**Westinghouse
Astronuclear
Laboratory**

WANL-TME-1914

JULY 1970

DATA ITEM T-119

**RADIATION ENVIRONMENT IN
VARIOUS REACTOR COMPONENTS**

FINAL REPORT

MASTER

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Westinghouse Astronuclear Laboratory

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DATA ITEM T-119

RADIATION ENVIRONMENT IN VARIOUS REACTOR COMPONENTS

FINAL REPORT

APPROVED BY:

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NERVA Design Engineering

INFORMATION CATEGORY

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ABSTRACT

The radiation environment was experimentally measured in the core and reflector of the PAX/G reactor assembly which provides preliminary radiation information characteristic of the graphite-fueled R-1 reactor. The results obtained include neutron flux densities obtained with sulfur ($E > 2.9$ Mev), U-238 ($E > 1.5$ Mev), bare and cadmium-covered Dy ($E < 0.4$ ev). Fast neutron dose rates from Phylatrons and gamma dose rates from CaF_2 thermoluminescent dosimeters and from Bragg-Gray ion chambers were also determined.

Results were compared with analytical calculations based on a similar reactor model. Normalization factors to be applied to the calculational techniques were statistically determined along with 1σ standard deviations. The normalizations and standard deviations will be used in the radiation design efforts for the R-1 design.

Good agreement was obtained for the gamma dose rate from TLD's and for the fast flux ($E > 1.5$ MeV). Reasonable agreement was obtained for the fast neutron dose rate. The agreement obtained for fast neutron flux ($E > 2.9$ MeV), for thermal flux ($E < 0.4$ eV), and for Bragg-Gray dose rate was somewhat poor. However, these differences can be explained in part by the differences between the experimental and analytical models. Further calculation and investigation are planned to resolve the differences.

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TABLE OF CONTENTS

<u>Section</u>		<u>Page</u>
1.0	INTRODUCTION	1-1
1.1	TEST IDENTIFICATION	1-1
1.1.1	Type of Report	1-1
1.1.2	Scope of Test Document	1-1
1.1.3	Functional Category of Test	1-1
1.1.4	Contractor	1-1
1.1.5	Test Facility	1-1
1.2	PURPOSE	1-1
2.0	REFERENCES AND ASSUMPTIONS	2-1
2.1	REFERENCES	2-1
2.1.1	Specification	2-1
2.1.2	Data Item Description	2-1
2.1.3	Scope of Test Document	2-1
2.1.4	Drawing	2-1
2.1.5	TME	2-2
2.1.6	Data Release Memorandum	2-2
2.2	ASSUMPTIONS	2-2
3.0	TEST ARTICLE CONFIGURATION	3-1
3.1	PAX-G1A REACTOR	3-1
3.2	REACTOR MODIFICATIONS FOR PASSIVE DOSIMETRY	3-3
3.3	REACTOR MODIFICATIONS FOR ACTIVE DOSIMETRY	3-3

TABLE OF CONTENTS

<u>Section</u>		<u>Page</u>
4.0	SUMMARY OF TESTS	4-1
4.1	NARRATIVE	4-1
4.1.1	Test Description	4-1
4.1.2	Passive Dosimetry	4-2
4.1.3	Active Dosimetry	4-2
4.1.4	Power Level Calibrations	4-11
4.2	TEST RESULTS	4-11
4.2.1	General	4-11
4.2.2	Dysprosium-Aluminum Foils	4-13
4.2.3	Sulfur Pellets	4-13
4.2.4	Uranium-238 Foils	4-13
4.2.5	Thorium Foils	4-13
4.2.6	Comparison of the S and U-238 Measurements	4-13
4.2.7	Phylatrons	4-14
4.2.8	Thermoluminescent Dosimeters	4-14
4.2.9	Bragg-Gray Chambers	4-14
5.0	ANALYSIS OF TESTS	5-1
5.1	INTRODUCTION	5-1
5.2	ANALYTICAL MODEL	5-1
5.3	METHOD OF ANALYSIS	5-2
5.3.1	Calculational Procedure	5-2
5.3.2	Determination of Analytical and Experimental Quantities	5-3
5.3.2.1	Activation and Equivalent Neutron Flux Data	5-4
5.3.2.2	DOT Response Functions	5-6

TABLE OF CONTENTS

<u>Section</u>	<u>Page</u>
5.4 COMPARISON OF RESULTS	5-7
5.4.1 Fission Distribution Data	5-7
5.4.2 Sulfur ($E > 2.9$ Mev) Fast Neutron Flux Data	5-8
5.4.3 U-238 ($E > 1.5$ Mev) Fast Neutron Flux Data	5-9
5.4.4 Bare Dysprosium Neutron Activation Data	5-11
5.4.5 Dysprosium ($E < 0.4$ ev) Thermal Neutron Flux Data	5-12
5.4.6 Fast Neutron Absorbed Dose Rate Data	5-14
5.4.7 Photon Absorbed Dose Rate Data	5-15
5.4.8 Bragg-Gray Ion Chamber Photon Heating Rate Data	5-16
5.5 STATISTICAL ANALYSIS OF THE DATA	5-20
5.5.1 Introduction	5-20
5.5.2 Method of Analysis	5-21
5.5.3 Statistical Results	5-26
5.6 REFERENCES	5-28
6.0 CONCLUSIONS	6-1
APPENDICES	
A Special Reactor Components	A-1
B Foil Calibration Reports	B-1

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LIST OF ILLUSTRATIONS

<u>Figure</u>		<u>Page</u>
3-1	PAX-G 1A Reflector Assembly	3-2
3-2	Location of Dosimetry Positions in R-1 Test Sector	3-4
3-3	Position of Bragg-Gray Detectors and Be Slug in Radial Heating Rate Measurements in the Reflector	3-5 3-6
3-4	Traverse System Employed on In-Core and Reflector Tie Rod Hole Measurements	
4-1	PAX-G/1 Reflector Geometry	4-18
4-2	Bare Dy Activity Radial Traverses	4-19
4-3	Bare Dy Activity Midplane Radial	4-20
4-4	Bare Dy Activity Reflector Detail	4-21
4-5	Bare Dy Activity Axial Traverses	4-22
4-6	(Cd) Dy Activity Radial Traverses	4-23
4-7	(Cd) Dy Activity Midplane Radial	4-24
4-8	(Cd) Dy Activity Reflector Detail	4-25
4-9	(Cd) Dy Activity Axial Traverses	4-26
4-10	Dy Thermal Flux Radial Traverses	4-27
4-11	Dy Thermal Flux Midplane Radial	4-28
4-12	Dy Thermal Flux Reflector Detail	4-29
4-13	Dy Thermal Flux Axial Traverses	4-30
4-14	Dy Cadmium Ratio Radial Traverses	4-31
4-15	Dy Cadmium Ratio Midplane Radial	4-32
4-16	Dy Cadmium Ratio Reflector Detail	4-33
4-17	Dy Cadmium Ratio Axial Traverses	4-34
4-18	Sulfur Fast Flux Radial Traverses	4-35
4-19	Sulfur Fast Flux Midplane Radial	4-36
4-20	Sulfur Fast Flux Reflector Detail	4-37
4-21	Sulfur Fast Flux Axial Traverses	4-38
4-22	U-238 Fast Flux Radial Traverses	4-39
4-23	U-238 Fast Flux Midplane Radial	4-40

LIST OF ILLUSTRATIONS (Continued)

<u>Figure</u>		<u>Page</u>
4-24	U-238 Fast Flux Reflector Detail	4-41
4-25	U-238 Fast Flux Axial Traverses	4-42
4-26	S, U Comparison Central Axial	4-43
4-27	S, U Comparison Reflector Detail	4-44
4-28	S, U Comparison Midplane Radial	4-45
4-29	Phylatron Fast Neutron Dose Rate - Radial Traverses	4-46
4-30	Phylatron Fast Neutron Dose Rate - Midplane Radial	4-47
4-31	Phylatron Fast Neutron Dose Rate - Reflector Detail	4-48
4-32	Phylatron Fast Neutron Dose Rate - Axial Traverses	4-49
4-33	CaF ₂ Gamma Dose Radial Traverses	4-50
4-34	CaF ₂ Gamma Dose Midplane Radial	4-51
4-35	CaF ₂ Gamma Dose Reflector Detail	4-52
4-36	CaF ₂ Gamma Dose Axial Traverses	4-53
4-37	Absorbed Gamma Dose Rates in Fuel Element Void Q11B - Radius 1.914 cm	4-54
4-38	Absorbed Gamma Dose Rates in Fuel Element Void 4E5C - Radius 22.08 cm	4-55
4-39	Absorbed Gamma Dose Rates in Fuel Element Void 4H7B - Radius 34.62 cm	4-56
4-40	Absorbed Gamma Dose Rates in Fuel Element Void 4J8C - Radius 40.34 cm	4-57
4-41	Absorbed Gamma Dose Rates in Fuel Element Void 4P7E - Radius 47.16 cm	4-58
4-42	Absorbed Gamma Dose Rates as a Function of Elevation in the Reflector Tie Bolt Hole T-1 (Radius = 54.36 cm)	4-59
4-43	Absorbed Gamma Dose Rates as a Function of Elevation in the Reflector Tie Bolt Hole T-2 (Radius = 60.72 cm)	4-60
4-44	Absorbed Gamma Dose Rates as a Function of Radius in the Reflector, Position S-1	4-61
4-45	Absorbed Gamma Dose Rates as a Function of Radius in the Reflector, Position S-2	4-62

LIST OF ILLUSTRATIONS

<u>Figure</u>		<u>Page</u>
5-1	PAX-G1 DOT Geometry Mockup	5-35
5-2	PAX-G1 Computational Flow Chart	5-36
5-3	Comparison of Calculated and Experimental Relative Radial Fission Flux	5-37
5-4	Comparison of Calculated and Experimental Relative Axial Fission Flux	5-38
5-5	Comparison of DOT Calculated and Experimental Sulfur ($E > 2.9$ Mev) Fast Neutron Flux as a Function of Axial Position	5-39
5-6	Comparison of DOT Calculated and Experimental Sulfur ($E > 2.9$ Mev) Fast Neutron Flux as a Function of Axial Position	5-40
5-7	Comparison of DOT Calculated and Experimental Sulfur ($E > 2.9$ Mev) Fast Neutron Flux as a Function of Axial Position	5-41
5-8	Comparison of DOT Calculated and Experimental Sulfur ($E > 2.9$ Mev) Fast Neutron Flux as a Function of Axial Position	5-42
5-9	Comparison of DOT Calculated and Experimental Sulfur ($E > 2.9$ Mev) Fast Neutron Flux as a Function of Axial Position	5-43
5-10	Comparison of DOT Calculated and Experimental Sulfur ($E > 2.9$ Mev) Fast Neutron Flux as a Function of Axial Position	5-44
5-11	Comparison of DOT Calculated and Experimental Sulfur ($E > 2.9$ Mev) Fast Neutron Flux as a Function of Axial Position	5-45
5-12	Comparison of DOT Calculated and Experimental Sulfur ($E > 2.9$ Mev) Fast Neutron Flux as a Function of Radial Position	5-46
5-13	Comparison of DOT Calculated and Experimental Sulfur ($E > 2.9$ Mev) Fast Neutron Flux as a Function of Radial Position	5-47
5-14	Comparison of DOT Calculated and Experimental U-238 ($E > 1.5$ Mev) Neutron Flux as a Function of Axial Position	5-48
5-15	Comparison of DOT Calculated and Experimental U-238 ($E > 1.5$ Mev) Neutron Flux as a Function of Axial Position	5-49
5-16	Comparison of DOT Calculated and Experimental U-238 ($E > 1.5$ Mev) Neutron Flux as a Function of Axial Position	5-50

LIST OF ILLUSTRATIONS

<u>Figure</u>		<u>Page</u>
5-17	Comparison of DOT Calculated and Experimental U-238 ($E > 1.5$ Mev) Neutron Flux as a Function of Axial Position	5-51
5-18	Comparison of DOT Calculated and Experimental U-238 ($E > 1.5$ Mev) Neutron Flux as a Function of Axial Position	5-52
5-19	Comparison of DOT Calculated and Experimental U-238 ($E > 1.5$ Mev) Neutron Flux as a Function of Axial Position	5-53
5-20	Comparison of DOT Calculated and Experimental U-238 ($E > 1.5$ Mev) Neutron Flux as a Function of Radial Position	5-54
5-21	Comparison of DOT Calculated and Experimental U-238 ($E > 1.5$ Mev) Neutron Flux as a Function of Radial Position	5-55
5-22	Comparison of DOT Calculated and Experimental Bare Dysprosium Activation Data as a Function of Axial Position	5-56
5-23	Comparison of DOT Calculated and Experimental Bare Dysprosium Activation Data as a Function of Axial Position	5-57
5-24	Comparison of DOT Calculated and Experimental Bare Dysprosium Activation Data as a Function of Axial Position	5-58
5-25	Comparison of Dot Calculated and Experimental Bare Dysprosium Activation Data as a Function of Axial Position	5-59
5-26	Comparison of DOT Calculated and Experimental Bare Dysprosium Activation Data as a Function of Axial Position	5-60
5-27	Comparison of DOT Calculated and Experimental Bare Dysprosium Activation Data as a Function of Axial Position	5-61
5-28	Comparison of DOT Calculated and Experimental Bare Dysprosium Activation Data as a Function of Radial Position	5-62
5-29	Comparison of DOT Calculated and Experimental Bare Dysprosium Activation Data as a Function of Radial Position	5-63

LIST OF ILLUSTRATIONS

<u>Figure</u>		<u>Page</u>
5-30	Comparison of DOT Calculated and Experimental Dysprosium ($E < 0.4$ eV) Thermal Neutron Flux as a Function of Axial Position	5-64
5-31	Comparison of DOT Calculated and Experimental Dysprosium ($E < 0.4$ eV) Thermal Neutron Flux as a Function of Axial Position	5-65
5-32	Comparison of DOT Calculated and Experimental Dysprosium ($E < 0.4$ eV) Thermal Neutron Flux as a Function of Axial Position	5-66
5-33	Comparison of DOT Calculated and Experimental Dysprosium ($E < 0.4$ eV) Thermal Neutron Flux as a Function of Axial Position	5-67
5-34	Comparison of DOT Calculated and Experimental Dysprosium ($E < 0.4$ eV) Thermal Neutron Flux as a Function of Axial Position	5-68
5-35	Comparison of DOT Calculated and Experimental Dysprosium ($E < 0.4$ eV) Thermal Neutron Flux as a Function of Axial Position	5-69
5-36	Comparison of DOT Calculated and Experimental Dysprosium ($E < 0.4$ eV) Thermal Neutron Flux as a Function of Axial Position	5-70
5-37	Comparison of DOT Calculated and Experimental Dysprosium ($E < 0.4$ eV) Thermal Neutron Flux as a Function of Radial Position	5-71
5-38	Comparison of DOT Calculated and Experimental Dysprosium ($E < 0.4$ eV) Thermal Neutron Flux as a Function of Radial Position	5-72
5-39	Comparison of DOT Calculated and Experimental Dysprosium ($E < 0.4$ eV) Thermal Neutron Flux as a Function of Radial Position	5-73
5-40	Comparison of DOT Calculated and Experimental (Silicon Diode) Fast Neutron Absorbed Dose Rate as a Function of Axial Position	5-74
5-41	Comparison of DOT Calculated and Experimental (Silicon Diode) Fast Neutron Absorbed Dose Rate as a Function of Axial Position	5-75
5-42	Comparison of DOT Calibrated and Experimental (Silicon Diode) Fast Neutron Absorbed Dose Rate as a Function of Axial Position	5-76

LIST OF ILLUSTRATIONS

<u>Figure</u>		<u>Page</u>
5-43	Comparison of DOT Calculated and Experimental (Silicon Diode) Fast Neutron Absorbed Dose Rate as a Function of Axial Position	5-77
5-44	Comparison of DOT Calculated and Experimental (Silicon Diode) Fast Neutron Absorbed Dose Rate as a Function of Axial Position	5-78
5-45	Comparison of DOT Calculated and Experimental (Silicon Diode) Fast Neutron Absorbed Dose Rate as a Function of Axial Position	5-79
5-46	Comparison of DOT Calculated and Experimental (Silicon Diode) Fast Neutron Absorbed Dose Rate as a Function of Axial Position	5-80
5-47	Comparison of Calculated and Experimental (Silicon Diode) Fast Neutron Absorbed Dose Rate as a Function of Radial Position	5-81
5-48	Comparison of DOT Calculated and Experimental (Silicon Diode) Fast Neutron Absorbed Dose Rate as a Function of Radial Position	5-82
5-49	Comparison of DOT Calculated and Experimental (CaF ₂ TLD) Gamma Absorbed Dose Rate as a Function of Axial Position	5-83
5-50	Comparison of DOT Calculated and Experimental (CaF ₂ TLD) Gamma Absorbed Dose Rate as a Function of Axial Position	5-84
5-51	Comparison of DOT Calculated and Experimental (CaF ₂ TLD) Gamma Absorbed Dose Rate as a Function of Axial Position	5-85
5-52	Comparison of DOT Calculated and Experimental (CaF ₂ TLD) Gamma Absorbed Dose Rate as a Function of Axial Position	5-86
5-53	Comparison of DOT Calculated and Experimental (CaF ₂ TLD) Gamma Absorbed Dose Rate as a Function of Axial Position	5-87
5-54	Comparison of DOT Calculated and Experimental (CaF ₂ TLD) Gamma Absorbed Dose Rate as a Function of Axial Position	5-88
5-55	Comparison of DOT Calculated and Experimental (CaF ₂ TLD) Gamma Absorbed Dose Rate as a Function of Radial Position	5-89
5-56	Comparison of DOT Calculated and Experimental (CaF ₂ TLD) Gamma Absorbed Dose Rate as a Function of Radial Position	5-90

