= 0.8 \text{ rem-ft}^2/\text{hr} \end{aligned}$$

Shield Region 3:

$$\begin{aligned} D_{sg}(3) &= 8.60 \times 15 \pi \times 16.34 \times 10^5 \times 15 \pi \int_8^{10-1/3} e^{-1.075h} dh \\ &= 4.5 \times 10^5 S \text{ rem-ft}^2/\text{hr} \end{aligned}$$

Where S = probability of surviving a scattering from the snow wall and escaping from the tunnel.

If a scattering point on the snow wall is selected at 3 ft above core mid-plane and isotropic re-direction of the reflected radiation is assumed, 20 percent of the scattered radiation will escape from the tunnel. Applying well known formulas for scattering from a thick slab, S is found to be $(20\%) \times 19 \times 10^{-3}$.

$$D_{sg}(3) = 1.6 \times 10^3 \text{ rem-ft}^2/\text{hr}$$

Total gamma leakage from the reactor tunnel is:

$$\begin{aligned} D_{sg} &= D_{sg}(1) + D_{sg}(2) + D_{sg}(3) \\ &= 50 + 0.8 + 1650 = 1.7 \times 10^3 \text{ rem-ft}^2/\text{hr} \end{aligned}$$

From Ref. (4) the following relationships were obtained:

$$D_1 (20 \text{ ft}) = 3.4 \times 10^{-2} D_{sg} = 58 \text{ mrem/hr}$$

$$D_1 (80 \text{ ft}) = 4.1 \times 10^{-3} D_{sg} = 7.0$$

$$D_1 (467 \text{ ft}) = 1.4 \times 10^{-4} D_{sg} = 0.23$$

$$D_1 (867 \text{ ft}) = 2.1 \times 10^{-5} D_{sg} = 0.035$$

The resultant dose rates on the snow surface are plotted in Fig. 2.4.

The neutron fluxes escaping from the primary tunnel will be composed almost entirely of those neutrons leaking from the side of the primary shield and scattering out of the tunnel. The increased effectiveness of steel for neutron attenuation deep within the primary shield approximately compensates for the loss of hydrogenous material in the insulation, in air spaces in the primary shield, and in the reduced density primary water. Hence, the primary shield approximates a right circular cylinder with the reactor located on the centerline. Leakage out the top and bottom of the cylinders is sufficiently small that it may be neglected for this analysis. From ORNL-2518, ⁽⁵⁾ the thermal neutron flux, ϕ_{th} is found to be 4.8×10^2 n/cm²-sec, and the fast neutron dose rate is found to be 3.2×10^{-1} ergs/gm (tissue)-hr.

Since at least one scattering must occur before any of these neutrons which leave the site are reduced in energy by at least the following amounts,

$$\frac{E}{E_0} = \frac{1 + 2\cos \theta + 1}{4} \approx 0.4$$

if the scattering occurs from hydrogen; or

$$\frac{E}{E_0} = \frac{16^2 + 2 \cos \theta + 1}{17^2} \approx 0.9$$

if the scattering is an elastic scattering from oxygen. Only a very small percentage of the fast neutrons incident upon the snow wall will be reflected with sufficient energy to cause a subsequent inelastic scattering in air. Inelastic scatterings with oxygen in the snow wall will result in very low energy reflected neutrons. Of the neutrons elastically scattered from the snow wall and above the 6 Mev energy inelastic scattering threshold, 25 percent are scattered by oxygen in the snow wall. Of these, only 20 percent will escape from the tunnel on first scattering. Therefore, the gamma dose rate due to inelastic scattering of neutrons in air will be small.

The neutrons incident upon the snow wall will be reflected with about 82 percent efficiency. That is, the albedo for an infinite water reflector is 0.82. Also, as in the above analysis for gammas, 20 percent of the neutrons which scatter once will escape. Or, if the same albedo holds true for subsequent scatterings and the probability of escape from the tunnel is taken equal to that fraction of the tunnel walls open to the atmosphere, then the escape fraction from successive scatterings is:

1st Scattering	$= 0.82 \times 0.20\phi$
2nd Scattering	$= (0.82)^2 \times 0.80 \times 0.14\phi$
3rd Scattering	$= (0.82)^3 \times 0.80 \times 0.14\phi$
n-th Scattering	$= (0.82)^n \times 0.80 \times (0.86)^{n-2} \times 0.14\phi$

Where

ϕ = total neutron leakage from the primary shield.

Totalling these, we see that 43% of the neutrons leaking from the primary shield will eventually make their way out of the primary tunnel.

The neutron leakage from the side of the primary shield will be of the form

$$\phi' = \frac{\phi_0}{2} \left(1 \cos \frac{\pi Z}{3.6} \right)$$

Where

Z = feet from core midplane

in the region near the core midplane where the greatest portion of the leakage occurs. The total leakage from the shield is

$$\phi = 15\pi \int_{-3.6}^{3.6} \frac{\phi_0}{2} \left(1 \cos \frac{\pi Z}{3.6} \right) dZ = 1.7 \times 10^2 \phi_0$$

If ϕ_0 is taken equal to $\phi_{th} + \phi_f$ then,

$$\phi_0 = 4.8 \times 10^2 + 2.5 \times 10^2 = 7.3 \times 10^2 \text{ n/cm}^2\text{-sec}$$

or

$$\phi_0 = 1.7 \times 10^2 \times 7.3 \times 10^2 \times (30.48)^2 = 1.15 \times 10^8 \text{ n/sec}$$

is the rate at which neutrons are escaping the tunnel. From Ref. (4) the following relationships are found

$$D_2 (20) = 1.4 \times 10^{-10} \phi = 1.6 \times 10^{-2} \text{ mrem/hr}$$

$$D_2 (80) = 3.5 \times 10^{-11} \phi = 4.0 \times 10^{-3} \text{ mrem/hr}$$

$$D_2 (467) = 3.5 \times 10^{-13} \phi = 4.0 \times 10^{-4} \text{ mrem/hr}$$

$$D_2 (867) = 2.1 \times 10^{-13} = 2.4 \times 10^{-4} \text{ mrem/hr}$$

The dose rates, plotted in Fig. 2.5, are roughly three orders of magnitude below the dose rates from air scattering of gamma radiation; however, these results will serve to show the neutron leakage effects upon camp gamma dose rates. The total air scatter dose is essentially that shown in Fig. 2.4.

Figure 2.5. Dose Rates on Snow Surface Over Byrd Station Due to Neutron Captures in Air

2.3 NEUTRON TRANSMISSION TO ONE MILE

Modifications to the PL-3 technical provisions have eliminated neutron flux at one mile as a basic limiting criterion of design. However, this flux is estimated on the basis of bomb blast measurements as follows.

The total number of neutrons escaping from the reactor tunnel from the above analysis is 1.15×10^8 n/sec. The neutron flux at one mile,

ϕ (1 mile), resulting from this leakage flux is,

$$\phi (1 \text{ mile}) = \frac{1.15 \times 10^8}{2\pi (1610)^2} e^{-\Sigma_a 1610} \times 60 = 0.29 \text{ n/m}^2\text{-min}$$

Where,

Σ_a = effective removal cross section for neutrons in air from bomb blast data = 0.0045 m^{-1} .

Thus, even with margin allowed for additional neutron streaming at shield penetrations during reactor operation, it appears that with a 15 ft diam primary shield, the neutron flux at 1 mile will remain below $1 \text{ n/m}^2\text{-min}$.

3.0 MACHINE CODES USED

The survey calculations of gamma dose rates were done for the most part by machine codes, described in this section. The major calculational work was done by the ROC primary shielding codes, ⁽⁶⁾ developed by Alco Products to obtain gamma dose rates through a primary shield configuration and programmed for use on an IBM-650 computer.

