Westinghouse
Astronuclear
Laboratory

SNPO-C

Records *aws*

WANL-TME-2757
FEBRUARY 1971

DATA ITEM S-100

**RADIATION AND SHIELDING
ANALYSIS REPORT
NUCLEAR SUBSYSTEM**

VOLUME I CHAPTERS 1 THRU 6

MASTER

DISTRIBUTION OF THIS DOCUMENT UNLIMITED

DISCLAIMER

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

DISCLAIMER

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

Westinghouse Astronuclear Laboratory

Subcontract NP-1
CODE IDENT NO. 14683

WANL-TME-2757
FEBRUARY 1971

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

DATA ITEM S-100

RADIATION AND SHIELDING ANALYSIS REPORT, NUCLEAR SUBSYSTEM.

APPROVED BY:

F. D. Retallick

F. D. Retallick, Manager
NERVA Engineering

INFORMATION CATEGORY

Unclassified

J. M. Hughes 3/5/71
AUTHORIZED CLASSIFIER DATE

leg

PAGE BLANK

TABLE OF CONTENTS

<u>Section</u>	<u>Page</u>
See Volume I for Chapters 1 through 6.	
1.0 INTRODUCTION	1-1
1.1 PURPOSE OF REPORT	1-1
1.2 CONTENTS OF REPORT	1-1
2.0 NUCLEAR SUBSYSTEM	2-1
2.1 INTRODUCTION	2-1
2.1.1 Description of the Nuclear Subsystem	2-1
2.1.2 Content of Chapter	2-2
2.2 SUMMARY OF RESULTS AND CONCLUSIONS	2-4
2.2.1 External Radiation Environment	2-4
2.2.2 Internal Radiation Environment	2-8
2.2.3 Radiation Heating Rates	2-10
2.2.4 Radiation Sources	2-12
2.3 REQUIREMENTS	2-14
2.3.1 Specification Identification	2-14
2.3.2 Applicable Requirements	2-14
2.4 METHOD OF ANALYSIS (Design Practices)	2-16
2.4.1 Analytical Procedures	2-16
2.4.2 Analytical Models	2-35
2.4.3 Status/Maturity of Input Data	2-38
2.4.4 Validity of Analysis	2-40
2.5 RESULTS, ANALYSIS, AND INTERPRETATION	2-57
2.5.1 Composite Core	2-57
2.5.2 Graphite Core	2-70
2.5.3 Activation Analysis	2-81

TABLE OF CONTENTS

(Continued)

<u>Section</u>	<u>Page</u>
2.6 REFERENCES	2-83
3.0 FUEL ELEMENTS	3-1
3.1 INTRODUCTION	3-1
3.2 SUMMARY OF RESULTS AND CONCLUSIONS	3-1
4.0 CLUSTER HARDWARE	4-1
4.1 INTRODUCTION	4-1
4.2 SUMMARY OF RESULTS AND CONCLUSIONS	4-1
5.0 CORE PERIPHERY	5-1
5.1 INTRODUCTION	5-1
5.2 SUMMARY OF RESULTS AND CONCLUSIONS	5-1
6.0 SUPPORT PLATE AND PLENA	6-1
6.1 INTRODUCTION	6-1
6.1.1 Description of Support Plate and Plena	6-1
6.1.2 Content of Chapter	6-2
6.2 SUMMARY OF RESULTS AND CONCLUSIONS	6-3
6.2.1 Composite Core	6-3
6.2.2 Graphite Core	6-5
6.3 REQUIREMENTS	6-6
6.3.1 Specification Identification	6-6
6.3.2 Applicable Requirements	6-6

TABLE OF CONTENTS (Continued)

<u>Section</u>	<u>Page</u>
6.4 METHOD OF ANALYSIS (Design Practices)	6-7
6.4.1 Analytical Procedures	6-7
6.4.2 Analytical Models	6-7
6.4.3 Status/Maturity of Input Data	6-9
6.4.4 Validity of Analysis	6-9
6.5 RESULTS, ANALYSIS, AND INTERPRETATION	6-10
6.5.1 Composite Core	6-10
6.5.2 Graphite Core	6-15
6.6 REFERENCES	6-19
* * * * * See Volume II for Chapters 7 through 11. * * * * *	
7.0 INTERNAL SHIELD	7-1
7.1 INTRODUCTION	7-1
7.1.1 Description of the Internal Shield	7-1
7.1.2 Content of Chapter	7-2
7.2 SUMMARY OF RESULTS AND CONCLUSIONS	7-4
7.2.1 External Radiation Environment	7-4
7.2.2 Internal Radiation Environment	7-6
7.2.3 Radiation Heating Rates	7-7
7.2.4 Radiation Sources	7-7
7.3 REQUIREMENTS	7-9
7.3.1 Specification Identification	7-9
7.3.2 Applicable Requirements	7-9

TABLE OF CONTENTS (Continued)

<u>Section</u>	<u>Page</u>
7.4 METHOD OF ANALYSIS (Design Practices)	7-11
7.4.1 Analytical Procedures	7-11
7.4.2 Analytical Models	7-11
7.4.3 Status/Maturity of Input Data	7-13
7.4.4 Validity of Analysis	7-14
7.5 RESULTS, ANALYSIS, AND INTERPRETATION	7-19
7.5.1 Composite Core	7-19
7.5.2 Graphite Core	7-25
7.5.3 Shield Design to Meet Current Specifications	7-30
7.6 REFERENCES	7-34
8.0 REFLECTOR ASSEMBLY	8-1
8.1 INTRODUCTION	8-1
8.1.1 Description of Aluminum Structure Reflector	8-1
8.1.2 Content of Chapter	8-2
8.2 SUMMARY OF RESULTS AND CONCLUSIONS	8-2
8.2.1 Composite Core	8-2
8.2.2 Graphite Core	8-4
8.3 REQUIREMENTS	8-5
8.3.1 Specification Identification	8-5
8.3.2 Applicable Requirements	8-5
8.4 METHOD OF ANALYSIS (Design Practices)	8-6
8.4.1 Analytical Procedures	8-6

TABLE OF CONTENTS (Continued)

<u>Section</u>	<u>Page</u>
8.4.2 Analytical Models	8-6
8.4.3 Status/Maturity of Input Data	8-7
8.4.4 Validity of Analysis	8-8
8.5 RESULTS, ANALYSIS, AND INTERPRETATION	8-9
8.5.1 Composite Core	8-9
8.5.2 Graphite Core	8-14
8.6 REFERENCES	8-18
9.0 CONTROL DRUM DRIVE ASSEMBLY	9-1
9.1 INTRODUCTION	9-1
9.1.1 Description of Control Drum Drive Assembly	9-1
9.1.2 Content of Chapter	9-2
9.2 SUMMARY OF RESULTS AND CONCLUSIONS	9-3
9.2.1 Fast Neutron Environment	9-4
9.2.2 Intermediate Neutron Environment	9-5
9.2.3 Thermal Neutron Environment	9-6
9.2.4 Gamma Dose Environment	9-7
9.2.5 Radiation Heating Environment	9-8
9.3 REQUIREMENTS	9-9
9.3.1 Specification Identification	9-9
9.3.2 Applicable Requirements	9-9

TABLE OF CONTENTS (Continued)

<u>Section</u>	<u>Page</u>
9.4 METHOD OF ANALYSIS (Design Practices)	9-10
9.4.1 Analytical Procedures	9-10
9.4.2 Analytical Models	9-10
9.4.3 Status/Maturity of Input Data	9-11
9.4.4 Validity of Analysis	9-12
9.5 RESULTS, ANALYSIS, AND INTERPRETATION	9-13
9.5.1 Composite Core	9-13
9.5.2 Graphite Core	9-17
9.5.3 Unshielded Core	9-21
9.5.4 Activation Analysis	9-22
9.6 REFERENCES	9-24
10.0 STRUCTURAL SUPPORT COOLANT ASSEMBLY	10-1
10.1 INTRODUCTION	10-1
10.1.1 Description of Structural Support Coolant Assembly	10-1
10.1.2 Content of Chapter	10-1
10.2 SUMMARY OF RESULTS AND CONCLUSIONS	10-2
10.2.1 Fast Neutron Environment	10-3
10.2.2 Intermediate Neutron Environment	10-3
10.2.3 Thermal Neutron Environment	10-4
10.2.4 Gamma Dose Environment	10-4
10.2.5 Radiation Heating Environment	10-5
10.3 REQUIREMENTS	10-5
10.3.1 Specification Identification	10-6
10.3.2 Applicable Requirements	10-6

TABLE OF CONTENTS (Continued)

<u>Section</u>	<u>Page</u>
10.4 METHOD OF ANALYSIS (Design Practices)	10-7
10.4.1 Analytical Procedures	10-7
10.4.2 Analytical Models	10-7
10.4.3 Status/Maturity of Input Data	10-7
10.4.4 Validity of Analysis	10-7
10.5 RESULTS, ANALYSIS, AND INTERPRETATION	10-8
10.5.1 Composite Core	10-9
10.5.2 Graphite Core	10-12
10.5.3 Unshielded Core	10-15
10.6 REFERENCES	10-17
11.0 NUCLEAR SUBSYSTEM INSTRUMENTATION	11-1
11.1 INTRODUCTION	11-1
11.1.1 Description of Nuclear Subsystem Instrumentation	11-1
11.1.2 Content of Chapter	11-2
11.2 SUMMARY OF RESULTS AND CONCLUSIONS	11-2
11.2.1 Fast Neutron Environment	11-3
11.2.2 Intermediate Neutron Environment	11-3
11.2.3 Thermal Neutron Environment	11-3
11.2.4 Gamma Dose Environment	11-4
11.2.5 Radiation Heating Environment	11-4

TABLE OF CONTENTS
(Continued)

<u>Section</u>	<u>Page</u>
11.3 REQUIREMENTS	11-4
11.3.1 Specification Identification	11-4
11.3.2 Applicable Requirements	11-4
11.4 METHOD OF ANALYSIS (Design Practices)	11-6
11.4.1 Analytical Procedures	11-6
11.4.2 Analytical Models	11-6
11.4.3 Status/Maturity of Input Data	11-6
11.4.4 Validity of Analysis	11-7
11.5 RESULTS, ANALYSIS, AND INTERPRETATION	11-9
11.5.1 Composite Core	11-9
11.5.2 Graphite Core	11-12
11.6 REFERENCES	11-16

LIST OF TABLES

<u>Table No.</u>	<u>Title</u>	<u>Page No.</u>
2-1	Radiation Leakage Limits at a Plane 63 Inches Forward of the Core Center	2-86
2-2	Radiation Leakage Limits at Critical Component Locations	2-87
2-3	Selected Locations Internal to the NSS for Radiation Environment Data (Composite Core)	2-88
2-4	R-Z Model Region Description	2-94
2-5	Mesh Line Coordinates for the Neutron Basic Reactor Geometry DOT Problem	2-97
2-6	Mesh Line Coordinates for the Neutron Shield Geometry DOT Problem	2-98
2-7	Mesh Line Coordinates for the Photon Basic Reactor Geometry DOT Problem	2-99
2-8	Mesh Line Coordinates for the Photon Shield Geometry DOT Problem	2-100
2-9	Material Description for R-1 Reference Design (Dwg. 939J723) Employing Composite Fuel	2-101
2-10	Material Description for R-1 Reference Design (Dwg. 939J723) Employing Graphite Fuel	2-102
2-11	Uncertainty Factors to be Applied to Calculated Radiation Levels Forward of the R-1 Reactor	2-103
2-12	Statistical Analysis of the Difference Between Calculated and Experimental Data in the Core and Reflector	2-104
2-13	Statistical Analysis of the Difference Between Calculated and Experimental Data Internal to the Shield Mockups	2-105
2-14	Statistical Analysis of the Difference Between Calculated and Experimental Data in the CHESH	2-106
2-15	Comparison of Measured and Calculated Radiation Heating in the Beryllium Mass in the PAX-G1 Reflector	2-107

LIST OF TABLES
(Continued)

<u>Table No.</u>	<u>Title</u>	<u>Page No.</u>
2-16	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Fuel Element ECC (Composite Fueled Core)	2-108
2-17	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Core Periphery ECC (Composite Fueled Core)	2-109
2-18	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Reflector Assembly ECC (Composite Fueled Core)	2-110
2-19	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Cluster Hardware ECC (Composite Fueled Core)	2-111
2-20	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Support Plate Portion of the Support Plate and Plena ECC (Composite Fueled Core)	2-112
2-21	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Plena Portion of the Support Plate and Plena ECC (Composite Fueled Core)	2-113
2-22	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Internal Shield ECC (Composite Fueled Core)	2-114
2-23	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Control Drum Actuator Portion of the Control Drum Drive Assembly ECC (Composite Fueled Core)	2-115
2-24	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Fuel Element ECC (Graphite Fueled Core)	2-116

LIST OF TABLES
(Continued)

<u>Table No.</u>	<u>Title</u>	<u>Page No.</u>
2-25	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Core Periphery ECC (Graphite Fueled Core)	2-117
2-26	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Reflector Assembly ECC (Graphite Fueled Core)	2-118
2-27	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Cluster Hardware ECC (Graphite Fueled Core)	2-119
2-28	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Support Plate Portion of the Support Plate and Plena ECC (Graphite Fueled Core)	2-120
2-29	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Internal Shield ECC (Graphite Fueled Core)	2-121
2-30	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Plena Portion of the Support Plate and Plena ECC (Graphite Fueled Core)	2-122
2-31	Regionwise and Pointwise Uncertainty Factors, F, for the Radiation Environment in the Reactor Regions Associated with the Control Drum Actuator Portion of the Control Drum Drive Assembly ECC (Graphite Fueled Core)	2-123
2-32	NERVA Flight Uncertainty Factors for the Total Heat Deposited in each ECC	2-124
2-33	Uncertainty of Photon and Beta Sources for Key Regions of the NSS	2-125
2-34	Comparison of Internal Radiation Environmental Data at Key NSS Locations for Composite and Graphite Fueled Cores	2-126

LIST OF TABLES
(Continued)

<u>Table No.</u>	<u>Title</u>	<u>Page No.</u>
2-35	Radiation Environment at Key NSS Locations - Composite Fuel in the Core	2-127
2-36	Pressure Vessel and Reactor Assembly Source Strengths Composite Core	2-128
2-37	Fission Product Gamma Ray Source Strength (MeV/sec) for Various Times Following Termination of a 10-Minute Operation at Full Power (1511 Mw Thermal)	2-132
2-38	Fission Product Gamma Ray Source Strength (MeV/sec) for Various Times Following Termination of a 30-Minute Operation at Full Power (1511 Mw Thermal)	2-134
2-39	Fission Product Gamma Ray Source Strength (MeV/sec) for Various Times Following Termination of a 60-Minute Operation at Full Power (1511 Mw Thermal)	2-136
2-40	Neutron and Photon Radiation Environment (Composite Core) Data Point Location -- Tangent Plane (Max)	2-138
2-41	Neutron and Photon Radiation Environment (Composite Core) Data Point Location -- TPA	2-139
2-42	Neutron and Photon Radiation Environment (Composite Core) Data Point Location -- LTS - 1	2-140
2-43	Neutron and Photon Radiation Environment (Composite Core) Data Point Location -- LTS - 2	2-141
2-44	Neutron and Photon Radiation Environment (Composite Core) Data Point Location -- LTS - 3	2-142
2-45	Relative Contribution from Various Source Zones to the Overall Photon Dose Rate at Several Locations on a 10-Foot Meridian Ring	2-143
2-46	Delayed Gamma Ray Dose Rate Versus Time after Shutdown Following a 10-Minute Run at Full Power (1511 Mw Thermal Power) -- Composite Core	2-144

LIST OF TABLES
(Continued)

<u>Table No.</u>	<u>Title</u>	<u>Page No.</u>
2-47	Delayed Gamma Ray Dose Rate Versus Time After Shutdown Following a 30-Minute Run at Full Power (1511 Mw Thermal Power) -- Composite Core	2-148
2-48	Delayed Gamma Ray Dose Rate Versus Time After Shutdown Following a 60-Minute Run at Full Power (1511 Mw Thermal Power) -- Composite Core	2-152
2-49	R-1 Reference Design (WANL Dwg. 939J723) Heating Rate by Region at Full Power (1511 Mw Thermal) with Composite Fuel in the Core	2-156
2-50	Comparison of Heating Rates at Key NSS Locations for Composite and Graphite Fueled Cores	2-163
2-51	Time Integrated Radiation Environment at Selected Locations (Composite Core)	2-164
2-52	Gamma Ray Source Terms for Composite Core at Transient State Points	2-194
2-53	Factors to Convert Extra-Core Secondary Gamma Source Terms to Transient State Points	2-195
2-54	Radiation Environment at Key NSS Locations - Graphite Fuel in the Core	2-196
2-55	Pressure Vessel and Reactor Assembly Source Strengths Graphite Core	2-197
2-56	Neutron and Photon Radiation Environment (Graphite Core) Data Point Location -- Tangent Plane (Max)	2-201
2-57	Neutron and Photon Radiation Environment (Graphite Core) Data Point Location -- TPA	2-202
2-58	Neutron and Photon Radiation Environment (Graphite Core) Data Point Location -- LTS - 1	2-203
2-59	Neutron and Photon Radiation Environment (Graphite Core) Data Point Location -- LTS - 2	2-204

LIST OF TABLES
(Continued)

<u>Table No.</u>	<u>Title</u>	<u>Page No.</u>
2-60	Neutron and Photon Radiation Environment (Graphite Core) Data Point Location -- LTS - 3	2-205
2-61	Relative Contribution from Various Source Zones to the Overall Photon Dose Rate at Several Locations on a 10-Foot Meridian Ring	2-206
2-62	Delayed Gamma Ray Dose Rate Versus Time After Shutdown Following a 10-Minute Run at Full Power (1511 Mw Thermal Power) -- Graphite Core	2-207
2-63	Delayed Gamma Ray Dose Rate Versus Time After Shutdown Following a 30-Minute Run at Full Power (1511 Mw Thermal Power) -- Graphite Core	2-211
2-64	Delayed Gamma Ray Dose Rate Versus Time After Shutdown Following a 60-Minute Run at Full Power (1511 Mw Thermal Power) -- Graphite Core	2-215
2-65	R-1 Reference Design (WANL Dwg. 939J723) Heating Rate by Region at Full Power (1511 Mw Thermal) With Graphite Fuel in the Core	2-219
2-66	Time Integrated Radiation Environment at Selected Locations (Graphite Core)	2-223
2-67	Gamma Ray Source Terms for Graphite Core at Transient State Points	2-254
2-68	After Shutdown Gamma Dose Rate 10 Feet from Core Center for R-1 Reference Design (939J723) with a Composite Core	2-255
2-69	After Shutdown Gamma Dose Rate 10 Feet from Core Center for R-1 Reference Design (939J723) with a Composite Core	2-256
2-70	After Shutdown Gamma Dose Rate 10 Feet from Core Center for R-1 Reference Design (939J723) with a Composite Core	2-257

LIST OF TABLES
(Continued)

<u>Table No.</u>	<u>Title</u>	<u>Page No.</u>
6-1	Maximum Radiation Environment at Support Plate Composite and Graphite Fuel in the Reactor Core	6-20
6-2	Maximum Heating Rate by Component for the Support Plate and Plena	6-21
6-3	Maximum Time Integrated Radiation Environment in the Support Plate and Plena (Composite Core)	6-24
6-4	Maximum Time Integrated Radiation Environment in the Support Plate and Plena (Graphite Core)	6-25
7-1	Maximum Radiation Environment at Internal Shield Composite and Graphite Fuel in the Reactor Core	7-35
7-2	Maximum Heating Rate by Component for the Internal Shield	7-36
7-3	Maximum Time Integrated Radiation Environment in the Internal Shield (Composite Core)	7-38
7-4	Maximum Time Integrated Radiation Environment in the Internal Shield (Graphite Core)	7-39
7-5	Neutron and Photon Radiation Environment (Composite Core) Data Point Location - Tangent Plane (Max)	7-40
7-6	Neutron and Photon Radiation Environment (Composite Core) Data Point Location - TPA	7-41
7-7	Neutron and Photon Radiation Environment (Composite Core) Data Point Location - LTS-1	7-42
7-8	Neutron and Photon Radiation Environment (Composite Core) Data Point Location - LTS-2	7-43
7-9	Neutron and Photon Radiation Environment (Composite Core) Data Point Location - LTS-3	7-44
7-10	Neutron and Photon Radiation Environment (Graphite Core) Data Point Location - Tangent Plane (Max)	7-45
7-11	Neutron and Photon Radiation Environment (Graphite Core) Data Point Location - TPA	7-46

LIST OF TABLES
(Continued)

<u>Table No.</u>	<u>Title</u>	<u>Page No.</u>
7-12	Neutron and Photon Radiation Environment (Graphite Core) Data Point Location - LTS-1	7-47
7-13	Neutron and Photon Radiation Environment (Graphite Core) Data Point Location - LTS-2	7-48
7-14	Neutron and Photon Radiation Environment (Graphite Core) Data Point Location - LTS-3	7-49
7-15	Neutron and Photon Radiation Environment (Composite Core) Data Point Location - Tangent Plane (Max.)	7-50
7-16	Neutron and Photon Radiation Environment (Composite Core) Data Point Location - LTS-1	7-51
7-17	Neutron and Photon Radiation Environment (Composite Core) Data Point Location - LTS-2	7-52
7-18	Neutron and Photon Radiation Environment (Composite Core) Data Point Location - LTS-3	7-53
7-19	Neutron and Photon Radiation Environment (Composite Core) Data Point Location - TPA	7-54
8-1	Maximum Radiation Environment at Reflector Assembly Composite and Graphite Fuel in the Reactor Core	8-19
8-2	Maximum Heating Rate by Component in the Reflector, Aluminum Structure	8-20
8-3	Maximum Time Integrated Radiation Environment in the Reflector (Composite Core)	8-22
8-4	Maximum Time Integrated Radiation Environment in the Reflector (Graphite Core)	8-23

LIST OF TABLES
(Continued)

<u>Table No.</u>	<u>Title</u>	<u>Page No.</u>
9-1	Maximum Nominal Radiation Environment at Several CDDA Locations for the Composite Core	9-25
9-2	Maximum Nominal Radiation Environment at Several CDDA Locations for the Graphite Core	9-26
9-3	Maximum Nominal Radiation Environment at Several CDDA Locations for Operation in Space with the Internal Shield Removed	9-27
9-4	Estimates of the Radiation Environment at the CDDA Actuator Location Due to the Presence of Test Cell C	9-28
9-5	Time Integrated Radiation Environment for Several Control Drum Drive Assembly Locations for the Composite Fueled Core	9-29
9-6	Time Integrated Radiation Environment for Several Control Drum Drive Assembly Locations for the Graphite Fueled Core	9-31
9-7	Time Integrated Radiation Environment for Several Control Drum Drive Assembly Locations for an Unshielded Core	9-33
10-1	Nominal Radiation Environment At Several SSCA Locations for the Composite Core	10-18
10-2	Nominal Radiation Environment At Several SSCA Locations for the Graphite Core	10-19
10-3	Nominal Radiation Environment At Several SSCA Locations for Operation in Space with the Internal Shield Removed	10-20
10-4	Time Integrated Radiation Environment for Several Structural Support Coolant Assembly Locations for the Composite Fueled Core	10-21
10-5	Time Integrated Radiation Environment for Several Structural Support Coolant Assembly Locations for the Graphite Fueled Core	10-22
10-6	Time Integrated Radiation Environment for Several Structural Support Coolant Assembly Locations for an Unshielded Core	10-23

LIST OF TABLES
(Continued)

<u>Table No.</u>	<u>Title</u>	<u>Page No.</u>
11-1	Radiation Environment at Instrumentation Locations for the Composite Fuel Core	11-17
11-2	Time Integrated Radiation Environment and Uncertainty Factors at Instrumentation Locations for the Composite Fuel Core	11-22
11-3	Maximum Nominal Radiation Environment at the Instrumentation Locations for the Composite and Graphite Cores	11-63
11-4	Radiation Environment at Instrumentation Locations for the Graphite Fuel Core	11-64
11-5	Time Integrated Radiation Environment and Uncertainty Factors at Instrumentation Locations for the Graphite Fuel Core	11-69

LIST OF ILLUSTRATIONS

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-1	Frequency Distribution of Possible Radiation Levels	2-258
2-2	Flow Chart for Nuclear, Thermal, and Structural Analysis	2-259
2-3	Nuclear and Radiation Design NSS Analysis (Block 11)	2-260
2-4	Flow Chart to Calculate Neutron and Photon Environment and Heating Rates Internal to the Nuclear Subsystem (Blocks 11-7, 11-8, and 11-9)	2-261
2-5	Flow Chart to Calculate Neutron and Photon Environment External to the Nuclear Subsystem (Block 11-10)	2-262
2-6	Radiation Environment Drawing	2-263
2-7	Basic Reactor Geometry Model for the Neutron DOT Problem	2-265
2-8	Shield Geometry Model for the Neutron DOT Problem	2-266
2-9	Basic Reactor Geometry Model for the Photon DOT Problem	2-267
2-10	Shield Geometry Model for the Photon DOT Problem	2-268
2-11	Heating Rate Model for the Composite Core	2-269
2-12	Heating Rate Model for the Graphite Core	2-270
2-13	Common Radiation Analysis Model for the NSS Employing Composite Fuel	2-271
2-14	Common Radiation Analysis Model for the NSS Employing Graphite Fuel	2-272
2-15	Gamma Ray Dose Rate as a Function of Axial Location on the Reactor Centerline (Composite Core)	2-273
2-16	Gamma Ray Dose Rate as a Function of Axial Location at the Core Mid-Radius (Composite Core)	2-274
2-17	Gamma Ray Dose Rate as a Function of Axial Location at the Core Edge (Composite Core)	2-275
2-18	Gamma Ray Dose Rate as a Function of Axial Location at the Outer Radius of the Core Periphery (Composite Core)	2-276

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-19	Gamma Ray Dose Rate as a Function of Axial Location at the Outer Radius of the Lateral Support (Composite Core)	2-277
2-20	Gamma Ray Dose Rate as a Function of Axial Location at the Inner Radius of the Reflector (Composite Core)	2-278
2-21	Gamma Ray Dose Rate as a Function of Axial Location on the Control Drum Centerline (Composite Core)	2-279
2-22	Gamma Ray Dose Rate as a Function of Axial Location at the Outer Radius of the Reflector (Composite Core)	2-280
2-23	Gamma Ray Dose Rate as a Function of Axial Location at the Pressure Vessel Side Surface (Composite Core)	2-281
2-24	Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the CHESH (Composite Core)	2-282
2-25	Gamma Ray Dose Rate as a Function of Radial Location at the Core Exit (Composite Core)	2-283
2-26	Gamma Ray Dose Rate as a Function of Radial Location at the Core Midplane (Composite Core)	2-284
2-27	Gamma Ray Dose Rate as a Function of Radial Location at the Core Inlet (Composite Core)	2-285
2-28	Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the Support Plate (Composite Core)	2-286
2-29	Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the Support Plate (Composite Core)	2-287
2-30	Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the Shield Plate (Composite Core)	2-288
2-31	Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the BATH Central Shield (Composite Core)	2-289
2-32	Gamma Ray Dose Rate as a Function of Radial Location at the BATH Central Shield Midplane (Composite Core)	2-290

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-33	Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the BATH Central Shield (Composite Core)	2-291
2-34	Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Composite Core)	2-292
2-35	Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the Flow Baffle (Composite Core)	2-293
2-36	Gamma Ray Dose Rate as a Function of Radial Location at the Pressure Vessel Dome (Composite Core)	2-294
2-37	Fast Neutron Flux as a Function of Axial Location on the Reactor Centerline (Composite Core)	2-295
2-38	Fast Neutron Flux as a Function of Axial Location at the Core Mid-Radius (Composite Core)	2-296
2-39	Fast Neutron Flux as a Function of Axial Location at the Core Edge (Composite Core)	2-297
2-40	Fast Neutron Flux as a Function of Axial Location at the Outer Radius of the Core Periphery (Composite Core)	2-298
2-41	Fast Neutron Flux as a Function of Axial Location at the Outer Radius of the Lateral Support (Composite Core)	2-299
2-42	Fast Neutron Flux as a Function of Axial Location at the Inner Radius of the Reflector (Composite Core)	2-300
2-43	Fast Neutron Flux as a Function of Axial Location on the Control Drum Centerline (Composite Core)	2-301
2-44	Fast Neutron Flux as a Function of Axial Location at the Outer Radius of the Reflector (Composite Core)	2-302
2-45	Fast Neutron Flux as a Function of Axial Location at the Pressure Vessel Side Surface (Composite Core)	2-303
2-46	Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the CHESH (Composite Core)	2-304

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-47	Fast Neutron Flux as a Function of Radial Location at the Core Exit (Composite Core)	2-305
2-48	Fast Neutron Flux as a Function of Radial Location at the Core Midplane (Composite Core)	2-306
2-49	Fast Neutron Flux as a Function of Radial Location at the Core Inlet (Composite Core)	2-307
2-50	Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the Support Plate (Composite Core)	2-308
2-51	Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the Support Plate (Composite Core)	2-309
2-52	Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the Shield Plate (Composite Core)	2-310
2-53	Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the BATH Central Shield (Composite Core)	2-311
2-54	Fast Neutron Flux as a Function of Radial Location at the BATH Central Shield Midplane (Composite Core)	2-312
2-55	Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the BATH Central Shield (Composite Core)	2-313
2-56	Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Composite Core)	2-314
2-57	Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the Flow Baffle (Composite Core)	2-315
2-58	Fast Neutron Flux as a Function of Radial Location at the Pressure Vessel Dome (Composite Core)	2-316
2-59	Intermediate Neutron Flux as a Function of Axial Location on the Reactor Centerline (Composite Core)	2-317
2-60	Intermediate Neutron Flux as a Function of Axial Location at the Core Mid-Radius (Composite Core)	2-318

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-61	Intermediate Neutron Flux as a Function of Axial Location at the Core Edge (Composite Core)	2-319
2-62	Intermediate Neutron Flux as a Function of Axial Location at the Outer Radius of the Core Periphery (Composite Core)	2-320
2-63	Intermediate Neutron Flux as a Function of Axial Location at the Outer Radius of the Lateral Support (Composite Core)	2-321
2-64	Intermediate Neutron Flux as a Function of Axial Location at the Inner Radius of the Reflector (Composite Core)	2-322
2-65	Intermediate Neutron Flux as a Function of Axial Location on the Control Drum Centerline (Composite Core)	2-323
2-66	Intermediate Neutron Flux as a Function of Axial Location on the Outer Radius of the Reflector (Composite Core)	2-324
2-67	Intermediate Neutron Flux as a Function of Axial Location at the Pressure Vessel Side Surface (Composite Core)	2-325
2-68	Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the CHESH (Composite Core)	2-326
2-69	Intermediate Neutron Flux as a Function of Radial Location at the Core Exit (Composite Core)	2-327
2-70	Intermediate Neutron Flux as a Function of Radial Location at the Core Midplane (Composite Core)	2-328
2-71	Intermediate Neutron Flux as a Function of Radial Location at the Core Inlet (Composite Core)	2-329
2-72	Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the Support Plate (Composite Core)	2-330
2-73	Intermediate Neutron Flux as a Function of Radial Location at the Forward Surface of the Support Plate (Composite Core)	2-331
2-74	Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the Shield Plate (Composite Core)	2-332

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-75	Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the BATH Central Shield (Composite Core)	2-333
2-76	Intermediate Neutron Flux as a Function of Radial Location at the BATH Central Shield Midplane (Composite Core)	2-334
2-77	Intermediate Neutron Flux as a Function of Radial Location at the Forward Surface of the BATH Central Shield (Composite Core)	2-335
2-78	Intermediate Neutron Flux as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Composite Core)	2-336
2-79	Intermediate Neutron Flux as a Function of Radial Location at the Forward Surface of the Flow Baffle (Composite Core)	2-337
2-80	Intermediate Neutron Flux as a Function of Radial Location at the Pressure Vessel Dome (Composite Core)	2-338
2-81	Thermal Neutron Flux as a Function of Axial Location on the Reactor Centerline (Composite Core)	2-339
2-82	Thermal Neutron Flux as a Function of Axial Location on the Core Mid-Radius (Composite Core)	2-340
2-83	Thermal Neutron Flux as a Function of Axial Location at the Core Edge (Composite Core)	2-341
2-84	Thermal Neutron Flux as a Function of Axial Location at the Outer Radius of the Core Periphery (Composite Core)	2-342
2-85	Thermal Neutron Flux as a Function of Axial Location at the Outer Radius of the Lateral Support (Composite Core)	2-343
2-86	Thermal Neutron Flux as a Function of Axial Location at the Inner Radius of the Reflector (Composite Core)	2-344
2-87	Thermal Neutron Flux as a Function of Axial Location on the Control Drum Centerline (Composite Core)	2-345
2-88	Thermal Neutron Flux as a Function of Axial Location at the Outer Radius of the Reflector (Composite Core)	2-346

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-89	Thermal Neutron Flux as a Function of Axial Location at the Pressure Vessel Side Surface (Composite Core)	2-347
2-90	Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the CHESH (Composite Core)	2-348
2-91	Thermal Neutron Flux as a Function of Radial Location at the Core Exit (Composite Core)	2-349
2-92	Thermal Neutron Flux as a Function of Radial Location at the Core Midplane (Composite Core)	2-350
2-93	Thermal Neutron Flux as a Function of Radial Location at the Core Inlet (Composite Core)	2-351
2-94	Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the Support Plate (Composite Core)	2-352
2-95	Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the Support Plate (Composite Core)	2-353
2-96	Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the Shield Plate (Composite Core)	2-354
2-97	Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the BATH Central Shield (Composite Core)	2-355
2-98	Thermal Neutron Flux as a Function of Radial Location at the BATH Central Shield Midplane (Composite Core)	2-356
2-99	Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the BATH Central Shield (Composite Core)	2-357
2-100	Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Composite Core)	2-358
2-101	Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the Flow Baffle (Composite Core)	2-359
2-102	Thermal Neutron Flux as a Function of Radial Location at the Pressure Vessel (Composite Core)	2-360
2-103	Average 155 Group Neutron Flux Versus Neutron Energy in the Composite Core	2-361

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-104	Average 155 Group Neutron Flux Versus Neutron Energy in the Outer Reflector Region	2-362
2-105	Average 155 Group Neutron Flux Versus Neutron Energy in the BATH Central Shield	2-363
2-106	Average 155 Group Neutron Flux Versus Neutron Energy in the Pressure Vessel Dome	2-364
2-107	Average 155 Group Neutron Flux Versus Neutron Energy in the Side Pressure Vessel	2-365
2-108	Photon Energy Flux as a Function of Photon Energy at the Center of the Composite Core	2-366
2-109	Photon Energy Flux as a Function of Photon Energy at the Inner Radius of the Reflector on the Core Midplane	2-367
2-110	Photon Energy Flux as a Function of Photon Energy at the Outer Radius of the Pressure Vessel on the Core Midplane	2-368
2-111	Photon Energy Flux as a Function of Photon Energy at the Aft Face of the BATH Shield on the Centerline	2-369
2-112	Photon Energy Flux as a Function of Photon Energy at the Surface of the Pressure Vessel Dome on the Centerline	2-370
2-113	Photon Heating Rate in Aluminum 6061 ($\rho = 2.71$ gm/cc) as a Function of Axial Location on the Reactor Centerline	2-371
2-114	Photon Heating Rate in Stainless Steel 304 ($\rho = 7.92$ gm/cc) as a Function of Axial Location on the Reactor Centerline	2-372
2-115	Photon Heating Rate in Aluminum 6061 ($\rho = 2.71$ gm/cc) as a Function of Axial Location on the Reactor Centerline	2-373
2-116	Photon Heating Rate in Stainless Steel 304 ($\rho = 7.92$ gm/cc) as a Function of Axial Location on the Reactor Centerline	2-374
2-117	Relative Radial Distribution of Photon Source in Region 7, Pressure Vessel Side - 1	2-375

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-118	Relative Axial Distribution of Photon Source in Region 7, Pressure Vessel Side - 1	2-376
2-119	Relative Radial Distribution of Photon Source in Region 34, Pressure Vessel Dome	2-377
2-120	Relative Axial Distribution of Photon Source in Region 34, Pressure Vessel Dome	2-378
2-121	Relative Radial Distribution of Photon Source in Region 35, Pressure Vessel Side - 2	2-379
2-122	Relative Axial Distribution of Photon Source in Region 35, Pressure Vessel Side - 2	2-380
2-123	Photon Kerma Rate on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design (939J723) With a Composite Core	2-381
2-124	Fast ($E > 1.0$ MeV) Neutron Flux on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design (939J723) With a Composite Core	2-382
2-125	Intermediate ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) Neutron Flux on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design (939J723) With a Composite Core	2-383
2-126	Thermal ($E < 0.4 \text{ eV}$) Neutron Flux on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design (939J723) With a Composite Core	2-384
2-127	Gamma Ray Kerma Rate Vs. Polar Angle for Meridian Rings of Various Radii for the R-1 Reference Design (939J723)	2-385
2-128	Fast ($E > 1.0 \text{ MeV}$) Neutron Flux Vs. Polar Angle for Meridian Rings of Various Radii for the R-1 Reference Design (939J723)	2-386
2-129	Intermediate ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) Neutron Flux Vs. Polar Angle for Meridian Rings of Various Radii for the R-1 Reference Design (939J723)	2-387

LIST OF ILLUSTRATIONS
(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-130	Thermal ($E < 0.4$ eV) Neutron Flux Vs. Polar Angle for Meridian Rings of Various Radii for the R-1 Reference Design (939J723)	2-388
2-131	Gamma Ray Kerma Rate on the Reactor Axis (Forward End) Vs. Distance from Core Center for R-1 Reference Design (939J723)	2-389
2-132	Neutron Fluxes on the Reactor Axis (Forward End) Vs. Distance from Core Center for R-1 Reference Design (939J723)	2-390
2-133	Gamma Ray Isodose Rate (Rads-C/hr) Contours at Full Power (1511 Mw Thermal) Around the R-1 Reference Design Reactor (939J723) with Test Car and Test Cell C Shield Removed (Composite Fuel)	2-391
2-134	Fast Neutron ($E > 1.0$ MeV) Isoflux (Neutrons/cm ² -sec) Contours at Full Power (1511 Mw Thermal) Around the R-1 Reference Design (939J723) with Test Car and Test Cell C Shield Removed (Composite or Graphite Fuel)	2-392
2-135	Intermediate Neutron ($0.4 \text{ eV} < E < 1.0$ MeV) Isoflux (Neutrons/cm ² -sec) Contours at Full Power (1511 Mw Thermal) Around the R-1 Reference Design Reactor (939J723) with Test Car and Test Cell C Shield Removed (Composite or Graphite Fuel)	2-393
2-136	Thermal Neutron ($E < 0.4$ eV) Isoflux (Neutrons/cm ² -sec) Contours at Full Power (1511 Mw Thermal) Around the R-1 Reference Design Reactor (939J723) with Test Car and Test Cell C Shield Removed (Composite or Graphite Fuel)	2-394
2-137	Relative Heating Rate as a Function of Reactor Power Level	2-395
2-138	Relative Radial Distribution of Heating in the Pressure Vessel Side, Region 22 (Composite Core)	2-396
2-139	Relative Axial Distribution of Heating in the Pressure Vessel Side, Region 22 (Composite Core)	2-397
2-140	Relative Radiation Distribution of Heating in the Pressure Vessel Side-B, Region 42 (Composite Core)	2-398
2-141	Relative Axial Distribution of Heating in the Pressure Vessel Side-B, Region 42 (Composite Core)	2-399

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-142	Relative Radial Distribution of Heating in the Pressure Vessel Dome, Region 53 (Composite Core)	2-400
2-143	Relative Axial Distribution of Heating in the Pressure Vessel Dome, Region 53 (Composite Core)	2-401
2-144	Relative Increase Due to the Test Cell C Facility Shield on the Gamma Dose Rate as a Function of Radius on the Core Midplane	2-402
2-145	Relative Increase Due to the Test Cell C Facility Shield on the Fast Neutron Flux ($E > 1.0$ MeV) as a Function of Radius on the Core Midplane	2-403
2-146	Relative Increase Due to the Test Cell C Facility Shield on the Intermediate Neutron Flux ($0.4 \text{ eV} < E < 1.0$ MeV) as a Function of Radius on the Core Midplane	2-404
2-147	Relative Increase Due to the Test Cell C Facility Shield on the Thermal Neutron Flux ($E < 0.4$ eV) as a Function of Radius on the Core Midplane	2-405
2-148	Relative Increase Due to the Test Cell C Facility Shield on the Al-7039 Heating Rate as a Function of Radius on the Core Midplane	2-406
2-149	Relative Increase Due to the Test Cell C Facility Shield on the Al-6061 Heating Rate as a Function of Radius on the Core Midplane	2-407
2-150	Relative Increase Due to the Test Cell C Facility Shield on the Beryllium Heating Rate as a Function of Radius on the Core Midplane	2-408
2-151	Relative Increase Due to the Test Cell C Facility Shield on the Carbon Heating Rate as a Function of Radius on the Core Midplane	2-409
2-152	Relative Increase Due to the Test Cell C Facility Shield on the A-286 Heating Rate as a Function of Radius on the Core Midplane	2-410

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-153	Relative Increase Due to the Test Cell C Facility Shield on the Thermal Neutron Flux ($E < 0.4$ eV) as a Function of Axial Height on the Core Centerline	2-411
2-154	Relative Increase Due to the Test Cell C Facility Shield on the Intermediate Neutron Flux ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) as a Function of Axial Height on the Core Centerline	2-412
2-155	Relative Increase Due to the Test Cell C Facility Shield on the Fast Neutron Flux ($E > 1.0 \text{ MeV}$) as a Function of Axial Height on the Core Centerline	2-413
2-156	Relative Increase Due to the Test Cell C Facility Shield on the Gamma Dose Rate as a Function of Axial Height on the Core Centerline	2-414
2-157	Relative Increase Due to the Test Cell C Facility Shield on the Al-6061 Heating Rate as a Function of Axial Height on the Core Centerline	2-415
2-158	Relative Increase Due to the Test Cell C Facility Shield on the Al-7039 Heating Rate as a Function of Axial Height on the Core Centerline	2-416
2-159	Relative Increase Due to the Test Cell C Facility Shield on the A-286 Heating Rate as a Function of Axial Height on the Core Centerline	2-417
2-160	Relative Increase Due to the Test Cell C Facility Shield on the Carbon Heating Rate as a Function of Axial Height on the Core Centerline	2-418
2-161	R-1 Power Profile for Both a Composite or Graphite Core	2-419
2-162	Neutron Integrating Factor as a Function of Run Time	2-420
2-163	Prompt Integrating Factor as a Function of Run Time	2-421
2-164	Delayed Integrating Factor as a Function of Run Time	2-422
2-165	Prompt and Decay Power as a Function of Reactor Run Time	2-423

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-166	Delayed Photon Plus Beta Power as a Function of Time After Shutdown Following a 60 Minute Full Power Run (1511 Mw)	2-424
2-167	Gamma Ray Dose Rate as a Function of Axial Location on the Reactor Centerline (Graphite Core)	2-425
2-168	Gamma Ray Dose Rate as a Function of Axial Location at the Core Mid-Radius (Graphite Core)	2-426
2-169	Gamma Ray Dose Rate as a Function of Axial Location at the Core Edge (Graphite Core)	2-427
2-170	Gamma Ray Dose Rate as a Function of Axial Location at the Outer Radius of the Core Periphery (Graphite Core)	2-428
2-171	Gamma Ray Dose Rate as a Function of Axial Location at the Outer Radius of the Lateral Support (Graphite Core)	2-429
2-172	Gamma Ray Dose Rate as a Function of Axial Location at the Inner Radius of the Reflector (Graphite Core)	2-430
2-173	Gamma Ray Dose Rate as a Function of Axial Location at the Control Drum Centerline (Graphite Core)	2-431
2-174	Gamma Ray Dose Rate as a Function of Axial Location at the Outer Radius of the Reflector (Graphite Core)	2-432
2-175	Gamma Ray Dose Rate as a Function of Axial Location at the Pressure Vessel Side Surface (Graphite Core)	2-433
2-176	Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the CHESH (Graphite Core)	2-434
2-177	Gamma Ray Dose Rate as a Function of Radial Location at the Core Exit (Graphite Core)	2-435
2-178	Gamma Ray Dose Rate as a Function of Radial Location at the Core Midplane (Graphite Core)	2-436
2-179	Gamma Ray Dose Rate as a Function of Radial Location at the Core Inlet (Graphite Core)	2-437

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-180	Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the Support Plate (Graphite Core)	2-438
2-181	Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the Support Plate (Graphite Core)	2-439
2-182	Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the Shield Plate (Graphite Core)	2-440
2-183	Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the BATH Central Shield (Graphite Core)	2-441
2-184	Gamma Ray Dose Rate as a Function of Radial Location at the BATH Central Shield Midplane (Graphite Core)	2-442
2-185	Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the BATH Central Shield (Graphite Core)	2-443
2-186	Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Graphite Core)	2-444
2-187	Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the Flow Baffle (Graphite Core)	2-445
2-188	Gamma Ray Dose Rate as a Function of Radial Location at the Pressure Vessel Dome (Graphite Core)	2-446
2-189	Fast Neutron Flux as a Function of Axial Location on the Reactor Centerline (Graphite Core)	2-447
2-190	Fast Neutron Flux as a Function of Axial Location at the Core Mid-Radius (Graphite Core)	2-448
2-191	Fast Neutron Flux as a Function of Axial Location at the Core Edge (Graphite Core)	2-449
2-192	Fast Neutron Flux as a Function of Axial Location at the Outer Radius of the Core Periphery (Graphite Core)	2-450
2-193	Fast Neutron Flux as a Function of Axial Location at the Outer Radius of the Lateral Support (Graphite Core)	2-451

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-194	Fast Neutron Flux as a Function of Axial Location at the Inner Radius of the Reflector (Graphite Core)	2-452
2-195	Fast Neutron Flux as a Function of Axial Location on the Control Drum Centerline (Graphite Core)	2-453
2-196	Fast Neutron Flux as a Function of Axial Location at the Outer Radius of the Reflector (Graphite Core)	2-454
2-197	Fast Neutron Flux as a Function of Axial Location at the Pressure Vessel Side Surface (Graphite Core)	2-455
2-198	Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the CHESH (Graphite Core)	2-456
2-199	Fast Neutron Flux as a Function of Radial Location at the Core Exit (Graphite Core)	2-457
2-200	Fast Neutron Flux as a Function of Radial Location at the Core Midplane (Graphite Core)	2-458
2-201	Fast Neutron Flux as a Function of Radial Location at the Core Inlet (Graphite Core)	2-459
2-202	Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the Support Plate (Graphite Core)	2-460
2-203	Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the Support Plate (Graphite Core)	2-461
2-204	Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the Shield Plate (Graphite Core)	2-462
2-205	Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the BATH Central Shield (Graphite Core)	2-463
2-206	Fast Neutron Flux as a Function of Radial Location at the BATH Central Shield Midplane (Graphite Core)	2-464
2-207	Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the BATH Central Shield (Graphite Core)	2-465

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-208	Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Graphite Core)	2-466
2-209	Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the Flow Baffle (Graphite Core)	2-467
2-210	Fast Neutron Flux as a Function of Radial Location at the Pressure Vessel Dome (Graphite Core)	2-468
2-211	Intermediate Neutron Flux as a Function of Axial Location on the Reactor Centerline (Graphite Core)	2-469
2-212	Intermediate Neutron Flux as a Function of Axial Location at the Core Mid Radius (Graphite Core)	2-470
2-213	Intermediate Neutron Flux as a Function of Axial Location at the Core Edge (Graphite Core)	2-471
2-214	Intermediate Neutron Flux as a Function of Axial Location at the Outer Radius of the Core Periphery (Graphite Core)	2-472
2-215	Intermediate Neutron Flux as a Function of Axial Location at the Outer Radius of the Lateral Support (Graphite Core)	2-473
2-216	Intermediate Neutron Flux as a Function of Axial Location at the Inner Radius of the Reflector (Graphite Core)	2-474
2-217	Intermediate Neutron Flux as a Function of Axial Location on the Control Drum Centerline (Graphite Core)	2-475
2-218	Intermediate Neutron Flux as a Function of Axial Location at the Outer Radius of the Reflector (Graphite Core)	2-476
2-219	Intermediate Neutron Flux as a Function of Axial Location at the Pressure Vessel Side Surface (Graphite Core)	2-477
2-220	Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the CHESH (Graphite Core)	2-478
2-221	Intermediate Neutron Flux as a Function of Radial Location at the Core Exit (Graphite Core)	2-479

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-222	Intermediate Neutron Flux as a Function of Radial Location at the Core Midplane (Graphite Core)	2-480
2-223	Intermediate Neutron Flux as a Function of Radial Location at the Core Inlet (Graphite Core)	2-481
2-224	Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the Support Plate (Graphite Core)	2-482
2-225	Intermediate Neutron Flux as a Function of Radial Location at the Forward Surface of the Support Plate (Graphite Core)	2-483
2-226	Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the Shield Plate (Graphite Core)	2-484
2-227	Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the BATH Central Shield (Graphite Core)	2-485
2-228	Intermediate Neutron Flux as a Function of Radial Location at the BATH Central Shield Midplane (Graphite Core)	2-486
2-229	Intermediate Neutron Flux as a Function of Radial Location at the Forward Surface of the BATH Central Shield (Graphite Core)	2-487
2-230	Intermediate Neutron Flux as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Graphite Core)	2-488
2-231	Intermediate Neutron Flux as a Function of Radial Location at the Forward Surface of the Flow Baffle (Graphite Core)	2-489
2-232	Intermediate Neutron Flux as a Function of Radial Location at the Pressure Vessel Dome (Graphite Core)	2-490
2-233	Thermal Neutron Flux as a Function of Axial Location on the Reactor Centerline (Graphite Core)	2-491
2-234	Thermal Neutron Flux as a Function of Axial Location at the Core Mid Radius (Graphite Core)	2-492

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-235	Thermal Neutron Flux as a Function of Axial Location at the Core Edge (Graphite Core)	2-493
2-236	Thermal Neutron Flux as a Function of Axial Location at the Outer Radius of the Core Periphery (Graphite Core)	2-494
2-237	Thermal Neutron Flux as a Function of Axial Location at the Outer Radius of the Lateral Support (Graphite Core)	2-495
2-238	Thermal Neutron Flux as a Function of Axial Location at the Inner Radius of the Reflector (Graphite Core)	2-496
2-239	Thermal Neutron Flux as a Function of Axial Location on the Control Drum Centerline (Graphite Core)	2-497
2-240	Thermal Neutron Flux as a Function of Axial Location at the Outer Radius of the Reflector (Graphite Core)	2-498
2-241	Thermal Neutron Flux as a Function of Axial Location at the Pressure Vessel Side Surface (Graphite Core)	2-499
2-242	Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the CHESH (Graphite Core)	2-500
2-243	Thermal Neutron Flux as a Function of Radial Location at the Core Exit (Graphite Core)	2-501
2-244	Thermal Neutron Flux as a Function of Radial Location at the Core Midplane (Graphite Core)	2-502
2-245	Thermal Neutron Flux as a Function of Radial Location at the Core Inlet (Graphite Core)	2-503
2-246	Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the Support Plate (Graphite Core)	2-504
2-247	Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the Support Plate (Graphite Core)	2-505
2-248	Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the Shield Plate (Graphite Core)	2-506

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-249	Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the BATH Central Shield (Graphite Core)	2-507
2-250	Thermal Neutron Flux as a Function of Radial Location at the BATH Central Shield Midplane (Graphite Core)	2-508
2-251	Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the BATH Central Shield (Graphite Core)	2-509
2-252	Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Graphite Core)	2-510
2-253	Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the Flow Baffle (Graphite Core)	2-511
2-254	Thermal Neutron Flux as a Function of Radial Location at the Pressure Vessel Dome (Graphite Core)	2-512
2-255	Average 155 Group Neutron Flux Versus Neutron Energy in the Outer Reflector Region	2-513
2-256	Average 155 Group Neutron Flux Versus Neutron Energy in the Graphite Core	2-514
2-257	Average 155 Group Neutron Flux Versus Neutron Energy in the BATH Central Shield	2-515
2-258	Average 155 Group Neutron Flux Versus Neutron Energy in the Pressure Vessel Dome	2-516
2-259	Average 155 Group Neutron Flux Versus Neutron Energy in the Pressure Vessel Side	2-517
2-260	Photon Energy Flux as a Function of Photon Energy at the Center of the Graphite Core	2-518
2-261	Photon Energy Flux as a Function of Photon energy at the Inner Radius of the Reflector on the Core Midplane	2-519
2-262	Photon Energy Flux as a Function of Photon Energy at the Surface of the Pressure Vessel on the Core Midplane	2-520

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-263	Photon Energy Flux as a Function of Photon Energy at the Aft Face of the BATH Shield on the Centerline	2-521
2-264	Photon Energy Flux as a Function of Photon Energy at the Surface of the Pressure Vessel on the Centerline	2-522
2-265	Photon Heating Rate in Aluminum 6061 ($\rho=2.71$ gm/cc) as a Function of Axial Location on the Reactor Centerline	2-523
2-266	Photon Heating Rate in Aluminum 6061 ($\rho=2.71$ gm/cc) as a Function of Axial Location on the Reactor Centerline	2-524
2-267	Photon Heating Rate in Stainless Steel 304 ($\rho=7.92$ gm/cc) as a Function of Axial Location on the Reactor Centerline	2-525
2-268	Photon Heating Rate in Stainless Steel 304 ($\rho=7.92$ gm/cc) as a Function of Axial Location on the Reactor Centerline	2-526
2-269	Photon Kerma Rate on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design (939J723) With a Graphite Core	2-527
2-270	Fast ($E>1.0$ MeV) Neutron Flux on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design With a Graphite Core	2-528
2-271	Intermediate ($0.4\text{ eV}<E<1.0$ MeV) Neutron Flux on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design (939J723) With a Graphite Core	2-529
2-272	Thermal ($E<0.4$ eV) Neutron Flux on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design (939J723) With a Graphite Core	2-530
2-273	Gamma Ray Isodose Rate (Rad-C/hr) Contours at Full Power (1511 Mw Thermal) Around the R-1 Reference Design Reactor (939J723) With Test Car and Test Cell C Shield Removed (Graphite Fuel)	2-531
2-274	Relative Radial Distribution of Heating in the Pressure Vessel Side-A, Region 22 (Graphite Core)	2-533

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-275	Relative Axial Distribution of Heating in the Pressure Vessel Side, Region 22 (Graphite Core)	2-534
2-276	Relative Radial Distribution of Heating in the Pressure Vessel Side-B, Region 42 (Graphite Core)	2-535
2-277	Relative Axial Distribution of Heating in the Pressure Vessel Side-B, Region 42 (Graphite Core)	2-536
2-278	Relative Radial Distribution of Heating in the Pressure Vessel Dome, Region 53 (Graphite Core)	2-537
2-279	Relative Axial Distribution of Heating in the Pressure Vessel Dome, Region 53 (Graphite Core)	2-538
2-280	Relative Increase Due to the Test Cell C Facility Shield on the Gamma Dose Rate as a Function of Radius on the Core Midplane	2-538
2-281	Relative Increase Due to the Test Cell C Facility Shield on the Fast Neutron Flux ($E > 1.0$ MeV) as a Function of Radius on the Core Midplane	2-540
2-282	Relative Increase Due to the Test Cell C Facility Shield on the Intermediate Neutron Flux ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) as a Function of Radius on the Core Midplane	2-541
2-283	Relative Increase Due to the Test Cell C Facility Shield on the Thermal Neutron Flux ($E < 0.4 \text{ eV}$) as a Function of Radius on the Core Midplane	2-542
2-284	Relative Increase Due to the Test Cell C Facility Shield on the Al 7039 Heating Rate as a Function of Radius on the Core Midplane	2-543
2-285	Relative Increase Due to the Test Cell C Facility Shield on the A-286 Heating Rate as a Function of Radius on the Core Midplane	2-544
2-286	Relative Increase Due to the Test Cell C Facility Shield on the Beryllium Heating Rate as a Function of Radius on the Core Midplane	2-545

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
2-287	Relative Increase Due to the Test Cell C Facility Shield on the Carbon Heating Rate as a Function of Radius on the Core Midplane	2-546
2-288	Relative Increase Due to the Test Cell C Facility Shield on the Al-6061 Heating Rate as a Function of Radius on the Core Midplane	2-547
2-289	Relative Increase Due to the Test Cell C Facility Shield on the Thermal Neutron Flux ($E < 0.4$ eV) as a Function of Axial Height on the Core Centerline	2-548
2-290	Relative Increase Due to the Test Cell C Facility Shield on the Intermediate Neutron Flux ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) as a Function of Axial Height on the Core Centerline	2-549
2-291	Relative Increase Due to the Test Cell C Facility Shield on the Fast Neutron Flux ($E > 1.0 \text{ MeV}$) as a Function of Axial Height on the Core Centerline	2-550
2-292	Relative Increase Due to the Test Cell C Facility Shield on the Gamma Dose Rate as a Function of Axial Height on the Core Centerline	2-551
2-293	Relative Increase Due to the Test Cell C Facility Shield on the Al-6061 Heating Rate as a Function of Axial Height on the Core Centerline	2-552
2-294	Relative Increase Due to the Test Cell C Facility Shield on the Al-7039 Heating Rate as a Function of Axial Height on the Core Centerline	2-553
2-295	Relative Increase Due to the Test Cell C Facility Shield on the A-286 Heating Rate as a Function of Axial Height on the Core Centerline	2-554
2-296	Relative Increase Due to the Test Cell C Facility Shield on the Carbon Heating Rate as a Function of Axial Height on the Core Centerline	2-555

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
6-1	Relative Radial Distribution of Heating in the Core Plenum, Region 28 (Composite Core)	6-26
6-2	Relative Radial Distribution of Heating in the Core Plenum, Region 28 (Composite Core)	6-27
6-3	Relative Axial Distribution of Heating in the Core Plenum, Region 28 (Composite Core)	6-28
6-4	Relative Axial Distribution of Heating in the Core Plenum, Region 28 (Composite Core)	6-29
6-5	Relative Radial Distribution of Heating in the Support Plate, Region 29 (Composite Core)	6-30
6-6	Relative Axial Distribution of Heating in the Support Plate, Region 29 (Composite Core)	6-31
6-7	Relative Radial Distribution of Heating in the Lateral Support - Forward, Region 30 (Composite Core)	6-32
6-8	Relative Axial Distribution of Heating in the Lateral Support - Forward, Region 30 (Composite Core)	6-33
6-9	Relative Radial Distribution of Heating in the Support Plate Plenum, Region 32 (Composite Core)	6-34
6-10	Relative Axial Distribution of Heating in the Support Plate Plenum, Region 32 (Composite Core)	6-35
6-11	Relative Radial Distribution of Heating in the Instrumentation Ring, Region 33 (Composite Core)	6-36
6-12	Relative Axial Distribution of Heating in the Instrumentation Ring, Region 33 (Composite Core)	6-37
6-13	Relative Radial Distribution of Heating in the Flow Baffle I, Region 37 (Composite Core)	6-38
6-14	Relative Axial Distribution of Heating in the Flow Baffle I, Region 37 (Composite Core)	6-39
6-15	Relative Radial Distribution of Heating in the Flow Baffle II, Region 50 (Composite Core)	6-40

LIST OF ILLUSTRATIONS
(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
6-16	Relative Axial Distribution of Heating in the Flow Baffle II, Region 50 (Composite Core)	6-41
6-17	Relative Radial Distribution of Heating in the Peripheral Dome Plenum-A, Region 52 (Composite Core)	6-42
6-18	Relative Axial Distribution of Heating in the Peripheral Dome Plenum-A, Region 52 (Composite Core)	6-43
6-19	Relative Radial Distribution of Heating in the Peripheral Dome Plenum-B, Region 52 (Composite Core)	6-44
6-20	Relative Axial Distribution of Heating in the Peripheral Dome Plenum-B, Region 52 (Composite Core)	6-45
6-21	Relative Radial Distribution of Heating in the Peripheral Dome Plenum-C, Region 52 (Composite Core)	6-46
6-22	Relative Axial Distribution of Heating in the Peripheral Dome Plenum-C, Region 52 (Composite Core)	6-47
6-23	Relative Radial Distribution of Photon Source in Region 9, Core Plenum	6-48
6-24	Relative Axial Distribution of Photon Source in Region 9, Core Plenum	6-49
6-25	Relative Radial Distribution of Photon Source in Region 10, Lateral Support Forward	6-50
6-26	Relative Axial Distribution of Photon Source in Region 10, Lateral Support Forward	6-51
6-27	Relative Radial Distribution of Photon Source in Region 11, Support Plate	6-52
6-28	Relative Axial Distribution of Photon Source in Region 11, Support Plate	6-53
6-29	Relative Radial Distribution of Photon Source in Region 13, Support Plate Plenum	6-54