LIST OF ILLUSTRATIONS

<u>Figure</u>		<u>Page</u>
5-57	Comparison of Calculated and Experimental Carbon Photon Heating Rate as a Function of Axial Position	5-91
5-58	Comparison of Calculated and Experimental Carbon Photon Heating Rate as a Function of Axial Position	5-92
5-59	Comparison of Calculated and Experimental Carbon Photon Heating Rate as a Function of Axial Position	5-93
5-60	Comparison of Calculated and Experimental Carbon Photon Heating Rate as a Function of Axial Position	5-94
5-61	Comparison of Calculated and Experimental Carbon Photon Heating Rate as a Function of Axial Position	5-95
5-62	Comparison of Calculated and Experimental Carbon Photon Heating Rate as a Function of Radial Position	5-96
5-63	Comparison of Calculated and Experimental Stainless Steel Photon Heating Rate as a Function of Axial Position	5-97
5-64	Comparison of Calculated and Experimental Stainless Steel Photon Heating Rate as a Function of Axial Position	5-98
5-65	Comparison of Calculated and Experimental Stainless Steel Photon Heating Rate as a Function of Axial Position	5-99
5-66	Comparison of Calculated and Experimental Stainless Steel Photon Heating Rate as a Function of Axial Position	5-100
5-67	Comparison of Calculated and Experimental Stainless Steel Photon Heating Rate as a Function of Axial Position	5-101
5-68	Comparison of Calculated and Experimental Stainless Steel Photon Heating Rate as a Function of Radial Position	5-102
5-69	Comparison of Calculated and Experimental Beryllium Photon Heating Rate as a Function of Axial Position	5-103

LIST OF ILLUSTRATIONS

<u>Figure</u>		<u>Page</u>
5-70	Comparison of Calculated and Experimental Beryllium Photon Heating Rate as a Function of Axial Position	5-104
5-71	Comparison of Calculated and Experimental Beryllium Photon Heating Rate as a Function of Radial Position	5-105
5-72	Comparison of Calculated and Experimental Aluminum Photon Heating Rate as a Function of Axial Position	5-106
5-73	Comparison of Calculated and Experimental Aluminum Photon Heating Rate as a Function of Axial Position	5-107
5-74	Comparison of Calculated and Experimental Aluminum Photon Heating Rate as a Function of Radial Position	5-108
5-75	Probability Plot of Sulfur Data	5-109
5-76	Probability Plot of U-238 Data in the Core	5-110
5-77	Probability Plot of U-238 Data in the Reflector	5-111
5-78	Probability Plot of Bare Dysprosium Data in the Core	5-112
5-79	Probability Plot of Bare Dysprosium Data in the Reflector	5-113
5-80	Probability Plot of Dysprosium ($E < 0.4$ eV) Data in the Core	5-114
5-81	Probability Plot of Dysprosium ($E < 0.4$ eV) Data in the Reflector	5-115
5-82	Probability Plot of Neutron Dose Rate Data	5-116
5-83	Probability Plot of Gamma Ray Dose Rate Data	5-117
5-84	Probability Plot of Carbon Heating Rate Data	5-118
5-85	Probability Plot of Stainless Steel Heating Rate Data	5-119
5-86	Probability Plot of Beryllium Heating Rate Data	5-120
5-87	Probability Plot of Aluminum Heating Rate Data	5-121

LIST OF ILLUSTRATIONS (Continued)

<u>Figure</u>		<u>Page</u>
A-1	Detail of Modified Fuel Elements	A-2
A-2	Fuel Element with Large Dosimeter Slots	A-3
A-3	Fuel Element with Small Dosimeter Slots	A-4
A-4	Beryllium Rod with Dosimetry Slots	A-6
A-5	Beryllium Slugs	A-7

BLANK

LIST OF TABLES

<u>Table</u>		<u>Page</u>
4-1	Passive Dosimeters	4-3
4-2	Constants Used in Data Reduction	4-5
4-3	Sources of Random and Biasing Errors	4-7
4-4	Sources of Systematic Errors in Absolute Activity and Flux	4-8
4-5	TLD and Phylatron Measurement Uncertainties	4-10
5-1	PAX-G1 Atom Densities, (Atoms/barn-cm)	5-29
5-2	Radial (R) and Axial (Z) DOT Mesh	5-30
5-3	Effective Cross Sections Employed to Obtain an Equivalent Flux from Activation Data	5-32
5-4	Comparison of Measured and Calculated Radiation Heating in the Beryllium Mass in the PAX-G1 Reflector	5-33
5-5	Statistical Analysis of the Difference between Calculated and Experimental Data	5-34

1.0 INTRODUCTION

1.1 TEST IDENTIFICATION

1.1.1 Type of Report

This report is a complete Final Report.

1.1.2 Scope of Test Document

The applicable Scope of Test Document Number is M66DW-270-1.

1.1.3 Functional Category of Test

This is a Development test.

1.1.4 Contractor

The Westinghouse Astronuclear Laboratory (WANL/14683) under Subcontract NP-1, CY' 69, is the contractor.

1.1.5 Test Facility

These tests were performed at the Westinghouse Astronuclear Experimental Facility located at the Westinghouse Waltz Mill Site, July 3 to September 30, 1969.

1.2 PURPOSE

The purpose of this experiment was to:

1. Experimentally measure the typical radiation environment for the following NERVA reactor components in the PAX reactor:

- A. Fuel elements, EC 677566.
- B. Support stem and miscellaneous cluster hardware, EC 677558
- C. Core periphery, EC 677564
- D. Reflector, control drums, and axial support, EC 677559
- E. Lateral support, EC 677563

2. Provide experimental data for comparison with analytical predictions in order to generate normalization factors and standard deviations for use in the radiation design efforts for the R-1 design. These experiments are being performed to satisfy, in part, the requirements of paragraphs 3.1 and 4.1 of the applicable specifications for the preceeding engineering components.

2.0 REFERENCES AND ASSUMPTIONS

2.1 REFERENCES

2.1.1 Specification

EC 677566 - Part I. Performance/Design and Qualification Requirements
Fuel Element for 677555A NERVA Nuclear Subsystem, June, 1968.

EC 677558 - Part I. Performance/Design and Qualification Requirements Support
Stem and Cluster Miscellaneous Hardware for 677555A NERVA Nuclear Subsystem, June, 1968.

EC 677564 - Part I. Performance/Design and Qualification Requirements
Core Periphery for 677555A NERVA Nuclear Subsystem, June, 1968.

EC 677559 - Part I. Performance/Design and Qualification Requirements Reflector,
Control Drums and Axial Support for 677555A NERVA Nuclear Subsystem, June, 1968.

EC 677563 - Part I. Performance/Design and Qualification Requirements
Lateral Support for 677555A NERVA Nuclear Subsystem, June, 1968.

2.1.2 Data Item Description

Form 9 Data Item Description T-119, dated 4/2/69

2.1.3 Scope of Test Document

M66DW-270-1 - Scope of Test for Experimentally Determining the Radiation
Environment in Various NERVA Reactor Components

2.1.4 Drawing

711J496 - Reactor/Vessel Assembly, R-1 PAX

711J498B - Segment, Reflector Assembly, R-1 PAX

939J685A - NERVA R-1 Conceptual Reference Design

2.1.5 TME

WANL-TME-1798, Volume 6 - NERVA Engineering Data and Design Concepts -
Summary & Recommendations

WANL-TME-1852, Volume 1 - R-1 Parametric Shield Measurements Conducted
on the NERVA Permanent Assembly Experiment (PAX) Reactor

WANL-TME-1911 - PAX-G0 Power Distribution Measurements

WANL-TME-1915 - Radiation Environment on CHESH Components

WANL-TME-1917 - PAX-G1 Gross Power Measurements

2.1.6 Data Release Memorandum

51150 - Parts Inventory in the PAX-G0 Reactor

51150A - Parts Inventory in the PAX-G0A, B, and C Reactors

51163 - Parts Inventory in the PAX-G1 Reactor

51163A - Parts Inventory in the PAX-G1A Reactor

51166 - PAX-G1 Shim Wire Inventory

51166A - PAX-G1 Shim Wire Inventory

51176 - Calibration of Thermoluminescent Dosimeters (TLDs)

51176A - Calibration of Thermoluminescent Dosimeters (TLDs)

51177 - Calibration of Bragg-Gray Heating Rate Detectors for Use at WANEF

51179 - Use of Bragg-Gray Detectors for Heating Rate Measurements

51180 - Gamma Ray Heating Rates in PAX/R-1 Core and Reflector

51187 - Passive Dosimetry Data from the Measurement of the Radiation

Environment in Various Reactor Components

51187A - Passive Dosimetry Data from the Measurement of the Radiation

Environment in Various Reactor Components

51190 - Passive Dosimetry Data from the Measurement of Thermal Flux in
Core and on Support Stems

2.2 ASSUMPTIONS

1. The passive dosimeters, the dosimeter packets, and the voids created for passive dosimeter placement do not significantly perturb the flux or dose rate being measured.
2. The PAX-G1A mockup provides an adequate simulation of the radiation environment characteristic of the graphite fueled R-1 reactor.
3. The general assumptions concerning the use of Bragg-Gray ionization chambers for heating rate measurements in a reactor environment which are discussed in DRM 51179 and DRM 51177 are valid.
4. It is assumed that the Bragg-Gray detectors as used for these measurements, if constructed of dissimilar material to that in which they are embedded, do not seriously violate the Bragg-Gray principle (see DRM 51179). As the secondary electron production in aluminum and beryllium does not differ greatly, it is very likely that the use of an aluminum detector in the beryllium reflector does not violate the Bragg-Gray principle sufficiently to introduce a significant error in results. However, the use of a stainless steel detector in the reactor core may involve some violation of the Bragg-Gray principle and result in significant uncertainty.
5. It is assumed that the voids produced by removing a fuel element from the core do not affect the material heating rate either by change in local flux level or by streaming through the void. Some experimental measurements reported herein indicate that the streaming effect is small.
6. It is assumed that the center of detection of the Bragg-Gray ionization chambers is 1.27 cm from the physical end of the outer electrode.
7. The Bragg-Gray principle would be correctly applied for the graphite detector if the surrounding media in which it is imbedded were graphite. It is assumed that the presence of U-235 in the fueled elements surrounding the detector does not violate the principle.

3.0 TEST ARTICLE CONFIGURATION

3.1 PAX-G1A REACTOR

The PAX reactor, modified as shown in Figure 3-1 and designated PAX-G1A has been used to perform standard reactor experiments during the initial phases of the R-1 design program to support the release of R-1 graphite fuel specifications and provide information on the adequacy of the control capability of the R-1 reflector system and hydrogen in the support stems. Basically, the G1A reactor has a larger core (no inner reflector) than the previous PAX assemblies and a 60-degree mockup of the R-1 18-control drum configuration in the reflector.

The present R-1 reactor core design contains 1878 fueled and 349 unfueled elements. The PAX/R-1 also contains a total of 2227 elements, but the number of fueled elements is greater. This R-1 design calls for uranium loadings as high as 630 mg/cm^3 . In order to mockup the total uranium content using available fuel (loadings less than 500 mg/cm^3), approximately one-half of the central elements in PAX/R-1 are fueled. These are made from existing fuel elements by drilling and tapping the ends. To accommodate the increase in the total number of elements, an extension to the core support plate is inserted into the support plate region of the PAX/R-1 as seen in Figure 3-1. Approximately 245 kg of highly enriched uranium are contained in the PAX-G1A core.

The PAX-G1A reflector is obtained by replacing two 30-degree KIWI sectors of the 12-drum NRX configuration with a 60-degree R-1 reflector mockup segment. This reflector mockup segment is constructed of laminated beryllium slabs. The non-beryllium fraction in this segment is the same as that of the present R-1 reflector. The lateral support hardware is mocked up in the plunger holes by the same amount of material as in R-1. This reflector segment contains three control drums whose vanes are Al-B¹⁰, 100 mils thick, with a B¹⁰ surface concentration of 0.13 gram per square centimeter. The R-1 mockup control drums have a vane span of 120 degrees.

The PAX-G1A reactor build (including shim wire inventory, parts inventory, fuel element listing, and/or representative as-built weights) are detailed in Drawings 711J496 and 711J498B, and DRMs 51166, 51166A, 51163, 51163A, 51150 and 51150A.

612397-3

Figure 3-1. PAX Assembly for R-1 (U)

3.2 REACTOR MODIFICATIONS FOR PASSIVE DOSIMETRY

A number of changes were made in the reactor in order to provide spaces in which to insert passive dosimeters.

1. Five fuel clusters were equipped with one modified fuel element and a modified cluster block in each. The fuel elements were at locations Q11B, 4E5C, 4H7B, 4J8C and 4P7E. All were code 99 except the one in location 4P7E which was code 59.
2. A special filler strip was used in place of the regular filler strip number 21.
3. Slotted beryllium rods were used in two tie bolt holes in the beryllium reflector.
4. Beryllium slugs were used to fill two lateral support holes in the beryllium reflector near the reactor midplane.

Figure 3-2 shows the locations in the reactor of the above special hardware. Appendix A describes the pieces in detail.

3.3 REACTOR MODIFICATIONS FOR ACTIVE DOSIMETRY

1. Beryllium slugs similar to those described in Appendix A, Section A.5 were used in the two lateral support holes in front of the Bragg-Gray chambers as shown in Figure 3-3.
2. Fuel element segments of code 9 fuel were used for some experiments to partially fill the fuel element void in which Bragg-Gray measurements were being taken.
3. The traverse mechanism used to perform axial traverses of Bragg-Gray chambers in fuel element voids in the core and in the tie bolt holes in the reflector is shown in Figure 3-4.

Figure 3-2. Location of Dosimetry Positions in R-1 Test Sector

Figure 3-3. Positions of Bragg-Gray Detectors and Be Slug in Radial Heating Rate Measurements in the Reflector

Figure 3-4. Traverse System Employed on In-Core and Reflector Tie Rod Hole Measurements

4.0 SUMMARY OF TESTS

4.1 NARRATIVE

4.1.1 Test Description

The radiation environment was measured in the core and reflector regions of the PAX-G1A reactor with both active and passive dosimetry. Axial traverses were made at five fuel element locations, one filler strip (passive dosimetry only) and two tie bolt holes as shown in Figure 3-2. An axial traverse was also made on the pressure vessel surface for a few passive dosimeter types. Radial traverses were made through the reflector near the core midplane at two azimuthal locations. All measurements were made at azimuths near the center of the R-1 90 degree mockup sector.

The passive dosimeters consisted of Dy-Al foils to measure thermal neutron flux; sulfur, U-238, and thorium foils to measure fast neutron flux; thermoluminescent dosimeters to measure gamma dose rate; and Phylatrons to measure fast neutron dose rate.

The active dosimeters were Bragg-Gray chambers which are used to measure gamma heating rates. They were all of basically similar design but were constructed of four different wall materials: graphite, stainless steel, beryllium, and aluminum.

The active measurements fall into three general categories:

1. Graphite and stainless steel detectors were employed to perform measurements in the reactor core by insertion of the Bragg-Gray detectors in a void made by removing a specific fuel element, the locations and radii of which are shown in Figure 3-2. The measurements were made as a function of detector elevation above the core midplane using the remote control traverse mechanism shown in Figure 3-4 to move the detectors and station them at fixed locations.

- 1A. An additional set of heating rate measurements with the graphite chamber was made in the central fuel element void Q11B with the void filled with various lengths of code 9 fuel to investigate the effects of possible streaming through the void which could increase the measured heating rate.

2. Axial traverse measurements were performed in the reflector tie bolt holes T-1 and T-2 at locations shown in Figure 3-2, and as a function of detector elevation, employing the same traverse mechanism. These measurements were made with aluminum and beryllium detectors.

3. Heating rate measurements were made with aluminum and beryllium detectors near the core midplane in the lateral support spring plunger holes S-1 and S-2 of the reflector as a function of radial position as shown in Figure 3-3.

The measurements were performed with the core shimmed to a cold critical drum bank of $115^{\circ} \pm 2^{\circ}$. The five drums nearest the test sector were held at angles of $115.0^{\circ} \pm 0.1^{\circ}$ during each run.

4.1.2 Passive Dosimetry

4.1.2.1 Dosimeter Types

The types and sizes of passive dosimeters used are presented in Table 4-1. A more detailed description of each type appears in Appendix A of WANL-TME-1915.

The Dy-Al and the thorium foils were irradiated simultaneously in the same cadmium covers and were separated from each other by a layer of aluminum foil.