The ROC shielding codes are divided into four programs, with the first two programs used to prepare data for the latter two. The latter two programs calculate gamma dose rates from the results of neutron captures in the materials of the shield, and gamma dose rates from the activity of core material. The code begins by calculating volumetric sources for the reactor core. Taking five energy groups ranging from greater than 7 Mev to 0.75 Mev. Input for this program is composed of volume fractions for each material in the core and average fast and thermal flux values for the core. In the second program, volumetric source values are calculated for each of the shield materials, also taking the 5 energy groups.

Of the latter programs, Code 653 calculates the gamma dose rate from the sources in the shield and reports these in Roentgens per hour for each of the five energy groups. Code 654 calculates the contribution of the dose rate arising from activity buildup of materials of the core. Each of these four codes depends on a library of previously calculated material cards. Libraries for shut-down times from 0.1 day to 10 days and operating conditions are available. This survey concerned itself with operating conditions and 0.1 of a day after shut-down from infinite operation.

The ROC Codes depend upon the VALPROD ⁽⁷⁾ code for fast and thermal fluxes which are used as input in Program 652. The VALPROD code is a modified two-group multi-region code. This code prepares input for Problem 22 which is run immediately following VALPROD. Input for VALPROD consists of fast and thermal core constants, axial buckling and geometry values. Regions and mesh spacings are specified. Flux values for each mesh point are generated. The code is limited to 19 regions and 79 mesh points which is adequate for this survey.

The fluxes derived from VALPROD and Problem 22 are in an unnormalized form. In order to normalize these flux values for the ROC codes, we can employ Programs 39 and 39 S ⁽⁸⁾ which multiply each flux by a normalizing factor. This gives the true fluxes throughout the shield configuration to be used in Program 652 of the ROC code.

To obtain the normalized values of the flux, the normalizing factor is calculated from the following formula:

$$NF = \nu_c \frac{\delta \times P}{V}$$

Where,

NF = normalizing factor

ν_c = effective number of neutrons per fission
(from VALPROD output)

δ = fission per watt-sec. (3.24×10^{10})

P = reactor power level in watts

V = Core volume (cm^3)

4.0 THE HIGHLY ENRICHED BWR CONCEPT

4.1 DESCRIPTION OF CONFIGURATION

This configuration consisted of a highly enriched boiling water type reactor with a water reflector and carbon steel reactor pressure vessel. The reactor vessel was 1-3/8 in. steel, clad with 1/4 in. stainless steel and had a 59.5 in. ID. The vessel was enclosed in a steel containment vessel of 1-1/4 in. wall thickness and 70 in. ID. A shield water tank extended out to a total radius of 7.5 ft. In this water the heavy lead shield was positioned.

For survey purposes 1-1/2 in. and 4 in. lead thicknesses were investigated. Dose rates were determined by the ROC code through these thicknesses of lead and through the water of the primary shield tank without lead (0 in. of lead). Table 4-1 gives a detailed layout of this geometrical configuration.

TABLE 4-1
5x5 BWR REACTOR CONFIGURATION

<u>Region</u>	<u>Material</u>	<u>Thickness, (In.)</u>	<u>Outside Radius, (In.)</u>
Core	-----	-----	13
Reflector	Water	16.75	29.75
Clad	Stainless Steel	0.25	30.00
Reactor Vessel	Carbon Steel	1.375	31.375
Insulation	Glass Wool	2	33.375
Insulation Can	Stainless Steel	0.125	33.50
Air Gap	Void	1.50	35.0
Containment Vessel	Carbon Steel	1.25	36.25
Water Gap	Water	2	38.25
Gamma Shield	Lead	0, 1.5, 4	42.25
Water	Water	Varied	90.00

4.2 CORE NUCLEAR CONSTANTS

The core nuclear constants used were those of the PL-2 core and water reflector listed in Table 4-2.

TABLE 4-2
BWR NUCLEAR CONSTANTS

	<u>Core</u>	<u>Water Reflector</u>
D_{th}	0.29545	0.2490
D_f	1.70179	1.62974
Σ_{th}^a	0.118971	0.00975
τ	57.726	42.5575
p	0.788813	0.98664
K_f	1.32790	0
K_{th}	1.72208	0
B^2 (axial)	0.0020706	

4.3 PRIMARY SHIELD NUCLEAR CONSTANTS

The materials of the primary shield were water, carbon steel, stainless steel, glass wool insulation, and lead. Table 4-3 lists the nuclear constants used in the VALPROD code to obtain the neutron distribution for each case investigated.

TABLE 4-3
SHIELD NUCLEAR CONSTANTS

	<u>Stainless (9)</u> <u>Steel</u>	<u>Carbon (9)</u> <u>Steel</u>	<u>Water (9)</u>	<u>Insulation</u>	<u>Lead (10)</u>
D_{th}	0.326055	0.303789	0.1434	9.999	1.039
D_f	1.314459	1.22727	1.6758612	9.999	3.600
Σ_{th}^a	0.144415	0.219076	0.0112226	zero	0.00478
τ	161.635805	150.3994	34.963813	999999.	104.500
p	zero	zero	0.9296183	1.0000	0.99532
K_f	zero	zero	zero	zero	zero
K_{th}	zero	zero	zero	zero	zero

4.4 NEUTRON DISTRIBUTION

The normalized neutron fluxes are shown in Fig. 4.1. This is the neutron radial distribution through the primary shield and into the shield water. These neutron flux values were used in Program 652, a primary shielding code to obtain source data for Program 653.

4.5 GAMMA DOSE RATE

The gamma dose rate through the primary shield was calculated by the ROC codes discussed in Section 3.0. The parameters needed for Program 651 are volume fractions and average fast and thermal flux values in the reactor core. The volume fractions and average fluxes used for the BWR core were as follows:

VF U ²⁵⁰²	=	0.012551
VF 304 Stainless	=	0.123039
VF Water	=	0.860017
$\bar{\phi}$ fast	=	5.078×10^{13}
$\bar{\phi}$ th	=	1.049×10^{13}

For this configuration the number of shield regions was 10.

Program 653 calculates the dose rate due to sources arising from the shield materials. It requires the distance from core surface to last point in the shield (195.6 cm) and the core radius (33 cm).

Input for Program 654, which calculates the dose rate contributed by the reactor core, requires the reactor core radius previously quoted and the height of the reactor core (66.04 cm). Material files do not differ from those of Program 653.

4.6 OUTPUT

Table 4-4 gives the output from the machine calculations, with only shutdown cases run for this configuration.

Figure 4.1. Neutron Flux Traverse - BWR 5 x 5 Core

TABLE 4-4
OUTPUT FROM MACHINE CALCULATIONS FOR THE BWR CONFIGURATION

<u>Reactor Condition</u>	<u>Gamma Shield Thickness, (In.)</u>	<u>Distance from Reactor Centerline, (In.)</u>	<u>Gamma Dose Rate, mr/hr</u>
Shutdown	4 Pb	36.3	2.48×10^4
Shield		38.3	2.02×10^4
Contribution		42.3	5.48×10^2
		55.0	7.33×10^1
Shutdown	0 Pb	36.3	8.31×10^4
Shield		43.4	3.17×10^4
Contribution		75.7	2.80×10^2
		104.3	4.77×10^{-3}
Shutdown	0 Pb	36.3	3.45×10^7
Core		43.4	1.19×10^7
Contribution		75.6	4.26×10^4
Shutdown	4 Pb	36.3	2.77×10^7
Core		38.3	2.16×10^7
Contribution		42.3	1.24×10^5
		55.0	2.16×10^4
Shutdown	1 1/2	36.3	2.77×10^7
Core		38.3	2.16×10^7
Contribution		39.7	3.07×10^6
		60.9	1.50×10^5

Results are plotted in Fig. 4.2 through 4.4 as dose rate vs. distance from core centerline. The dashed lines are operating doses, inferred from other calculations for similar configurations.