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
6-30	Relative Axial Distribution of Photon Source in Region 13, Support Plate Plenum	6-55
6-31	Relative Radial Distribution of Photon Source in Region 14, Instrumentation Ring	6-56
6-32	Relative Axial Distribution of Photon Source in Region 14, Instrumentation Ring	6-57
6-33	Relative Radial Distribution of Photon Source in Region 28, Flow Baffle 1	6-58
6-34	Relative Axial Distribution of Photon Source in Region 28, Flow Baffle 1	6-59
6-35	Relative Radial Distribution of Photon Source in Region 29, Flow Baffle 2	6-60
6-36	Relative Axial Distribution of Photon Source in Region 29, Flow Baffle 2	6-61
6-37	Relative Radial Distribution of Photon Source in Region 30, Central Dome Plenum	6-62
6-38	Relative Axial Distribution of Photon Source in Region 30, Central Dome Plenum	6-63
6-39	Relative Radial Distribution of Photon Source in Region 31, Peripheral Dome Plenum 1	6-64
6-40	Relative Axial Distribution of Photon Source in Region 31, Peripheral Dome Plenum 1	6-65
6-41	Relative Radial Distribution of Photon Source in Region 32, Peripheral Dome Plenum 2	6-66
6-42	Relative Axial Distribution of Photon Source in Region 32, Peripheral Dome Plenum 2	6-67
6-43	Relative Radial Distribution of Photon Source in Region 33, Peripheral Dome Plenum 3	6-68

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
6-44	Relative Axial Distribution of Photon Source in Region 33, Peripheral Dome Plenum 3	6-69
6-45	Relative Radial Distribution of Heating in the Core Plenum, Region 28 (Graphite Core)	6-70
6-46	Relative Axial Distribution of Heating in the Core Plenum, Region 28 (Graphite Core)	6-71
6-47	Relative Radial Distribution of Heating in the Core Plenum, Region 28 (Graphite Core)	6-72
6-48	Relative Axial Distribution of Heating in the Core Plenum, Region 28 (Graphite Core)	6-73
6-49	Relative Radial Distribution of Heating in the Support Plate, Region 29 (Graphite Core)	6-74
6-50	Relative Axial Distribution of Heating in the Support Plate, Region 29 (Graphite Core)	6-75
6-51	Relative Radial Distribution of Heating in the Lateral Support - Forward, Region 30 (Graphite Core)	6-76
6-52	Relative Axial Distribution of Heating in the Lateral Support - Forward, Region 30 (Graphite Core)	6-77
6-53	Relative Radial Distribution of Heating in the Support Plate - Plenum, Region 32 (Graphite Core)	6-78
6-54	Relative Axial Distribution of Heating in the Support Plate - Plenum, Region 32 (Graphite Core)	6-79
6-55	Relative Radial Distribution of Heating in the Instrumentation Ring, Region 33 (Graphite Core)	6-80
6-56	Relative Axial Distribution of Heating in the Instrumentation Ring, Region 33 (Graphite Core)	6-81
6-57	Relative Radial Distribution of Heating in the Flow Baffle I, Region 37 (Graphite Core)	6-82

LIST OF ILLUSTRATIONS
(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
6-58	Relative Axial Distribution of Heating in the Flow Baffle I, Region 37 (Graphite Core)	6-83
6-59	Relative Radial Distribution of Heating in the Flow Baffle II, Region 50 (Graphite Core)	6-84
6-60	Relative Axial Distribution of Heating in the Flow Baffle II, Region 50 (Graphite Core)	6-85
6-61	Relative Radial Distribution of Heating in the Peripheral Dome Plenum A, Region 52-A (Graphite Core)	6-86
6-62	Relative Axial Distribution of Heating in the Peripheral Dome Plenum A, Region 52-A (Graphite Core)	6-87
6-63	Relative Radial Distribution of Heating in the Peripheral Dome Plenum B, Region 52-B (Graphite Core)	6-88
6-64	Relative Axial Distribution of Heating in the Peripheral Dome Plenum B, Region 52-B (Graphite Core)	6-89
6-65	Relative Radial Distribution of Heating in the Peripheral Dome Plenum C, Region 52-C (Graphite Core)	6-90
6-66	Relative Axial Distribution of Heating in the Peripheral Dome Plenum C, Region 52-C (Graphite Core)	6-91
6-67	Relative Radial Distribution of Photon Source in Region 9, Core Plenum	6-92
6-68	Relative Axial Distribution of Photon Source in Region 9, Core Plenum	6-93

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
7-1	Percent "Void" in Shield Regions Versus Radius	7-55
7-2	Percent of Non-BATH Material in Shield Regions Versus Radius	7-56
7-3	Relative Radial Distribution of Heating in the Aft Central Shield Plate, Region 35 (Composite Core)	7-57
7-4	Relative Axial Distribution of Heating in the Aft Central Shield Plate, Region 35 (Composite Core)	7-58
7-5	Relative Radial Distribution of Heating in the BATH Central Shield, Region 36 (Composite Core)	7-59
7-6	Relative Radial Distribution of Heating in the BATH Central Shield, Region 36 (Composite Core)	7-60
7-7	Relative Axial Distribution of Heating in the BATH Central Shield, Region 36 (Composite Core)	7-61
7-8	Relative Axial Distribution of Heating in the BATH Central Shield, Region 36 (Composite Core)	7-62
7-9	Relative Radial Distribution of Heating in the BATH Peripheral Shield I, Region 38 (Composite Core)	7-63
7-10	Relative Radial Distribution of Heating in the BATH Peripheral Shield I, Region 38 (Composite Core)	7-64
7-11	Relative Axial Distribution of Heating in the BATH Peripheral Shield I, Region 38 (Composite Core)	7-65
7-12	Relative Axial Distribution of Heating in the BATH Peripheral Shield I, Region 38 (Composite Core)	7-66
7-13	Relative Radial Distribution of Heating in the BATH Peripheral Shield II, Region 39 (Composite Core)	7-67
7-14	Relative Axial Distribution of Heating in the BATH Peripheral Shield II, Region 39 (Composite Core)	7-68
7-15	Relative Radial Distribution of Heating in the BATH Peripheral Shield III, Region 40 (Composite Core)	7-69
7-16	Relative Radial Distribution of Heating in the BATH Peripheral Shield III, Region 40 (Composite Core)	7-70

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
7-17	Relative Axial Distribution of Heating in the BATH Peripheral Shield III, Region 40 (Composite Core)	7-71
7-18	Relative Axial Distribution of Heating in the BATH Peripheral Shield III, Region 40 (Composite Core)	7-72
7-19	Relative Radial Distribution of Heating in the BATH Peripheral Shield IV, Region 41 (Composite Core)	7-73
7-20	Relative Radial Distribution of Heating in the BATH Peripheral Shield IV, Region 41 (Composite Core)	7-74
7-21	Relative Axial Distribution of Heating in the BATH Peripheral Shield IV, Region 41 (Composite Core)	7-75
7-22	Relative Axial Distribution of Heating in the BATH Peripheral Shield IV, Region 41 (Composite Core)	7-76
7-23	Relative Radial Distribution of Heating in the Lead Central Shield, Region 43 (Composite Core)	7-77
7-24	Relative Radial Distribution of Heating in the Lead Central Shield, Region 43 (Composite Core)	7-78
7-25	Relative Axial Distribution of Heating in the Lead Central Shield, Region 43 (Composite Core)	7-79
7-26	Relative Axial Distribution of Heating in the Lead Central Shield, Region 43 (Composite Core)	7-80
7-27	Relative Radial Distribution of Heating in the Lead Peripheral Shield I, Region 44 (Composite Core)	7-81
7-28	Relative Radial Distribution of Heating in the Lead Peripheral Shield I, Region 44 (Composite Core)	7-82
7-29	Relative Axial Distribution of Heating in the Lead Peripheral Shield I, Region 44 (Composite Core)	7-83
7-30	Relative Axial Distribution of Heating in the Lead Peripheral Shield I, Region 44 (Composite Core)	7-84

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
7-31	Relative Radial Distribution of Heating in the Lead Peripheral Shield II, Region 45 (Composite Core)	7-85
7-32	Relative Axial Distribution of Heating in the Lead Peripheral Shield II, Region 45 (Composite Core)	7-86
7-33	Relative Radial Distribution of Heating in the Lead Peripheral Shield III, Region 46 (Composite Core)	7-87
7-34	Relative Radial Distribution of Heating in the Lead Peripheral Shield III, Region 46 (Composite Core)	7-88
7-35	Relative Axial Distribution of Heating in the Lead Peripheral Shield III, Region 46 (Composite Core)	7-89
7-36	Relative Axial Distribution of Heating in the Lead Peripheral Shield III, Region 46 (Composite Core)	7-90
7-37	Relative Radial Distribution of Heating in the Lead Peripheral Shield IV, Region 47 (Composite Core)	7-91
7-38	Relative Radial Distribution of Heating in the Lead Peripheral Shield IV, Region 47 (Composite Core)	7-92
7-39	Relative Axial Distribution of Heating in the Lead Peripheral Shield IV, Region 47 (Composite Core)	7-93
7-40	Relative Axial Distribution of Heating in the Lead Peripheral Shield IV, Region 47 (Composite Core)	7-94
7-41	Relative Radial Distribution of Heating in the Peripheral Shield Plate, Region 48 (Composite Core)	7-95
7-42	Relative Axial Distribution of Heating in the Peripheral Shield Plate, Region 48 (Composite Core)	7-96
7-43	Relative Radial Distribution of Heating in the Shield Plenum, Region 49 (Composite Core)	7-97
7-44	Relative Axial Distribution of Heating in the Shield Plenum, Region 49 (Composite Core)	7-98

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
7-45	Relative Radial Distribution of Photon Source in Region 15, Central Shield Plate - Aft	7-99
7-46	Relative Axial Distribution of Photon Source in Region 15, Central Shield Plate - Aft	7-100
7-47	Relative Radial Distribution of Photon Source in Region 16, BATH Central Shield	7-101
7-48	Relative Axial Distribution of Photon Source in Region 16, BATH Central Shield	7-102
7-49	Relative Radial Distribution of Photon Source in Region 17, BATH Peripheral Shield - 1	7-103
7-50	Relative Axial Distribution of Photon Source in Region 17, BATH Peripheral Shield - 1	7-104
7-51	Relative Radial Distribution of Photon Source in Region 18, BATH Peripheral Shield - 2	7-105
7-52	Relative Axial Distribution of Photon Source in Region 18, BATH Peripheral Shield - 2	7-106
7-53	Relative Radial Distribution of Photon Source in Region 19, BATH Peripheral Shield - 3	7-107
7-54	Relative Axial Distribution of Photon Source in Region 19, BATH Peripheral Shield - 3	7-108
7-55	Relative Radial Distribution of Photon Source in Region 20, BATH Peripheral Shield - 4	7-109
7-56	Relative Axial Distribution of Photon Source in Region 20, BATH Peripheral Shield - 4	7-110
7-57	Relative Radial Distribution of Photon Source in Region 21, Lead Central Shield	7-111
7-58	Relative Axial Distribution of Photon Source in Region 21, Lead Central Shield	7-112

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
7-59	Relative Radial Distribution of Photon Source in Region 22, Lead Peripheral Shield -1	7-113
7-60	Relative Axial Distribution of Photon Source in Region 22, Lead Peripheral Shield - 1	7-114
7-61	Relative Radial Distribution of Photon Source in Region 23, Lead Peripheral Shield - 2	7-115
7-62	Relative Axial Distribution of Photon Source in Region 23, Lead Peripheral Shield - 2	7-116
7-63	Relative Radial Distribution of Photon Source in Region 24, Lead Peripheral Shield - 3	7-117
7-64	Relative Axial Distribution of Photon Source in Region 24, Lead Peripheral Shield - 3	7-118
7-65	Relative Radial Distribution of Photon Source in Region 25, Lead Peripheral Shield - 4	7-119
7-66	Relative Axial Distribution of Photon Source in Region 25, Lead Peripheral Shield - 4	7-120
7-67	Relative Radial Distribution of Photon Source in Region 26, Peripheral Shield Plate	7-121
7-68	Relative Axial Distribution of Photon Source in Region 26, Peripheral Shield Plate	7-122
7-69	Relative Radial Distribution of Photon Source in Region 27, Shield Plenum	7-123
7-70	Relative Axial Distribution of Photon Source in Region 27, Shield Plenum	7-124
7-71	Relative Radial Distribution of Heating in the Aft Central Shield Plate, Region 35 (Graphite Core)	7-125
7-72	Relative Axial Distribution of Heating in the Aft Central Shield Plate, Region 35 (Graphite Core)	7-126

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No</u>
7-73	Relative Radial Distribution of Heating in the BATH Central Shield, Region 36 (Graphite Core)	7-127
7-74	Relative Axial Distribution of Heating in the BATH Central Shield, Region 36 (Graphite Core)	7-128
7-75	Relative Radial Distribution of Heating in the BATH Central Shield, Region 36 (Graphite Core)	7-129
7-76	Relative Axial Distribution of Heating in the BATH Central Shield, Region 36 (Graphite Core)	7-130
7-77	Relative Radial Distribution of Heating in the BATH Peripheral Shield I, Region 38 (Graphite Core)	7-131
7-78	Relative Axial Distribution of Heating in the BATH Peripheral Shield I, Region 38 (Graphite Core)	7-132
7-79	Relative Radial Distribution of Heating in the BATH Peripheral Shield I, Region 38 (Graphite Core)	7-133
7-80	Relative Axial Distribution of Heating in the BATH Peripheral Shield I, Region 38 (Graphite Core)	7-134
7-81	Relative Radial Distribution of Heating in the BATH Peripheral Shield II, Region 39 (Graphite Core)	7-135
7-82	Relative Axial Distribution of Heating in the BATH Peripheral Shield II, Region 39 (Graphite Core)	7-136
7-83	Relative Radial Distribution of Heating in the BATH Peripheral Shield III, Region 40 (Graphite Core)	7-137
7-84	Relative Radial Distribution of Heating in the BATH Peripheral Shield III, Region 40 (Graphite Core)	7-138
7-85	Relative Axial Distribution of Heating in the BATH Peripheral Shield III, Region 40 (Graphite Core)	7-139
7-86	Relative Axial Distribution of Heating in the BATH Peripheral Shield III, Region 40 (Graphite Core)	7-140

LIST OF ILLUSTRATIONS
(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
7-87	Relative Radial Distribution of Heating in the BATH Peripheral Shield IV, Region 41 (Graphite Core)	7-141
7-88	Relative Radial Distribution of Heating in the BATH Peripheral Shield IV, Region 41 (Graphite Core)	7-142
7-89	Relative Axial Distribution of Heating in the BATH Peripheral Shield IV, Region 41 (Graphite Core)	7-143
7-90	Relative Axial Distribution of Heating in the BATH Peripheral Shield IV, Region 41 (Graphite Core)	7-144
7-91	Relative Radial Distribution of Heating in the Lead Central Shield, Region 43 (Graphite Core)	7-145
7-92	Relative Radial Distribution of Heating in the Lead Central Shield, Region 43 (Graphite Core)	7-146
7-93	Relative Axial Distribution of Heating in the Lead Central Shield, Region 43 (Graphite Core)	7-147
7-94	Relative Axial Distribution of Heating in the Lead Central Shield, Region 43 (Graphite Core)	7-148
7-95	Relative Radial Distribution of Heating in the Lead Peripheral Shield I, Region 44 (Graphite Core)	7-149
7-96	Relative Radial Distribution of Heating in the Lead Peripheral Shield I, Region 44 (Graphite Core)	7-150
7-97	Relative Axial Distribution of Heating in the Lead Peripheral Shield I, Region 44 (Graphite Core)	7-151
7-98	Relative Axial Distribution of Heating in the Lead Peripheral Shield I, Region 44 (Graphite Core)	7-152
7-99	Relative Radial Distribution of Heating in the Lead Peripheral Shield II, Region 45 (Graphite Core)	7-153
7-100	Relative Axial Distribution of Heating in the Lead Peripheral Shield II, Region 45 (Graphite Core)	7-154

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
7-101	Relative Radial Distribution of Heating in the Lead Peripheral Shield III, Region 46 (Graphite Core)	7-155
7-102	Relative Radial Distribution of Heating in the Lead Peripheral Shield III, Region 46 (Graphite Core)	7-156
7-103	Relative Axial Distribution of Heating in the Lead Peripheral Shield III, Region 46 (Graphite Core)	7-157
7-104	Relative Axial Distribution of Heating in the Lead Peripheral Shield III, Region 46 (Graphite Core)	7-158
7-105	Relative Radial Distribution of Heating in the Lead Peripheral Shield IV, Region 47 (Graphite Core)	7-159
7-106	Relative Radial Distribution of Heating in the Lead Peripheral Shield IV, Region 47 (Graphite Core)	7-160
7-107	Relative Axial Distribution of Heating in the Lead Peripheral Shield IV, Region 47 (Graphite Core)	7-161
7-108	Relative Axial Distribution of Heating in the Lead Peripheral Shield IV, Region 47 (Graphite Core)	7-162
7-109	Relative Radial Distribution of Heating in the Peripheral Shield Plate, Region 48 (Graphite Core)	7-163
7-110	Relative Axial Distribution of Heating in the Peripheral Shield Plate, Region 48 (Graphite Core)	7-164
7-111	Relative Radial Distribution of Heating in the Shield Plenum, Region 49 (Graphite Core)	7-165
7-112	Relative Axial Distribution of Heating in the Shield Plenum, Region 49 (Graphite Core)	7-166
7-113	Maximum Gamma Ray Kerma Rate on the Tangent Plane Versus Centerline Thickness of BATH Material	7-167
7-114	Maximum Fast ($E > 1.0$ MeV) Flux on the Tangent Plane Versus Centerline Thickness of BATH Material	7-168

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
8-1	Relative Radial Distribution of Heating in the Lateral Support, Region 17 (Composite Core)	8-24
8-2	Relative Axial Distribution of Heating in the Lateral Support, Region 17 (Composite Core)	8-25
8-3	Relative Radial Distribution of Heating in the Structure, Region 18 (Composite Core)	8-26
8-4	Relative Radial Distribution of Heating in the Structure, Region 18 (Composite Core)	8-27
8-5	Relative Axial Distribution of Heating in the Structure, Region 18 (Composite Core)	8-28
8-6	Relative Axial Distribution of Heating in the Structure, Region 18 (Composite Core)	8-29
8-7	Relative Radial Distribution of Heating in the Reflector, Regions 19 - 21 (Composite Core)	8-30
8-8	Relative Axial Distribution of Heating in the Reflector, Regions 19 - 21 (Composite Core)	8-31
8-9	Relative Radial Distribution of Heating in the Reflector, Regions 19 - 21 (Composite Core)	8-22
8-10	Relative Axial Distribution of Heating in the Reflector, Regions 19 - 21 (Composite Core)	8-33
8-11	Relative Radial Distribution of Heating in the Reflector, Regions 19 - 21 (Composite Core)	8-34
8-12	Relative Axial Distribution of Heating in the Reflector, Regions 19 - 21 (Composite Core)	8-35
8-13	Relative Radial Distribution of Heating in the Aft Reflector Hardware, Region 26 (Composite Core)	8-36
8-14	Relative Axial Distribution of Heating in the Aft Reflector Hardware, Region 26 (Composite Core)	8-37

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
8-15	Relative Radial Distribution of Heating in the Forward Reflector Hardware I, Region 31 (Composite Core)	8-38
8-16	Relative Axial Distribution of Heating in the Forward Reflector Hardware I, Region 31 (Composite Core)	8-39
8-17	Relative Radial Distribution of Heating in the Forward Reflector Hardware II, Region 34 (Composite Core)	8-40
8-18	Relative Axial Distribution of Heating in the Forward Reflector Hardware II, Region 34 (Composite Core)	8-41
8-19	Azimuthal Heating Rate Distribution for the Control Drum Vanes With the Control Drums Positioned at an Angle of 50 Degrees	8-42
8-20	Relative Radial Distribution of Photon Source in Region 4, Lateral Support	8-43
8-21	Relative Axial Distribution of Photon Source in Region 4, Lateral Support	8-44
8-22	Relative Radial Distribution of Photon Source in Region 5, Aluminum Structure	8-45
8-23	Relative Axial Distribution of Photon Source in Region 5, Aluminum Structure	8-46
8-24	Relative Radial Distribution of Photon Source in Region 6, Reflector	8-47
8-25	Relative Axial Distribution of Photon Source in Region 6, Reflector	8-48
8-26	Relative Radial Distribution of Photon Source in Region 8, Aft Reflector Hardware	8-49
8-27	Relative Axial Distribution of Photon Source in Region 8, Aft Reflector Hardware	8-50
8-28	Relative Radial Distribution of Photon Source in Region 12, Forward Reflector Hardware	8-51

LIST OF ILLUSTRATIONS (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
8-29	Relative Axial Distribution of Photon Source in Region 12, Forward Reflector Hardware	8-52
8-30	Relative Radial Distribution of Heating in the Lateral Support, Region 17 (Graphite Core)	8-53
8-31	Relative Axial Distribution of Heating in the Lateral Support, Region 17 (Graphite Core)	8-54
8-32	Relative Radial Distribution of Heating in the Aluminum Structure, Region 18 (Graphite Core)	8-55
8-33	Relative Axial Distribution of Heating in the Structure, Region 18 (Graphite Core)	8-56
8-34	Relative Radial Distribution of Heating in the Aluminum Structure, Region 18 (Graphite Core)	8-57
8-35	Relative Axial Distribution of Heating in the Structure, Region 18 (Graphite Core)	8-58
8-36	Relative Radial Distribution of Heating in the Reflector, Regions 19 - 21 (Graphite Core)	8-59
8-37	Relative Axial Distribution of Heating in the Reflector, Regions 19 - 21 (Graphite Core)	8-60
8-38	Relative Radial Distribution of Heating in the Reflector, Regions 19 - 21 (Graphite Core)	8-61
8-39	Relative Axial Distribution of Heating in the Reflector, Regions 19 - 21 (Graphite Core)	8-62
8-40	Relative Radial Distribution of Heating in the Reflector, Regions 19 - 21 (Graphite Core)	8-63
8-41	Relative Axial Distribution of Heating in the Reflector, Regions 19 - 21 (Graphite Core)	8-64
8-42	Relative Radial Distribution of Heating in the Aft Reflector Hardware, Region 26 (Graphite Core)	8-65

LIST OF ILLUSTRATIONS

(Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page No.</u>
8-43	Relative Axial Distribution of Heating in the Aft Reflector Hardware, Region 26 (Graphite Core)	8-66
8-44	Relative Radial Distribution of Heating in the Forward Reflector Hardware, Region 31 (Graphite Core)	8-67
8-45	Relative Axial Distribution of Heating in the Forward Reflector Hardware I, Region 31 (Graphite Core)	8-68
8-46	Relative Radial Distribution of Heating in the Forward Reflector Hardware II, Region 34 (Graphite Core)	8-69
8-47	Relative Axial Distribution of Heating in the Forward Reflector Hardware II, Region 34 (Graphite Core)	8-70
8-48	Azimuthal Heating Rate Distribution for the Control Drum Vanes with the Control Drums Positioned at an Angle of 50 Degrees	8-71

1.0 INTRODUCTION

1.1 PURPOSE OF REPORT

The purpose of this report is to document the Radiation and Shielding Analysis of the NUCLEAR SUBSYSTEM (NSS) and its Engineering Critical Components (ECC). This report constitutes the primary source of all steady state and transient radiation and shielding data. The detailed Radiation and Shielding Analysis consisted of the following:

Support for the derivation and establishment of appropriate performance and design requirements for the NSS and its ECs. Establishment and substantiation of the appropriate parameters of the selected design.

The following detailed information is contained in this report:

- 1) Radiation sources and distributions for both a composite and graphite fueled Nuclear Subsystem,
- 2) Radiation heating rates and distributions for both a composite and graphite fueled Nuclear Subsystem,
- 3) Development of an internal shield design to meet the Engine Specification radiation limitations for a composite fueled Nuclear Subsystem,
- 4) Radiation environment internal and external to both a composite and graphite fueled Nuclear Subsystem, and,
- 5) Development of uncertainty factors related to radiation analyses data.

1.2 CONTENTS OF REPORT

This report presents radiation and shielding results with both composite and graphite fueled cores. Radiation and shielding data are presented for the Nuclear Subsystem (NSS) as a whole (Chapter 2) and for the following Type A Engineering Critical Components (ECC):

- 1) Support Plate and Plena, EC-677561 (Chapter 6),

- 2) Internal Shield, EC-677562 (Chapter 7),
- 3) Reflector Assembly, EC-677559 (Chapter 8),
- 4) Control Drum Drive Assembly, EC-677585 (Chapter 9),
- 5) Structural Support Coolant Assembly, EC-677575 (Chapter 10), and,
- 6) Nuclear Subsystem Instrumentation, EC-677565 (Chapter 11).

Radiation and shielding data for the following Type C ECCs are not presented because these ECCs are involved in Trade Studies whose results are to be reported for NSS PDR:

- 1) Fuel Element, EC-677566 (Chapter 3),
- 2) Cluster Hardware, EC-677558 (Chapter 4),
- 3) Core Periphery, EC-677564 (Chapter 5), and,
- 4) The Lateral Support portion of the Reflector Assembly, EC-677562 (Chapter 8).

The radiation and shielding data for these ECCs are reported in the appropriate S-54 Data Item as follows:

- 1) Fuel Element -- Trade Study No. 772,
- 2) Cluster Hardware -- Trade Study No. 774,
- 3) Core Periphery -- Trade Study No. 778, and,
- 4) Lateral Support -- Trade Study No. 782.

In general, this Data Item contains the following information for the NSS and each ECC with both composite and graphite fuel in the core:

- 1) Description of the component,
- 2) Summary of results and conclusions,
- 3) The requirements towards which the information is directed,
- 4) The method of analysis which includes
 - a) a description of the analytical procedures employed,
 - b) a description of the analytical models used,

- c) a description of the status/maturity of the input data upon which the results are based, and,
 - d) a discussion of the validity of the analysis, including comparisons with experimental results and the derivation of the uncertainties in the radiation analysis data.
- 5) The results, analysis, and interpretation of the data which contains the following technical information for both steady state and transient operation of the NSS:
 - a) the internal radiation environment, both neutron fluxes and photon dose rates, the time integrated internal radiation environment, both neutron fluences and photon dose, and the uncertainties in the internal environment data,
 - b) the radiation sources which include the axial and radial distribution of the sources by reactor region and the total source strength by photon energy group and reactor region,
 - c) the radiation heat deposition which includes the axial and radial distribution of the heat by reactor region, the total heat deposited by reactor region, the effect of Test Cell C on the total heat deposited in major reactor regions, and the uncertainties in radiation heating data.

In addition, the NSS and Internal Shield chapters (Chapter 2.0 and 7.0, respectively) contain information on: 1) the external radiation environment at the Engine Specification locations, 2) the external radiation environment in the vicinity of the NSS under "space like" conditions for preliminary evaluations of the effect of the NSS on Test Cell C, and, 3) the uncertainties in the external radiation environment. The recommended internal shield design to meet the external radiation levels in the Engine Specification is described in Chapter 7.

2.0 NUCLEAR SUBSYSTEM

2.1 INTRODUCTION

The NSS consists of a hydrogen cooled, graphite moderated, intermediate neutron energy reactor and its associated reactivity control devices consisting of neutron absorbing control drums with their actuation system and the Structural Support Coolant Assembly.

2.1.1 Description of the Nuclear Subsystem

The Nuclear Subsystem includes a core, reflector, core support plate and associated piping and an internal shield. The core contains fuel element clusters. Each cluster, except for the irregular clusters in the periphery, consists of 6 fueled elements surrounding a central unfueled element. A counterflow support stem passes through each central element to the support plate. The support stems carry the core pressure drop load to the core support plate and also accommodate a varying hydrogen inventory for core reactivity control.

The core support plate transmits the core pressure drop load to the reflector and also incorporates plenums for support stem hydrogen feed, exhaust and bypass.

The reflector surrounds the core and, as its name implies, reflects neutrons into the core to sustain reactivity. A lateral support system is also housed in the reflector. This supplies radial restraint for the core acting through the core periphery which consists of thermal and flow barriers backed up by filler strips which round the core into a cylindrical shape for interfacing with the lateral support. Eighteen control drums are contained in the reflector. Each control drum has a neutron absorbing plate covering approximately 120 degrees of its external surface. Synchronous rotation of the control drums provides the means then to establish reactor criticality.

An internal shield, forward of the core support plate, completes the major components of the reactor. This shield provides radiation protection for engine components.

The NSS is comprised of the following Engineering Critical Components (ECC):

- 1) Fuel Element (FE), EC-677566,
- 2) Cluster Hardware (CH), EC-677558,
- 3) Core Periphery (CP), EC-677564,
- 4) Support Plate and Plena (SP/P), EC-677561,
- 5) Internal Shield (IS), EC-677562,
- 6) Reflector Assembly (RA), EC-677559,
- 7) Control Drum Drive Assembly (CDDA), EC-677585,
- 8) Structural Support Coolant Valve Actuator (SSCVA), EC-677576,
- 9) Nuclear Subsystem Instrumentation (NSSI), EC-677565.

2.1.2 Content of Chapter

This chapter presents the results of radiation and shielding analyses of both the composite and graphite candidate designs in support of the NSS for Data Item S-100. Section 2.2 summarizes the major radiation and shielding characteristics of each design which provide a basis for design evaluation with respect to the NSS Specification Requirements itemized in Section 2.3. Section 2.4 describes the analytical methods and models employed in this study and the results of application of these methods to analyses of radiation and shielding related experiments performed on the PAX/R-1 reactor. Based upon these analyses, the validity of these methods to the NSS design was determined and uncertainty factors applicable to the radiation data were developed. Section 2.5 presents in detail the results and analysis of the NSS. In particular, the following data are included:

- 1) fast, intermediate, and thermal neutron flux, gamma ray dose rate, and neutron and photon spectra at selected locations,
- 2) the photon source strength by energy group for each reactor region defined by the NSS portion of the Common Radiation Analysis Model (CRAM),

- 3) the total heat deposited in each reactor region defined by the heating rate model,
- 4) estimates of the effect of Test Cell C on the NSS environment,
- 5) the fission product photon source strength for various times after shutdown following full power operation of 10, 30, and 60 minutes,
- 6) comparisons of the calculated radiation environment data with the radiation related Engine Specifications,
- 7) neutron flux and gamma ray dose rate as a function of polar angle on several meridian rings and in the form of isodose and isoflux contours,
- 8) the gamma ray dose rate as a function of time after shutdown,
- 9) the decay heat generated after shutdown,
- 10) the transient photon sources, and,
- 11) the radiation environment at key locations at selected times during operation.

2.2 SUMMARY OF RESULTS AND CONCLUSIONS

This section presents some of the highlights of Radiation and Shielding analysis of the overall Nuclear Subsystem. Analysis is discussed for both composite fuel and graphite fuel in the core.

2.2.1 External Radiation Environment

Analysis of major importance is that related to the Nuclear Radiation Shielding Function of the NSS (CP-677555A). As defined therein, the radiation shielding requirements for the unmanned configuration of the NERVA engine shall be satisfied by NSS shielding internal to the pressure vessel to limit the radiation leakage 1) through a plane 63 inches forward of the core center (perpendicular to the engine axis), and, 2) at critical components in the engine system. Specifically, the "critical components" are three locations pertaining to the lower thrust structure and one location pertaining to the turbopump. The radiation leakage values specified by ANSC are given in Tables 2-1 and 2-2 of this report in terms of photon kerma rate and fast, intermediate and thermal neutron flux.

The radiation environment on the plane and at the critical components was calculated for the R-1 reference design employing the BATH-lead Internal Shield. (The detailed analysis is presented in Section 2.5) These results show that all the specifications are met with one exception--the intermediate neutron specification out to a radius of approximately 13 inches on the tangent plane which is approximately 16 percent higher than the specification. However, the calculated gamma radiation environment is about a factor of two lower than the specification values on the plane. This means that more shielding thickness of BATH at the centerline is needed to meet the intermediate neutron specification while lead could be removed and stay within the other requirements.

The above results are basically true for either a composite or graphite fueled Nuclear Subsystem because: 1) no significant difference in neutron fluxes external to the reactor was observed, and, 2) the gamma ray dose rate for the composite fueled core is only about 15 percent less than for the graphite fueled core.

The calculated radiation environment on the plane and at the critical component locations referred to above apply to the nominal calculations adjusted by 1σ uncertainties and by factors to account for radiation streaming through penetrations in the Internal Shield.* The 1σ uncertainties for photons and neutrons are 1.4 and 3.1 respectively.

The one standard deviation in the radiation environment was derived by comparing calculated and experimentally determined external radiation environment. (Section 2.4.4 describes this analysis.) With the one standard deviation adjustment to the radiation environment at the engine specification locations, the probability of exceeding the specified radiation environment is 0.16. This represents the one sided probability for exceeding a specified limit when possible one standard deviations in the data have been considered.

Figure 2-1 displays this situation whereby the calculated radiation environment represents the nominal value of the environment with a normal distribution of possible radiation environment values about the nominal value. Since the shield is to be designed to account for possible one standard deviations in the calculated data, then by definition 68 percent (integral under the curve in Figure 2-1 from -1σ to $+1\sigma$) of the possible radiation values lie within plus or minus one standard deviation of the nominal value with 16 percent (integral under the curve from -1σ to $-\infty$) of the remaining values lying below the minus one standard deviation and 16 percent (integral under the curve from $+1\sigma$ to $+\infty$) of the remaining values lying above plus one standard deviation. Therefore, from the requirement that the shield be designed to include possible one standard deviations in the radiation environment, it follows by the definition of a normal probability distribution that the possibility of the calculated radiation environment exceeding the specified limit is 0.16.

The factors to account for radiation streaming depend upon the location of the critical component and/or radius on the plane. Briefly, these factors for photons range from 1.2 to 1.5, and for neutrons from 1.5 to 2.0.

The locations of the Internal Shield specifications are limited to short distances from the reactor. The largest distance from the center of the core is less than nine feet, i.e., at

* Per Letter, R. W. Schroeder to A. L. Feldman, "Shielding Discussions of May 28, 1970," 1 June 1970.

the turbopump location. Hence, radiation streaming at these distances is not as significant as that which might occur at large distances from the reactor.

No radiation streaming factor was applied to the calculated environment at the turbopump location. The TPA specification location is at a radius of 36 inches, while the outer radius of the pressure vessel is only approximately 27.8 inches. Hence, the radiation incident at the TPA location is primarily due to radiation leakage through the side surface of the subsystem, rather than through the top surface. Specifically, the two-dimensional discrete ordinates analysis shows that about 80 percent of the total neutron flux and photon dose rate at the TPA is due to side leakage.

It was pointed out above that the internal shield mass thicknesses should be modified to include more BATH and less lead in order to meet the intermediate neutron specification on the plane. (If the specification on the plane is met, calculations have demonstrated that the specification at the critical components are also met.) The centerline mass thickness of the BATH and lead in the reference design shield is about 104 gm/cm^2 . The radiation analysis computer code model of this shield was comprised of the following approximate weights: BATH, 2160 pounds; lead, 950 pounds; and structure, 260 pounds (total weight of approximately 3370 pounds).

Analysis described in Section 7.5.3 of Chapter 7.0 identifies a shield that meets the specifications in all respects. This shield is defined as an "all BATH" shield, having a centerline mass thickness of approximately 87 gm/cm^2 of BATH material (not including structure). Assuming a BATH shield region density of 2.60 gm/cc , which implies a coolant fraction of 0.0887 in the central shield, the centerline thickness is 13.2 inches. Adding the current aluminum forward and aft end plate thickness of 1.2 inches, the total shield assembly thickness would be 14.4 inches and the total weight would be about 2870 pounds.

The "all BATH" shield described above exceeds the allocated overall shield length of 13.6 inches. Additional analysis (described in Chapter 7.0) identified a 13.6 inch thick BATH-lead shield consisting of: 12.15 inches of BATH, 0.25 inches of lead, 1.2 inches of

aluminum structure. This shield weighs approximately 2820 pounds, i.e., about 500 pounds less than the reference design. This shield meets all the specifications on the 63 inch plane except the intermediate neutron flux which is exceeded by only about 10 percent.

The question remaining to be answered is: In the case of a graphite fueled Nuclear Subsystem, would the same shield used with a composite core perform as well against the specification? To answer this question, a detailed analysis was also performed for the graphite fueled case. The results (see Chapter 7.0) show that all the specifications are met with the all BATH shield with one exception--the gamma ray dose rate on the plane is exceeded by about ten percent. For the required 13.6 inch length, the lead-BATH shield designed for a composite core is expected to meet the gamma ray dose rate specification also to within ten percent.

One of the major reasons why the Internal Shield mass can be decreased is that the current reference NSS design now includes 15 grams/cm^2 of material forward of the Internal Shield on the centerline, whereas the previous design (939J685E) had only approximately 4 grams/cm^2 of material forward of the Internal Shield.

Any subsequent major changes in the Nuclear Subsystem design would, of course, require further evaluation of the shields defined above. Furthermore, any weight "quoted" for the Internal Shield is very dependent on the diameter of the subsystem.

Other radiation environment data were developed in the form of meridian ring plots and isodose and isoflux plots so that preliminary analysis related to the Test Cell C environment could be evaluated.

It is important to note that the photon radiation level at a polar angle of 90 degrees (10 feet from the core midplane) is a factor of 70 higher than at a polar angle of 0 degrees, i. e., out the "dome end". This is mainly due to the attenuation by the Internal Shield and other hardware in the axial direction. (The current reference design has a mass thickness of about 175 gm/cm^2 forward of the core on the centerline.)

The fast neutron flux displays a factor of 60 increase from 0° to 90° , while the intermediate and thermal neutron flux increase by factors of 200 and 240, respectively.

The delayed gamma ray dose rate for various times after shutdown following full power (1511 MW thermal) runs of 10, 30 and 60 minutes were also developed for both a composite and a graphite fueled core. The increase in the gamma ray dose rate from 0° to a polar angle 90° (at 10 feet from the core center), one hour after shutdown, is almost a factor of 400 compared to the factor of 70 cited above during operation. This is due to the softer photon spectrum exhibited after shutdown. At 90° , one hour after shutdown, the gamma ray dose rate for the composite core case is 75 percent less than for the graphite core case because of the higher density core of the composite core and, hence, more "self-attenuation".

Of course, the gamma ray dose rate decreases as a function of time after shutdown; for example, following a 60 minute full power run, the dose rate at a polar angle of 0° is only 6 Rad-C/Hr one day following shutdown compared to 1.8×10^3 only one hour following shutdown. The comparable dose rate during operation is 9.5×10^5 .

2.2.2 Internal Radiation Environment

The radiation environment, i.e., neutron fluxes and photon dose rates as well as fluence and dose, are presented in Section 2.5 in graphical and tabular form for both a composite and graphite fueled nuclear subsystem.

Comparison of the photon dose rates internal to the NSS show a definite trend: The dose rate for the graphite fuel case is always higher than that for the composite, ranging from a factor of 1.4 in the core to a factor of 1.05 at the vessel dome. This is to be expected since the composite core has a higher density (2.65 gm/cc) than the graphite core (1.91 gm/cc) and, hence, provides more "self attenuation".

A comparison of the neutron fluxes in the graphite and composite core nuclear subsystems shows that the fast and intermediate neutron fluxes for the composite cores are consistently greater than those for the graphite core. The ratio of the graphite to composite fast and intermediate fluxes is typically in the range of 0.82 to 0.98. This is due to the lower carbon to uranium ratio for the composite core. The removal of the efficient carbon moderator (fast neutron thermalizer) yields a harder (more fast energy neutrons than low

energy neutrons) neutron spectrum. This also creates a condition where the median neutron energy causing fission is higher for the composite core.

The presence of more thermal neutrons in the graphite core (factor of approximately 5 at the core center) vividly illustrates the harder neutron spectrum created by the removal of carbon moderator. This large difference in thermal neutrons between the two cores at the core center is almost eliminated in non-core regions. For example, at the inner surface of the reflector at the core midplane, the graphite core thermal neutron flux is a factor of 1.07 greater than that for the composite core. At the aft surface of the BATH central shield, however, the graphite core thermal flux is a factor of 0.96 that of the composite core. The reflector ratio is greater than one since the thermal neutrons at this location are due to a combination of the thermal flux leakage from the core (higher for graphite) and fast neutron (lower for graphite) thermalization in the reflector region. The 1.07 factor, thus reflects the relative importance of each component.

In the forward axial direction, a similar combination of leakage and thermalization exists; however, the presence of an efficient absorber (boron) in the poison plate reduces the importance of the core leakage component. This enhances the fast neutron thermalization contribution. The ratio being less than one is consistent with that obtained for the fast and intermediate neutron fluxes.

The maximum photon dose rate, as well as fast ($E > 1.0$ MeV) and intermediate ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) neutron flux occur, of course, at the center of the core. These values are, respectively, $2.5 \times 10^{10} \text{ Rad-C/Hr}$ and 4.8×10^{15} and $2.4 \times 10^{16} \text{ n/cm}^2\text{-sec}$. The maximum thermal ($E < 0.4 \text{ eV}$) flux occurs in the beryllium reflector and is $5.9 \times 10^{14} \text{ n/cm}^2\text{-sec}$.

Following 60 ten minute (at rated temperature of 4250°R) operating cycles, the maximum environment (fluence and dose) in the Nuclear Subsystem with a composite fueled core are:

$$\begin{aligned}
 E > 1.0 \text{ MeV}, & 1.7 \times 10^{20} \text{ n/cm}^2 \\
 0.4 \text{ eV} < E < 1.0 \text{ MeV}, & 8.5 \times 10^{20} \text{ n/cm}^2 \\
 E < 0.4 \text{ eV}, & 2.2 \times 10^{19} \text{ n/cm}^2 \\
 \text{photon dose}, & 2.5 \times 10^{11} \text{ Rads-C}
 \end{aligned}$$

Uncertainties applicable to the internal radiation environment data were developed from the latest available experimental data. These uncertainties depend on the type of radiation and the location of interest within the NSS. Typical factors in the R-1 composite fueled core are: 1.03 for fast and intermediate neutron flux, 1.35 for thermal neutron flux and 1.14 for the photon dose rate. Uncertainties for the R-1 graphite fueled core are about 20 percent lower.

It is important to note that at the dome end of the NSS, about 22 percent of the total neutron flux is due to neutrons with energies greater than 1 MeV, whereas at the side surface of the NSS only 9 percent of the total flux is due to neutrons with energies greater than 1 MeV. The "harder" spectrum at the dome end is due to the Internal Shield. For a similar reason, 56 percent of the total photon energy flux at the dome end is due to photons with energies greater than 3 MeV, while this value is only 40 percent at the side of the subsystem.

2.2.3 Radiation Heating Rates

The total heat deposition for both a composite and graphite fueled core are presented in tabular form in Section 2.5 for each material in each region in the radiation heating rate models. Heating rate distributions are also presented graphically.

In the case of the composite fueled core, the total heat deposited in a few key regions is as follows:

Core Periphery	- 5.14 Mw
Lateral Support	- 3.66 Mw
Aluminum Structure	- 3.70 Mw
Beryllium Reflector	- 12.3 Mw
Internal Shield	- 1.65 Mw
Support Plate and Plena	- 3.6 Mw

Of the total heat deposited in the beryllium reflector, 3.48 Mw are deposited in the control vane at a 50° drum angle. This is an experimentally adjusted value, i.e., the calculated value decreased by a factor of 1.5.

The average heat deposition in either the composite or graphite fueled core is approximately 45 Mw/ft^3 . The heating rate in the unfueled element ranges from $1.37 \times 10^4 \text{ Btu/hr-in}^3$ at the core center to 5.9×10^3 at the core edge.

The maximum heating rate in the beryllium material is $2.39 \times 10^3 \text{ Btu/hr-in}^3$ for the composite fueled case. The maximum heating rate in the BATH material is $4.6 \times 10^3 \text{ Btu/hr-in}^3$ and in the aluminum in the support plate, $2 \times 10^3 \text{ Btu/hr-in}^3$.

A review of the heating rate data indicates that in general for regions outside the core:

- 1) the heating rate in materials primarily heated by photons (i.e., metals) decreases by 4 to 20 percent with the maximum decreases being 33 percent in the CHESH for a composite fueled core.
- 2) the heating rate in materials which have a significant amount of neutron heating (i.e., hydrogen, beryllium, and carbon) decreased by 5 to 10 percent with the maximum change being 18 percent again occurring in the CHESH for a composite fueled core, and
- 3) the heating rate in materials which have an (n, α) reaction display a mixed change of plus 4 percent to minus 6 percent for the composite fueled core.

Based on the latest available comparisons between calculated and experimental data, 1σ uncertainties applicable to the heating rates were developed. These uncertainties are dependent upon the type of heat deposition and upon the location within the subsystem. As an example, the regionwise uncertainties applicable to the kinetic neutron and photon heat deposition in the reflector are 1.13 and 1.06, respectively. These uncertainties apply to a composite fueled core. Uncertainties for a graphite fueled core are generally about

20 percent lower.

The maximum possible increase in the beryllium heating rate due to the presence of the Test Cell C 360° shield was observed to be about 20 percent. In the axial direction, the maximum increase occurs, of course, at the dome of the reactor; the maximum possible increase was about a factor of 3.

2.2.4 Radiation Sources

The total source in the composite fueled core is 9.04×10^{20} MeV/sec compared to 8.96×10^{20} in the graphite fueled core. The difference in the total source is noted to be insignificant. However, due to the zirconium in the composite core, the photon spectrum is much harder than in the case of the graphite core. For example, the photon source for $E > 7.0$ MeV is a factor of 2 higher for the composite fueled core.

The total source in the reflector region is approximately 1.9×10^{19} MeV/sec for either the composite or graphite fueled core. Less than 10 percent of this source is due to the beryllium material itself. The source is mainly due to capture gamma rays from the Fe, Cr and Ni in the hardware in the reflector.

The total source in the Support Plate and Plena is about 1.9×10^{19} MeV/sec for either the composite or graphite fueled core. The TZM moly orifices in this region account for about half of this total source. It is important to minimize secondary gamma production in the Support Plate and Plena regions since these regions contribute about 54 percent of the total photon dose rate on the centerline, 10 feet from the core center. In contrast, at this same location, the entire reactor core contributes only 28 percent of the total photon dose rate.

The total photon source in the Internal Shield is only 1.8×10^{18} MeV/sec, compared to 6.3×10^{19} MeV/sec for all the NSS regions, excluding the core. For the graphite fueled core, the total source in the shield is about 10 percent lower. About 70 percent of the total photon source in the shield results from the low energy (0.478 MeV) gamma accompanying the boron (n,α) reaction in boron carbide in the BATH material. Since the major source in the shield consists of low energy photons, the shield source contributes only about

10 percent to the total gamma ray dose rate on the reactor centerline at locations external to the NSS.

The total photon source in the lead portion of the internal shield is only 3.5×10^{15} MeV/sec. Eighty-five percent of this source results from the gamma production due to neutron inelastic scattering.

The contribution by source region to the total photon dose rate shows a slight shift in the case of the graphite fueled reactor for reasons stated previously. On the centerline, 10 feet from the core center the composite fueled core contributes 28 percent to the total photon dose rate, whereas the graphite fueled core contributes 37 percent.

It is important to minimize secondary gamma ray sources whenever possible for the following reason: If the photon radiation level on the centerline is reduced by only 25 percent, the Internal Shield weight can be reduced by roughly 200 pounds.

Estimates of uncertainties in the prompt, delayed, and secondary photon sources are presented in Section 5.

2.3 REQUIREMENTS

This section provides the NSS requirements which pertain to radiation data.

2.3.1 Specification Identification

The current governing specification for the Nuclear Subsystem (NSS) is Specification No. CP-677555A.

2.3.2 Applicable Requirements

The applicable requirements for the NSS which are either evaluated from a radiation standpoint or require radiation information to be evaluated are as follows:

"3.1.1.1.11. Nuclear Radiation Shielding Function

3.1.1.1.11.1. Unmanned Configuration - The radiation shielding requirements for the unmanned configuration of the NERVA engine shall be satisfied by NSS shielding internal to the pressure vessel which shall have the following characteristics:

(a) The Internal Shield shall limit the radiation leakage through a plane 63 inches forward of the core center, perpendicular to the engine axis, to the levels given in Table 9.

(b) The internal shield shall limit the pressure vessel leakage radiation at critical components in the engine system to the levels given in Table 10.

3.1.1.1.11.2. Manned Configuration - The NSS shielding requirements are identical to those specified in 3.1.1.1.11.1 (Unmanned Configuration).

3.1.2.1 Reliability. - (tbd)

3.1.2.4.2 Induced Environment

3.1.2.4.2.1 Nuclear Environment - The NSS components external to the pressure vessel shall be capable of withstanding the induced radiation environment specified in Table 12 (tbd) and Figures 3 (tbd) and 4 (tbd). NSS components internal to the pressure vessel shall be capable of withstanding the radiation environment which is attendant to the operation of the NSS over its Operating Service Life. The NSS shall be capable of functioning with or without an external shield.

3.1.2.4.2.3. Thermal Environment - The NSS shall be capable of withstanding the thermal environment which is attendant to the operation of the NSS over its Operating Service Life as specified in the Performance Characteristics section.

3.3.1.3. Nuclear Criteria

3.3.1.3.1. Internal Shield Design - The shield design shall be based on the following: The shield leakage shall be calculated on the basis of nominal formulae. For preliminary design, leakage values shall then be increased by one sigma analytical uncertainty before comparison with the limits specified in Tables 9 and 10.

The Internal Shield shall not exceed 18 inches in thickness including structural support and cooling regions.

3.3.3.1.2. Radiation - Materials shall be selected and qualified by appropriate tests to be compatible with the radiation environment specified in 3.1.2.4 (Environments). "

2.4 METHOD OF ANALYSIS – DESIGN PRACTICES

This section describes the analytical methods and models employed in the radiation analysis of the overall NSS. Also included in this section is a discussion of the status/maturity of the data. The validity of the radiation analysis is described and uncertainty factors applicable to the calculated radiation data are developed.

2.4.1 Analytical Procedures

The purpose of this section is to describe: 1) the overall framework of nuclear, radiation, thermal, and structural analyses, 2) the relationship of radiation and shielding computations to this overall framework, and 3) the details of the radiation and shielding procedures.

2.4.1.1 Overall Design Engineering Procedures

Figure 2-2 displays the flowchart of nuclear, thermal and structural analyses. In this figure, and in all subsequent figures describing analytical procedures, the following symbolism is employed:

- 1) An inverted trapezoid (shorter baseline on the bottom) denotes an input of information.
- 2) An upright trapezoid (shorter baseline on the top) denotes an output of information.
- 3) A rectangular block represents an operation.
- 4) All functions have a number in the upper left-hand corner which identifies the function on any subsequent flowchart.
- 5) When the operations performed for a given function are expanded into several subfunctions, the base identification number will remain the same and suffix numbers identifying the subfunctions will be added (i. e., the subfunctions for block 11 will be numbered 11-1, 11-2, etc.)
- 6) When a specific function is performed by an existing operational computer code, the name of the computer code will appear at the bottom of the function block.

The procedure for performing nuclear, radiation, thermal, and structural analyses has been broken down into four steps which are as follows:

- Step 1. The first step consists of developing or obtaining the data necessary to define the engineering model of the NSS.
- Step 2. The second step consists of translating the information in the engineering model of the NSS into the analytical models required to perform nuclear, thermal, and structural analyses.
- Step 3. The third step consists of employing the analytical model, developed in Step 2, to obtain nuclear, thermal and structural results for the NSS.
- Step 4. The fourth step consists of: 1) employing the detailed component models developed in Step 2 and the boundary conditions from the NSS calculational results developed in Step 3 to accomplish individual component thermal and structural analyses, and 2) reporting the results of nuclear, radiation, thermal, and structural analyses. Note that the nuclear analysis is performed only on the NSS as a whole and no component nuclear analysis is performed. The fourth step also briefly indicates the Design Engineering validation approach which includes: 1) comparing analytical and experimental results to obtain uncertainties in the calculational technique, and 2) employing experimental or other reference information to validate/improve the analytical model or technique.

Before a detailed description of the four steps followed in the analytic process, it is important to describe the incorporation of uncertainties and propagation of error methods throughout the analysis chain. These items, which constitute the essence of the reliability-related calculations, must begin with the input information of Step 1 being described probabilistically, namely, that drawings be a source not only of nominal dimensions but also of geometric uncertainties, that the materials properties be expressed as means, standard deviations, and (where available) test sample size, and that engine component simulation in the NSS performance models be described by nominal characteristics as well as uncertainties. The NSS level analysis, Step 3, is performed probabilistically in that the nuclear analytic results include uncertainty estimates, while the thermal and structural analyses are performed in random trial, or Monte Carlo, fashion. The thermal and structural NSS computer

representations are equipped with the necessary programming logic to select random values of all input parameters, perform a selected number of random trial (Monte Carlo runs), and reduce the resulting output data in terms of mean values and standard deviations. At present, 30 Monte Carlo cases for a given operating condition have been selected as a reasonable compromise between available resources and approach to an exact solution. In current practice, the Monte Carlo analyses are restricted to steady state conditions. The Step 4 component analyses must depend, of course, on the geometric and property information of Step 1 and boundary conditions as generated by the Step 4 gross NSS analysis. The probabilistic aspects of Step 1 and 4 as just described provide the raw material for probabilistic analysis at the Step 4 level. At this level of detailed component analysis, no fixed method of propagation of error analysis is followed.

If Monte Carlo techniques were applied to every component finite difference or element model, the required computer time would far exceed available program resources. The propagation of error analysis at the component level is performed by some scheme created by the ingenuity of the engineering analyst and best suited to the individual component analysis. Some suggested methods are available in NRPs 403, 411, and 605. The final result of probabilistic component analysis is, of course, a statistical description of component failure-governing "stress," stress in a broad sense which could actually be temperature, predicted reactivity loss, or structural load. The failure-governing stress is combined with the "strength," or the component ability to sustain the particular form of "stress," through the stress interference method as described in NRP 307. The interference analysis yields the probability of occurrence of a given component failure mechanism. The failure mechanism probabilities are indicated at the bottom of Figure 2-2 as a unique output of the probabilistic design process, distinct from the radiation environment, heating distribution, temperatures, pressures, stresses, and displacement which result from a conventional reactor analysis.

A brief description of each step follows.

From Figure 2-2, it is seen that Step 1 consists of the following inputs of information:

- 1) Block 1 is NSS and/or component drawings.
- 2) Block 2 is the bill of materials defining the type, quantity, and composition of individual parts.
- 3) Block 3 is material properties information which includes the nominal value and standard deviation for each property.
- 4) Block 4 is the run profile, state points, and engine component data information.
- 5) Block 5 consists of optional input data for malfunction and emergency mode which can be input into the analysis process in another reliability-related task, namely, prediction of the effects of engine component malfunctions.

The information from Blocks 1 through 5 is combined together to generate the engineering model of the NSS in Block 7. A Design Engineering goal is automation of this entire procedure. A computer description of each component and the bill of materials would be handled by the Weights And Moments of Intertia (WAMI) code to generate the weights, center of gravity, and moments of inertia for each component (Block 6). The WAMI code would also act as the file manager for the majority of the information in the engineering model and would provide the information needed to generate analytical models. The Nuclear, Thermal Stress (NUTS) code handles the material properties information and provides the information in a format which is directly useable by the design computation codes. Codes, such as the Design Performance Reliability (DPR) code, are being developed which will automatically generate probability of failure mechanism information via the Monte Carlo technique to evaluate propagation of errors (NRP 605) for the NSS as a whole. The DPR code is being developed as a specific tool to predict core assembly reliability.

A vast quantity of information is obtained from the drawings. The process of translating drawing information to computer input is being investigated. This process can be automated via the use of a digitizer. This machine takes the x and y coordinates of a point and/or any grouping of symbols and produces computer input from this information on punched cards (a magnetic tape is also available). Therefore, an operator using a digitizer, essentially traces a drawing to produce a computer description of the component and any other information

shown on the drawing. The computer description can then be used directly in calculations with WAMI to generate analytical models.

Step 2 in the nuclear, thermal, and structural analyses consists of developing the analytical models required by each discipline (Blocks 8, 9, and 10) from the information contained in the common engineering data bank (Block 7). This procedure may also be automated with the Nuclear Analytical Model (NAM), Thermal Analytical Model (TAM), and Stress Analytical Model (SAM) codes in order to make the optimum utilization of manpower and avoid duplication of effort. The NAM, TAM, and SAM codes are currently only names which may apply to several computer codes, some of which are already in existence, used to generate an analytical mode. The LOBO and MESHER codes that generate spatial mesh patterns for nuclear and thermal calculations have been programmed. The codes to translate data from WAMI into weights and atom densities for nuclear and thermal calculations have been formulated.

Step 3 in the nuclear, thermal, and structural analyses consists of performing the analysis related to the NSS as a whole. The nuclear analytical model developed in Block 8 is employed to perform the nuclear and radiation calculations in Block 11 (the details of the radiation portion of the nuclear computations will be described below). In addition, in order to perform the nuclear computations, the region average temperatures and hydrogen densities are required (Block 12) from the thermal analyses. However, note that the thermal analyses, both for the NSS and components (Blocks 13 and 19), require heating rates, fission distribution, and reactivity coefficients in order to be performed (Block 15). Therefore, as can be seen from Figure 2-2, the nuclear and thermal computations form a closed loop which require some iteration in order to obtain nuclear and thermal data.

The thermal calculations for the NSS (Block 13) employ the thermal analytical model (Block 9) and nuclear information (Block 15) to generate thermal data for the NSS and thermal-flow boundary conditions (Block 16) which are used in the thermal calculations for components (Block 19) and in the stress calculations for both the NSS (Block 14) and components (Block 21). The stress calculations for the NSS employ the stress analytical model (Block 10) and the thermal-flow boundary conditions (Block 16) to generate the stress data for

the NSS and the loads and displacement boundary conditions (Block 17) which are used in the stress calculations for components (Block 21).

Step 4 in the nuclear, radiation, thermal, and structural analyses consists of the following three phases:

- 1) Thermal and stress calculations are performed on a component level (Note that nuclear and radiation calculations are not performed on a component level).
- 2) Analytical and experimental results are compared to derive uncertainties or experimental results or other reference information are employed to improve the analytical model or technique (Block 18). This function provides feedback to the second step of this procedure.
- 3) Nuclear, radiation, thermal and stress computations are reported in the appropriate data item. In addition, the results of malfunction and emergency mode condition calculations to evaluate FMECAs and failure mode probabilities with the aid of interference theory are reported. These results are obtained by Monte Carlo analyses which employ essentially the same calculational techniques used in analyses at design conditions.

The thermal calculations for component analysis (Block 19) employ the thermal analytical model (Block 19), the nuclear information (Block 15), and the thermal-flow boundary conditions from the NSS computations (Block 16) to generate thermal data for the components and the thermal boundary conditions (Block 20) for the components. The stress calculations for component analysis (Block 21) employ the stress analytical model (Block 10), the thermal-flow boundary conditions from the NSS and component analyses (Blocks 16 and 20), and the loads and displacement boundary conditions from the NSS computations (Block 17) to generate the stress data for the components. Radiation data are also provided to Thermal and Structural Analysis to modify radiation sensitive property data in the analyses.

2.4.1.2 Nuclear and Radiation and Shielding Design Procedure

Figure 2-3 displays the general outline of the procedure for performing nuclear and radiation design analysis for the NSS (Block 11). The atom densities and problem spatial mesh description, which constitute the nuclear analytical model (Block 8), and the

temperatures and hydrogen densities (Block 12) are employed to generate neutron spectra and neutron and photon cross sections (Block 11-1). These cross sections are employed to generate neutron and/or photon fluxes over the NSS (Block 11-2). This information is then processed to obtain:

- 1) the reactivity analysis for the NSS (Block 11-3) which includes determining the absolute criticality of the system and reactivity shimming,
- 2) the nuclear kinetic response for the NSS (Block 11-4) which consists of determining cold-to-hot reactivity changes,
- 3) the control drum worth (Block 11-5),
- 4) the power map and fuel zoning (Block 11-6),
- 5) the internal radiation environment (Block 11-7),
- 6) the radiation heating rates (Block 11-8),
- 7) the radiation sources (Block 11-9), and,
- 8) the external radiation environment (Block 11-10).

The information from Blocks 11-3, 11-4, and 11-5 and the results from Blocks 11-6, 11-7, and 11-8 which are processed by additional computer codes form the heating rate, fission distribution, and reactivity coefficient (Block 15) data required to perform thermal analyses. The data in Blocks 11-3 through 11-10 are reported in the appropriate data item (Block 22) and are compared with experimental results to obtain uncertainties in the analytical results and experimental results are used to improve the analytical technique (Block 17).

2.4.1.3 Radiation and Shielding Design Procedure

The WANL Integrated Nuclear, Radiation and Shielding Standard Design Method⁽²⁾ (WISDM), was employed to obtain the analytical results presented in this Data Item. The following sections describe the details of the radiation analyses and the computer codes employed. The description of the radiation analyses has been separated into the six subsections for ease of presentation, understanding, and discussion. The six subsections will present, respectively, the analytical procedures for determining the following information:

- 1) the radiation environment, radiation heating rates, and photon sources internal to the NSS,
- 2) the radiation environment external to the NSS,
- 3) the effect of radiation streaming on the external radiation environment at the Engine Specification locations,
- 4) the estimate of the effect of "Test Cell C" on the environment in the NSS, and,
- 5) unshielded environment.