4.1.2.2 Dosimeter Calibrations

The passive dosimeters (Dy, S, U-238 and Th) which were counted on the beta or gamma detector instrumentation required calibration factors to enable count rates to be converted to activities and/or fluxes. In each case, this was accomplished by counting one or more irradiated dosimeters on the WANEF instrumentation and then sending them out of the facility for chemical or activation analysis to have their absolute activity determined. The documentation of these calibrations is given in Appendix B. The accuracies obtained are included in the uncertainty analysis described in Section 4.1.2.5. The procedures used for the calibrations are described in more detail in WANL-TME-1915, Appendix A.

The calibration procedures for the TLD chips and the Phylatrons are also described in WANL-TME-1915, Appendix A. A more detailed description of the TLD calibration appears in DRM 51176 and DRM 51176A.

TABLE 4-1
PASSIVE DOSIMETERS

Type	Length (l) Thickness (t) (Inches)	Diameter (Inches)	Average Weight (Grams)	Radiation Quantity Measured	B = Bare C = Cd-covered P = Plastic-covered
Dy-Al Foils ⁽¹⁾	0.010 t	1/4	0.0232	Thermal Neutron Flux ($E < 0.4$ eV)	B, C
S Pellets	0.266 t	0.43	1.13	Fast Neutron Flux ($E > 2.9$ MeV)	B
U-238 Foils ⁽²⁾	0.005 t	1/4	0.0729	Fast Neutron Flux ($E > 1.5$ MeV)	C
Th-232 Foils	0.004 t	1/4	0.0437	Fast Neutron Flux ($E > 1.35$ MeV)	C
CaF ₂ TLD ⁽³⁾	1/16 t	1/8 - square	0.0476	Gamma Dose Rate	P
LiF TLD ⁽³⁾	1/16 t	1/8 - square	0.0399	Gamma Dose Rate	P
Phylatron	0.290 l	0.115 Max.	0.213	Fast Neutron Dose Rate	B

(1) 5.87 weight percent dysprosium in an aluminum matrix. (See Appendix A of WANL-TME-1915.)

(2) 190 ppm U-235.

(3) Hot-pressed chips.

4.1.2.3 Positioning Techniques and Accuracies

The bare dysprosium foils which were inserted in the fuel elements and the filler strip were taped between a disc of light cardboard and a pellet of graphite to make a close fit into the dosimeter slots. The Dy foils in the beryllium rods were taped to one side of the dosimeter slot.

The cadmium-covered dysprosium and thorium foils were irradiated together inside the same cadmium box separated from each other by aluminum foil. The cadmium covers were taped closed and made a close fit into the dosimeter slots in the fuel elements, filler strip, and beryllium rods. The cadmium covers are described in Appendix A of WANL-TME-1915.

The cadmium-covered uranium foils were treated similarly but were not combined with any other foils for irradiation.

The TLDs were irradiated in Tygon packets as described in Appendix A of WANL-TME-1915. The TLDs and Phylatrons were irradiated together.

The various passive dosimeter measurements in the lateral support holes were inserted into the recesses in the beryllium slugs. The slugs were then taped together and the dosimeters in the outer recesses of the assembly were taped in place.

The accuracy of placement of any of the dosimeters is known to within ± 0.2 cm in any direction as referenced from the bottom and from the centerline of the PAX core. The measured positions are reported in the data tables.

4.1.2.4 Data Reduction Techniques

The data reduction techniques used are described in Section 4.1.2.2 of WANL-TME-1915. The cross-sections and certain other constants used in the data reduction are shown in Table 4-2.

TABLE 4-2. CONSTANTS USED IN DATA REDUCTION

	$\text{Dy}^{164} \longrightarrow \text{Dy}^{165}$		$\text{S}^{32} \longrightarrow \text{P}^{32}$		U^{238}
Atomic Number	66	66	16	15	92
Isotopic Weight ¹⁾	163.9392		31.97207		238.0508
Abundance ¹⁾	0.2818		0.950		
Half Life ¹⁾		141 min.		14.3 days	
Decay Constant		$4.9159 \times 10^{-3} \text{ min.}^{-1}$		$3.3661 \times 10^{-5} \text{ min.}^{-1}$	
Thermal Cross-Section ²⁾	2700 barns + 7.4%				
Fast Cross-Section			0.300 barns		0.555 barns
Effective Threshold Energy			2.9 Mev		1.5 MeV

1. Chart of Nuclides, July 1966
2. BNL 325, 2nd Edition, Supp. 2

4.1.2.5 Uncertainty Analysis

The analysis of the counting data included an uncertainty analysis for each dosimeter based on counting statistics and uncertainties of background rates, system resolving times, dosimeter weights and counting time. In addition to these quantities, the other possible sources of uncertainty were estimated for each type of dosimeter and were inserted into the final steps of the data reduction. Tables 4-3 and 4-4 list the significant sources of uncertainty and estimated magnitudes in terms of one-sigma standard deviations in percent.

In order to more completely describe the uncertainty calculation, consider E_0 to be defined as the uncertainty calculated from the quantities listed in the first sentence of the preceding paragraph, and the quantities E_1 , E_2 , and E_3 to be defined as shown in Tables 4-3 and 4-4. Then the uncertainty calculated for the fast flux is the vector sum of E_0 , E_1 , and E_2 . The uncertainty calculated for the cadmium ratios is the vector sum of E_{0B} , E_{0C} , E_{1B} , and E_{1C} , where the subscripts, B and C, denote the bare and Cd-covered activities, respectively. Using the same subscripts, the "statistical" uncertainty calculated for the thermal flux is composed of E_{0B} , E_{0C} , E_{1B} , E_{1C} , T1 and T2, where T1 and T2 are, respectively, the percent uncertainties in the C_1 and C_2 parameters of the thermal activity calculations:

$$A_T = C_1 [A_B - (C_2) A_C]$$

where C_1 and C_2 are the first order correction factors relating to perturbations of the sub-cadmium and epi-cadmium flux by the foils and the cadmium covers, and where A_B and A_C are, respectively, the bare and Cd-covered activities. The "total" uncertainty calculated for the thermal flux is the vector sum of the "statistical" uncertainty described above plus the uncertainties E_{2B} and E_{3B} .

TABLE 4-3
SOURCES OF RANDOM AND BIASING UNCERTAINTIES

Source of Uncertainty	One-sigma Uncertainty in Percent			
	Dy	S	U-238	Th
Spectral Dependent Variations of Cross Section	1.0	7.0	5.0	5.0
Variations in Alloy Content	2.0	--	--	--
Counter Inter-Normalization	2.0	1.0	5.0	5.0
Contributions from Unwanted Reactions	--	0.5	5.0	5.0
TOTAL	E1	3.0	7.1	8.6

TABLE 4-4
SOURCES OF SYSTEMATIC UNCERTAINTIES IN ABSOLUTE ACTIVITY AND FLUX

		One-sigma Uncertainty in Percent				
Source	Detector Type	Dy	S	U-238	Th	
Isotopic Abundance		0.1	0.1	-	-	
Alloy Content		7.0	-	-	-	
Normalizer Count Rate		1.0	1.0	1.0	1.0	
Saturation Exposure Calculation		0.1	1.0	-	-	
Reactor Power		6.0	6.0	6.0	6.0	
Absolute Counting Efficiency of Normalizer		6.0	6.0	5.0	7.0	
Contribution of Unwanted Activities		-	1.0	-	-	
SUBTOTAL		E2	11.0	8.7	7.9	9.3
Cross Section (For Flux Computation Only)		E3	7.4	7.0	10.0	10.0
TOTAL			13.3	11.1	12.7	13.6

The accuracy of the TLD and Phylatron measurements have also been estimated as shown in Table 4-5. This table summarizes the principal sources of uncertainty on the basis of one-sigma estimates given in percent. The total uncertainty is in the range of ± 16 percent to ± 19 percent for the TLD's and $(+ 23 \text{ percent } \pm 5 \text{ rads})$ for the Phylatrons.

4.1.3 Active Dosimetry

4.1.3.1 Method

The gamma radiation heating rates at various locations in the PAX-G1A core and R-1 sector mockup reflector were measured by means of 4 different Bragg-Gray type ionization chambers with walls of graphite, beryllium, aluminum, and stainless steel. A general description of this instrument and the theory of its operation as a gamma heating rate indicator is given in DRM 51179. Briefly, the Bragg-Gray ionization chamber measures the heating rate primarily due to gamma rays in the specific material of which it is made in a layer of the order of 1-4 mm thickness around the detector cavity.

4.1.3.2 Calibration

The Bragg-Gray calibration constants K_m were obtained by exposing the chamber to a known (traceable to NBS radiation standards) gamma ray field from a Co-60 source as described in DRM 51177. The calibration constants of the detectors employed in these measurements are given in DRM 51180.

4.1.3.3 Positioning Techniques and Accuracies

Positioning of the Bragg-Gray chambers for the axial traverses in core and reflector was accomplished by means of the traverse system shown in Figure 3-4.

The uncertainty in axial position, including uncertainty in the center of detection, is of the order of ± 1 centimeter as determined from visual observation of the operation of the traverse system.

TABLE 4-5
TLD AND PHYLATRON MEASUREMENT UNCERTAINTIES

Source of Uncertainty	One-Sigma Uncertainty in Percent	
	TLD	Phylatron
Reproducibility	5	2
Uniformity	12	10
Power	6	6
Calibration	6.4*	20
Readout Accuracy	1 to 10	5** (rads)
TOTAL	16 to 19***	23% + 5** (rads)

* See DRM 51176A.

** Absolute uncertainties [rads (tissue)]

*** An additional uncertainty results from use in a mixed gamma neutron field as a direct consequence of the 3 mole percent Mn:F₂ contained in the CaF₂:Mn TLD's. It is felt, however, that an upper limit of approximately 30 percent (due to neutron capture by the Mn) would be a reasonable assumption for these measurements.

Positioning for the radial traverse in the reflector is shown in Figure 3-3.

The uncertainty in radial position is approximately ± 0.25 centimeters.

4.1.3.4 Data Reduction Techniques

The data reduction consists of converting the ionization chamber current readings to the form of the absorbed gamma ray dose rate in the detector material in $\text{Rads}_m/\text{hr-watt}$, or watts/gm-megawatt as described in WANL-TME-1915, Section 4.1.3.4.

4.1.3.5 Uncertainty Analysis

The uncertainty in these measurement results from the vectorial sum of uncertainties in detector calibration (about 6%), uncertainty in current reading with the picoammeter from manufacturer's specifications (approximately 1%), and uncertainties in the reactor power (approximately 6%). Assuming that these uncertainties are independent, the vector sum of the results is ($\pm 8.5\%$). In addition, there is the uncertainty due to axial positioning errors of ± 1 cm. This can be estimated for each data point from the slope of the experimental results as shown on the graphs.

4.1.4 Power Level Calibrations

The procedures used for determining the reactor power levels are described in Appendix B of WANL-TME-1915.

4.2 TEST RESULTS

4.2.1 General

The results of the measurements are presented in graphical form in this document. Complete tables of the results are reported in DRMs 51187 and 51187A. Positions are reported in terms of the radius in centimeters from the centerline of the reactor, the distance in centimeters above the bottom of the PAX core, and the reactor azimuthal angle in degrees referenced from a theta-zero halfway between PAX drums 1 and 2, (see Figure 4-1). This choice of theta-zero was chosen to confirm to usage by the Nuclear Design group in their analytical codes.

The thermal and fast neutron data are presented as neutron flux density in units of neutrons per centimeter-squared per second per watt. For the fast neutron flux density the flux is given above an energy of 2.9 MeV for sulfur and above 1.5 MeV for U-238. Bare and cadmium-covered Dy activities are presented for the Dy-Al measurements in units of disintegrations per second per gram of nuclide per watt of reactor power, and the difference is presented as thermal neutron flux density for an energy less than 0.4 ev. The cadmium ratio, defined as the ratio of the bare to the cadmium-covered activity, is also presented.

The TLD data are presented as dose rate in rads(C) per hour per watt.

The Phylatron data are presented as dose rate in rads(tissue) per hour per watt.

Estimates of uncertainty are presented for each data point in terms of percent standard deviation on a one-sigma basis.

For each type of passive dosimetry measurement, four graphs are presented. The first is a radial plot of all the data points for that dosimeter. The second is a radial plot of all the data from near the reactor midplane only, (that is, at a distance of 66.04 cm above the bottom of the fuel in the core and tie bolts, and 70.64 cm in the filler strip, and 73.28 cm in the lateral support holes above the same reference.) The third plot is a more detailed distribution of data in the midplane of the reflector, (2 points from the tie bolt measurements, and 10 points from the lateral support hole measurements). The fourth plot of each dosimeter type presents all the data, except the lateral support hole data, as axial traverses.

Additional thermal flux measurements with Dy-Al dosimeters were performed at a later date in order to check the validity of the data reported herein and the validity of the data reported in WANL-TME-1915. These additional measurements are reported in DRM-51190 and the results are in general agreement with and confirm the validity of the data in both this report and WANL-TME-1915.

4.2.2 Dysprosium-Aluminum Foils

Figures 4-2 and 4-5 show the activity of the bare Dy-Al foils. Figures 4-6 to 4-9 show the activity of the cadmium-covered Dy-Al foils. Figures 4-10 to 4-13 show the thermal flux obtained from Dy-Al foils by assuming a thermal activity equal to the difference of the bare and cadmium-covered activities. Figures 4-14 to 4-17 show the cadmium ratios of the Dy-Al foils calculated as the ratio of the bare activity to the cadmium-covered activity.

4.2.3 Sulfur Pellets

Figures 4-18 to 4-21 show the fast flux above 2.9 MeV obtained from sulfur pellets.

4.2.4 U-238 Foils

Figures 4-22 to 4-25 show the fast flux above 1.5 MeV obtained from U-238 foils.

4.2.5 Thorium Foils

The data obtained from the thorium foils were inconsistent with other fast flux data and had larger uncertainties. Hence, they will not be reported.

4.2.6 Comparisons of the Sulfur and Uranium-238 Measurements

Figures 4-26 to 4-28 show comparisons of the sulfur and uranium-238 measurements for the midplane radial, reflector midplane and the axial distribution in the central cluster. Based on these comparisons, it is apparent that there are inconsistencies in the U-238 measurement data in the outer radial reflector region and increased uncertainties in the data. An investigation of these anomalies showed that they were caused by the extremely low count rates resulting from excessive decay of the foil samples produced by attempting to run too many foils in one reactor run, thus not permitting enough time to count the foils before they decayed to a level approaching the background activity.

4.2.7 Phylatrons

Figures 4-29 to 4-32 show the fast neutron dose rates obtained from the Phylatrons.

4.2.8 Thermoluminescent Dosimeters

Figures 4-33 to 4-36 show the gamma ray dose rates obtained from CaF_2 dosimeters. The results obtained from the LiF dosimeters are not presented because they are inconsistent with the CaF_2 and Bragg-Gray chamber data due to an apparent larger impurity of Li^6 than was originally quoted by the vendor. The CaF_2 TLD results are consistently lower than the Bragg-Gray chamber gamma ray dose rate results as described below. The reasons for these differences are not fully understood. The Bragg-Gray chambers may have an unknown fast and/or thermal neutron response. However, the CaF_2 detectors are also known to have a thermal neutron and low energy gamma response but tend to increase the measured dose rate. Fading effects of the TLDs are also known to occur. All of these effects should be investigated further to help understand the measured differences.

4.2.9 Bragg-Gray Chambers

4.2.9.1 In-Core Heating Rate Measurements

The results of the heating rate measurements made as a function of elevation above the core midplane at various radii are given in Figures 4-37 to 4-41. It is noted that in all cases the measured heating rate in the graphite detector is higher than that in stainless steel. The heating rate in a material should be proportional to the value of the mass energy absorption coefficient μ_m/ρ at the energy of the incident gamma rays. At the energy of the detector calibration, 1.25 MeV, the value of μ_m/ρ for graphite is about 7% higher

than for stainless steel, (iron). At other gamma ray energies this is not true; at 50 keV the value for μ_m/ρ is about a factor of 10 higher for stainless steel than for graphite. Similarly, at 6 MeV μ_m/ρ is about 40% higher for steel. Although the details of the gamma ray spectrum in the reactor environment are not known, it is believed to cover a wide range of energies from 50 keV to 9 MeV. Hence, it would seem likely that in reactor experiments the heating rate in steel should exceed that in graphite.

Several causes may be postulated for the higher indicated heating rate in graphite. It may be due to the presence of fast neutrons in the reactor environment which can cause knock-on carbon atoms by elastic collision. However, because of the larger mass of carbon than that of the neutron, the range of the recoil carbon atom will be small and only a very few of these will be able to emerge from the graphite into the ionization chamber cavity. Those that do cause ionization in the cavity base and result in a detectable current. It would be expected that the addition to the Bragg-Gray currents from this cause would be small. Fast and thermal neutron-produced charged particle reactions in carbon or the air contained in the chamber (e.g., n, p reaction in N^{14}) may also result in additional detector current; however, the magnitude of this correction is not known.

Other causes for the high heating rate in graphite may be related to the possible violation of the Bragg-Gray principle in the high atomic weight stainless steel detector and its use in the large mass of the dissimilar graphite. Included in this possible cause is the validity of the assumption that a detector calibration at 1.25 MeV gamma ray energy is correct when employed in a reactor spectrum.