Figure 4.2. Gamma Dose Rate Vs Distance - BWR - 0 In. Pb, Shutdown

Figure 4.3. Gamma Dose Rate Vs Distance - BWR - 1-1/2 In. Pb, Shutdown

Figure 4.4. Gamma Dose Rate Vs Distance - BWR - 4 In. Pb, Shutdown

5.0 TYPE 3 CORE

5.1 DESCRIPTION OF CONFIGURATION

The Type 3 core shielding configuration is based upon an SM-2 type core (PWR) with a 5 x 5 fuel element array. This configuration used a 2-1/2 in. steel reflector placed next to the core, composed of three 13/16 in. rings and one 1/2-in. ring separated from each other by 1/4-in. water annuli. The pressure vessel is carbon steel with 52-in. diam clad on the inside with stainless steel.

The radial configuration is indicated by Table 5.1 and on Dwg. AEL-705.

TABLE 5-1
TYPE 3 CORE RADIAL CONFIGURATION

<u>Region</u>	<u>Material</u>	<u>Thickness, (In.)</u>	<u>Outside Radius, (In.)</u>
Core			8.29
Water	Water	0.25	8.54
Reflector Ring	Steel	0.50	9.04
Water	Water	0.25	9.29
Reflector Ring	Steel	0.8125	10.10
Water	Water	0.25	10.35
Reflector Ring	Steel	0.8125	11.16
Water	Water	0.25	11.41
Reflector Ring	Steel	0.8125	12.22
Water	Water	13.78	26.00
P.V. Clad	Stainless		
	Steel	0.25	26.25
Pressure Vessel	Carbon		
	Steel	1	27.25
Insulation	Glass		
	Wool	1.5	28.75
Support ring	Steel	1.5	30.25
Water	Water	1.5	31.75
Gamma Shield	Lead	0, 1.5, 4	35.75
Water	Water	Varied	90.0

5.2 CORE NUCLEAR CONSTANTS

The core constants for the 5x5 Type 3 core are as follows for use in the VALPROD code:

TABLE 5-2
TYPE 3 CORE NUCLEAR CONSTANTS

D_f	=	1.394013
D_{th}^2	=	0.234241
Σ_f^a	=	0.012860
Σ_f^f	=	0.017782
τ	=	42.0338
Σ_{th}^a	=	0.312872
$\nu \Sigma_f^{th}$	=	0.514604
K_f	=	1.3827
K_{th}	=	1.64477
ρ	=	0.601436
B^2 (axial)	=	0.00192913

5.3 NEUTRON DISTRIBUTION

The neutron distribution from VALPROD is shown in Fig. 5.1 for the Type 3 core. This distribution was normalized to true flux in the core and then the fast and thermal fluxes were plotted for further use in Program 652 of the ROC codes.

SECTION A-A

NOTES:
 MATERIAL - 304 STAINLESS STL.
 DESIGN PRESSURE - 1350 P.S.I.

PRELIMINARY

LAYOUT			
ALCO PRODUCTS INC. SCHENECTADY, NEW YORK ATOMIC ENERGY DEPT.			
MODEL	PL-9	DESIGN	REVISED
DATE		DATE	
APPROVED		APPROVED	
DATE		DATE	
APPROVED		APPROVED	
DATE		DATE	
APPROVED		APPROVED	
TITLE			
REACTOR VESSEL			
CONCEPT A-1			
LAYOUT NO. AEL 705			

5.4 GAMMA DOSE RATE

The dose rate calculation was run as before for shutdown at 0.1 day after infinite reactor operation. Volume fractions for the core, needed for Program 651, are as follows:

$$\text{VF } \text{U}^{25}\text{O}_2 = 0.00240$$

$$\text{VF } 304 \text{ SS} = 0.237$$

$$\text{VF } \text{H}_2\text{O} = 0.739$$

The average flux values were

$$\bar{\phi}_{\text{fast}} = 1.62 \times 10^{14} \text{ n/cm}^2\text{-sec}$$

$$\bar{\phi}_{\text{th}} = 1.04 \times 10^{13} \text{ n/cm}^2\text{-sec}$$

Three cases were surveyed for this configuration. The gamma shield varied from 4 in. to 1-1/2 in. to 0 in. (i.e. for the latter, water only appeared in the primary shield tank).

This core radius was 21.05 cm, the height of the core was 55.88 cm and the width of the shield from the edge of the core to the last point in the shield was 207.6 cm. These parameters are used to run Programs 653 and 654.

Table 5-3 displays the output from the ROC codes, with dose point distances where the doses were measured. These results are plotted in Fig. 5.2 thru 5.4.

TABLE 5-3
TYPE 3 CORE GAMMA DOSE RATE RESULTS

<u>Reactor Condition</u>	<u>Gamma Shield Thickness, (In.)</u>	<u>Distance From Core, (In.)</u>	<u>Gamma Dose Rate, mr/hr</u>
Shutdown Core Contribution	0	30.2	4.69×10^6
		46.5	2.82×10^5
Shutdown Core Contribution	1 1/2	30.2	4.69×10^6
		31.7	3.79×10^6
		33.3	6.70×10^5
		54.5	$\sim 3.41 \times 10^4$
Shutdown Shield Contribution	4	30.2	5.15×10^5
		31.7	4.21×10^5
		35.7	5.16×10^3
		51.2	5.47×10^2
Shutdown Core Contribution	4	30.2	4.69×10^6
		31.7	3.79×10^6
		35.7	2.78×10^4
		51.2	4.05×10^3

Figure 5.1. Neutron Flux Traverse - 5 x 5 PWR-SM-2 Type 3 Core

Figure 5.2. Gamma Dose Rate Vs Distance - 5 x 5 PWR - 0 In. Pb, Shutdown

Figure 5.3. Gamma Dose Rate Vs Distance -
5 x 5 PWR - 1 1/2 In. Pb
Shutdown

Figure 5.4. Gamma Dose Rate Vs Distance -
5 x 5 PWR - 4 In. Pb, Shutdown

6.0. TYPE 2 CORE

6.1 DESCRIPTION OF CONFIGURATION

The Type 2 core is a 7x7 fuel element array (SM-1 type core). This layout has a water reflector and a carbon steel pressure vessel with 51-1/2 in. ID. The inside surface is clad with 0.25 in. stainless steel. A support ring is placed next to the reactor vessel insulation. The lead gamma shield is then placed in the water tank outside the support ring vessel and 2 in. from it in a radial direction.

Reference drawings AEL-728 and 732 depict the layout. Table 6-1 lists this configuration as used in the shielding survey.

TABLE 6-1
TYPE 2 CORE RADIAL CONFIGURATION

<u>Region</u>	<u>Material</u>	<u>Thickness, (In.)</u>	<u>Outside Radius, (In.)</u>
Core			11.1
Reflector	Water	14.65	25.75
P. V. Clad	Stainless Steel	0.25	26.00
Pressure Vessel	Carbon Steel	1.125	27.12
Insulation	Glass Wool	1.666	28.79
Support Ring	Stainless Steel	1.5	30.29
Water	Water	2	32.29
Gamma Shield	Lead	0,1.5,4	36.29
Water	Water	Varied	90.0

Reactor Vessel Detail Concept A-3

Primary System Complex A-3 (PWR)

6.2 CORE NUCLEAR CONSTANTS

The nuclear constants used for the Type 2 Core were for a temperature of 440°F and were as follows:

TABLE 6-2
TYPE 2 CORE NUCLEAR CONSTANTS

D_{t1}	0.247063
D_f	1.334972
Σ_a^{th}	0.206452
τ	38.826884
ρ	0.729426
K_f	1.289046
K_{th}	1.570528
B^2 (axial)	0.002129

6.3 NEUTRON DISTRIBUTION

The VALPROD and Problem 22 Output was normalized in the usual manner for true flux to be used as input to Programs 652 and 653. The fast and slow neutron distribution may be seen in Fig. 6.1.

6.4 GAMMA DOSE RATE

The gamma dose rates were calculated for 0.1 day after shutdown and operating condition.