Internal Radiation Environment, Sources and Heating Rates

The basic analytical procedure employed to calculate the neutron and photon radiation environment, sources, and heating rates internal to the NSS is shown schematically in Figure 2-4. In addition to depicting the flow of information throughout the calculation, Figure 2-4 illustrates the type of information derived from each computation as well as the sequence of calculations. The encircled sequence numbers shown in Figure 2-4 are subsequently employed to facilitate discussion of the interconnection between various phases of the internal calculation and the external calculation.

The entire operational procedure shown in Figure 2-4, which is fully automated via either magnetic tape and/or punched card input and/or output data, corresponds to data blocks 11-7, 11-8, and 11-9 in Figure 2-3.

As shown in Figure 2-4 as calculational sequence 1, atom densities⁽³⁾, neutron spectra, microscopic neutron and photon cross sections, and photon production library data were input to the APPROPOS code.⁽⁴⁾ The 52 group (with 51 downscatters) neutron cross sections input to the APPROPOS code were generated via the GAMBIT code,⁽⁵⁾ while 13 group photon cross sections were generated via the GAMLEG⁽⁴⁾ code. The 52 group spectra input to the APPROPOS code were generated via appropriate one dimensional ANISN⁽⁶⁾ transport calculations.

The APPROPOS code generates several types of data as follows: 1) photon production data for input to the NAGS code,⁽⁷⁾ 2) 16 group, region dependent, macroscopic, P_0 transport corrected neutron cross sections (with upscatter removed) for input to the DOT code,⁽⁸⁾ and, 3) 13 group, P_1 , region dependent, photon cross sections for use in DOT.

In order to accurately analyze the reactor, approximately 8500 spatial mesh cells are required for a typical neutron problem and 5200 for a typical photon problem. In order to fit this computation onto the CDC6600 computer, with 128K core memory locations, the reactor was modeled into two problems. The first problem, which is designated as basic reactor geometry on Figure 2-4, consisted of those regions from the aft pressure vessel flange to the support plate while the second problem, which is designated as shield geometry, consisted of those regions from the support plate to the forward end of the NSS. (These models are discussed in Section 2.4.2).

A total of five DOT problems are needed to obtain the neutron and photon fluxes throughout the NSS at the end of 60 minutes of full power operation. These calculations are as follows:

- 1) a fixed source neutron calculation to obtain the neutron fluxes over the basic reactor geometry (calculational sequence 2); a radially flat fission distribution in the core with a cosine axial variation represented the fixed neutron source,

- 2) a boundary source neutron calculation to transport the fluxes from the boundary at the support plate through the shield geometry (calculational sequence 3),
- 3) a fixed source photon calculation to obtain the photon fluxes over the basic reactor geometry (calculational sequence 6),
- 4) a boundary source photon calculation to transport the photons originating in the basic reactor geometry from the boundary at the support plate through the shield geometry (calculational sequence 7), and,
- 5) a fixed source photon calculation to obtain the photon fluxes over the shield geometry due to photons originating in the shield regions and regions above the shield (calculational sequence 8).

An S_8 angular quadrature was employed in all the DOT problems.

The fixed source for the basic reactor photon DOT problem was generated by the NAGS code (calculational sequence 4).

Neutron and photon scalar fluxes from the DOT basic reactor geometry problems are input to the NAGS code (calculational sequence 4) which calculates sources for use in the CRAM (Common Radiation Analysis Model) and heating rates. Heating rate distributions in each NAGS region are automatically plotted by the NAGS code (with the exception of the control vane heating which is discussed later in this section).

The neutron and photon scalar fluxes from the three DOT shield geometry problems are input to an additional NAGS problem (Calculational sequence 5) which calculates both the sources for the CRAM and the heating rates throughout that portion of the NSS.

The tedious and time consuming process of obtaining radiation environment data (neutron fluxes, dose rates, etc.) from the DOT output has also been automated by means of the CHEAPER code⁽⁹⁾ (See data block 11-7 of Figure 2-3). Scalar fluxes from the DOT

problem are input via tape to the CHEAPER code. The CHEAPER code interpolates and/or extrapolates the DOT data at region boundaries, or at any arbitrary set of R,Z coordinates within the problem geometry to provide the environment at specific locations of interest. This code also plots the data as an option. In this manner, the environment traverses given in Sections 2.5.1.1.1 and 2.5.2.1.1 were automatically plotted.

Radiation data at selected locations from the CHEAPER code is also automatically processed through the SNARE code⁽¹⁰⁾ to provide transient environment data given in Sections 2.5.1.2 and 2.5.2.2.

Thermal analysis also requires as input data, the heating rate due to delayed gamma rays as a separate quantity as presented in Section 2.5.1 and 2.5.2. To calculate these data, "delayed gamma" photon DOT problems are also run, but are not schematically shown in Figure 2-4.

A notable exception to the R,Z DOT heating rate procedure shown in Figure 2-4 is the calculation of the radiation heating in the control vanes. The heating rate in the control vanes is due to three nuclear reactions which are as follows: 1) the photon heating of the control vane material, 2) the neutron kinetic heating of the control vane material, and 3) the (n, α) reaction of B^{10} in the control vanes which liberates 2.311 MeV per neutron capture event as an alpha particle and 0.478 MeV per neutron capture event as a gamma ray. The energy from the alpha particle resulting from the (n, α) reaction is assumed to be deposited in the control vane material while the energy from the gamma ray associated with the (n, α) event is assumed to be deposited solely in the beryllium in the reflector.

The photon and neutron kinetic heating of the control vane material is based on the preceding heating rate analysis with the DOT code by assuming that the heating rate due to neutrons and photon on a per unit mass basis is the same in the CuB^{10} material as in A-286. The heat due to the (n, α) reaction is based on R- θ geometry, diffusion theory calculations with the EXTERMINATOR-II code⁽¹¹⁾ which explicitly represents the control vanes positioned at the 50 degree position (See the S-101 Data Item for the details of the EXTERMINATOR calculation). The EXTERMINATOR code provides the total number of absorptions (neutron capture events) which occurs in the control vane. This data is then

converted to the amount of heat generated by the (n, α) reaction. However, the results of the control vane heating rate experiment (Reference 12), clearly and consistently indicated that the calculated heating rate data for the control vane was too high by a factor of 1.49. It was determined that this occurred because effective neutron cross sections which accounted for the flux depression through the control vane were not employed in the calculation. Therefore, because the experimental results indicated a consistent difference and because a deficiency in the calculation technique was discovered, the heat generated in the control vanes by the (n, α) reaction was decreased by a factor of 1.49. However, it must be noted and emphasized that none of the calculated heating rates in any other material have been adjusted because the experimental data does not display a consistently clear difference between calculated and experimental results nor has any deficiency in the calculational technique been discovered.

The azimuthal distribution of the heating rate over the vane is obtained from the EXTERMINATOR results by processing the calculated neutron fluxes on the inner and outer surface of the control vane through the ARCE code (Reference 13). The ARCE code determines the B^{10} reaction rate at each nodal point on the surface of the control vane by the following equation:

$$B^{10} \text{ Reaction Rate, } F(\alpha) = \sum_{i=1}^{16} \sum_a^i (B^{10}) \Phi^i \quad \text{where,}$$

$$\sum_a^i (B^{10}) = \text{the } i^{\text{th}} \text{ neutron energy group absorption cross section for } B^{10}$$

$$\Phi^i = \text{the } i^{\text{th}} \text{ neutron energy group flux at a nodal point on the surface of the control vane}$$

$$\alpha = \text{the azimuthal position on the control vane.}$$

The B^{10} reaction rate data is then numerically integrated and normalized such that

$$\frac{\int_0^{120^\circ} F(\alpha) d\alpha}{\int_0^{120^\circ} d\alpha} = 1.0$$

External Radiation Environment

Two techniques were employed to calculate the radiation environment external to the nuclear subsystem as defined by data block 11-10 in Figure 2-3. The first technique used the MAP code,⁽¹⁴⁾ while the second approach employed the KAP-V code.⁽¹⁵⁾ These techniques are schematically illustrated in Figure 2-5.

The MAP code, which was employed to perform the analysis of the neutron external environment, performs a ray tracing calculation from the surface of the NSS, across a void, to a detector location. The source of the ray trace is the neutron angular flux leakage at the surface of the NSS as obtained from the preceding DOT transport calculations (entry points 2 and 3 on Figure 2-5)

In addition to the discrete ordinates method, the point kernel approach was also applied to the analysis of the neutron and photon radiation levels external to the NSS. Neutron and photon source terms for use in the point kernel code were obtained directly from the NAGS source calculations performed during the course of the two dimensional internal analysis. These particular NAGS calculations are referenced as sequence numbers 4 and 5 on Figure 2-5. In the case of neutrons, NAGS supplied relative radial and axial fission distributions as well as the total number of fissions in the reactor core. For the photon analysis, NAGS generates the relative radial and axial distributions and the total photon source strength in all regions of the NSS.

The point kernel neutron analysis was carried out using the infinite media moments method data with carbon as the reference material. A bivariate polynomial expression was employed to calculate the differential neutron spectra based on infinite media carbon data. The bivariate expression was solved in terms of initial neutron energy and of material penetration.

The photon analysis was carried out using the standard 13 energy group scheme that was employed in the discrete ordinates analysis. Material attenuation coefficients for each element present in the problem were generated internal to the KAP-V code. Infinite media water dose buildup factors were applied to all of the photon dose rate calculations. As in the case of the attenuation coefficients, the infinite media buildup factors were generated internal to the KAP-V code.

However, the KAP-V code does not have the capability to calculate thermal ($E < 0.4$ eV) and intermediate ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) energy neutron fluxes. Therefore, in order to overcome this deficiency, the following technique was employed to obtain the external radiation environment:

- 1) The MAP code was employed to determine thermal, intermediate, and fast energy neutron fluxes at the detector point of interest (calculational sequence 10).
- 2) From the MAP data, the ratio of the thermal to fast neutron flux and the ratio of the intermediate to fast neutron flux was determined at the locations of interest.
- 3) These ratios obtained from the MAP data were then applied to the KAP calculation of the fast neutron flux in order to obtain the thermal and intermediate neutron flux at the positions of interest (calculational sequence 11).

The DOT-MAP calculational techniques is not strictly employed herein to provide all external radiation environment because at NSS PDR, the internal shield is to be designed for a Composite fueled core. Due to the numerous calculations required to adequately analyze this reactor-shield geometry, the application of this technique to the shield design is prohibitive at this time for schedular reasons. The shield was "sized" based on the point kernel technique; the more sophisticated S_n transport technique was employed to "verify" the point kernel calculations.

Radiation Streaming Through Shield Penetrations

In order to determine if the internal shield meets specifications, it is necessary to develop factors which when applied to nominal calculations account for streaming through shield penetrations. It should be remembered that the nominal calculations are based on analytical models in which the shield is represented as a homogeneous mixture of the shield materials and the shield penetrations.

The locations of the Internal Shield specifications are limited to short distances from the reactor. The largest distance from the center of the core is less than 9 feet, i.e., at the TPA location. Hence, radiation "streaming" at these short distances is not as significant a problem as at large distances from the reactor.

Essentially, there are two types of shield penetrations in the R-1 Internal Shield reference design. The main coolant channels are relatively small (.188 inches in diameter) whereas the peripheral piping penetrations have a diameter of 0.75 inches. Consequently, two factors for gamma ray streaming and two factors for neutron streaming are required.

The analysis performed to determine the effect of streaming through coolant channels as well as through peripheral piping penetrations in the internal shield was based on the following equation:⁽¹⁶⁾

$$R = \frac{\left\{\frac{a}{z}\right\}^2 + 2 \int_{r_o}^{\infty} \frac{z}{r^2} e^{-[u \ell(r)]} dr}{2 uz \int_{uz}^{\infty} \frac{e^{-t}}{t^2} dt}$$

where:

a = hole radius

z = shield thickness

uz = no. of mean free paths in shield

$$r_o = \sqrt{a^2 + z^2}$$

$$\ell(r) = r \left[1 - \sqrt{\frac{a}{r^2 - z^2}} \right]$$

The parameter, R , in the above equation is the ratio of photon or neutron flux with the hole to that calculated for a homogeneous shield with no hole present. As stated above, the equation was evaluated for both neutrons and photons for coolant channels and for peripheral piping penetrations. The results are as follows:

LOCATION	PHOTONS	NEUTRONS
FORWARD OF SHIELD $0 < R < 68 \text{ CM}$ (EXCEPT IN LINE WITH SUPPORT STEM COOLANT LINES)	$R = 1.2$	$R = 1.5$
FORWARD OF SHIELD IN LINE WITH SUPPORT STEM COOLANT LINES	$R = 1.5$	$R = 2.0$

Estimate of the Effect of "Test Cell C" Environment

Preliminary estimates of the effect of "Test Cell C" on radiation environment internal to the NSS was determined. Since the Test Cell C facility shields for R-1 testing have not been designed, the NRX-A6 360° facility shield and privy roof shield designs⁽¹⁷⁾ were employed herein.

To determine an upper limit estimate of the effect of the facility shields on the NSS environment, radial and axial one-dimensional ANISN transport problems were run with and without the facility shields. In this preliminary analysis the facility shielding was placed directly tangent to the side and/or top of the pressure vessel.

Results of this analysis are presented as relative data in Section 2.5 of this chapter.

The NSS-TCC geometry is difficult to handle for nuclear and radiation analysis with available computer codes. Realizing the oversimplification of the physical situation

when analysis is performed with 1-D geometry, it was deemed necessary to obtain an estimate of the degree of validity of the upper limit calculation that is expected with the procedure outlined above. Since there was not any experimental data available for comparison with the calculations, the experimental data obtained from the NRX-A6 test series was used to obtain estimates of the effect of TCC on the radiation environment at the forward pressure vessel dome closure. Engineering calculations were performed to ascertain the validity of the upper limit estimate obtained from the transport calculations.

The engineering calculations were performed for data points on the surface of the forward pressure vessel dome closure and the effect of Test Cell C at this location was assumed to be independent of radius. The data presented are heating rate (watts/gm) of stainless steel and gamma Kerma rate (Rad-C/hr).⁽¹⁸⁾ These data were obtained in the following manner using NRX-A6 test data in Test Cell C⁽¹⁹⁾ and DOT calculated⁽²⁰⁾ environment without TCC being present:

1. It was assumed that the full power DOT calculations of the gamma heating rate at the forward dome closure of A6 was errorless. Thus, it represented the heating rate due to the NSS only (called direct).
2. The total heating rate due to the direct plus TCC induced at the forward dome closure for full power operation was obtained from low power TLD measurements made on the dome closure and privy roof plus calorimeter measurements made at full power on the privy roof at an adjacent location to the TLD. No calorimeter measurements were made on the forward dome closure to obtain a direct result. The total full power heating rate at the dome closure surface due to the direct plus TCC induced heating was obtained by multiplying the full power calorimeter heating rate on the privy roof by the ratio of the TLD determined dose rate at the dome closure to that at the location on the privy roof adjacent to the calorimeter.

3. The TCC induced heating rate on the NRX-A6 NSS was obtained by subtracting the direct quantity obtained by the procedures outlined in item 1 (DOT calculations) from the total obtained by the procedure outlined in item 2 (calorimeter and TLD).
4. From the use of gamma albedos, it was deduced that the TCC heating was a result of secondary gamma production in the TCC material and not due to gamma scatter.
5. The secondary gamma production in TCC due to the presence of an unshielded R-1 in TCC rather than an NRX-A6 reactor was assumed to be proportional to the ratio of the total neutron leakage from the forward dome closure of R-1 to that from the NRX-A6. The total neutron leakage from the unshielded R-1 was obtained from the KAP-V (530 05/25/70) and 1-D axial ANISN (332 03/05/70) calculations. The total neutron leakage from A6 was obtained from Reference 20. Thus, the TCC induced heating rate due to TCC for an unshielded R-1 was obtained by multiplying the heating rate obtained in item 3 by the ratio of the total neutron leakage from NRX-A6. As an estimate, this value is also applied to a shielded R-1 reactor.

Data obtained in the above manner are presented in Chapters 9.0, 10.0 and 11.0 for the following specified ECC: 1) NSSI at the CDDA position indication, and 2) CDDA at the actuator.

Unshielded Analysis

Data for the unshielded core are presented for the following ECCs at the specified locations:

NSSI ECC:

1. CDDA position indication.
2. SSCV position indication

CDDA ECC:

1. Actuator
2. Controller
3. A-MOD

SSCA ECC:

1. SSCV actuators.

The procedure used to obtain all of the unshielded data, with the exception of the A-MOD location, was the following:

1. The BATH, lead, and forward and aft aluminum end plates were removed from the NSS geometry. These material regions were replaced by a "void" region having the same geometric boundaries. This geometry is referred to as the "unshielded NSS."
2. The secondary photon sources associated with these materials were deleted in the photon calculations.
3. By virtue of preserving the forward pressure vessel dome closure and reactor geometry, the position of the actuator with respect to the core center remained unchanged when the internal shield was removed.
4. The KAP-V point kernel code was used to calculate the gamma dose rate, photon heating in stainless steel and fast neutron flux ($E > 1.0$ MeV) at the desired locations.
5. To obtain the thermal and intermediate neutron fluxes, one-dimensional axial ANISN transport calculations were performed on the unshielded geometry to obtain ratios of thermal flux to fast flux and intermediate flux to fast flux at the forward pressure vessel dome closure surface. These ratios were multiplied times the KAP-V calculated fast neutron flux at the desired locations to obtain the thermal and intermediate neutron flux values.
6. As stated in item 3, the calculations were performed with the actuator location being in the same location as the shielded geometry. The zero shield concept (Dwg. 940J420), however, shows that the dome is moved 13 inches toward the core when the shield is removed. The data presented in Reference 21 were used to obtain the increase

in radiation level when the actuator is moved 13 inches toward the core. The geometry factor obtained in this manner was 1.6.

The unshielded data at the A-MOD location for both the graphite and composite fueled cores were assumed to be the same as that obtained for the shielded calculations. This is reasonable since the data point for the A-MOD (Station - 3.6) is below the shield and the removal of the shield is not expected to significantly perturb the radiation environment at this location.

2.4.2 Analytical Models

Several different models are required in order to perform the radiation analysis of the NSS and to present the results of the analysis in a logical manner. Specifically these models are: 1) the radiation environment drawing, 2) the R,Z calculational models employed in the two dimensional discrete ordinates transport analysis, 3) models for presentation of heating rate data, and 4) the Common Radiation Analysis Model (CRAM).

Each model will be discussed in detail in the following sections.

2.4.2.1 Radiation Environment Drawing

Figure 2-6 is a simplified layout drawing of the Nuclear Subsystem defined by WANL Dwg. 939J740. This is the basic model which is the starting point of the Radiation and Shielding Analysis, along with the detailed ECC drawings. To orient the reader, the various ECCs discussed throughout this report are indicated in Figure 2-6. Also shown on Figure 2-6 are specific selected locations at which radiation environment data are provided in Section 2-5. Table 2-3 presents a description of these locations, as well as, the coordinates of each point. The drawing defined by Figure 2-6 and data associated with this drawing are also supplied as input to the E 105 Data Item.

2.4.2.2 Two Dimensional Discrete Ordinates Transport Models

In order to provide the spatial mesh in the DOT code to accurately analyze the NSS and to minimize computer computational time, four DOT geometries were employed in

the computations. Two geometries applied to neutron calculations and two geometries applied to photon calculations. For both neutrons and photons, one geometry (the basic reactor geometry) represented the NSS from the aft pressure vessel flange to approximately 10 cm into the BATH portion of the internal shield. The second geometry (shield geometry) represented the NSS from the plane just forward of the support plate to the pressure vessel dome. The radial mesh for all four geometries is the same; whereas, a different axial mesh is employed in each of the geometries. These models are schematically presented in Figures 2-7 through 2-10. Figures 2-7 and 2-8 are, respectively, the basic reactor geometry and shield geometry for the neutron DOT problems. Figures 2-9 and 2-10 are similar models for the photon DOT problems. Mesh line locations and the DOT problem region numbers are indicated on each figure.

Table 2-4 gives the following data pertinent to each region in Figures 2-7 through 2-10:

- 1) the region description
- 2) the region boundaries in terms of radius and reactor station
- 3) the ECC assigned to each region in this report.

It is important to note the ECC assigned to each region as defined in Table 2-4 for use in the radiation analysis. Due to the nature of the radiation analysis, ECCs cannot be analyzed as separate entities, but rather the entire NSS must be analyzed. In some instances, actual mechanical components overlap into different regions in the radiation models. Therefore, in order to logically present data for each ECC, the assignments given in Table 2-4 by region were established and are used consistently throughout this report. For example, all data pertinent to the Internal Shield ECC defined by regions 35, 36, etc. (Table 2-4) are presented in the appropriate chapter, i.e., Chapter 7.

Tables 2-5 through 2-8 provide radial and axial mesh for the DOT models presented in Figures 2-7 through 2-10. The neutron problems employed a total of 8450 mesh, whereas the photon problems required only 5250 mesh.

The reader will note from Figure 2-4 that the DOT problems employ macroscopic region dependent cross section data. To obtain these region cross sections, it is necessary to define the material composition data for each region in the DOT model. Thus, the following detailed information is available in Reference 3:

- 1) the element composition data for all materials in the NSS,
- 2) the masses and element atom densities of the materials in the non-core regions, and
- 3) the partial element densities for each non-core region.

Masses and atom densities for the composite and graphite core regions were obtained from References 22 and 23.

2.4.2.3 Heating Rate Models

The heating rate models, presented schematically in Figures 2-11 and 2-12 for composite and graphite fueled cores, respectively, are essentially derivatives of the DOT models. The only significant difference in the geometry between the two heating rate models is the length of the fuel elements in the two cores (See regions 1-15 in Figures 2-11 and 2-12).

The region numbering system and the region description is the same as previously defined in Table 2-4.

Two sets of dimensions are shown in Figures 2-11 and 2-12: inches and centimeters. Heating rate distributions presented throughout this report are given in units of inches to be compatible with thermal analysis codes. The internal radiation environment traverses presented in Section 2.5 are given in units of centimeters since the CHEAPER code automatic plot routine employed these units.

The total heat deposition for each region defined in Figures 2-11 and 2-12 is presented in Section 2.5.

2.4.2.4 Common Radiation Analysis Model (CRAM)

Figure 2-13 provides the region description for the R-Z geometry CRAM model of the NSS with composite fuel in the core. The dimensions given for the region boundaries

are in centimeters. The axial reference plane ($Z=0.0$) is located at the pressure vessel-nozzle mating plane with the positive portion of the Z-axis orientated towards the forward end of the NSS. Table 2-9 provides a brief description for the 35 regions comprising this model and the region material densities (gm/cm^3) for each subregion of the NSS with composite fuel in the core defined by Figure 2-13. Figure 2-14 and Table 2-10 provide similar information for the NSS with graphite fuel in the core. It should be noted that the graphite fuel model employs 36 regions to describe this geometry. Region 36 refers to the unfueled tip region which is not present in the Composite fueled core.

The CRAM models are a simplified version of the DOT models.

2.4.2.5 Computer Program Traceability

To insure traceability of all computer programs, tapes, and problems employed in the radiation analysis reported herein, Reference 24 provides the following information:

- 1) Reference Problem Number
- 2) Computer Program Name
- 3) Problem Description
- 4) Input Tape Identification, and,
- 5) Output Tape Identification where applicable.

The major program tapes are also identified in DRM-52850.

The Reference Problem Numbers are utilized throughout this report to identify computer output data related to graphs and tables wherever possible. These identification numbers are located in the lower right-hand corner of the appropriate graph or table.

2.4.3 Status/Maturity of Input Data

As can be seen from Figure 2-4 and Sections 2.4.1 and 2.4.2, the primary input data needed to perform these computations consisted of the following information:

- 1) NSS description,
- 2) element atom densities and partial densities,
- 3) neutron spectra, and,

- 4) microscopic neutron transport, photon transport, and photon production cross sections.

The status/maturity of this input data is as follows:

- 1) The latest description of the NSS as defined by WANL Drawing 939J740 was employed in these analyses with a few exceptions which are noted in Status/Maturity of Input Data Section for each ECC. Subsequent evaluation of these slight differences indicated that these exceptions would not significantly affect the results presented in this report.
- 2) The element atom densities and partial densities are based on the component masses as determined from the NSS description which will be slightly altered by the above refinements. The determination of all component masses was performed twice by separate individuals to insure validity of the data.
- 3) The neutron spectra employed in these calculations was based on the calculated spectra for the composite and graphite cores in the Conceptual Reference Design.⁽²⁵⁾ The difference between the Conceptual and Preliminary Reference Designs has a very small effect on the calculated neutron spectra, and hence, a negligible effect on the calculated macroscopic neutron cross sections obtained by spectrally weighting and employed in this analysis for the Preliminary Reference Design.
- 4) The status/maturity of the microscopic cross section employed in this analysis is described in detail in Reference 2.

2.4.4 Validity of Analysis

The validity of the basic analytical procedures described in Section 2.4.1 and shown schematically as Block 18 in Figure 2-2 was verified by performing several experiments and comparing analytical and experimental results. From these comparisons, the uncertainties in the R-1 radiation environment and heating rates were obtained for the NSS. NERVA flight uncertainties were then developed from the R-1 NSS uncertainties and from engineering judgement gained via the NRX test series. The following sections provide:

- 1) a brief description of each experiment,
- 2) the conclusions reached,
- 3) the use of uncertainties,
- 4) the differences between calculated and experimental results, and
- 5) the uncertainties to be applied to the radiation environment and heating rates.

Before describing the comparisons between the calculated and experimental data and the resulting uncertainties in radiation data, it was necessary to define the relationship of an uncertainty factor, F , to the mean, $\bar{x}$, and to the standard deviation, σ , in the difference between calculated and experimental data. Historically, uncertainty factors have always been applied to radiation data; whereas, $\bar{x}$ and σ have always been the results of a statistical analysis. In comparing the calculated and experimental data, it was found that a normal distribution of the difference, Δ , between calculated and experimental results occurred only when $\Delta = \log_e (\text{Experimental data, } E) - \log_e (\text{Calculated data, } C) = \log_e [E/C]$. Since $\bar{x}$ and σ have meaning only for a normal distribution, this dictated that the above equation for Δ be used. With this background information, it is now possible to define the relationship between F , $\bar{x}$, and σ :

$\bar{x}$ = the mean or average value of Δ as defined above in the logarithmic plane

σ = the standard deviation for Δ as defined above in the logarithmic plane

$$F = e^{\bar{x} + n\sigma}$$

Therefore, when n equals zero, F represents the average ratio of experimental to calculated results. If n equals plus or minus one, then F represents the one standard deviation upper

or lower limit in the ratio of experimental to calculated results. With n equal to plus or minus one, the product of F and the calculated radiation data represents the one standard deviation upper or lower limit in the experimental data. Specifically, for reasons which will be discussed in Section 2.4.4.2, the uncertainty factor, F , for radiation data will be defined by the preceding equation with $\bar{x}$ set equal to zero (no average difference between calculated and experimental results) and n set equal to one; therefore, F represents in the linear plane the one standard deviation of the ratio of experimental to calculated data.

2.4.4.1 Comparison with Experimental Results

Four experiments were performed to ascertain the validity of the analytical techniques employed in the analyses of the NSS. These four experiments consisted of:

- 1) the "Nuclear Subsystem Shielding Provisions Study," which determined the radiation environment and heating rates internal and external to mockups of the internal shield,
- 2) the "Control Vane Heating Distribution Experiment," which determined the radiation environment and heating rates on the surface of the control vane,
- 3) the "Radiation Environment in Various Reactor Components," which determined the radiation environment and heating rates in the core, core periphery, and reflector assembly, and
- 4) the "Radiation Environment on CHESH Components," which determined the radiation environment and heating rates in the CHESH.

The following is a brief description and the conclusions reached regarding each experiment.

2.4.4.1.1 Nuclear Subsystem Shielding Provisions Study

Reference 26 provides a detailed comparison of the analytical and experimental results obtained in the Nuclear Subsystem Shielding Provisions Study. In this experiment neutron and photon radiation environment were measured internal and external to three BATH shield mockups with mass thicknesses of 69, 89, and 109 gm/cm², and a borated steel-liquid hydrogen shield mockup, placed at the aft end of the PAX-GOC reactor assembly. The WANL Integrated Nuclear, Radiation and Shielding Standard Design Method (WISDM) was employed to analyze these experiments.

Based on the comparisons between the measurements and calculations, the following conclusions were noted:

- 1) The ability of the WISDM technique, employing the two dimensional discrete ordinates code (DOT IIW), to predict neutron and photon radiation levels internal to and near mockups of the BATH shield was clearly demonstrated,
- 2) The capability and feasibility of coupling two successive transport problems at an arbitrary "boundary source" plane was demonstrated. This boundary source feature is extremely important in that a judiciously chosen boundary source may, for a given reactor, be applied to parametric shield studies in such a way that the analysis is facilitated and the overall cost greatly reduced,
- 3) Effects associated with the transport of particles through and around the corners of the shadow shields were observed. The resultant overattenuation was observed to a larger degree in the point kernel analytical results, but, also, to some extent in the discrete ordinates calculations.
- 4) The discrete ordinates technique produced anomalous results at large distances from the reactor on the centerline of the system due to low order quadrature (S_8). This behavior was intensified with increasing shield opacity.

5) Further studies employing both the discrete ordinates technique and the point kernel method are warranted. The discrete ordinates method should be examined in terms of higher order angular quadrature and an increased number of mesh cells employed in the preceding S_n transport calculations. Although the point kernel approach does have an inherent difficulty in handling short circuit paths through and around shadow shields, some of the observed differences between point kernel results and measurements may be resolved through an examination of the applicability of infinite media moments data or infinite media buildup factors to finite media shield systems.

6) In order to improve the comparisons between analytical results and experimental data external to mockups of the shadow shields, all avenues of approach toward accurate determination, as well as reduction of the background environment should be thoroughly explored, especially the use of detector collimators.

2.4.4.1.2 Control Vane Heating Distribution Experiment

Reference 12 provides a detailed comparison of the analytical and experimental results obtained in the Control Vane Heating Distribution Experiment. In this experiment, the R-1 test sector of the PAX-G2A reactor was used to obtain radiation measurements on the surfaces of a CuB (24 v/o ^{10}B) R-1 prototype control vane located at a drum angle of 115° . Based upon the measurements, the expected heating rate in an 18 control vane NERVA R-1 reactor with the vanes at 115° is 0.00198 ± 0.0002 megawatts per megawatt of reactor thermal power. The $^{10}\text{B}(n,\alpha)$ heating rate was analytically determined with the (R, θ) EXTERMINATOR-2 neutron diffusion code (See Section 2.4.1 for calculational details) using both infinitely dilute and effective absorption cross sections for the vane. The ratio of the calculated to experimentally derived $^{10}\text{B}(n,\alpha)$ heating was $1.17 (+0.16/-0.13)$ and $1.49 (+0.2/-0.17)$ for effective and dilute vane cross sections, respectively. The activation rates of ^{164}Dy and ^6Li on the inside and outside surfaces of the vane were compared with

*This number represents the one standard deviation in the experimental data.

(R,θ) EXTERMINATOR-2 calculations. The use of effective, rather than dilute, absorption cross sections for the vane material considerably enhanced the agreement of the calculations with the experimental activation rates.

2.4.4.1.3 Radiation Environment in Various Reactor Components

Reference 27 provides a detailed comparison of the analytical and experimental results obtained in the Radiation Environment in Various Reactor Components experiment. In this experiment the radiation environment was measured in the core and reflector of the PAX-G1A reactor assembly. The results obtained include neutron fluxes 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 detectors and from Bragg-Gray ion chambers were also obtained. The WISDM technique was employed to calculate the radiation environment. Based on comparisons between the measurements and calculations the following conclusions were made:

1) The average multiplicative factor to be applied to the calculated fast neutron flux ($E > 1.0$ MeV) 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, 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.

2) The average multiplicative factor to be applied to the calculated thermal neutron flux ($E < 0.4$ eV) 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, 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. It should be noted that for the reflector this uncertainty applies only to the azimuthally average thermal neutron flux.

*This number represents the average ratio of experimental to calculated data of $\bar{x}$ defined in Section 2.4.4.

**These numbers represent the one sigma lower and upper limit in the ratio of experimental to calculated data, $e^{\bar{x}-\sigma}$ or $e^{\bar{x}+\sigma}$ as defined in Section 2.4.4.

3) 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.

4) 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.

5) 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.

6) The statistical analysis indicated that there was no statistically significant difference between the shape of the calculated and the shape of experimental data.

2.4.4.1.4 Radiation Environment on CHESH Components

Reference 28 provides a detailed comparison of the analytical and experimental results in the Radiation Environment on CHESH Components. In this experiment, the PAX-GIA reactor was employed for measurements of the radiation environment in mockups of the R-1 Cluster Hot End Support Hardware (CHESH).

Based on the analysis and comparisons presented in WANL-TME-1915 the following conclusions were made:

1) The uncertainty factors obtained from this analysis which employs an explicit representation of the CHESH are applicable to design computations which employ a smeared representation of the CHESH.

2) The average uncertainty factor to be applied to the calculated fast neutron flux in the CHESH is 0.82. The one standard deviation (one sigma) variation in this factor is 0.81 to 0.84.

3) The average uncertainty factor at ambient conditions, to be applied to the calculated thermal neutron flux in the CHESH is 0.24. The one sigma variation in this factor is 0.17 to 0.33.

4) The average uncertainty factor, at ambient conditions, to be applied to the calculated neutron heating data in the CHESH is 0.62. The one sigma variation in this factor is 0.56 to 0.71.

5) The average uncertainty factor, at ambient conditions, to be applied to the calculated photon heating rate data in the CHESH is 0.83. The one sigma variation in this factor is 0.63 to 1.10.

2.4.4.2 Uncertainty Analysis

The uncertainty factors associated with the radiation environment and radiation heating rates are derived in this section in accordance with the procedures outlined in NRP-600 and NRP-605. A detailed summary of the Failure Modes Effects and Criticality Analysis (FMECA) for the NSS is given in Reference 29. The FMECA's are listed for each ECC or CEI which comprise a part of the NSS. These various failure modes are due to various mechanisms, such as, fatigue, creep, changes in material properties due to radiation, thermal and mechanical stresses, etc. These mechanisms are created by various environmental factors, such as, nuclear radiation, heating, vibration, thermal shock, etc.

Reliability predictions and assessments require the determination of probabilities for the various failure modes and mechanisms that have been identified. The uncertainty factors in the radiation environment and radiation heating rates represent a portion of the information needed to generate these probabilities. The various disciplines that analyze the performance of the NSS, such as Thermal and Stress Analysis, use these uncertainties in radiation heating, and neutron flux and fluence to determine the uncertainties in the reactor temperatures and changes in material properties due to radiation. These overall uncertainties are then used to assist in the determination of an overall probability of success of the components operating in the specified thermal and nuclear environment.

The following sections describe the statistical results obtained by comparing calculated and experimental data for the external and internal radiation environment. Following these two sections, the uncertainty factors for the radiation environment, heating rates, and sources are derived from this statistical comparison between the calculated and experimental data.

2.4.4.2.1 Statistical Results Obtained for the External Radiation Environment Data

One of the major purposes of the study presented in Reference26 was to determine uncertainty factors applicable to the calculated radiation environment external to the nuclear subsystem employing a typical NERVA internal BATH shield. The comparisons gained through this study were not sufficiently comprehensive to develop such factors to a great degree of precision; however, it is possible to employ the comparisons so as to estimate the possible error margins in radiation levels calculated above the shielded end of the NSS.

The current reactor design includes approximately 175 gm/cm^2 of material forward of the reactor core. Of this mass thickness, about 110 gm/cm^2 is the internal shield material; the remaining $\sim 65 \text{ gm/cm}^2$ is comprised of the core support plate and other hardware associated with the reactor design. Also, there are several plena which contain hydrogen of various densities located on the axis of the nuclear subsystem. The thickest BATH shield employed in Reference26 consisted of only 109 gm/cm^2 forward of the reactor core. Based on the mass thickness difference alone, in order to determine uncertainty factors applicable to the analysis of the R-1 reactor, it is necessary to extrapolate the experimental data from 109 gm/cm^2 to 175 gm/cm^2 . Extrapolations of this sort are presented in Reference 26.

Based on a comparison of the extrapolation of experimental and analytical data, it is possible to estimate uncertainty factors to be applied to the analysis of a 175 gm/cm^2 shield. The uncertainty factor is defined as:

$$F = \frac{\{\phi_{\text{exp}}\}_{175}}{\{\phi_{\text{calc}}\}_{175}}$$

where:

$\{\phi_{\text{exp}}\}_{175}$ = Estimated experimental fast neutron flux or photon dose rate forward of a 175gm/cm² shield

$\{\phi_{\text{calc}}\}_{175}$ = Estimated analytical fast neutron flux or photon dose rate forward of a 175gm/cm² shield

Uncertainty factors determined via the method outlined above are summarized in Table 2-11. From Table 2-11 it is seen that on or close to the PVARA tangent plane, uncertainty factors applicable to fast neutron calculations range from 1.3 to 1.8 for the discrete ordinates method and from 1.8 to 3.1 for the point kernel technique. For photon analyses a slight reversal is observed. That is, for the discrete ordinates approach the uncertainty factor ranges from 1.7 to 2.0, while for the point kernel method the variation is from 1.2 to 1.4.

In view of the relatively large variation in uncertainty factors along the plane, it is fully realized that additional experimental data are required to further define the analytical uncertainties. Nevertheless, in lieu of experimental data directly applicable to a 175 gm/cm² configuration, these extrapolations are the only recourse. Until measurements are available for an assembly which more closely approximates the R-1 axial configuration, all R-1 design studies will be based on the more conservative uncertainty factors; i.e., those obtained near the periphery of the shield assembly. Further, the fast neutron uncertainty

factor will be applied to neutrons of all energies in lieu of applicable experimental data for neutrons of other energies. As more and better experimental data become available, the uncertainties for the flight type reactor will be considerably less.

From the preceding discussion, it can be seen that a precise understanding of the uncertainties in the calculated external radiation environment cannot be obtained until more experimental data is obtained. The uncertainty factors in Table 2-11 really represent the average ratio of extrapolated experimental to calculated data. Due to the preliminary nature of the current results, the application of the possible one standard deviation to this average difference between extrapolated calculational and experimental results is not warranted. That is, there is not sufficient data of the right type to perform a rigorous statistical analysis. Therefore, this uncertainty factor based on extrapolated average ratio of experimental to calculated results will be used as the one standard deviation to be applied to the calculated results. By this assumption, the probability of failure that the calculated external radiation environment adjusted by the uncertainty factor exceeds the reported values is 0.16. This probability of failure applies to the R-1 reactor; however, as more and better experiments are performed, the design of the internal shield will be refined so that the probability of failure for a flight type reactor will be considerably less. The uncertainty factors in Table 2-11 and the external radiation environment at the engine specification locations (Section 2.5) which was adjusted by these uncertainty factors represent the WANL input to ANSC to evaluate possible failure modes attributed to the radiation environment generated by the NSS.

2.4.4.2.2 Statistical Results Obtained for Internal Radiation Environment Data

The calculated and experimental internal radiation environment data from the experiments described in References 26, 27, and 28 were analyzed by the statistical procedures outlined in NRP-600. This analysis revealed that the statistical variations between the calculated and experimental data have a normal distribution (See Reference 28 for details), if and only if, the statistical variation between calculated and experimental data is represented as the logarithm (log) of the experimental data minus the log of the calculated

data. Therefore, performing the statistical analysis in the log plane, the average value and standard deviation in the statistical variation between calculated and experimental results were obtained. This information was then converted back to the arithmetic plane where upon the data represented the average ratio of experimental to calculated data and the possible one standard deviation in this ratio. (See Section 2.4.4 for details.)

These analyses yielded statistical information on the radiation environment and heating rates in the core, reflector, shield, and CHESH. These statistical results are presented for these regions, respectively, in Tables 2-12, 2-13, and 2-14 for the detectors employed in each experiment. In addition, the total heat deposited in the beryllium portion of the PAX-G1 reflector was obtained by integrating the calculated and experimental results over the volume of the reflector. The results of this comparison are presented in Table 2-15 and indicate that: 1) the calculated heating due to gamma rays is low by 10 percent, 2) the calculated heating due to neutrons is low by 3 percent, and 3) the calculated total heating in the beryllium is low by 8 percent. Caution must be exercised with this data because this represents only the difference between the calculated and experimental data for the PAX-G1 reflector and does not represent the possible differences based on this data for any reflector design. This difference is derived in the following section.

2.4.4.2.3 Radiation Environment and Heating Rate Uncertainty Factors

In the preceding section on the external radiation environment, the paucity and dubious nature of the statistical results was indicated. In view of these arguments, the uncertainty factor of 3.1 (See Table 2-11) is applied to all neutron data at a point external to the NSS while an uncertainty factor of 1.4 is applied to all photon data at a point external to the NSS. This uncertainty factor is assumed to apply to 68 percent (1σ) of the data to a 95 percent confidence level. The following discussion for the internal environment data will amplify this statement.

The results presented in the preceding section for the internal radiation environment and heating rates was analyzed by the procedures presented in NRP 600 to provide a pointwise, F_p , and regionwise average, F_R , uncertainty factors for radiation data. The pointwise uncertainty factor was derived so that to a 95 percent confidence level (probability) 68 percent (1σ) of the actual radiation data, E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal data. The pointwise uncertainty factor to include more or less than 68 percent (more or less than 1σ) of the actual radiation data is derived from the uncertainty factor for one standard deviation, F_p , as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 3.0, 2.3, etc.)

The regionwise average uncertainty factor, F_R , was derived so that to a 95 percent confidence level (probability) the actual value of regionwise average radiation data, $\bar{E}^A$, will be written within the following limits:

$$E_R F_R^{-1} \leq \bar{E}^A \leq E_R F_R \quad \text{where,}$$

E_R = the reported value of the regionwise average radiation data.

In deriving these uncertainty factors, the following assumptions were made which are based on the experimental results, previous experience, and engineering judgment:

- 1) The uncertainty factors for the radiation environment and heating rates were based on the results for the following experimental detectors:
 - a) the uncertainty in the fast neutron flux ($E > 1.0$ MeV) was based on measurements with the U-238 detector,

- b) the uncertainty in the intermediate neutron flux ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) was also based on measurements with the U-238 detector,
 - c) the uncertainty in the thermal neutron flux ($E < 0.4 \text{ eV}$) was based on the bare minus episcadmium cutoff dysprosium data,
 - d) the uncertainty in the gamma ray dose rate was based on measurements with the CaF_2 thermoluminescent detector (TLD),
 - e) the uncertainty in the neutron heating rate was based on measurements with the p-i-n junction Si diode (Phylatron), and
 - f) the uncertainty in the photon heating rate was based on measurements with the CaF_2 TLD.
- 2) The uncertainty factors are based on experiments performed with graphite fuel in the core; therefore, the uncertainty factors from these experiments were increased by 20 percent to account for the added uncertainty in the use of composite fuel in the core.
- 3) Because experiments were not performed for every reactor region, the following assumptions were made:
- a) the uncertainty factors for the support plate were based on the measurements performed in the internal shield, and
 - b) the uncertainty factors for i) the plena portion of the SP/P ECC (those regions forward of the internal shield), ii) the actuator portion of the CDDA ECC, and iii) the SSCVA were taken to be the same as the uncertainty factors for the external radiation data.

In addition, in deriving the uncertainty factors, it was necessary to remove any average difference or bias between the calculated and experimental results. This was done because no deficiency in the analytical procedure was found which would lead to a bias between a calculated and experimental result and because the experimental results did not

provide a consistently clear indication that biases between calculated and experimental results existed (the exception to this is the control vane heating described in Section 2.4.1).

By consistently clear indication of a bias, the following possibilities are referred to:

- 1) The possible statistical variation in the absolute calibration of the experimental detectors which would lead to a bias,
- 2) A possible discrepancy in the calculated cross section, such as D_y , or a response function which would lead to a bias between calculated and experimental results, but because the detector material does not exist in the NSS the bias would have no significance when applied to calculational results for the NSS, and
- 3) the instances where calculational results for a given quantity, such as photon heating, agree with the experimental results for some detectors but disagree with the experimental results with other detectors. References 26, 27 and 28 provide detailed discussion regarding the validity of the experimental data which might lead to a bias between calculated and experimental results.

Tables 2-16 through 2-31 provide the pointwise and regionwise uncertainty factors both for a composite fueled core and a graphite fueled core for the ECCs which comprise the NSS. The definition of the radiation analysis model in Section 2.4.2 defines regions in the model which are associated with each ECC.

The uncertainty factors are provided in these tables for the fast neutron flux, intermediate neutron flux, thermal neutron flux, gamma ray dose rate, neutron kinetic heating rate, and photon heating rate. When considering the total heat deposition in a material which consists of the sum of neutron and photon heating, the procedure for determining propagation of errors as outlined in NRP-605 was employed. The uncertainty factors reported in Sections 2.5.1.2 and 2.5.2.2 for the total heat deposited in a material at selected state points on the reactor operating profile were derived from the neutron and photon heating rate uncertainty factors reported in Tables 2-16 through 2-31 by analysis of the propagation of error.

Table 2-32 provides the NERVA flight regionwise average uncertainty factors for the total heat deposited in the various ECCs. This data is provided for both composite and graphite fuel in the core and is based on previous experience and engineering judgment.

2.4.4.2.4 Uncertainties in Sources

There are three sources of photons considered in the analysis of the NSS:

1) prompt photons released from fission, 2) delayed photons released from fission product decay, and 3) secondary photons released from neutron interactions with the elements.

The measurement of the individual contribution of each of the three components in an NSS is not possible with the state-of-the-art techniques used in radiation measurements. When photon measurements are made in the PAX testing at WANEF, the result includes the sum of the three components. This results in an uncertainty that is the total of the three individual components. Thus, any comparison of the results of the experimental data with calculations can only be made on a total gamma dose basis to obtain an uncertainty that reflects the statistical combination of the individual ones. Due to the differences in photon energy of each of the source components, the contribution of each source component to the total dose at a location removed from the source location will depend upon the transport of these photons through the media between the source-detector locations. In this case, the comparison of calculation with experiment, an additional uncertainty enters due to the analytical transport of the source radiation to a detector location. In essence, the comparison of calculations with experiment not only includes an uncertainty in the photon source, but also includes the transport of these sources to a detector location. The uncertainties derived and presented in this data item include the combination of these uncertainties.

The literature is continuously reviewed in order to determine the best available data on the magnitude of the prompt and delayed photon and beta energy release per thermal

neutron fission of ^{235}U . At the same time, the determination of realistic uncertainties associated with these values are obtained. Radiation and Shielding presently uses 8.178 MeV/Fission, 6.92 MeV/Fission, and 8.23 MeV/Fission for the integral energy release per fission of prompt photons ($0 < t < 10^{-3}$ sec), delayed photons ($t > 10^{-3}$ sec), and total beta ($t > 0$ sec), respectively. The data generated in this data item use these values. In line with the continuous review of the literature, it has been suggested that these values be changed⁽³⁰⁾ to those presented in Table 2-33. The one sigma uncertainties for these values are also presented. These data obtained very recently are currently under evaluation by Radiation and Shielding to determine if there is a concurrence with the recommendations in Reference 30.

It is observed that the prompt photon energy release data employed in this report is the one sigma upper limit of the newly recommended value; thus, the photon source data are slight (8 percent) overestimates. The delayed photons are slightly underestimated (4 percent).

It is important to note that the uncertainty factors obtained from a comparison of the calculations with experiments presented in this data item were generated using the photon sources based upon the energy release presently in the code. If the newly recommended values had been used for the present calculations, the uncertainty factors would not apply. New factors would have to be generated based upon recalculations of the PAX data or the generation of new data. The uncertainty factors are, thus, consistent with the basic data used in the present calculations.

The determination of uncertainties in the secondary photon sources due to neutron interactions is a complicated and sometimes ambiguous process. There is presently no known method available in the state-of-the-art radiation measurements that permits the direct determination of these sources in a fissioning assembly. An indirect method of the determination of these sources is one that uses activation detectors that infer the accuracy of the calculated neutron spectra in the various regions of the NSS. The activation detectors are chosen such

that their response is due mainly to various neutron energy groups that span the total neutron energy range. These detectors are typically bare and cadmium covered Dy ($E < 0.4$ eV and $E > 0.4$ eV), S ($E > 2.9$ MeV) and U^{238} ($E > 1.5$ MeV). The ratio of the experimental to calculated neutron activation of these detectors is assumed to be directly proportional to the uncertainty in the calculated secondary photon sources. This technique does not include uncertainties in the photon yield per reaction nor biased uncertainties, such as inaccuracies in the activation cross section of the foil monitor. In lieu of any better technique, a comparison⁽³¹⁾ of the neutron activation per neutron energy region in the PAX reactor with the calculated photon source distribution yields the uncertainty factors for key NSS regions presented in Table 2-33.

Due to the recognized inadequacies of the technique to determine uncertainties in secondary photons, the data are presented only as being representative of possible deviations in these data with the intent being that further expansion of present experimental techniques will result in a considerable improvement in the uncertainty analysis.

The photon source data presented in Table 2- 33 were analyzed in Reference 31 to determine the importance of the sources to the photon dose on a 30 foot meridian ring. This was done for a polar angle of 0° (forward axis of reactor) and at a 30° polar angle. The importance of the uncertainties associated with the dose calculations was determined for the 939J685E reactor assembly. The results indicated that at the 0° polar angle the percentage contribution of the individual regions to the total photon dose were 39.6% (core), 10.0% (reflector), 28.9% (support plate), 9.0% (shield), and 12.5% (other). The combined dose using the uncertainty factors in the sources was 6 percent lower than the calculated value with a possible 5 percent greater value and 17 percent lower value based upon the experimental uncertainties in the data. These were statistically weighted with the relative importance of the sources to the total dose. Although the shield sources have a large experimental uncertainty, this contributed a relatively small contribution to the total dose since these contributed only 4 percent (calculated adjusted by experimental) to the total dose. Similar conclusions on the shield sources were obtained at the 30° polar angle.

2.5 RESULTS, ANALYSIS, AND INTERPRETATION

This section presents the results of the radiation analysis of the overall NSS and interpretation of the significance of these results.

2.5.1 Composite Core

The radiation analysis was performed for the NSS containing both a composite fueled core and a graphite fueled core. This section presents the results obtained for the NSS containing a composite fueled core.

2.5.1.1 Steady State

This section provides radiation data for the NSS at steady state conditions which are defined to be: 1) a composite fueled core in the Start Of Life (SOL) condition, 2) nuclear properties which exist at the end of 60 minutes of full power operation, and 3) thermal-hydraulic conditions which exist at the End Of Life (EOL). In addition, radiation sources and dose rates for the NSS after shutdown are also provided in this section.

2.5.1.1.1 Internal Radiation Environment

Figures 2-15 through 2-36 present the gamma ray dose rate internal to the NSS with a composite fueled core on several axial and radial traverses. Figures 2-15 through 2-23 present the data on the following axial traverses:

- 1) Reactor centerline ($r=0.0$ cm),
- 2) Core mid radius ($r=23.694$ cm),
- 3) Outer radius of the core ($r=47.388$ cm),
- 4) Outer radius of the core periphery ($r=49.469$ cm),
- 5) Outer radius of the lateral support ($r=53.632$ cm),
- 6) Outer radius of the aluminum structure ($r=56.426$ cm),
- 7) Control drum centerline ($r=62.230$ cm),
- 8) Outer radius of the reflector ($r=68.580$ cm), and
- 9) Surface of the pressure vessel side ($r=70.841$ cm).

Figures 2-24 through 2-36 present the data on the following radial traverses:

- 1) Aft surface of the CHESH ($z=23.035$ cm),
- 2) Core exit ($z=28.448$ cm),
- 3) Core midplane ($z=94.488$ cm),
- 4) Core inlet ($z=160.528$ cm),
- 5) Aft surface of the support plate ($z=169.418$ cm),
- 6) Forward surface of the support plate ($z=184.658$ cm),
- 7) Aft surface of the shield plate ($z=188.036$),
- 8) Aft surface of the BATH central shield ($z=190.576$ cm),
- 9) Midplane of BATH central shield ($z=204.673$ cm),
- 10) Forward surface of BATH central shield ($z=218.770$ cm),
- 11) Forward surface of lead central shield ($z=222.072$ cm),
- 12) Forward surface of the flow baffle ($z=238.633$ cm), and,
- 13) Surface of the pressure vessel dome ($z=254.508$ cm).

It should be noted that the CRAM model coordinate system has been employed in reporting this data (See Section 2.4.2) and the uncertainty factors for this data are reported in Section 2.4.4.2.

The following data presented in Figures 2-37 through 2-102 at the same axial and radial traverses described above for the gamma ray dose rate data:

- 1) Figures 2-37 through 2-58 present the axial and radial traverses of the fast neutron ($E>1.0$ MeV) flux,
- 2) Figures 2-59 through 2-80 present the axial and radial traverses of the intermediate neutron (0.4 eV $<E<1.0$ MeV) flux, and,
- 3) Figures 2-81 through 2-102 present the axial and radial traverses of the thermal neutron ($E<0.4$ eV) flux.

Figures 2-103 through 2-107 present the neutron flux as a function of the neutron energy for 155 energy groups at the following spatial locations:

- 1) Average neutron flux in the center core region (region 1 in the DOT model, See Section 2.4.2 for region definition),

- 2) Average neutron flux in the Reflector (region 21),
- 3) Average neutron flux in the side pressure vessel (region 22),
- 4) Average neutron flux in the BATH central shield (region 36), and
- 5) Average neutron flux in the pressure vessel dome (region 53).

These data were obtained by interpolation of the 52 energy group neutron spectra which were employed to spectrally weight the neutron cross sections used in this analysis (See Section 2.4.1). The 52 group spectra were obtained from the one dimensional, discrete ordinate code, ANISN.

An examination of Figures 2-103 through 2-107 reveals that, with the exception of the neutron spectrum for the core center, all of the spectra generally have the same shape and peak at an energy of approximately .04 eV. The spectrum at the core center peaks at approximately 1 eV which is indicative of a highly absorbing medium with a low moderator to absorber (C/U) ratio.

Figures 2-108 through 2-112 present the photon energy flux as a function of energy for 13 energy groups (standard WISDM photon energy group structure) at the following spatial locations:

- 1) Core center,
- 2) Inner radius of the beryllium portion of the reflector on the core midplane,
- 3) Surface of the pressure vessel side on the core midplane,
- 4) Aft surface of the BATH central shield on the reactor centerline, and
- 5) Surface of the pressure vessel dome on the reactor centerline.

These data were obtained directly from the R-Z geometry, discrete ordinates transport calculations with the DOT code.

Figures 108 through 112 reveal that with the exception of the photon energy flux at the pressure vessel dome, all of the spectra have generally the same shape and peak at an energy of approximately 1 MeV. The energy flux at the pressure vessel dome peaks at approximately 2.5 MeV which is indicative of the presence of the internal shield and the consequent attenuation of low energy gamma rays.

Figures 2-113 through 2-116 present the photon radiation heating rate for stainless steel 304 and aluminum 6061 on the reactor centerline internal to the NSS. Two figures present these data in units of watts/gm, while the other two figures present these data in units of Btu/hr-in³. These data are not the total heat deposition in these materials but represents a measure of the photon dose rate which is radiation environment data internal to the NSS.

Examination of Figures 2-113 through 2-116 reveals that the maximum photon heating rate occurs at the core center and is 68 w/gm-Al and 70 w/gm-stainless steel. The heating rates in these materials at the pressure vessel dome are, respectively, 0.013 w/gm-Al and 0.014 w/gm-stainless steel, for the reactor operating at full power (1511 megawatts thermal) in space.

Table 2-34 provides a comparison between the internal radiation environment in the NSS with a composite fueled core and the environment in the NSS with a graphite fueled core for the following key locations:

- 1) Reactor centerline on the core midplane,
- 2) Inner radius of the beryllium portion of the reflector on the core midplane,
- 3) Surface of the pressure vessel side on the core midplane,
- 4) Aft surface of the CHESH on the reactor centerline,
- 5) Aft surface of the BATH central shield on the reactor centerline, and
- 6) Surface of the pressure vessel dome on the reactor centerline.

A survey of this table and the radiation environment plots presented above for the NSS with a composite fueled core and presented in Section 2.5.2.1 for the NSS with a graphite fueled core reveals that in general for the regions outside the core:

- 1) the gamma ray dose rate is 5 to 25 percent lower with composite fuel,
- 2) the fast ($E > 1.0$ MeV) neutron flux is slightly higher by 2 to 8 percent with composite fuel,
- 3) the intermediate ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) neutron flux is slightly higher by 3 to 11 percent with composite fuel, and
- 4) the difference in the thermal ($E < 0.4 \text{ eV}$) neutron flux varied from 6 percent higher to 21 lower with composite fuel.

Table 2-35 presents the radiation environment at the preceding key locations internal to the NSS. This table presents the fast, intermediate, and thermal neutron fluxes, the heating rate (w/gm-SS), and the gamma ray dose rate (Rads-C/hr). Also presented on this table are the neutron fluences and gamma ray doses for one hour operation at full power (1511 Mw) and for 60 ten minute cycles as defined by the power profile in Section 2.5.1.2. The radiation environment at these key locations in the NSS, plus all of the other locations defined by Figure 2-6 and Table 2-3, have been integrated for the nominal reactor operating profile (ten minutes at rated temperature) to provide fluences and doses at several state points during the operation. These data are given in Section 2.5.1.2.1.

The primary purpose of the internal radiation environment data is to provide support to thermal analyses so that thermal properties at the proper irradiated conditions are obtained. In addition, the internal radiation environment data is employed to evaluate the effects of radiation damage on the reactor materials. The S-110 Data Item describes the NSS materials and the safe radiation level which can occur before significant radiation damage occurs. By comparing these levels with the preceding results and the radiation data to be presented in Section 2.5.1.2 for a graphite fueled core, the effect of radiation damage on NERVA materials can be assessed.

2.5.1.1.2 Sources for the Nuclear Subsystem

The geometric model of the R-1 Reference Design with composite fuel in the core is shown in Figure 2-13 which depicts a cross sectional view of the reactor taken in the R-Z plane. The model shown in Figure 2-13 is referenced to the intersection of the reactor axis with the pressure vessel-nozzle mating plane and all dimensions are given in centimeters. The material composition of each of the 35 geometric regions shown in Figure 2-13 is given in Table 2-9.

The gamma ray source strengths for each of the zones depicted in Figure 2-13 are given in Table 2-36, and are presented in the standard WISDM 13 energy group structure. These data represent the total gamma ray energy release rate (MeV/sec) in each region for maximum gamma activity during operation. This condition occurs at the end of an assumed 60 minute reactor burn at a power level of 1511 megawatts (Thermal).

Because the pressure vessel is an ANSC component, the photon source distributions which pertain to the pressure vessel portion (Regions 7, 34, and 35) of the NSS CRAM model are presented in this chapter of the S-100 Data Item. Figures 2-117 through 2-122 present the relative radial and relative axial distribution of the photon sources in the three pressure vessel regions.

The photon source strength at any location and for any given energy group in the pressure vessel is determined in the following manner:

- 1) Determine the value of the relative radial distribution of photon source at the position of interest,
- 2) Determine the value of the relative axial distribution of the photon source at the position of interest,
- 3) Determine the source strength for the energy group and region of interest from Table 2-36, and
- 4) The source strength for the energy group and position of interest is the product of the source strength, the value of the relative radial distribution, and the value of the relative axial distribution.

The gamma ray source terms resulting from decay of fission fragments following termination of 10, 30, and 60 minute full power (1511 Mw Thermal) reactor runs are presented in Tables 2-37, 2-38, and 2-39, respectively. The data presented in Tables 2-37 through 2-39 encompass a time period ranging from 30 minutes to 1 year after reactor shutdown; and from a group structure standpoint, is compatible with the full power data presented in Table 2-36.

2.5.1.1.3 External Radiation Environment

Tables 2-40 through 2-44 present the radiation environment external to the NSS with composite fuel in the core at the engine specification locations. The data presented in Table 2-40 for the plane located at 63 inches from the core center apply to the center-line only. Data for other radii in this plane are presented graphically in subsequent figures of this chapter.

These tables provide the following information:

- 1) The location of the engine specification data point relative to the CRAM model defined in Figure 2-13.

- 2) The photon kerma rate (Rad-C/hr), the fast ($E > 1.0$ MeV) neutron flux, the intermediate ($0.4 \text{ eV} < E < 1.0$ MeV) neutron flux, and the thermal ($E < 0.4 \text{ eV}$) neutron flux nominal values as determined by the discrete ordinates, DOT-MAP, and KAP-V point kernel techniques described in Section 2.4.1.
- 3) The KAP-V radiation environment which has been adjusted for the one standard deviation uncertainty factor derived in Section 2.4.4.2.
- 4) The KAP-V radiation environment which has been adjusted both for the one standard deviation uncertainty factor and for the effects of radiation streaming described in Section 2.4.1.
- 5) The Engine Specification values of the radiation levels.
- 6) The ratio of the specification values to the KAP-V radiation environment values adjusted for both uncertainties and streaming.