4.2.9.2 Streaming Effects in Core Voids

Measurements of the possible effects of gamma ray streaming through voids in the reactor core caused by the removal of fuel elements during the in-core measurements described above indicate that the effect of streaming on the measured heating rates is less than 2%. These results were obtained by comparing measurements made with and without partial length fuel elements placed in the void below the Bragg-Gray detector.

4.2.9.3 Reflector Heating Rates as a Function of Elevation

The results of the heating rate measurements made vertically in the reflector tie bolt holes T-1 and T-2 by means of aluminum and beryllium detectors are given in Figures 4-41 and 4-42. It is observed that the heating rate in the aluminum is consistently higher than that of beryllium and at the energy of detector calibration, 1.25 MeV, the value of the mass energy absorption coefficient for aluminum is 9.3% higher than for beryllium. Unlike the situation with graphite and stainless steel discussed above, the value for μ_m/ρ is higher for aluminum at all energies so that the measured results in the reflector are qualitatively as expected.

The results obtained in the Al-Be measurements tend to refute in a qualitative manner the discussion given in Section 4.2.9.1 on the possible fast neutron sensitivity of a graphite chamber by knock-on carbon atoms. The atomic weight of Be (9.02) is less than carbon (12.0) and thus should receive more of the fast neutron energy in an elastic collision, so that more beryllium atoms would emerge into the cavity with higher energy and result in larger ionization currents than for the case with carbon. This discussion assumes that the ionization currents resulting from recoil Al (27) and Fe (55) atoms are negligible because of their high mass. Other fast and thermal neutron-induced charged particle reactions may contribute to the detector current but the magnitude of the effect is not known.

4.2.9.4 Heating Rate Measurements in the Reflector Spring Plunger Holes S-1 and S-2 (U)

The results of heating rate measurements at three positions radially across the reflector in spring plunger holes S-1 and S-2 are given in Figures 4-43 and 4-44. As in the vertical measurements made in the reflector, the measured heating rate in the aluminum detector is slightly higher than that of beryllium. It will be noted that the physical position of the tie bolt holes T-1 and T-2 are equivalent to radial distances of 2.54 cm and 8.90 cm, respectively, from the inside of the beryllium reflector. Thus, the tie bolt hole data obtained near the core midplane, elevation of 7 cm, should be at the same radial position as the spring hole heating rates (at an equivalent radial distance of 2.54 cm and 8.90 cm from

inside surface of the beryllium reflector) and the agreement is within the relative uncertainty of the measurements.

When the reactor is at power, the poison elements of the R-1 sector control drums at 115° will be physically closer to the S-1 spring hole than the S-2. The heating rates measured in the S-1 and S-2 position, however, are the same within the error of measurement.

Figure 4-1. PAX-G1 Reflector Geometry

FIGURE 4-2 MREVRX EXPT. - BARE DY ACTIVITY RADIAL TRAVERSES

FIGURE 4-3

MREVRG EXPT. - BARE DY ACTIVITY MIDPLANE RADIAL

FIGURE 4-4 MREVRG EXPT. - BARE DY ACTIVITY REFLECTOR DETAIL

FIGURE 4-5 MREVRG EXPT. - BARE DY ACTIVITY AXIAL TRAVERSES

FIGURE 4-6

MREVC EXPT. - (CD) BY ACTIVITY RADIAL TRAVERSES

FIGURE 4-7 MREVRG EXPT. - CCD BY ACTIVITY MIDPLANE RADIAL

FIGURE 4-8

MREVRG EXPT. - (CD) BY ACTIVITY REFLECTOR DETAIL

FIGURE 4-9 MREVRG EXPT. - (CD) BY ACTIVITY AXIAL TRAVERSES

FIGURE 4-10

MREVR EXPT. - DY THERMAL FLUX RADIAL TRAVERSES

FIGURE 4-11 MREVRG EXPT. - DY THERMAL FLUX MIDPLANE RADIAL

FIGURE 4-12 MREVIC EXPT. - DY THERMAL FLUX REFLECTOR DETAIL

FIGURE 4-13

MREVRG EXPT. - DY THERMAL FLUX AXIAL TRAVERSES

FIGURE 4-14 MREVRG EXPT. - DY CADMIUM RATIO RADIAL TRAVERSES

FIGURE 4-15 MREVRG EXPT. - DY CADMIUM RATIO MIDPLANE RADIAL

FIGURE 4-16 MREVRG EXPT. - DY CADMIUM RATIO REFLECTOR DETAIL

FIGURE 4-17 MREVRG EXPT. - DY CADMIUM RATIO AXIAL TRAVERSES

FIGURE 4-18

MREVRX EXPT. - SULFUR FAST FLUX RADIAL TRAVERSES

FIGURE 4-19 MREVRG EXPT. - SULFUR FAST FLUX MIDPLANE RADIAL

FIGURE 4-20 MREVRG EXPT. - SULFUR FAST FLUX REFLECTOR DETAIL

FIGURE 4-21 MREVRG EXPT. - SULFUR FAST FLUX AXIAL TRAVERSES

FIGURE 4-22 MREVRG EXPT. - U-238 FAST FLUX RADIAL TRAVERSES

FIGURE 4-23 MREVRG EXPT. - U-238 FAST FLUX MIDPLANE RADIAL

FIGURE 4-24 MREVRG EXPT. - U-238 FAST FLUX REFLECTOR DETAIL

FIGURE 4-25 MREVRG EXPT. - U-238 FAST FLUX AXIAL TRAVERSES

FIGURE 4-26

MREVRC EXPT. - S U COMPARISON CENTRAL AXIAL

FIGURE 4-27

MREVRG EXPT. - S U COMPARISON REFLECTOR DETAIL

FIGURE 4-28 MREVRC EXPT. - S U COMPARISON MIDPLANE RADIAL

FAST NEUTRON DOSE RATE - RADS(TISSUE)/WATT-HOUR

Figure 4-29 Phylatron Fast Neutron Dose Rate - Radial Traverses

FAST NEUTRON DOSE RATE - RADS(TISSUE)/WATT-HOUR

Figure 4-30 Phylatron Fast Neutron Dose Rate - Midplane Radial

Figure 4-31 Phylatron Fast Neutron Dose Rate - Reflector Detail

Figure 4-32 Phylatron Fast Neutron Dose Rate - Axial Traverses

FIGURE 4-33

MREVRG EXPT. - CAF2 GAMMA DOSE RADIAL TRAVERSES

FIGURE 4-34

MREVR EXPT. - CAF2 GAMMA DOSE MIDPLANE RADIAL

FIGURE 4-35

MREVRC EXPT. - CAF2 GAMMA DOSE REFLECTOR DETAIL

FIGURE 4-36 MREVRG EXPT. - CAF2 GAMMA DOSE AXIAL TRAVERSES

Figure 4-37. Absorbed Gamma Dose Rates in Fuel Element Void Q11B - Radius 1.914 cm

Figure 4-38. Absorbed Gamma Dose Rates in Fuel Element Void 4E5C - Radius 22.08 cm

Figure 4-39. Absorbed Gamma Dose Rates in Fuel Element Void 4H7B Radius 34.62 cm

Figure 4-40. Absorbed Gamma Dose Rates in Fuel Element Void 4J8C - Radius 40.34 cm

Figure 4-41. Absorbed Gamma Dose Rates in Fuel Element Void 4P7E - Radius 47.16 cm

Figure 4-42. Absorbed Gamma Dose Rates as a Function of Elevation in the Reflector Tie Bolt Hole T-1 (Radius = 54.36 cm)

Figure 4-43. Absorbed Gamma Dose Rates as a Function of Elevation in the Reflector Tie Bolt Hole T-2 (Radius = 60.72 cm)

Figure 4-44. Absorbed Gamma Dose Rates as a Function of Radius in the Reflector, Position S-1

Figure 4-45. Absorbed Gamma Dose Rates as a Function of Radius in the Reflector, Position S-2

5.0 ANALYSIS OF TESTS

5.1 INTRODUCTION

This section presents a comparison of the experimentally measured and analytically calculated radiation environment in the core, lateral support, and reflector regions of the test sector* in the PAX-G1 reactor. The radiation environment compared consists of the fission distribution, neutron fluxes (Sulfur and U-238 activation data), thermal neutron fluxes (Dysprosium activation data), fast neutron dose rates (p-i-n Si diode data), photon heating rates (Bragg-Gray ion chambers data) for carbon, stainless steel, beryllium, and aluminum, and photon dose rates (CaF₂ thermoluminescent detector data). These data are presented as a function of axial location at various radial positions and midplane data which have been cross-plotted to obtain the comparison as a function of radius.

The geometrical model and atomic number densities employed in the calculations are also presented along with the method of analysis used in obtaining the calculated results. Finally, a statistical analysis of the difference between calculated and experimental data is given along with a discussion of possible reasons for the differences between the two sets of data. The statistical analysis of the data provides a set of uncertainty factors to be applied to calculated radiation data.

5.2 ANALYTICAL MODEL

Figure 5-1 displays the following information for the PAX-G1 analytical model:

1. Region number,
2. Region dimensions (cm), and
3. Spatial mesh intervals employed in the calculations.

The axial reference plane ($Z=0$) employed in this figure is located at the dome end of the pressure vessel with the positive portion of the Z-axis orientated towards the aft end of the reactor. This is the same coordinate system employed in the two-dimensional calculations.

* The test sector is that portion of the reactor which had been modified to closely resemble the R-1 reactor configuration given by WANL Drawing 939J685A.

However, the coordinate system employed in comparing calculated and experimental data has the axial reference plane ($Z=0$) located at the inlet end of the fuel with the same orientation of the positive portion of the Z-axis.

Table 5-1 provides the following information for the PAX-G1 analytical model presented in Figure 5-1 for regions outside the active core:

1. Region number,
2. Region description, and
3. Atomic number densities for each element in each region.

Calculations were performed at ambient conditions with no hydrogen to conform to the PAX-G1 operation.

Table 5-2 presents the radial and axial mesh lines employed in the DOT code.

5.3 METHOD OF ANALYSIS

5.3.1 Calculational Procedure

Figure 5-2 is a schematic diagram of the computational procedure employed to obtain the analytical results presented in this report. This procedure is basically the WISDM (WANL Integrated Nuclear, Radiation and Shielding Standard Design Method) technique.⁽¹⁾

A converged, region dependent, 155 fine energy group neutron spectrum was calculated for the PAX-G1 reactor via an iterative loop employing the GAMB1T⁽²⁾-ANISN⁽³⁾-ANISIG codes. On the first iteration, GAMB1T utilizes the P_1 equations to generate 52 group, P_0 transport corrected, neutron cross sections for an infinite media spectrum. These cross sections are employed in the one-dimensional, discrete ordinates transport code ANISN, to obtain 52 neutron energy group average fluxes and currents for each reactor region. These data are interpolated by the ANISIG code to 155 neutron energy groups which then becomes the region-dependent fine energy group spectrum employed in GAMB1T to generate new spectrum averaged cross sections. This process is repeated until a converged spectrum (change in spectrum between two iterations is small)

is obtained. This converged spectrum is then employed in the GAMBIT code to generate region dependent 16 energy group P_0 , transport corrected macroscopic neutron transport and microscopic photon production cross sections. These cross sections plus a tape containing microscopic photon cross sections are input to the APPROPOS code which generates P_0 and P_1 13 group photon transport cross sections for use in the DOT⁽⁴⁾ photon code, and microscopic and macroscopic cross section libraries to be employed in the NAGS⁽⁵⁾ data processing code.

The 16 group neutron transport cross sections were input to both radial and axial ANISN transport problems. The ANISN problems provide neutron cross sections with upscatter terms removed and a radial and an axial neutron flux guess for the two-dimensional discrete ordinates transport code, DOT. The first neutron DOT problem employed an S_2 angular quadrature scheme and provided a flux guess for a neutron DOT calculation employing an S_6 angular quadrature scheme. It should be noted that the DOT calculations contained response functions which provided the activation, equivalent flux, and dose rate calculated data given in this report (see Section 5.3.2 for details).

The DOT neutron fluxes and the library data from the APPROPOS code were employed in the NAGS code to generate photon sources. These sources were then used in a fixed source, 13 energy group, full P_1 , photon DOT with an S_6 angular quadrature scheme to generate photon fluxes. The neutron and photon fluxes from the DOT calculations were then employed in the NAGS code to generate the calculated radiation heating rate information presented in this report.

5.3.2 Determination of Analytical and Experimental Quantities

As previously mentioned (Section 5.3.1), the calculated neutron activation, neutron equivalent flux, neutron dose rate, and photon dose rate were obtained directly from the DOT code via response functions while the heating rate data were obtained from the NAGS code. This section contains a discussion of: (1) the relationship between activation and equivalent flux, (2) the relationship between the calculated and experimental data, and (3) the response functions employed to obtain the calculated data from DOT.

5.3.2.1 Activation and Equivalent Neutron Flux Data

Basically, a foil is irradiated at a constant power level for a period of time in a neutron flux, $\Phi^r(E)$, to produce a radioactive nuclide(s) that emits β particles or γ rays. The foil is then placed in a nuclear counter that measures the β or γ decay rate of the radioactive nuclide(s). Through the use of basic production and decay relationships, the measured decay rate is translated into the rate of production of the radioactive nuclide which existed in the reactor. This rate of production is commonly referred to as the saturated activity, A^{e**} , with units of disintegrations per unit volume per second (also may have the units of dis/gm-sec).

The DOT code calculates the saturated activity, A^{c***} , through the use of the following analytical expression:

$$A^c = \int_0^{10 \text{ Mev}} \Sigma^c(E) \Phi^c(E) dE \quad \text{where,} \quad (1)$$

$\Sigma^c(E)$ = The calculated neutron cross section (response function) for the production of the radioactive nuclide.

$\Phi^c(E)$ = The calculated differential energy flux (total flux per unit energy).

Both the experimental and the calculated data are reduced to an equivalent flux, $\bar{\Phi}$, by employing the same effective cross section, $\bar{\Sigma}$, for the nuclear reaction under consideration. The following expressions represent this reduction to equivalent flux:

$$\begin{aligned} \bar{\Phi}^e &= \text{Experimentally Determined Equivalent Neutron Flux} \\ &= A^e / \bar{\Sigma} \end{aligned} \quad (2)$$

$$\begin{aligned} \bar{\Phi}^c &= \text{Calculated Equivalent Neutron Flux} \\ &= A^c / \bar{\Sigma} \end{aligned} \quad (3)$$

From these relationships, it can be seen that basically only a saturated activity

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- * The superscript, r, denotes the physically real, or actual, quantity.
 - ** The superscript, e, denotes an experimentally measured quantity.
 - *** The superscript, c, denotes a calculated quantity.

has been determined which is translated into the more familiar units of a flux. The conversions in equations (2) and (3) yield the actual neutron flux if, and only if, the following relationship could be employed to obtain $\bar{\Sigma}$:

$$\bar{\Sigma} = \frac{\int \Sigma(E) \Phi(E) dE}{\int \Phi(E) dE} \quad (4)$$

This relationship is possible to evaluate analytically; however, experimentally this relationship cannot be evaluated. Therefore, the same estimate of the effective cross section, $\bar{\Sigma}$, has been employed in converting both the analytical and experimental data to an equivalent flux. Table 5-3 displays the effective cross sections employed to convert calculated and experimental neutron activation data to an equivalent flux.

For purposes of brevity, Φ^e and Φ^c will be referred to in the remainder of this report as a flux rather than the equivalent neutron flux. It should also be noted that the neutron detectors employed to measure the neutron flux have a threshold or cutoff in their response such that the cross section for the production of the radioactive nuclide(s) is nearly zero over a particular energy range. Therefore, the flux will also be referred to as the fast flux ($E > 2.9$ Mev), the thermal flux ($E < 0.4$ ev), etc.

In comparing the calculated and experimentally determined fluxes, it should be remembered that:

- 1) the experimental flux, Φ^e , may differ from the real flux, Φ^r , because of experimental uncertainties, and
- 2) the calculated equivalent flux, Φ^c , may differ from the experimental flux, Φ^e , not because of a discrepancy in the calculation of the flux itself, but, because of a discrepancy in the calculated cross section, $\Sigma^c(E)$ (see equations (1) and (3) employed to obtain Φ^c).

These points will be discussed in more detail when the uncertainty factors for the calculated radiation environment are derived.

5.3.2.2 DOT Response Functions

The response functions in DOT employed to obtain the calculated data are essentially cross sections and were obtained basically from the GAMBIT and APPROPOS code cross section libraries. However, the neutron response functions had to be corrected on an energy group-wise basis for self-shielding, flux depression, and the effect of cadmium shielding on the detectors. These corrections are described in detail in Reference 6. Outlined below is a brief description of these corrections to the neutron response functions.

The resonance self-shielding of the U-238 foils was calculated with the GAMBIT code. The correction for resonance self-shielding in the isotope of Dy being measured in the Dy foils was negligible while in sulfur the resonance self-shielding is virtually non-existent.