The volume fractions and flux averages needed for program 651 are as follows:

VF $U^{235}O_2$	0.008572
VF 304 Stainless Steel	0.191627
VF H_2O	0.799801
$\bar{\phi}_f$	9.60×10^{13}
$\bar{\phi}_{th}$	1.26×10^{13}

Dose rates calculated for this configuration are listed in Table 6-3 and Fig. 6.2 to 6.7.

Figure 6.1. Neutron Flux Traverse Type 2 Core

TABLE 6-3
TYPE 2 CORE GAMMA DOSE RATE RESULTS

<u>Reactor Condition</u>	<u>Lead Shield Thickness, (In.)</u>	<u>Contribution</u>	<u>Distance from Core Centerline (In.)</u>	<u>Gamma Dose Rate, mr/hr</u>
Shutdown	0	Core	30.2	3.24×10^7
			37.2	1.25×10^7
			50.9	$\sim 1.10 \times 10^6$
Shutdown	1-1/2	Core	30.2	3.24×10^7
			32.3	2.52×10^7
			33.8	3.45×10^6
			48.5	4.25×10^5
Shutdown	4	Core	30.2	3.24×10^7
			32.3	2.52×10^7
			36.3	1.52×10^5
			47.0	3.31×10^4
Shutdown	0	Shield	30.2	1.26×10^5
			37.2	5.21×10^4
			50.9	6.92×10^3
Operating	0	Core	30.2	3.67×10^8
			37.2	1.49×10^8
			50.9	2.45×10^7
Operating	1-1/2	Core	30.3	3.67×10^8
			32.3	2.77×10^8
			33.8	2.87×10^7
			48.6	4.37×10^6
Operating	4	Core	30.3	3.67×10^8
			32.3	2.77×10^8
			36.3	7.27×10^5
			47.0	1.78×10^5
Operating	4	Shield	30.3	4.40×10^8
			32.3	3.49×10^8
			36.3	2.76×10^6
			47.0	9.09×10^5

Figure 6.2. Gamma Dose Rate Vs Distance -
7 x 7 PWR - 0 In. Pb, Shutdown

Figure 6.3. Gamma Dose Rate Vs Distance -
7 x 7 PWR - 1 1/2 In. Pb,
Shutdown

Figure 6.4. Gamma Dose Rate Vs Distance -
7 x 7 PWR - 4 In. Pb, Shutdown

Figure 6.5 Gamma Dose Rate Vs Distance -
7 x 7 PWR - 0 In. Pb, Operating

Figure 6.6. Gamma Dose Rate Vs Distance -
7 x 7 PWR - 1 1/2 In. Pb,
Operating

Figure 6.7. Gamma Dose Rate Vs Distance -
7 x 7 PWR - 4 In. Pb, Operating

7.0 TYPE 2 CORE - STEEL-WATER SHIELD CONFIGURATION

7.1 DESCRIPTION OF CONFIGURATION

To compare the effects of a laminated steel-water shield with the lead shielding dose rates previously calculated, three steel-water shield configurations were set up and calculated. There was no change in the basic geometry from the edge of the core to the outside edge of the containment vessel. However, after a 1 in. water gap, steel rings were placed in the shield tank water instead of the lead previously used.

Three configurations were studied: one used one 2-in. steel ring; the second used three 2-in. steel rings separated by a 1-in. water gap; the third used four 2-in. rings separated by 1-in. water gaps. Therefore, the total steel in the water tank was 2, 6, and 8 in. Table 7-1 indicates the four shield ring configuration, from which the other configurations can easily be obtained.

TABLE 7-1
TYPE 2 CORE STEEL-WATER SHIELD CONFIGURATION

<u>Region</u>	<u>Material</u>	<u>Thickness</u> <u>In.</u>	<u>Outer Radius,</u> <u>In.</u>
Support Ring	Stainless Steel	1.5	30.30
Water Annulus	Water	1	31.30
1st Shield Ring	Steel	2	33.30
Water Annulus	Water	1	34.30
2nd Shield Ring	Steel	2	36.30
Water Annulus	Water	1	37.30
3rd Shield Ring	Steel	2	39.30
Water Annulus	Water	1	40.30
4th Shield Ring	Steel	2	42.30
Water Annulus	Water	Varied	90.00

7.2 GAMMA DOSE RATE DISTRIBUTION FOR THE TYPE 2 CORE STEEL-WATER SHIELD CONFIGURATION

Since the neutron distribution was the same as the lead case through the containment vessel, no new input was needed. The only change in ROC code input is a substitution of the steel material file for the lead previously used, thus facilitating calculation of the results. All cases run were for the shutdown 0.1 day case; results are displayed in Table 7-2. The dose rates were plotted and are shown in Fig. 7.1 through 7.3.

TABLE 7-2
TYPE 2 CORE STEEL-WATER GAMMA DOSE RATE RESULTS

<u>Reactor Condition</u>	<u>Gamma Shield Thickness, In.</u>	<u>Contribution</u>	<u>Distance from Core Centerline, In.</u>	<u>Gamma Dose Rate, mr/hr</u>
Shutdown	2 Steel	Core	30.3	3.24×10^7
			31.3	2.84×10^7
			33.6	4.53×10^6
			48.4	$\sim 4.67 \times 10^5$
Shutdown	6 Steel	Core	30.3	3.24×10^7
			31.3	2.84×10^7
			33.3	4.53×10^6
			34.3	4.03×10^6
			36.3	6.70×10^5
			37.3	5.97×10^5
			39.3	9.05×10^4
Shutdown	8 Steel	Core	57.4	7.03×10^3
			30.2	3.24×10^7
			31.3	2.84×10^7
			33.3	4.53×10^6
			34.3	4.03×10^6
			36.3	6.70×10^5
			37.3	5.97×10^5
			39.3	9.05×10^4
			40.3	7.69×10^4
			42.3	7.37×10^3
Shutdown	6 Steel through the last two steel shield rings	Shield	53.3	$\sim 1.70 \times 10^3$
			30.3	1.5×10^5
			34.3	3.8×10^4
			36.2	1.2×10^4
			37.2	1.1×10^4
			39.3	3.2×10^3
			57.4	2.4×10^2

Figure 7.1. Gamma Dose Rate Vs Distance -
7 x 7 PWR - 2 In. Fe, 1 Iron
Shield Ring, Shutdown

Figure 7.2. Gamma Dose Rate Vs Distance -
7 x 7 PWR - 6 In. Fe, 3 Iron
Shield Rings, Shutdown

Figure 7.3. Gamma Dose Rate Vs Distance - 7 x 7 PWR - 8 In. Fe, 4 Iron Shield Rings, Shutdown

8.0 TYPE 2 CORE-REFERENCE CONFIGURATION

8.1 DESCRIPTION OF CONFIGURATION

The Type 2 core reference configuration is basically a 7 x 7 fuel element array, SM-1 type core, with a stainless steel pressure vessel. This survey used a pressure vessel wall thickness of 2-1/2 in. and vessel ID of 40 in. Two and one-half inches of lead was selected as the gamma shielding for the reference core. The lead was clad on the inside and outside surfaces with 1/4-in. of stainless steel. A detailed layout may be seen in Table 8-1 and on reference drawing AEL-762.

TABLE 8-1
TYPE 2 CORE REFERENCE CONFIGURATION

<u>Region</u>	<u>Material</u>	<u>Thickness, In.</u>	<u>Outside Radius, In.</u>
Core			11.1
Reflector	Water	8.9	20.0
Pressure Vessel	Stainless Steel	2.5	22.5
Insulation	Glass Wool	2	24.5
PV Support Ring	Stainless Steel	1	25.5
Water	Water	1	26.5
Shield Clad	Stainless Steel	0.25	26.75
Gamma Shield	Lead	2.5	29.25
Shield Clad	Stainless Steel	0.25	29.50
Water	Water	60	90.00

8.2 NEUTRON DISTRIBUTION

The VALPROD and Problem 22 Codes were rerun for this configuration with the same nuclear constants as previously quoted. The fluxes were again normalized to true power using 3.84×10^{12} as a normalizing factor. The resultant fast and thermal fluxes are shown in Fig. 8.1.