Figures 2-123 through 2-126 present plots of the information in items two through five above as a function of radial position on the plane located 63 inches from the core center. It should be noted that the change in the peak in the curve adjusted for streaming at a radius of approximately 55 centimeters is due to the increased streaming assessed to the piping penetrations located in the peripheral portion of the shield. Section 2.4.1 summarizes and describes the derivation of the factors employed for radiation streaming in this chapter.

From these tables and figures, it is seen that the Engine Specification radiation levels are exceeded only for the intermediate energy neutron flux out to a radius of approximately 30 cm on the tangent plane. The specification values are exceeded by a maximum of 16 percent at the reactor centerline of the tangent plane. All other Engine Specification radiation levels have been met. It should be noted that these results are for the internal shield defined by the NSS reference design (WANL Drawing 939J723). Further, the gamma ray radiation levels are at least a factor of two lower than the specification indicating the need for more neutron shielding and less photon shielding. The design of an internal shield that meets all specification radiation levels is discussed in Chapter 7.0 and in Section 2.2.

Figures 2-127 through 2-130 present the nominal radiation environment external to the NSS with composite fuel in the core as a function of polar angle on 10, 20, and 30 foot meridian rings. These figures have an inset diagram which defines the relationship of the meridian ring angle to the core center.

Figures 2-131 and 2-132 present the nominal external radiation environment for the NSS with composite fuel in the core as a function of axial distance from the core center. This data is presented on the reactor centerline and extends from the pressure vessel dome in the forward direction. It is interesting to note from Figure 2-131 that the photon dose rate on the centerline forward of the shield is only about 15% higher for the case of the reactor core employing graphite fuel.

Figure 2-133 presents the gamma ray iso-dose plot external to the NSS for a "space-like" environment. Figures 2-134 through 2-136 present iso-flux plots for the fast, intermediate, and thermal neutron fluxes. These data have been supplied so that preliminary evaluations of the effect of the NSS on Test Cell C can be made.

It is also important to note the relative importance of the various source regions in the NSS to the total photon dose rate at polar angles on the meridian ring. Table 2-45 provides these comparisons in terms of relative dose rate in percent.

The gamma ray dose rate on a ten foot meridian ring due to the decay of fission fragments following the termination of 10, 30, and 60 minutes of full power (1511 Mw Thermal) reactor runs are presented in Tables 2-46, 2-47, and 2-48, respectively, as a function of polar angle and time after shutdown. These data, which encompass a time from 30 minutes to 1 year after reactor shutdown, are based on the decay photon sources presented in Tables 2-37 through 2-39, and were generated with the KAP-V code.

At 0°, the gamma ray dose rate one hour after termination of a 60 minute full power run is 1.8×10^3 Rads-C/hr compared to 9.5×10^5 during operation.

2.5.1.1.4 Total Heat Deposition by Region

Table 2-49 provides the NSS heating rate by region at full power (1511 Mw Thermal). This table provides the following information for the materials located within

the Nuclear Subsystem with a composite fueled core:

- 1) Weight and density of materials in each region of the Nuclear Subsystem as defined by Figure 2-11, and
- 2) Neutron, prompt photon, delayed photon, and total heating rates for the materials located within the Nuclear Subsystem.

Figure 2-137 provides the relative heating rate as a function of reactor power level. This information can be employed to adjust the heating rate data if the reactor is operated at other than 1511 Mw Thermal.

This data is provided so that thermal computations can be performed. In addition, the heating rates in Table 2-49 have been provided as neutron and photon heating rates so that the neutron and photon heating rate uncertainty factors developed in Section 2.4.4.2 can be applied to the data to obtain a total heating rate uncertainty factor according to the following formula which was derived via the propagation of errors outlined in NRP 605:

$$F_t = \text{Total heating rate uncertainty factor}$$

$$F_t = 1.0 + \frac{\sqrt{[H_n (F_n - 1.0)]^2 + [H_\gamma (F_\gamma - 1.0)]^2}}{H_t}$$

where,

H_n = The heating due to neutrons,

H_γ = The heating due to photons,

$H_t = H_n + H_\gamma$ = the total heating,

F_n = The neutron heating uncertainty factor, and

F_γ = The photon heating uncertainty factor.

Table 2-50 provides a comparison between the heating rates with a composite fueled core and the heating rates with a graphite fueled core for an assortment of materials and regions. A comparison of the heating rate for every material in every region can be made by examining the data presented in this section and the data presented in Section

2.5.2.1.4. A survey of the data in Table 2-50 indicates that in general for regions outside the core:

- 1) the heating rate primarily caused by photons in materials (i.e., metals) decreases by 4 to 20 percent with the maximum decreases being 33 percent in the CHESH for a composite fueled core,
- 2) the heating rate in materials which have a significant amount of neutron heating (i.e., hydrogen, beryllium, and carbon) decreased by 5 to 10 percent with the maximum change being 18 percent again occurring in the CHESH for a composite fueled core, and
- 3) the heating rate in materials which have an (n,α) reaction display a mixed change of plus 4 percent to minus 6 percent for the composite fueled core.

The heating rate distributions for all of the Type A components within the NSS are presented in the appropriate ECC chapter in this report. However, the pressure vessel is not a WANL component and hence is not an ECC. Therefore, the heating rate distributions for the pressure vessel are presented within this section for the Nuclear Subsystem. It should be noted that all of the volumetric heating rates provided in this report are per cubic inch of solid material.

Figures 2-138 through 2-143 provide the relative radial and relative axial distribution of radiation heating for the pressure vessel, regions 22, 42, and 53 of the heating rate model as defined by Figure 2-11. The heating rate at any location within the pressure vessel regions is determined in the following manner:

- 1) Determine the value of the relative radial distribution of heating for the position of interest,
- 2) Determine the value of the relative axial distribution of heating for the location of interest,

- 3) Determine the average value of the heating rate from the appropriate curve (the average value of the heating rate appears on both the radial and axial distribution curves for each region), and
- 4) The heating rate at the location of interest is the product of the average heating rate, the value of the relative radial distribution, and the value of the relative axial distribution.

2.5.1.1.5 Effect of Test Cell C on the Radiation Data Internal to the NSS

Figures 2-144 through 2-152 present the ratio of radiation data with Test Cell C present to the radiation data in absence of Test Cell C as a function of radial position internal to the NSS for the following quantities:

- 1) the gamma ray dose rate,
- 2) the fast ($E > 1.0$ MeV) neutron flux,
- 3) the intermediate ($0.4 \text{ eV} < E < 1.0$ MeV) neutron flux,
- 4) the thermal ($E < 0.4$ eV) neutron flux,
- 5) the heating rate in Al 6061,
- 6) the heating rate in Al 7039,
- 7) the heating rate in A-286,
- 8) the heating rate in carbon, and
- 9) the heating rate in beryllium.

With the exception of the heating rate in beryllium, Figures 2-153 through 2-160 provide plots of similar information as a function of axial position. It should be noted, that the data presented in these figures represent the maximum effect that Test Cell C can have on the radiation data because the Test Car and the Test Cell C shield have been placed directly adjacent to the NSS (See Section 2.4.1 for details of the calculational procedure).

An examination of the data in the preceding figures reveals that, in the radial direction, the presence of Test Cell C creates a maximum increase in:

- 1) the gamma ray dose rate of 44 percent,

- 2) the fast neutron flux of 23 percent,
- 3) the intermediate neutron flux of a factor of 2.0,
- 4) the thermal neutron flux of a factor of 2.9, and
- 5) the heating rate of 44 percent.

While in the axial direction, the presence of Test Cell C creates a maximum increase in:

- 1) the gamma ray dose rate of a factor of 5.0,
- 2) the fast neutron flux of 28 percent,
- 3) the intermediate neutron flux of 78 percent,
- 4) the thermal neutron flux of a factor of 2.3, and
- 5) the heating rate of a factor of 3.4.

2.5.1.2 Transient

Figure 2-161 displays the power profile, percent full power as a function of time, for the NSS which employs either a composite fueled core or a graphite fueled core. It should be noted that the power profile shown is for ten minutes at a chamber temperature of 4250 R with a throttled hold of 65 percent and nominal specification startup and shutdown ramps of 150°R per second. Identified on this power profile are six state points labeled A, B, C, D, E, and F. The radiation data provided in the following sections are based on the power profile displayed in Figure 2-161 and, in some cases, reference either the times at which the state points occur or the state points themselves which are shown in Figure 2-161.

2.5.1.2.1 Radiation Environment

Figures 2-162 through 2-164 display, respectively, the integrating factor as a function of time during the run to convert

- 1) the neutron flux or neutron heating rate divided by 3600 to the corresponding neutron fluence or neutron heat,
- 2) the prompt photon dose rate divided by 3600 or the prompt photon heating rate divided by 3600 to the corresponding prompt photon dose or prompt photon heat, and

- 3) the delayed photon dose rate divided by 3600 or the delayed photon heating rate divided by 3600 to the corresponding delayed photon dose or delayed photon heat.

The conversion factors presented in these figures can be applied to any of the previously reported radiation data to obtain the data integrated over the power profile up to the time of interest.

Table 2-51 presents radiation data for the NSS with a composite fueled core at each of the selected locations defined in Figure 2-6 and Table 2-3 integrated over the power profile from 1) State Point A to State Point B (identified as State Point B on Table), 2) State Point A to State Point C (identified as State Point C on Table), 3) State Point A to State Point D (identified as State Point D on Table), and 4) State Point A to State Point E (identified as State Point E on Table). The radiation data at State Point A is zero while the radiation data integrated from State Point A to State Point F is the same as the data reported at State Point E to the significance to which the data is reported in Table 2-51.

Table 2-51 presents the following information for each selected location and state point:

- 1) the gamma dose and associated uncertainty factor (See Section 2.4.4.2 for the details concerning the uncertainty factor),
- 2) the fast neutron fluence and associated uncertainty factor,
- 3) the intermediate neutron fluence and associated uncertainty factor,
- 4) the thermal neutron fluence and associated uncertainty factor, and
- 5) total heat deposition in selected materials and the associated uncertainty factor.

An approximation to the radiation data for the n^{th} operation of the NSS over the power profile defined by Figure 2-161 can be made by multiplying the data in Table 2-51 at State

Point E by $n-1$ and then adding the radiation data at the state point of interest in the n^{th} operation of the NSS.

2.5.1.2.2 Total Heat Deposition by Region

Figure 2-165 presents the prompt total reactor power and the delayed total reactor power as a function of run time for the power profile defined in Figure 2-161. Figure 2-166 presents the power after shutdown due to delayed photons and beta particles as a function of time after shutdown.*

Figures 2-162 through 2-164 provided the integrating factors to convert the total heating rate for a region or material in a region, as supplied by Table 2-49, to the total heat deposition at any point in time during the run profile defined by Figure 2-161. It should be noted that the neutron, prompt, and delayed integrating factors must be applied to the corresponding heating rate in Table 2-49.

2.5.1.2.3 Sources for the Nuclear Subsystem

The gamma ray source strength for the composite core defined in Figure 2-13 are given in Table 2-52 for the times at which the State Points defined on Figure 2-161 occur. These data are presented in the standard WISDM 13 energy structure. Table 2-53 provides the factors to convert the gamma ray source terms provided at steady state conditions for the non-core regions of the NSS with a composite fueled core to the source terms which occur at the State Points defined by Figure 2-161.

2.5.2 Graphite Core

The radiation analysis of the NSS was performed for the NSS containing a composite fueled core and for the NSS containing a graphite fueled core. This section presents the results obtained for the NSS containing a graphite fueled core.

* Power due to delayed neutrons, which is important for approximately 100 seconds after shutdown is not included in these data.

2.5.2.1 Steady State

This section provides radiation data for the NSS at steady state conditions which are defined to be 1) a graphite fueled core in the Start Of Life (SOL) condition, 2) nuclear properties which exist at the end of 60 minutes of full power operation, and 3) thermal-hydraulic conditions which exist at the End Of Life (EOL). In addition, radiation sources and dose rates for the NSS after shutdown are also provided in this section.

2.5.2.1.1 Internal Radiation Environment

Figures 2-167 through 2-188 present the gamma ray dose rate internal to the NSS with a graphite fueled core on several axial and radial traverses. Figures 2-167 through 2-175 present the data on the following axial traverses:

- 1) Reactor centerline ($r=0.0$ cm),
- 2) Core mid radius ($r=23.694$ cm),
- 3) Outer radius of the core ($r=47.388$ cm),
- 4) Outer radius of the core periphery ($r=49.469$ cm),
- 5) Outer radius of the lateral support ($r=53.632$ cm),
- 6) Outer radius of the aluminum structure ($r=56.426$ cm),
- 7) Control drum centerline ($r=62.230$ cm),
- 8) Outer radius of the reflector ($r=68.580$ cm), and
- 9) Surface of the pressure vessel side ($r=70.841$ cm).

Figures 2-176 through 2-188 present the data on the following radial traverses:

- 1) Aft surface of the CHESH ($z=23.035$ cm),
- 2) Core exit ($z=28.448$ cm),
- 3) Core midplane ($z=94.488$ cm),
- 4) Core inlet ($z=160.528$ cm),
- 5) Aft surface of the support plate ($z=169.418$ cm),
- 6) Forward surface of the support plate ($z=184.658$ cm),
- 7) Aft surface of the shield plate ($z=188.036$ cm),

- 8) Aft surface of the BATH central shield ($z=190.576$ cm),
- 9) Midplane of BATH central shield ($z=204.673$ cm),
- 10) Forward surface of BATH central shield ($z=218.770$ cm),
- 11) Forward surface of lead central shield ($z=222.072$ cm),
- 12) Forward surface of the flow baffle ($z=238.633$ cm),
- 13) Surface of the pressure vessel dome ($z=254.508$ cm).

It should be noted that the CRAM model coordinate system has been employed in reporting this data (See Section 2.4.2) and the uncertainty factors for this data are reported in Section 2.4.4.2.

The following data are presented in Figures 2-189 through 2-254 at the same axial and radial traverses described above:

- 1) Figures 2-189 through 2-210 present the fast neutron ($E > 1.0$ MeV) flux data,
- 2) Figures 2-211 through 2-232 present the intermediate neutron ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) flux data, and,
- 3) Figures 2-233 through 2-254 present the thermal neutron ($E < 0.4 \text{ eV}$) flux data.

Figures 2-255 through 2-259 present the neutron flux as a function of the neutron energy for 155 energy groups in the following regions:

- 1) Average neutron flux in the center core region (region 1 in Figure 2-7),
- 2) Average neutron flux in the reflector (region 21),
- 3) Average neutron flux in the pressure vessel side (region 22),
- 4) Average neutron flux in the BATH central shield (region 36), and,
- 5) Average neutron flux in the pressure vessel dome (region 53).

These data were obtained by the procedure described in Section 2.5.1.1.1, and, as in the case of the composite fueled core, the core center spectrum for a graphite fueled core displays a different shape and peaks at a different energy than the spectrums for the other regions. This occurs because the core is a highly absorbing medium with a low moderator to absorber (C/U) ratio.

Figures 2-260 through 2-264 present the photon energy flux as a function of photon energy for 13 energy groups (Standard WISDM photon energy group structure) at the following spatial locations.

- 1) Core center,
- 2) Inner radius of the beryllium portion of the reflector on the core midplane,
- 3) Surface of the pressure vessel side on the core midplane,
- 4) Aft surface of the BATH central shield on the reactor centerline, and
- 5) Surface of the pressure vessel dome on the reactor centerline.

These data were obtained directly from the R-Z geometry, discrete ordinates transport calculations with the DOT code.

Figures 2-260 through 2-264 reveal that with the exception of the photon energy flux at the pressure vessel dome, all of the data have generally the same shape, and peak at an energy of approximately 1 MeV. The fluxes at the pressure vessel dome peak at approximately 2.5 MeV which is indicative of the presence of the internal shield and the consequent attenuation of the low energy gamma rays.

Figures 2-265 through 2-268 present the photon radiation heating rate for stainless steel 304 and aluminum 6061 on the reactor centerline internal to the NSS. Two figures are presented in units of watts/gm, while the other two figures present this data in units of Btu/hr-in³. These data are not the total heat deposition in these materials but represent a measure of the photon dose rate which is radiation environment data internal to the NSS.

Examination of Figures 2-265 through 2-268 reveals that the maximum photon

heating rates in these materials at the pressure vessel dome are, respectively, 0.014 w/gm-Al and 0.015 w/gm-stainless steel.

Table 2-34 presented a comparison of the radiation environment in the NSS with composite fuel to the radiation environment in the NSS with graphite fuel. The details of this comparison are presented in Section 2.5.1.1.1 and indicate that 1) the gamma ray dose rate is 5 to 25 higher with graphite fuel, 2) the fast and intermediate neutron fluxes slightly lower, and 3) the change in the thermal neutron flux was both higher and lower.

Table 2-54 presents the radiation environment at some key locations internal to the NSS with a graphite fueled core. This table presents the fast, intermediate and thermal neutron fluxes, the heating rate (w/gm-SS), and the gamma ray dose rate (Rads-C/hr). Also presented on this table are the neutron fluences and gamma ray doses for one hour operation at full power (1511 Mw) and for 60 ten minute cycles as defined by the Power Profile in Section 2.5.1.2. It should be noted that the radiation environment at these key locations in the NSS plus all of the other locations defined by Figure 2-6 and Table 2-3 have been integrated for the nominal reactor operating profile (10 minutes at rated temperature) to provide fluences and doses at several state points during the operation. These data are given in Section 2.5.2.2.1. In addition, radiation damage limits for the NSS materials are given in the S-110 Data Item.

2.5.2.1.2 Sources for the Nuclear Subsystem

The geometric model of the R-1 Reference Design with graphite fuel in the core is shown in Figure 2-14 which depicts a cross sectional view of the reactor taken in the R-Z plane. The model shown in Figure 2-14 is referenced to the intersection of the reactor axis with the pressure vessel-nozzle mating plane and all dimensions are given in centimeters. The material composition of each of the 36 geometric regions shown in Figure 2-14 is given in Table 2-10.

The gamma ray source strengths for each of the zones depicted in Figure 2-14 are given in Table 2-55, and are presented in the standard WISDM 13 energy group structure. These data represent the total gamma ray energy release rate (MeV/sec) in each region for maximum gamma activity during operation. This condition occurs at the end of an assumed

60 minute reactor burn at a power level of 1511 megawatts (Thermal).

Because the pressure vessel is an ANSC component, the photon source distributions which pertain to the pressure vessel portion (Regions 7, 34, and 35) of the NSS CRAM model are presented in this chapter of the S-100 Data Item. Figures 2-117 through 2-122 present the relative radial and relative axial distribution of the photon sources in the three pressure vessel regions, and are the same pressure vessel source strength distributions presented for the NSS with a composite fueled core because no change in the distributions occurs with the change in cores. Section 2.5.1.1.2 also describes the manner in which the source strength at any location and energy group is obtained.

The gamma ray source terms resulting from decay of fission fragments following termination of 10, 30, and 60 minute full power (1511 Mw Thermal) reactor runs are presented in Tables 2-37, 2-38, and 2-39, respectively. The data presented in Tables 2-37 through 2-39 encompass a time period ranging from 30 minutes to 1 year after reactor shutdown; and from a group structure standpoint, is compatible with the full power data presented in Table 2-55.

2.5.2.1.3 External Radiation Environment

Tables 2-56 through 2-60 present the radiation environment external to the NSS with graphite fuel in the core at the engine specification locations. These tables provide the following information:

- 1) The location of the engine specification data point relative to the CRAM model defined in Figure 2-14.
- 2) The nominal photon kerma rate (rad-C/hr), the fast ($E > 1.0$ MeV) neutron flux, the intermediate ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) neutron flux, and the thermal ($E < 0.4 \text{ eV}$) neutron flux as determined by the discrete ordinates, DOT-MAP, and point kernel techniques described in Section 2.4.1.
- 3) The KAP-V radiation environment which has been adjusted for the one standard deviation uncertainty factor derived in

Section 2.4.4.2.

- 4) The KAP-V radiation environment which has been adjusted both for the one standard deviation uncertainty factor and for the effects of radiation streaming described below.
- 5) The Engine Specification values of the radiation levels.
- 6) The ratio of the specification values to the KAP-V radiation environment values adjusted for both uncertainties and streaming.

Figures 2-269 through 2-272 also present plots of the information in items two through five above as a function of radial position on the plane located 63 inches from the core center. It should be noted that the peak in the radiation curve adjusted for streaming at a radius of approximately 55 centimeters is due to the increased streaming assessed to the piping penetrations located in the peripheral portion of the shield. Section 2.4.1 summarizes and describes the derivation of the factors employed for radiation streaming.

From these tables and figures, it is seen that the Engine Specification radiation levels are exceeded only for the intermediate energy neutron flux out to a radius of approximately 30 cm on the tangent plane. The specification values are exceeded by a maximum of 16 percent at the reactor centerline of the tangent plane. All other Engine Specification radiation levels have been met. These results are for the BATH-lead shield described in Chapter 7.0. It should be noted that the gamma ray radiation levels are at least a factor of two lower than the specifications. The design of an internal shield that meets all specification radiation levels is discussed in Chapter 7.0 and in Section 2.2.

Figures 2-127 through 2-130 previously presented the external radiation environment for the NSS with graphite fuel in the core as a function of polar angle on 10, 20, and 30 foot meridian rings. These figures have an inset diagram which defines the relationship of the meridian ring angle to the core center.

Figures 2-131 and 2-132 also presented the external radiation environment for the NSS with graphite fuel in the core as a function of axial distance from the core center. This data is presented on the reactor centerline and extends from the pressure vessel dome in the forward direction.

Figure 2-273 presents the gamma ray iso-dose plot external to the NSS for a "space-like" environment. Figures 2-134 through 2-136 present iso-flux plots for the fast, intermediate, and thermal neutron fluxes. These data have been supplied so that preliminary evaluations of the effect of the NSS on Test Cell C can be made.

It is also interesting to note the relative importance of the various source regions in the NSS to the total photon dose rate at polar angles on the meridian ring. Table 2-61 provides these comparisons in terms of relative dose rate in percent.

The gamma ray dose rate on a ten foot meridian ring due to the decay of fission fragments following the termination of 10, 30, and 60 minutes of full power (1511 Mw Thermal) reactor runs are presented in Table 2-62, 2-63, and 2-64, respectively, as a function of polar angle and time after shutdown. The data presented in Tables 2-63 through 2-65 encompass a time from 30 minutes to 1 year after reactor shutdown and are based on the decay photon sources presented in Tables 2-37 through 2-39. At 0° , the gamma ray dose rate one hour after termination of a 60 minute full power run is a factor of 1.4 higher for the graphite fueled NSS compared to the composite fueled NSS.

2.5.2.1.4 Total Heat Deposition by Region

Table 2-65 provides the NSS heating rate by region at full power (1511 Mw Thermal). This table provides the following information for the materials located within the Nuclear Subsystem with a graphite fueled core:

- 1) Weight and density of materials in each region of the Nuclear Subsystem as defined by Figure 2-12, and
- 2) Neutron, prompt photon, delayed photon, and total heating rates for the materials located within the Nuclear Subsystem.

Figure 2-137 provides the relative heating rate as a function of reactor power level. This information can be employed to adjust the heating rate data if the reactor is operated at other than 1511 Mw Thermal.

This data is provided so that thermal computations can be performed. In addition, the heating rates in Table 2-65 have been given as neutron and photon heating so that the neutron and photon heating rate uncertainty factors developed in Section 2.4.4.2 can be applied to this data to obtain a total heating rate uncertainty factor according to the formula provided in Section 2.5.1.1.4.

Section 2.5.1.1.4 also provides a comparison (Table 2-50) between the heating rates for the NSS with a composite fueled core and the heating rates with a graphite fueled core. Briefly, it was found that 1) the heating rate in metals was 4 to 20 percent higher, 2) the heating rates in H, C, and Be was 5 to 10 percent higher, and 3) the heating rate in CuB and BATH, (n, α) reaction, displayed a mixed change with a graphite fueled core.

The heating rate distributions for all of the Type A components within the NSS are presented in the appropriate ECC chapter in this report. However, the pressure vessel is not a WANL component and hence, is not an ECC. Therefore, the heating rate distributions for the pressure vessel are presented within this section for the Nuclear Subsystem.

Figures 2-274 through 2-279 provide the relative radial and relative axial distribution of radiation heating for the pressure vessel, regions 22, 42, and 53 of the heating rate model as defined by Figure 2-12. Section 2.5.1.1.4 describes how the heating rate at any location within the pressure vessel is obtained from this data.

2.5.2.1.5 Effect of Test Cell C on the Radiation Data Internal to the NSS

Figures 2-280 through 2-288 present the ratio of radiation data with Test Cell C present to the radiation data in absence of Test Cell C as a function of radial position internal to the NSS with a graphite fueled core for the following quantities:

- 1) the gamma ray dose rate,
- 2) the fast ($E > 1.0$ MeV) neutron flux,
- 3) the intermediate ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) neutron flux,

- 4) the thermal ($E < 0.4$ eV) neutron flux,
- 5) the heating rate in Al 6061,
- 6) the heating rate in Al 7039,
- 7) the heating rate in A-286,
- 8) the heating rate in carbon, and
- 9) the heating rate in beryllium.

With the exception of the heating rate in beryllium, Figures 2-289 through 2-296 provide plots of similar information as a function of axial position. It should be noted, that the data presented in these figures represent the maximum effect that Test Cell C can have on the radiation data because the Test Car and the Test Cell C shield have been placed directly adjacent to the NSS (See Section 2.4.1 for details of the calculational procedure).

An examination of the data in the preceding figures reveals that, in the radial direction, the presence of Test Cell C causes a maximum increase in:

- 1) the gamma ray dose rate of 37 percent,
- 2) the fast neutron flux of 23 percent,
- 3) the intermediate neutron flux of a factor of 2.0,
- 4) the thermal neutron flux of a factor of 2.8, and
- 5) the heating rate of 36 percent.

While, in the axial direction, the presence of Test Cell C causes a maximum increase in:

- 1) the gamma ray dose rate of a factor of 5.0,
- 2) the fast neutron flux of 28 percent,
- 3) the intermediate neutron flux of 78 percent,
- 4) the thermal neutron flux of a factor of 2.3, and
- 5) the heating rate of a factor of 3.3.

2.5.2.2 Transient

Figure 2-161 displays the power profile, percent full power as a function of time, for the NSS which employs either a composite fueled core or a graphite fueled core. Section

2.5.1.2 describes this profile and how the transient data provided in this section relates to the State Points indicated on the profile.

2.5.2.2.1 Radiation Environment

Figures 2-162 through 2-164 provide the integrating factors to convert fluxes, dose rates and heating rates to fluences, doses, and heat. Section 2.5.1.2.1 describes the manner in which these integrating factors are applied to radiation data to obtain integrated data over the reactor operating profile.

Table 2-66 presents radiation data for the NSS with a graphite fueled core at each of the selected locations defined by Figure 2-6 and Table 2-3 integrated over the power profile to the point in time at which each State Point occurs.

Table 2-66 presents the following information for each selected location and state point:

- 1) the gamma dose and associated uncertainty factor (See Section 2.4.4.2 for the details concerning the uncertainty factor),
- 2) the fast neutron fluence and associated uncertainty factor,
- 3) the intermediate neutron fluence and associated uncertainty factor,
- 4) the thermal neutron fluence and associated uncertainty factor, and
- 5) total heat deposition in selected materials and the associated uncertainty factor.

Section 2.5.1.2.1 describes the radiation data in more detail.

2.5.2.2.2 Total Heat Deposition by Region

Figure 2-165 presents the prompt total reactor power and the delayed total reactor power as a function of run time for the power profile defined in Figure 2-161. Figure 2-166 presents the power after shutdown due to delayed photons and beta particles as a function of time after shutdown for the NSS with either a graphite or composite fueled core.

Figures 2-162 through 2-164 provided the integrating factors to convert the total heating rate for a region or material in a region, as supplied by Table 2-65, to the total heat deposition at any point in time during the run profile defined by Figure 2-161. It should be noted that the neutron, prompt, and delayed integrating factors must be applied to the corresponding heating rate in Table 2-65.

2.5.2.2.3 Sources for the Nuclear Subsystem

The gamma ray source strength for the graphite core defined in Figure 2-14 is given in Table 2-67 for the times at which the State Points defined on Figure 2-161 occur. These data are presented in the standard WISDM 13 energy structure. Table 2-53 provides the factors to convert the gamma ray source terms provided at steady state conditions for the non-core regions of the NSS with a graphite fueled core to the source terms which occur at the State Points defined by Figure 2-161.

2.5.3 Activation Analysis

In addition to determining, at various times after shutdown, the photon dose rates external to the nuclear subsystem due to the fission product inventories in the reactor core, calculations were completed to include the effect on those dose rates of the activation of components in NSS regions external to the core.

The activation sources at various times following termination of a 60 minute full power reactor run were obtained via the ACT II code using the full power neutron flux levels from the DOT problem for the composite fueled reactor and the latest available information regarding basic material composition and impurities for each reactor region as input. These sources were then employed in the KAP-V point kernel code to calculate the photon dose rate on a 10 foot meridian ring at various times after shutdown.

The photon dose rates on the meridian ring due to fission product inventories, as well as activation sources, are summarized in Tables 2-68 through 2-70 for 30 minutes, 1 month, and 1 year after shutdown, respectively. Data are presented on the reactor axis as well as on the core midplane. From Tables 2-68 and 2-69, it is seen that at 30 minutes and

1 month following termination of the reactor run, the core source is the predominant contributor to the photon dose rates external to the NSS. However, as illustrated in Table 2-70, at 1 year after shutdown the activation sources do become a significant contributor (42% of total) to the photon radiation levels at the shielded end of the reactor. At 1 year after shutdown, the fission products still represent the dominant source for the core midplane.

2.6 REFERENCES

1. Aerojet Nuclear System Company, "CEI Specification for the NERVA 75K Full Flow Engine, " CP-90290,
2. L. I. Ortenberg, "The Evaluation of the WANL Integrated Nuclear, Radiation and Shielding Standard Design Method (WISDM)," WANL-TME-2689, to be published.
3. D. J. Hill, "DOT Model for the Radiation Analysis of the R-1 Preliminary Reference Design (WANL Dwg. 937J723), " DRM 52842 (September 14, 1970).
4. R. G. Soltesz, et.al, "Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation, " Volume 3, "Cross Section Generation and Data Processing Techniques, " WANL-PR-(LL)-034 (August, 1970).
5. G. Collier, et.al, "Second Version of the GAMBIT Code, " WANL-TME-1969 (November, 1969).
6. R. G. Soltesz, "Revised WANL ANISN Program User's Manual, " WANL-TMI-1967 (April, 1969).
7. R. K. Disney, et.al, "Synthesis of Calculational Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems, " Volume 7, NAGS, WANL-PR(LL)-010 (June, 1967).
8. R. K. Disney, and R. G. Soltesz, "User's Manual for the DOT-IIW Discrete Ordinates Transport Computer Code, " WANL-TME-1982 (December, 1969).
9. A. H. Fero, "CHEAPER User's Manual, " to be published.
10. R. G., Soltesz, "SNARE User's Manual, " to be published.
11. R. B. Fowler, et. al, "EXTERMINATOR-2, A Fortran IV Code for Solving Multigroup Neutron Diffusion Equations In Two Dimensions, " ORNL-4078 (April 1967).
12. V. S. Oblock and W. D. Rankin, "Control Vane Heating Measurements, " WANL-TME-1913 (September, 1970).
13. Unpublished internal memorandum.
14. R. K. Disney, et.al, "User's Manual for the MAP Radiation Transport Code, " WANL-TME-2706 (August, 1970).
15. R. K. Disney and M. A. Capo, "Synthesis of Calculational Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems, " Volume 4, KAPV Code, WANL-PR(LL)-010 (June, 1967).

2.6 REFERENCES (CONTINUED)

16. W. E. Selph and H. C. Claiborne, "Methods for Calculating Effects of Ducts, Access Ways and Holes in Radiation Shields, " ORNL-RSIC-20 (January, 1968).
17. M. A. Capo, et.al, "NRX-A6 Radiation Analysis Model, " WANL-TME-1700, (September, 1967).
18. V. S. Oblock, "Estimated Heating Rates at a Control Drum Actuator Operating in an Unshielded R-1 Reactor in Test Cell C, " DRM 52369 (May 7, 1970).
19. W. L. Knecht, et.al, "NRX-A6 Reactor Test Analysis Report, " WANL-TNR-223 (August, 1968).
20. A. Shalette, et.al, "NRX-A6 Nuclear Subsystem Thermal and Nuclear Design Data Book, " WANL-TME-1642 (November, 1967).
21. R. N. Nassano, et.al, "Radiation and Shielding Input to the Axial Optimization Trade Study, " DRM 52376 (June, 1970).
22. D. G. Bevard, "Reference R-1 Calculational Model Composite Core - Phase I Elements, " DRM 52772 (August, 1970).
23. D. G. Bevard, "Reference R-1 Calculational Model Graphite Core - Phase I Elements, " DRM 52774 (September, 1970).
24. M. A. Capo, "Computer Program Traceability for the R-1 Reference Design (939J723) Analysis, " DRM 52850, (December, 1970).
25. S. L. Anderson, et.al, "KAPV and DOT Nuclear Models for the R-1 Conceptual Design Analysis Reported in DRM 52379, " DRM 52840 (July, 1970).
26. S. L. Anderson and J. R. Magaw, "Nuclear Subsystem Shielding Provision Study Experiments, " WANL-TME-1912 (September, 1970).
27. D. J. Hill and W. D. Rankin, "Radiation Environment In Various Reactor Components, " WANL-TME-1914 (July, 1970).
28. D. J. Hill and W. D. Rankin, "Radiation Environment on CHESH Components, " WANL-TME-1915 (August, 1970).

2.6 REFERENCES (CONTINUED)

29. R. A. Schatz, "NSS Failure Mode Effects and Criticality Analysis - Preliminary R-202 Input in Support of Engine PDR, " DRM 53022 (August, 1970).
30. Letter from Dr. A. Foderaro to M. A. Capo, "Energy Distribution in U²³⁵ Thermal Fission, " (June, 1970).
31. Aerojet Nuclear Systems Company, "Shield System Trade Study, " Volume 2, S054-023 (July, 1970).

TABLE 2-1

RADIATION LEAKAGE LIMITS AT A PLANE
63 INCHES FORWARD OF THE CORE CENTER*

<u>TYPE OF RADIATION</u>	<u>RADIATION LEAKAGE LIMITS WITHIN PRESSURE VESSEL OUTSIDE RADIUS</u>
Gamma Carbon KERMA Rate	$1.8 \times 10^7 \text{ Rad(C)}/\text{hr}$
Fast Neutron Flux ($E > 1.0 \text{ MeV}$)	$2.0 \times 10^{12} \text{ n/cm}^2\text{-sec}$
Intermediate Neutron Flux ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$)	$3.0 \times 10^{12} \text{ n/cm}^2\text{-sec}$
Thermal Neutron Flux ($E < 0.4 \text{ eV}$)	$6.0 \times 10^{11} \text{ n/cm}^2\text{-sec}$

* For convenience, this plane is commonly referred to as the "tangent plane" in this report.

TABLE 2-2
RADIATION LEAKAGE LIMITS AT CRITICAL COMPONENT LOCATIONS
(FULL POWER OPERATING RADIATION LEVELS)

	<u>Location</u>		<u>Gamma KERMA Rate Rads (Carbon)/hr</u>	<u>Neutron Flux (n/cm²-sec)</u>		
	<u>Radius (inches)</u>	<u>Distance From Core Center (in.)</u>		<u>Thermal E<0.4 eV</u>	<u>Intermediate 0.4 eV<E<1.0 MeV</u>	<u>Fast E>1.0 MeV</u>
Lower Thrust Structure - 1	19.5	65.5	2.1×10^7	---	---	2.8×10^{14}
Lower Thrust Structure - 2	19.5	67.5	1.8×10^7	---	---	2.8×10^{14}
Lower Thrust Structure - 3	19.5	76.5	1.8×10^7	---	---	2.8×10^{14}
Turbopump - 1	36.0	107.0	7.0×10^6	1.0×10^{12}	5.0×10^{12}	1.0×10^{12}

TABLE 2-3

SELECTED LOCATIONS INTERNAL TO THE NSS FOR RADIATION ENVIRONMENT DATA*

(Composite Core)

Location Number	Description		Coordinates	
			Radius (inches)	Reactor Station (inches)
1	Reactor Centerline	Aft Surface of CHESH	0.000	54.131
2	Reactor Centerline	Core Exit	0.000	52.000
3	Reactor Centerline	Core Midplane	0.000	26.000
4	Reactor Centerline	Core Inlet	0.000	0.000
5	Reactor Centerline	Aft Face of Support Plate	0.000	- 3.500
6	Reactor Centerline	Forward Face of Support Plate	0.000	- 9.500
7	Reactor Centerline	Aft Face of Central Shield Plate	0.000	-10.830
8	Reactor Centerline	Aft Face of BATH Central Shield	0.000	-11.830
9	Reactor Centerline	BATH Central Shield Midplane	0.000	-17.380
10	Reactor Centerline	Forward Face of BATH Central Shield	0.000	-22.930
11	Reactor Centerline	Forward Face of Lead Central Shield	0.000	-24.230
12	Reactor Centerline	Aft Face of Slow Baffle II	0.000	-29.630
13	Reactor Centerline	Forward Face of Flow Baffle II	0.000	-30.750
14	Reactor Centerline	Aft Surface of the Pressure Vessel Dome	0.000	-36.040
15	Reactor Centerline	Forward Surface of the Vessel Dome	0.000	-37.000
16	Core Edge	Aft Surface of CHESH	18.657	54.131
17	Core Edge	Core Exit	18.657	52.000
18	Core Edge	Core Midplane	18.657	26.000
19	Core Edge	Core Inlet	18.657	0.000
20	Outer Radius of Core Periphery	Core Exit	19.476	52.000
21	Outer Radius of Core Periphery	Core Midplane	19.476	26.000
22	Outer Radius of Core Periphery	Core Inlet	19.476	0.000

*See Figure 2-6 for locations.

TABLE 2-3 (Continued)

SELECTED LOCATIONS INTERNAL TO THE NSS FOR RADIATION ENVIRONMENT DATA*

(Composite Core)

Location Number	Description		Coordinates	
			Radius (inches)	Reactor Station (inches)
23	Inner Radius of Structure	Aft End of Structure	21.115	55.200
24	Inner Radius of Structure	Core Midplane	21.115	26.000
25	Inner Radius of Structure	Forward End of Structure	21.115	-3.500
26	Outer Radius of Structure	Aft End of Structure	22.215	55.200
27	Inner Radius of Reflector	Aft End of Reflector	22.215	52.000
28	Inner Radius of Reflector	Core Midplane	22.215	26.000
29	Inner Radius of Reflector	Forward End of Reflector	22.215	0.000
30	Outer Radius of Reflector	Aft End of Reflector	27.000	52.000
31	Outer Radius of Reflector	Core Midplane	27.000	26.000
32	Outer Radius of Reflector	Forward End of Reflector	27.000	0.000
33	Pressure Vessel Side Surface	Pressure Vessel - Nozzle Mating Flange	27.890	63.200
34	Pressure Vessel Side Surface	Core Midplane	27.890	26.000

*See Figure 2-6 for locations.

TABLE 2-3 (Continued)

SELECTED LOCATIONS INTERNAL TO THE NSS FOR RADIATION ENVIRONMENT DATA*
(Composite Core)

Location Number	Description	Coordinates	
		Radius (inches)	Reactor Station (inches)
35	Pressure Vessel Side Surface	27.890	-37.000
36	Outer Radius of Support Plate	22.215	- 3.500
37	Outer Radius of Support Plate	22.215	- 9.500
38	Outer Radius of BATH Central Shield	19.476	-10.830
39	Outer Radius of BATH Central Shield	19.476	-11.830
40	Outer Radius of BATH Central Shield	19,476	-17.380
41	Outer Radius of BATH Central Shield	19.476	-22.930
42	Outer Radius of BATH Central Shield	19.476	-24.230
43	Outer Radius of BATH Central Shield	19.476	-29.630

*See Figure 2-6 for locations.

TABLE 2-3 (Continued)

SELECTED LOCATIONS INTERNAL TO THE NSS FOR RADIATION ENVIRONMENT DATA*
(Composite Core)

Location Number	Description	Coordinates	
		Radius (inches)	Reactor Station (inches)
44	Outer Radius of BATH Central Shield	19.476	-30.750
45	Inner Radius of BATH Peripheral Shield I	20.505	-11.040
46	Inner Radius of BATH Peripheral Shield I	20.505	-22.140
47	Inner Radius of BATH Peripheral Shield I	20.505	-23.440
48	Outer Radius of BATH Peripheral Shield I	22.038	-11.040
49	Outer Radius of BATH Peripheral Shield I	22.038	-22.140
50	Outer Radius of BATH Peripheral Shield I	22.038	-23.440
51	Outer Radius of BATH Peripheral Shield II	23.625	-11.040

*See Figure 2-6 for locations.

TABLE 2-3 (Continued)

SELECTED LOCATIONS INTERNAL TO THE NSS FOR RADIATION ENVIRONMENT DATA*
(Composite Core)

Location Number	Description		Coordinates	
			Radius (inches)	Reactor Station (inches)
52	Outer Radius of BATH Peripheral Shield II	Forward Face of BATH Peripheral Shield II	23.625	-22.140
53	Outer Radius of BATH Peripheral Shield II	Forward Face of Lead Peripheral Shield II	23.625	-23.440
54	Outer Radius of BATH Peripheral Shield III	Aft Face of Peripheral Shield	25.375	-11.040
55	Outer Radius of BATH Peripheral Shield III	Forward Face of BATH Peripheral Shield III	25.375	-22.140
56	Outer Radius of BATH Peripheral Shield	Forward Face of Lead Peripheral Shield III	25.375	-23.440
57	Outer Radius of BATH Peripheral Shield IV	Aft Face of Peripheral Shield	27.000	-11.040
58	Outer Radius of BATH Peripheral Shield IV	Forward Face of BATH Peripheral Shield IV	27.000	-20.640
59	Outer Radius of BATH Peripheral Shield IV	Forward Face of Lead Peripheral Shield IV	27.000	-21.940

*See Figure 2-6 for locations

TABLE 2-3 (Continued)

SELECTED LOCATIONS INTERNAL TO THE NSS FOR RADIATION ENVIRONMENT DATA*

Location Number	Description	Coordinates	
		Radius (inches)	Reactor Station (inches)
	FOR GRAPHITE CORE ONLY		
60	Reactor Centerline	0.000	51.630
61	Core Edge, Forward Face of Unfueled Tips	18.657	51.630

*See Figure 2-6 for locations.

TABLE 2-4
R-Z MODEL REGION DESCRIPTION

Boundaries						Engineering Critical Component
Region* Number	Description	Reactor	Station	Radius		
		Forward	Aft	Inner	Outer	
1	Core-1	0.00"	52.000"	0.000"	3.163"	FE
2	Core-2	0.00"	52.000"	3.163"	5.535"	FE
3	Core-3	0.00"	52.000"	5.535"	7.028"	FE
4	Core-4	0.00"	52.000"	7.028"	7.907"	FE
5	Core-5	0.00"	52.000"	7.907"	8.911"	FE
6	Core-6	0.00"	52.000"	8.911"	9.978"	FE
7	Core-7	0.00"	52.000"	9.978"	10.834"	FE
8	Core-8	0.00"	52.000"	10.834"	11.587"	FE
9	Core-9	0.00"	52.000"	11.587"	12.743"	FE
10	Core-10	0.00"	52.000"	12.743"	14.331"	FE
11	Core-11	0.00"	52.000"	14.331"	15.710"	FE
12	Core-12	0.00"	52.000"	15.710"	16.267"	FE
13	Core-13	0.00"	52.000"	16.267"	16.833"	FE
14	Core-14	0.00"	52.000"	16.833"	17.729"	FE
15	Core-15	0.00"	52.000"	17.729"	18.657"	FE
16	Core Periphery	0.00"	52.000"	18.657"	19.476"	CP
17	Lateral Support	0.00"	52.000"	19.476"	21.115"	RA
18	Structure	-3.50"	55.200"	21.115"	22.215"	RA
19	Outer Reflector-A	0.00"	52.000"	22.215"	23.411"	RA
20	Ring R _x	0.00"	52.000"	23.411"	23.515"	RA
21	Outer Reflector-B	0.00"	52.000"	23.515"	27.000"	RA
22	Pressure Vessel Side-A	-9.500"	63.200"	27.000"	27.890"	NSS

*Refer to Figures 2-11 and 2-12 for region locations.

TABLE 2-4
(Continued)
R-Z MODEL REGION DESCRIPTION

Boundaries							Engineering Critical Component
Region* Number	Description	Reactor	Station	Radius			
		Forward	Aft	Inner	Outer		
23	CHESH	52.000"	54.131"	0.000"	18.657"	CH	
24	Nozzle Chamber	Note 1	Note 1	Note 1	Note 1	NSS	
25	Nozzle Barrel	55.200"	63.200"	21.115"	22.215"	NSS	
26	Aft Reflector Hardware	52.000"	53.938"	22.215"	27.000"	RA	
27	Aft Reflector Plenum	53.938"	63.200"	22.215"	27.000"	NSS	
28	Core Plenum	- 3.500"	0.000"	0.000"	18.657"	SP/P	
29	Support Plate	- 9.500"	-3.500"	0.000"	22.215"	SP/P	
30	Lateral Support - Forward	- 3.500"	0.00"	18.657"	21.115"	SP/P	
31	Forward Reflector Hardware I	- 9.500"	0.00"	22.215"	27.000"	RA	
32	Support Plate Plenum	-10.830"	-9.500"	0.000"	19.476"	SP/P	
33	Instrumentation Ring	-11.040"	-9.500"	19.476"	22.215"	SP/P	
34	Forward Reflector Hardware II	-11.040"	-9.500"	22.215"	27.000"	SP/P	
35	Aft Central Shield Plate	-11.830"	-10.830"	0.000"	19.476"	IS	
36	BATH Central Shield	-22.930"	-11.830"	0.000"	19.476"	IS	
37	Flow Baffle I	-30.750"	-11.040"	19.476"	20.505"	SP/P	
38	BATH Peripheral Shield I	-22.140"	-11.040"	20.505"	22.038"	IS	
39	BATH Peripheral Shield II	-22.140"	-11.040"	22.038"	23.625"	IS	
40	BATH Peripheral Shield III	-22.140"	-11.040"	23.625"	25.375"	IS	
41	BATH Peripheral Shield IV	-20.640"	-11.040"	25.375"	27.000"	IS	
42	Pressure Vessel Side-B	-37.000"	- 9.500"	27.000"	27.890"	NSS	
43	Lead Central Shield	-24.230"	-22.930"	0.000"	19.476"	IS	
44	Lead Peripheral Shield I	-23.440"	-22.140"	20.505"	22.038"	IS	

*Refer to Figures 2-11 and 2-12 for region locations.

TABLE 2-4
(Continued)
R-Z MODEL REGION DESCRIPTION

		Boundaries				Engineering Critical Component
Region* Number		Reactor	Station	Radius		
		Forward	Aft	Inner	Outer	
45	Lead Peripheral Shield II	-23.440"	-22.140"	22.038"	23.625"	IS
46	Lead Peripheral Shield III	-23.440"	-22.140"	23.625"	25.375"	IS
47	Lead Peripheral Shield IV	-21.940"	-20.640"	25.375"	27.000"	IS
48	Peripheral Shield Plate	-24.190"	-23.440"	20.505"	25.375"	IS
49	Shield Plenum	-29.630"	-24.230"	0.000"	19.476"	IS
50	Flow Baffle II	-30.750"	-29.630"	0.000"	19.476"	SP/P
51	Central Dome Plenum	-36.040"	-30.750"	0.000"	11.032"	SP/P
52	Peripheral Dome Plenum	Note 2	Note 2	Note 2	Note 2	SP/P
53	Pressure Vessel Dome	-37.000"	-36.040"	0.000"	27.000"	NSS

Note 1:

24	Nozzle Chamber-A	54.131"	63.200"	0.000"	18.657"	NSS
24	Nozzle Chamber-B	52.000"	63.200"	18.657"	21.115"	NSS

Note 2:

52	Peripheral Dome Plenum-A	-36.040"	-30.750"	11.032"	20.505"	SP/P
52	Peripheral Dome Plenum-B	-36.040"	-24.190"	20.505"	25.375"	SP/P
52	Peripheral Dome Plenum-C	-36.040"	-21.940"	25.375"	27.000"	SP/P

*Refer to Figures 2-11 and 2-12 for region locations.

TABLE 2-5
MESH LINE COORDINATES FOR THE NEUTRON BASIC REACTOR GEOMETRY DOT PROBLEM

Mesh Line No.	RADIUS	R MIDPOINT	HEIGHT	H MIDPOINT
1	0.	1.5000000F+00	-2.0000000F+02	-1.4941100F+02
2	3.0000000E+00	4.5000000F+00	-1.4962000F+02	-1.4943500F+02
3	6.0000000E+00	7.0165000F+00	-1.4925000F+02	-1.4946000F+02
4	8.0330000E+00	9.5165000F+00	-1.4887000F+02	-1.4848000F+02
5	1.1000000E+01	1.2529000F+01	-1.4849000F+02	-1.4830500F+02
6	1.4058000E+01	1.5029000F+01	-1.4812000F+02	-1.4793000F+02
7	1.6000000E+01	1.6925000F+01	-1.4774000F+02	-1.4755000F+02
8	1.7850000E+01	1.8425000F+01	-1.4736000F+02	-1.4717500F+02
9	1.9000000E+01	1.9542000F+01	-1.4699000F+02	-1.4680000F+02
10	2.0084000E+01	2.0792000F+01	-1.4661000F+02	-1.4642000F+02
11	2.1500000E+01	2.2066500F+01	-1.4623000F+02	-1.4604000F+02
12	2.2633000E+01	2.3316500F+01	-1.4586000F+02	-1.4567000F+02
13	2.4000000E+01	2.4672000F+01	-1.4548000F+02	-1.4529000F+02
14	2.5344000E+01	2.5822000F+01	-1.4510000F+02	-1.4491000F+02
15	2.6300000E+01	2.6909500F+01	-1.4472000F+02	-1.4453000F+02
16	2.7519000E+01	2.7770150F+01	-1.4435000F+02	-1.4415000F+02
17	2.8021300E+01	2.8726150F+01	-1.4397000F+02	-1.4377000F+02
18	2.9431000E+01	3.0215500F+01	-1.4359000F+02	-1.4339000F+02
19	3.1000000E+01	3.1684000F+01	-1.4322000F+02	-1.4302000F+02
20	3.2368000E+01	3.3184000F+01	-1.4284000F+02	-1.4264000F+02
21	3.4000000E+01	3.5200000F+01	-1.4246000F+02	-1.4227000F+02
22	3.6400000E+01	3.7200000F+01	-1.4209000F+02	-1.4190000F+02
23	3.8000000E+01	3.8951500F+01	-1.4171000F+02	-1.4152000F+02
24	3.9903000E+01	4.0201500F+01	-1.4133000F+02	-1.4114500F+02
25	4.0500000E+01	4.0909000F+01	-1.4096000F+02	-1.4077000F+02
26	4.1314000E+01	4.1659000F+01	-1.4057000F+02	-1.4038500F+02
27	4.2000000E+01	4.2378500F+01	-1.4019000F+02	-1.4000000F+02
28	4.2757000E+01	4.3378500F+01	-1.3980000F+02	-1.3961500F+02
29	4.4000000E+01	4.4516000F+01	-1.3943000F+02	-1.3924000F+02
30	4.5032000E+01	4.5516000F+01	-1.3905000F+02	-1.3886000F+02
31	4.6000000E+01	4.6494000F+01	-1.3868000F+02	-1.3848000F+02
32	4.7388000E+01	4.8428500F+01	-1.3831000F+02	-1.3811000F+02
33	4.9469000E+01	5.0775850F+01	-1.3794000F+02	-1.3774000F+02
34	5.2082700E+01	5.2857350F+01	-1.3757000F+02	-1.3737000F+02
35	5.3632000E+01	5.4804000F+01	-1.3720000F+02	-1.3700000F+02
36	5.5976000E+01	5.6201000F+01	-1.3683000F+02	-1.3663000F+02
37	5.6426000E+01	5.6963000F+01	-1.3646000F+02	-1.3626000F+02
38	5.7500000E+01	5.8250000F+01	-1.3609000F+02	-1.3589000F+02
39	5.9000000E+01	5.9232000F+01	-1.3572000F+02	-1.3552000F+02
40	5.9464000E+01	5.9482000F+01	-1.3535000F+02	-1.3515000F+02
41	5.9500000E+01	5.9550000F+01	-1.3498000F+02	-1.3478000F+02
42	5.9600000E+01	5.9664500F+01	-1.3461000F+02	-1.3441000F+02
43	5.9729000E+01	5.9848000F+01	-1.3424000F+02	-1.3404000F+02
44	6.0007000E+01	6.0503500F+01	-1.3387000F+02	-1.3367000F+02
45	6.1000000E+01	6.2000000F+01	-1.3350000F+02	-1.3330000F+02
46	6.3000000E+01	6.3726000F+01	-1.3313000F+02	-1.3293000F+02
47	6.4452000E+01	6.5726000F+01	-1.3276000F+02	-1.3256000F+02
48	6.7000000F+01	6.7740000F+01	-1.3239000F+02	-1.3219000F+02
49	6.8580000E+01	6.9290000F+01	-1.3202000F+02	-1.3182000F+02
50	7.0000000E+01	7.0423000E+01	-9.7320000E+01	-9.3010000F+01
51	7.0846000E+01		-8.8700000E+01	-8.4395000F+01
52			-8.0090000E+01	-7.5780000F+01
53			-7.1470000F+01	-6.7165000F+01
54			-6.2860000E+01	-5.8550000F+01
55			-5.4240000E+01	-4.9935000F+01
56			-4.5630000E+01	-4.1315000F+01
57			-3.7000000E+01	-3.2690000F+01
58			-3.3000000F+01	-3.2000000F+01
59			-3.1000000E+01	-3.0195000F+01
60			-2.9390000F+01	-2.8919000F+01
61			-2.8449000E+01	-2.7549000F+01
62			-2.6650000E+01	-2.5047500F+01
63			-2.3525000E+01	-2.3270000F+01
64			-2.3035000E+01	-2.1677500F+01
65			-2.0320000E+01	-1.7190000F+01
66			-1.4060000E+01	-1.1815000F+01
67			-9.5700000F+00	-7.3250000F+00
68			-5.0000000F+00	-3.8100000F+00
69			-2.5400000E+00	-1.2700000F+00
70			0.	

TABLE 2-6
MESH LINE COORDINATES FOR THE NEUTRON SHIELD GEOMETRY DOT PROBLEM

Mesh Line No	RADIUS	R MINPOINT	HEIGHT	X MINPOINT
1	0.	1.5000000E+00	-2.5450000E+02	-2.5175400E+02
2	3.0000000E+00	4.5000000E+00	-2.5100000E+02	-2.5257450E+02
3	6.0000000E+00	7.0165000E+00	-2.5200000E+02	-2.5030450E+02
4	9.0370000E+00	9.4145000E+00	-2.4870000E+02	-2.4690500E+02
5	1.1000000E+01	1.2520000E+01	-2.4511000E+02	-2.4371500E+02
6	1.4050000E+01	1.5020000E+01	-2.4152000E+02	-2.4007650E+02
7	1.8000000E+01	1.8725000E+01	-2.3803000E+02	-2.3762650E+02
8	1.7850000E+01	1.8425000E+01	-2.3620000E+02	-2.3620400E+02
9	1.9000000E+01	1.9540000E+01	-2.3578000E+02	-2.3394400E+02
10	2.0084000E+01	2.0720000E+01	-2.3210000E+02	-2.3051000E+02
11	2.1500000E+01	2.2066500E+01	-2.2897000E+02	-2.2737000E+02
12	2.2673000E+01	2.3116500E+01	-2.2574000E+02	-2.2416000E+02
13	2.4000000E+01	2.4472000E+01	-2.2254000E+02	-2.2272600E+02
14	2.5344000E+01	2.5827000E+01	-2.2207000E+02	-2.2207150E+02
15	2.6700000E+01	2.6909500E+01	-2.2197100E+02	-2.2137050E+02
16	2.7519000E+01	2.7770150E+01	-2.2075000E+02	-2.2040750E+02
17	2.8721000E+01	2.8721450E+01	-2.2006500E+02	-2.1974750E+02
18	2.9431000E+01	2.9155000E+01	-2.1943000E+02	-2.1910000E+02
19	3.1000000E+01	3.1846000E+01	-2.1877000E+02	-2.1846000E+02
20	3.2349000E+01	3.3184000E+01	-2.1835000E+02	-2.1817500E+02
21	3.4000000E+01	3.5200000E+01	-2.1792000E+02	-2.1771000E+02
22	3.6400000E+01	3.7200000E+01	-2.1750000E+02	-2.1725000E+02
23	3.8000000E+01	3.8951500E+01	-2.1700000E+02	-2.1684500E+02
24	3.9903000E+01	4.0291500E+01	-2.1676300E+02	-2.1650950E+02
25	4.0500000E+01	4.0800000E+01	-2.1625600E+02	-2.1607000E+02
26	4.1310000E+01	4.1850000E+01	-2.1590000E+02	-2.1571000E+02
27	4.2000000E+01	4.2374000E+01	-2.1552000E+02	-2.1534000E+02
28	4.2757000E+01	4.3178500E+01	-2.1516000E+02	-2.1498000E+02
29	4.4000000E+01	4.4514000E+01	-2.1480000E+02	-2.1461000E+02
30	4.5032000E+01	4.5516000E+01	-2.1442000E+02	-2.1423500E+02
31	4.6000000E+01	4.6494000E+01	-2.1405000E+02	-2.1387000E+02
32	4.7380000E+01	4.8428500E+01	-2.1367000E+02	-2.1351500E+02
33	4.9449000E+01	5.0775000E+01	-2.1331000E+02	-2.1317500E+02
34	5.2082700E+01	5.2857350E+01	-2.1295000E+02	-2.1275150E+02
35	5.3672000E+01	5.4804000E+01	-2.1255000E+02	-2.1233000E+02
36	5.5976000E+01	5.6701000E+01	-2.1217000E+02	-2.1190000E+02
37	5.6426000E+01	5.6943000E+01	-2.1179000E+02	-2.1160000E+02
38	5.7500000E+01	5.8250000E+01	-2.1141000E+02	-2.1122000E+02
39	5.9000000E+01	5.9727000E+01	-2.1103000E+02	-2.1084000E+02
40	5.9446000E+01	5.9482000E+01	-2.1065000E+02	-2.1046000E+02
41	5.9500000E+01	5.9550000E+01	-2.1027000E+02	-2.1008000E+02
42	5.9400000E+01	5.9444500E+01	-2.0989000E+02	-2.0970000E+02
43	5.9729000E+01	5.9888000E+01	-2.0951000E+02	-2.0932000E+02
44	6.0007000E+01	6.0507500E+01	-2.0913000E+02	-2.0894000E+02
45	6.1000000E+01	6.2000000E+01	-2.0875000E+02	-2.0856000E+02
46	6.3000000E+01	6.3726000E+01	-2.0837000E+02	-2.0818000E+02
47	6.4452000E+01	6.4726000E+01	-2.0799000E+02	-2.0779000E+02
48	6.7000000E+01	6.7700000E+01	-2.0761000E+02	-2.0742000E+02
49	6.8500000E+01	6.9200000E+01	-2.0723000E+02	-2.0704000E+02
50	7.0000000E+01	7.0423000E+01	-2.0685000E+02	-2.0665500E+02
51	7.0846000E+01		-2.0646000E+02	-2.0627000E+02
52			-2.0608000E+02	-2.0589000E+02
53			-2.0570000E+02	-2.0551000E+02
54			-2.0532000E+02	-2.0513000E+02
55			-2.0494000E+02	-2.0475000E+02
56			-2.0456000E+02	-2.0437000E+02
57			-2.0418000E+02	-2.0399000E+02
58			-2.0380000E+02	-2.0361000E+02
59			-2.0342000E+02	-2.0323000E+02
60			-2.0304000E+02	-2.0285000E+02
61			-2.0266000E+02	-2.0247000E+02
62			-2.0228000E+02	-2.0209000E+02
63			-2.0190000E+02	-2.0171000E+02
64			-2.0152000E+02	-2.0133000E+02
65			-2.0114000E+02	-2.0095000E+02
66			-2.0076000E+02	-2.0057000E+02
67			-2.0038000E+02	-2.0019000E+02
68			-2.0000000E+02	-1.9981000E+02
69			-1.9962000E+02	-1.9943000E+02
70			-1.9925000E+02	-1.9905000E+02
71			-1.9887000E+02	-1.9867000E+02
72			-1.9849000E+02	-1.9829000E+02
73			-1.9812000E+02	-1.9791000E+02
74			-1.9774000E+02	-1.9753000E+02
75			-1.9736000E+02	-1.9715000E+02
76			-1.9700000E+02	-1.9680000E+02
77			-1.9664000E+02	-1.9644000E+02
78			-1.9628000E+02	-1.9608500E+02
79			-1.9592000E+02	-1.9573000E+02
80			-1.9556000E+02	-1.9537000E+02
81			-1.9520000E+02	-1.9501000E+02
82			-1.9484000E+02	-1.9465000E+02
83			-1.9448000E+02	-1.9429000E+02
84			-1.9412000E+02	-1.9393000E+02
85			-1.9376000E+02	-1.9357000E+02
86			-1.9340000E+02	-1.9321000E+02
87			-1.9304000E+02	-1.9285000E+02
88			-1.9268000E+02	-1.9249000E+02
89			-1.9232000E+02	-1.9213000E+02
90			-1.9196000E+02	-1.9177000E+02
91			-1.9160000E+02	-1.9141000E+02
92			-1.9124000E+02	-1.9105000E+02
93			-1.9088000E+02	-1.9069000E+02
94			-1.9052000E+02	-1.9033000E+02
95			-1.9016000E+02	-1.9007000E+02
96			-1.8980000E+02	-1.8971000E+02
97			-1.8944000E+02	-1.8935000E+02
98			-1.8908000E+02	-1.8899000E+02
99			-1.8872000E+02	-1.8863000E+02
100			-1.8836000E+02	-1.8827000E+02
101			-1.8800000E+02	-1.8791000E+02

TABLE 2-7
MESH LINE COORDINATES FOR THE PHOTON BASIC REACTOR GEOMETRY DOT PROBLEM

Mesh Line No.	RADIUS	R-MIDPOINT	HEIGHT	HEIGHT
1	0.	1.5000000E+00	-2.0000000E+00	-1.4943300E+02
2	3.0000000E+00	4.5000000E+00	-1.0000000E+00	-1.4943300E+02
3	6.0000000E+00	7.0165300E+00	-1.0774000E+00	-1.4717000E+02
4	8.0330000E+00	9.5145000E+00	-1.0661000E+00	-1.4604000E+02
5	1.1000000E+01	1.2549000E+01	-1.0549000E+00	-1.4491000E+02
6	1.4050000E+01	1.5024000E+01	-1.0435000E+00	-1.4378000E+02
7	1.6000000E+01	1.6925000E+01	-1.0322000E+00	-1.4265000E+02
8	1.7850000E+01	1.8420000E+01	-1.0210000E+00	-1.4152000E+02
9	1.9000000E+01	1.9542000E+01	-1.0133000E+00	-1.4039000E+02
10	2.0000000E+01	2.0170000E+01	-1.0070000E+00	-1.3926000E+02
11	2.1500000E+01	2.2065000E+01	-1.0000000E+00	-1.3813000E+02
12	2.2530000E+01	2.3116500E+01	-1.0000000E+00	-1.3699000E+02
13	2.4000000E+01	2.4672000E+01	-1.0000000E+00	-1.3586000E+02
14	2.5344000E+01	2.5422000E+01	-1.0000000E+00	-1.3473000E+02
15	2.6300000E+01	2.5945000E+01	-1.0000000E+00	-1.3360000E+02
16	2.7514000E+01	2.7770100E+01	-1.0000000E+00	-1.3247000E+02
17	2.8021300E+01	2.8726100E+01	-1.0000000E+00	-1.3134000E+02
18	2.9131000E+01	3.0215500E+01	-1.0000000E+00	-1.3021000E+02
19	3.1000000E+01	3.1544000E+01	-1.0000000E+00	-1.2908000E+02
20	3.2500000E+01	3.3104000E+01	-1.0000000E+00	-1.2795000E+02
21	3.4000000E+01	3.5200000E+01	-1.0000000E+00	-1.2682000E+02
22	3.6000000E+01	3.7200000E+01	-1.0000000E+00	-1.2569000E+02
23	3.8000000E+01	3.8945000E+01	-1.0000000E+00	-1.2456000E+02
24	3.9500000E+01	4.0213500E+01	-1.0000000E+00	-1.2343000E+02
25	4.0500000E+01	4.0940000E+01	-1.0000000E+00	-1.2230000E+02
26	4.1310000E+01	4.1690000E+01	-1.0000000E+00	-1.2117000E+02
27	4.2000000E+01	4.2378500E+01	-1.0000000E+00	-1.2004000E+02
28	4.2757000E+01	4.3374500E+01	-1.0000000E+00	-1.1891000E+02
29	4.4000000E+01	4.4516000E+01	-1.0000000E+00	-1.1778000E+02
30	4.5032000E+01	4.5516000E+01	-1.0000000E+00	-1.1665000E+02
31	4.6000000E+01	4.6440000E+01	-1.0000000E+00	-1.1552000E+02
32	4.7380000E+01	4.8428500E+01	-1.0000000E+00	-1.1439000E+02
33	4.9469000E+01	5.0775500E+01	-1.0000000E+00	-1.1326000E+02
34	5.2182700E+01	5.2447350E+01	-9.7320000E+01	-9.3010000E+01
35	5.3432000E+01	5.4140000E+01	-8.4700000E+01	-8.4310000E+01
36	5.5760000E+01	5.6201000E+01	-8.0000000E+01	-7.5700000E+01
37	5.6424000E+01	5.6473000E+01	-7.1470000E+01	-6.7145000E+01
38	5.7500000E+01	5.8250000E+01	-6.2460000E+01	-5.8550000E+01
39	5.9000000E+01	5.9232000E+01	-5.4240000E+01	-4.9935000E+01
40	5.9440000E+01	5.9440000E+01	-4.6430000E+01	-4.1315000E+01
41	5.9500000E+01	5.9500000E+01	-3.7000000E+01	-3.2000000E+01
42	5.9600000E+01	5.9664500E+01	-3.3000000E+01	-2.2000000E+01
43	5.9725000E+01	5.9664500E+01	-3.1000000E+01	-1.2000000E+01
44	6.0007000E+01	6.0503500E+01	-2.0300000E+01	-2.6919000E+01
45	6.1000000E+01	6.2000000E+01	-2.0448000E+01	-2.7549000E+01
46	6.3000000E+01	6.4726000E+01	-2.6650000E+01	-2.5037500E+01
47	6.4452000E+01	6.5726000E+01	-2.3525000E+01	-2.3240000E+01
48	6.7000000E+01	6.7743000E+01	-2.1350000E+01	-2.1617500E+01
49	6.8500000E+01	6.9200000E+01	-2.0320000E+01	-1.7140000E+01
50	7.0000000E+01	7.0423000E+01	-1.4000000E+01	-1.1815000E+01
51	7.0846000E+01		-9.5700000E+00	-7.3250000E+00
52			-5.0000000E+00	-3.8130000E+00
53			-2.5400000E+00	-1.2700000E+00
54			0.	