The thermal self-shielding and thermal flux depression were determined by using the overall self-shielding factor derived by the use of the first flight transport approximation for slab geometry⁽⁷⁾. This overall self-shielding for an infinite slab, f_o , is expressed as:

$$f_o = \frac{1 - a}{2 X_o} \quad \text{where,} \quad (4)$$

$$X_o = \sum a t$$

$$t = \text{total thickness of foil}$$

$$\sum a = \text{macroscopic absorption cross section for the detector}$$

$$a = e^{-X_o} - X_o e^{-X_o} + X_o^2 E_1(X_o)$$

The evaluation of this expression for the U-238 and Dy foils indicated that the overall self-shielding factor was less than one percent.

In this series of experiments, cadmium covered U-238 and Dy foils were employed. Therefore, it was necessary to calculate a series of energy dependent transmission factors, $T(E)$, to determine the cadmium filtered activity in these foils. The following expression

was employed which applies to optically thin ($\sum \tau < 1$) detectors:

$$T(E) = \left\{ E_2 \left[\sum_t^f(E) t_f \right] + E_2 \left[\sum_a^f(E) t_f \right] \right\} / 2 \quad (5)$$

$$E_2(X) = \int_0^1 e^{-X/\mu} d\mu = \int_1^\infty u^{-2} e^{-Xu} du$$

$$\mu = c \cos \theta$$

$$\theta = \text{angle of incident neutrons}$$

$$\sum_t^f(E) = \text{energy dependent total macroscopic cross section of the filter}$$

$$\sum_a^f(E) = \text{energy dependent absorption macroscopic cross section of the filter, and}$$

$$t_f = \text{filter thickness, cm.}$$

Equation (5) was derived on a semi-empirical basis. While an expression of the form of equation (5) yields the correct transmission when absorption is the only process occurring, the expression which accounts for the removal of neutrons by scattering is quite complex. Quite clearly, the true transmission factor lies between an expression employing $\sum_a^f$ and an expression employing $\sum_t^f$ only. By comparing results obtained with an expression of the form of equation (5) with the experimentally measured cadmium ratios of 0.005 inch gold foils measured in the center of the FCX reactor for various thickness of cadmium, it was determined that equation (5), which is the arithmetic average of the expressions using the absorption and total cross section, yielded good agreement with the experimental measurements.

5.4 COMPARISON OF RESULTS

5.4.1 Fission Distribution Data

Figures 5-3 and 5-4 compare, respectively, radial and axial calculated and experimentally measured fission per gram of uranium data which have been normalized to unity at the center of the core. The radial data were obtained on the reactor midplane, while the axial data were obtained on the reactor centerline. The experimental results shown in these figures are reported in Reference 8.

From these two figures, it can be seen that there was good agreement between the calculated and experimental data. The radial calculation of the fission distribution tended to be approximately 7 percent low in the 30 to 40 centimeter region of the core. The axial calculation displayed no significant difference in the distribution. The accuracy in predicting the fission distribution provides a fundamental check of the calculational scheme and indicates, in a basic manner, the adequacy of the technique for radiation analysis internal to the Nuclear Subsystem.

5.4.2 Sulfur ($E > 2.9$ Mev) Fast Neutron Flux Data

Figures 5-5 through 5-13 provide a comparison of the calculated and experimentally measured neutron flux data obtained with sulfur pellets. Because the $S^{32}(n, p)$ reaction data is a threshold reaction, this data corresponds approximately to the integrated fast neutron flux with an energy greater than 2.9 Mev.

Figures 5-5 through 5-11 present a comparison of the calculated and experimental sulfur data as a function of axial position at the following radial locations:*

1. fuel element 000B ($R=1.52$ cm),
2. fuel element 4E5C ($R=21.69$ cm),
3. fuel element 4H7B ($R=34.23$ cm),
4. fuel element 4J8C ($R=39.95$ cm),
5. fuel element 4P7E ($R=46.78$ cm),
6. filler strip ($R=48.77$ cm),
7. inner reflector tie bolt hole ($R=54.31$ cm),
8. outer reflector tie bolt hole ($R=60.64$ cm),
9. outer surface of the aluminum pressure vessel.

It should be noted that as output, the DOT code produces a value, such as the flux, at the midpoint of each mesh interval in the problem. These mid mesh intervals do not of course coincide with the position at which the experimental data were obtained.

* Center of "detector" radial coordinates are indicated and varies somewhat with different detectors.

Hence, the DOT output data were numerically (not graphically) double interpolated, i.e., both in the R and Z direction, to calculate a value at exactly the same location as the position of the detector.

Figure 5-12 presents the midplane data from the preceding figures plotted as a function of radius. Figure 5-13 presents calculated and experimental sulfur data as a function of radial position in two reflector spring holes. These holes were located azimuthally on each side of a control drum in the reflector and slightly aft of the core midplane axially.

From these figures, it can be seen that the calculated shape of the sulfur data agrees well with the measured data. On an average pointwise basis (average derived in Section 5.5) the calculations are 32 percent lower than the measurements. It should be noted that similar results have been obtained in other comparisons of calculations with experiments.

The difference in absolute magnitude between the calculated and measured data may be due to: 1) the use of only one energy group above 2.9 Mev in the calculations, 2) the use of P_0 transport corrected cross sections, 3) the use of the Cranberg fission spectrum, and/or 4) a possible thermal neutron response in either impurities or another isotope of sulfur.* Each of these reasons postulated for differences between experimental and calculated data must be further systematically evaluated to isolate the cause of the difference.

5.4.3 U-238 ($E > 1.5$ Mev) Fast Neutron Flux Data

Figures 5-14 through 5-21 compare the calculated and experimentally measured neutron flux data obtained with U-238 foils. These foils contained 190 ppm of U-235. Cadmium covers were employed with the foils to partially suppress the fissioning in U-235 so that the primary activation measured was that due to fast fission in U-238. Calculated data indicate that less than 2% of the activation is due to U-235 fissions. Therefore, the U-238 foil data does correspond approximately to the integrated fast neutron flux with an energy greater than 1.5 Mev.

* The calculated results were obtained for S^{32} only. A thermal neutron response does occur in another isotope of sulfur, but the presence of this isotope has been corrected experimentally by allowing the activation due to this isotope to decay before the detectors are counted.

With the exception of the traverse on the outer surface of the pressure vessel which was not obtained experimentally, the U-238 data presented are at the same traverses as presented previously for sulfur. Therefore, Section 5.4.2 provides a detailed description which is also applicable to the U-238 data traverses.

From Figures 5-14 through 5-18, it can be seen that good agreement between calculated and experimental data was obtained in the core with the average pointwise difference being 4 percent (average difference derived in Section 5.5). However, as is clearly shown by the reflector spring hole data, Figure 5-21, the experimental data become progressively lower than the calculations as the radial distance from the core increases. This difference occurred because anomalies were encountered in the U-238 reflector experimental data. The cause of these anomalies is described in Section 4.2.6. The fact that there are experimental anomalies is illustrated by comparing the experimentally obtained sulfur and U-238 data in the spring holes shown, respectively, by Figures 5-13 and 5-21. These figures indicate that the U-238 experimental data are higher than the sulfur data at the inner radius of the reflector, but lower than the sulfur data at the outer radius of the reflector. The high energy fluxes represented by these two sets of data should have approximately the same slope. In addition, the U-238 data should be higher than the sulfur data at all locations because the U-238 data represent the integral neutron flux over a wider energy range which includes the energy range of the sulfur data. Further, experimental data obtained on the surface of the pressure vessel during the PAX-E5 series of measurements agree with current sulfur data, but are considerably higher than the current U-238 data.

In the preceding section, it was noted that the calculated sulfur data were low. For example, the calculated sulfur data which represents the fast flux ($E > 2.9$ Mev) is 22% low at the core center while the calculated U-238 data which represents the fast flux ($E > 1.5$ Mev) is 4% high at the core center. This indicates that the calculated neutron spectrum is too "soft" relative to the experimental results at high energies.

Considering: 1) the uniformity of the difference between the calculated and experimental sulfur data (see Statistical Analysis Section 5.5 for details) even in the reflector, 2) the good agreement between the calculated and experimental U-238 data in the core, and 3) the relationship between the sulfur and U-238 information, it is concluded that the calculated and experimental U-238 data in the reflector would have been in considerably better agreement provided experimental anomalies had not been encountered. Therefore, the uncertainty obtained from the statistical analysis of the U-238 data (see Section 5.5) in the core will be assumed to be applicable to the reflector also for representing the difference between calculations and experiment. Since the U-238 data most nearly represent the fast neutron flux with energies greater than 1.0 Mev which is required for the predicted internal radiation environment for use in radiation effects, the uncertainty to be assigned to the R-1 calculated fast flux $E > 1.0$ Mev will be based directly on the U-238 core data.

5.4.4 Bare Dysprosium Neutron Activation Data

Figures 5-22 through 5-29 compare the calculated and experimentally measured bare (i.e., not cadmium covered) Dy foil activation data. Since the bare Dy data does not correspond directly to the integrated neutron flux over a particular energy range, the Dy activation data were not converted to an equivalent neutron flux.

With the exception of the traverse on the outer surface of the pressure vessel, which was not experimentally measured, the Dy data were obtained along the same traverses as those for sulfur. Therefore, Section 5.4.2 provides a detailed discussion of the traverses of data presented.

From examining these figures, it can be seen that there is a significant difference between the calculated and experimental data at several locations in the core; however, there is reasonably better agreement between the two sets of data in the reflector. On the average pointwise basis (the average difference is derived in Section 5.5), the calculated data in the core is 22% low; whereas, in the reflector, the calculated data is 4% low. Currently, no explanation for this difference exists; however, as was noted in Section 5.3.2.1, the discrepancy may be due to several reasons other than a discrepancy in the calculated neutron flux.

Examination of Figure 5-29 which displays activation data in the spring holes in the reflector shows an interesting phenomenon which will be discussed in more detail in the next section on the Dy ($E < 0.4$ ev) thermal neutron flux data. As can be seen the experimental data in the two spring holes are significantly different. This occurs because one spring hole is closer azimuthally to the poison in the control vane than the other. The control vane causes a local flux depression which is reflected in the experimental data. Since the calculations were performed for an R,Z model, the azimuthal perturbation is not observed in calculated data for these traverses.

5.4.5 Dysprosium ($E < 0.4$ ev) Thermal Neutron Flux Data

Figures 5-30 through 5-39 compare the calculated and experimentally measured Dy foil data which corresponds to the integrated thermal neutron flux for energies less than 0.4 ev. These data are the difference between bare Dy data reported in the previous section and cadmium covered Dy data. Due to the Cd cutoff of neutrons, the difference in the two sets of information provides the Dy activation, and hence the equivalent flux for neutrons with energies less than 0.4 ev. The cadmium covered Dy data will not be presented in Section 5 because the bare Dy data and the difference between the bare and cadmium covered data is presented; therefore, the presentation of the cadmium covered Dy data would be redundant.

With the exception of the traverse on the outer surface of the pressure vessel, the Dy data presented are along the same traverses as those obtained for sulfur. Therefore, Section 5.4.2 provides a description of the traverses of data presented.

From examining these figures, it can be seen that there is reasonably good agreement at most locations in the core except near the core outlet. In the reflector, there is reasonably good agreement out to a radius of 54 cm. This can be seen most clearly in Figure 5-38 which presents a radial traverse of the Dy data through the spring holes in the reflector.

The difference between the calculated and experimental data in the reflector occurs for two reasons. First as was noted in the preceding section, one of the spring holes

is closer azimuthally to the poison in the control vane than the other spring hole. The control vane causes a local thermal flux depression which is larger for the hole closest to the vane and is reflected in the experimental data. Second, the calculations employed a poison ring to represent the control vanes. This ring was adjusted to yield: (1) the correct total absorption rate, (2) eigenvalue, and (3) the correct power at the edge of the core. Therefore, on the average, the poison ring should represent the control vanes. However, due to the difference between the actual vane geometry and poison ring, the shape of the local flux depression differs. In reality, neutrons can reach the outer portion of the reflector without passing through the control vanes; whereas, in the calculational model, all neutrons reaching the outer portion of the reflector must pass through the poison ring.

The following procedure was employed to test the validity of the assumption that the calculated data represented the average azimuthal thermal neutron flux. Two-dimensional, R- θ geometry, diffusion theory calculations were performed with the EXTERMINATOR⁽⁹⁾ code which explicitly represented the control vanes. From this calculation, the average azimuthal thermal neutron flux was obtained at various radial locations in the reflector. The ratio of the thermal neutron flux in the spring hole to the average azimuthal thermal neutron was determined from the EXTERMINATOR data and multiplied by the data calculated by the R, Z DOT (Figure 5-38) to obtain DOT calculated data which had been adjusted for azimuthal asymmetries. These adjusted data are presented in Figure 5-39, and it can be seen that considerably better agreement exists between the calculations and experiment. This tends to indicate that the DOT data represent the average thermal neutron flux.

Therefore, the adjusted data which are more representative of the difference between the calculated and experimental data were employed to determine the uncertainty in the reflector for the Dy ($E < 0.4$ ev) data. Since these Dy data most nearly represent the thermal neutron flux with energies less than 0.4 ev which are data required for the predicted internal radiation environment, the uncertainty in the thermal neutron flux ($E < 0.4$ ev) will be based on the Dy data.

An additional discrepancy between the experimental and calculated data

occurred. This anomaly is clearly seen in Figure 5-30 through 5-33 which are axial traverses of the experimental data at various radial locations in the core. It can be seen that the experimental data consistently do not decrease axially, but sometimes increase at the outlet end of the fuel. Since the measurements were performed without the Boral enclosure* in place at the aft end of the core, it is believed that this inconsistency is due to reflection of thermal neutrons by the test cell and its environment.

5.4.6 Fast Neutron Absorbed Dose Rate Data

Figures 5-40 through 5-48 compare the calculated and experimentally measured fast neutron absorbed dose rate obtained with silicon diodes. With the exception of the traverse on the outer surface of the pressure vessel, the fast neutron absorbed dose rate data presented are along the same traverses as those obtained for sulfur. Therefore, Section 5.4.2 provides a detailed description of the traverses of data presented.

From examining these figures, it can be seen that there is reasonable agreement between the calculated and experimental data with no significant trends in the data occurring. The average pointwise difference between the calculated and experimental data (derived in Section 5.5) over the core and reflector is 2%.

The fast neutron dose rate in Rads-Tissue/hr-watt can be related to the neutron kinetic heat deposition in tissue. No direct measurements of the neutron heating rate were made; therefore, these data will be employed to obtain the neutron kinetic heating rate uncertainty factors for all materials. This is as valid of an assumption as, and analogous to (with the same uncertainties), the assumption that the uncertainty in the fast neutron flux ($E > 1.0$ Mev) is represented by the difference between the calculated and experimentally measured U-238 foil data.

The experimental fast neutron absorbed dose rate information will be employed

* The Boral enclosure is a box-like structure containing Boral sheeting. This structure can be placed on the aft or outlet end of the PAX reactor where the boron in the boral sheeting will absorb the thermal neutron being reflected by the test cell environment.

to obtain the total heat deposition in the beryllium portion of the PAX-G1 reflector. This comparison along with the Bragg-Gray photon heating rate data is presented in Section 5.4.8.

5.4.7 Photon Absorbed Dose Rate Data

Figures 5-49 through 5-56 compare the calculated and experimentally measured gamma ray absorbed dose rate obtained with CaF_2 thermoluminescent detectors (TLD). The gamma ray absorbed dose rate data presented are along the same traverses as those obtained for sulfur. Therefore, Section 5.4.2 provides a detailed description of the traverses of data presented.

These figures show excellent agreement between the calculated and experimental data with the average pointwise difference between the two sets of information being 4% (derived in Section 5.5). This average difference was derived from the axial traverses of the experimental data.

One anomaly in the data occurs. The calculated data agree quite well with all of the axial traverses of the experimental data. However, the calculated data display a statistically significant difference, approximately 15%, from the experimental data obtained in the reflector spring holes (Figure 5-56). There are two possible explanations for this difference. First, the experimental data in the spring holes may be higher due to photon streaming through the spring holes. These spring holes were not explicitly represented in the calculation, and, therefore, any streaming which occurred would not be represented in the calculated data. The experimental data in the reflector tie bolt holes would not be subject to significant streaming because of the large L/D ratio and axial orientation of the hole. Second, the CaF_2 thermoluminescent detector may respond not only to photons but also to thermal neutrons. As was discussed in Section 5.4.5, the thermal neutron flux is subject to a local flux depression due to the control vane. In addition, the flux depression is different between the experimental and DOT calculated data. Therefore, the effect of this difference would be at a maximum in a spring hole which is close to the control vane and minimal in the tie bolt holes which are located farther away from the control vane.

It should be noted that the gamma ray absorbed dose rate in Rads-Carbon/hr-watt can be related to photon heating rate in carbon. Therefore, these data along with the Bragg-Gray photon heating rate information will be discussed in the next section.

5.4.8 Bragg-Gray Ion Chamber Photon Heating Rate Data

Figure 5-57 through 5-62 compare the calculated and experimentally measured photon heating rate for carbon in the core. Figures 5-63 through 5-68 compare similar information for stainless steel in the core. Figures 5-69 through 5-71 compare the calculated and experimentally measured photon heating rate for beryllium in the reflector. Figures 5-72 through 5-74 compare similar information for aluminum in the reflector. All of the preceding experimental data were obtained with Bragg-Gray ion chambers composed of the material being measured. It should be noted that the calculated heating rate information compared with the Bragg-Gray data was obtained from the NAGS program (see Section 5.3 and Figure 5-2), while the calculated photon dose rate data (Rads-C/hr), which are essentially a photon heating rate, were obtained directly from DOT via response function. Even though the two sets of calculated data were obtained from different portions of the calculational scheme, a comparison of the carbon data indicates that the difference between the two sets of calculated data is less than 0.5 percent at all locations in the core and reflector.