LAYOUT	
ALCO PRODUCTS INC.	
SCHENECTADY, NEW YORK	
ATOMIC ENERGY DEPT.	
MODEL	PL-3
DATE	8-12-62
REFERENCE	8-12-62
AUTHORITY	8-12-62
DATE	8-12-62
APPROVED	
TITLE	
PRIMARY SYSTEM	
COMPLEX C5 (PWR)	
LAYOUT NO.	
REL-762	

8.3 GAMMA DOSE RATE DISTRIBUTION

Gamma dose rates were calculated for the operating reactor and shutdown reactor conditions as before.

Program 651 uses the same volume functions and average fluxes as quoted in Section 6.4, since there has been no change in the core.

A detailed listings of the gamma dose rates calculated for the reference configuration will be seen in Table 8-2. These are plotted in Fig. 8.2 and 8.3.

TABLE 8-2
REFERENCE CONFIGURATION GAMMA DOSE RATE RESULTS

<u>Reactor Condition</u>	<u>Gamma Shield Thickness In.</u>	<u>Contribution</u>	<u>Distance Reactor from Centerline, In.</u>	<u>Gamma Dose Rate, mr/hr</u>
Operating	2-1/2	Core	36.1	5.86×10^7
Operating	2-1/2	Core	49.5	1.24×10^7
Operating	2-1/2	Core	62	2.3×10^6
Operating	2-1/2	Shield	62	2.10×10^6
Shutdown	2-1/2	Core	62	8.25×10^3
Shutdown	2-1/2	Shield	62	4.21×10^3
		Shield	87.2	1.09×10^2

Figure 8.2. Gamma Dose Rate Vs Distance, Operating Condition - 7 x 7 Reference Core

Figure 8.3. Gamma Dose Rate Vs Distance, 2.4 Hr After Shutdown - 7 x 7 Reference Core

9.0 CONCLUSIONS AND RECOMMENDATIONS

Cross plots giving gamma shield material thickness required as a function of radius are given in Fig. 9.1, 9.2, 9.3, and 9.4.

The extent of the shield radius which now appears most compatible with other design considerations is 7.5 ft. For this radius shield around the Type 2 7 x 7 core, either 5.8 in. of steel or 3.8 in. of lead will reduce the shutdown dose to 90 mrem/hr at 2.4 hr after shutdown.

Lead is recommended as the gamma shielding material rather than steel because of the weight penalty of steel, the increased problems associated with the activation of steel, and higher operating gamma dose rates associated with the steel-water shield. The lead should be positioned as close to the pressure vessel support ring as is consistent and feasible with mechanical design considerations.

A fabricated shield of 7.5 ft radius as described above would require 9 percent more weight of material to be shipped if steel were used for the gamma shielding. Since all shipments will be made by air, this is a significant factor regarding shipping weights and logistics requirements.

A more compact shield would result if lead were used, since steel would require a water annulus for each steel shield ring used. Nuclear instrumentation may be placed nearer the core using lead shielding than with steel shielding. The use of lead also results in a saving of 8 in. on the gamma shield diameter.

Figure 9.3 shows that the lead shielding might be reduced if the shield radius were increased over 7.5 ft, with water substituted for lead shielding.

It is felt that additional work using more sophisticated machine codes is required to finalize the primary shield configuration. The code tabulation in the Appendix gives several codes which might serve this purpose.

Figure 9.1. Lead Shield Thickness Vs Shield Radius - To Give 90 Mr/Hr at 2.4 Hr after Shutdown - BWR Configuration

Figure 9.2. Lead Shield Thickness Vs Shield Radius - To Give 90 Mr/Hr at 2.4 Hr after Shutdown - 5 x 5 PWR Configuration

Figure 9.3. Lead Shield Thickness Vs Shield Radius - To Give 90 Mr/Hr at 2.4 Hr after Shutdown - 7 x 7 PWR Configuration

Figure 9.4. Steel Shield Thickness Vs Shield Radius - To Give 90 Mr/Hr at 2.4 Hr after Shutdown - 7 x 7 PWR Configuration

10.0 REFERENCES

1. Bobe, P.E., Selection of Core for the PWR PL-3 Conceptual Design," AP Note 419, Alco Products, Inc., May 10, 1962.
2. Hauenstein, G. C., Pomeroy, D.L., "Spent Fuel Transfer, Storage and Shipment for PL-3," APAE Memo-310, March 7, 1962.
3. Bopp, C.D., Sissman O., "How to Calculate Gamma Radiation Induced in Reactor Materials," Nucleonics, Vol. 14, No. 1, p. 46, January 1956.
4. Wheeler, B.L., "Military Field Expedient Shielding Experiment," FZK 122, Convair, October 18, 1960.
5. "Attenuation in Water of Radiation from the Bulk Shielding Reactor," ORNL-2518, July 8, 1958.
6. Rosen, S. S., Oby, P.V., Caton, R.L., "Primary Shield Calculations on the IBM-650 (ROC Codes)," APAE Memo-142, October 15, 1958.
7. Oby, P.V., "Modified Two-Group Multiregion Calculation Using the VALPROD Code for the IBM-650," AP Note 24 Revised, Alco Products, Inc., August 14, 1957.
8. Bagley, R.O., "Flux and Power Normalization on the IBM-650," AP Note-113, Alco Products, Inc., September 10, 1958.
9. Rosen, S.S., et al, "Shielding Experiments and Analyses, Task VI," APAE-35, October 15, 1958.
10. Crouch, A.N., "SM-2 Primary Shield Survey," APAE Memo-247, February 19, 1960.

APPENDIX A - SURVEY OF SHIELDING MACHINE CODES

ABSTRACT

This appendix describes several recent machine codes relating to the reactor shielding field. References are given for further examination if interest is warranted.

LIST OF CODES

<u>CODE</u>	<u>MACHINE</u>	<u>REFERENCE</u>	<u>PAGE NO.</u>
RAS I	IBM 650	AECU 3778	A4
		AP NOTE 63	
MAGNET III	IBM 704	CWR 900-13-3	A5
MAGNET IV	IBM 704	CWR-900-13-4	A6
9-NIOBE	IBM 7090	NDA 2143-18	A7
PATCH	IBM 704	NAA-SR-MEMO 5299	A8
FARSE	Not given	NAA-SR-MEMO 5772	A9
GRACE I	IBM 704	NS & E 2 #1	A10
	IBM 709		
GRACE II	IBM 704, 709	NAA-SR-MEMO 4649	A11
FANTASIA & TRIPOD	1103A UNIVAC	WADC-TR-59-443	A12
C-17, D-53	IBM 704, 7090	PROCEDURE C-17, D-53	A13
SYMMETRIC SHIELDS	IBM 704	ORNL 3016	A16
N-NEUTRON FLUXES	IBM 704	KAPL 2090	A17
PENETRATION PROBLEMS	IBM 704	WADC-TR-59-443	A18
DISTRIBUTION OF PENETRATING NEUTRONS IN REACTOR SHIELDS	FERRANTI-MERCURY	AERE-R-3082	A19
SHIELDING PROGRAMS		APEX-387	A20

LIST OF CODES (CONT'D)

<u>CODE</u>	<u>MACHINE</u>	<u>REFERENCE</u>	<u>PAGE NO.</u>
01-0, 02-0, 03-0	IBM 704	APEX 428	A21
04-0	IBM 704	APEX 396	A22
09-0	IBM 704	APEX 533	A23
10-0	IBM 704	APEX 503	A24
12-0	IBM 704	APEX 404	A25
14-0, 14-1	IBM 704	XDC 59-2-16	A26
14-2	IBM 704	XDC 59-6-173	A27
14-3	IBM 704	XDC 59-3-52	A28
39 S	IBM 650	AP NOTE 113	A29
ROC	IBM 650	APAE MEMO 142	A29
PROGRAM 655	IBM 650	APAE MEMO 162	A29
CAMEL CODE	IBM 650	AP NOTE 73 REV	A29
FLAP	IBM 650	AP NOTE 233	A29
LAP	IBM 650	AP NOTE 238	A30
SPIC I	IBM 704		A31
SPAN 2	IBM 704		A32
MISCELLANEOUS DESCRIPTIONS			A33
REACTOR SHIELD PENETRATIONS	IBM 704		
NEUTRON ATTENUATION	IBM 650		
MARTIN GENERALIZED SHIELDING CODE			
MARTIN LINE SOURCE SHIELDING CODE			
SPIC I			

Name of Code RAS-1

Computer and Program System IBM 650

Nature of Problem Solved

Calculates gamma dose rates at miscellaneous points in a reactor system given system geometry, shadow shield, and source data. Primary and secondary sources may be used.