TABLE 2-8

MESH LINE COORDINATES FOR THE PHOTON SHIELD GEOMETRY DOT PROBLEM

Mesh Line No.	RADIUS	R MIDPOINT	HEIGHT	H MIDPOINT
1	0.	1.500000E+00	-2.545000E+02	-2.517540E+02
2	3.000000E+00	4.500000E+00	-2.530000E+02	-2.525345E+02
3	6.000000E+00	7.016530E+00	-2.520000E+02	-2.503445E+02
4	8.033000E+00	9.516530E+00	-2.487000E+02	-2.464050E+02
5	1.100000E+01	1.252400E+01	-2.451100E+02	-2.433150E+02
6	1.405000E+01	1.502400E+01	-2.415200E+02	-2.400765E+02
7	1.700000E+01	1.672530E+01	-2.385300E+02	-2.376265E+02
8	1.785000E+01	1.802400E+01	-2.360200E+02	-2.362040E+02
9	1.900000E+01	1.952400E+01	-2.357500E+02	-2.339440E+02
10	2.100000E+01	2.079200E+01	-2.321000E+02	-2.305100E+02
11	2.150000E+01	2.206450E+01	-2.280200E+02	-2.273300E+02
12	2.263300E+01	2.331450E+01	-2.257400E+02	-2.241400E+02
13	2.400000E+01	2.467200E+01	-2.225800E+02	-2.223260E+02
14	2.534400E+01	2.542200E+01	-2.220120E+02	-2.227215E+02
15	2.630000E+01	2.690450E+01	-2.210710E+02	-2.213605E+02
16	2.751900E+01	2.777010E+01	-2.207500E+02	-2.204075E+02
17	2.802130E+01	2.872410E+01	-2.200650E+02	-2.197475E+02
18	2.943100E+01	3.021550E+01	-2.194300E+02	-2.191000E+02
19	3.100000E+01	3.184000E+01	-2.187700E+02	-2.185600E+02
20	3.236000E+01	3.318400E+01	-2.183500E+02	-2.177300E+02
21	3.400000E+01	3.520000E+01	-2.178000E+02	-2.168815E+02
22	3.640000E+01	3.720000E+01	-2.167030E+02	-2.164075E+02
23	3.800000E+01	3.951500E+01	-2.162500E+02	-2.158800E+02
24	3.990300E+01	4.020150E+01	-2.154700E+02	-2.151600E+02
25	4.050000E+01	4.090000E+01	-2.148000E+02	-2.142450E+02
26	4.131800E+01	4.165000E+01	-2.136900E+02	-2.133215E+02
27	4.200000E+01	4.237850E+01	-2.129530E+02	-2.123715E+02
28	4.275700E+01	4.337850E+01	-2.117900E+02	-2.110300E+02
29	4.400000E+01	4.451800E+01	-2.102700E+02	-2.095100E+02
30	4.503200E+01	4.551600E+01	-2.087500E+02	-2.074000E+02
31	4.600000E+01	4.664400E+01	-2.084500E+02	-2.068500E+02
32	4.738800E+01	4.842850E+01	-2.053200E+02	-2.045500E+02
33	4.946400E+01	5.077585E+01	-2.034000E+02	-2.030400E+02
34	5.208270E+01	5.285735E+01	-2.022400E+02	-2.017100E+02
35	5.363200E+01	5.480400E+01	-2.011400E+02	-2.005700E+02
36	5.597600E+01	5.620100E+01	-2.000000E+02	-1.994350E+02
37	5.662800E+01	5.691300E+01	-1.988700E+02	-1.983050E+02
38	5.750000E+01	5.825000E+01	-1.977400E+02	-1.971700E+02
39	5.900000E+01	5.923200E+01	-1.966100E+02	-1.960400E+02
40	5.946400E+01	5.942200E+01	-1.954400E+02	-1.949150E+02
41	5.950000E+01	5.955000E+01	-1.943500E+02	-1.937850E+02
42	5.960000E+01	5.966450E+01	-1.932200E+02	-1.926550E+02
43	5.972400E+01	5.966800E+01	-1.920900E+02	-1.915250E+02
44	6.000700E+01	6.050350E+01	-1.909600E+02	-1.903950E+02
45	6.100000E+01	6.200000E+01	-1.905700E+02	-1.900300E+02
46	6.300000E+01	6.372800E+01	-1.895000E+02	-1.894345E+02
47	6.445200E+01	6.572800E+01	-1.885600E+02	-1.884345E+02
48	6.700000E+01	6.770000E+01	-1.883000E+02	-1.881600E+02
49	6.850000E+01	6.929000E+01	-1.880300E+02	-1.875100E+02
50	7.000000E+01	7.062300E+01	-1.870000E+02	-1.865000E+02
51	7.086000E+01		-1.860000E+02	-1.855000E+02
52			-1.850000E+02	-1.848290E+02
53			-1.845000E+02	

TABLE 2-9

MATERIAL DESCRIPTION FOR R-1 REFERENCE DESIGN (DWG. 939J723)
EMPLOYING COMPOSITE FUEL

This table which contained Confidential Restricted Data, was purposely deleted so that this report could be published as an unclassified document. This information is published as Table 1 in DRM 52846, "CRAM Model for the R-1 Reference Design Employing Composite Fuel."

TABLE 2-10

**MATERIAL DESCRIPTION FOR R-1 REFERENCE DESIGN (DWG. 939J723)
EMPLOYING GRAPHITE FUEL**

This table which contained Confidential Restricted Data, was purposely deleted so that this report could be published as an unclassified document. This information is published as Table 1 in DRM 52847, "CRAM Model for the R-1 Reference Design Employing Graphite Fuel."

TABLE 2-11
UNCERTAINTY FACTORS TO BE APPLIED
TO CALCULATED RADIATION LEVELS FORWARD* OF THE R-1 REACTOR

	DISCRETE ORDINATES		POINT KERNEL	
	NEAR REACTOR CENTERLINE	NEAR REACTOR PERIPHERY	NEAR REACTOR CENTERLINE	NEAR REACTOR PERIPHERY
FAST NEUTRONS	1.3	1.8	1.8	3.1
PHOTONS	1.7	2.0	1.2	1.4

* NOTE: Factors are applicable on the PVARA tangent plane or at locations in close proximity to the plane.

TABLE 2-12

STATISTICAL ANALYSIS OF THE DIFFERENCE BETWEEN CALCULATED AND
EXPERIMENTAL DATA IN THE CORE AND REFLECTOR

Data Type and Location	Average Ratio of Experimental to Calculated Data	One Standard Deviation in the Ratio of Experimental to Calculated Data	
		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, core periphery, and reflector.

TABLE 2-13

STATISTICAL ANALYSIS OF THE DIFFERENCE BETWEEN CALCULATED AND EXPERIMENTAL DATA
INTERNAL TO THE SHIELD MOCKUPS

Shield Configuration and Data Type	Average Ratio of Experimental to Calculated Data	One Standard Deviation in the Ratio of Experimental to Calculated Data	
		Upper Limit	Lower Limit
BATH (109 gm/cm ²)			
Shield Configuration 16C			
Sulfur Data ⁽¹⁾	1.08	1.16	1.01
U-238 ⁽²⁾	0.82	1.08	0.61
Dy Thermal Flux ⁽³⁾	0.60	0.83	0.43
CaF ₂ TLD Data ⁽⁴⁾	0.92	0.96	0.88

⁽¹⁾Neutron Flux E > 2.9 MeV

⁽²⁾Neutron Flux E > 1.5 MeV

⁽³⁾Thermal Neutron Flux E < 0.4 eV

⁽⁴⁾Photon Dose Rate

TABLE 2-14

STATISTICAL ANALYSIS OF THE DIFFERENCE BETWEEN CALCULATED AND
EXPERIMENTAL DATA IN THE CHESH

Data Type and Location	Average Ratio of Experimental to Calculated Data	One Standard Deviation in the Ratio of Experimental to Calculated Data	
		Upper Limit	Lower Limit
U-238 Data ⁽¹⁾ - Internal to CHESH	0.81	0.83	0.80
U-238 Data ⁽¹⁾ - At CHESH Surface	0.83	0.84	0.81
Sulfur Data ⁽²⁾ - Internal to CHESH	1.60	1.66	1.54
Sulfur Data ⁽²⁾ - at CHESH Surface	1.64	1.73	1.55
Dy Thermal Flux Data ⁽⁴⁾ - Internal to CHESH	0.24	0.29	0.19
Dy Thermal Flux Data ⁽⁴⁾ - at CHESH Surface	0.24	0.37	0.16
Phylatron Data ⁽⁵⁾ - internal to CHESH	0.68	0.77	0.60
Phylatron Data ⁽⁵⁾ - at CHESH Surface	0.58	0.65	0.52
CaF ₂ TLD Data ⁽⁶⁾ - Internal to CHESH	0.85	1.01	0.71
CaF ₂ TLD Data ⁽⁶⁾ - at CHESH Surface	0.82	1.22	0.55

(1) Neutron Flux ($E > 1.5$ MeV)

(6) Photon Dose Rate

(2) Neutron Flux ($E > 2.9$ MeV)(4) Neutron Flux ($E < 0.4$ eV)

(5) Neutron Dose Rate

TABLE 2-15

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% ⁽⁵⁾	8.5
NEUTRON KINETIC ENERGY ⁽³⁾	3.60 $\pm$ 23% ⁽⁵⁾	3.5
	<hr/> 12.99 $\pm$ 8.9%	<hr/> 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 2-16

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE REACTOR REGIONS ASSOCIATED WITH THE FUEL ELEMENT ECC (COMPOSITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	1.03	1.03	1.35	1.14	1.31	1.14
Regionwise Average, F_R	---	---	---	---	1.13	1.06

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, E^A , will be within the following limits:

$$E_R F_R^{-1} \leq E^A \leq E_R F_R \quad \text{where,}$$

E_R = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-17

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F , FOR THE RADIATION ENVIRONMENT IN THE REACTOR REGIONS ASSOCIATED WITH THE CORE PERIPHERY ECC (COMPOSITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
Pointwise, F_p	1.16	1.16	1.35	1.14	1.31	1.14
Regionwise Average, F_R	---	---	---	---	1.13	1.06

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, $\bar{E}^A$, will be within the following limits:

$$\bar{E}_R F_R^{-1} \leq \bar{E}^A \leq \bar{E}_R F_R \quad \text{where,}$$

$\bar{E}_R$ = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-18

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE REACTOR REGIONS ASSOCIATED WITH THE REFLECTOR ASSEMBLY ECC (COMPOSITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	1.16	1.16	1.35	1.14	1.31	1.14
Regionwise Average, F_R	---	---	---	---	1.13	1.06

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, $\bar{E}^A$, will be within the following limits:

$$\bar{E}_R F_R^{-1} \leq \bar{E}^A \leq \bar{E}_R F_R \quad \text{where,}$$

$\bar{E}_R$ = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-19

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE REACTOR REGIONS ASSOCIATED WITH THE CLUSTER HARDWARE ECC (COMPOSITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	1.05	1.05	2.04	1.44	1.28	1.44
Regionwise Average, F_R	---	---	---	---	1.14	1.18

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, E^A , will be within the following limits:

$$E_R F_R^{-1} \leq E^A \leq E_R F_R \quad \text{where,}$$

E_R = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-20

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE REACTOR REGIONS ASSOCIATED WITH THE SUPPORT PLATE PORTION OF THE SUPPORT PLATE AND PLENA ECC (COMPOSITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	1.71	1.71	1.88	1.09	1.31	1.09
Regionwise Average, F_R	---	---	---	---	1.13	1.04

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, $\bar{E}^A$, will be within the following limits:

$$\bar{E}_R F_R^{-1} \leq \bar{E}^A \leq \bar{E}_R F_R \quad \text{where,}$$

$\bar{E}_R$ = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-21

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE REACTOR REGIONS ASSOCIATED WITH THE PLENA PORTION OF THE SUPPORT PLATE AND PLENA ECC (COMPOSITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	3.1	3.1	3.1	1.4	3.1	1.4
Regionwise Average, F_R	---	---	---	---	2.1	1.2

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, E^A , will be within the following limits:

$$E_R F_R^{-1} \leq E^A \leq E_R F_R \quad \text{where,}$$

E_R = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-22

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE REACTOR REGIONS ASSOCIATED WITH THE INTERNAL SHIELD ECC (COMPOSITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	1.71	1.71	1.88	1.09	1.31	1.09
Regionwise Average, F_R	---	---	---	---	1.13	1.04

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, $\bar{E}^A$, will be within the following limits:

$$\bar{E}_R F_R^{-1} \leq \bar{E}^A \leq \bar{E}_R F_R \quad \text{where,}$$

$\bar{E}_R$ = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-23

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE REACTOR REGIONS ASSOCIATED WITH THE CONTROL DRUM ACTUATOR PORTION OF THE CONTROL DRUM DRIVE ASSEMBLY ECC (COMPOSITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	3.1	3.1	3.1	1.4	3.1	1.4
Regionwise Average, F_R	---	---	---	---	2.1	1.2

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, E^A , will be within the following limits:

$$E_R F_R^{-1} \leq E^A \leq E_R F_R \quad \text{where,}$$

E_R = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-24

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE REACTOR REGIONS ASSOCIATED WITH THE FUEL ELEMENT ECC (GRAPHITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	1.02	1.02	1.29	1.12	1.26	1.12
Regionwise Average, F_R	---	---	---	---	1.11	1.05

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_{p(m)} = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, E^A , will be within the following limits:

$$E_R F_R^{-1} \leq E^A \leq E_R F_R \quad \text{where,}$$

E_R = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-25

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE
REACTOR REGIONS ASSOCIATED WITH THE CORE PERIPHERY ECC (GRAPHITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	1.13	1.13	1.29	1.12	1.26	1.12
Regionwise Average, F_R	---	---	---	---	1.11	1.05

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p^{(m)} = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, $\bar{E}^A$, will be within the following limits:

$$\bar{E}_R F_R^{-1} \leq \bar{E}^A \leq \bar{E}_R F_R \quad \text{where,}$$

$\bar{E}_R$ = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-26

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE REACTOR REGIONS ASSOCIATED WITH THE REFLECTOR ASSEMBLY ECC (GRAPHITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	1.13	1.13	1.29	1.12	1.26	1.12
Regionwise Average, F_R	---	---	---	---	1.11	1.05

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, E^A , will be within the following limits:

$$E_R F_R^{-1} \leq E^A \leq E_R F_R \quad \text{where,}$$

E_R = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-27

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE
REACTOR REGIONS ASSOCIATED WITH THE CLUSTER HARDWARE ECC (GRAPHITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	1.04	1.04	1.87	1.37	1.23	1.37
Regionwise Average, F_R	---	---	---	---	1.12	1.15

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, $\bar{E}^A$, will be within the following limits:

$$\bar{E}_R F_R^{-1} \leq \bar{E}^A \leq \bar{E}_R F_R \quad \text{where,}$$

$\bar{E}_R$ = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-28

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE REACTOR REGIONS ASSOCIATED WITH THE SUPPORT PLATE PORTION OF THE SUPPORT PLATE AND PLENA ECC (GRAPHITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	1.59	1.59	1.73	1.07	1.26	1.07
Regionwise Average, F_R	---	---	---	---	1.11	1.03

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, $\bar{E}^A$, will be within the following limits:

$$\bar{E}_R F_R^{-1} \leq \bar{E}^A \leq \bar{E}_R F_R \quad \text{where,}$$

$\bar{E}_R$ = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-29

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE REACTOR REGIONS ASSOCIATED WITH THE INTERNAL SHIELD ECC (GRAPHITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	1.59	1.59	1.73	1.07	1.26	1.07
Regionwise Average, F_R	---	---	---	---	1.11	1.03

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, $\bar{E}^A$, will be within the following limits:

$$E_R F_R^{-1} \leq \bar{E}^A \leq E_R F_R \quad \text{where,}$$

E_R = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-30

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE REACTOR REGIONS ASSOCIATED WITH THE PLENA PORTION OF THE SUPPORT PLATE AND PLENA ECC (GRAPHITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	3.1	3.1	3.1	1.4	3.1	1.4
Regionwise Average, F_R	---	---	---	---	2.1	1.2

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, E^A , will be within the following limits:

$$E_R F_R^{-1} \leq E^A \leq E_R F_R \quad \text{where,}$$

E_R = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-31

REGIONWISE AND POINTWISE UNCERTAINTY FACTORS, F, FOR THE RADIATION ENVIRONMENT IN THE REACTOR REGIONS ASSOCIATED WITH THE CONTROL DRUM ACTUATOR PORTION OF THE CONTROL DRUM DRIVE ASSEMBLY ECC (GRAPHITE FUELED CORE)

Type of Uncertainty Factor	Uncertainty Factor, F					
	Fast Neutron Data	Intermediate Neutron Data	Thermal Neutron Data	Photon Dose Data	Heating Rate Data	
					Neutron	Photon
Pointwise, F_p	3.1	3.1	3.1	1.4	3.1	1.4
Regionwise Average, F_R	---	---	---	---	2.1	1.2

- NOTE: 1. Section 2.4.2 defines the reactor regions in the Radiation and Shielding model associated with each CEI and ECC.
2. There is a 95% confidence level (probability) that 68% (1σ) of the actual pointwise radiation environment data E^A , will be within the following limits:

$$E_p F_p^{-1} \leq E^A \leq E_p F_p \quad \text{where,}$$

E_p = the calculated nominal radiation environment data, and

F_p = the pointwise uncertainty factor.

The uncertainty factor for other than one standard deviation is determined as follows:

$$F_p(m) = F_p^m \quad \text{where,}$$

m = the desired level of the standard deviation in units of σ (i.e., 1.0, 2.3, etc.)

3. There is a 95% confidence level (probability) that the actual value of the regionwise average radiation environment, E^A , will be within the following limits:

$$E_R F_R^{-1} \leq E^A \leq E_R F_R \quad \text{where,}$$

E_R = the calculated nominal value of the regionwise average radiation environment, and

F_R = the regionwise average uncertainty factor.

4. Factors for neutron heating rate data apply to neutron kinetic heating only.

TABLE 2-32

NERVA FLIGHT UNCERTAINTY FACTORS FOR THE TOTAL HEAT DEPOSITED IN EACH ECC¹

Engineering Critical Component	NERVA Flight Uncertainty Factors on Total Heat ²	
	Composite Fueled Core	Graphite Fueled Core
Fuel Element	1.06	1.04
Core Periphery	1.06	1.04
Reflector Assembly	1.10	1.08
Cluster Hardware	1.07	1.05
Support Plate Portion of SP/P ECC	1.09	1.07
Internal Shield	1.18	1.15
Plena Portion of SP/P ECC (Regions above the Internal Shield)	1.24	1.20
Control Drum Actuator Portion of CDDA ECC	1.24	1.20

(1) See Section 2.4.2 for a definition of the reactor regions in the radiation analysis model associated with each ECC.

(2) There is a 95 percent confidence level (probability) that the actual value of the total heat H^A for the design defined by WANL Drawing 939J723, to be within the following limits:

$$H^R F_F^{-1} \leq H^A \leq H^R F_F \text{ where}$$

H^R = the calculated nominal value of the total heat

F_F = NERVA flight uncertainty factor on total heat

TABLE 2-33

UNCERTAINTY OF PHOTON AND BETA SOURCES FOR KEY REGIONS OF THE NSS

<u>Fission Sources</u>	
Prompt Photons ($t < 10^{-3}$ sec)	7.6 ± 0.6 MeV/Fission
Delayed Photons ($t > 10^{-3}$ sec)	7.2 ± 1.3 MeV/Fission
Beta ($0 < t < \infty$)	$7.7 \frac{+0.3}{-1.3}$ MeV/Fission

<u>Secondary Photon Sources</u>		
<u>Component</u>	<u>Correction Factor *</u>	<u>Percent Standard Deviation in the Experimental Data **</u>
Reactor Core	1.07	$\pm 1.0\%$
Reflector	1.08	± 14.1
Support Plate	0.84	± 35.4
Shield	0.44	± 186.0
Aft End Hardware	2.53	± 11.2

*Factor to be used to adjust calculated data to nominal measurement weighted according to absorptions by neutron group, except in the core.

**Uncertainty in experimental data weighted according to absorptions by neutron.

TABLE 2-34

COMPARISON OF INTERNAL RADIATION ENVIRONMENTAL DATA AT KEY NSS LOCATIONS
FOR COMPOSITE AND GRAPHITE FUELED CORES

Location	Coordinates		Relative ⁽¹⁾ Fast Neutron Flux ($E > 1.0$ MeV)		Relative ⁽¹⁾ Intermediate Neutron Flux ($0.4 \text{ eV} < E < 1.0$ MeV)		Relative ⁽¹⁾ Thermal Neutron Flux ($E < 0.4 \text{ eV}$)		Relative ⁽¹⁾ Gamma Ray Dose Rate	
	Radius (in.)	Reactor Station (in.)	Composite Fuel	Graphite Fuel	Composite Fuel	Graphite Fuel	Composite Fuel	Graphite Fuel	Composite Fuel	Graphite Fuel
Reactor Centerline - Core Midplane	0.000	26.000	1.00	0.95	1.00	0.97	1.00	4.95	1.00	1.38
Inner Surface of Reflector - Core Midplane	22.215	26.000	1.00	0.96	1.00	0.94	1.00	1.07	1.00	1.23
Pressure Vessel Surface - Core Midplane	27.890	26.000	1.00	0.98	1.00	0.93	1.00	1.00	1.00	1.20
Reactor Centerline - Alt Face of CHESH	0.000	54.131	1.00	0.92	1.00	0.89	1.00	1.21	1.00	1.40
Reactor Centerline - Alt Face of BATH Control Shield	0.000	-11.830	1.00	0.94	1.00	0.89	1.00	0.96	1.00	1.04
Reactor Centerline - Pressure Vessel Dome	0.000	-37.000	1.00	0.97	1.00	0.94	1.00	0.94	1.00	1.05

(1) Data are reported relative to the composite fueled core, i.e., $\frac{\text{Graphite}}{\text{Composite}}$.

TABLE 2-35
RADIATION ENVIRONMENT AT KEY NSS LOCATIONS - COMPOSITE FUEL IN THE CORE

Location	Coordinates		Gamma Environment				NEUTRON ENVIRONMENT								
							Fast ($E > 1.0$ mev)			Intermediate ($0.4 \text{ ev} \leq E \leq 1.0$ mev)			Thermal ($E < 0.4 \text{ ev}$)		
	Radius (IN.)	Reactor Station (IN.)	Heating Rate (w/gm - SS)	Dose Rate (RADS-C/HR)	1 HR. Dose (RADS-C)	10 HR. Dose (RADS-C)	Flux ($\text{N/cm}^2 \text{ - sec}$)	1 HR. Fluency (N/cm^2)	10 HR. Fluency (N/cm^2)	Flux ($\text{N/cm}^2 \text{ - sec}$)	1 HR. Fluency (N/cm^2)	10 HR. Fluency (N/cm^2)	Flux ($\text{N/cm}^2 \text{ - sec}$)	1 HR. Fluency (N/cm^2)	10 HR. Fluency (N/cm^2)
Reactor Centerline - Core Midplane	0.000	26.000	6.88 (1)	2.46 (10)	2.46 (10)	2.46 (11)	4.76 (15)	1.71 (19)	1.71 (20)	2.37 (16)	8.53 (19)	8.53 (20)	3.68 (12)	1.32 (16)	1.32 (17)
Inner Surface of Reflector - Core Midplane	22.215	26.000	1.19 (1)	4.10 (9)	4.10 (9)	4.10 (10)	7.10 (14)	2.56 (18)	2.56 (19)	7.76 (15)	2.79 (19)	2.79 (20)	5.12 (14)	1.84 (18)	1.84 (19)
Pressure Vessel Surface - Core Midplane	27.890	26.000	3.14 (0)	1.05 (9)	1.05 (9)	1.05 (10)	6.07 (13)	2.19 (17)	2.19 (18)	5.20 (14)	1.87 (18)	1.87 (19)	9.86 (13)	3.55 (17)	3.55 (18)
Reactor Centerline - Alt Face of CHESH	0.000	54.131	4.18 (0)	1.44 (9)	1.44 (9)	1.44 (10)	3.61 (14)	1.30 (18)	1.30 (19)	1.26 (15)	4.54 (18)	4.54 (19)	3.89 (12)	1.40 (16)	1.40 (17)
Reactor Centerline - Alt Face of BATH Central Shield	0.000	-11.830	3.51 (0)	1.16 (9)	1.16 (9)	1.16 (10)	1.20 (14)	4.32 (17)	4.32 (18)	1.00 (15)	3.60 (18)	3.60 (19)	3.92 (13)	1.41 (17)	1.41 (18)
Reactor Centerline - Pressure Vessel Dome	0.000	-37.000	1.31 (-2)	4.16 (6)	4.16 (6)	4.17 (7)	6.42 (11)	2.31 (15)	2.31 (16)	1.87 (12)	6.73 (15)	6.73 (16)	3.10 (11)	1.12 (15)	1.12 (16)

Numbers in Parentheses Refer to Powers of Ten to be Applied to the Data.

TABLE 2-36

PRESSURE VESSEL AND REACTOR ASSEMBLY SOURCE STRENGTHS

Composite Core

GAMMA SOURCE STRENGTH (MEV/SEC)									
Energy (Mev)	REGION 1 Hot End Support Hardware	REGION 2 Core	REGION 3 Core Periphery	REGION 4 Lateral Support	REGION 5 Aluminum Structure	REGION 6 Reflector	REGION 7 Pressure Vessel Side - 1	REGION 8 Aft Reflector Hardware	REGION 9 Core Plenum
8.50	2.23 (17)	2.29 (18)	1.48 (17)	2.58 (14)	1.41 (18)	5.13 (18)	1.44 (17)	5.92 (16)	7.96 (17)
7.25	1.89 (17)	4.63 (18)	2.54 (17)		4.51 (17)	1.32 (18)	5.70 (16)	2.79 (16)	5.15 (17)
6.50	1.91 (17)	1.30 (19)	5.92 (17)		5.96 (17)	2.51 (18)	7.37 (16)	2.85 (16)	7.83 (17)
5.50	1.59 (17)	1.68 (19)	6.58 (17)		5.57 (17)	1.53 (18)	7.23 (16)	2.76 (16)	1.10 (18)
4.50	1.60 (17)	3.82 (19)	1.02 (18)	3.42 (17)	8.60 (17)	1.57 (18)	1.43 (17)	2.98 (16)	1.24 (18)
3.50	1.81 (17)	8.16 (19)	1.11 (18)	8.21 (15)	1.20 (18)	1.64 (18)	2.02 (17)	2.73 (16)	1.81 (16)
2.80	1.42 (17)	4.35 (19)	3.48 (17)		8.64 (17)	7.55 (17)	1.49 (17)	1.57 (16)	6.73 (17)
2.50	2.90 (17)	7.37 (19)	1.03 (18)	2.10 (16)	6.32 (17)	8.58 (17)	1.03 (17)	1.98 (16)	6.11 (17)
2.00	2.26 (17)	8.62 (19)	7.40 (17)		4.00 (17)	6.16 (17)	7.14 (16)	1.38 (16)	5.98 (17)
1.50	1.74 (17)	1.12 (20)	5.11 (17)		4.39 (17)	9.56 (17)	6.24 (16)	1.65 (16)	6.65 (17)
1.00	1.11 (17)	1.46 (20)	5.77 (17)	2.86 (15)	1.07 (18)	8.21 (17)	1.39 (17)	1.61 (16)	6.44 (17)
0.70	4.68 (16)	1.94 (20)	9.54 (16)		7.05 (17)	8.15 (17)	1.04 (17)	1.43 (16)	7.22 (17)
0.30	6.97 (15)	9.06 (19)	4.90 (16)		1.76 (17)	1.73 (17)	3.51 (16)	2.86 (15)	5.81 (16)
TOTAL	2.10 (16)	9.04 (20)	7.13 (18)	3.74 (17)	9.36 (18)	1.87 (19)	1.36 (18)	2.99 (17)	1.02 (19)

- NOTE: 1) Numbers in Parentheses Refer to Powers of 10.
 2) Refer to Figure 2-13 for Region Location.

643 9/8/70

TABLE 2-36
(Continued)

PRESSURE VESSEL AND REACTOR ASSEMBLY SOURCE STRENGTHS

Composite Core

Energy (MeV)	GAMMA SOURCE STRENGTH (MEV/SEC)								
	REGION 10 Lateral Support Forward	REGION 11 Support Plate	REGION 12 Forward Reflector Hardware	REGION 13 Support Plate Plenum	REGION 14 Instrument Ring	REGION 15 Central Shield Plate - Afr	REGION 16 BATH Central Shield	REGION 17 BATH Peripheral Shield - 1	REGION 18 BATH Peripheral Shield - 2
8.50	2.37 (17)	1.77 (18)	4.06 (17)	2.12 (17)	1.83 (16)	1.42 (16)	5.42 (15)	7.39 (14)	5.47 (14)
7.25	6.16 (16)	5.75 (17)	1.32 (17)	6.81 (16)	6.13 (15)	5.63 (15)	2.21 (15)	2.99 (14)	2.23 (14)
6.50	8.29 (16)	5.51 (17)	1.38 (17)	6.10 (16)	6.13 (15)	7.26 (15)	9.10 (16)	1.14 (16)	9.40 (15)
5.50	7.17 (16)	5.96 (17)	1.36 (17)	6.81 (16)	6.59 (15)	7.20 (15)	9.01 (15)	1.15 (15)	9.25 (14)
4.50	6.90 (16)	5.78 (17)	1.24 (17)	5.48 (16)	7.85 (15)	1.45 (16)	2.43 (16)	3.13 (15)	2.55 (15)
3.50	6.67 (16)	5.96 (17)	1.04 (17)	4.60 (16)	8.86 (15)	2.18 (16)	3.66 (16)	4.57 (15)	3.76 (15)
2.80	3.08 (16)	3.34 (17)	5.13 (16)	1.93 (16)	5.92 (15)	1.73 (16)	2.14 (16)	2.63 (15)	2.19 (15)
2.50	3.36 (16)	3.80 (17)	1.01 (17)	2.89 (16)	5.48 (15)	1.21 (16)	2.20 (16)	2.82 (15)	2.24 (15)
2.00	3.10 (16)	2.71 (17)	5.29 (16)	2.46 (16)	3.69 (15)	7.05 (15)	9.08 (15)	1.16 (15)	9.31 (14)
1.50	5.88 (16)	4.08 (17)	8.11 (16)	4.30 (16)	4.51 (15)	6.10 (15)	3.44 (16)	4.33 (15)	3.55 (15)
1.00	4.83 (16)	4.25 (17)	5.79 (16)	2.49 (16)	6.40 (15)	2.20 (16)	1.12 (17)	1.19 (16)	1.12 (16)
0.70	4.80 (16)	4.39 (17)	6.75 (16)	3.83 (16)	5.48 (15)	1.29 (16)	8.02 (17)	1.03 (17)	8.38 (16)
0.30	7.16 (15)	7.38 (16)	1.79 (16)	5.56 (15)	1.12 (15)	3.00 (15)	2.53 (15)	3.51 (14)	2.59 (14)
TOTAL	8.46 (17)	6.99 (18)	1.49 (18)	6.94 (17)	8.65 (16)	1.51 (17)	1.17 (18)	1.46 (17)	1.22 (17)

NOTE: 1) Numbers in Parentheses Refer to Powers of 10.
2) Refer to Figure 2-13 for Region Location.

643 9/8/70

TABLE 2-36

(Continued)

PRESSURE VESSEL AND REACTOR ASSEMBLY SOURCE STRENGTHS

Composite Core

Energy (Mev)	GAMMA SOURCE STRENGTH (MEV/SEC)								
	REGION 19	REGION 20	REGION 21	REGION 22	REGION 23	REGION 24	REGION 25	REGION 26	REGION 27
	BATH Peripheral Shield - 3	BATH Peripheral Shield - 4	Lead Central Shield	Lead Peripheral Shield - 1	Lead Peripheral Shield - 2	Lead Peripheral Shield - 3	Lead Peripheral Shield - 4	Peripheral Shield Plate	Shield Penum
8.50	3.01 (15)	3.84 (14)	4.57 (14)	4.72 (13)	5.44 (13)	1.12 (14)	1.75 (13)	1.17 (14)	5.14 (13)
7.25	1.10 (15)	1.57 (14)	1.73 (11)	2.73 (11)		1.97 (13)	3.14 (11)	3.76 (13)	1.96 (13)
6.50	8.57 (15)	5.53 (15)	2.30 (11)	3.66 (11)		2.29 (13)	4.55 (11)	4.64 (13)	2.38 (13)
5.50	1.75 (15)	5.92 (14)	2.39 (11)	3.85 (11)		2.17 (13)	5.18 (11)	4.53 (13)	2.41 (13)
4.50	3.02 (15)	1.61 (15)	4.97 (11)	8.04 (11)		1.97 (13)	1.13 (12)	6.82 (13)	4.49 (13)
3.50	4.36 (15)	2.64 (15)	1.00 (12)	1.71 (12)		2.33 (13)	3.29 (12)	1.04 (14)	6.21 (13)
2.50	2.27 (15)	1.69 (15)	2.31 (15)	2.83 (14)	3.25 (14)	2.41 (14)	2.97 (14)	7.79 (13)	5.46 (13)
2.10	2.29 (15)	1.68 (15)	6.92 (12)	6.15 (12)	6.48 (11)	9.27 (12)	2.48 (12)	6.08 (13)	3.14 (14)
2.00	1.41 (15)	5.83 (14)	2.22 (11)	3.53 (11)		1.17 (13)	4.34 (11)	3.23 (13)	2.15 (13)
1.50	4.16 (15)	2.11 (15)	2.05 (11)	3.29 (11)		2.23 (13)	4.51 (11)	3.72 (13)	2.01 (13)
1.00	1.05 (16)	8.86 (15)	8.88 (13)	1.29 (13)	1.25 (13)	2.63 (13)	1.61 (13)	1.01 (14)	6.28 (13)
0.70	6.96 (16)	4.28 (16)	3.60 (14)	4.49 (13)	5.12 (13)	5.66 (13)	4.76 (13)	5.93 (13)	3.57 (13)
0.30	3.05 (14)	1.72 (14)	2.43 (14)	2.98 (13)	3.46 (13)	2.69 (13)	3.10 (13)	1.06 (13)	7.13 (12)
TOTAL	1.11 (17)	7.48 (16)	3.47 (15)	4.28 (14)	4.78 (14)	6.13 (14)	4.18 (14)	7.97 (14)	7.43 (14)

- NOTE: 1) Numbers in Parentheses Refer to Powers of 10.
2) Refer to Figure 2-13 for Region Location.

643 9/8/70

TABLE 2-36
(Continued)

PRESSURE VESSEL AND REACTOR ASSEMBLY SOURCE STRENGTHS

Composite Core

GAMMA SOURCE STRENGTH (MEV/SEC)								
Energy (Mev)	REGION 28 Flow Baffle 1	REGION 29 Flow Baffle 2	REGION 30 Central Dome Plenum	REGION 31 Peripheral Dome Plenum - 1	REGION 32 Peripheral Dome Plenum - 2	REGION 33 Peripheral Dome Plenum - 3	REGION 34 Pressure Vessel Dome	REGION 35 Pressure Vessel Side - 2
8.50	2.85 (14)	4.88 (15)		2.43 (14)	3.13 (14)	7.66 (13)	1.50 (14)	1.28 (15)
7.25	1.14 (14)	1.79 (15)		6.41 (13)	8.29 (13)	2.03 (13)	5.94 (13)	4.99 (14)
6.50	1.45 (14)	1.91 (15)		7.76 (13)	1.00 (14)	2.46 (13)	7.75 (13)	6.80 (14)
5.50	1.44 (14)	2.04 (15)		7.13 (13)	9.22 (13)	2.27 (13)	7.79 (13)	6.97 (14)
4.50	2.91 (14)	2.90 (15)		6.94 (13)	9.60 (13)	2.23 (13)	1.58 (14)	1.40 (15)
3.50	4.21 (14)	4.66 (15)		6.30 (13)	8.44 (13)	2.28 (13)	2.63 (14)	2.78 (15)
2.50	5.42 (14)	3.39 (15)		3.46 (13)	4.60 (13)	1.22 (13)	2.08 (14)	2.16 (15)
2.50	2.18 (14)	2.59 (15)	1.41 (14)	3.90 (14)	5.02 (14)	1.22 (14)	1.36 (14)	1.65 (15)
2.00	1.41 (14)	1.44 (15)		2.90 (13)	3.85 (13)	1.02 (13)	7.51 (13)	6.51 (14)
1.50	1.23 (14)	1.69 (15)		4.57 (13)	6.10 (13)	1.67 (13)	7.15 (13)	6.63 (14)
1.00	3.52 (14)	5.14 (15)		3.84 (13)	5.27 (13)	1.58 (13)	2.18 (14)	2.96 (15)
0.70	2.06 (14)	3.21 (15)		4.01 (13)	5.39 (13)	1.53 (13)	1.25 (14)	1.62 (15)
0.30	4.47 (13)	5.76 (14)		7.66 (12)	9.98 (12)	2.53 (12)	3.34 (13)	4.48 (14)
TOTAL	2.83 (15)	3.62 (16)	1.41 (14)	1.17 (15)	1.53 (15) ^a	3.84 (14)	1.65 (15)	1.75 (16)

- NOTE: 1) Numbers in Parentheses Refer to Powers of 10.
2) Refer to Figure 2-13 for Region Location.

643 9/8/70

TABLE 2-37

FISSION PRODUCT GAMMA RAY SOURCE STRENGTH (MEV/SEC) FOR VARIOUS TIMES

FOLLOWING TERMINATION OF A 10-MINUTE OPERATION AT FULL POWER (1511 MW THERMAL)

ENERGY (Mev)	TIME AFTER SHUTDOWN (SEC)								
	1.8 (3)	2.4 (3)	3.0 (3)	3.6 (3)	4.8 (3)	6.0 (3)	7.2 (3)	1.8 (4)	3.6 (4)
2.80	2.68 (17)	1.89 (17)	1.46 (17)	1.17 (17)	8.08 (16)	5.91 (16)	4.50 (16)	1.14 (16)	3.35 (15)
2.50	1.17 (18)	7.48 (17)	5.13 (17)	3.75 (17)	2.33 (17)	1.66 (17)	1.27 (17)	3.68 (16)	1.18 (16)
2.00	9.73 (17)	5.25 (17)	3.09 (17)	1.98 (17)	1.03 (17)	6.94 (16)	5.51 (16)	2.20 (16)	6.99 (15)
1.50	1.95 (18)	1.49 (18)	1.15 (18)	9.08 (17)	5.89 (17)	4.08 (17)	3.00 (17)	8.08 (16)	2.92 (16)
1.00	1.22 (18)	8.59 (17)	6.37 (17)	4.92 (17)	3.26 (17)	2.39 (17)	1.88 (17)	6.07 (16)	3.10 (16)
0.70	1.24 (18)	1.10 (18)	9.86 (17)	8.87 (17)	7.27 (17)	6.03 (17)	5.05 (17)	1.37 (17)	5.73 (16)
0.30	2.23 (17)	1.79 (17)	1.42 (17)	1.14 (17)	7.54 (16)	5.37 (16)	4.14 (16)	1.67 (16)	1.06 (16)
TOTAL	7.04 (18)	5.09 (18)	3.89 (18)	3.09 (18)	2.13 (18)	1.60 (18)	1.26 (18)	3.65 (17)	1.50 (17)

ENERGY (Mev)	TIME AFTER SHUTDOWN (SEC)								
	7.2 (4)	8.64 (4)	1.728 (5)	4.32 (5)	6.05 (5)	8.64 (5)	1.728 (6)	2.592 (6)	5.184 (6)
2.80	6.47 (14)	5.34 (14)	6.67 (14)	8.96 (14)	8.82 (14)	7.91 (14)	4.73 (14)	2.75 (14)	5.42 (13)
2.50	2.29 (15)	1.44 (15)	8.29 (14)	9.59 (14)	9.28 (14)	8.27 (14)	4.93 (14)	2.87 (14)	5.66 (13)
2.00	1.18 (15)	7.12 (14)	2.32 (14)	1.91 (14)	1.82 (14)	1.61 (14)	9.64 (13)	5.65 (13)	1.17 (13)
1.50	7.14 (15)	4.65 (15)	8.10 (14)	2.77 (14)	2.11 (14)	1.46 (14)	5.82 (13)	3.06 (13)	6.00 (12)
1.00	1.31 (16)	9.57 (15)	2.27 (15)	4.65 (14)	2.68 (14)	1.43 (14)	3.81 (13)	1.78 (13)	3.69 (12)
0.70	3.32 (16)	2.81 (16)	1.25 (16)	3.22 (15)	1.97 (15)	1.17 (15)	4.38 (14)	3.08 (14)	2.11 (14)
0.30	7.12 (15)	6.23 (15)	3.15 (15)	1.20 (15)	8.14 (14)	5.08 (14)	1.76 (14)	8.91 (13)	2.62 (13)
TOTAL	6.47 (16)	5.12 (16)	2.05 (16)	7.21 (15)	5.26 (15)	3.74 (15)	1.77 (15)	1.06 (15)	3.70 (14)

NOTE: Numbers in parentheses refer to powers of ten.

331 9/29/70

TABLE 2-37
(Continued)

FISSION PRODUCT GAMMA RAY SOURCE STRENGTH (MEV/SEC) FOR VARIOUS TIMES

FOLLOWING TERMINATION OF A 10-MINUTE OPERATION AT FULL POWER (1511 MW THERMAL)

ENERGY (Mev)	TIME AFTER SHUTDOWN (SEC)						
	7.776 (6)	1.037 (7)	1.296 (7)	1.555 (7)	2.074 (7)	2.592 (7)	3.154 (7)
2.80	1.07 (13)	2.10 (12)	4.14 (11)	8.16 (10)	3.16 (9)	1.23 (8)	3.63 (6)
2.50	1.11 (13)	2.19 (12)	4.32 (11)	8.51 (10)	3.30 (9)	1.28 (8)	3.79 (6)
2.00	2.82 (12)	1.03 (12)	6.41 (11)	5.33 (11)	4.46 (11)	3.85 (11)	3.29 (11)
1.50	1.31 (12)	3.79 (11)	1.87 (11)	1.41 (11)	1.15 (11)	9.87 (10)	8.43 (10)
1.00	9.89 (11)	3.75 (11)	2.04 (11)	1.36 (11)	7.50 (10)	4.65 (10)	3.08 (10)
0.70	1.63 (14)	1.26 (14)	9.63 (13)	7.27 (13)	4.06 (13)	2.25 (13)	1.20 (13)
0.30	1.25 (13)	7.17 (12)	4.50 (12)	3.03 (12)	1.67 (12)	1.14 (12)	8.59 (11)
TOTAL	2.03 (14)	1.39 (14)	1.03 (14)	7.67 (13)	4.29 (13)	2.41 (13)	1.33 (13)

NOTE: Numbers in parentheses refer to powers of ten.

331 9/29/70

TABLE 2-38

FISSION PRODUCT GAMMA RAY SOURCE STRENGTH (MEV/SEC) FOR VARIOUS TIMES

FOLLOWING TERMINATION OF A 30-MINUTE OPERATION AT FULL POWER (1511 MW THERMAL)

ENERGY (Mev)	TIME AFTER SHUTDOWN (SEC)								
	1.8 (3)	2.4 (3)	3.0 (3)	3.6 (3)	4.8 (3)	6.0 (3)	7.2 (3)	1.8 (4)	3.6 (4)
2.80	6.03 (17)	4.51 (17)	3.59 (17)	2.94 (17)	2.09 (17)	1.56 (17)	1.20 (17)	3.25 (16)	9.69 (15)
2.50	2.43 (18)	1.63 (18)	1.18 (18)	8.98 (17)	5.93 (17)	4.37 (17)	3.44 (17)	1.06 (17)	3.44 (16)
2.00	1.81 (18)	1.03 (18)	6.44 (17)	4.37 (17)	2.54 (17)	1.86 (17)	1.53 (17)	6.34 (16)	2.03 (16)
1.50	4.59 (18)	3.55 (18)	2.79 (18)	2.22 (18)	1.48 (18)	1.06 (18)	7.97 (17)	2.32 (17)	8.51 (16)
1.00	2.71 (18)	1.99 (18)	1.52 (18)	1.21 (18)	8.41 (17)	6.37 (17)	5.11 (17)	1.75 (17)	9.16 (16)
0.70	3.32 (18)	2.97 (18)	2.67 (18)	2.41 (18)	1.99 (18)	1.66 (18)	1.39 (18)	3.89 (17)	1.69 (17)
0.30	5.44 (17)	4.35 (17)	3.48 (17)	2.81 (17)	1.92 (17)	1.41 (17)	1.13 (17)	4.88 (16)	3.15 (16)
TOTAL	1.60 (19)	1.21 (19)	9.51 (18)	7.76 (18)	5.56 (18)	4.27 (18)	3.43 (18)	1.05 (18)	4.42 (17)

NOTE: Numbers in parentheses refer to powers of ten.

ENERGY (Mev)	TIME AFTER SHUTDOWN (SEC)								
	7.2 (4)	8.64 (4)	1.728 (5)	4.32 (5)	6.05 (5)	8.64 (5)	1.728 (6)	2.592 (6)	5.184 (6)
2.80	1.92 (15)	1.59 (15)	2.01 (15)	2.69 (15)	2.65 (15)	2.37 (15)	1.42 (15)	8.25 (14)	1.63 (14)
2.50	6.72 (15)	4.27 (15)	2.49 (15)	2.88 (15)	2.78 (15)	2.48 (15)	1.48 (15)	8.61 (14)	1.69 (14)
2.00	3.44 (15)	2.09 (15)	6.93 (14)	5.73 (14)	5.46 (14)	4.84 (14)	2.89 (14)	1.69 (14)	3.51 (13)
1.50	2.10 (16)	1.37 (16)	2.41 (15)	8.29 (14)	6.33 (14)	4.38 (14)	1.74 (14)	9.18 (13)	1.80 (13)
1.00	3.87 (16)	2.83 (16)	6.76 (15)	1.39 (15)	8.02 (14)	4.28 (14)	1.14 (14)	5.33 (13)	1.11 (13)
0.70	9.89 (16)	8.36 (16)	3.74 (16)	9.64 (15)	5.91 (15)	3.49 (15)	1.31 (15)	9.24 (14)	6.33 (14)
0.30	2.12 (16)	1.86 (16)	9.39 (15)	3.59 (15)	2.44 (15)	1.52 (15)	5.27 (14)	2.67 (14)	7.86 (13)
TOTAL	1.92 (17)	1.52 (17)	6.12 (16)	2.16 (16)	1.58 (16)	1.12 (16)	5.31 (15)	3.19 (15)	1.11 (15)

TABLE 2-38
(Continued)

FISSION PRODUCT GAMMA RAY SOURCE STRENGTH (MEV/SEC) FOR VARIOUS TIMES
FOLLOWING TERMINATION OF A 30-MINUTE OPERATION AT FULL POWER. (1511 MW THERMAL)

ENERGY (Mev)	TIME AFTER SHUTDOWN (SEC)						
	7.776 (6)	1.037 (7)	1.296 (7)	1.555 (7)	2.074 (7)	2.592 (7)	3.154 (7)
2.80	3.20 (13)	6.31 (12)	1.24 (12)	2.45 (11)	9.49 (9)	3.68 (8)	1.09 (7)
2.50	3.34 (13)	6.58 (12)	1.29 (12)	2.55 (11)	9.90 (9)	3.84 (8)	1.14 (7)
2.00	8.47 (12)	3.09 (12)	1.92 (12)	1.59 (12)	1.34 (12)	1.16 (12)	9.86 (11)
1.50	3.93 (12)	1.14 (12)	5.59 (11)	4.23 (11)	3.44 (11)	2.96 (11)	2.53 (11)
1.00	2.97 (12)	1.13 (12)	6.11 (11)	4.08 (11)	2.25 (11)	1.39 (11)	9.24 (10)
0.70	4.89 (14)	3.78 (14)	2.89 (14)	2.18 (14)	1.22 (14)	6.74 (13)	3.58 (13)
0.30	3.76 (13)	2.15 (13)	1.35 (13)	9.09 (12)	5.01 (12)	3.41 (12)	2.58 (12)
TOTAL	6.08 (14)	4.18 (14)	3.08 (14)	2.30 (14)	1.28 (14)	7.24 (13)	3.97 (13)

NOTE: Numbers in parentheses refer to powers of ten.

173 9/25/70

TABLE 2-39

FISSION PRODUCT GAMMA RAY SOURCE STRENGTH (MEV/SEC) FOR VARIOUS TIMES
 FOLLOWING TERMINATION OF A 60-MINUTE OPERATION AT FULL POWER (1511 MW THERMAL)

ENERGY (Mev)	TIME AFTER SHUTDOWN (SEC)								
	1.8 (3)	2.4 (3)	3.0 (3)	3.6 (3)	4.8 (3)	6.0 (3)	7.2 (3)	1.8 (4)	3.6 (4)
2.80	8.97 (17)	6.97 (17)	5.68 (17)	4.73 (17)	3.45 (17)	2.63 (17)	2.07 (17)	6.10 (16)	1.84 (16)
2.50	3.33 (18)	2.35 (18)	1.77 (18)	1.40 (18)	9.78 (17)	7.47 (17)	6.02 (17)	1.98 (17)	6.57 (16)
2.00	2.24 (18)	1.35 (18)	8.98 (17)	6.50 (17)	4.21 (17)	3.28 (17)	2.78 (17)	2.00 (17)	3.85 (16)
1.50	6.81 (18)	5.35 (18)	4.27 (18)	3.46 (18)	2.39 (18)	1.76 (18)	1.37 (18)	4.39 (17)	1.63 (17)
1.00	3.92 (18)	2.98 (18)	2.36 (18)	1.94 (18)	1.41 (18)	1.10 (18)	9.01 (17)	3.35 (17)	1.79 (17)
0.70	5.74 (18)	5.16 (18)	4.66 (18)	4.23 (18)	3.51 (18)	2.94 (18)	2.48 (18)	7.25 (17)	3.33 (17)
0.30	8.25 (17)	6.65 (17)	5.40 (17)	4.45 (17)	3.18 (17)	2.45 (17)	2.01 (17)	9.43 (16)	6.23 (16)
TOTAL	2.38 (19)	1.86 (19)	1.51 (19)	1.26 (19)	9.37 (18)	7.38 (18)	6.04 (18)	1.97 (18)	8.60 (17)

ENERGY (Mev)	TIME AFTER SHUTDOWN (SEC)								
	7.2 (4)	8.64 (4)	1.728 (5)	4.32 (5)	6.05 (5)	8.64 (5)	1.728 (6)	2.592 (6)	5.184 (6)
2.80	3.77 (15)	3.18 (15)	4.02 (15)	5.38 (15)	5.29 (15)	4.75 (15)	2.83 (15)	1.65 (15)	3.25 (14)
2.50	1.30 (16)	8.34 (15)	4.98 (15)	5.75 (15)	5.57 (15)	4.96 (15)	2.96 (15)	1.72 (15)	3.39 (14)
2.00	6.66 (15)	4.09 (15)	1.38 (15)	1.15 (15)	1.09 (15)	9.68 (14)	5.78 (14)	3.39 (14)	7.02 (13)
1.50	4.09 (16)	2.68 (16)	4.77 (15)	1.66 (15)	1.26 (15)	8.75 (14)	3.49 (14)	1.84 (14)	3.60 (13)
1.00	7.59 (16)	5.56 (16)	1.34 (16)	2.77 (15)	1.60 (15)	8.55 (14)	2.28 (14)	1.07 (14)	2.21 (13)
0.70	1.96 (17)	1.66 (17)	7.44 (16)	1.92 (16)	1.18 (16)	6.99 (15)	2.63 (15)	1.85 (15)	1.27 (15)
0.30	4.21 (16)	3.69 (16)	1.87 (16)	7.18 (15)	4.87 (15)	3.04 (15)	1.05 (15)	5.35 (14)	1.57 (14)
TOTAL	3.78 (17)	3.01 (17)	1.22 (17)	4.31 (16)	3.15 (16)	2.24 (16)	1.06 (16)	6.38 (15)	2.22 (15)

NOTE: Numbers in parentheses refer to powers of ten.

173 9/25/70

TABLE 2-39
(Continued)

FISSION PRODUCT GAMMA RAY SOURCE STRENGTH (MEV/SEC) FOR VARIOUS TIMES
FOLLOWING TERMINATION OF A 60-MINUTE OPERATION AT FULL POWER (1511 MW THERMAL)

ENERGY (Mev)	TIME AFTER SHUTDOWN (SEC)						
	7.776 (6)	1.037 (7)	1.296 (7)	1.555 (7)	2.074 (7)	2.592 (7)	3.154 (7)
2.80	6.40 (13)	1.26 (13)	2.48 (12)	4.89 (11)	1.90 (10)	7.36 (8)	2.18 (7)
2.50	6.68 (13)	1.32 (13)	2.59 (12)	5.10 (11)	1.98 (10)	7.68 (8)	2.27 (7)
2.00	1.69 (13)	6.17 (12)	3.84 (12)	3.20 (12)	2.68 (12)	2.31 (12)	1.97 (12)
1.50	7.86 (12)	2.27 (12)	1.12 (12)	8.45 (11)	6.87 (11)	5.92 (11)	5.06 (11)
1.00	5.93 (12)	2.25 (12)	1.22 (12)	8.17 (11)	4.50 (11)	2.79 (11)	1.85 (11)
0.70	9.78 (14)	7.56 (14)	5.78 (14)	4.36 (14)	2.44 (14)	1.35 (14)	7.17 (13)
0.30	7.51 (13)	4.30 (13)	2.70 (13)	1.82 (13)	1.00 (13)	6.81 (12)	5.15 (12)
TOTAL	1.22 (15)	8.36 (14)	6.16 (14)	4.60 (14)	2.57 (14)	1.45 (14)	7.95 (13)

NOTE: Numbers in parentheses refer to powers of ten.

173 9/25/70

TABLE 2-40

NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)DATA POINT LOCATION -- TANGENT PLANE (MAX)

$$\begin{bmatrix} R = 0 \text{ cm.} \\ Z = 254.508 \text{ cm.} \end{bmatrix}$$

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncert. & Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	4.17 (6)	4.80 (6)	6.73 (6)	8.08 (6)	1.8 (7)	2.23
Fast Neutron Flux ($E > 1.0$ Mev) (N/cm ² -sec)	6.41 (11)	2.64 (11)	8.18 (11)	1.23 (12)	2.0 (12)	1.63
Int. Neutron Flux ($0.4 \text{ ev} < E < 1.0 \text{ Mev}$) (N/cm ² -sec)	1.87 (12)	7.69 (11)	2.38 (12)	3.57 (12)	3.0 (12)	0.84
Thermal Neutron Flux ($E < 0.4 \text{ ev}$) (N/cm ² -sec)	3.09 (11)	1.27 (11)	3.94 (11)	5.90 (11)	6.0 (11)	1.02

NOTE: 1) Numbers in parentheses refer to powers of ten.
 2) Data location coordinates are referenced to CRAM model 0.0.