The photon heating rate data in the core is presented as a function of axial position at the following radial locations:

1. fuel element 000B ($r = 0.753$ in),
2. fuel element 4E5C ($r = 8.69$ in),
3. fuel element 4H7B ($r = 13.62$ in),
4. fuel element 4J8C ($r = 15.88$ in), and
5. fuel element 4P7E ($r = 18.57$ in).

The midplane data from the preceding axial traverses were replotted as function of radius and are also shown.

The photon heating rate data in the reflector are presented for the following traverses:

1. inner reflector tie bolt hole ($r = 21.40$ in),
2. outer reflector tie bolt hole ($r = 23.91$ in), and
3. reflector spring hole.

It should be noted that these traverses are similar to those for the sulfur data; however, the radial or axial location is given in inches rather than centimeters and the experimental axial traverses were obtained only over the aft half of the reactor.

The statistical analysis of the data (see Section 5.5) indicates that the average pointwise difference between the calculated and experimental Bragg-Gray data is as follows:

1. the calculated carbon data in the core is 17% low,
2. the calculated steel data in the core is 3% low,
3. the calculated beryllium data in the reflector is 28% low, and
4. the calculated aluminum data in the reflector is 24% low.

Comparing differences between the Bragg-Gray and CaF_2 TLD experimental data, the first difference is that the Bragg-Gray data obtained in the tie bolts in the reflector do not display a statistically significant difference from the experimental data obtained in the reflector spring holes. As was discussed in Section 5.4.7, the CaF_2 TLD experimental data did display a difference between the tie bolt and spring hole information. The second difference is that the CaF_2 TLD experimental data were generally lower than the calculations; whereas, the Bragg-Gray data are generally higher than the calculations.

The following discussion illustrates the second difference numerically. On the reactor centerline ($r = 0.753$ inches) at the core midplane, the heating rate for carbon is (1) 7.5×10^{-2} W/gm-Mw with the Bragg-Gray ion chamber, (2) 6.3×10^{-2} W/gm-Mw with the CaF_2 thermoluminescent detector (TLD), and (3) 6.6×10^{-2} W/gm-Mw which is the calculated heating rate due to photons. Randomly selecting data at another radius on the core midplane for comparison ($r = 13.7$ inches), the heating rate for carbon is (1) 6.1×10^{-2}

W/gm-Mw with the Bragg-Gray ion chamber, (2) 5.0×10^{-2} W/gm-Mw with the CaF_2 TLD, and (3) 4.9×10^{-2} W/gm-Mw which is the calculated heating rate due to photons.

From these data it can be seen that there is generally good agreement between calculations and the TLD data with the Bragg-Gray data being consistently and significantly higher than either the TLD or calculated data.

Another comparison, which displays differences between the data, is the calculated and the experimental Bragg-Gray data for carbon and stainless steel in the core. For example, on the reactor centerline at the core midplane, the heating rate in stainless steel is 6.5×10^{-2} W/gm-Mw with the Bragg Gray detector and 6.4×10^{-2} W/gm-Mw via calculations while the data for carbon is 7.5×10^{-2} W/gm-Mw with the Bragg-Gray detector and 6.6×10^{-2} W/gm-Mw via calculations. It can be seen that for stainless steel there is good agreement between calculations and the Bragg-Gray experimental data while for carbon the experimental Bragg-Gray data are significantly higher than the calculated data. Comparing the calculated and experimental Bragg-Gray data for beryllium and aluminum on the midplane in the reflector ($r = 20.9$ inches), it is found that the heating rate for beryllium is 1.7×10^{-2} W/gm-Mw with the Bragg-Gray detector and 1.4×10^{-2} W/gm-Mw via calculations, while for aluminum the heating rate is 1.6×10^{-2} W/gm-Mw with the Bragg-Gray detector and 1.2×10^{-2} W/gm-Mw via calculations.

These differences could be due to two possible causes: (1) uncertainties in the calculations or the energy absorption coefficients (cross sections) employed to obtain the calculated heating rate data and (2) a neutron response in the detectors. Since the calculated data agreed with some of the experimental results, it would seem less probable that the differences are due to uncertainties in the calculated photon fluxes. In addition, the possible inaccuracies in the energy-absorption coefficients are quite small (in order of approximately 3%). Therefore, while uncertainties in the calculated data may account for some of the differences, it seems unlikely that these could account for differences of 20 and 30 percent. Therefore, the most probable cause for these differences is a neutron response in the detectors. It should be noted that the calculated data included only heating due to photons and contained absolutely no heating due to neutrons.

Because the Bragg-Gray data has the possibility of having a neutron response, the difference between the Bragg-Gray data and calculations for the photon heating* would not be representative of the uncertainty in photon heating. Therefore, the uncertainty in photon heating will be based on the difference between the CaF_2 TLD data and calculations.

Table 5-4 provides a comparison of the measured and calculated total heat deposition in the beryllium mass in the PAX-G1 reflector. The calculated total heat deposition was obtained by integrating the heating rate distribution over the beryllium volume. The experimental total heat deposition was obtained by integrating the beryllium Bragg-Gray data for photon heating and the fast neutron absorbed dose rate for neutron heating over the beryllium reflector volume. The neutron heating for tissue, which was the nature of the experimental data, was converted to neutron heating in beryllium by multiplying the experimental result by the ratio of the calculated neutron heating in beryllium to the calculated neutron heating in tissue. The calculated data were integrated by the NAGS code to obtain the total heat deposition. Both sets of data were then adjusted to a power level of 1575 Mw thermal power in the reactor.

This comparison of the total heat deposited in the beryllium portion of the reflector was made before it was known that the Bragg-Gray detector might have a significant neutron response. The CaF_2 TLD information is the data that should be employed to determine the photon heating in beryllium. However, as was noted in Section 5.4.7, there was a difference between the TLD data measured in the reflector spring holes and the data obtained in the reflector tie bolt holes. If the TLD data in the reflector tie bolt holes were employed to obtain the experimentally determined total heat deposition in beryllium, then

* One of the conclusions of this report is that an analytical or experimental determination of the neutron response of Bragg-Gray ion chambers should be investigated. When this investigation is completed, a valid comparison between calculated and experimental Bragg-Gray data can be made. However, at present, the exact magnitude of the neutron response in these Bragg-Gray detectors is not known.

there would be no significant difference between the calculated and experimental results because of the good agreement between calculated data and the tie bolt measurements. If the TLD data in the spring holes were employed, then the difference between the calculated and experimental total heat deposition in beryllium would be similar to the difference obtained with the Bragg-Gray detectors. Therefore, the use of the Bragg-Gray data represents a conservative estimate of the uncertainty in the total heat deposition in the beryllium portion of the PAX-G1 reflector.

From Table 5-4, it can be seen that the calculated neutron heating is 3% low while the calculated photon heating is 10% low. The total calculated heat deposition is 7.8% low relative to the experimental data. It should be noted that these differences are not in conflict with the average differences previously presented. The previous differences are the pointwise average differences for midplane data (see Section 5.5); whereas, the differences in the total heat deposition is the difference integrated over the reflector region. Further, a pointwise difference always represents the upper limit on a region integrated difference which is consistent with the results given.

It must be emphasized that the data presented in Table 5-4 represent only the total heat deposition in the beryllium in the PAX-G1 reflector which has been adjusted to a power level of 1575 Mw thermal. The PAX-G1 reflector heat deposition in beryllium cannot be directly compared with the R-1 reflector total heat deposition data because (1) the R-1 employs a more massive reflector (i.e., more heat deposited) and (2) the data in Table 5-4 do not include the heat deposited in aluminum, Inconel, steel, carbon and titanium which are materials present in the R-1 reflector that would increase the total heat deposition.

5.5 STATISTICAL ANALYSIS OF THE DATA

5.5.1 Introduction

This section provides the statistical analysis of the difference between the calculated and experimental data. Derived in this section are: (1) the best functional form for

representing the difference between calculations and experiments, (2) the average difference between calculations and experiments, (3) the one sigma limits on the average difference, and (4) a statistical indication of whether the shape of the calculated and experimental data is the same.

5.5.2 Method of Analysis

The R,Z DOT calculated output data are obtained at the midpoints of the mesh intervals given in Table 5-2. It would have been fortuitous if a calculated data point occurred at the same radial and axial location as an experimental data point. Because of the steep slopes in the data, a considerable difference between the calculated and experimental data would result if the calculated data point nearest to the experimental data point was employed in determining the difference between the two sets of data. Graphical interpolation of the calculated data to the experimental data point locations is extremely tedious in two dimensions and can lead to errors of from five to ten percent. Therefore, double linear interpolation of the calculated data was utilized to obtain the calculated data at exactly the location specified as the experimental detector center as follows:

r_i = Radial Location

Z_i = Axial Location

C_i = Calculated Data at center of mesh cell

C = The calculated data at the position of interest

$$C = \frac{C_2 (Z_1 - Z_3) + C_1 (Z_3 - Z_2)}{(Z_1 - Z_2)} + \quad (1)$$

$$\frac{C_4 (Z_1 - Z_3) + C_3 (Z_3 - Z_2) + C_2 (Z_3 - Z_1) + C_1 (Z_2 - Z_3)}{(Z_1 - Z_2)} \times \frac{(r_3 - r_1)}{(r_2 - r_1)}$$

The statistical analysis of the difference between the calculated and experimental data was performed with a graphical method. This method was employed because: 1) it is relatively easy to apply, and 2) it yields information concerning the data which is very difficult to obtain from a straight numerical analysis. (The veracity of this statement will become apparent later in the discussion).

The graphical analysis consists of plotting the difference between the calculated and experimental data as a function of the cumulative percent on a probability scale. If the differences between calculations and experiment have a normal distribution, the plot will represent the integral of a normal distribution curve. The probability scale is employed because the integral of a normal distribution will appear as a straight line in the plot. Therefore, the best straight line through the data will provide the following information:

- 1) The value of the difference between the calculated and experimental data obtained from the straight line at a cumulative probability of 50% is the average value of the difference. This occurs because the average value of a normal distribution occurs at the 50% location in the integral of the distribution.
- 2) The difference between the values on the straight line at the 50% and 84% locations (or the 16% and 50% locations) provides the value of one standard deviation in the differences between calculations and experiments. This occurs because in a normal distribution 68% (plus and minus 34%) of the data points fall within one standard deviation of the average of the distribution. Therefore, in the integral of a normal distribution plotted as a function of cumulative percent, 68% of data lie between the 16% and 84% locations.
- 3) The ability of a straight line to represent the data provides an indication of whether the variations in the differences between calculated and experimental data are randomly distributed. If

the data follow a straight line closely, then the variation in the differences is random and there is no statistically significant difference between the shape of the calculated data and the shape of the experimental data (note that this does not rule out the possibility of an average difference, or constant bias, between the two sets of data).

Because of the accuracy required in these comparisons the average difference and standard deviation was determined numerically and employed to plot the best straight line. Therefore, the primary use of the plots was to determine the correlation between the shapes of the two sets of data. This analysis is impossible to perform numerically for a function of two variables (even for one variable) which is represented by a high order transcendental function when there is not considerably more data available.

By examining the plots of the difference between the calculated and experimental data, bad data points will immediately become apparent and can be excluded from the analysis. In addition, the data over the "entire reactor"* were first plotted with the location in the reactor from which the data were obtained being tagged. By examining this plot, it was possible to determine whether the experimental and calculated data agreed in either magnitude, or shape, or both over portions of the reactor but not over the entire reactor. If this appeared to be the case, then the plot was redone over sections of the reactor.

The plot of the data is constructed in the following manner:

- 1) The differences between calculational and experimental data points are ranked in order of increasing magnitude,
- 2) The cumulative percent interval is determined by dividing 100% by number of data points in the plot,

* Entire reactor is defined here as the active core, filler strip, lateral support and reflector.

- 3) The differences are then plotted in order of increasing magnitude at the midpoint of the cumulative percent intervals on a probability scale. That is, the smallest algebraic difference is plotted at the midpoint of the first interval (determined in step 2). The second smallest algebraic difference is plotted at the sum of the interval plus one-half the sum of the interval. Succeeding values are plotted in a similar fashion.

Because of the amount of effort involved and because the two variables (r and Z) would confuse and complicate the analysis, the midplane data as a function of radius were analyzed to obtain average differences, standard deviations, and comparisons of shape. Selected axial shapes were examined, and no differences in the shapes of the data were found which could not be attributed to either random variations or differences such as test cell environment reflection which was not represented in the calculations.

In determining the difference between calculated and experimental data, it was first necessary to determine the best analytical form for the difference because the choice of an incorrect form would yield differences which did not have a normal distribution, and hence, the averages and standard deviations would have no meaning.

In order to determine the form of the difference, the data for sulfur were analyzed using different forms. The sulfur data were chosen because the shape is representative of most of the shapes obtained and visual comparison of the calculated and experimental data indicated that the two sets of data had the same shape, with a constant difference between the two sets of data. The following forms for the difference between the two sets of data were tested:

E = experimental value

C = calculated value

$$1) \quad E - C \quad ,$$

$$2) \quad \frac{E - C}{E} \quad ,$$

- 3) $\log (E - C)$, and
- 4) $\log E - \log C$.

Of these four forms for the difference between calculated and experimental data, the form of the difference represented by $\log E - \log C$ was chosen to perform the analysis for all data because the statistical analysis of the sulfur data employing this form indicated that the difference between calculated and experimental data had a normal distribution. This was determined by plotting the differences obtained for the four forms on a probability scale and determining which form yielded the best straight line.* The difference between the calculated and experimental sulfur data using analytical form four is plotted in Figure 5-75, and, as can be readily seen, a straight line, which indicated a normal distribution, quite readily, fits the data.

It should be emphasized that form four was chosen strictly on a statistical basis; however, the use of this form led to the use of uncertainty factors to describe possible variations in radiation levels. This is demonstrated as follows:

$$\begin{aligned}\Delta_i &= \text{the difference between } E_i \text{ and } C_i \\ &= \log E_i - \log C_i\end{aligned}\quad (2)$$

$$\Delta_i = \text{a random variable}$$

$$\bar{\Delta} = \text{average difference of } \Delta_i$$

$$s = \text{standard deviation of } \Delta_i$$

$$\text{Therefore, for 68\% of the observations } \bar{\Delta} - s \leq \log E_i - \log C_i \leq \bar{\Delta} + s \quad \text{or} \quad (3)$$

$$\log C_i + \bar{\Delta} - s \leq \log E_i \leq \log C_i + \bar{\Delta} + s \quad (4)$$

Raising equation (4) to the power of ten and setting

$$\bar{\Delta} - s = \log k_1 \quad \text{and} \quad (5)$$

$$\bar{\Delta} + s = \log k_2 \quad \text{equation (6) is obtained.}$$

$$k_1 C_i \leq E_i \leq k_2 C_i \quad (6)$$

$$\text{If } \bar{\Delta} = 0, \quad \text{then}$$

$$k_1 = 1/k_2 \quad \text{and} \quad (7)$$

$$C_i/k_2 \leq E_i \leq k_2 C_i \quad (8)$$

* It should be noted that the differences given by some of the forms yielded a probability plot which had an apparently random distribution of data.

However, equation (8) also represents the manner in which a one sigma uncertainty factor is applied to calculated data to obtain the possible variation in the data. Therefore, the use of uncertainty factors to represent variations in radiation levels has been derived from strictly statistical considerations.

5.5.3 Results

Figures 5-75 through 5-87 are the probability plots for the differences between the calculated and experimental reactor midplane data in this report. Table 5-5 provides the average difference between the calculated and experimental data and the one sigma limits on this difference. As an example of the use of these data, Table 5-5 indicates that, on the average over the reactor, the calculated neutron absorbed dose rate was 2% less than the experimental data. Further, the experimental data lay between the product of 0.89 and 1.16 times the calculated data for 68% of the observations.

Examination of the data, indicates that it was necessary to subdivide the U-238 and bare dysprosium data into core and reflector data in an attempt to obtain an analytical shape which was comparable to the experimental shape. The remainder of the data could be analyzed on an "entire reactor" basis. It should be noted that although the thermal flux ($E < 0.4$ ev) dysprosium data were divided into a core and a reflector component, this was done because the core midplane data were compared directly with experimental data while, in the reflector, the calculated data were adjusted for azimuthal variations (see Section 5.4.5) and were compared with the experimental data in the spring holes. Examination of the ($E < 0.4$ ev) dysprosium data in Table 5-5 indicates that the core and reflector calculated data have essentially the same average difference and standard deviation from experimental data. Therefore, only a single set of uncertainty factors which apply to the "entire reactor" will be employed with these data.