Restrictions

The equations apply only to gamma radiation due to the activation of pressurized water in primary coolant systems (6 MEV decay gamma source).

References

AECU 3778
AP NOTE 63

Authors, AECU 3778

E. Czapek, R. Gulino, S. Pierce, and J. Shapiro.

Issuing Facility

Electric Boat Co., Groton, Conn.

Name of Code

MAGNET III

Computer and Program System IBM 704

Nature of Problem Solved

The purpose of the code is to provide dose rate or leakage data from systems of given spherical shapes.

Method Used

Monte Carlo

Restriction

One, two and three dimension geometry.

References

CWR 900-13-3 Monte Carlo Calculations for Neutron and Gamma particle interactions and transports in one two and three dimensional nuclear assemblies. Magnet III Program for one and two dimensional Geometry, Gamma heating and leakage.

Authors

J.F. Logue, W.J. Roberts, K.L. Rooney and J.A. De Vries, 236 pp.

Issuing Facility

Curtiss Wright Corp., Research Division, Quehanna, Pa.

Name of Code MAGNET IV

Computer and Program System IBM 704

Nature of Problem Solved

Attacks two basic problems in fast reactor analysis

1. fast reactor criticality, lifetime, isotopic absorption ratios, inelastic scattering and two dimensional cylindrical geometries.
2. for an arbitrary source type, transmission thru spherical and cylindrical annuli; also absorption and energy loss in such annuli are determined.

Method Used

Monte Carlo

Restrictions

Up to three dimensional nuclear assemblies.

References

CWR 900-13-4 Monte Carlo Calculation for Neutron and Gamma Particle interactions and transport in one, two, and three dimensional nuclear assemblies - Magnet IV program for one and two dimensional geometry neutron interactions, plain sampling.

Authors

J.F. Logue, W.J. Roberts, Kevin L. Rooney, and J.A. De Vries, 179 pp (5-59) Curtiss Wright Corp., Research Division, Quehanna, Pa.

Issuing Facility

Curtiss Wright Corp., Research Div., Quehanna, Pa.

Name of Code 9-NIOBE

Computer and Program System IBM 7090

Nature of Problem Solved

Calculates angular distribution, total fluxes and currents for neutrons (or photons) as a function of energy (or photons), as a function of energy (or wave length) in a finite, multilayered spherically symmetric configuration.

Method of Solutions

Numerically integrates the time independent neutron or gamma ray Boltzmann transport equation.

Restriction

200 radial points, 16 directions, 5 materials, 50 regions.

References

NDA 2143-18 Military Compact Reactor Program (MCR). Description of Input Preparation and Operating Procedures for 9-NIOBE an IBM 7090 Code.

Authors

D. Yetman, B. Eiseman and G. Rabinowitz. United Nuclear Corp., successor to Nuclear Development Corp. of America, General Motors Corporation.

Issuing Facility

General Motors Corp., United Nuclear Corporation.

Name of Code

PATCH

Computer and Program System

IBM 709 FORTRAN

Nature of Problem Solved

Computes the dose energy and energy absorption buildup factor for multilayered slab shields.

Input Data Necessary

Number of regions, number of materials and their atomic numbers, and gamma and energy absorptions.

Restrictions

Code allows for n regions and six materials with a maximum of seven⁽⁷⁾ mesh points per region.

Running Time

For a multiregion problem is linearly proportional to the running time for a one region problem. One region problem with five (5) internal mesh points is less than 1 min.

References

NAA-SR-MEMO 5299 (5-60), 78 pp.

Author

Spiegler, P.A.

Issuing Facility

Atoms International, Division of North American Aviation.

Name of Code FARSE

Computer and Program System Not given

Nature of Problem Solved

Investigates the effect of complex shield geometries on overall weight and payloads. Computes dose deposit at each target mesh from each source mesh.

Method of Solution

Uses an attenuation model based on mean free paths traversed along a straight line trajectory.

Restrictions

Requires 51 pieces of input information per case.

Options

User may specify the shield and obtain the dose, or conversely specify the dose rates and tolerances across the pay load and determine the shield system satisfying these conditions.

Basis of Code

Attenuation Model

Running Time

Not Given in Abstract

References

NAA-SR-MEMO 5772 FARSE - A first order approximation of reactor shields for Snap Systems Part I - Non-Scattered Dose.

Author

Rooney, K.L., Boling, M.A.

Issuing Facility

Atomics International, Div. of North American Aviation, Conoga Park, California.

Name of Code

GRACE I

Computer and Program System

IBM 704 FORTRAN
IBM 709

Nature of Problem Solved

Computes the gamma flux distribution in a multi-region finite or semi-infinite slab shield.

Option

Dose rates or heat generation rates.

Restrictions

Regions 30

Gamma Energy Group 20.

References

An IBM 704-709 Program designed for computing Gamma Ray Attenuation and Heating in Reactor Shields - Nuclear Science and Engineering 2 #1 Suppl., 237-8 (1959) June.

Authors

D.S. Duncan and A.B. Speir

Issuing Facility

Atomics International Div., North American Aviation, Inc.,
Conoga Park, California.

Name of Program

GRACE II

Computer and Program System

IBM 704

IBM 709 FORTRAN II

Nature of Problem Solved

Multigroup, multiregion gamma ray attenuation code. Computes total dose rate or heat generation rate from either a spherical or cylindrical source.

Latitude

Source may be located in concentric shell region or in the central region. Source distribution may be uniform, exponential or have a polynomial variation in the radial direction. In the case of cylindrical geometry the distribution may have a polynomial variation in the axial direction. Buildup factor is represented analytically in a double exponential form. Shield region outside the source volume may be either semi-infinite slabs or concentric shells.

Option

Dose rate or heat generation rate.

Restrictions

Regions, 22 max, mesh points per region, 10. Gamma ray energy groups, 20. Shield materials, 20, material buildup factors 20.

References

NAA-SR-MEMO 4649 (A.I.) Div. of NAA GRACE II - An IBM 704 program for computing gamma-ray attenuation and heating in cylindrical and spherical geometries.

Authors

Duncan and Speirs

Issuing Facility

Atomics International, Div. of North American Aviation, Conoga Park, California.

Remarks

This document may be of pertinent interest.

Name of Code FANTASIA & TRIPOD

Computer and Program System 1103A UNIVAC

Nature of Problem Solved

Computes neutron transmission through Laminated Slab Shields and slowing down density within the shield.

Method of Solution

Monte Carlo

Basis

C.E. Code VALPROD

Status

Theory and application of FANTASIA & TRIPOD are discussed and operating instructions are included in writeup.

References

WADC-TR-59-443 (9-59) (109 pp).

Author

Steinberg, Heitnez and Arson, TRG, Inc. Syosset, NY

Issuing Facility

TRG, Inc. Syosset, NY

Name of Code

MOMENTS METHOD SHIELD PENETRATION
C-17 and D-53

Computer and Program System

IBM 7090 (C-17)
and IBM 704 (D-53)

Nature of Problem Solved

Computes the fast-neutron and gamma-ray number spectra and dose rates at point detectors taking into account the energy and spatial distribution of the source radiation, the changes in energy distribution as the shield is penetrated, and the geometric arrangement of the various shield materials.