148 9/25/70
 242 10/5/70
 239 10/5/70

TABLE 2-41

NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)

DATA POINT LOCATION -- TPA

$$\begin{bmatrix} R = 91.44 \text{ cm.} \\ Z = 366.27 \text{ cm.} \end{bmatrix}$$

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncert. & Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	1.69 (6)	1.27 (6)	1.78 (6)	1.78 (6)	7.0 (6)	3.94
Fast Neutron Flux ($E > 1.0$ Mev) (N/cm ² -sec)	1.22 (11)	4.28 (10)	1.33 (11)	1.33 (11)	1.0 (12)	7.52
Int. Neutron Flux ($0.4 \text{ ev} < E < 1.0$ Mev) (N/cm ² -sec)	7.65 (11)	2.68 (11)	8.31 (11)	8.31 (11)	5.0 (12)	6.15
Thermal Neutron Flux ($E < 0.4$ ev) (N/cm ² -sec)	1.37 (11)	4.79 (10)	1.48 (11)	1.48 (11)	1.0 (12)	6.75

113 10/2/70 164 9/25/70
 179 9/25/70 163 9/25/70
 717 9/24/70 209 9/25/70

NOTE: 1) Numbers in parentheses refer to powers of ten.
 2) Data location coordinates are referenced to CRAM model 0.0.

TABLE 2-42

NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)

DATA POINT LOCATION -- LTS-1

$$\begin{bmatrix} R = 49.53 \text{ cm.} \\ Z = 260.86 \text{ cm.} \end{bmatrix}$$

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncert. & Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	2.10 (6)	2.60 (6)	3.64 (6)	5.46 (6)	2.1 (7)	3.84
Fast Neutron Flux ($E > 1.0$ Mev) (N/cm ² -sec)	2.69 (11)	1.34 (11)	4.15 (11)	8.30 (11)	2.8 (14)	337.0
Int. Neutron Flux ($0.4 \text{ ev} < E < 1.0$ Mev) (N/cm ² -sec)	7.76 (11)	3.86 (11)	1.20 (12)	2.40 (12)	N/A	N/A
Thermal Neutron Flux ($E < 0.4$ ev) (N/cm ² -sec)	1.60 (11)	7.97 (10)	2.47 (11)	4.94 (11)	N/A	N/A

NOTE: 1) Numbers in parentheses refer to powers of ten.
 2) Data location coordinates are referenced to CRAM model 0.0.

113 10/2/70 164 9/25/70
 179 9/25/70 163 9/25/70

TABLE 2-43

NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)

DATA POINT LOCATION -- LTS-2

$$\left[\begin{array}{l} R = 49.53 \text{ cm.} \\ Z = 265.94 \text{ cm.} \end{array} \right]$$

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncert. & Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	2.09 (6)	2.48 (6)	3.47 (6)	5.21 (6)	1.8 (7)	3.45
Fast Neutron Flux ($E > 1.0$ Mev) (N/cm ² -sec)	2.74 (11)	1.30 (11)	4.03 (11)	8.06 (11)	2.8 (14)	247.0
Int. Neutron Flux ($0.4 \text{ ev} < E < 1.0 \text{ Mev}$) (N/cm ² -sec)	7.06 (11)	3.34 (11)	1.04 (12)	2.08 (12)	N/A	N/A
Thermal Neutron Flux ($E < 0.4 \text{ ev}$) (N/cm ² -sec)	1.40 (11)	6.63 (10)	2.06 (11)	4.12 (11)	N/A	N/A

NOTE: 1) Numbers in parentheses refer to powers of ten.
 2) Data location coordinates are referenced to CRAM model 0.0.

113 10/2/70 164 9/25/70
 179 9/25/70 163 9/25/70

TABLE 2-44
NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)

DATA POINT LOCATION -- LTS-3

$R = 49.53 \text{ cm.}$
 $Z = 288.80 \text{ cm.}$

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncert. & Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	1.42 (6)	2.01 (6)	2.81 (6)	4.22 (6)	1.8 (7)	4.27
Fast Neutron Flux ($E > 1.0 \text{ Mev}$) (N/cm ² -sec)	1.84 (11)	1.12 (11)	3.47 (11)	6.94 (11)	2.8 (14)	403.0
Int. Neutron Flux ($0.4 \text{ ev} < E < 1.0 \text{ Mev}$) (N/cm ² -sec)	4.65 (11)	2.83 (11)	8.77 (11)	1.75 (12)	N/A	N/A
Thermal Neutron Flux ($E < 0.4 \text{ ev}$) (N/cm ² -sec)	9.11 (10)	5.54 (10)	1.72 (11)	3.44 (11)	N/A	N/A

NOTE: 1) Numbers in parentheses refer to powers of ten.
2) Data location coordinates are referenced to CRAM model 0.0.

113 10/2/70 164 9/25/70
179 9/25/70 163 9/25/70

TABLE 2-45
RELATIVE CONTRIBUTION FROM VARIOUS SOURCE ZONES TO THE OVERALL PHOTON DOSE RATE
AT SEVERAL LOCATIONS ON A 10-FOOT MERIDIAN RING
(R-1 REFERENCE DESIGN REACTOR WITH COMPOSITE FUEL)

SOURCE ZONE	RELATIVE DOSE RATE (%)		
	0° POLAR ANGLE	90° POLAR ANGLE	180° POLAR ANGLE
CORE	28.2	75.5	87.4
REFLECTOR	3.8	10.8	2.6
AI STRUCTURE + LATERAL SUPPORT + CORE PERIPHERY	2.0	2.6	3.1
CORE PLENUM	17.4	2.3	Negligible
SUPPORT PLATE	29.2	2.0	Negligible
SHIELD (PLATES + PLENA)	10.0	1.4	Negligible
PRESSURE VESSEL	1.3	1.3	0.37
CHESH	Negligible	Negligible	4.9
OTHER	8.1	4.1	1.63
TOTAL	100.0	100.0	100.0
TOTAL DOSE RATE (Rads(C)/hr)	9.64×10^5	6.52×10^7	6.35×10^7

047

10/3/70

041

9/30/70

TABLE 2-46

DELAYED GAMMA RAY DOSE RATE VERSUS TIME AFTER SHUTDOWN FOLLOWING
A 10-MINUTE RUN AT FULL POWER (1511 MW THERMAL POWER)--COMPOSITE CORE

	Polar Angle						
Time	0.0 DEG.	2.5 DEG.	5.0 DEG.	7.5 DEG.	10.0 DEG.	12.5 DEG.	15.0 DEG.
1.8(3) SEC.	1.41255E+03	1.39247E+03	1.34293E+03	1.26806E+03	1.16892E+03	1.04353E+03	9.90952E+02
2.4(3) SEC.	9.19173E+02	9.06162E+02	8.73581E+02	8.24752E+02	7.59762E+02	6.77896E+02	6.43946E+02
3.0(3) SEC.	6.43697E+02	6.34623E+02	6.11640E+02	5.77396E+02	5.31649E+02	4.74200E+02	4.50599E+02
3.6(3) SEC.	4.78430E+02	4.71706E+02	4.54538E+02	4.29059E+02	3.94935E+02	3.52181E+02	3.34749E+02
4.8(3) SEC.	3.00196E+02	2.95986E+02	2.85176E+02	2.69174E+02	2.47705E+02	2.20878E+02	2.10040E+02
6.0(3) SEC.	2.12468E+02	2.09484E+02	2.01839E+02	1.90511E+02	1.75320E+02	1.56358E+02	1.48728E+02
7.2(3) SEC.	1.61715E+02	1.59440E+02	1.53629E+02	1.45002E+02	1.33450E+02	1.19031E+02	1.13238E+02
1.8(4) SEC.	4.71692E+01	4.65021E+01	4.48105E+01	4.22879E+01	3.89242E+01	3.47126E+01	3.30102E+01
3.6(4) SEC.	1.57920E+01	1.55676E+01	1.49902E+01	1.41356E+01	1.29948E+01	1.15722E+01	1.10082E+01
7.2(4) SEC.	3.51055E+00	3.45974E+00	3.32499E+00	3.12831E+00	2.86585E+00	2.54409E+00	2.42714E+00
8.64(4) SEC.	2.36806E+00	2.33374E+00	2.24226E+00	2.10367E+00	1.93071E+00	1.71356E+00	1.63702E+00
1.728(5) SEC.	1.14776E+00	1.13158E+00	1.09159E+00	1.03019E+00	9.49533E-01	8.49972E-01	8.12997E-01
4.320(5) SEC.	1.17152E+00	1.15561E+00	1.11805E+00	1.05860E+00	9.80667E-01	8.82267E-01	8.41484E-01
6.050(5) SEC.	1.12338E+00	1.10819E+00	1.07252E+00	1.01585E+00	9.41578E-01	8.47532E-01	8.08065E-01
8.640(5) SEC.	9.92426E-01	9.79040E-01	9.47721E-01	8.97842E-01	8.32491E-01	7.49599E-01	7.14550E-01
1.728(6) SEC.	5.86044E-01	5.78150E-01	5.59763E-01	5.30401E-01	4.91953E-01	4.43114E-01	4.22334E-01
2.592(6) SEC.	3.40721E-01	3.36130E-01	3.25441E-01	3.08307E-01	2.86015E-01	2.57627E-01	2.45562E-01
5.184(6) SEC.	6.80337E-02	6.71111E-02	6.49585E-02	6.15262E-02	5.70368E-02	5.13608E-02	4.89984E-02
7.776(6) SEC.	1.41326E-02	1.39365E-02	1.34744E-02	1.27428E-02	1.17888E-02	1.06029E-02	1.01493E-02
1.037(7) SEC.	3.38300E-03	3.33271E-03	3.21110E-03	3.02218E-03	2.77808E-03	2.48893E-03	2.40728E-03
1.296(7) SEC.	1.15395E-03	1.13475E-03	1.08642E-03	1.01339E-03	9.20359E-04	8.18210E-04	8.06724E-04
1.555(7) SEC.	6.22406E-04	6.11251E-04	5.82396E-04	5.39518E-04	4.85409E-04	4.28647E-04	4.28670E-04
2.074(7) SEC.	3.54343E-04	3.48013E-04	3.31454E-04	3.06875E-04	2.75957E-04	2.43105E-04	2.42577E-04
2.592(7) SEC.	2.46299E-04	2.42112E-04	2.31169E-04	2.14790E-04	1.94161E-04	1.71277E-04	1.68942E-04
3.154(7) SEC.	1.77846E-04	1.74981E-04	1.67505E-04	1.56212E-04	1.41962E-04	1.25419E-04	1.22282E-04

Note: 3.6(3) sec = 1 hour
 3.6(4) sec = 10 hours
 8.64(4) sec = 1 day
 2.592(6) sec = 1 month
 3.154(7) sec = 1 year

NOTE: Units are Rads-C/hr
 Meridian ring is located at 10 feet
 from the core center.

475 10/16/70

TABLE 2-46
(Continued)

Time	Polar Angle						
	17.5 DEG.	20.0 DEG.	25.0 DEG.	30.0 DEG.	40.0 DEG.	50.0 DEG.	60.0 DEG.
1.8(3) SEC.	9.22691E+02	8.46337E+02	4.50244E+03	2.24737E+04	9.93907E+04	1.95134E+05	2.84383E+05
2.4(3) SEC.	5.99814E+02	5.49489E+02	2.99352E+03	1.52732E+04	6.86704E+04	1.35761E+05	1.98760E+05
3.0(3) SEC.	4.19870E+02	3.84309E+02	2.13319E+03	1.10800E+04	5.04997E+04	1.00412E+05	1.47567E+05
3.6(3) SEC.	3.12017E+02	2.85415E+02	1.60776E+03	8.47212E+03	3.90409E+04	7.79885E+04	1.14964E+05
4.8(3) SEC.	1.95880E+02	1.79088E+02	1.02727E+03	5.52130E+03	2.58444E+04	5.19725E+04	7.69566E+04
6.0(3) SEC.	1.38753E+02	1.26850E+02	7.33965E+02	3.99000E+03	1.88550E+04	3.80137E+04	5.65344E+04
7.2(3) SEC.	1.05671E+02	9.66008E+01	5.61865E+02	3.07696E+03	1.46308E+04	2.96228E+04	4.40664E+04
1.8(4) SEC.	3.08064E+01	2.81480E+01	1.63297E+02	8.91775E+02	4.23235E+03	8.56345E+03	1.27338E+04
3.6(4) SEC.	1.02950E+01	9.36245E+00	5.76911E+01	3.30610E+02	1.61951E+03	3.31716E+03	4.97082E+03
7.2(4) SEC.	2.28674E+00	2.05257E+00	1.55931E+01	1.04452E+02	5.63337E+02	1.19645E+03	1.83441E+03
8.64(4) SEC.	1.54566E+00	1.38413E+00	1.10864E+01	7.71104E+01	4.25241E+02	9.10708E+02	1.40366E+03
1.728(5) SEC.	7.64200E-01	6.98789E-01	4.64980E+00	3.01879E+01	1.63802E+02	3.50528E+02	5.40926E+02
4.320(5) SEC.	7.85211E-01	7.30207E-01	3.52666E+00	1.71369E+01	7.75071E+01	1.55552E+02	2.30662E+02
6.050(5) SEC.	7.53416E-01	7.01894E-01	3.24613E+00	1.48897E+01	6.40701E+01	1.25879E+02	1.84041E+02
8.640(5) SEC.	6.65913E-01	6.21067E-01	2.79571E+00	1.23355E+01	5.11585E+01	9.88320E+01	1.42834E+02
1.728(6) SEC.	3.93445E-01	3.67304E-01	1.61640E+00	6.89334E+00	2.76099E+01	5.24439E+01	7.48914E+01
2.592(6) SEC.	2.28775E-01	2.13574E-01	9.43158E-01	4.04792E+00	1.63201E+01	3.10935E+01	4.44991E+01
5.184(6) SEC.	4.56957E-02	4.25896E-02	2.01427E-01	9.60955E-01	4.28276E+00	8.53539E+00	1.26009E+01
7.776(6) SEC.	9.50271E-03	8.80012E-03	5.23356E-02	3.21849E-01	1.70923E+00	3.63515E+00	5.59100E+00
1.037(7) SEC.	2.28119E-03	2.07080E-03	2.01899E-02	1.66173E-01	1.00591E+00	2.22498E+00	3.50071E+00
1.296(7) SEC.	7.81243E-04	6.83292E-04	1.15900E-02	1.11322E-01	7.08447E-01	1.58774E+00	2.51632E+00
1.555(7) SEC.	4.21696E-04	3.58324E-04	8.09148E-03	8.13846E-02	5.24521E-01	1.17913E+00	1.87182E+00
2.074(7) SEC.	2.38255E-04	2.01854E-04	4.55832E-03	4.57332E-02	2.94332E-01	6.61285E-01	1.04942E+00
2.592(7) SEC.	1.63997E-04	1.40991E-04	2.68094E-03	2.61228E-02	1.66651E-01	3.73585E-01	5.92147E-01
3.154(7) SEC.	1.17277E-04	1.02383E-04	1.57382E-03	1.46602E-02	9.22491E-02	2.06096E-01	3.26077E-01

475 10/16/70

TABLE 2-46
(Continued)

Time	Polar Angle						
	70.0 DEG.	80.0 DEG.	90.0 DEG.	100.0 DEG.	110.0 DEG.	120.0 DEG.	130.0 DEG.
1.8(3) SEC.	3.46540E+05	3.98174E+05	4.13064E+05	3.88822E+05	3.29022E+05	2.57014E+05	2.03150E+05
2.4(3) SEC.	2.42822E+05	2.79644E+05	2.90277E+05	2.73019E+05	2.30475E+05	1.79277E+05	1.41347E+05
3.0(3) SEC.	1.80661E+05	2.08449E+05	2.16483E+05	2.03475E+05	1.71427E+05	1.32881E+05	1.04538E+05
3.6(3) SEC.	1.40986E+05	1.62918E+05	1.69264E+05	1.59007E+05	1.33750E+05	1.03384E+05	8.11830E+04
4.8(3) SEC.	9.46078E+04	1.09561E+05	1.13895E+05	1.06909E+05	8.97192E+04	6.90688E+04	5.40839E+04
6.0(3) SEC.	6.96087E+04	8.07186E+04	8.39415E+04	7.87538E+04	6.59951E+04	5.06768E+04	3.96078E+04
7.2(3) SEC.	5.43120E+04	6.30348E+04	6.55670E+04	6.14950E+04	5.14833E+04	3.94686E+04	3.08089E+04
1.8(4) SEC.	1.56931E+04	1.82116E+04	1.89429E+04	1.77668E+04	1.48752E+04	1.14064E+04	8.90331E+03
3.6(4) SEC.	6.15328E+03	7.16777E+03	7.46331E+03	6.99042E+03	5.82858E+03	4.44344E+03	3.44698E+03
7.2(4) SEC.	2.29989E+03	2.70823E+03	2.82799E+03	2.63845E+03	2.17425E+03	1.62083E+03	1.23858E+03
8.64(4) SEC.	1.76501E+03	2.08356E+03	2.17710E+03	2.02936E+03	1.66785E+03	1.23717E+03	9.41271E+02
1.728(5) SEC.	6.81066E+02	8.04924E+02	8.41328E+02	7.83857E+02	6.43463E+02	4.76057E+02	3.60289E+02
4.320(5) SEC.	2.84526E+02	3.30449E+02	3.43857E+02	3.22356E+02	2.69689E+02	2.06235E+02	1.59610E+02
6.050(5) SEC.	2.25240E+02	2.59799E+02	2.69856E+02	2.53609E+02	2.13759E+02	1.65532E+02	1.29236E+02
8.640(5) SEC.	1.73649E+02	1.99112E+02	2.06500E+02	1.94484E+02	1.64970E+02	1.29108E+02	1.01541E+02
1.728(6) SEC.	9.04031E+01	1.02997E+02	1.06638E+02	1.00668E+02	8.59803E+01	6.80498E+01	5.39400E+01
2.592(6) SEC.	5.37746E+01	6.13268E+01	6.35108E+01	5.99331E+01	5.11341E+01	4.04010E+01	3.19936E+01
5.184(6) SEC.	1.54894E+01	1.79358E+01	1.86482E+01	1.75005E+01	1.46885E+01	1.12945E+01	8.78365E+00
7.776(6) SEC.	7.02047E+00	8.27969E+00	8.61882E+00	8.06360E+00	6.63510E+00	4.92900E+00	3.74139E+00
1.037(7) SEC.	4.44498E+00	5.29128E+00	5.54005E+00	5.14833E+00	4.19366E+00	3.05881E+00	2.29123E+00
1.296(7) SEC.	3.20595E+00	3.82708E+00	4.00979E+00	3.72262E+00	3.02306E+00	2.19263E+00	1.63587E+00
1.555(7) SEC.	2.38653E+00	2.85059E+00	2.98711E+00	2.77260E+00	2.25012E+00	1.63009E+00	1.21528E+00
2.074(7) SEC.	1.33773E+00	1.59761E+00	1.67405E+00	1.55391E+00	1.26129E+00	9.14052E-01	6.81665E-01
2.592(7) SEC.	7.54459E-01	9.00682E-01	9.43680E-01	8.76073E-01	7.11404E-01	5.15978E-01	3.84955E-01
3.154(7) SEC.	4.15180E-01	4.95388E-01	5.18966E-01	4.81871E-01	3.91528E-01	2.84295E-01	2.12182E-01

475 10/16/70

TABLE 2-46
(Continued)

Time	Polar Angle				
	140.0 DEG.	150.0 DEG.	160.0 DEG.	170.0 DEG.	180.0 DEG.
1.8(3) SEC.	2.09270E+05	2.83808E+05	3.92155E+05	4.54417E+05	4.74900E+05
2.4(3) SEC.	1.46617E+05	2.00733E+05	2.78576E+05	3.23078E+05	3.37752E+05
3.0(3) SEC.	1.09068E+05	1.50498E+05	2.09598E+05	2.43243E+05	2.54357E+05
3.6(3) SEC.	8.51039E+04	1.18173E+05	1.65038E+05	1.91629E+05	2.00426E+05
4.8(3) SEC.	5.70970E+04	8.00113E+04	1.12189E+05	1.30357E+05	1.36379E+05
6.0(3) SEC.	4.20014E+04	5.91940E+04	8.32031E+04	9.67194E+04	1.01205E+05
7.2(3) SEC.	3.27649E+04	4.63463E+04	6.52478E+04	7.58691E+04	7.93960E+04
1.8(4) SEC.	9.44932E+03	1.33463E+04	1.87878E+04	2.18503E+04	2.28674E+04
3.6(4) SEC.	3.69338E+03	5.28860E+03	7.49567E+03	8.73188E+03	9.14392E+03
7.2(4) SEC.	1.36535E+03	2.03465E+03	2.93826E+03	3.43682E+03	3.60466E+03
8.64(4) SEC.	1.04326E+03	1.56813E+03	2.27421E+03	2.66266E+03	2.79378E+03
1.728(5) SEC.	3.96557E+02	5.96411E+02	8.67601E+02	1.01689E+03	1.06757E+03
4.320(5) SEC.	1.65010E+02	2.30536E+02	3.25648E+02	3.79648E+02	3.97888E+02
6.050(5) SEC.	1.30973E+02	1.77891E+02	2.47999E+02	2.88334E+02	3.01903E+02
8.640(5) SEC.	1.01340E+02	1.34416E+02	1.85173E+02	2.14724E+02	2.24622E+02
1.728(6) SEC.	5.30594E+01	6.86541E+01	9.33096E+01	1.07856E+02	1.12698E+02
2.592(6) SEC.	3.16349E+01	4.11801E+01	5.60920E+01	6.48503E+01	6.77652E+01
5.184(6) SEC.	9.13538E+00	1.27258E+01	1.78672E+01	2.07740E+01	2.17490E+01
7.776(6) SEC.	4.15211E+00	6.24463E+00	9.04341E+00	1.05736E+01	1.10906E+01
1.037(7) SEC.	2.63728E+00	4.11892E+00	6.04731E+00	7.08683E+00	7.43890E+00
1.296(7) SEC.	1.90649E+00	3.01221E+00	4.43925E+00	5.20522E+00	5.46470E+00
1.555(7) SEC.	1.42098E+00	2.25110E+00	3.31993E+00	3.89299E+00	4.08709E+00
2.074(7) SEC.	7.96748E-01	1.26152E+00	1.86003E+00	2.18098E+00	2.28966E+00
2.592(7) SEC.	4.48477E-01	7.08488E-01	1.04422E+00	1.22449E+00	1.24555E+00
3.154(7) SEC.	2.45756E-01	3.86818E-01	5.69906E-01	6.68439E-01	7.01841E-01

475 10/16/70

TABLE 2-47

DELAYED GAMMA RAY DOSE RATE VERSUS TIME AFTER SHUTDOWN FOLLOWING
A 30-MINUTE RUN AT FULL POWER (1511 MW THERMAL POWER)--COMPOSITE CORE

Time	Polar Angle								
	0.0 DEG.	2.5 DEG.	5.0 DEG.	7.5 DEG.	10.0 DEG.	12.5 DEG.	15.0 DEG.		
1.8(3) SEC.	2.97572E+03	2.93355E+03	2.82845E+03	2.67048E+03	2.46059E+03	2.19585E+03	2.08571E+03		
2.4(3) SEC.	2.03725E+03	2.00850E+03	1.93590E+03	1.82755E+03	1.68295E+03	1.50121E+03	1.42638E+03		
3.0(3) SEC.	1.49608E+03	1.47503E+03	1.42143E+03	1.34178E+03	1.23519E+03	1.10156E+03	1.04697E+03		
3.6(3) SEC.	1.15020E+03	1.13405E+03	1.09271E+03	1.03144E+03	9.49299E+02	8.46514E+02	8.04774E+02		
4.8(3) SEC.	7.61861E+02	7.51171E+02	7.23750E+02	6.83138E+02	6.28663E+02	5.60027E+02	5.33192E+02		
6.0(3) SEC.	5.59249E+02	5.51390E+02	5.31278E+02	5.01452E+02	4.61480E+02	4.11585E+02	3.91523E+02		
7.2(3) SEC.	4.36084E+02	4.29941E+02	4.14282E+02	3.91016E+02	3.59879E+02	3.21009E+02	3.05390E+02		
1.8(4) SEC.	1.35412E+02	1.33496E+02	1.28640E+02	1.21398E+02	1.11742E+02	9.96516E+01	9.47631E+01		
3.6(4) SEC.	4.59552E+01	4.53018E+01	4.36214E+01	4.11342E+01	3.78140E+01	3.36744E+01	3.20337E+01		
7.2(4) SEC.	1.03278E+01	1.01783E+01	9.78170E+00	9.20288E+00	8.43048E+00	7.48381E+00	7.14028E+00		
8.64(4) SEC.	7.00516E+00	6.90362E+00	6.63307E+00	6.23789E+00	5.71155E+00	5.06935E+00	4.84315E+00		
1.728(5) SEC.	3.44456E+00	3.39602E+00	3.27610E+00	3.09190E+00	2.84996E+00	2.55127E+00	2.44028E+00		
4.320(5) SEC.	3.51685E+00	3.46908E+00	3.35635E+00	3.17789E+00	2.94395E+00	2.64857E+00	2.52614E+00		
6.050(5) SEC.	3.37047E+00	3.32491E+00	3.21789E+00	3.04786E+00	2.82503E+00	2.54286E+00	2.42446E+00		
8.640(5) SEC.	2.97516E+00	2.93502E+00	2.84113E+00	2.69160E+00	2.49569E+00	2.24719E+00	2.14210E+00		
1.728(6) SEC.	1.75908E+00	1.73538E+00	1.68019E+00	1.59206E+00	1.47666E+00	1.33007E+00	1.26769E+00		
2.592(6) SEC.	1.02199E+00	1.00822E+00	9.76159E-01	9.24946E-01	8.57901E-01	7.72755E-01	7.36567E-01		
5.184(6) SEC.	2.03934E-01	2.01170E-01	1.94718E-01	1.84430E-01	1.70973E-01	1.53958E-01	1.46878E-01		
7.776(6) SEC.	4.23985E-02	4.18099E-02	4.04237E-02	3.82285E-02	3.53665E-02	3.18086E-02	3.04476E-02		
1.037(7) SEC.	1.01608E-02	1.00098E-02	9.64459E-03	9.07723E-03	8.34415E-03	7.47570E-03	7.23035E-03		
1.296(7) SEC.	3.45666E-03	3.39914E-03	3.25429E-03	3.03546E-03	2.75670E-03	2.45069E-03	2.41646E-03		
1.555(7) SEC.	1.86369E-03	1.83028E-03	1.74385E-03	1.61542E-03	1.45335E-03	1.28339E-03	1.28357E-03		
2.074(7) SEC.	1.06446E-03	1.04545E-03	9.95700E-04	9.21859E-04	8.28979E-04	7.30289E-04	7.28717E-04		
2.592(7) SEC.	7.40116E-04	7.27542E-04	6.94680E-04	6.45491E-04	5.83536E-04	5.14772E-04	5.07683E-04		
3.154(7) SEC.	5.32321E-04	5.23749E-04	5.01385E-04	4.67546E-04	4.24961E-04	3.75447E-04	3.66016E-04		

NOTE: Units are Rads-C/hr
Meridian ring is located at 10 feet
from the core center.

472 10/16/70

TABLE 2-47
(Continued)

Time	Polar Angle						
	17.5 DEG.	20.0 DEG.	25.0 DEG.	30.0 DEG.	40.0 DEG.	50.0 DEG.	60.0 DEG.
1.8(3) SEC.	1.94255E+03	1.78031E+03	9.62884E+03	4.88183E+04	2.18496E+05	4.31157E+05	6.30469E+05
2.4(3) SEC.	1.32899E+03	1.21668E+03	6.72384E+03	3.47828E+04	1.58051E+05	3.13867E+05	4.60879E+05
3.0(3) SEC.	9.75797E+02	8.92773E+02	5.00970E+03	2.63061E+04	1.20912E+05	2.41280E+05	3.55431E+05
3.6(3) SEC.	7.50294E+02	6.86176E+02	3.89574E+03	2.07058E+04	9.60725E+04	1.92480E+05	2.94299E+05
4.8(3) SEC.	4.97334E+02	4.54696E+02	2.61817E+03	1.41448E+04	6.65005E+04	1.33988E+05	1.98657E+05
6.0(3) SEC.	3.65304E+02	3.33955E+02	1.93719E+03	1.05670E+04	5.00767E+04	1.01242E+05	1.50455E+05
7.2(3) SEC.	2.84997E+02	2.60537E+02	1.51652E+03	8.31701E+03	3.95988E+04	8.02226E+04	1.19387E+05
1.8(4) SEC.	8.84362E+01	8.08046E+01	4.68531E+02	2.55703E+03	1.21796E+04	2.45373E+04	3.64820E+04
3.6(4) SEC.	2.99582E+01	2.72440E+01	1.68053E+02	9.64157E+02	4.72725E+03	9.68645E+03	1.45192E+04
7.2(4) SEC.	6.72795E+00	6.03821E+00	4.60006E+01	3.08756E+02	1.66717E+03	3.54239E+03	5.43275E+03
8.64(4) SEC.	4.57302E+00	4.09527E+00	3.28246E+01	2.28490E+02	1.26077E+03	2.70075E+03	4.16325E+03
1.728(5) SEC.	2.29370E+00	2.09769E+00	1.39324E+01	9.03544E+01	4.90020E+02	1.04875E+03	1.61780E+03
4.320(5) SEC.	2.35719E+00	2.19211E+00	1.05840E+01	5.14125E+01	2.32463E+02	4.66484E+02	6.91676E+02
6.050(5) SEC.	2.26049E+00	2.10593E+00	9.73877E+00	4.46674E+01	1.92192E+02	3.77592E+02	5.52049E+02
8.640(5) SEC.	1.99629E+00	1.86185E+00	8.37975E+00	3.69633E+01	1.53250E+02	2.96017E+02	4.27764E+02
1.728(6) SEC.	1.18098E+00	1.10252E+00	4.85118E+00	2.06841E+01	8.28284E+01	1.57313E+02	2.24632E+02
2.592(6) SEC.	6.86215E-01	6.40621E-01	2.82895E+00	1.21412E+01	4.89493E+01	9.32589E+01	1.33466E+02
5.184(6) SEC.	1.36979E-01	1.27669E-01	6.03816E-01	2.88086E+00	1.28405E+01	2.55918E+01	3.77826E+01
7.776(6) SEC.	2.85078E-02	2.63996E-02	1.57017E-01	9.65660E-01	5.12839E+00	1.09071E+01	1.67756E+01
1.037(7) SEC.	6.85151E-03	6.21981E-03	6.06022E-02	4.98655E-01	3.01824E+00	6.67589E+00	1.05034E+01
1.296(7) SEC.	2.34030E-03	2.04665E-03	3.47639E-02	3.34003E-01	2.12576E+00	4.76428E+00	7.55071E+00
1.555(7) SEC.	1.26279E-03	1.07291E-03	2.42550E-02	2.43994E-01	1.57256E+00	3.53613E+00	5.41185E+00
2.074(7) SEC.	7.15742E-04	6.06397E-04	1.36962E-02	1.37423E-01	8.84399E-01	1.98701E+00	3.15326E+00
2.592(7) SEC.	4.92751E-04	4.23706E-04	8.03772E-03	7.82830E-02	4.99331E-01	1.11931E+00	1.77411E+00
3.154(7) SEC.	3.50996E-04	3.06461E-04	4.70076E-03	4.37658E-02	2.75356E-01	6.15161E-01	9.73272E-01

472 10/16/70

TABLE 2-47
(Continued)

Time	Polar Angle						
	70.0 DEG.	80.0 DEG.	90.0 DEG.	100.0 DEG.	110.0 DEG.	120.0 DEG.	130.0 DEG.
1.8(3) SEC.	7.69715E+05	8.85897E+05	9.19436E+05	8.64956E+05	7.30633E+05	5.68963E+05	4.48878E+05
2.4(3) SEC.	5.63978E+05	6.50459E+05	6.75454E+05	6.34960E+05	5.35184E+05	4.15164E+05	3.26766E+05
3.0(3) SEC.	4.35715E+05	5.03324E+05	5.22884E+05	4.91257E+05	4.13372E+05	3.19725E+05	2.51166E+05
3.6(3) SEC.	3.49029E+05	4.03711E+05	4.19544E+05	3.93983E+05	3.31062E+05	2.55440E+05	2.00336E+05
4.8(3) SEC.	2.44398E+05	2.83204E+05	2.94455E+05	2.76330E+05	2.31742E+05	1.78191E+05	1.39409E+05
6.0(3) SEC.	1.85333E+05	2.14997E+05	2.23605E+05	2.09755E+05	1.75698E+05	1.34817E+05	1.05311E+05
7.2(3) SEC.	1.47178E+05	1.70849E+05	1.77722E+05	1.66673E+05	1.39507E+05	1.06911E+05	8.34279E+04
1.8(4) SEC.	4.49576E+04	5.21694E+04	5.42636E+04	5.08958E+04	4.26149E+04	3.26809E+04	2.55109E+04
3.6(4) SEC.	1.79760E+04	2.09426E+04	2.18069E+04	2.04241E+04	1.70270E+04	1.29596E+04	1.00445E+04
7.2(4) SEC.	6.81234E+03	8.02288E+03	8.37792E+03	7.81606E+03	6.44004E+03	4.79960E+03	3.66689E+03
8.64(4) SEC.	5.23550E+03	6.18089E+03	6.45851E+03	6.02006E+03	4.94723E+03	3.66918E+03	2.79116E+03
1.728(5) SEC.	2.03683E+03	2.40715E+03	2.51599E+03	2.34415E+03	1.92439E+03	1.42385E+03	1.07766E+03
4.320(5) SEC.	8.53157E+02	9.90819E+02	1.03101E+03	9.66554E+02	8.08674E+02	6.18449E+02	4.78658E+02
6.050(5) SEC.	6.75620E+02	7.79277E+02	8.09441E+02	7.60711E+02	6.41186E+02	4.96531E+02	3.87660E+02
8.640(5) SEC.	5.20020E+02	5.96242E+02	6.18355E+02	5.82385E+02	4.94035E+02	3.86674E+02	3.04132E+02
1.728(6) SEC.	2.71147E+02	3.08909E+02	3.19824E+02	3.01923E+02	2.57883E+02	2.04117E+02	1.41801E+02
2.592(6) SEC.	1.61285E+02	1.83936E+02	1.90486E+02	1.79756E+02	1.53366E+02	1.21174E+02	9.59584E+01
5.184(6) SEC.	4.64441E+01	5.37803E+01	5.59166E+01	5.24749E+01	4.40426E+01	3.38650E+01	2.63361E+01
7.776(6) SEC.	2.10648E+01	2.48431E+01	2.59507E+01	2.41947E+01	1.99084E+01	1.47893E+01	1.12258E+01
1.037(7) SEC.	1.33365E+01	1.58756E+01	1.66219E+01	1.54467E+01	1.25825E+01	9.17763E+00	6.87467E+00
1.296(7) SEC.	9.62016E+00	1.14840E+01	1.20323E+01	1.11706E+01	9.07134E+00	6.57937E+00	4.90869E+00
1.555(7) SEC.	7.15496E+00	8.54620E+00	8.95546E+00	8.31239E+00	6.74599E+00	4.88715E+00	3.64356E+00
2.074(7) SEC.	4.01956E+00	4.80044E+00	5.03012E+00	4.66914E+00	3.78989E+00	2.74651E+00	2.04825E+00
2.592(7) SEC.	2.26038E+00	2.69843E+00	2.82725E+00	2.62471E+00	2.13139E+00	1.54592E+00	1.15339E+00
3.154(7) SEC.	1.23922E+00	1.47863E+00	1.54900E+00	1.43828E+00	1.16863E+00	8.48561E-01	6.33314E-01

TABLE 2-47
(Continued)

Time	Polar Angle				
	140.0 DEG.	150.0 DEG.	160.0 DEG.	170.0 DEG.	180.0 DEG.
1.8(3) SEC.	4.64765E+05	6.34732E+05	8.79875E+05	1.02020E+06	1.06645E+06
2.4(3) SEC.	3.40501E+05	4.69049E+05	6.52742E+05	7.57409E+05	7.91972E+05
3.0(3) SEC.	2.63020E+05	3.64712E+05	5.09037E+05	5.90983E+05	6.18085E+05
3.6(3) SEC.	2.10665E+05	2.93711E+05	4.10923E+05	4.77282E+05	4.99255E+05
4.8(3) SEC.	1.47482E+05	2.07222E+05	2.90892E+05	3.38070E+05	3.53716E+05
6.0(3) SEC.	1.11824E+05	1.57862E+05	2.22049E+05	2.58153E+05	2.70137E+05
7.2(3) SEC.	8.87777E+04	1.25677E+05	1.76997E+05	2.05824E+05	2.15397E+05
1.8(4) SEC.	2.70671E+04	3.82179E+04	5.37946E+04	6.25630E+04	6.54753E+04
3.6(4) SEC.	1.07864E+04	1.54523E+04	2.19065E+04	2.55210E+04	2.67259E+04
7.2(4) SEC.	4.04349E+03	6.02844E+03	8.70764E+03	1.01856E+04	1.06832E+04
8.64(4) SEC.	3.09387E+03	4.65152E+03	6.74691E+03	7.89961E+03	8.28873E+03
1.728(5) SEC.	1.18606E+03	1.78357E+03	2.59438E+03	3.04072E+03	3.19226E+03
4.320(5) SEC.	4.94814E+02	6.91212E+02	9.76312E+02	1.13819E+03	1.19286E+03
6.050(5) SEC.	3.92866E+02	5.33587E+02	7.43868E+02	8.64848E+02	9.05545E+02
8.640(5) SEC.	3.03483E+02	4.02442E+02	5.54348E+02	6.42802E+02	6.72425E+02
1.728(6) SEC.	1.59143E+02	2.05884E+02	2.79799E+02	3.23413E+02	3.37931E+02
2.592(6) SEC.	9.48826E+01	1.23511E+02	1.68235E+02	1.94504E+02	2.03246E+02
5.184(6) SEC.	2.73918E+01	3.81598E+01	5.35782E+01	6.22949E+01	6.52188E+01
7.776(6) SEC.	1.24579E+01	1.87363E+01	2.71339E+01	3.17252E+01	3.32764E+01
1.037(7) SEC.	7.91281E+00	1.23580E+01	1.81435E+01	2.12623E+01	2.23186E+01
1.296(7) SEC.	5.72085E+00	9.03900E+00	1.33213E+01	1.56198E+01	1.63985E+01
1.555(7) SEC.	4.26048E+00	6.74946E+00	9.95399E+00	1.16721E+01	1.22540E+01
2.074(7) SEC.	2.39407E+00	3.79062E+00	5.58903E+00	6.55341E+00	6.87998E+00
2.592(7) SEC.	1.34368E+00	2.12261E+00	3.12841E+00	3.66846E+00	3.85139E+00
3.154(7) SEC.	7.33455E-01	1.15441E+00	1.70082E+00	1.99490E+00	2.09459E+00

472 10/16/70

TABLE 2-48

DELAYED GAMMA RAY DOSE RATE VERSUS TIME AFTER SHUTDOWN FOLLOWING
A 60-MINUTE RUN AT FULL POWER (1511 MW THERMAL POWER)--COMPOSITE CORE

	Polar Angle							
Time	0.0 DEG.	2.5 DEG.	5.0 DEG.	7.5 DEG.	10.0 DEG.	12.5 DEG.	15.0 DEG.	
1.8(3) SEC.	4.12462E+03	4.06632E+03	3.91992E+03	3.70076E+03	3.40882E+03	3.04143E+03	2.88960E+03	
2.4(3) SEC.	2.96001E+03	2.91831E+03	2.81250E+03	2.65498E+03	2.44442E+03	2.18026E+03	2.07208E+03	
3.0(3) SEC.	2.25637E+03	2.22466E+03	2.14368E+03	2.02350E+03	1.86254E+03	1.66100E+03	1.57903E+03	
3.6(3) SEC.	1.79502E+03	1.76982E+03	1.70527E+03	1.60961E+03	1.48136E+03	1.32101E+03	1.25611E+03	
4.8(3) SEC.	1.25184E+03	1.23422E+03	1.18923E+03	1.12248E+03	1.03300E+03	9.21274E+02	8.76282E+02	
6.0(3) SEC.	9.50537E+02	9.37160E+02	9.03013E+02	8.52310E+02	7.84419E+02	6.99666E+02	6.65596E+02	
7.2(3) SEC.	7.61677E+02	7.50937E+02	7.23604E+02	6.82956E+02	6.28593E+02	5.60716E+02	5.33436E+02	
1.8(4) SEC.	2.81742E+02	2.77770E+02	2.67704E+02	2.52596E+02	2.32575E+02	2.07323E+02	1.97013E+02	
3.6(4) SEC.	8.78408E+01	8.65912E+01	8.33755E+01	7.86177E+01	7.22662E+01	6.43507E+01	6.12190E+01	
7.2(4) SEC.	2.01047E+01	1.98136E+01	1.90409E+01	1.79134E+01	1.64088E+01	1.45855E+01	1.38981E+01	
8.64(4) SEC.	1.37577E+01	1.35583E+01	1.30270E+01	1.22508E+01	1.12170E+01	9.95601E+00	9.51249E+00	
1.728(5) SEC.	6.87476E+00	6.77789E+00	6.53870E+00	6.17122E+00	5.68856E+00	5.09261E+00	4.87100E+00	
4.320(5) SEC.	7.02946E+00	6.93399E+00	6.70868E+00	6.35199E+00	5.88440E+00	5.29398E+00	5.04923E+00	
6.050(5) SEC.	6.73825E+00	6.64715E+00	6.43319E+00	6.09329E+00	5.64781E+00	5.08373E+00	4.84699E+00	
8.640(5) SEC.	5.95628E+00	5.87595E+00	5.68799E+00	5.38864E+00	4.99643E+00	4.49893E+00	4.28857E+00	
1.728(6) SEC.	3.51224E+00	3.46492E+00	3.35472E+00	3.17874E+00	2.94831E+00	2.65561E+00	2.53107E+00	
2.592(6) SEC.	2.04337E+00	2.01584E+00	1.95173E+00	1.84934E+00	1.71528E+00	1.54504E+00	1.47264E+00	
5.184(6) SEC.	4.08336E-01	4.02798E-01	3.89878E-01	3.69276E-01	3.42330E-01	3.08263E-01	2.94084E-01	
7.776(6) SEC.	8.47697E-02	8.35929E-02	8.08212E-02	7.64324E-02	7.07102E-02	6.35968E-02	6.08761E-02	
1.037(7) SEC.	2.03249E-02	2.00228E-02	1.92923E-02	1.81574E-02	1.66910E-02	1.49540E-02	1.44631E-02	
1.296(7) SEC.	6.91870E-03	6.80356E-03	6.51369E-03	6.07576E-03	5.51788E-03	4.90542E-03	4.83676E-03	
1.555(7) SEC.	3.73343E-03	3.66652E-03	3.49345E-03	3.23625E-03	2.91170E-03	2.57122E-03	2.57132E-03	
2.074(7) SEC.	2.12879E-03	2.09076E-03	1.99128E-03	1.84360E-03	1.65785E-03	1.46049E-03	1.45735E-03	
2.592(7) SEC.	1.47777E-03	1.45264E-03	1.38698E-03	1.28872E-03	1.16494E-03	1.02764E-03	1.01363E-03	
3.154(7) SEC.	1.06442E-03	1.04727E-03	1.00254E-03	9.34963E-04	8.49693E-04	7.50684E-04	7.31866E-04	

Note: 3.6(3) sec = 1 hour
 3.6(4) sec = 10 hours
 8.64(4) sec = 1 day
 2.592(6) sec = 1 month
 3.154(7) sec = 1 year

474 10/16/70

NOTE: Units are Rads-C/hr
 Meridian ring is located at 10 feet
 from the core center.

TABLE 2-48
(Continued)

Time	Polar Angle						
	17.5 DEG.	20.0 DEG.	25.0 DEG.	30.0 DEG.	40.0 DEG.	50.0 DEG.	60.0 DEG.
1.8(3) SEC.	2.69204E+03	2.46576E+03	1.35213E+04	6.95136E+04	3.14547E+05	6.23620E+05	9.14767E+05
2.4(3) SEC.	1.93113E+03	1.76727E+03	9.87442E+03	5.16786E+04	2.37013E+05	4.72557E+05	6.95758E+05
3.0(3) SEC.	1.47208E+03	1.34649E+03	7.62349E+03	4.04322E+04	1.87339E+05	3.75129E+05	5.53891E+05
3.6(3) SEC.	1.17136E+03	1.07109E+03	6.12360E+03	3.28171E+04	1.53299E+05	3.08029E+05	4.55864E+05
4.8(3) SEC.	8.17491E+02	7.47374E+02	4.31979E+03	2.34586E+04	1.10768E+05	2.23601E+05	3.31948E+05
6.0(3) SEC.	6.21095E+02	5.67803E+02	3.29923E+03	1.80496E+04	8.57617E+04	1.73591E+05	2.58182E+05
7.2(3) SEC.	4.97843E+02	4.55103E+02	2.65178E+03	1.45673E+04	6.94583E+04	1.40804E+05	2.09637E+05
1.8(4) SEC.	1.83731E+02	1.67874E+02	9.61647E+02	5.18384E+03	2.43584E+04	4.90741E+04	7.27780E+04
3.6(4) SEC.	5.72624E+01	5.20589E+01	3.22705E+02	1.85930E+03	9.14246E+03	1.87548E+04	2.81325E+04
7.2(4) SEC.	1.30977E+01	1.17520E+01	8.99342E+01	6.05484E+02	3.27515E+03	6.96351E+03	1.06838E+04
8.64(4) SEC.	8.98265E+00	8.04403E+00	6.46053E+01	4.50446E+02	2.48798E+03	5.33163E+03	8.22081E+03
1.728(5) SEC.	4.57821E+00	4.18752E+00	2.77643E+01	1.79875E+02	9.75091E+02	2.08667E+03	3.21848E+03
4.320(5) SEC.	4.71150E+00	4.38157E+00	2.11472E+01	1.02670E+02	4.64031E+02	9.31015E+02	1.38031E+03
6.050(5) SEC.	4.51918E+00	4.21019E+00	1.94672E+01	8.92712E+01	3.84047E+02	7.54471E+02	1.10300E+03
8.640(5) SEC.	3.99664E+00	3.72752E+00	1.67756E+01	7.39939E+01	3.06770E+02	5.92548E+02	8.56267E+02
1.728(6) SEC.	2.35793E+00	2.20125E+00	9.68798E+00	4.13195E+01	1.65507E+02	3.14380E+02	4.48949E+02
2.592(6) SEC.	1.37201E+00	1.28084E+00	5.65664E+00	2.42798E+01	9.78987E+01	1.86526E+02	2.66953E+02
5.184(6) SEC.	2.74262E-01	2.55617E-01	1.20928E+00	5.77133E+00	2.57293E+01	5.12837E+01	7.57169E+01
7.776(6) SEC.	5.69977E-02	5.27830E-02	3.13952E-01	1.93091E+00	1.02549E+01	2.18103E+01	3.35454E+01
1.037(7) SEC.	1.37052E-02	1.24417E-02	1.21210E-01	9.97316E-01	6.03644E+00	1.33517E+01	2.10067E+01
1.296(7) SEC.	4.68418E-03	4.09664E-03	6.95424E-02	6.68066E-01	4.25174E+00	9.52896E+00	1.51070E+01
1.555(7) SEC.	2.52946E-03	2.14936E-03	4.85291E-02	4.88101E-01	3.14579E+00	7.07179E+00	1.12262E+01
2.074(7) SEC.	1.43141E-03	1.21272E-03	2.73917E-02	2.74839E-01	1.76874E+00	3.97387E+00	6.30629E+00
2.592(7) SEC.	9.83966E-04	8.45927E-04	1.60854E-02	1.56731E-01	9.99842E-01	2.24134E+00	3.55259E+00
3.154(7) SEC.	7.01872E-04	6.12774E-04	9.40938E-03	8.76264E-02	5.51347E-01	1.23175E+00	1.04882E+00

474 10/16/70

TABLE 2-48
(Continued)

Time	Polar Angle						
	70.0 DEG.	80.0 DEG.	90.0 DEG.	100.0 DEG.	110.0 DEG.	120.0 DEG.	130.0 DEG.
1.8(3) SEC.	1.11876E+06	1.28964E+06	1.33902E+06	1.25897E+06	1.06171E+06	8.24391E+05	6.49194E+05
2.4(3) SEC.	8.52666E+05	9.84710E+05	1.02291E+06	9.61125E+05	8.08963E+05	6.26006E+05	4.91902E+05
3.0(3) SEC.	6.79877E+05	7.86262E+05	8.17064E+05	7.67327E+05	6.44890E+05	4.97737E+05	3.90432E+05
3.6(3) SEC.	5.60264E+05	6.48653E+05	6.74265E+05	6.32964E+05	5.31332E+05	4.09233E+05	3.20544E+05
4.8(3) SEC.	4.08669E+05	4.73848E+05	4.92755E+05	4.62318E+05	3.87460E+05	2.97582E+05	2.32612E+05
6.0(3) SEC.	3.18178E+05	3.69247E+05	3.84072E+05	3.60231E+05	3.01611E+05	2.31262E+05	1.80541E+05
7.2(3) SEC.	2.58502E+05	3.00141E+05	3.12233E+05	2.92746E+05	2.45016E+05	1.87692E+05	1.46417E+05
1.8(4) SEC.	8.95522E+04	1.03789E+05	1.07918E+05	1.01262E+05	8.49002E+04	6.52640E+04	5.10284E+04
3.6(4) SEC.	3.48446E+04	4.06093E+04	4.22892E+04	3.96026E+04	3.30029E+04	2.51025E+04	1.94851E+04
7.2(4) SEC.	1.33999E+04	1.57840E+04	1.64834E+04	1.53768E+04	1.26671E+04	9.43694E+03	7.20748E+03
8.64(4) SEC.	1.03395E+04	1.22079E+04	1.27566E+04	1.18901E+04	9.76999E+03	7.24437E+03	5.50965E+03
1.728(5) SEC.	4.05224E+03	4.78887E+03	5.00537E+03	4.66356E+03	3.82856E+03	2.83286E+03	2.14412E+03
4.320(5) SEC.	1.70246E+03	1.97708E+03	2.05725E+03	1.92867E+03	1.61371E+03	1.23423E+03	9.55296E+02
6.050(5) SEC.	1.34986E+03	1.55693E+03	1.61718E+03	1.51984E+03	1.28107E+03	9.92098E+02	7.74589E+02
8.640(5) SEC.	1.04093E+03	1.19350E+03	1.23777E+03	1.16577E+03	9.88919E+02	7.74017E+02	6.08793E+02
1.728(6) SEC.	5.41938E+02	6.17436E+02	6.39260E+02	6.03470E+02	5.15424E+02	4.07937E+02	3.23355E+02
2.592(6) SEC.	3.22601E+02	3.67913E+02	3.81016E+02	3.59551E+02	3.06760E+02	2.42365E+02	1.91927E+02
5.184(6) SEC.	9.30771E+01	1.07781E+02	1.12063E+02	1.05165E+02	8.82637E+01	6.78651E+01	5.27759E+01
7.776(6) SEC.	4.21224E+01	4.96779E+01	5.18927E+01	4.83814E+01	3.98102E+01	2.95734E+01	2.24476E+01
1.037(7) SEC.	2.66727E+01	3.17508E+01	3.32435E+01	3.08930E+01	2.51646E+01	1.83551E+01	1.37492E+01
1.296(7) SEC.	1.92410E+01	2.29688E+01	2.40654E+01	2.23419E+01	1.81433E+01	1.31593E+01	9.81781E+00
1.555(7) SEC.	1.43132E+01	1.70964E+01	1.79152E+01	1.66287E+01	1.34951E+01	9.77640E+00	7.28855E+00
2.074(7) SEC.	8.03880E+00	9.60049E+00	1.00598E+01	9.33790E+00	7.57948E+00	5.49284E+00	4.09638E+00
2.592(7) SEC.	4.52635E+00	5.40357E+00	5.66153E+00	5.25593E+00	4.26804E+00	3.09563E+00	2.30959E+00
3.154(7) SEC.	2.48135E+00	2.96071E+00	3.10163E+00	2.87993E+00	2.33999E+00	1.69911E+00	1.26812E+00

474 10/16/70

TABLE 2-48
(Continued)

Time	Polar Angle				
	140.0 DEG.	150.0 DEG.	160.0 DEG.	170.0 DEG.	180.0 DEG.
1.8(3) SEC.	6.75437E+05	9.28507E+05	1.29092E+06	1.49763E+06	1.56586E+06
2.4(3) SEC.	5.14721E+05	7.12986E+05	9.94660E+05	1.15467E+06	1.20758E+06
3.0(3) SEC.	4.10361E+05	5.71755E+05	7.99689E+05	9.28776E+05	9.71512E+05
3.6(3) SEC.	3.38125E+05	4.73323E+05	6.63372E+05	7.70740E+05	8.06321E+05
4.8(3) SEC.	2.46581E+05	3.47363E+05	4.88169E+05	5.67457E+05	5.93765E+05
6.0(3) SEC.	1.91937E+05	2.71390E+05	3.82012E+05	4.44188E+05	4.64832E+05
7.2(3) SEC.	1.55905E+05	2.20893E+05	3.11215E+05	3.61930E+05	3.78776E+05
1.8(4) SEC.	5.39248E+04	7.57466E+04	1.06378E+05	1.23665E+05	1.29399E+05
3.6(4) SEC.	2.09032E+04	2.99852E+04	4.25361E+04	4.95609E+04	5.19035E+04
7.2(4) SEC.	7.95128E+03	1.18625E+04	1.71401E+04	2.00508E+04	2.10310E+04
8.64(4) SEC.	6.10879E+03	9.18807E+03	1.33297E+04	1.56077E+04	1.63768E+04
1.728(5) SEC.	2.35946E+03	3.54771E+03	5.16035E+03	6.04812E+03	6.34953E+03
4.320(5) SEC.	9.87311E+02	1.37885E+03	1.94741E+03	2.27028E+03	2.37933E+03
6.050(5) SEC.	7.84946E+02	1.06601E+03	1.48604E+03	1.72770E+03	1.80899E+03
8.640(5) SEC.	6.07494E+02	8.05581E+02	1.10965E+03	1.28670E+03	1.34400E+03
1.728(6) SEC.	3.18099E+02	4.11613E+02	5.59436E+02	6.46645E+02	6.75676E+02
2.592(6) SEC.	1.89780E+02	2.47067E+02	3.36544E+02	3.89094E+02	4.06584E+02
5.184(6) SEC.	5.48977E+01	7.64870E+01	1.07395E+02	1.24868E+02	1.30729E+02
7.776(6) SEC.	2.49120E+01	3.74672E+01	5.42602E+01	6.34415E+01	6.65434E+01
1.037(7) SEC.	1.58254E+01	2.47156E+01	3.62865E+01	4.25240E+01	4.46365E+01
1.296(7) SEC.	1.14421E+01	1.80785E+01	2.66433E+01	3.12404E+01	3.27978E+01
1.555(7) SEC.	8.52218E+00	1.35008E+01	1.99110E+01	2.33479E+01	2.45120E+01
2.074(7) SEC.	4.78805E+00	7.58108E+00	1.11778E+01	1.31065E+01	1.37596E+01
2.592(7) SEC.	2.69076E+00	4.25074E+00	6.26500E+00	7.34650E+00	7.71282E+00
3.154(7) SEC.	1.46870E+00	2.31168E+00	3.40585E+00	3.99471E+00	4.19434E+00

474 10/16/70

TABLE 2-49

R-1 REFERENCE DESIGN (WANL DWG. 939J723) HEATING RATE BY REGION AT FULL POWER
(1511 MW THERMAL) WITH COMPOSITE FUEL IN THE CORE

Region Number	Region Description	Material	Weight (Lbs)	Density (Lb/In ³)	Heating Rate				
					Btu/Hour				Mw
					Neutron	Prompt Gamma	Delayed Gamma	Total	Total
1-15	Core	Fueled Element	4.60 (3)	---	4.42 (9)	2.72 (8)	2.88 (8)*	4.98 (9)	1.46 (3)
		Unfueled Element	5.74 (2)	0.120	8.48 (6)	2.62 (7)	1.13 (7)	4.60 (7)	1.35 (1)
		Pyro Sleeve	8.77 (1)	0.0694	2.91 (6)	3.72 (6)	1.65 (6)	8.28 (6)	2.43 (0)
		A-286	1.71 (2)	0.286	2.63 (5)	5.12 (6)	3.25 (6)	1.10 (7)	3.22 (0)
		SS-304	1.24 (1)	0.287	1.84 (4)	5.37 (5)	2.34 (5)	7.89 (5)	2.31 (-1)
		Hydrogen	1.18 (0)	---	1.60 (6)	9.65 (4)	4.35 (4)	1.74 (6)	5.10 (-1)
		Total	5.45 (3)	---	4.43 (9)	3.08 (8)	3.04 (8)	5.05 (9)	1.48 (3)
16	Core Periphery	Graphitite-G	2.09 (2)	0.0658	3.33 (6)	4.28 (6)	1.76 (6)	9.37 (6)	2.74 (0)
		Pyrofoil	2.20 (1)	0.0405	3.52 (5)	4.51 (5)	1.86 (5)	9.89 (5)	2.90 (-1)
		ZrC (60% Dense)	1.92 (2)	0.146	4.38 (5)	4.48 (6)	1.77 (6)	6.69 (6)	1.96 (0)
		TZM Moly	1.50 (1)	0.359	6.86 (3)	3.67 (5)	1.43 (5)	5.17 (5)	1.52 (-1)
		Hydrogen	8.81 (-3)	---	5.84 (3)	3.54 (2)	1.46 (2)	6.34 (3)	1.86 (-3)
		Total	4.38 (2)	---	4.13 (6)	9.58 (6)	3.86 (6)	1.76 (7)	5.14 (0)
17	Lateral Support	P03 Graphite	6.19 (0)	0.0668	6.95 (4)	8.80 (4)	3.35 (4)	1.91 (5)	5.59 (-2)
		ZTA Graphite	3.85 (2)	0.0703	4.32 (6)	5.50 (6)	2.08 (6)	1.19 (7)	3.48 (0)
		Pyrofoil	7.90 (0)	0.0405	8.86 (4)	1.13 (5)	4.28 (4)	2.44 (5)	7.14 (-2)
		Hydrogen	3.75 (-1)	---	1.78 (5)	1.10 (4)	3.99 (3)	1.93 (5)	5.65 (-2)
		Total	3.99 (2)	---	4.66 (6)	5.71 (6)	2.16 (6)	1.25 (7)	3.66 (0)
18	Structure	P03 Graphite	8.64 (0)	0.0668	6.45 (4)	8.86 (4)	2.99 (4)	1.83 (5)	5.38 (-2)
		Al-6061	7.51 (2)	0.0975	1.18 (6)	7.77 (6)	2.55 (6)	1.15 (7)	3.37 (0)
		A-286	5.12 (1)	0.286	1.85 (4)	5.68 (5)	1.81 (5)	7.68 (5)	2.25 (-1)
		Hydrogen	4.95 (-1)	---	1.59 (5)	9.63 (3)	3.37 (3)	1.72 (5)	5.04 (-2)
		Total	8.11 (2)	---	1.42 (6)	8.44 (6)	2.76 (6)	1.26 (7)	3.70 (0)

NOTE: Numbers in parentheses refer to powers of ten.
Refer to Figure 2-11 for region locations.