Examination of Figures 5-75 through 5-87 reveals that the difference between several types of calculated and experimental data is fit reasonably well by a straight line. This indicates that variations in the difference is random and, therefore, the shape of the

calculated data is the same as the shape of the experimental data. This conclusion is true for the following sets of data:

- 1) Sulfur data over the "entire reactor", fast neutron flux ($E > 2.9$ Mev),
- 2) U-238 data in the core, fast neutron flux ($E > 1.5$ Mev),
- 3) Dysprosium data over the "entire reactor", thermal neutron flux ($E < 0.4$ ev),
- 4) Fast Neutron absorbed dose rate data over the "entire reactor",
- 5) Gamma ray absorbed dose rate data over the "entire reactor", and
- 6) The Bragg-Gray heating rate information over the region in which it was measured.

While the U-238 data in the reflector and the bare dysprosium data do not appear to be fit by a straight line, it can only be concluded that the shapes of the calculated and experimental data may be different or that insufficient data were obtained to evaluate the hypothesis.

5.6 REFERENCES

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TABLE 5-1
PAX-G1 ATOM DENSITIES (ATOMS/BARN-CM)

REGION \ ELEMENT	Be	C	Fe	Ni	Cr	Mn	Nb	Ta	Ti	Al	Cu	Gd
1. Core - 1												
2. Core - 2												
3. Core - 3												
4. Core - 4												
5. Core - 5												
6. Core - 6												
7. Core - 7												
8. Core - 8												
9. Core - 9												
10. Core - 10												
11. Core - 11												
12. Core - 12												
13. Core - 13												
14. Core - 14												
15. Core - 15												
16. Filler Strip		9.3000 (-2)										
17. Lateral Support		4.6720 (-2)										
18. Inner Reflector	8.3822 (-2)	4.2720 (-3)										
19. Outer Reflector I	1.00313 (-1)		4.27 (-4)	5.02 (-4)	2.35 (-4)					2.450 (-3)		
20. Vane	Ring Prescription (See Text)											
21. Outer Reflector II	1.00313 (-1)		4.27 (-4)	5.02 (-4)	2.35 (-4)					2.450 (-3)		
22. Void												
23. Pressure Vessel										5.9982 (-2)		
24. Pressure Vessel										5.9982 (-2)		
25. Void												
26. Simulated Shield										5.507 (-2)	1.491 (-3)	
27. Void												
28. Support Plate			2.894 (-3)	3.708 (-4)	8.371 (-4)	8.347 (-5)				2.998 (-2)	8.559 (-4)	
29. Cluster Plate Hardware			6.846 (-3)	8.773 (-4)	1.980 (-3)	1.975 (-4)				1.945 (-3)		
30. Cluster Plate			3.714 (-2)	4.757 (-3)	1.074 (-2)	1.071 (-3)				1.945 (-3)		
31. Cluster Plate Hardware			6.846 (-3)	8.773 (-4)	1.980 (-3)	1.975 (-4)				1.945 (-3)		1.54096 (-4)
32. Support Blocks		5.722 (-2)	1.093 (-3)	1.400 (-4)	3.161 (-4)	3.152 (-5)						
33. Tie Rod Ends			1.016 (-3)	1.302 (-4)	2.940 (-4)	2.932 (-5)						
34. Drum Drive			7.906 (-4)	2.320 (-4)	2.319 (-4)	2.300 (-5)			1.767 (-3)	3.911 (-2)	1.059 (-3)	
35. Drum Drive			8.637 (-4)	1.786 (-4)	2.533 (-4)	2.538 (-5)			4.143 (-3)	4.493 (-4)		
36. Lateral Support/Aft		9.609 (-2)	4.394 (-4)	5.494 (-5)	1.249 (-4)					9.255 (-4)		
37. Reflector/Aft			1.584 (-3)	1.285 (-4)	4.782 (-4)	4.595 (-5)	5.719 (-5)	8.862 (-3)		2.242 (-2)		
38. Void												

NOTE: The core atom densities represent Confidential Restricted Data. Therefore, these atom densities have been deleted from this table so that this report could be unclassified.

TABLE 5-2

RADIAL (R) AND AXIAL (Z) DOT MESH

<u>Mesh Line Numbers</u>	<u>R (CM)</u>	<u>Z (CM)</u>
1	0.0	0.0
2	4.000	0.848
3	8.033	5.239
4	11.000	11.200
5	14.058	18.700
6	17.850	27.300
7	20.084	34.800
8	22.633	40.385
9	25.344	48.000
10	27.519	55.349
11	29.431	60.000
12	32.368	65.000
13	34.400	70.589
14	36.400	71.500
15	37.900	72.400
16	39.903	73.129
17	40.600	73.230
18	41.318	73.330
19	42.757	73.446
20	43.900	73.921
21	45.032	78.000
22	45.800	84.000
23	46.600	92.000
24	47.388	101.000
25	47.427	110.000
26	49.122	120.000
27	51.054	130.000
28	51.816	140.000
29	52.600	150.000
30	53.338	160.000

(Continued next page)

TABLE 5-2
(Continued)

RADIAL (R) AND AXIAL (Z) DOT MESH

<u>Mesh Line Numbers</u>	<u>R (CM)</u>	<u>Z (CM)</u>
31	54.200	170.000
32	55.000	179.000
33	55.900	188.000
34	56.700	196.000
35	57.600	202.000
36	58.400	206.001
37	59.0738	207.000
38	59.1979	208.000
39	59.322	209.000
40	60.500	210.000
41	61.700	211.716
42	62.840	212.800
43	63.000	213.984
44	63.185	
45	63.600	
46	64.033	

TABLE 5-3

EFFECTIVE CROSS SECTIONS EMPLOYED TO OBTAIN AN EQUIVALENT FLUX
FROM ACTIVATION DATA

DETECTOR	EFFECTIVE CROSS SECTIONS, BARNS
Sulfur	0.300
U-238	0.555
Dysprosium	2700

TABLE 5-4

COMPARISON OF MEASURED AND CALCULATED RADIATION
HEATING ⁽¹⁾ IN THE BERYLLIUM MASS IN THE PAX-G1 REFLECTOR

<u>RADIATION COMPONENT</u>	<u>BASED ON MEASURED DATA (MW)</u>	<u>CALCULATED ⁽⁴⁾ (MW)</u>
GAMMA RAY ⁽²⁾	$9.39 \pm 8.5\% \text{ }^{(5)}$	8.5
NEUTRON KINETIC ENERGY ⁽³⁾	$3.60 \pm 23\% \text{ }^{(5)}$	3.5
	<hr/>	<hr/>
	$12.99 \pm 8.9\%$	12.0

- NOTE:
- 1) Both the calculated and experimental data have been scaled to a power level of 1575 Mw (thermal) in the reactor.
 - 2) Measurements for gamma rays from Bragg-Gray chambers -- traverse through bolt holes.
 - 3) Neutron heating component based on measured fast-neutron dose rate.
 - 4) Calculated by NAGS for PAX-G1 at ambient conditions.
 - 5) Uncertainty assigned to experimental measurement by WANEF.

TABLE 5-5

STATISTICAL ANALYSIS OF THE DIFFERENCE BETWEEN CALCULATED AND EXPERIMENTAL DATA

Data Type and Location	Average Difference Between Calculations and Experiments	One Standard Deviation in the Difference Between Calculations and Experiments	
		Upper Limit	Lower Limit
Neutron Dose Rate - Entire Reactor ⁽⁵⁾	1.02	1.16	0.89
Gamma Ray Dose Rate - Entire Reactor ⁽⁵⁾	0.96	1.02	0.90
U-238 Data ⁽¹⁾ - Core	0.96	0.97	0.95
U-238 Data ⁽¹⁾ - Reflector	0.84	0.89	0.79
Sulfur Data ⁽²⁾ - Entire Reactor ⁽⁵⁾	1.32	1.54	1.13
Bare Dysprosium Data - Core	1.22	1.53	0.96
Bare Dysprosium Data - Reflector	1.04	1.09	0.99
Dysprosium (E<0.4 ev) Data ⁽³⁾ - Core	1.00	1.15	0.87
Dysprosium (E<0.4 ev) Data ⁽³⁾ - Reflector	0.99	1.14	0.87
Beryllium Heating Rate ⁽⁴⁾ - Reflector	1.28	1.33	1.24
Carbon Heating Rate ⁽⁴⁾ - Core	1.17	1.19	1.16
Aluminum Heating Rate ⁽⁴⁾ - Reflector	1.24	1.28	1.21
Stainless Steel Heating Rate ⁽⁴⁾ - Core	1.03	1.06	1.00

- (1) Neutron Flux (E>1.5 Mev)
 (2) Neutron Flux (E>2.9 Mev)
 (3) Neutron Flux (E<0.4 ev)
 (4) Photon Heating Rate Only

- (5) Entire reactor refers only to the location in which these measurements were performed, namely, the core, lateral support, and reflector.

NOTE: All Dimensions are in centimeters.
Refer to Table 5-1 for Description of
Regions and Compositions. Refer to Table 5-2
for the radial and axial position of all mesh lines.

FIGURE 5-1
PAX-G1 DOT GEOMETRY MOCKUP

FIGURE 5-2
PAX-G1 CALCULATIONAL FLOW CHART

FIGURE 5-3
COMPARISON OF CALCULATED AND EXPERIMENTAL RELATIVE RADIAL FISSION FLUX

FIGURE 5-4
COMPARISON OF CALCULATED AND EXPERIMENTAL RELATIVE AXIAL FISSION FLUX

FIGURE 5-5

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL SULFUR
($E > 2.9$ Mev) FAST NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-6

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL SULFUR ($E > 2.9$ Mev) FAST NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-7

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL SULFUR
($E > 2.9$ Mev) FAST NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-8
COMPARISON OF DOT CALCULATED AND EXPERIMENTAL SULFUR
($E > 2.9$ Mev) FAST NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5 -9

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL SULFUR
(E > 2.9 Mev) FAST NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-10

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL SULFUR ($E > 2.9$ Mev) FAST NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-11

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL SULFUR ($E > 2.9$ Mev) FAST NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-12

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL SULFUR ($E > 2.9$ Mev) FAST NEUTRON FLUX AS A FUNCTION OF RADIAL POSITION

FIGURE 5-13

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL SULFUR
(E > 2.9 Mev) FAST NEUTRON FLUX AS A FUNCTION OF RADIAL POSITION

FIGURE 5-14

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL U-238
($E > 1.5$ Mev) NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-15

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL U-238
(E > 1.5 Mev) NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-16

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL U-238
($E > 1.5$ Mev) NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-17

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL U-238
($E > 1.5$ Mev) NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-18

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL U-238
($E > 1.5$ Mev) NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-19

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL U-238
($E > 1.5$ Mev) NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5 -20

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL
U-238 ($E > 1.5$ Mev) NEUTRON FLUX AS A FUNCTION OF RADIAL POSITION

FIGURE 5-21

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL U-238
(E > 1.5 Mev) NEUTRON FLUX AS A FUNCTION OF RADIAL POSITION

FIGURE 5-22
COMPARISON OF DOT CALCULATED AND EXPERIMENTAL BARE DYSPROSIUM
ACTIVATION DATA AS A FUNCTION OF AXIAL POSITION

FIGURE 5-23

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL BARE DYSPROSIUM
ACTIVATION DATA AS A FUNCTION OF AXIAL POSITION

FIGURE 5-24

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL BARE DYSPROSIUM ACTIVATION DATA AS A FUNCTION OF AXIAL POSITION

FIGURE 5-25

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL BARE DYSPROSIUM
ACTIVATION DATA AS A FUNCTION OF AXIAL POSITION

FIGURE 5-26
COMPARISON OF DOT CALCULATED AND EXPERIMENTAL BARE DYSPROSIUM
ACTIVATION DATA AS A FUNCTION OF AXIAL POSITION

FIGURE 5-27
COMPARISON OF DOT CALCULATED AND EXPERIMENTAL BARE DYSPROSIUM
ACTIVATION DATA AS A FUNCTION OF AXIAL POSITION

FIGURE 5-28
COMPARISON OF DOT CALCULATED AND EXPERIMENTAL BARE DYSPROSIUM
ACTIVATION DATA AS A FUNCTION OF RADIAL POSITION

FIGURE 5-29

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL BARE DYSPROSIUM
ACTIVATION DATA AS A FUNCTION OF RADIAL POSITION

FIGURE 5-30
COMPARISON OF DOT CALCULATED AND EXPERIMENTAL DYSPROSIUM ($E < 0.4$ eV)
THERMAL NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-31

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL DYSPROSIUM ($E < 0.4$ eV) THERMAL NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-32
COMPARISON OF DOT CALCULATED AND EXPERIMENTAL DYSPROSIUM ($E < 0.4 \text{ eV}$)
THERMAL NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-33

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL DYSPROSIUM ($E < 0.4$ eV)
THERMAL NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-34

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL DYSPROSIUM ($E < 0.4$ eV) THERMAL NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-35

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL DYSPROSIUM ($E < 0.4$ eV) THERMAL NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-36
COMPARISON OF DOT CALCULATED AND EXPERIMENTAL DYSPROSIUM ($E < 0.4$ eV)
THERMAL NEUTRON FLUX AS A FUNCTION OF AXIAL POSITION

FIGURE 5-37

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL DYSPROSIUM ($E < 0.4$ eV)
THERMAL NEUTRON FLUX AS A FUNCTION OF RADIAL POSITION

FIGURE 5-38
COMPARISON OF DOT CALCULATED AND EXPERIMENTAL DYSPROSIUM ($E < 0.4$ eV)
THERMAL NEUTRON FLUX AS A FUNCTION OF RADIAL POSITION

FIGURE 5-39

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL DYSPROSIUM ($E < 0.4$ eV) THERMAL NEUTRON FLUX AS A FUNCTION OF RADIAL POSITION

FIGURE 5-40

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (SILICON DIODE) FAST NEUTRON ABSORBED DOSE RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-41

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (SILICON DIODE)
FAST NEUTRON ABSORBED DOSE RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-42

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (SILICON DIODE) FAST NEUTRON ABSORBED DOSE RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-43

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (SILICON DIODE)
FAST NEUTRON ABSORBED DOSE RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-44

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (SILICON DIODE) FAST NEUTRON ABSORBED DOSE RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-45

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (SILICON DIODE) FAST NEUTRON ABSORBED DOSE RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-46

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (SILICON DIODE) FAST NEUTRON ABSORBED DOSE RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-47

COMPARISON OF CALCULATED AND EXPERIMENTAL (SILICON DIODE) FAST NEUTRON
ABSORBED DOSE RATE AS A FUNCTION OF RADIAL POSITION

FIGURE 5-48

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (SILICON DIODE)
FAST NEUTRON ABSORBED DOSE RATE AS A FUNCTION OF RADIAL POSITION

FIGURE 5-49

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (CaF_2 TLD) GAMMA ABSORBED DOSE RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-50

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (CaF₂ TLD) GAMMA ABSORBED DOSE RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-51

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (CaF₂ TLD) GAMMA ABSORBED DOSE RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-52

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (CaF₂ TLD) GAMMA ABSORBED DOSE RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-53
COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (CaF₂ TLD)
GAMMA ABSORBED DOSE RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-54

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (CaF_2 TLD) GAMMA ABSORBED DOSE RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-55

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (CaF₂ TLD)
GAMMA ABSORBED DOSE RATE AS A FUNCTION OF RADIAL POSITION

FIGURE 5-56

COMPARISON OF DOT CALCULATED AND EXPERIMENTAL (CaF_2 TLD) GAMMA ABSORBED DOSE RATE AS A FUNCTION OF RADIAL POSITION

FIGURE 5-57

COMPARISON OF CALCULATED AND EXPERIMENTAL CARBON PHOTON
HEATING RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-58
COMPARISON OF CALCULATED AND EXPERIMENTAL CARBON PHOTON HEATING RATE
AS A FUNCTION OF AXIAL POSITION

FIGURE 5-59

COMPARISON OF CALCULATED AND EXPERIMENTAL CARBON PHOTON HEATING RATE
AS A FUNCTION OF AXIAL POSITION

FIGURE 5-60
COMPARISON OF CALCULATED AND EXPERIMENTAL CARBON PHOTON HEATING RATE
AS A FUNCTION OF AXIAL POSITION

FIGURE 5-61
COMPARISON OF CALCULATED AND EXPERIMENTAL CARBON PHOTON HEATING RATE
AS A FUNCTION OF AXIAL POSITION

FIGURE 5-62
COMPARISON OF CALCULATED AND EXPERIMENTAL CARBON PHOTON HEATING RATE
AS A FUNCTION OF RADIAL POSITION

FIGURE 5-63
COMPARISON OF CALCULATED AND EXPERIMENTAL STAINLESS STEEL
PHOTON HEATING RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-64
COMPARISON OF CALCULATED AND EXPERIMENTAL STAINLESS STEEL
PHOTON HEATING RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-65
COMPARISON OF CALCULATED AND EXPERIMENTAL STAINLESS STEEL
PHOTON HEATING RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-66
COMPARISON OF CALCULATED AND EXPERIMENTAL STAINLESS STEEL
PHOTON HEATING RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-67
COMPARISON OF CALCULATED AND EXPERIMENTAL STAINLESS STEEL
PHOTON HEATING RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-68
COMPARISON OF CALCULATED AND EXPERIMENTAL STAINLESS STEEL
PHOTON HEATING RATE AS A FUNCTION OF RADIAL POSITION