The C-17 program additionally determines the heat generation rate due to the incident fast neutrons and gamma radiation from the number spectra and appropriate flux-to-heat conversion factors.

Method of Solution

The programs are accomplished by combining the differential energy spectra obtained by a moments method of solution to the Boltzmann transport equation with the concept of distributed point sources and line-of-sight penetration.

Restrictions

Both C-17 and D-53 require that the problem geometry be approximated as a combination of right parallelepipeds and frustra of circular conic shells.

Basis of Codes

Moments method solution to the Boltzmann transport equation.

Options

Performs neutron and gamma calculations, and heat generation rates.

Running Time

The estimated time to perform a neutron and gamma calculation using D-53 is $(1/4 \text{ to } n/4)$ sec/source point region where n is the number of x planes in each region. Specifically, if a calculation is to be made with a geometry containing six regions, each with 2 x -planes, and using 20 source points, then the problem will take on the order of $(1/4 + 1/2) (6) (20) = 90$ seconds. Time estimates for the C-17 program are similar to those for D-53.

References

Procedures.C-17 and D-53 Moments Method Shield Penetration Programs

Author

Peterson, D.M.

Issuing Facility

General Dynamics of Fort Worth (Formerly Convair), Fort Worth 1, Texas.

Name of Code

EFFICIENCY STUDIES WITH GAMMA RAY POR-
TION OF SPECIALIZED REACTOR SHIELD MONTE
CARLO PROGRAM 18-0

Computer and Program System

IBM 7090

Nature of Problem Solved

Due to confidential classification no information is available herein.

References

APEX 707, April 26, 1961

Author

M. Capo

Issuing Facility

General Electric Company, Cincinnati, Ohio

Name of Code

ANALYSIS OF AXIALLY SYMMETRIC REACTOR
SHIELDS

Computer and Program System

IBM 704

Nature of Problem Solved

Fast neutron and primary gamma doses outside of an axially symmetric shield can be computed for detector points located along the axis of symmetry.

Method of Solution

Furnishes neutron and gamma data applicable to reanalysis of shielding experiments and to general shield design.

References

ORNL 3016, Development for IBM 704 Analytical Code for Analysis of Axially Symmetric Reactor Shields.

Authors

F.B.K. Kam and H.E. Stern, Oak Ridge National Laboratory, Oak Ridge, Tenn.

Issuing Facility

Oak Ridge National Laboratory, Oak Ridge, Tenn.

Name of Code

SPACE-TIME VARIATION OF N NEUTRON FLUXES

Computer and Program System

IBM 704

Nature of Problem Solved

Theoretical aspects of program coupled lines and Groups (CLAG) for computing the space-time variation of neutron fluxes with a three dimensional model representation of the reactor are presented.

Method of Solution

A multichannel synthesis method is presented which enables neutron diffusion computation with simultaneous group inversion.

Restriction

3 Dimensional model. The model permits dynamics analysis with space and time variation not separated.

References

KAPL 2090 - Digital Computation of Space-time variation of N Neutron Fluxes.

Author

E.L. Wachspress

Issuing Facility

General Electric Co., KAPL, Schenectady, N.Y.

Name of Code

ANALYSIS OF PENETRATION PROBLEMS

Computer and Program System

IBM 704

Nature of Problem Solved

Computation of the transmission and reflection matrices are presented along with numerical results for gamma rays.

Method of Solution

Solution arrived at by the transmission matrix method.

References

WADC-TR-59-772 - The transmission matrix method for penetration problems (August 1960).

Authors

Yarmish, Fell, and Aronson, TRG, Inc., Syracuse, NY

Issuing Facility

TRG Inc., Syracuse, NY

Name of Code

DISTRIBUTION OF PENETRATING NEUTRONS IN
REACTOR SHIELDS

Computer and Program Systems : FERRANTI-MERCURY

Nature of Problem Solved

Evaluates the initial source distribution data for the neutron shielding program RASH B.

Options

Program can be used to obtain the removal flux distribution thru a reactor core and shield. Distribution of the average energy is also obtainable.

Restrictions

Limited to large cylindrical or plane systems

References

AERE-R-3082 (UK-AEA) a computer programme for determining the distribution of penetrating neutrons in reactor shields. October 1959, NSA 14-18541.

Authors

D.L. Bendall, Harwell, Becks, England (UK-AEA)

Issuing Facility

Atomic Energy of the United Kingdom.

Name of Code

SHIELDING COMPUTER PROGRAMS

Computer and Program System

DIGITAL

Nature of Problem Solved

Nuclear Analyses pertaining to shielding are presented.

Each program includes 1) the purpose of the program, 2) the theoretical or physical assumptions, and 3) the major computer equipment required for the program.

References

APEX 387 Shielding Computer Programs - October 1957.

Author

M.A. Capo

Issuing Facility

General Electric Co. ANP Dept. Cincinnati, Ohio

Name of Code 01-0, 02-0, and 03-0

Computer and Program System IBM 704

Nature of Problem Solved

Calculates direct beam fast neutron and gamma dose rates in and around cylindrical regions, coaxial with cylindrical source regions.

Method of Solution

Option

Computes weight of a reactor shield assembly

Components

Two memory frames, one magnetic drum, and two tape units.

References

APEX 428 Computer Programs 01-0, 02-0, and 03-0, Reactor Shield Analysis, February 1959.

Authors

M.A. Capo, W.E. Edwards, and J.E. MacDonald

Issuing Facility

General Electric Co., A.N.P. Dept., Cincinnati, Ohio

Name of Code PROGRAM 04-0

Computer and Program System IBM 704

Nature of Problem Solved

Computes reactor shield weight and fast n and gamma ray dose rates in and around complex reactor shield assemblies.

Method Used

Geometries are described by combinations of regions formed by rotating rectangles or trapezoids about the reactor axis or by translating rectangles parallel to the reactor axis. Compositions are expressed as volume fractions for each material in the field assembly. A cylindrical coordinate system is used to describe the location and dimensions of a volumetric source. Source distributions are assumed separable and may be described by cosine function. Modification of the Albert - Walton theory of neutron attenuation is used for fast neutron dose rate calculations. A combination of buildup factors is used with exponential attenuation to compute gamma ray dose rates.

References

APEX 396 Shielding Computer Program 04-0 Reactor Shield Analysis - Edwards, MacDonald, Goldberg, & Paine. (May 1958)

Issuing Facility

General Electric Co, ANP Dept., Cincinnati, Ohio

Name of Code 09-0

Computer and Program System IBM 704

Nature of Problem Solved

Calculates the fast neutron dose rate due to single-sintered radiation in a homogeneous, infinite medium for an anisotropic point source.

Method of Solution

Dose rate from the source spectrum is obtained by summation of the dose rates corrected for each initial energy.

Restrictions

10 Energy Levels

References

APEX 533 IBM 704 Program Report, ANP Shielding Program 09-0

Authors

J.W. Haffner, J.J. Loechler, J.E. MacDonald.

Issuing Facility

General Electric Co. ANP Dept., Cincinnati, Ohio

Name of Code 10-0

Computer and Program System IBM 704

Nature of Problem Solved

Calculates the fast neutron dose rate at a shielded point detector due to both direct beam and single-scattered radiation in a homogeneous, infinite medium from an anisotropic point source.

Method of Solution

The dose rate arising from the source spectrum is obtained by summation of the dose rates computed for each initial energy and for the rear side and front of the detector shield.

Restrictions

Both source and detector must be immersed in an infinite homogeneous medium.

References

APEX 503, IBM 704 Program Report Aircraft Nuclear Propulsion Shielding Program 10-0.