*Includes betas

TABLE 2-49
(Continued)

R-1 REFERENCE DESIGN (WANL DWG. 939J723) HEATING RATE BY REGION AT FULL POWER
(1511 MW THERMAL) WITH COMPOSITE FUEL IN THE CORE

Region Number	Region Description	Material	Weight (Lbs)	Density (Lb/In ³)	Heating Rate				
					Btu/Hour				Mw
					Neutron	Prompt Gamma	Delayed Gamma	Total	Total
19-21	Reflector	P03 Graphite	2.33 (1)	0.0668	6.74 (4)	1.46 (5)	4.14 (4)	2.55 (5)	7.49 (-2)
		Pyrofoil	8.42 (0)	0.0405	2.43 (4)	5.30 (4)	1.49 (4)	9.22 (4)	2.70 (-2)
		Beryllium	1.70 (3)	0.0665	8.70 (6)	1.16 (7)	2.67 (6)	2.30 (7)	6.73 (0)
		Al-6061	3.95 (2)	0.0975	2.43 (5)	2.53 (6)	6.89 (5)	3.46 (6)	1.01 (0)
		A-286	3.28 (2)	0.286	4.65 (4)	2.30 (6)	5.96 (5)	2.94 (6)	8.60 (-1)
		Control Vane	1.17 (2)	0.306	1.08 (7)	8.17 (5)	2.12 (5)	1.19 (7)	3.48 (0)
		Hydrogen	3.52 (0)	---	4.42 (5)	4.27 (4)	1.23 (4)	4.97 (5)	1.46 (-1)
		Total	2.58 (3)	---	2.03 (7)	1.75 (7)	4.24 (6)	4.21 (7)	1.23 (1)
22	Pressure Vessel Side A	Al-7039	9.07 (2)	0.0988	1.08 (5)	2.45 (6)	5.96 (5)	3.15 (6)	9.24 (-1)
		Hydrogen	1.07 (0)	---	2.53 (4)	5.30 (3)	1.40 (3)	3.20 (4)	9.39 (-3)
		Total	9.08 (2)	---	1.33 (5)	2.46 (6)	4.96 (5)	3.18 (6)	9.33 (-1)
23	CHESH	Pyrographite	8.01 (0)	0.0789	3.38 (4)	4.32 (4)	1.43 (4)	9.13 (4)	2.68 (-2)
		Pyrofoil	5.90 (-2)	0.0405	2.49 (2)	3.18 (2)	1.05 (2)	6.72 (2)	1.97 (-4)
		NbC/C Comp.	2.22 (1)	0.144	4.11 (4)	1.30 (5)	4.14 (4)	2.13 (5)	6.25 (-2)
		W-ThO ₂	1.01 (2)	0.675	3.00 (3)	9.77 (5)	3.10 (5)	1.29 (6)	3.77 (-1)
		A-286	4.05 (1)	0.286	7.91 (4)	1.62 (5)	7.31 (4)	3.14 (5)	9.20 (-2)
		SS-304	1.36 (0)	0.287	2.57 (2)	7.70 (3)	2.44 (3)	1.04 (4)	3.06 (-3)
		SS-316	9.24 (-2)	0.288	1.73 (1)	5.29 (2)	1.66 (2)	7.12 (2)	2.09 (-4)
		Hydrogen	1.34 (-1)	---	2.30 (4)	1.33 (3)	4.70 (2)	2.48 (4)	7.28 (-3)
		Total	1.73 (2)	---	1.81 (5)	1.32 (6)	4.42 (5)	1.94 (6)	5.69 (-1)
24	Nozzle Chamber	Hydrogen	1.52 (-1)	---	1.28 (4)	7.61 (2)	2.39 (2)	1.38 (4)	4.04 (-3)
		Total	1.52 (-1)	---	1.28 (4)	7.61 (2)	2.39 (2)	1.38 (4)	4.04 (-3)
25	Nozzle Barrel	SS-347	3.42 (2)	0.285	1.59 (4)	5.64 (5)	1.01 (5)	6.81 (5)	1.99 (-1)
		Total	3.42 (2)	---	1.59 (4)	5.64 (5)	1.01 (5)	6.81 (5)	1.99 (-1)

NOTE: Numbers in parentheses refer to powers of ten.
Refer to Figure 2-11 for region locations.

TABLE 2-49
(Continued)

R-1 REFERENCE DESIGN (WANL DWG. 939J723) HEATING RATE BY REGION AT FULL POWER
(1511 MW THERMAL) WITH COMPOSITE FUEL IN THE CORE

Region Number	Region Description	Material	Weight (Lbs)	Density (Lb/In ³)	Heating Rate				
					Btu/Hour				Mw
					Neutron	Prompt Gamma	Delayed Gamma	Total	Total
26	Aft Reflector Hardware	Al-6061	5.37 (1)	0.0975	6.96 (3)	9.49 (4)	2.21 (4)	1.24 (5)	3.63 (-2)
		A-286	1.18 (1)	0.286	3.48 (2)	2.28 (4)	5.05 (3)	2.82 (4)	8.28 (-3)
		SS-440C	9.06 (0)	0.280	2.97 (2)	1.73 (4)	3.83 (3)	2.14 (4)	6.27 (-3)
		Hydrogen	5.10 (-1)	---	1.32 (4)	1.68 (3)	4.19 (2)	1.53 (4)	4.50 (-3)
		Total	7.50 (1)	---	2.08 (4)	1.37 (5)	3.14 (4)	1.89 (5)	5.54 (-2)
27	Aft Reflector Plenum	Hydrogen	4.31 (0)	---	8.31 (4)	8.82 (3)	1.98 (3)	9.39 (4)	2.75 (-2)
		Total	4.31 (0)	---	8.31 (4)	8.82 (3)	1.98 (3)	9.39 (4)	2.75 (-2)
28	Core Plenum	TZM Moly	5.71 (1)	0.369	1.30 (4)	1.10 (6)	2.16 (5)	1.33 (6)	3.89 (-1)
		Copper- Boron	1.06 (1)	0.271	8.57 (5)	1.78 (5)	3.55 (4)	1.07 (6)	3.14 (-1)
		A-286	3.04 (1)	0.286	1.12 (4)	5.15 (5)	1.03 (5)	6.29 (5)	1.84 (-1)
		SS-302	3.49 (1)	0.286	1.24 (4)	5.87 (5)	1.17 (5)	7.16 (5)	2.10 (-1)
		SS-304	1.07 (2)	0.287	3.82 (4)	1.80 (6)	3.61 (5)	2.20 (6)	6.45 (-1)
		Hydrogen	7.73 (-1)	---	2.52 (5)	2.29 (4)	5.06 (3)	2.80 (5)	8.20 (-2)
		Total	2.41 (2)	---	1.18 (6)	4.20 (6)	8.38 (5)	6.22 (6)	1.82 (0)
29	Support Plate	Pyrofoil	2.08 (0)	0.0405	5.78 (3)	1.33 (4)	1.70 (3)	2.08 (4)	6.11 (-3)
		Al-6061	4.20 (2)	0.0975	2.45 (5)	2.77 (6)	3.36 (5)	3.35 (6)	9.83 (-1)
		A-286	1.68 (1)	0.286	2.21 (3)	1.23 (5)	1.40 (4)	1.39 (5)	4.09 (-2)
		SS-302	8.01 (-1)	0.286	1.01 (2)	5.81 (3)	6.61 (2)	6.57 (3)	1.93 (-3)
		SS-304	2.34 (2)	0.287	2.97 (4)	1.70 (6)	1.93 (5)	1.92 (6)	5.63 (-1)
		Hydrogen	3.52 (0)	---	4.09 (5)	4.24 (4)	5.63 (3)	4.57 (5)	1.34 (-1)
		Total	6.78 (2)	---	6.92 (5)	4.65 (6)	5.51 (5)	5.89 (6)	1.73 (0)
30	Lateral Support - Forward	Al-6061	1.44 (1)	0.0975	1.13 (4)	1.08 (5)	2.17 (4)	1.41 (5)	4.14 (-2)
		A-286	3.80 (1)	0.286	6.76 (3)	3.14 (5)	5.95 (4)	3.80 (5)	1.11 (-1)
		Hydrogen	1.99 (-1)	---	3.12 (4)	2.80 (3)	6.00 (2)	3.46 (4)	1.01 (-2)
		Total	5.26 (1)	---	4.93 (4)	4.25 (5)	8.18 (4)	5.56 (5)	1.62 (-1)

NOTE: Numbers in parentheses refer to powers of ten.
Refer to Figure 2-11 for region locations.

TABLE 2-49
(Continued)

R-1 REFERENCE DESIGN (WANL DWG. 939J723) HEATING RATE BY REGION AT FULL POWER
(1511 MW THERMAL) WITH COMPOSITE FUEL IN THE CORE

Region Number	Region Description	Material	Weight (Lbs)	Density (Lb/in ³)	Heating Rate				
					Btu/hour				Mw
					Neutron	Prompt Gamma	Delayed Gamma	Total	Total
31	Forward Reflector Hardware I	Al-6061	4.23 (1)	0.0975	8.11 (3)	1.26 (5)	1.39 (4)	1.53 (5)	4.49 (-2)
		A-286	3.53 (1)	0.286	1.53 (3)	1.17 (5)	1.64 (4)	1.35 (5)	3.97 (-2)
		SS-304	4.42 (1)	0.287	1.85 (3)	1.46 (5)	2.03 (4)	1.68 (5)	4.91 (-2)
		SS-440C	9.06 (0)	0.280	4.38 (2)	2.96 (4)	4.15 (3)	3.42 (4)	1.00 (-2)
		Hydrogen	2.46 (0)	---	9.38 (4)	1.30 (4)	2.19 (3)	1.09 (5)	3.21 (-2)
		Total	1.33 (2)	---	1.06 (5)	4.32 (5)	6.19 (4)	5.99 (5)	1.76 (-1)
32	Support Plate Plenum	A-286	5.33 (-1)	0.286	3.55 (1)	2.48 (3)	1.82 (2)	2.70 (3)	7.92 (-4)
		SS-304	5.37 (1)	0.287	3.44 (3)	2.47 (5)	1.82 (4)	2.69 (5)	7.89 (-2)
		Hydrogen	4.79 (-1)	---	2.81 (4)	3.69 (3)	3.14 (2)	3.21 (4)	9.41 (-3)
		Total	5.47 (1)	---	3.16 (4)	2.53 (5)	1.83 (4)	3.04 (5)	8.91 (-2)
33	Instrumentation Ring	Al-6061	3.20 (1)	0.0975	4.48 (3)	6.89 (4)	5.34 (3)	7.87 (4)	2.31 (-2)
		SS-304	5.99 (0)	0.287	1.82 (2)	1.43 (4)	1.04 (3)	1.55 (4)	4.53 (-3)
		Hydrogen	1.16 (-1)	---	3.24 (3)	4.61 (2)	3.87 (1)	3.74 (3)	1.10 (-3)
		Total	3.81 (1)	---	7.90 (3)	8.37 (4)	6.42 (3)	9.79 (4)	2.87 (-2)
34	Forward Reflector Hardware II	Al-6061	1.85 (1)	0.0975	1.74 (3)	3.07 (4)	3.05 (3)	3.55 (4)	1.04 (-2)
		A-286	4.25 (0)	0.286	9.00 (1)	7.84 (3)	7.30 (2)	8.66 (3)	2.54 (-3)
		SS-304	3.14 (0)	0.287	6.41 (1)	5.73 (3)	5.34 (2)	6.33 (3)	1.85 (-3)
		Hydrogen	3.61 (-1)	---	6.72 (3)	1.10 (3)	1.19 (2)	7.94 (3)	2.33 (-3)
		Total	2.63 (1)	---	8.61 (3)	4.54 (4)	4.43 (3)	5.84 (4)	1.71 (-2)
35	Aft Central Shield Plate	Al-6061	9.78 (1)	0.0975	2.45 (4)	3.43 (5)	2.59 (4)	3.93 (5)	1.15 (-1)
		Hydrogen	4.67 (-2)	---	2.33 (3)	3.05 (2)	2.47 (1)	2.66 (3)	7.79 (-4)
		Total	9.79 (1)	---	2.68 (4)	3.43 (5)	2.59 (4)	3.96 (5)	1.16 (-1)
36	BATH Central Shield	BATH	1.24 (3)	0.103	2.29 (6)	1.32 (6)	8.38 (4)	3.69 (6)	1.08 (0)
		Al-6061	3.35 (0)	0.0975	1.77 (2)	3.50 (3)	2.25 (2)	3.90 (3)	1.14 (-3)
		Hydrogen	2.93 (-1)	---	3.06 (3)	5.71 (2)	3.92 (1)	3.67 (3)	1.08 (-3)
		Total	1.24 (3)	---	2.29 (6)	1.32 (6)	8.41 (4)	3.69 (6)	1.08 (0)

NOTE: Numbers in parentheses refer to powers of ten.
Refer to Figure 2-11 for region locations.

TABLE 2-49
(Continued)

R-1 REFERENCE DESIGN (WANL DWG. 939J723) HEATING RATE BY REGION AT FULL POWER
(1511 MW THERMAL) WITH COMPOSITE FUEL IN THE CORE

Region Number	Region Description	Material	Weight (Lbs)	Density (Lb/in ³)	Heating Rate				
					Btu/Hour				Mw
					Neutron	Prompt Gamma	Delayed Gamma	Total	Total
37	Flow Baffle I	Al-6061	2.36 (2)	0.0975	5.97 (3)	1.02 (5)	6.81 (3)	1.15 (5)	3.37 (-2)
		SS-304	2.25 (1)	0.287	1.23 (2)	1.06 (4)	6.71 (2)	1.14 (4)	3.36 (-3)
		Hydrogen	1.83 (-2)	---	9.15 (1)	1.44 (1)	1.05 (0)	1.07 (2)	3.14 (-5)
		Total	2.59 (2)	---	6.18 (3)	1.13 (5)	7.48 (3)	1.27 (5)	3.71 (-2)
38	BATH Peripheral Shield I	BATH	1.84 (2)	0.103	2.88 (5)	1.28 (5)	8.23 (3)	4.24 (5)	1.24 (-1)
		Al-6061	1.99 (1)	0.0975	6.63 (2)	1.36 (4)	8.83 (2)	1.51 (4)	4.41 (-3)
		Hydrogen	5.02 (-1)	---	3.31 (3)	6.36 (2)	4.43 (1)	3.99 (3)	1.17 (-3)
		Total	2.05 (2)	---	2.92 (5)	1.42 (5)	9.16 (3)	4.43 (5)	1.30 (-1)
39	BATH Peripheral Shield II	BATH	2.48 (2)	0.103	2.38 (5)	1.44 (5)	9.93 (3)	3.92 (5)	1.15 (-1)
		Hydrogen	4.85 (-2)	---	2.42 (2)	5.12 (1)	3.84 (0)	2.97 (2)	8.71 (-5)
		Total	2.48 (2)	---	2.38 (5)	1.44 (5)	9.93 (3)	3.92 (5)	1.15 (-1)
40	BATH Peripheral Shield III	BATH	2.46 (2)	0.103	1.92 (5)	1.18 (5)	9.15 (3)	3.19 (5)	9.34 (-2)
		Al-6061	1.24 (1)	0.0975	2.70 (2)	5.83 (3)	4.59 (2)	6.56 (3)	1.92 (-3)
		A-286	3.12 (1)	0.286	1.52 (2)	1.61 (4)	1.20 (3)	1.75 (4)	5.13 (-3)
		SS-304	2.19 (1)	0.287	1.03 (2)	1.13 (4)	8.38 (2)	1.22 (4)	3.56 (-3)
		Hydrogen	1.14 (-1)	---	4.89 (2)	9.96 (1)	8.38 (0)	5.97 (2)	1.75 (-4)
		Total	3.12 (2)	---	1.93 (5)	1.51 (5)	1.17 (4)	3.56 (5)	1.04 (-1)
41	BATH Peripheral Shield IV	BATH	2.39 (2)	0.103	1.42 (5)	1.04 (5)	9.33 (3)	2.55 (5)	7.48 (-2)
		Al-6061	1.95 (1)	0.0975	3.97 (2)	8.38 (3)	7.56 (2)	9.53 (3)	2.79 (-3)
		Hydrogen	2.12 (-2)	---	8.46 (1)	1.68 (1)	1.64 (0)	1.03 (2)	3.02 (-5)
		Total	2.58 (2)	---	1.42 (5)	1.12 (5)	1.01 (4)	2.65 (5)	7.76 (-2)
42	Pressure Vessel Side B	Al-7039	3.43 (2)	0.0988	3.30 (3)	6.65 (4)	6.94 (3)	7.67 (4)	2.25 (-2)
		Hydrogen	4.04 (-1)	---	7.56 (2)	1.44 (2)	1.63 (1)	9.16 (2)	2.68 (-4)
		Total	3.43 (2)	---	4.06 (3)	6.66 (4)	6.96 (3)	7.76 (4)	2.28 (-2)

NOTE: Numbers in parentheses refer to powers of ten.
Refer to Figure 2-11 for region locations.

TABLE 2-49
(Continued)

R-1 REFERENCE DESIGN (WANL DWG. 939J723) HEATING RATE BY REGION AT FULL POWER
(1511 MW THERMAL) WITH COMPOSITE FUEL IN THE CORE

Region Number	Region Description	Material	Weight (Lbs)	Density (Lb/in ³)	Heating Rate				
					Btu/Hour				Mw
					Neutron	Prompt Gamma	Delayed Gamma	Total	Total
43	Lead Central Shield	Lead Alloy	5.30 (2)	0.408	7.71 (1)	6.47 (4)	3.32 (3)	6.81 (4)	1.99 (-2)
		Al-6061	3.07 (-1)	0.0975	1.42 (0)	1.97 (1)	1.12 (0)	2.22 (1)	6.51 (-6)
		Hydrogen	6.12 (-2)	---	5.55 (1)	7.06 (0)	4.35 (-1)	6.30 (1)	1.85 (-5)
		Total	5.30 (2)	---	1.34 (2)	6.47 (4)	3.32 (3)	6.82 (4)	1.99 (-2)
44	Lead Peripheral * Shield I	Lead Alloy	7.84 (1)	0.408	9.43 (0)	8.18 (3)	4.26 (2)	8.62 (3)	2.53 (-3)
		Al-6061	6.74 (-1)	0.0975	2.58 (0)	3.62 (1)	2.07 (0)	4.09 (1)	1.20 (-5)
		Hydrogen	7.13 (-2)	---	5.36 (1)	6.87 (0)	4.28 (-1)	6.09 (1)	1.79 (-5)
		Total	7.91 (1)	---	6.56 (1)	8.22 (3)	4.28 (2)	8.72 (3)	2.56 (-3)
45	Lead Peripheral Shield II	Lead Alloy	1.09 (2)	0.408	1.06 (1)	7.74 (3)	4.24 (2)	8.17 (3)	2.40 (-3)
		Hydrogen	1.09 (-2)	---	6.63 (0)	7.32 (-1)	4.85 (-2)	7.41 (0)	2.17 (-6)
		Total	1.09 (2)	---	1.72 (1)	7.74 (3)	4.24 (2)	8.18 (3)	2.40 (-3)
46	Lead Peripheral Shield III	Lead Alloy	1.08 (2)	0.408	7.58 (0)	6.09 (3)	3.42 (2)	6.44 (3)	1.89 (-3)
		Al-6061	1.46* (0)	0.0975	3.24 (0)	4.33 (1)	2.69 (0)	4.92 (1)	1.44 (-5)
		A-286	3.54 (0)	0.286	1.76 (0)	1.18 (2)	6.81 (0)	1.27 (2)	3.72 (-5)
		SS-304	2.57 (0)	0.287	1.24 (0)	8.48 (1)	4.89 (0)	9.09 (1)	2.67 (-5)
		Hydrogen	1.85 (-2)	---	8.06 (0)	9.93 (-1)	6.66 (-2)	9.12 (0)	2.67 (-6)
		Total	1.16 (2)	---	2.19 (1)	6.34 (3)	3.56 (2)	6.72 (3)	1.97 (-3)
47	Lead Peripheral Shield IV	Lead Alloy	1.25 (2)	0.408	9.45 (0)	8.81 (3)	5.15 (2)	9.33 (3)	2.73 (-3)
		Al-6061	2.64 (0)	0.0975	6.34 (0)	9.75 (1)	6.19 (0)	1.10 (2)	3.22 (-5)
		Hydrogen	5.39 (-3)	---	2.52 (0)	3.55 (-1)	2.47 (-2)	2.90 (0)	8.50 (-7)
		Total	1.28 (2)	---	1.83 (1)	8.91 (3)	5.21 (2)	9.44 (3)	2.76 (-3)
48	Peripheral Shield Plate	Al- 6061	4.35 (1)	0.0975	9.45 (1)	9.36 (2)	5.00 (1)	1.08 (3)	3.16 (-4)
		A-286	2.13 (0)	0.286	1.03 (0)	5.10 (1)	2.55 (0)	5.46 (1)	1.60 (-5)
		SS-304	1.48 (0)	0.287	6.99 (-1)	3.52 (1)	1.76 (0)	3.77 (1)	1.10 (-5)
		Hydrogen	5.24 (-2)	---	2.24 (1)	1.98 (0)	1.18 (-1)	2.45 (1)	7.19 (-6)
		Total	4.71 (1)	---	1.19 (2)	1.02 (3)	5.44 (1)	1.20 (3)	3.50 (-4)

NOTE: Numbers in parentheses refer to powers of ten.
Refer to Figure 2-11 for region locations.

TABLE 2-49
(Continued)

R-1 REFERENCE DESIGN (WANL DWG. 939J723) HEATING RATE BY REGION AT FULL POWER
(1511 MW THERMAL) WITH COMPOSITE FUEL IN THE CORE

Region Number	Region Description	Material	Weight (Lbs)	Density (Lb/in ³)	Heating Rate				
					Btu/Hour				Mw
					Neutron	Prompt Gamma	Delayed Gamma	Total	Total
49	Shield Plenum	Al-6061	1.88 (1)	0.0975	5.89 (1)	5.06 (2)	2.65 (1)	5.91 (2)	1.73 (-4)
		SS-304	3.03 (-1)	0.287	2.06 (-1)	9.00 (0)	4.41 (-1)	9.65 (0)	2.83 (-6)
		Hydrogen	1.59 (0)	---	9.78 (2)	7.76 (1)	4.38 (0)	1.06 (3)	3.10 (-4)
		Total	2.07 (1)	---	1.04 (3)	5.93 (2)	3.13 (1)	1.66 (3)	4.86 (-4)
50	Flow Baffle II	Al-6061	1.30 (2)	0.0975	2.99 (2)	3.09 (3)	1.53 (2)	3.54 (3)	1.04 (-3)
		Total	1.30 (2)	---	2.99 (2)	3.09 (3)	1.53 (2)	3.54 (3)	1.04 (-3)
51	Central Dome Plenum	Hydrogen	7.77 (-1)	---	2.78 (2)	3.22 (1)	1.79 (0)	3.12 (2)	9.14 (-5)
		Total	7.77 (-1)	---	2.78 (2)	3.22 (1)	1.79 (0)	3.12 (2)	9.14 (-5)
52	Peripheral Dome Plenum	Al-6061	2.58 (1)	0.0975	2.82 (1)	4.28 (2)	2.19 (1)	4.78 (2)	1.40 (-4)
		A-286	1.43 (1)	0.286	3.51 (0)	2.66 (2)	1.27 (1)	2.88 (2)	8.25 (-5)
		SS-304	8.24 (0)	0.287	1.97 (0)	1.52 (2)	7.27 (0)	1.61 (2)	4.73 (-5)
		Hydrogen	8.04 (0)	---	1.73 (3)	2.47 (2)	1.33 (1)	1.99 (3)	5.82 (-4)
		Total	5.63 (1)	---	1.76 (3)	1.09 (3)	5.52 (1)	2.91 (3)	3.52 (-4)
53	Pressure Vessel Dome	Al-7039	2.17 (2)	0.0988	1.84 (2)	2.90 (3)	1.47 (2)	3.23 (3)	9.46 (-4)
		Total	2.17 (2)	---	1.84 (2)	2.90 (3)	1.47 (2)	3.23 (3)	9.46 (-4)
	NERVA Nuclear Subsystem	---	1.53 (4)	---	4.47 (9)	3.66 (8)	3.19 (8)	5.16 (9)	1.511 (3)

NOTE: Numbers in parentheses refer to powers of ten.
Refer to Figure 2-11 for region locations.

TABLE 2-50

COMPARISON OF HEATING RATES AT KEY NSS LOCATIONS
FOR COMPOSITE AND GRAPHITE FUELED CORES

Component	Material	Relative ⁽¹⁾ Total Heating Rate	
		Composite Fuel	Graphite Fuel
Core	Hydrogen	1.00	0.95
		A-286	1.30
Core Periphery	Graphite-G	1.00	1.14
Reflector	Beryllium	1.00	1.08
	P03 Graphite	1.00	1.09
	Al-6061	1.00	1.19
	A-286	1.00	1.19
	Control Vane	1.00	0.98
Pressure Vessel Side	Al-7039	1.00	1.18
CHESH	Pyrographite	1.00	1.18
		A-286	1.33
Core Plenum	Copper-Boron	1.00	1.06
Support Plate	Pyrofoil	1.00	1.06
	Al-6061	1.00	1.10
	A-286	1.00	1.10
BATH Central Shield	Al-6061	1.00	1.04
	BATH	1.00	0.96
Pressure Vessel Dome	Al-7039	1.00	1.04

⁽¹⁾Data reported relative to the R-1 Reference Design with Composite fuel in the core, i. e., $\frac{\text{Graphite}}{\text{Composite}}$

TIME INTEGRATED RADIATION ENVIRONMENT AT SELECTED LOCATIONS (COMPOSITE CORE)

LOCATION - REACTOR CENTERLINE, AFT SURFACE OF CHESS
RADIUS (IN) 0.0 STATION (IN) 54.13

Point 1

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-NBC/C COMPOSITE	5.11E 00	1.35	1.90E 02	1.35	1.91E 02	1.35	2.00E 02	1.35
BTU/IN3-W-THO2	2.85E 01	1.44	1.08E 03	1.44	1.08E 03	1.44	1.13E 03	1.44
BTU/IN3-A-286	7.84E 00	1.43	2.97E 02	1.43	2.94E 02	1.43	3.13E 02	1.43
NEUTRONS/CM2, E GT 1.0MEV	5.95E 15	1.05	2.06E 17	1.05	2.07E 17	1.05	2.16E 17	1.05
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.08E 16	1.05	7.19E 17	1.05	7.22E 17	1.05	7.55E 17	1.05
NEUTRONS/CM2, E LT 0.4EV	6.42E 13	2.04	2.22E 15	2.04	2.23E 15	2.04	2.33E 15	2.04
GAMMA RAY DOSE, RADS (CARBON)	5.65E 06	1.44	2.72E 08	1.44	2.23E 08	1.44	2.34E 08	1.44

7410 11/20/70

LOCATION - REACTOR CENTERLINE, CORE EXIT
RADIUS (IN) 0.0 STATION (IN) 52.00

Point 2

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-NBC/C COMPOSITE	1.08E 01	1.13	4.09E 02	1.13	4.12E 02	1.13	4.31E 02	1.13
BTU/IN3-W-THO2	7.26E 01	1.14	2.81E 03	1.14	2.82E 03	1.14	2.96E 03	1.14
BTU/IN3-A-286	1.70E 01	1.14	6.54E 02	1.14	6.58E 02	1.14	6.90E 02	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.03E 16	1.03	3.57E 17	1.03	3.59E 17	1.03	3.75E 17	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.68E 16	1.03	1.62E 18	1.03	1.63E 18	1.03	1.70E 18	1.03
NEUTRONS/CM2, E LT 0.4EV	2.02E 13	1.35	7.00E 14	1.35	7.04E 14	1.35	7.35E 14	1.35
GAMMA RAY DOSE, RADS (CARBON)	1.30E 07	1.14	5.03E 08	1.14	5.05E 08	1.14	5.30E 08	1.14

TABLE 2-51
(Continued)

Page 2 of 30

LOCATION - REACTOR CENTERLINE, CORE MIDPLANE
RADIUS (IN) 0.0 STATION (IN) 26.00

Point 3

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-A-286	1.22E 02	1.14	4.79E 03	1.14	4.82E 03	1.14	5.05E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	7.84E 16	1.03	2.71E 18	1.03	2.73E 18	1.03	2.85E 18	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.91E 17	1.03	1.35E 19	1.03	1.36E 19	1.03	1.42E 19	1.03
NEUTRONS/CM2, E LT 0.4EV	6.07E 13	1.35	2.10E 15	1.35	2.11E 15	1.35	2.20E 15	1.35
GAMMA RAY DOSE, RADS (CARBON)	9.46E 07	1.14	3.74E 09	1.14	3.76E 09	1.14	3.95E 09	1.14

LOCATION - REACTOR CENTERLINE, CORE INLET
RADIUS (IN) 0.0 STATION (IN) 0.0

Point 4

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-TZM MOLY	6.02E 01	1.14	2.27E 03	1.14	2.28E 03	1.14	2.39E 03	1.14
BTU/IN3-CU-18V/O BN	1.37E 02	1.62	4.86E 03	1.62	4.88E 03	1.62	5.11E 03	1.62
BTU/IN3-A-286	4.17E 01	1.14	1.57E 03	1.14	1.58E 03	1.14	1.66E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.95E 16	1.03	6.76E 17	1.03	6.79E 17	1.03	7.10E 17	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.03E 17	1.03	3.58E 18	1.03	3.60E 18	1.03	3.76E 18	1.03
NEUTRONS/CM2, E LT 0.4EV	3.43E 14	1.35	1.19E 16	1.35	1.19E 16	1.35	1.25E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	3.16E 07	1.14	1.20E 09	1.14	1.20E 09	1.14	1.26E 09	1.14

TABLE 2-51
(Continued)

LOCATION - REACTOR CENTERLINE, AFT FACE OF SUPPORT PLATE
RADIUS (IN) 0.0 STATION (IN) -3.

Point 5

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR

BTU/IN3-TZM MOLY	3.93E 01	1.09	1.41E 03	1.09	1.42E 03	1.09	1.48E 03	1.09
BTU/IN3-CU-18V/O BN	2.45E 02	1.79	8.50E 03	1.79	8.55E 03	1.79	8.93E 03	1.79
BTU/IN3-AL-6061	9.02E 00	1.09	3.24E 02	1.09	3.26E 02	1.09	3.41E 02	1.09
BTU/IN3-A-286	2.70E 01	1.09	9.69E 02	1.09	9.74E 02	1.09	1.02E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.06E 16	1.71	3.66E 17	1.71	3.68E 17	1.71	3.84E 17	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.10E 16	1.71	2.46E 18	1.71	2.47E 18	1.71	2.56E 19	1.71
NEUTRONS/CM2, E LT 0.4EV	2.33E 15	1.88	8.08E 16	1.88	8.12E 16	1.88	8.48E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.96E 07	1.09	7.08E 08	1.09	7.12E 08	1.09	7.45E 08	1.09

LOCATION - REACTOR CENTERLINE, FORWARD FACE OF SUPPORT PLATE
RADIUS (IN) 0.0 STATION (IN) -9.50

Point 6

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR

BTU/IN3-AL-6061	3.27E 00	1.09	1.15E 02	1.09	1.15E 02	1.09	1.21E 02	1.09
BTU/IN3-A-286	1.02E 01	1.09	3.57E 02	1.09	3.59E 02	1.09	3.75E 02	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.94E 15	1.71	1.02E 17	1.71	1.02E 17	1.71	1.07E 17	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.60E 16	1.71	8.99E 17	1.71	9.03E 17	1.71	9.44E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	2.63E 15	1.88	9.09E 16	1.88	9.14E 16	1.88	9.55E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	7.17E 06	1.09	2.52E 08	1.09	2.54E 08	1.09	2.65E 08	1.09

TABLE 2-51
(Continued)

LOCATION - REACTOR CENTERLINE, AFT FACE OF CENTRAL SHIELD PLATE
RADIUS (IN) 0.0 STATION (IN) -10.83

Point 7

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-AL-6061	2.78E 00	1.09	9.75E 01	1.09	9.80E 01	1.09	1.02E 02	1.09
BTU/IN3-A-286	8.61E 00	1.09	3.01E 02	1.09	3.03E 02	1.09	3.17E 02	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.57E 15	1.71	8.88E 16	1.71	8.92E 16	1.71	9.33E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.07E 16	1.71	7.17E 17	1.71	7.21E 17	1.71	7.54E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	1.54E 15	1.88	5.33E 16	1.88	5.36E 16	1.88	5.60E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	6.10E 06	1.09	2.15E 08	1.09	2.16E 08	1.09	2.25E 08	1.09

LOCATION - REACTOR CENTERLINE, AFT FACE OF BATH CENTRAL SHIELD
RADIUS (IN) 0.0 STATION (IN) -11.83

Point 8

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-BATH	1.60E 01	1.75	5.53E 02	1.75	5.56E 02	1.75	5.81E 02	1.75
BTU/IN3-AL-6061	2.37E 00	1.09	8.27E 01	1.09	8.31E 01	1.09	8.68E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.97E 15	1.71	6.82E 16	1.71	6.86E 16	1.71	7.17E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.65E 16	1.71	5.71E 17	1.71	5.74E 17	1.71	6.00E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	6.47E 14	1.88	2.24E 16	1.88	2.25E 16	1.88	2.35E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	5.28E 06	1.09	1.85E 08	1.09	1.86E 08	1.09	1.94E 08	1.09

2-167

TABLE 2-51
(Continued)

LOCATION - REACTOR CENTERLINE, BATH CENTRAL SHIELD MIDPLANE
 RADIUS (IN) 0.0 STATION (IN) -17.0

Point 9

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)
 NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	1.12E 00	1.47	3.88E 01	1.47	3.90E 01	1.47	4.07E 01	1.47
BTU/IN3-AL-6061	5.02E-01	1.09	1.75E 01	1.09	1.76E 01	1.09	1.84E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	3.03E 14	1.71	1.05E 16	1.71	1.05E 16	1.71	1.10E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.45E 14	1.71	5.02E 16	1.71	5.04E 16	1.71	5.27E 16	1.71
NEUTRONS/CM2, E LT 0.4EV	2.20E 12	1.88	7.60E 13	1.88	7.63E 13	1.88	7.98E 13	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.15E 06	1.09	4.02E 07	1.09	4.04E 07	1.09	4.22E 07	1.09

LOCATION - REACTOR CENTERLINE, FORWARD FACE OF BATH CENTRAL SHIELD
 RADIUS (IN) 0.0 STATION (IN) -22.0

Point 10

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)
 NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	2.65E-01	1.55	9.19E 00	1.55	9.23E 00	1.55	9.65E 00	1.55
BTU/IN3-LEAD ALLOY	9.02E-01	1.09	3.13E 01	1.09	3.15E 01	1.09	3.29E 01	1.09
BTU/IN3-AL-6061	9.55E-02	1.09	3.32E 00	1.09	3.34E 00	1.09	3.49E 00	1.09
NEUTRONS/CM2, E GT 1.0MEV	5.50E 13	1.71	1.90E 15	1.71	1.91E 15	1.71	2.00E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.13E 14	1.71	7.38E 15	1.71	7.41E 15	1.71	7.75E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	3.22E 12	1.88	1.11E 14	1.88	1.12E 14	1.88	1.17E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.15E 05	1.09	7.49E 06	1.09	7.53E 06	1.09	7.87E 06	1.09

TABLE 2-51
(Continued)

LOCATION - REACTOR CENTERLINE, FORWARD FACE OF LEAD CENTRAL SHIELD
RADIUS (IN) 0.0 STATION (IN) -24.23

Point 11

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-LEAD ALLOY	1.32E-01	1.09	4.58E-00	1.09	4.60E-00	1.09	4.81E-00	1.09
BTU/IN3-AL-6061	1.95E-02	1.09	6.78E-01	1.09	6.82E-01	1.09	7.12E-01	1.09
NEUTRONS/CM2, E GT 1.0MEV	3.60E 13	1.71	1.25E 15	1.71	1.25E 15	1.71	1.31E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.44E 14	1.71	4.47E 15	1.71	5.00E 15	1.71	5.23E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.21E 13	1.88	4.70E 14	1.88	4.22E 14	1.88	4.41E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.99E 04	1.09	1.39E 06	1.09	1.40E 06	1.09	1.46E 06	1.09

LOCATION - REACTOR CENTERLINE, AFT FACE OF FLOW BAFFLE II
RADIUS (IN) 0.0 STATION (IN) -29.63

Point 12

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	1.64E-02	1.41	5.86E-01	1.41	5.89E-01	1.41	6.16E-01	1.41
NEUTRONS/CM2, E GT 1.0MEV	2.64E 13	3.10	9.13E 14	3.10	9.18E 14	3.10	9.59E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.09E 14	3.10	3.76E 15	3.10	3.78E 15	3.10	3.95E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.53E 13	3.10	5.30E 14	3.10	5.33E 14	3.10	5.57E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	3.53E 04	1.40	1.23E 06	1.40	1.23E 06	1.40	1.24E 06	1.40

TABLE 2-51
(Continued)

LOCATION - REACTOR CENTERLINE, FORWARD FACE OF FLOW BAFFLE 11
RADIUS (IN) 0.0 STATION (IN) -30.75

Point 13

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	1.45E-02	1.41	5.03E-01	1.41	5.05E-01	1.41	5.28E-01	1.41
NEUTRONS/CM2, E GT 1.0MEV	2.00E 13	3.10	6.93E 14	3.10	6.96E 14	3.10	7.28E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.62E 13	3.10	2.98E 15	3.10	3.00E 15	3.10	3.13E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.53E 13	3.10	5.31E 14	3.10	5.33E 14	3.10	5.57E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	3.07E 04	1.40	1.07E 06	1.40	1.07E 06	1.40	1.12E 06	1.40

LOCATION - REACTOR CENTERLINE, AFT SURFACE OF PRESSURE VESSEL DOME
RADIUS (IN) 0.0 STATION (IN) -36.04

Point 14

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-7039	1.10E-02	1.40	3.83E-01	1.40	3.85E-01	1.40	4.02E-01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.37E 13	3.10	4.73E 14	3.10	4.76E 14	3.10	4.97E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.76E 13	3.10	1.65E 15	3.10	1.66E 15	3.10	1.73E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	8.30E 12	3.10	2.87E 14	3.10	2.89E 14	3.10	3.02E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.33E 04	1.40	8.12E 05	1.40	8.16E 05	1.40	8.53E 05	1.40

TABLE 2-51
(Continued)

LOCATION - REACTOR CENTERLINE, TANGENT PLANE
 RADIUS (IN) 0.0 STATION (IN) -37.00

Point 15

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

4TU/IN3-AL-7039	8.95E-03	1.40	3.11E-01	1.40	3.13E-01	1.40	3.27E-01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.06E 13	3.10	3.66E 14	3.10	3.68E 14	3.10	3.85E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.09E 13	3.10	1.07E 15	3.10	1.07E 15	3.10	1.12E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	5.11E 12	3.10	1.77E 14	3.10	1.78E 14	3.10	1.86E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	1.91E 04	1.40	6.65E 05	1.40	6.67E 05	1.40	6.99E 05	1.40

LOCATION - CORE EDGE, AFT SURFACE OF CHESH
 RADIUS (IN) 18.66 STATION (IN) 54.13

Point 16

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-NBC/C COMPOSITE	2.78E 00	1.36	1.02E 02	1.36	1.03E 02	1.36	1.08E 02	1.36
BTU/IN3-W-TH02	1.75E 01	1.44	6.60E 02	1.44	6.64E 02	1.44	6.95E 02	1.44
BTU/IN3-A-286	4.38E 00	1.43	1.63E 02	1.43	1.64E 02	1.43	1.72E 02	1.43
NEUTRONS/CM2, E GT 1.0MEV	2.83E 15	1.05	9.81E 16	1.05	9.86E 16	1.05	1.03E 17	1.05
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.30E 16	1.05	4.50E 17	1.05	4.52E 17	1.05	4.73E 17	1.05
NEUTRONS/CM2, E LT 0.4EV	4.52E 14	2.04	1.56E 16	2.04	1.57E 16	2.04	1.64E 16	2.04
GAMMA RAY DOSE, RADS (CARBON)	3.20E 06	1.44	1.20E 08	1.44	1.21E 08	1.44	1.27E 08	1.44

TABLE 2-51
(Continued)

LOCATION - CORE EDGE, CORE EXIT		STATION (IN)		Point 17	
RADIUS (IN)		52.00			

STATE POINT (B)		STATE POINT (C)		STATE POINT (D)	
STATE POINT (E)					
NOMINAL UNC FACTR		NOMINAL UNC FACTR		NOMINAL UNC FACTR	
NOMINAL UNC FACTR					
BTU/IN3-GRAPHITITE G		2.67E 00	1.14	9.92E 01	1.14
BTU/IN3-NBC/C COMPOSITE		5.07E 00	1.13	1.92E 02	1.13
BTU/IN3-ZRC(0.6DENSE)		4.04E 00	1.13	1.77E 02	1.13
BTU/IN3-W-THO2		3.58E 01	1.14	1.33E 03	1.14
BTU/IN3-A-286		8.08E 00	1.14	3.10E 02	1.14
NEUTRONS/CM2, E GT 1.0MEV		4.51E 15	1.03	1.56E 17	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV		2.14E 16	1.03	7.40E 17	1.03
NEUTRONS/CM2, E LT 0.4EV		4.68E 14	1.35	1.62E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)		6.13E 06	1.14	2.37E 08	1.14

LOCATION - CORE EDGE, CORE MIDPLANE		STATION (IN)		26.00		Point 18	
RADIUS (IN)		18.66					

STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
NOMINAL UNC FACTR		NOMINAL UNC FACTR		NOMINAL UNC FACTR		NOMINAL UNC FACTR	
BTU/IN3-GRAPHITITE G							
2.00E 01		1.14		7.51E 02		1.14	
BTU/IN3-ZRC(0.6DENSE)		3.33E 01		1.30E 03		1.13	
NEUTRONS/CM2, E GT 1.0MEV		3.49E 16		1.21E 18		1.03	
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV		1.86E 17		6.44E 18		1.03	
NEUTRONS/CM2, E LT 0.4EV		3.09E 15		1.07E 17		1.35	
GAMMA RAY DOSE, RADS (CARBON)		4.43E 07		1.75E 09		1.14	

TABLE 2-51
(Continued)

LOCATION - CORE EDGE, CORE INLET
RADIUS (IN) 18.66 STATION (IN) 0.0 Point 19

	STATE POINT (B)	STATE POINT (C)	STATE POINT (D)	STATE POINT (E)
	NOMINAL UNC FACTR	NOMINAL UNC FACTR	NOMINAL UNC FACTR	NOMINAL UNC FACTR
BTU/IN3-GRAPHITITE G	5.80E 00 1.14	2.14E 02 1.14	2.15E 02 1.14	2.25E 02 1.14
BTU/IN3-ZRC(0.6DENSE)	1.07E 01 1.13	4.02E 02 1.13	4.04E 02 1.13	4.24E 02 1.13
NEUTRONS/CM2, E GT 1.0MEV	8.71E 15 1.03	3.01E 17 1.03	3.03E 17 1.03	3.17E 17 1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.88E 16 1.03	1.69E 18 1.03	1.70E 18 1.03	1.77E 18 1.03
NEUTRONS/CM2, E LT 0.4EV	1.17E 15 1.35	4.05E 16 1.35	4.07E 16 1.35	4.26E 16 1.35
GAMMA RAY DOSE, RADS (CARBON)	1.40E 07 1.14	5.32E 08 1.14	5.34E 08 1.14	5.60E 08 1.14

LOCATION - OUTER RADIUS OF CORE PERIPHERY, CORE EXIT
RADIUS (IN) 19.48 STATION (IN) 52.00 Point 20

	STATE POINT (B)	STATE POINT (C)	STATE POINT (D)	STATE POINT (E)
	NOMINAL UNC FACTR	NOMINAL UNC FACTR	NOMINAL UNC FACTR	NOMINAL UNC FACTR
BTU/IN3-GRAPHITITE G	2.17E 00 1.14	8.07E 01 1.14	8.11E 01 1.14	8.49E 01 1.14
BTU/IN3-ZRC(0.6DENSE)	3.82E 00 1.13	1.45E 02 1.13	1.46E 02 1.13	1.53E 02 1.13
NEUTRONS/CM2, E GT 1.0MEV	3.62E 15 1.16	1.25E 17 1.16	1.26E 17 1.16	1.31E 17 1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.96E 16 1.16	6.80E 17 1.16	6.83E 17 1.16	7.14E 17 1.16
NEUTRONS/CM2, E LT 0.4EV	7.23E 14 1.35	2.50E 16 1.35	2.52E 16 1.35	2.63E 16 1.35
GAMMA RAY DOSE, RADS (CARBON)	5.00E 06 1.14	1.93E 08 1.14	1.94E 08 1.14	2.03E 08 1.14

TABLE 2-51
(Continued)

LOCATION - OUTER RADIUS OF CORE PERIPHERY, CORE MIDPLANE
RADIUS (IN) 19.48 STATION (IN) 26.00

Point 21

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-GRAPHITITE G	1.47E 01	1.14	5.50E 02	1.14	5.53E 02	1.14	5.79E 02	1.14
BTU/IN3-ZRC(0.6DENSE)	2.42E 01	1.13	9.38E 02	1.13	9.43E 02	1.13	9.89E 02	1.13
NEUTRONS/CM2, E GT 1.0MEV	2.60E 16	1.16	9.01E 17	1.16	9.06E 17	1.16	9.47E 17	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.71E 17	1.16	5.92E 18	1.16	5.95E 18	1.16	6.22E 18	1.16
NEUTRONS/CM2, E LT 0.4EV	5.51E 15	1.35	1.91E 17	1.35	1.92E 17	1.35	2.00E 17	1.35
GAMMA RAY DOSE, RADS (CARBON)	3.17E 07	1.14	1.24E 09	1.14	1.25E 09	1.14	1.31E 09	1.14

LOCATION - OUTER RADIUS OF CORE PERIPHERY, CORE INLET
RADIUS (IN) 19.48 STATION (IN) 0.0

Point 22

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-GRAPHITITE G	4.59E 00	1.14	1.69E 02	1.14	1.69E 02	1.14	1.77E 02	1.14
BTU/IN3-ZRC(0.6DENSE)	8.55E 00	1.13	3.19E 02	1.13	3.20E 02	1.13	3.35E 02	1.13
NEUTRONS/CM2, E GT 1.0MEV	6.85E 15	1.16	2.37E 17	1.16	2.38E 17	1.16	2.49E 17	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.52E 16	1.16	1.56E 18	1.16	1.57E 18	1.16	1.64E 18	1.16
NEUTRONS/CM2, E LT 0.4EV	1.84E 15	1.35	6.38E 16	1.35	6.41E 16	1.35	6.71E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	1.10E 07	1.14	4.14E 08	1.14	4.16E 08	1.14	4.36E 08	1.14

TABLE 2-51
(Continued)

LOCATION - INNER RADIUS OF STRUCTURE, AFT END OF STRUCTURE
 RADIUS (IN) 21.11 STATION (IN) 55.7

Point 23

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	1.41E 00	1.13	5.26E 01	1.13	5.23E 01	1.13	5.54E 01	1.13
BTU/IN3-A-286	4.10E 00	1.14	1.53E 02	1.14	1.54E 02	1.14	1.62E 02	1.14
NEUTRONS/CM2, E GT 1.0MEV	2.22E 15	1.16	7.68E 16	1.16	7.72E 16	1.16	8.07E 16	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.19E 16	1.16	4.11E 17	1.16	4.13E 17	1.16	4.32E 17	1.16
NEUTRONS/CM2, E LT 0.4EV	5.42E 14	1.35	1.87E 16	1.35	1.86E 16	1.35	1.97E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	3.02E 06	1.14	1.14E 08	1.14	1.15E 08	1.14	1.20E 08	1.14

LOCATION - INNER RADIUS OF STRUCTURE, CORE MIDPLANE
 RADIUS (IN) 21.11 STATION (IN) 26.00

Point 24

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	1.00E 01	1.13	3.84E 02	1.13	3.85E 02	1.13	4.05E 02	1.13
BTU/IN3-A-286	2.85E 01	1.14	1.10E 03	1.14	1.10E 03	1.14	1.15E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.62E 16	1.16	5.62E 17	1.16	5.63E 17	1.16	5.90E 17	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.45E 17	1.16	5.03E 18	1.16	5.05E 18	1.16	5.28E 18	1.16
NEUTRONS/CM2, E LT 0.4EV	7.84E 15	1.35	2.71E 17	1.35	2.73E 17	1.35	2.85E 17	1.35
GAMMA RAY DOSE, RADS (CARBON)	2.15E 07	1.14	8.33E 08	1.14	8.38E 08	1.14	8.79E 08	1.14

TABLE 2-51
(Continued)

LOCATION - INNER RADIUS OF STRUCTURE, FORWARD END OF STRUCTURE
 RADIUS (IN) 21.11 STATION (IN) -3.50

Point 25

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-AL-6061	2.80E 00	1.13	1.01E 02	1.13	1.01E 02	1.13	1.06E 02	1.13
BTU/IN3-A-286	8.51E 00	1.14	3.06E 02	1.14	3.07E 02	1.14	3.22E 02	1.14
NEUTRONS/CM2, E GT 1.0MEV	3.12E 15	1.16	1.08E 17	1.16	1.08E 17	1.16	1.13E 17	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.34E 16	1.16	8.08E 17	1.16	8.12E 17	1.16	8.49E 17	1.16
NEUTRONS/CM2, E LT 0.4EV	2.17E 15	1.35	7.50E 16	1.35	7.54E 16	1.35	7.88E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	6.07E 06	1.14	2.20E 08	1.14	2.21E 08	1.14	2.31E 08	1.14

LOCATION - OUTER RADIUS OF STRUCTURE, AFT END OF STRUCTURE
 RADIUS (IN) 22.21 STATION (IN) 55.20

Point 26

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-AL-6061	1.03E 00	1.13	3.81E 01	1.13	3.83E 01	1.13	4.01E 01	1.13
BTU/IN3-A-286	3.08E 00	1.14	1.14E 02	1.14	1.14E 02	1.14	1.20E 02	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.37E 15	1.16	4.73E 16	1.16	4.75E 16	1.16	4.97E 16	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.89E 15	1.16	3.08E 17	1.16	3.09E 17	1.16	3.23E 17	1.16
NEUTRONS/CM2, E LT 0.4EV	8.69E 14	1.35	3.01E 16	1.35	3.02E 16	1.35	3.16E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	2.24E 06	1.14	8.34E 07	1.14	8.39E 07	1.14	8.78E 07	1.14

TABLE 2-51
(Continued)

LOCATION - INNER RADIUS OF REFLECTOR, AFT END OF REFLECTOR
 RADIUS (IN) 22.21 STATION (IN) 52.00

Point 27

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-BERYLLIUM	1.57E 00	1.16	5.68E 01	1.16	5.71E 01	1.16	5.97E 01	1.16
BTU/IN3-AL-6061	1.47E 00	1.13	5.52E 01	1.13	5.55E 01	1.13	5.81E 01	1.13
BTU/IN3-A-286	4.34E 00	1.14	1.63E 02	1.14	1.63E 02	1.14	1.71E 02	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.92E 15	1.16	6.63E 16	1.16	6.66E 16	1.16	6.96E 16	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.63E 16	1.16	5.63E 17	1.16	5.66E 17	1.16	5.92E 17	1.16
NEUTRONS/CM2, E LT 0.4EV	1.17E 15	1.35	4.06E 16	1.35	4.09E 16	1.35	4.27E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	3.21E 06	1.14	1.21E 08	1.14	1.22E 08	1.14	1.28E 08	1.14

LOCATION - INNER RADIUS OF REFLECTOR, CORE MIDPLANE
 RADIUS (IN) 22.21 STATION (IN) 26.00

Point 28

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-BERYLLIUM	9.21E 00	1.18	3.34E 02	1.18	3.35E 02	1.18	3.51E 02	1.18
BTU/IN3-AL-6061	7.73E 00	1.13	2.92E 02	1.13	2.93E 02	1.13	3.07E 02	1.13
BTU/IN3-A-286	2.22E 01	1.14	8.41E 02	1.14	8.45E 02	1.14	8.86E 02	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.17E 16	1.16	4.05E 17	1.16	4.07E 17	1.16	4.25E 17	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.28E 17	1.16	4.43E 18	1.16	4.45E 18	1.16	4.65E 18	1.16
NEUTRONS/CM2, E LT 0.4EV	8.44E 15	1.35	2.92E 17	1.35	2.93E 17	1.35	3.07E 17	1.35
GAMMA RAY DOSE, RADS (CARBON)	1.65E 07	1.14	6.31E 08	1.14	6.34E 08	1.14	6.64E 08	1.14

TABLE 2-51
(Continued)

LOCATION - INNER RADIUS OF REFLECTOR, FORWARD END OF REFLECTOR
 RADIUS (IN) 22.21 STATION (IN) 0.0

Point 29

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BERYLLIUM	2.91E 00	1.16	1.05E 02	1.16	1.05E 02	1.16	1.10E 02	1.16
BTU/IN3-AL-6061	2.87E 00	1.13	1.05E 02	1.13	1.06E 02	1.13	1.11E 02	1.13
BTU/IN3-A-286	8.57E 00	1.14	3.15E 02	1.14	3.17E 02	1.14	3.32E 02	1.14
NEUTRONS/CM2, E GT 1.0MEV	3.31E 15	1.16	1.14E 17	1.16	1.15E 17	1.16	1.20E 17	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.17E 16	1.16	1.10E 18	1.16	1.10E 18	1.16	1.15E 18	1.16
NEUTRONS/CM2, E LT 0.4EV	2.67E 15	1.35	9.24E 16	1.35	9.29E 16	1.35	9.71E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	6.24E 06	1.14	2.32E 08	1.14	2.33E 08	1.14	2.44E 08	1.14

LOCATION - OUTER RADIUS OF REFLECTOR, AFT END OF REFLECTOR
 RADIUS (IN) 27.00 STATION (IN) 52.00

Point 30

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BERYLLIUM	4.93E-01	1.13	1.80E 01	1.13	1.81E 01	1.13	1.89E 01	1.13
BTU/IN3-AL-7039	6.58E-01	1.14	2.43E 01	1.14	2.45E 01	1.14	2.56E 01	1.14
BTU/IN3-A-286	2.03E 00	1.14	7.49E 01	1.14	7.52E 01	1.14	7.88E 01	1.14
NEUTRONS/CM2, E GT 1.0MEV	3.33E 14	1.16	1.15E 16	1.16	1.16E 16	1.16	1.21E 16	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.88E 15	1.16	9.96E 16	1.16	1.00E 17	1.16	1.05E 17	1.16
NEUTRONS/CM2, E LT 0.4EV	5.16E 14	1.35	1.78E 16	1.35	1.79E 16	1.35	1.87E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	1.48E 06	1.14	5.51E 07	1.14	5.54E 07	1.14	5.80E 07	1.14

TABLE 2-51
(Continued)

LOCATION - OUTER RADIUS OF REFLECTOR, CORE MIDPLANE
 RADIUS (IN) 27.00 STATION (IN) 26.00

Point 31

	STATE POINT (B)	STATE POINT (C)	STATE POINT (D)	STATE POINT (E)
	NOMINAL UNC FACTR	NOMINAL UNC FACTR	NOMINAL UNC FACTR	NOMINAL UNC FACTR
BTU/IN3-BERYLLIUM	2.09E 00	1.13	7.68E 01	1.13
BTU/IN3-AL-7039	2.70E 00	1.14	1.01E 02	1.14
BTU/IN3-A-286	8.26E 00	1.14	3.08E 02	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.54E 15	1.16	5.31E 16	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.61E 16	1.16	5.59E 17	1.16
NEUTRONS/CM2, E LT 0.4EV	3.05E 15	1.35	1.06E 17	1.35
GAMMA RAY DOSE, RADS (CARBON)	6.07E 06	1.14	2.29E 08	1.14

LOCATION - OUTER RADIUS OF REFLECTOR, FORWARD END OF REFLECTOR
 RADIUS (IN) 27.00 STATION (IN) 0.0

Point 32

	STATE POINT (B)	STATE POINT (C)	STATE POINT (D)	STATE POINT (E)
	NOMINAL UNC FACTR	NOMINAL UNC FACTR	NOMINAL UNC FACTR	NOMINAL UNC FACTR
BTU/IN3-BERYLLIUM	9.68E-01	1.13	3.49E 01	1.13
BTU/IN3-AL-7039	1.27E 00	1.14	4.63E 01	1.14
BTU/IN3-A-286	3.94E 00	1.14	1.43E 02	1.14
NEUTRONS/CM2, E GT 1.0MEV	6.77E 14	1.16	2.34E 16	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.09E 15	1.16	2.11E 17	1.16
NEUTRONS/CM2, E LT 0.4EV	1.03E 15	1.35	3.55E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	2.84E 06	1.14	1.04E 08	1.14

TABLE 2-51
(Continued)

LOCATION - PRESSURE VESSEL SIDE SURFACE, PRESSURE VESSEL - NOZZLE MATING FLANGE
 RADIUS (IN) 27.89 STATION (IN) 63.20

Point 33

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-7039	2.00E-01	1.40	7.36E 00	1.40	7.40E 00	1.40	7.74E 00	1.40
NEUTRONS/CM2, E GT 1.0MEV	2.63E 14	3.10	9.11E 15	3.10	9.16E 15	3.10	9.57E 15	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.20E 14	3.10	2.47E 16	3.10	2.50E 16	3.10	2.62E 16	3.10
NEUTRONS/CM2, E LT 0.4EV	7.42E 13	3.10	2.57E 15	3.10	2.58E 15	3.10	2.7CE 15	3.10
GAMMA RAY DOSE, RADS (CARBON)	4.26E 05	1.40	1.58E 07	1.40	1.59E 07	1.40	1.67E 07	1.40

LOCATION - PRESSURE VESSEL SIDE SURFACE, CORE MIDPLANE
 RADIUS (IN) 27.89 STATION (IN) 26.00

Point 34

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-7039	1.91E 00	1.39	7.15E 01	1.39	7.18E 01	1.39	7.52E 01	1.39
NEUTRONS/CM2, E GT 1.0MEV	1.00E 15	3.10	3.46E 16	3.10	3.48E 16	3.10	3.63E 16	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.57E 15	3.10	2.97E 17	3.10	2.98E 17	3.10	3.12E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	1.62E 15	3.10	5.62E 16	3.10	5.65E 16	3.10	5.91E 16	3.10
GAMMA RAY DOSE, RADS (CARBON)	4.32E 06	1.40	1.63E 08	1.40	1.64E 08	1.40	1.71E 08	1.40

TABLE 2-51
(Continued)

LOCATION - PRESSURE VESSEL SIDE SURFACE, TANGENT PLANE
 RADIUS (IN) 27.89 STATION (IN) -37.00

Point 35

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-AL-7039	2.37E-03	1.39	8.24E-02	1.39	8.28E-02	1.39	8.65E-02	1.39
NEUTRONS/CM2, E GT 1.0MEV	1.74E 12	3.10	6.03E 13	3.10	6.06E 13	3.10	6.34E 13	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.35E 12	3.10	1.51E 14	3.10	1.51E 14	3.10	1.58E 14	3.10
NEUTRONS/CM2, E LT 0.4EV	7.56E 11	3.10	2.62E 13	3.10	2.63E 13	3.10	2.75E 13	3.10
GAMMA RAY DOSE, RADS (CARBON)	5.15E 03	1.40	1.80E 05	1.40	1.81E 05	1.40	1.89E 05	1.40

LOCATION - OUTER RADIUS OF SUPPORT PLATE, AFT FACE OF SUPPORT PLATE
 RADIUS (IN) 22.21 STATION (IN) -3.00

Point 36

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-AL-6061	2.19E 00	1.09	7.86E 01	1.09	7.90E 01	1.09	8.26E 01	1.09
BTU/IN3-A-286	6.74E 00	1.09	2.42E 02	1.09	2.43E 02	1.09	2.54E 02	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.15E 15	1.71	7.43E 16	1.71	7.47E 16	1.71	7.81E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.79E 16	1.71	6.14E 17	1.71	6.22E 17	1.71	6.50E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	2.22E 15	1.88	7.64E 16	1.88	7.73E 16	1.88	8.08E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	4.79E 06	1.09	1.73E 08	1.09	1.74E 08	1.09	1.82E 08	1.09

TABLE 2-51
(Continued)

LOCATION - OUTER RADIUS OF SUPPORT PLATE, FORWARD FACE OF SUPPORT PLATE
 RADIUS (IN) 22.21 STATION (IN) -9.50

Point 37

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-AL-6061	1.03E 00	1.09	3.62E 01	1.09	3.63E 01	1.09	3.80E 01	1.09
BTU/IN3-A-286	3.23E 00	1.09	1.13E 02	1.09	1.14E 02	1.09	1.19E 02	1.09
NEUTRONS/CM2, E GT 1.0MEV	7.81E 14	1.71	2.70E 16	1.71	2.72E 16	1.71	2.84E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.82E 15	1.71	2.36E 17	1.71	2.37E 17	1.71	2.48E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	1.01E 15	1.88	3.48E 16	1.88	3.50E 16	1.88	3.66E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.28E 06	1.09	8.04E 07	1.09	8.08E 07	1.09	8.44E 07	1.09

LOCATION - OUTER RADIUS OF BATH CENTRAL SHIELD, AFT FACE OF CENTRAL SHIELD PLATE
 RADIUS (IN) 19.48 STATION (IN) -10.83

Point 38

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-AL-6061	1.71E 00	1.09	4.25E 01	1.09	4.27E 01	1.09	4.47E 01	1.09
BTU/IN3-A-286	3.77E 00	1.09	1.32E 02	1.09	1.33E 02	1.09	1.38E 02	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.08E 15	1.71	3.75E 16	1.71	3.77E 16	1.71	3.94E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.48E 15	1.71	2.93E 17	1.71	2.95E 17	1.71	3.08E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	7.38E 14	1.88	2.55E 16	1.88	2.57E 16	1.88	2.69E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.67E 06	1.09	9.38E 07	1.09	9.42E 07	1.09	9.85E 07	1.09

TABLE 2-51
(Continued)

LOCATION - OUTER RADIUS OF BATH CENTRAL SHIELD, INT FACE OF BATH CENTRAL SHIELD
 RADIUS (IN) 19.48 STATION (IN) -11.3

Point 39

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)
 NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	9.24E 00	1.78	3.20E 02	1.78	3.22E 02	1.78	3.36E 02	1.78
BTU/IN3-AL-6061	1.01E 00	1.09	3.53E 01	1.09	3.55E 01	1.09	3.71E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	8.52E 14	1.71	2.95E 16	1.71	2.96E 16	1.71	3.10E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.74E 15	1.71	2.33E 17	1.71	2.35E 17	1.71	2.45E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	3.37E 14	1.88	1.17E 16	1.88	1.17E 16	1.88	1.23E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.25E 06	1.09	7.87E 07	1.09	7.91E 07	1.09	8.26E 07	1.09