FIGURE 5-69

COMPARISON OF CALCULATED AND EXPERIMENTAL BERYLLIUM PHOTON
HEATING RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-70
COMPARISON OF CALCULATED AND EXPERIMENTAL BERYLLIUM PHOTON
HEATING RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-71
COMPARISON OF CALCULATED AND EXPERIMENTAL BERYLLIUM PHOTON
HEATING RATE AS A FUNCTION OF RADIAL POSITION

FIGURE 5-72
COMPARISON OF CALCULATED AND EXPERIMENTAL ALUMINUM PHOTON
HEATING RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-73
COMPARISON OF CALCULATED AND EXPERIMENTAL ALUMINUM PHOTON
HEATING RATE AS A FUNCTION OF AXIAL POSITION

FIGURE 5-74

COMPARISON OF CALCULATED AND EXPERIMENTAL ALUMINUM PHOTON HEATING RATE AS A FUNCTION OF RADIAL POSITION

FIGURE 5-75
PROBABILITY PLOT OF SULFUR DATA

FIGURE 5-76
PROBABILITY PLOT OF U-238 DATA IN THE CORE

FIGURE 5-77
PROBABILITY PLOT OF U-238 DATA IN THE REFLECTOR

FIGURE 5-78
PROBABILITY PLOT OF BARE DYSPROSIUM DATA IN THE CORE

FIGURE 5-79
PROBABILITY PLOT OF BARE DYSPROSIUM DATA IN THE REFLECTOR

FIGURE 5-80
PROBABILITY PLOT OF DYSPROSIUM ($E < 0.4$ eV) DATA IN THE CORE

FIGURE 5-81
PROBABILITY PLOT OF DYSPROSIUM ($E < 0.4$ eV) DATA IN THE REFLECTOR

FIGURE 5-82
PROBABILITY PLOT OF NEUTRON DOSE RATE DATA

FIGURE 5-83
PROBABILITY PLOT OF GAMMA RAY DOSE RATE DATA

FIGURE 5-84
PROBABILITY PLOT OF CARBON HEATING RATE DATA

FIGURE 5-85
PROBABILITY PLOT OF STAINLESS STEEL HEATING RATE DATA

FIGURE 5-86
PROBABILITY PLOT OF BERYLLIUM HEATING RATE DATA

FIGURE 5-87
PROBABILITY PLOT OF ALUMINUM HEATING RATE DATA

6.0 CONCLUSIONS

Based on the preceding analysis and discussion, the following conclusions were made:

- 1) The need for possible variations in the actual radiation level, R^a , to be represented in terms of the calculated radiation level, R^c , by an equation of the following form was derived from a statistical analysis of the data:
$$R^c / K \leq R^a \leq K R^c, \quad \text{where}$$
$$K = \text{some multiple of the standard deviation}$$
Therefore, a multiplicative factor (uncertainty factor) should be employed in reporting the possible variations in all radiation data.
- 2) The pointwise uncertainty in the fast neutron flux required for radiation effects analysis ($E > 1.0$ MeV) is based on the difference between the calculated and experimental U-238 data in the core. The average multiplicative factor to be applied to the calculated fast neutron flux data in both the core and the reflector is 0.96. The one sigma variation in this factor is 0.95 to 0.97. Therefore, at ambient conditions, it is concluded that the actual fast neutron flux lies between the product of 0.95 and the calculated fast neutron flux and the product of 0.97 and the calculated flux approximately 68 percent of the time.
- 3) The pointwise uncertainty in the thermal neutron flux ($E < 0.4$ eV) is based on the difference between the calculated and experimental sub-cadmium dysprosium data. The average multiplicative factor to be applied to the calculated thermal neutron flux data in both the core and the reflector is 1.00. The one sigma variation in this factor is 0.87 to 1.15. Therefore, at ambient conditions, it is concluded that the actual thermal neutron flux lies between the product of 0.87 and

the calculated thermal neutron flux and the product of 1.15 and the flux approximately 68 percent of the time. Further, it should be noted that for the reflector this uncertainty applies only to the azimuthally average thermal neutron flux.

- 4) The pointwise uncertainty in neutron heating and neutron dose rate is based on the difference between the calculated and experimental fast neutron absorbed dose rate. The average multiplicative factor to be applied to neutron heating and neutron dose rate in both the core and reflector is 1.02. The one-sigma variation in this factor is 0.89 to 1.16. Therefore, at ambient conditions, it is concluded that the actual neutron heating or fast neutron absorbed dose rate lies between the product of 0.89 and the calculated neutron data and the product of 1.16 and the calculated data 68 percent of the time.
- 5) The pointwise uncertainty in photon heating and photon dose rate is based on the difference between the calculated and experimental CaF_2 thermoluminescent detector data. The average multiplicative factor to be applied to the calculated photon heating and photon dose rate in both the core and reflector is 0.96. The one sigma variation in this factor is 0.90 to 1.02. Therefore, at ambient conditions, it is concluded that the actual photon heating or photon absorbed dose rate lies between the product of 0.90 and the calculated photon data and the product of 1.02 and the calculated data 68 percent of the time. It should be noted that this uncertainty has not been based on the Bragg-Gray data because of the component of neutron heating which may be present in this data.
- 6) The total heat deposition uncertainty in the beryllium portion of the reflector was obtained by integrating the calculated and experimental over the region of interest. This computation indicated that the total

calculated heat deposition for reflector beryllium material was 7.6 percent low.

- 7) The statistical analysis indicated that there was no statistically significant difference between the shape of the calculated and the shape of experimental data for the following measurements:
- a) Sulfur data,
 - b) U-238 data (in the core only),
 - c) Subcadmium dysprosium data,
 - d) Fast neutron absorbed dose rate data,
 - e) Photon absorbed dose rate data, and
 - f) Bragg-Gray heating rate data.

As a result of this analysis, it was determined that the following areas merit further investigation which may lead to the elimination of some of the differences between calculated and experimental data:

- 1) The effect of thermal activation in sulfur should be evaluated analytically in an effort to understand the difference between the calculated and experimental sulfur data.
- 2) The effect of fission spectrum, anisotropic cross sections, and group structure on the calculated sulfur data should be evaluated.
- 3) Since a neutron response in the Bragg-Gray detectors may be an appreciable effect, the neutron response should be evaluated analytically, and, if possible, this effect should be evaluated experimentally in order to better understand the heating rate data supplied by these detectors.
- 4) The effect of delayed photon buildup, of low energy spectral response, and neutron effects on CaF_2 thermoluminescent detectors should be evaluated.

- 5) The effect of the difference between the control vane and the poison ring, which is used to represent the control vanes in the calculations on the thermal neutron flux, should be investigated further.

APPENDIX A

SPECIAL REACTOR COMPONENTS

A.1 MODIFIED FUEL ELEMENTS ...

Standard fuel elements were modified. These elements contained machined slots which provided sufficient space for the passive dosimeters. Each element had nine milled slots axially spaced in one of the element faces as shown in Figures A-1 through A-3. There were two sets of elements, one with large slots and one with small. The larger slots were required for the sulfur, TLD and Phylatron dosimeters. The smaller slots are used for the foil dosimeters such as Dy, U-238 and Th.

A.2 MODIFIED CLUSTER BLOCKS

The cluster blocks, for those clusters containing the modified fuel elements described in the preceding section, were modified in order to facilitate removal of those elements following the irradiation. A hexagonal cylinder, with cross section identical to a fuel element cross section, was removed from each of the support blocks so that the fuel element could be removed without disassembling the cluster. In order to replace the material in the void thus created, 2-1/4 inch long sections of unfueled graphite elements were fabricated. These were attached to the modified fuel elements with two 0.040 inch diameter Zr wires running the length of the fuel element and graphite segment through diagonally opposite corner flow channels of the element.

A.3 SPECIAL FILLER STRIP

A special filler strip was manufactured. The special strip has holes 0.450 inch in diameter and 0.265 inch deep milled into the outer surface at nine axial locations in order to provide for insertion of passive dosimeters. Due to an error in manufacturing the axial position, these holes were displaced from the axial positions of the corresponding dosimetry slots in the fuel elements.

9 Slots per Element at Axial Locations (Z) = 0.5, 6.5,
13.0, 19.5, 26.0, 32.5, 39.0, 45.5, 51.5

	<u>Length (L)</u>	<u>Width (W)</u>	<u>Depth (D)</u>
Type 1	0.375 ± 0.005	0.155 ± 0.010	0.380 ± 0.005
Type 2	0.440 ± 0.005	0.280 ± 0.010	0.440 ± 0.005

All Dimensions in Inches

Figure A-1. Detail of Modified Fuel Elements

Figure A-2. Fuel Element With Large Dosimeter Slots

Figure A-3. Fuel Element With Small Dosimeter Slots

A.4 BERYLLIUM RODS FOR TIE BOLT HOLES

Special beryllium rods were manufactured for insertion into the reflector. These rods were 1/2 inch diameter and 52 inches long and were equipped with nine slots spaced so as to be at the same axial height as the slots in the fuel elements. The lowest slot in the beryllium rods, however, was an inch higher than the corresponding slot in the fuel elements because the beryllium rod did not reach to the bottom of the core. Figure A-4 shows a sketch of the beryllium rods.

A.5 LATERAL SUPPORT HOLE INSERTS

Special hardware was manufactured to permit measurement in the lateral support spring holes near the reflector midplane. Beryllium slugs were machined of various thicknesses and with recesses in the ends for the different types of dosimeters as shown in Figure A-5. These allowed the spring hole to be essentially filled with beryllium with only enough void to conveniently accommodate five dosimeters equally spaced in a radial direction from the inner to the outer surface of the beryllium reflector.

Holes drilled in the pressure vessel provided access to the lateral support holes. The pressure vessel holes were provided with aluminum plugs.

Figure A-4. Beryllium Rod With Dosimetry Slots

Type	Length (L)	HOLE	
		Diameter (D)	Depth (d)
for Sulfur, TLD	1.008	0.450	0.265
for Cadmium covers	1.045	0.400	0.140
for Bare foils	1.075	0.260	0.020

Type	Length (L)	HOLE	
		Diameter (D)	Depth (d)
for Sulfur, TLD	1.288	0.450	0.265
for Cadmium covers	1.177	0.400	0.140
for Bare foils	1.087	0.260	0.020

Assembly

Slug Diameter $0.625 + 0.002$
Tolerances $+ 0.002$ on slug length
 ± 0.005 on hole diameter
 and depth

Figure A-5. Beryllium Slugs

APPENDIX B

FOIL CALIBRATION REPORTS

Table B-1 presents the report on absolute activity of dysprosium-aluminum foils which was used in the absolute calibration of the Dy-Al foil measurements.

Table B-2 presents the report on absolute activity of sulfur pellets which was used in the absolute calibration of the sulfur pellet measurements.

Table B-3 presents the report on total fissions in U-238 foils which was used in the calibration of the U-238 foil measurements.

CHEMICAL ANALYSIS REPORT
FORM 54741

ANAL. SERV. REQUEST NO.

1433

ORIGINATOR

DEPT. & GRP.

EXT.

ROOM NO.

FACILITY

R. BRAWDY

WANE

WM.

APPROVAL SIGNATURE

DATE RECEIVED

DATE REPORTED

C. Blackburn

7-23-69

7-24-69

METHOD	ANALYST	REFERENCE	METHOD	ANALYST	REFERENCE
<u>8 Spectrometry</u>	CAR	FILE-WANKE JUN-JULY-1969			

RESULTS OF ANALYSIS

[illegible]

REMARKS:

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TABLE B-1 (CONTINUED)

CHEMICAL ANALYSIS REQUEST <small>FORM 54700</small>		WESTINGHOUSE ATOMIC POWER DIVISIONS ANALYTICAL SERVICE LABORATORIES WALTZ MILL SITE		ANAL. SERV. REQUEST NO. 1433	
ORIGINATOR R. Bandy	DEPT. & GRP. WANEF	EST. 5226	INITIATING DOCUMENT REFERENCES		DATE SUBMITTED 7-23-69
CHARGE NO. 04704			IWR		REQUESTED COMPLETION DATE
ORIGINATOR'S SIGNATURE & DATE		AUTHORIZING SIGNATURE & DATE		ACCEPTANCE SIGNATURE & DATE C. Chapman 7-23-69	
ORIGINATOR'S SAMPLE NO.	ANAL. SERV. LAB NO.	ANALYSIS			
#66	8704	ABSOLUTE ACTIVITY QUANTITATIVE FOR: QUALITATIVE FOR: APPROX. CONCENTRATION (S): RESULTS IN UNITS OF: ☐ Wt. % ☐ PPM ☐ Other (specify) DISPOSITION OF SAMPLE: ☐ Discard ☒ Return to Requestor ☐ Other (specify) MATERIAL CLASSIFICATION: ☒ Unclassified ☐ Confidential ☐ Secret			
#67	8705				
#68	8706				
		SAMPLE DESCRIPTION			
		APPROX. WT. AND/OR VOL. ENRICHMENT:			
		MATERIAL: FORM AND COMPOSITION, SPECIAL HANDLING, ETC. Dy-AL Foils .010 x .250			
		RADIOACTIVITY: Content and Level _____ M.R./hr. @ _____ in.			
		PREVIOUS HISTORY (IRRADIATIONS, PHYSICAL TREATMENT, ETC., WHICH COULD INFLUENCE SAMPLE TREATMENT): 25 wt/mas			
		REMARKS:			
ANALYTICAL SERVICE LABORATORY USE ONLY					
REMARKS:					

Spec ☐	Gas ☐	Radio Chem. ☒	Chem ☐	Na ☐	Mass ☐	Pu ☐
ANAL. SERV. REQUEST NO. 1433			CHECK COPIES FOR LEGIBILITY		ORIGINATOR COPY	

TABLE B-2
SULFUR ACTIVITY CALIBRATION REPORT

PSL:462

From : Physical Sciences Laboratory
WIN :
Date : July 11, 1969
Subject : Analysis of WANEF Sulfur Pellets

ASTRONUCLEAR LABORATORY

To: R. W. Brawdy, WANEF

cc: F. S. Frantz, WANEF
T. O. Harves
V. S. Oblock, WANEF
W. D. Rankin, WANEF
J. Roesmer

Three sulfur samples dated July 3, 1969 have been analyzed for P-32 activity in the Physical Sciences Laboratory. The P-32 activity was corrected for decay to 16:33, July 3, 1969 using a half life value of 14.3 days. The following results were obtained:

Sample Designation	Sample Type	Weight (Grams)	Activity Determined	Total DPM
59	Sulfur Pellet	1.1113	P-32	$6.86 \times 10^3 \pm 5\%$
60	Sulfur Pellet	1.0547	P-32	$6.56 \times 10^3 \pm 5\%$
177	Sulfur Pellet	1.0905	P-32	$7.47 \times 10^3 \pm 5\%$

Wayne A. Henninger
Wayne A. Henninger
Physical Sciences Laboratory

Approved by:

G. T. Rymer, Supervisor
Physical Sciences Laboratory

TABLE B-3
U-238 CALIBRATION REPORT

PSL:465

From : Physical Sciences Laboratory
WIN :
Date : July 18, 1969
Subject : Analysis of WANEF Fission Foils

ASTRONUCLEAR LABORATORY

To: R. W. Brawdy, WANEF

cc: F. S. Frantz, WANEF
V. S. Oblock, WANEF
W. A. Henninger
W. D. Rankin, WANEF
G. T. Rymer

Three fission foils, dated July 11, 1969, have been analyzed for Mo-99 to determine the number of fissions. The results, corrected to 14:25, June 11, 1969 are summarized in the attached table.

J. Roesmer
Physical Sciences Laboratory

Approved by:

G. T. Rymer, Supervisor
Physical Sciences Laboratory

Attachment

/lfd

TABLE B-3 (CONTINUED)

ANALYSIS OF THREE WANEF FISSION FOILS

Sample Designation	Weight (grams)	Activity Measured	Total Number of Fissions
#84	0.0726	Mo-99	$5.62 \times 10^8 \pm 4\%$
#85	0.0782	Mo-99	$5.90 \times 10^8 \pm 6\%$
#86	0.0774	Mo-99	$6.17 \times 10^8 \pm 6\%$

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or Cut

TITLE Radiation Environment In Various Reactor Components	
	DOCUMENT NO. WANL-TME-1914	DATE 061070
ABSTRACT An experiment was performed to measure the radiation environment in the core and reflector of the PAX-G mockup of the R-1 reactor. Fast and thermal neutron fluxes, fast neutron dose rates, and gamma dose rates were obtained experimentally and compared with analytical calculations. Good agreement was obtained for fast fluxes ($E > 1.5$ Mev) in the core region and for TLD-measured gamma dose rates.		
DATA ITEM NO. T-119	SECURITY CLASSIFICATION OF DOCUMENT Unclassified	
CDRL LINE NO. TBD	TOTAL NO. OF PAGES 208	
KEY WORDS Radiation Measurements Gamma dose rate Experimental Data PAX/R-1 Reactor PAX-G Reactor Fast neutron flux Fast neutron dose rate Thermal neutron flux		
AUTHOR/EDITOR W. D. Rankin & D. J. Hill		PREPARING DEPT. WANEF
W Form 59943 DOCUMENT INFORMATION SUMMARY		

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CLEVELAND, OHIO