Authors

J.W. Haffner, J.J. Loechler, and J.E. MacDonald, G.E. Co.
ANPD, ANP Cincinnati, Ohio. March 1958

Issuing Facility

General Electric Co., ANP Dept., Cincinnati, Ohio

Name of Code

12-0

Computer and Program Systems IBM 704

Nature of Problem Solved

Computes the gamma ray flux, energy absorption rate or dose rate from an isotropic point source.

Method of Solution

Monte Carlo Calculations

Restrictions

20 initial energies

Equipment Needed

One frame of magnetic core memory and two tapes for utility routines, no magnetic drum is required.

References

APEX 404 Shielding Computer Program 12-0 Gamma Ray Penetration Point Source, August 1958.

Authors

W.E. Edwards and J.J. Loechler

Issuing Facility

General Electric Co., ANP Department, Cincinnati, Ohio

Name of Code 14-0, 14-1

Computer and Program System IBM 704

Nature of Problem Solved

Evaluate influence functions and integrate over source regions to perform reactor-shield penetration calculations for neutrons and gamma rays. Neutron and gamma ray fluxes, spectra, and dose and energy absorption rates can be computed for positions in and around complex shields containing multiple sources described in a cylindrical coordinate system. In addition, the programs can compute reactor-shield weight. Computation of any of these quantities in a single problem is optional.

Method of Solution

A modification of the Albert-Weldon theory of neutron attenuation is used for fast neutron flux or dose rate calculation in hydrozenous material. NDA differential scattered gamma-ray energy spectra are used in the computations of gamma ray spectra.

Restrictions

The programs are coded for use on an IBM 704 having 32 768 magnetic core memory locations. Magnetic tape units are used for the production package and output. No magnetic drum memory is required.

Options

Shield weight as well as dose points are calculated. Input may be either on tape or cards.

References

XCD 59-2-16 Shielding computer programs 14-0 and 14-1 Reactor Shield Analysis 1-23-59.

Authors

J.T. Martin, J.P. Galch, W.E. Edwards

Issuing Facility

General Electric Co., Nuclear Propulsion Dept., Cincinnati 15, Ohio

Name of Code 14-2

Computer and Program System IBM 704

Nature of Problem Solved

The same as for Code Nos. 14-0 and 14-1 except that instead of the sources being described in a cylindrical coordinate system, the sources for this program are described in a rectangular coordinate system.

The rest of the abstract describes the code similarly to Codes 14-0 and 14-1.

References

XDC 59-6-173 Shielding Computer Program 14-2 Reactor Shield Analysis (June 15, 1959).

Authors

J. T. Martin, J. P. Yalch, W. E. Edwards.

Issuing Facility

General Electric Co. Nuclear Propulsion Department, Cincinnati, 15, Ohio.

Name of Code 14-3

Computer and Program System IBM 704

Nature of Problem Solved

Provides an extensive check on the data which is used as input for Programs 14-0, 14-1, and 14-2. Purpose of the program is to save time by catching errors in input data which might result in a useless and costly run of one of the computational programs.

Method of Solution

The report explains the types of errors in each kind of data which are detected by 14-3 and illustrates, via the sample output sheets in the appendices, the printout obtained from a run of this program. The data is divided into two parts - non-geometrical and geometrical.

References

XDC 59-3-52 Shielding Computer Program 14-3 Data Check for Shielding Computer Programs 14-0, 14-1, 14-2, December 10, 1958.

Author

M. D. McDonald.

Issuing Facility

General Electric Co. Nuclear Propulsion Dept., Cincinnati, 15, Ohio.

LIST OF SHIELDING CODES WRITTEN BY
ALCO PRODUCTS, INC., SCHENECTADY, N.Y. FOR THE IBM 650

Program 39S AP Note 113 by Bagley & Stueck.

Normalizes VALPROD fluxes for use in ROC codes.

AP Note 113 Addendum by Bagley, Rosen, Stueck and Caton.

ROC Codes (Programs for 651, 652, 653, and 654).

APAE Memo 142 by Rosen, Oby, and Caton.

Calculates gamma dose rates in the primary shielding of thermal reactors for five separate gamma energies. Two-group fluxes are used to determine gamma source values with both shield activation and core gammas taken into account. May also be used for gamma heating calculations.

APAE Memo 142.

Collation of ROC Code Output (Program 655).

APAE Memo 162 by C. J. Stueck.

Sums the gamma dose values for all energy groups and for core and shield activation gammas. Results are total dose values for each specified point in the reactor shield.

The Camel Code (Program No. 73), AP Note 194 Rev. 1 by C. L. Padula.

The Camel Code is a program written to evaluate the radiation hazards from a nuclear incident. The code calculates the total integrated external dose and the dose due to rainout and fallout as well as other items of interest.

The FLAP Code (Program No. 82), AP Note 233, Ferguson, Lerman, Austin, and Padula.

The FLAP program is designed to calculate the gamma dose rate at a point for any of several simple geometries. Calculations can be made for any arrangement of shielding materials that the design engineer might expect to use. The different arrangements which may be calculated are 1. Point Source, 2. Size Source 3. Infinite slab or cylinder, 4. Cylindrical source with point at end.

The LAP Code AP Note 238, Ferguson, Lerman, Austin, and Padula.

The LAP codes were designed to be used in conjunction with the FLAP code. Common parameters for dose rate calculations may be obtained from the FLAP code and then used as part of the input for the LAP codes. The LAP I and LAP II codes calculate the gamma dose rate at a point due to line and cylindrical geometry sources, respectively.

Name of Code

SPIC I

Computer

IBM 704

Nature of Problem Solved

Calculates the fast neutron dose rate on the thermal neutron flux at a point outside a right circular cylindrical source which is surrounded by cylindrical shell shields and is capped by plane slab shields.

Method of Solution

Finds the fast-neutron dose rate or the thermal-neutron flux at a field point outside a right circular cylinder of finite height which contains a source distribution (r,z) in up to three concentric cylindrical regions.

Reference

WAPD-TM-196 SPIC I An IBM 704 Code to calculate the Neutron Distribution Outside a Right-Circular Cylindrical Source.

Author

P. Giller

Issuing Facility

Westinghouse - Bettis - Pittsburgh, Pa.

Name of Code

SPAN-2

Computer

IBM 704

Calculates the uncollided gamma flux at a point outside a right circular cylinder which is surrounded by cylindrical shell shields and above by plane slab shields.

References

WAPD-TM-176 SPAN2 - An IBM 704 Code to calculate Uncollided Flux Outside a Circular Cylinder (Aug. 1959).

Author

R. A. Gillis, T. J. Lawton, K. W. Brand

Issuing Facility

Westinghouse Bettis Plant, Pittsburgh, Pa.

MISCELLANEOUS SHIELDING CODES

Reactor Shield Penetration Calculations, Edwards and MacDonald, Paper presented at the Nuclear Engineering and Science Conference at Chicago, 1958.

Applications of point to point attenuation functions to the calculations of fast neutron and gamma ray dose and energy absorption rates in and around reactor shields is described. A description of IBM-704 programs capable of evaluating attenuation functions along source receiver paths in complex shields and performing a final integration over a source region is presented.

Name of Code

An IBM-650 Program to calculate the neutron attenuation in a water-metal reactor shield.

Nature of Problem Solved

Calculates the neutron attenuation in water-metal shields in one dimension of plane or cylindrical geometry for up to 14 regions.

Range

1 n 398 where n is the total number of divisions of range.

1 m 14 where m is the number of regions.

Speed

Time required per point is 2 n seconds.

Equipment Required

Tape system consisting of one 533, one on-line 407, IAS, one 727 magnetic tape unit, indexing registers and automatic floating point decimal arithmetic..

Shielding Codes Written at

Martin Company - Baltimore, Md.

GENERALIZED SHIELDING CODE

IBM 704 Computer

Calculates direct gamma or neutron flux and heating rate.

LINE SOURCE SHIELDING

IBM 704 Computer

Calculates gamma or neutron flux and heating from a shielded line source.