LOCATION - OUTER RADIUS OF BATH CENTRAL SHIELD, BATH CENTRAL SHIELD MIDPLANE
 RADIUS (IN) 19.48 STATION (IN) -11.3

Point 40

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)
 NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	5.99E-01	1.53	2.08E 01	1.53	2.09E 01	1.53	2.18E 01	1.53
BTU/IN3-AL-6061	2.30E-01	1.09	8.00E 00	1.09	8.04E 00	1.09	8.40E 00	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.48E 14	1.71	5.13E 15	1.71	5.16E 15	1.71	5.39E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.07E 14	1.71	2.79E 16	1.71	2.81E 16	1.71	2.93E 16	1.71
NEUTRONS/CM2, E LT 0.4EV	2.72E 12	1.88	9.40E 13	1.88	9.44E 13	1.88	9.88E 13	1.88
GAMMA RAY DOSE, RADS (CARBON)	5.23E 05	1.09	1.82E 07	1.09	1.83E 07	1.09	1.91E 07	1.09

TABLE 2-51
(Continued)

LOCATION - OUTER RADIUS OF BATH CENTRAL SHIELD, FORWARD FACE OF BATH CENTRAL SHIELD
RADIUS (IN) 19.48 STATION (IN) -22.93

Point 41

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	1.50E-01	1.59	5.21E 00	1.59	5.24E 00	1.59	5.47E 00	1.59
BTU/IN3-LEAD ALLOY	4.50E-01	1.09	1.56E 01	1.09	1.57E 01	1.09	1.64E 01	1.09
BTU/IN3-AL-6061	4.83E-02	1.09	1.68E 00	1.09	1.69E 00	1.09	1.76E 00	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.93E 13	1.71	1.01E 15	1.71	1.02E 15	1.71	1.06E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.36E 14	1.71	4.72E 15	1.71	4.74E 15	1.71	4.96E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	3.33E 12	1.88	1.15E 14	1.88	1.16E 14	1.88	1.21E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.08E 05	1.09	3.75E 06	1.09	3.77E 06	1.09	3.94E 06	1.09

LOCATION - OUTER RADIUS OF BATH CENTRAL SHIELD, FORWARD FACE OF LEAD CENTRAL SHIELD
RADIUS (IN) 19.48 STATION (IN) -24.23

Point 42

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-LEAD ALLOY	1.32E-01	1.09	4.56E 00	1.09	4.58E 00	1.09	4.79E 00	1.09
BTU/IN3-AL-6061	1.67E-02	1.09	5.78E-01	1.09	5.81E-01	1.09	6.07E-01	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.01E 13	1.71	6.96E 14	1.71	6.99E 14	1.71	7.31E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.81E 13	1.71	3.40E 15	1.71	3.41E 15	1.71	3.57E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	7.93E 12	1.88	2.74E 14	1.88	2.76E 14	1.88	2.88E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.54E 04	1.09	1.23E 06	1.09	1.24E 06	1.09	1.29E 06	1.09

TABLE 2-51
(Continued)

LOCATION - OUTER RADIUS OF BATH CENTRAL SHIELD, AFT FACE OF FLOW BAFFLE II
 RADIUS (IN) 19.48 STATION (IN) -29.63

Point 43

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	9.48E-03	1.40	3.28E-01	1.40	3.30E-01	1.40	3.45E-01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.24E 13	3.10	4.28E 14	3.10	4.31E 14	3.10	4.50E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.90E 13	3.10	2.04E 15	3.10	2.05E 15	3.10	2.14E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.09E 13	3.10	3.76E 14	3.10	3.77E 14	3.10	3.95E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.01E 04	1.40	6.98E 05	1.40	7.01E 05	1.40	7.32E 05	1.40

LOCATION - OUTER RADIUS OF BATH CENTRAL SHIELD, FORWARD FACE OF FLOW BAFFLE II
 RADIUS (IN) 19.48 STATION (IN) -30.75

Point 44

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	8.04E-03	1.40	2.78E-01	1.40	2.79E-01	1.40	2.92E-01	1.40
NEUTRONS/CM2, E GT 1.0MEV	9.09E 12	3.10	3.14E 14	3.10	3.16E 14	3.10	3.30E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.61E 13	3.10	1.59E 15	3.10	1.60E 15	3.10	1.67E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.11E 13	3.10	3.85E 14	3.10	3.87E 14	3.10	4.05E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	1.73E 04	1.40	5.99E 05	1.40	6.02E 05	1.40	6.29E 05	1.40

TABLE 2-51
(Continued)

LOCATION - I.R. OF BATH PERIPHERAL SHIELD 1, AFT FACE OF PERIPHERAL SHIELD
 RADIUS (IN) 20.50 STATION (IN) -11.04

Point 45

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	1.62E 01	1.82	5.62E 02	1.82	5.62E 02	1.82	5.91E 02	1.82
BTU/IN3-AL-6061	1.05E 00	1.09	3.69E 01	1.09	3.71E 01	1.09	3.87E 01	1.09
BTU/IN3-SS-304	3.26E 00	1.09	1.14E 02	1.09	1.14E 02	1.09	1.19E 02	1.09
NEUTRONS/CM2, E GT 1.0MEV	8.54E 14	1.71	2.96E 16	1.71	2.97E 16	1.71	3.11E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.98E 15	1.71	2.42E 17	1.71	2.43E 17	1.71	2.54E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	5.50E 14	1.88	1.90E 16	1.88	1.91E 16	1.88	2.00E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.34E 06	1.09	8.21E 07	1.09	8.25E 07	1.09	8.62E 07	1.09

LOCATION - I.R. OF BATH PERIPHERAL SHIELD 1, FLR. FACE OF BATH PER. SHIELD 1
 RADIUS (IN) 20.50 STATION (IN) -20.14

Point 46

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	2.27E-01	1.65	7.86E 00	1.65	7.9 E 00	1.65	8.26E 00	1.65
BTU/IN3-LEAD ALLOY	5.48E-01	1.09	1.90E 01	1.09	1.91E 01	1.09	2.00E 01	1.09
BTU/IN3-AL-6061	5.64E-02	1.09	1.96E 00	1.09	1.97E 00	1.09	2.05E 00	1.09
BTU/IN3-SS-304	1.78E-01	1.09	6.16E 00	1.09	6.19E 00	1.09	6.47E 00	1.09
NEUTRONS/CM2, E GT 1.0MEV	3.43E 13	1.71	1.19E 15	1.71	1.19E 15	1.71	1.25E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.64E 14	1.71	5.67E 15	1.71	5.70E 15	1.71	5.90E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	2.46E 12	1.88	8.51E 13	1.88	8.56E 13	1.88	8.94E 13	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.27E 05	1.09	4.41E 06	1.09	4.43E 06	1.09	4.63E 06	1.09

TABLE 2-51
(Continued)

LOCATION - I.R. OF BATH PERIPHERAL SHIELD I, FLR. FACE OF LEAD PER. SHIELD I
 RADIUS (IN) 20.50 STATION (IN) -23.44

Point 47

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-LEAD ALLOY	2.29E-01	1.09	7.95E-00	1.09	7.99E-00	1.09	8.35E-00	1.09
BTU/IN3-AL-6061	2.80E-02	1.09	9.71E-01	1.09	9.75E-01	1.09	1.02E-00	1.09
BTU/IN3-SS-304	8.75E-02	1.09	3.04E-00	1.09	3.05E-00	1.09	3.19E-00	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.41E-13	1.71	8.34E-14	1.71	8.39E-14	1.71	8.77E-14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.16E-14	1.71	4.00E-15	1.71	4.02E-15	1.71	4.20E-15	1.71
NEUTRONS/CM2, E LT 0.4EV	6.01E-12	1.88	2.08E-14	1.88	2.09E-14	1.88	2.14E-14	1.88
GAMMA RAY DOSE, RADS (CARBON)	6.08E-04	1.09	2.12E-06	1.09	2.13E-06	1.09	2.22E-06	1.09

LOCATION - O.R. OF BATH PERIPHERAL SHIELD I, AFT FACE OF PERIPHERAL SHIELD
 RADIUS (IN) 22.04 STATION (IN) -11.44

Point 48

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	8.59E-00	1.78	2.97E-02	1.78	2.97E-02	1.78	3.12E-02	1.78
BTU/IN3-AL-6061	9.11E-01	1.09	3.19E-01	1.09	3.20E-01	1.09	3.35E-01	1.09
BTU/IN3-SS-304	2.82E-00	1.09	9.86E-01	1.09	9.91E-01	1.09	1.04E-02	1.09
NEUTRONS/CM2, E GT 1.0MEV	6.29E-14	1.71	2.17E-16	1.71	2.19E-16	1.71	2.28E-16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.33E-15	1.71	1.84E-17	1.71	1.85E-17	1.71	1.94E-17	1.71
NEUTRONS/CM2, E LT 0.4EV	4.43E-14	1.88	1.53E-16	1.88	1.54E-16	1.88	1.61E-16	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.05E-06	1.09	7.18E-07	1.09	7.22E-07	1.09	7.54E-07	1.09

TABLE 2-51
(Continued)

LOCATION - O.R. OF BATH PERIPHERAL SHIELD 1, FOR FACE OF LEAD PER. SHIELD 1
RADIUS (IN) 22.04 STATION (IN) -1.1

Point 49

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	1.33E-01	1.56	4.63E-00	1.56	4.63E-00	1.56	4.81E-00	1.56
BTU/IN3-LEAD ALLOY	4.30E-01	1.09	1.50E-01	1.09	1.50E-01	1.09	1.57E-01	1.09
BTU/IN3-AL-6061	4.67E-02	1.09	1.63E-00	1.09	1.63E-00	1.09	1.71E-00	1.09
BTU/IN3-SS-304	1.47E-01	1.09	5.11E-00	1.09	5.11E-00	1.09	5.37E-00	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.95E-13	1.71	1.02E-15	1.71	1.03E-15	1.71	1.07E-15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.24E-14	1.71	4.28E-15	1.71	4.40E-15	1.71	4.43E-15	1.71
NEUTRONS/CM2, E LT 0.4EV	2.13E-12	1.88	7.37E-13	1.88	7.41E-13	1.88	7.75E-13	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.05E-05	1.09	3.65E-06	1.09	3.66E-06	1.09	3.93E-06	1.09

LOCATION - O.R. OF BATH PERIPHERAL SHIELD 1, FOR FACE OF LEAD PER. SHIELD 1
RADIUS (IN) 22.04 STATION (IN) -23.44

Point 50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-LEAD ALLOY	6.43E-02	1.09	2.93E-00	1.09	2.94E-00	1.09	3.08E-00	1.09
BTU/IN3-AL-6061	1.21E-02	1.09	4.22E-01	1.09	4.24E-01	1.09	4.43E-01	1.09
BTU/IN3-SS-304	3.58E-02	1.09	1.25E-00	1.09	1.25E-00	1.09	1.31E-00	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.17E-13	1.71	7.49E-14	1.71	7.53E-14	1.71	7.87E-14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	9.21E-13	1.71	3.19E-15	1.71	3.20E-15	1.71	3.35E-15	1.71
NEUTRONS/CM2, E LT 0.4EV	6.12E-12	1.88	2.12E-14	1.88	2.13E-14	1.88	2.22E-14	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.56E-04	1.09	4.71E-05	1.09	4.75E-05	1.09	9.14E-05	1.09

TABLE 2-51
(Continued)

LOCATION - O.R. OF BATH PERIPHERAL SHIELD II, AFT FACE OF PERIPHERAL SHIELD
 RADIUS (IN) 23.63 STATION (IN) -11.04

Point 51

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	7.94E 00	1.78	2.75E 02	1.78	2.77E 02	1.78	2.89E 02	1.78
BTU/IN3-AL-6061	8.50E-01	1.09	2.98E 01	1.09	3.01E 01	1.09	3.13E 01	1.09
BTU/IN3-SS-304	2.63E 00	1.09	9.23E 01	1.09	9.28E 01	1.09	9.70E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	5.70E 14	1.71	1.97E 16	1.71	1.98E 16	1.71	2.07E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.60E 15	1.71	1.59E 17	1.71	1.60E 17	1.71	1.67E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	4.13E 14	1.88	1.43E 16	1.88	1.44E 16	1.88	1.50E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.91E 06	1.09	6.73E 07	1.09	6.77E 07	1.09	7.07E 07	1.09

LOCATION - O.R. OF BATH PERIPHERAL SHIELD II, FOR. FACE OF BATH PER. SHIELD II
 RADIUS (IN) 23.63 STATION (IN) -22.14

Point 52

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	1.06E-01	1.57	3.67E 00	1.57	3.69E 00	1.57	3.86E 00	1.57
BTU/IN3-LEAD ALLOY	3.31E-01	1.09	1.15E 01	1.09	1.17E 01	1.09	1.21E 01	1.09
BTU/IN3-AL-6061	3.65E-02	1.09	1.27E 00	1.09	1.27E 00	1.09	1.34E 00	1.09
BTU/IN3-SS-304	1.15E-01	1.09	4.00E 00	1.09	4.02E 00	1.09	4.20E 00	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.22E 13	1.71	7.70E 14	1.71	7.74E 14	1.71	8.07E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.88E 13	1.71	3.07E 15	1.71	3.07E 15	1.71	3.23E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.67E 12	1.88	5.77E 13	1.88	5.8E 13	1.88	6.07E 13	1.88
GAMMA RAY DOSE, RADS (CARBON)	8.17E 04	1.09	2.85E 06	1.09	2.86E 06	1.09	2.99E 06	1.09

TABLE 2-51
(Continued)

LOCATION - D.R. OF BATH PERIPHERAL SHIELD II, FOR. FACE OF LEAD PFP. SHIELD II
 RADIUS (IN) 23.63 STATION (IN) -23.44

Point 53

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-LEAD ALLOY	5.63E-02	1.09	1.96E 00	1.09	1.97E 00	1.09	2.05E 00	1.09
BTU/IN3-AL-6061	8.25E-03	1.09	2.87E-01	1.09	2.88E-01	1.09	3.01E-01	1.09
BTU/IN3-SS-304	2.42E-02	1.09	8.39E-01	1.09	8.43E-01	1.09	8.81E-01	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.63E 13	1.71	5.66E 14	1.71	5.69E 14	1.71	5.94E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.80E 13	1.71	2.35E 15	1.71	2.37E 15	1.71	2.47E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	5.19E 12	1.88	1.80E 14	1.88	1.81E 14	1.88	1.89E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.67E 04	1.09	5.83E 05	1.09	5.85E 05	1.09	6.12E 05	1.09

LOCATION - D.R. OF BATH PERIPHERAL SHIELD III, AFT FACE OF PERIPHERAL SHIELD
 RADIUS (IN) 25.38 STATION (IN) -11.04

Point 54

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	6.64E 00	1.78	2.30E 02	1.78	2.31E 02	1.78	2.42E 02	1.78
BTU/IN3-AL-6061	7.56E-01	1.09	2.67E 01	1.09	2.69E 01	1.09	2.81E 01	1.09
BTU/IN3-SS-304	2.33E 00	1.09	8.23E 01	1.09	8.28E 01	1.09	8.65E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	5.71E 14	1.71	1.98E 16	1.71	1.99E 16	1.71	2.08E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.82E 15	1.71	1.32E 17	1.71	1.33E 17	1.71	1.39E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	3.24E 14	1.88	1.12E 16	1.88	1.13E 16	1.88	1.18E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.70E 06	1.09	6.02E 07	1.09	6.05E 07	1.09	6.33E 07	1.09

TABLE 2-51
(Continued)

Page 28 of 30

LOCATION - O.R. OF BATH PERIPHERAL SHIELD III, FOR. FACE OF BATH PER. SHIELD III
 RADIUS (IN) 25.38 STATION (IN) -22.14

Point 55

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	7.21E-02	1.59	2.50E 00	1.59	2.51E 00	1.59	2.63E 00	1.59
BTU/IN3-LEAD ALLOY	1.95E-01	1.09	6.78E 00	1.09	6.81E 00	1.09	7.12E 00	1.09
BTU/IN3-AL-6061	2.31E-02	1.09	8.05E-01	1.09	8.09E-01	1.09	8.45E-01	1.09
BTU/IN3-SS-304	7.30E-02	1.09	2.54E 00	1.09	2.55E 00	1.09	2.66E 00	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.62E 13	1.71	5.59E 14	1.71	5.62E 14	1.71	5.93E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.62E 13	1.71	1.94E 15	1.71	1.95E 15	1.71	2.04E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.68E 12	1.88	5.80E 13	1.88	5.83E 13	1.88	6.09E 13	1.88
GAMMA RAY DOSE, RADS (CARBON)	5.10E 04	1.09	1.78E 06	1.09	1.79E 06	1.09	1.87E 06	1.09

LOCATION - O.R. OF BATH PERIPHERAL SHIELD III, FOR. FACE OF LEAD PER. SHIELD III
 RADIUS (IN) 25.38 STATION (IN) -23.44

Point 56

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-LEAD ALLOY	5.97E-02	1.09	2.07E 00	1.09	2.08E 00	1.09	2.18E 00	1.09
BTU/IN3-AL-6061	8.17E-03	1.09	2.84E-01	1.09	2.86E-01	1.09	2.98E-01	1.09
BTU/IN3-SS-304	2.49E-02	1.09	8.66E-01	1.09	8.70E-01	1.09	9.09E-01	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.16E 13	1.71	4.02E 14	1.71	4.04E 14	1.71	4.23E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.44E 13	1.71	1.54E 15	1.71	1.54E 15	1.71	1.61E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	4.10E 12	1.88	1.42E 14	1.88	1.43E 14	1.88	1.49E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.71E 04	1.09	5.96E 05	1.09	5.99E 05	1.09	6.26E 05	1.09

TABLE 2-51
(Continued)

LOCATION - O.R. OF BATH PERIPHERAL SHIELD IV, AFT FACE OF PERIPHERAL SHIELD
 RADIUS (IN) 27.00 STATION (IN) -11.00

Point 57

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	5.58E 00	1.78	1.94E 02	1.78	1.95E 02	1.78	2.43E 02	1.78
BTU/IN3-AL-6061	6.12E-01	1.09	2.18E 01	1.09	2.19E 01	1.09	2.79E 01	1.09
BTU/IN3-SS-304	1.84E 00	1.09	6.72E 01	1.09	6.75E 01	1.09	7.65E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	4.94E 14	1.71	1.71E 16	1.71	1.72E 16	1.71	1.79E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.56E 15	1.71	8.87E 16	1.71	8.91E 16	1.71	9.21E 16	1.71
NEUTRONS/CM2, E LT 0.4EV	2.22E 14	1.88	7.66E 15	1.88	7.70E 15	1.88	8.35E 15	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.37E 06	1.09	4.89E 07	1.09	4.91E 07	1.09	5.13E 07	1.09

LOCATION - O.R. OF BATH PERIPHERAL SHIELD IV, FOR. FACE OF BATH PER. SHIELD IV
 RADIUS (IN) 27.00 STATION (IN) -20.00

Point 58

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	7.13E-02	1.48	2.48E 00	1.48	2.49E 00	1.48	2.60E 00	1.48
BTU/IN3-LEAD ALLOY	2.75E-01	1.09	9.59E 00	1.09	9.64E 00	1.09	1.01E 01	1.09
BTU/IN3-AL-6061	3.13E-02	1.09	1.09E 00	1.09	1.10E 00	1.09	1.15E 00	1.09
BTU/IN3-SS-304	1.00E-01	1.09	3.49E 00	1.09	3.50E 00	1.09	3.66E 00	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.82E 13	1.71	6.28E 14	1.71	6.31E 14	1.71	6.60E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.92E 13	1.71	1.70E 15	1.71	1.71E 15	1.71	1.79E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	5.21E 11	1.88	1.80E 13	1.88	1.81E 13	1.88	1.89E 13	1.88
GAMMA RAY DOSE, RADS (CARBON)	6.98E 04	1.09	2.44E 06	1.09	2.45E 06	1.09	2.56E 06	1.09

TABLE 2-51
(Continued)

LOCATION - O.R. OF BATH PERIPHERAL SHIELD IV, FUP. FACE OF LEAD PER. SHIELD IV
 RADIUS (IN) 27.00 STATION (IN) -21.94

Point 59

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-LEAD ALLOY	7.28E-02	1.09	2.54E-00	1.09	2.55E-00	1.09	2.67E-00	1.09
STU/IN3-AL-6061	9.37E-03	1.09	3.27E-01	1.09	3.29E-01	1.09	3.43E-01	1.09
BTU/IN3-SS-304	2.86E-02	1.09	9.98E-01	1.09	1.00E-00	1.09	1.05E-00	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.27E-13	1.71	4.41E-14	1.71	4.43E-14	1.71	4.63E-14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.70E-13	1.71	1.28E-15	1.71	1.29E-15	1.71	1.35E-15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.64E-12	1.88	5.67E-13	1.88	5.70E-13	1.88	5.96E-13	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.99E-04	1.09	6.96E-05	1.09	6.99E-05	1.09	7.30E-05	1.09

2-193

TABLE 2-52

GAMMA RAY SOURCE TERMS FOR COMPOSITE CORE AT TRANSIENT STATE POINTS

Gamma Group	Energy (MeV)	Source (MeV/sec) at State Points				
		B	C	D	E	F
1	7.5 - 10.	1.57(18)	1.52(18)	9.62(17)	9.78(17)	0
2	7.0 - 7.5	2.33(18)	2.25(18)	1.42(18)	1.45(18)	0
3	6.0 - 7.0	6.44(18)	6.24(18)	3.94(18)	4.01(18)	0
4	5.0 - 6.0	1.30(19)	1.37(19)	9.22(18)	9.07(18)	1.25(18)
5	4.0 - 5.0	3.35(19)	3.81(19)	2.71(19)	2.60(19)	6.70(18)
6	3.0 - 4.0	6.07(19)	7.29(19)	5.36(19)	5.05(19)	1.68(19)
7	2.6 - 3.0	3.54(19)	4.31(19)	3.20(19)	3.01(19)	1.06(19)
8	2.2 - 2.6	4.95(19)	6.00(19)	4.43(19)	4.17(19)	1.43(19)
9	1.8 - 2.2	6.71(19)	7.91(19)	5.75(19)	5.45(19)	1.68(19)
10	1.35 - 1.8	8.46(19)	1.02(20)	7.56(19)	7.11(19)	2.44(19)
11	0.9 - 1.35	1.16(20)	1.36(20)	9.90(19)	9.39(19)	2.89(19)
12	0.4 - 0.9	1.63(20)	1.86(20)	1.32(20)	1.27(20)	3.30(19)
13	0.0 - 0.4	8.47(19)	8.85(19)	5.93(19)	5.85(19)	7.63(18)
	TOTAL	7.18(20)	8.29(20)	5.94(20)	5.69(20)	1.60(20)

NOTE: Numbers in parentheses indicate powers of ten.

086-10/14/70

TABLE 2-53

FACTORS TO CONVERT EXTRA-CORE SECONDARY GAMMA SOURCE TERMS TO
TRANSIENT STATE POINTS

<u>State Point</u>	<u>R-1 w/Composite Fuel</u>	<u>R-1 w/Graphite Fuel</u>
B	1.030	1.007
C	0.997	0.976
D	0.630	0.617
E	0.641	0.627
F	0.000	0.000

TABLE 2-54

RADIATION ENVIRONMENT AT KEY NSS LOCATIONS - GRAPHITE FUEL IN THE CORE

Location	Coordinates		Gamma Environment				NEUTRON ENVIRONMENT								
	Radius (IN.)	Reactor Station (IN.)	Heating Rate (w/gm - SS)	Dose Rate (RADS-C/HR)	1 HR. Dose (RADS-C)	10 HR. Dose (RADS-C)	Fast ($E > 1.0$ mev)			Intermediate ($0.4 \text{ ev} \leq E \leq 1.0$ mev)			Thermal ($E < 0.4 \text{ ev}$)		
							Flux ($\text{N/cm}^2 \cdot \text{sec}$)	1 HR. Fluence (N/cm^2)	10 HR. Fluence (N/cm^2)	Flux ($\text{N/cm}^2 \cdot \text{sec}$)	1 HR. Fluence (N/cm^2)	10 HR. Fluence (N/cm^2)	Flux ($\text{N/cm}^2 \cdot \text{sec}$)	1 HR. Fluence (N/cm^2)	10 HR. Fluence (N/cm^2)
Reactor Centerline - Core Midplane	0.000	26.000	9.44 (1)	3.39 (10)	3.39 (10)	3.39 (11)	4.50 (15)	1.62 (19)	1.62 (20)	2.29 (16)	8.24 (19)	8.24 (20)	1.82 (13)	6.55 (16)	6.55 (17)
Inner Surface of Reflector - Core Midplane	22.215	26.000	1.45 (1)	5.03 (9)	5.03 (9)	5.03 (10)	6.84 (14)	2.46 (18)	2.46 (19)	7.32 (15)	2.64 (19)	2.64 (20)	5.46 (14)	1.97 (18)	1.97 (19)
Pressure Vessel Surface - Core Midplane	27.890	26.000	3.72 (0)	1.26 (9)	1.26 (9)	1.26 (10)	5.93 (13)	2.13 (17)	2.13 (18)	4.83 (14)	1.74 (18)	1.74 (19)	9.86 (13)	3.55 (17)	3.55 (18)
Reactor Centerline - Air Face of CHESH	0.000	54.131	5.84 (0)	2.02 (9)	2.02 (9)	2.02 (10)	3.31 (14)	1.19 (18)	1.19 (19)	1.12 (15)	4.03 (18)	4.03 (19)	4.71 (12)	1.70 (16)	1.70 (17)
Reactor Centerline - Air Face of BATH Central Shield	0.000	-11.830	3.65 (0)	1.21 (9)	1.21 (9)	1.21 (10)	1.13 (14)	4.07 (17)	4.07 (18)	8.92 (14)	3.21 (18)	3.21 (19)	3.77 (13)	1.36 (17)	1.36 (18)
Reactor Centerline - Pressure Vessel Dome	0.000	-37.000	1.37 (-2)	4.38 (6)	4.38 (6)	4.38 (7)	6.24 (11)	2.25 (15)	2.25 (16)	1.72 (12)	6.34 (15)	6.34 (16)	2.90 (11)	1.04 (15)	1.04 (16)

Numbers in Parentheses Refer to Powers of Ten to be Applied to the Data

TABLE 2-55

PRESSURE VESSEL AND REACTOR ASSEMBLY SOURCE STRENGTHS

Graphite Core

Energy (MeV)	GAMMA SOURCE STRENGTH (MEV/SEC)								
	REGION 1 Hot End Support Hardware	REGION 2 Core	REGION 3 Core Periphery	REGION 4 Lateral Support	REGION 5 Aluminum Structure	REGION 6 Reflector	REGION 7 Pressure Vessel Side - 1	REGION 8 Aft Reflector Hardware	REGION 9 Core Plenum
8.50	2.34 (17)	1.56 (18)	1.57 (17)	2.68 (14)	1.47 (18)	5.20 (16)	1.43 (17)	5.89 (16)	8.02 (17)
7.25	2.00 (17)	2.31 (18)	2.05 (17)		4.67 (17)	1.32 (18)	5.64 (16)	2.75 (16)	5.16 (17)
6.50	2.03 (17)	6.39 (16)	6.11 (17)		6.17 (17)	2.54 (18)	7.28 (16)	2.31 (16)	7.87 (17)
5.50	1.75 (17)	1.44 (17)	6.74 (17)		5.75 (17)	1.54 (18)	7.15 (16)	2.72 (16)	1.11 (18)
4.50	1.77 (17)	4.10 (19)	1.03 (18)	3.57 (17)	8.88 (17)	1.58 (18)	1.41 (17)	2.94 (16)	1.25 (18)
3.50	1.53 (17)	7.96 (19)	1.17 (18)	8.84 (15)	1.22 (18)	1.65 (18)	1.99 (17)	2.69 (16)	1.62 (18)
2.80	1.53 (17)	4.73 (19)	3.74 (19)		8.83 (17)	7.58 (17)	1.48 (17)	1.55 (16)	4.76 (17)
2.30	3.27 (17)	6.56 (19)	1.06 (18)	2.26 (16)	6.44 (17)	8.65 (17)	1.02 (17)	1.98 (16)	6.14 (17)
2.00	2.43 (17)	8.59 (19)	7.64 (17)		4.13 (17)	6.19 (17)	7.06 (16)	1.36 (16)	6.00 (17)
1.70	1.91 (17)	1.12 (20)	5.30 (17)		4.51 (17)	9.59 (17)	6.19 (16)	1.62 (16)	6.62 (17)
1.00	1.21 (17)	.43 (20)	6.20 (17)	3.07 (15)	1.06 (18)	8.21 (17)	1.36 (17)	1.58 (16)	6.44 (17)
0.70	4.84 (16)	2.00 (20)	1.00 (17)		6.99 (17)	8.16 (17)	1.02 (17)	1.40 (16)	7.25 (17)
0.30	8.12 (15)	9.29 (19)	5.07 (16)		1.71 (17)	1.73 (17)	3.36 (16)	2.80 (15)	3.82 (16)
TOTAL	2.29 (18)	8.96 (20)	7.45 (18)	3.92 (17)	9.56 (18)	1.88 (19)	1.34 (18)	2.93 (17)	1.03 (19)

NOTE: 1) Numbers in Parentheses Refer to Powers of 10.

2) Refer to Figure 2-14 for Region Location.

696 9/14/70

TABLE 2-55
(Continued)

PRESSURE VESSEL AND REACTOR ASSEMBLY SOURCE STRENGTHS

Graphite Core

Energy (MeV)	GAMMA SOURCE STRENGTH (MEV/SEC)								
	REGION 10 Lateral Support Forward	REGION 11 Support Plate	REGION 12 Forward Reflector Hardware	REGION 13 Support Plate Plenum	REGION 14 Instrument Ring	REGION 15 Central Shield Plate - Aft	REGION 16 BATH Central Shield	REGION 17 BATH Peripheral Shield - 1	REGION 18 BATH Peripheral Shield - 2
8.50	2.37 (17)	1.72 (18)	3.95 (17)	2.03 (17)	1.75 (16)	1.35 (16)	4.99 (15)	6.84 (14)	5.08 (14)
7.15	6.11 (16)	5.55 (17)	1.23 (17)	6.45 (16)	5.93 (15)	5.35 (15)	2.04 (15)	2.77 (14)	2.08 (14)
5.50	8.25 (16)	5.32 (17)	1.34 (17)	5.80 (16)	5.85 (15)	6.91 (15)	8.29 (16)	1.05 (16)	8.64 (15)
5.50	7.12 (16)	5.76 (17)	1.32 (17)	6.49 (16)	6.27 (15)	6.86 (15)	9.28 (15)	1.07 (15)	8.57 (14)
4.50	6.85 (16)	5.58 (17)	1.30 (17)	5.20 (16)	7.43 (15)	1.40 (16)	2.23 (15)	2.96 (15)	2.56 (15)
3.50	6.59 (16)	5.75 (17)	1.01 (17)	4.40 (16)	8.43 (15)	2.09 (16)	3.42 (16)	4.20 (15)	3.54 (15)
2.80	3.35 (16)	3.24 (17)	4.97 (16)	1.83 (16)	5.67 (15)	1.66 (16)	2.04 (16)	2.50 (15)	2.10 (15)
2.20	3.32 (16)	3.69 (17)	9.81 (16)	2.76 (16)	5.25 (15)	1.17 (16)	2.09 (16)	2.67 (15)	2.13 (15)
2.00	3.06 (16)	2.62 (17)	5.13 (16)	2.35 (16)	3.52 (15)	6.71 (15)	2.31 (15)	1.07 (15)	8.60 (14)
1.50	5.78 (16)	3.95 (17)	7.97 (16)	4.11 (16)	4.30 (15)	5.82 (15)	3.14 (16)	0.93 (15)	3.26 (15)
1.00	4.73 (16)	4.10 (17)	5.62 (16)	2.39 (16)	6.12 (15)	2.11 (15)	1.03 (17)	1.14 (16)	1.07 (16)
0.70	4.69 (16)	4.22 (17)	6.55 (16)	3.67 (16)	5.21 (15)	1.22 (16)	7.31 (17)	9.49 (16)	7.70 (16)
0.50	7.07 (15)	7.05 (16)	1.10 (16)	5.31 (15)	1.06 (15)	2.79 (15)	2.31 (15)	3.22 (14)	2.33 (14)
TOTAL	3.39 (17)	6.76 (18)	1.42 (18)	6.63 (17)	8.25 (16)	1.44 (17)	1.09 (19)	1.37 (17)	1.12 (17)

- NOTE: 1) Numbers in Parentheses Refer to Powers of 10.
2) Refer to Figure 2-14 for Region Location.

696 9/14/70

TABLE 2-55
(Continued)

Graphite Core

PRESSURE VESSEL AND REACTOR ASSEMBLY SOURCE STRENGTHS

GAMMA SOURCE STRENGTH (MEV/SEC)									
Energy (Mev)	REGION 19 BATH Peripheral Shield - 3	REGION 20 BATH Peripheral Shield - 4	REGION 21 Lead Central Shield	REGION 22 Lead Peripheral Shield - 1	REGION 23 Lead Peripheral Shield - 2	REGION 24 Lead Peripheral Shield - 3	REGION 25 Lead Peripheral Shield - 4	REGION 26 Peripheral Shield Plate	REGION 27 Shield Plenum
8.50	2.79 (15)	3.60 (14)	4.27 (14)	4.40 (13)	5.08 (13)	1.05 (14)	1.63 (13)	1.10 (14)	4.32 (13)
7.25	1.01 (15)	1.47 (14)	1.63 (13)	2.56 (11)		1.85 (13)	2.99 (11)	3.33 (13)	1.80 (13)
6.50	7.91 (15)	5.12 (15)	2.17 (11)	3.46 (11)		2.15 (13)	4.36 (11)	4.05 (15)	2.23 (10)
5.50	1.62 (15)	5.54 (14)	2.26 (11)	3.65 (11)		2.03 (13)	5.00 (11)	4.25 (13)	2.27 (13)
4.50	2.83 (15)	1.51 (15)	4.73 (11)	7.65 (11)		1.85 (13)	1.10 (12)	6.43 (10)	4.20 (13)
3.50	4.14 (15)	2.52 (15)	9.72 (11)	1.67 (12)		2.24 (13)	3.27 (12)	9.74 (10)	6.52 (12)
2.90	2.17 (15)	1.63 (15)	2.29 (15)	2.60 (14)	3.22 (14)	2.38 (14)	2.96 (14)	7.51 (13)	5.25 (13)
2.50	2.18 (15)	1.62 (15)	6.50 (12)	5.79 (12)	6.05 (11)	8.74 (12)	2.46 (12)	5.67 (10)	2.95 (14)
2.00	1.32 (15)	5.42 (14)	2.09 (11)	3.33 (11)		1.12 (13)	4.15 (11)	3.05 (10)	2.12 (13)
1.50	3.69 (15)	1.95 (15)	1.94 (11)	3.12 (11)		2.14 (13)	4.06 (11)	3.53 (10)	1.39 (13)
1.00	1.01 (16)	6.59 (15)	6.92 (13)	1.28 (13)	1.25 (13)	2.57 (13)	1.62 (13)	9.72 (10)	6.06 (13)
0.70	6.35 (16)	4.52 (16)	3.62 (14)	4.50 (13)	5.13 (13)	5.59 (13)	4.82 (13)	5.65 (10)	3.39 (10)
0.30	2.83 (14)	1.60 (14)	2.44 (14)	2.99 (13)	3.46 (13)	2.68 (13)	3.14 (13)	9.90 (10)	6.70 (12)
TOTAL	1.04 (17)	6.99 (16)	3.42 (15)	4.22 (14)	4.72 (14)	5.94 (14)	4.17 (14)	4.58 (14)	7.07 (14)

- NOTE: 1) Numbers in Parentheses Refer to Powers of 10.
2) Refer to Figure 2-14 for Region Location.

696 9/14/70

TABLE 2-55
(Continued)

PRESSURE VESSEL AND REACTOR ASSEMBLY SOURCE STRENGTHS

Graphite Core

Energy (MeV)	GAMMA SOURCE STRENGTH (MEV/SIC)								
	REGION 28 Flow Baffle 1	REGION 29 Flow Baffle 2	REGION 30 Control Dome Plenum	REGION 31 Peripheral Dome Plenum-1	REGION 32 Peripheral Dome Plenum-2	REGION 33 Peripheral Dome Plenum-3	REGION 34 Pressure Vessel Dome	REGION 35 Pressure Vessel Side - 2	REGION 36 Unfueled Taps
9.50	2.67 (14)	4.59 (15)		2.23 (14)	2.93 (14)	7.17 (13)	1.40 (14)	1.21 (15)	1.81 (15)
7.25	1.67 (14)	1.23 (15)		6.01 (13)	7.76 (13)	1.90 (13)	5.58 (13)	4.75 (14)	7.13 (15)
6.25	1.36 (14)	1.79 (15)		7.27 (13)	9.38 (13)	2.30 (13)	7.29 (13)	6.47 (14)	4.55 (16)
5.50	1.53 (14)	1.91 (15)		6.60 (13)	8.63 (13)	2.13 (13)	7.35 (13)	6.64 (14)	6.59 (16)
4.75	2.74 (14)	2.74 (15)		6.51 (13)	8.44 (13)	2.09 (13)	1.49 (14)	1.54 (15)	9.48 (16)
3.50	4.02 (14)	4.46 (15)		5.96 (13)	7.99 (13)	2.17 (13)	2.52 (14)	2.67 (15)	1.19 (17)
2.80	3.27 (14)	3.26 (15)		3.27 (15)	4.24 (13)	1.16 (13)	2.01 (14)	2.09 (15)	7.00 (16)
2.50	2.13 (14)	2.49 (15)	1.32 (14)	3.45 (14)	4.70 (14)	1.14 (14)	1.52 (14)	1.61 (15)	7.06 (16)
2.03	1.55 (14)	1.35 (15)		2.74 (15)	3.64 (13)	9.68 (12)	7.36 (13)	6.20 (14)	1.46 (17)
1.50	1.15 (14)	1.61 (15)		4.31 (13)	5.77 (13)	1.59 (13)	6.78 (13)	6.05 (14)	3.61 (16)
1.00	3.37 (14)	4.94 (15)		3.66 (13)	5.03 (13)	1.52 (13)	2.11 (14)	2.35 (15)	3.10 (16)
0.70	1.95 (14)	3.04 (15)		3.79 (13)	5.10 (13)	1.46 (13)	1.19 (14)	1.53 (15)	5.37 (15)
0.20	4.20 (13)	5.34 (14)		7.19 (12)	9.37 (12)	2.53 (12)	3.15 (13)	4.16 (14)	9.23 (14)
TOTAL	2.68 (15)	3.44 (16)	1.32 (14)	1.10 (15)	1.43 (15)	3.61 (14)	1.58 (15)	1.63 (16)	6.94 (17)

NOTE: 1) Numbers in Parentheses Refer to Powers of 10.
2) Refer to Figure 2-14 for Region Location.

696 9/14/70

TABLE 2-56

NEUTRON AND PHOTON RADIATION ENVIRONMENT (GRAPHITE CORE)

DATA POINT LOCATION -- TANGENT PLANE (MAX)

$$\begin{bmatrix} R = 0 \text{ cm.} \\ Z = 254.508 \text{ cm.} \end{bmatrix}$$

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncert. & Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	4.40 (6)	5.22 (6)	7.31 (6)	8.79 (6)	1.8 (7)	2.05
Fast Neutron Flux ($E > 1.0$ Mev) (N/cm ² -sec)	6.23 (11)	2.71 (11)	8.40 (11)	1.26 (12)	2.0 (12)	1.59
Int. Neutron Flux ($0.4 \text{ ev} < E < 1.0$ Mev) (N/cm ² -sec)	1.76 (12)	7.65 (11)	2.37 (12)	3.56 (12)	3.0 (12)	0.84
Thermal Neutron Flux ($E < 0.4$ ev) (N/cm ² -sec)	2.90 (11)	1.26 (11)	3.90 (11)	5.85 (11)	6.0 (11)	1.03

NOTE: 1) Numbers in parentheses refer to powers of ten.
 2) Data location coordinates are referenced to CRAM model 0.0.

140 9/25/70
 243 10/5/70
 240 10/5/70

TABLE 2-57

NEUTRON AND PHOTON RADIATION ENVIRONMENT (GRAPHITE CORE)

DATA POINT LOCATION -- TPA

$$\begin{bmatrix} R = 91.44 \text{ cm.} \\ Z = 366.27 \text{ cm.} \end{bmatrix}$$

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncert. & Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	1.87 (6)	1.36 (6)	1.90 (6)	1.90 (6)	7.0 (6)	3.68
Fast Neutron Flux ($E > 1.0$ Mev) (N/cm ² -sec)	1.19 (11)	4.41 (10)	1.37 (11)	1.37 (11)	1.0 (12)	7.30
Int. Neutron Flux ($0.4 \text{ ev} < E < 1.0$ Mev) (N/cm ² -sec)	7.09 (11)	2.63 (11)	8.15 (11)	8.15 (11)	5.0 (12)	6.15
Thermal Neutron Flux ($E < 0.4$ ev) (N/cm ² -sec)	1.35 (11)	4.98 (10)	1.54 (11)	1.54 (11)	1.0 (12)	6.5

112 10/2/70 178 9/25/70

180 9/25/70 181 9/25/70

718 9/24/70 208 9/25/70

- NOTE: 1) Numbers in parentheses refer to powers of ten.
 2) Data location coordinates are referenced to CRAM model 0.0.

TABLE 2-58

NEUTRON AND PHOTON RADIATION ENVIRONMENT (GRAPHITE CORE)

DATA POINT LOCATION -- LTS-1

$$\begin{bmatrix} R = 49.53 \text{ cm.} \\ Z = 260.86 \text{ cm.} \end{bmatrix}$$

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncert. & Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	2.20 (6)	2.84 (6)	3.98 (6)	5.97 (6)	2.1 (7)	3.52
Fast Neutron Flux ($E > 1.0$ Mev) (N/cm ² -sec)	2.62 (11)	1.38 (11)	4.28 (11)	8.56 (11)	2.8 (14)	326.0
Int. Neutron Flux ($0.4 \text{ ev} < E < 1.0$ Mev) (N/cm ² -sec)	7.32 (11)	3.85 (11)	1.19 (12)	2.38 (12)	N/A	N/A
Thermal Neutron Flux ($E < 0.4$ ev) (N/cm ² -sec)	1.50 (11)	7.91 (10)	2.45 (11)	4.90 (11)	N/A	N/A

112 10/2/70 178 9/25/70
180 9/25/70 181 9/25/70

- NOTE: 1) Numbers in parentheses refer to powers of ten.
2) Data location coordinates are referenced to CRAM model 0.0.

TABLE 2-59

NEUTRON AND PHOTON RADIATION ENVIRONMENT (GRAPHITE CORE)

DATA POINT LOCATION -- LTS-2

$$\begin{bmatrix} R = 49.53 \text{ cm.} \\ Z = 265.94 \text{ cm.} \end{bmatrix}$$

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncert. & Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	2.20 (6)	2.71 (6)	3.79 (6)	5.69 (6)	1.8 (7)	3.17
Fast Neutron Flux ($E > 1.0$ Mev) (N/cm ² -sec)	2.66 (11)	1.34 (11)	4.15 (11)	8.30 (11)	2.8 (14)	336.0
Int. Neutron Flux ($0.4 \text{ ev} < E < 1.0 \text{ Mev}$) (N/cm ² -sec)	6.65 (11)	3.35 (11)	1.04 (12)	2.08 (12)	N/A	N/A
Thermal Neutron Flux ($E < 0.4 \text{ ev}$) (N/cm ² -sec)	1.31 (11)	6.59 (10)	2.04 (11)	4.08 (11)	N/A	N/A

112 10/2/70 178 9/25/70

180 9/25/70 181 9/25/70

- NOTE: 1) Numbers in parentheses refer to powers of ten.
2) Data location coordinates are referenced to CRAM model 0.0.

TABLE 2-60
NEUTRON AND PHOTON RADIATION ENVIRONMENT (GRAPHITE CORE)

DATA POINT LOCATION -- LTS-3

$\left[\begin{array}{l} R = 49.53 \text{ cm.} \\ Z = 288.80 \text{ cm.} \end{array} \right]$

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncert. & Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	1.50 (6)	2.59 (6)	3.63 (6)	5.45 (6)	1.8 (7)	3.31
Fast Neutron Flux ($E > 1.0$ Mev) (N/cm ² -sec)	1.79 (11)	1.16 (11)	3.60 (11)	7.20 (11)	2.8 (14)	388.0
Int. Neutron Flux ($0.4 \text{ ev} < E < 1.0$ Mev) (N/cm ² -sec)	4.38 (11)	2.83 (11)	8.77 (11)	1.75 (12)	N/A	N/A
Thermal Neutron Flux ($E < 0.4$ ev) (N/cm ² -sec)	8.53 (10)	5.52 (10)	1.71 (11)	3.42 (11)	N/A	N/A

112 10/2/70 178 9/25/70
180 9/25/70 181 9/25/70

- NOTE: 1) Numbers in parentheses refer to powers of ten.
2) Data location coordinates are referenced to CRAM model 0.0.

TABLE 2-61
RELATIVE CONTRIBUTION FROM VARIOUS SOURCE ZONES TO THE OVERALL PHOTON DOSE RATE
AT SEVERAL LOCATIONS ON A 10-FOOT MERIDIAN RING
(R-1 REFERENCE DESIGN REACTOR WITH GRAPHITE FUEL)

SOURCE ZONE	RELATIVE DOSE RATE (%)		
	0° POLAR ANGLE	90° POLAR ANGLE	180° POLAR ANGLE
CORE	37.4	80.4	89.8
REFLECTOR	2.7	8.4	1.8
AL STRUCTURE + LATERAL SUPPORT + CORE PERIPHERY	2.0	4.1	2.5
CORE PLENUM	15.4	1.7	Negligible
SUPPORT PLATE	24.9	1.5	Negligible
SHIELD (PLATES + PLENA)	6.5	1.0	Negligible
PRESSURE VESSEL	1.1	1.0	0.2
CHESH	Negligible	Negligible	2.0
OTHER	10.0	1.9	3.7
TOTAL	100.0	100.0	100.0
TOTAL DOSE RATE (Rads(C)/hr)	1.09 (6)	8.42 (7)	9.33 (7)

045 10/3/70

TABLE 2-62

DELAYED GAMMA RAY DOSE RATE VERSUS TIME AFTER SHUTDOWN FOLLOWING
A 10-MINUTE RUN AT FULL POWER (1511 MW THERMAL POWER)--GRAPHITE CORE

Time	Polar Angle									
	0.0 DEG.	2.5 DEG.	5.0 DEG.	7.5 DEG.	10.0 DEG.	12.5 DEG.	15.0 DEG.	17.5 DEG.	20.0 DEG.	22.5 DEG.
1.8(3) SEC.	2.03884E+03	2.01227E+03	1.93994E+03	1.83641E+03	1.69855E+03	1.51725E+03	1.44100E+03	1.36593E+03	1.29086E+03	1.21579E+03
2.4(3) SEC.	1.32498E+03	1.30958E+03	1.26222E+03	1.19442E+03	1.10439E+03	9.85462E+02	9.36593E+02	8.87724E+02	8.38855E+02	7.89986E+02
3.0(3) SEC.	9.29512E+02	9.17296E+02	8.83965E+02	8.36258E+02	7.73061E+02	6.89935E+02	6.55535E+02	6.21135E+02	5.86735E+02	5.52335E+02
3.6(3) SEC.	6.91012E+02	6.81923E+02	6.57055E+02	6.21478E+02	5.74426E+02	5.12553E+02	4.87100E+02	4.61647E+02	4.36194E+02	4.10741E+02
4.8(3) SEC.	4.33744E+02	4.28061E+02	4.12390E+02	3.89997E+02	3.60435E+02	3.21600E+02	3.05751E+02	2.89902E+02	2.74053E+02	2.58204E+02
6.0(3) SEC.	3.07059E+02	3.03060E+02	2.91950E+02	2.76093E+02	2.55169E+02	2.27714E+02	2.16556E+02	2.05398E+02	1.94240E+02	1.83082E+02
7.2(3) SEC.	2.33736E+02	2.30705E+02	2.22241E+02	2.10170E+02	1.94244E+02	1.73366E+02	1.64901E+02	1.56436E+02	1.47971E+02	1.39506E+02
1.8(4) SEC.	6.81512E+01	6.72635E+01	6.47966E+01	6.12748E+01	5.66250E+01	5.05307E+01	4.80517E+01	4.55727E+01	4.30937E+01	4.06147E+01
3.6(4) SEC.	2.28194E+01	2.25181E+01	2.16793E+01	2.04780E+01	1.89045E+01	1.68456E+01	1.60258E+01	1.52060E+01	1.43862E+01	1.35664E+01
7.2(4) SEC.	5.08051E+00	5.01329E+00	4.81631E+00	4.53552E+00	4.17479E+00	3.70854E+00	3.53959E+00	3.37064E+00	3.20169E+00	3.03274E+00
8.64(4) SEC.	3.43032E+00	3.38561E+00	3.25075E+00	3.05956E+00	2.81482E+00	2.50023E+00	2.34971E+00	2.19919E+00	2.04867E+00	1.89815E+00
1.728(5) SEC.	1.66747E+00	1.64853E+00	1.58672E+00	1.50065E+00	1.38809E+00	1.24410E+00	1.18994E+00	1.13578E+00	1.08162E+00	1.02746E+00
4.320(5) SEC.	1.70167E+00	1.68297E+00	1.62459E+00	1.54252E+00	1.43308E+00	1.29112E+00	1.23060E+00	1.17008E+00	1.10956E+00	1.04904E+00
6.050(5) SEC.	1.63168E+00	1.61380E+00	1.55832E+00	1.48022E+00	1.37585E+00	1.24022E+00	1.18158E+00	1.12294E+00	1.06430E+00	1.00566E+00
8.640(5) SEC.	1.44145E+00	1.42570E+00	1.37696E+00	1.30830E+00	1.21642E+00	1.09689E+00	1.04477E+00	9.9265E+00	9.4053E+00	8.8841E+00
1.728(6) SEC.	8.51208E-01	8.41934E-01	8.13290E-01	7.72919E-01	7.18823E-01	6.48412E-01	6.17498E-01	5.86584E-01	5.55670E-01	5.24756E-01
2.592(6) SEC.	4.94900E-01	4.89517E-01	4.72853E-01	4.49383E-01	4.17928E-01	3.77001E-01	3.59053E-01	3.41105E-01	3.23157E-01	3.05209E-01
5.184(6) SEC.	9.88475E-02	9.77822E-02	9.44099E-02	8.96915E-02	8.33723E-02	7.51850E-02	7.14760E-02	6.77670E-02	6.40580E-02	6.03490E-02
7.776(6) SEC.	2.05556E-02	2.03418E-02	1.96049E-02	1.85988E-02	1.72549E-02	1.55414E-02	1.48719E-02	1.42024E-02	1.35329E-02	1.28634E-02
1.037(7) SEC.	4.93587E-03	4.88992E-03	4.68701E-03	4.42712E-03	4.08236E-03	3.66253E-03	3.54547E-03	3.42841E-03	3.31135E-03	3.19429E-03
1.296(7) SEC.	1.69257E-03	1.67997E-03	1.59440E-03	1.49396E-03	1.36201E-03	1.21256E-03	1.19896E-03	1.18536E-03	1.17176E-03	1.15816E-03
1.555(7) SEC.	9.15878E-04	9.10128E-04	8.57507E-04	7.98625E-04	7.21632E-04	6.38248E-04	6.41010E-04	6.43772E-04	6.46534E-04	6.49296E-04
2.074(7) SEC.	5.20259E-04	5.16637E-04	4.86729E-04	4.53132E-04	4.09040E-04	3.60980E-04	3.61874E-04	3.62768E-04	3.63662E-04	3.64556E-04
2.592(7) SEC.	3.59928E-04	3.56862E-04	3.37688E-04	3.15416E-04	2.85966E-04	2.52740E-04	2.50331E-04	2.47922E-04	2.45513E-04	2.43104E-04
3.154(7) SEC.	2.58666E-04	2.56054E-04	2.43413E-04	2.28138E-04	2.07776E-04	1.83937E-04	1.79960E-04	1.75983E-04	1.71906E-04	1.67829E-04

Note: 3.6(3) sec = 1 hour
3.6(4) sec = 10 hours
8.64(4) sec = 1 day
2.592(6) sec = 1 month
3.154(7) sec = 1 year

NOTE: Units are Rads-C/hr
Meridian ring is located at 10 feet
from the core center.

176 10/19/70

TABLE 2-62
(Continued)

Time	Polar Angle						
	17.5 DEG.	20.0 DEG.	25.0 DEG.	30.0 DEG.	40.0 DEG.	50.0 DEG.	60.0 DEG.
1.8(3) SEC.	1.33916E+03	1.22839E+03	6.23900E+03	3.11338E+04	1.37145E+05	2.65727E+05	3.83821E+05
2.4(3) SEC.	8.70722E+02	7.97770E+02	4.14790E+03	2.11539E+04	9.47502E+04	1.84826E+05	2.68266E+05
3.0(3) SEC.	6.09646E+02	5.58136E+02	2.95597E+03	1.53449E+04	6.96827E+04	1.36685E+05	1.99191E+05
3.6(3) SEC.	4.53138E+02	4.14626E+02	2.22807E+03	1.17330E+04	5.38763E+04	1.06155E+05	1.55202E+05
4.8(3) SEC.	2.84565E+02	2.60276E+02	1.42400E+03	7.64754E+03	3.56759E+04	7.07471E+04	1.03920E+05
6.0(3) SEC.	2.01606E+02	1.84404E+02	1.01767E+03	5.52765E+03	2.60352E+04	5.18339E+04	7.63601E+04
7.2(3) SEC.	1.53539E+02	1.40445E+02	7.79165E+02	4.26335E+03	2.02061E+04	4.03327E+04	5.95289E+04
1.8(4) SEC.	4.47268E+01	4.09054E+01	2.26401E+02	1.23553E+03	5.84455E+03	1.16597E+04	1.72017E+04
3.6(4) SEC.	1.49261E+01	1.36083E+01	7.99531E+01	4.57895E+02	2.23621E+03	4.51493E+03	6.71548E+03
7.2(4) SEC.	3.30897E+00	2.90934E+00	2.16327E+01	1.44745E+02	7.74993E+02	1.62927E+03	2.48219E+03
8.64(4) SEC.	2.23733E+00	2.01811E+00	1.53918E+01	1.06912E+02	5.88464E+02	1.24072E+03	1.90063E+03
1.728(5) SEC.	1.11371E+00	1.02138E+00	6.48006E+00	4.20028E+01	2.27467E+02	4.78982E+02	7.34518E+02
4.320(5) SEC.	1.14713E+00	1.06633E+00	4.92005E+00	2.39169E+01	1.07899E+02	2.13127E+02	3.13782E+02
6.050(5) SEC.	1.10090E+00	1.02486E+00	4.52908E+00	2.07920E+01	8.92327E+01	1.72591E+02	2.50425E+02
8.640(5) SEC.	9.73163E-01	9.06782E-01	3.90093E+00	1.72320E+01	7.12726E+01	1.35577E+02	1.94382E+02
1.728(6) SEC.	5.75051E-01	5.36258E-01	2.25563E+00	9.63323E+00	3.84763E+01	7.19767E+01	1.01925E+02
2.592(6) SEC.	3.34394E-01	3.11826E-01	1.31620E+00	5.65658E+00	2.27413E+01	4.26679E+01	6.05531E+01
5.184(6) SEC.	6.68205E-02	6.22088E-02	2.81179E-01	1.34180E+00	5.96324E+00	1.16959E+01	1.71348E+01
7.776(6) SEC.	1.39179E-02	1.28749E-02	7.31172E-02	4.48642E-01	2.37707E+00	4.97139E+00	7.59584E+00
1.037(7) SEC.	3.35680E-03	3.04454E-03	2.82391E-02	2.31317E-01	1.39769E+00	3.03908E+00	4.75299E+00
1.296(7) SEC.	1.15888E-03	1.01360E-03	1.62209E-02	1.54849E-01	9.83948E-01	2.16759E+00	3.41536E+00
1.555(7) SEC.	6.28786E-04	5.34888E-04	1.13253E-02	1.13173E-01	7.28349E-01	1.60945E+00	2.54019E+00
2.074(7) SEC.	3.54442E-04	3.00654E-04	6.37708E-03	6.35855E-02	4.08632E-01	9.02547E-01	1.42398E+00
2.592(7) SEC.	2.42483E-04	2.08595E-04	3.74785E-03	3.63138E-02	2.31346E-01	5.09914E-01	8.03519E-01
3.154(7) SEC.	1.72306E-04	1.50458E-04	2.19783E-03	2.03767E-02	1.28052E-01	2.81349E-01	4.42530E-01

176 10/19/70

TABLE 2-62
(Continued)

Time	Polar Angle						
	70.0 DEG.	80.0 DEG.	90.0 DEG.	100.0 DEG.	110.0 DEG.	120.0 DEG.	130.0 DEG.
1.8(3) SEC.	4.65007E+05	5.29602E+05	5.48916E+05	5.18120E+05	4.41991E+05	3.49520E+05	2.80579E+05
2.4(3) SEC.	3.25776E+05	3.71930E+05	3.85723E+05	3.63779E+05	3.09552E+05	2.43833E+05	1.95290E+05
3.0(3) SEC.	2.42362E+05	2.77247E+05	2.87667E+05	2.71116E+05	2.30231E+05	1.80764E+05	1.44498E+05
3.6(3) SEC.	1.89133E+05	2.16701E+05	2.24930E+05	2.11874E+05	1.79629E+05	1.40664E+05	1.12267E+05
4.8(3) SEC.	1.26924E+05	1.45750E+05	1.51372E+05	1.42475E+05	1.20509E+05	9.40082E+04	7.48582E+04
6.0(3) SEC.	9.33939E+04	1.07402E+05	1.11576E+05	1.04969E+05	8.86577E+04	6.89945E+04	5.48603E+04
7.2(3) SEC.	7.28747E+04	8.38822E+04	8.71600E+04	8.19740E+04	6.91707E+04	5.37446E+04	4.26938E+04
1.8(4) SEC.	2.10567E+04	2.42350E+04	2.51818E+04	2.36844E+04	1.99869E+04	1.55310E+04	1.23366E+04
3.6(4) SEC.	8.25413E+03	9.53924E+03	9.92097E+03	9.31830E+03	7.83068E+03	6.04291E+03	4.77967E+03
7.2(4) SEC.	3.08661E+03	3.60845E+03	3.76268E+03	3.52070E+03	2.92392E+03	2.21119E+03	1.72525E+03
8.64(4) SEC.	2.36973E+03	2.77762E+03	2.89804E+03	2.70933E+03	2.24410E+03	1.68903E+03	1.31308E+03
1.728(5) SEC.	9.16667E+02	1.07563E+03	1.12251E+03	1.04897E+03	8.68057E+02	6.51153E+02	5.04593E+02
4.320(5) SEC.	3.84188E+02	4.42564E+02	4.59878E+02	4.32297E+02	3.64778E+02	2.82786E+02	2.27018E+02
6.050(5) SEC.	3.04392E+02	3.48106E+02	3.61108E+02	3.40257E+02	2.89301E+02	2.27020E+02	1.80363E+02
8.640(5) SEC.	2.34823E+02	2.66879E+02	2.76438E+02	2.61011E+02	2.23367E+02	1.77086E+02	1.41575E+02
1.728(4) SEC.	1.22325E+02	1.38086E+02	1.42801E+02	1.35136E+02	1.16460E+02	9.33433E+01	7.51323E+01
2.592(6) SEC.	7.27454E+01	8.22023E+01	8.50298E+01	8.04389E+01	6.92469E+01	5.54120E+01	4.45726E+01
5.184(6) SEC.	2.09121E+01	2.40097E+01	2.49296E+01	2.34607E+01	1.98622E+01	1.54843E+01	1.22744E+01
7.776(6) SEC.	9.45512E+00	1.10665E+01	1.15420E+01	1.07949E+01	8.95585E+00	6.75361E+00	5.24985E+00
1.037(7) SEC.	5.97808E+00	7.06572E+00	7.38584E+00	6.88643E+00	5.65430E+00	4.18923E+00	3.22219E+00
1.296(7) SEC.	4.30938E+00	5.10850E+00	5.34353E+00	4.97762E+00	4.07416E+00	3.00220E+00	2.30198E+00
1.555(7) SEC.	3.20732E+00	3.80443E+00	3.98002E+00	3.70679E+00	3.03194E+00	2.23165E+00	1.71023E+00
2.074(7) SEC.	1.79767E+00	2.13201E+00	2.23033E+00	2.07733E+00	1.69940E+00	1.25122E+00	9.59112E-01
2.592(7) SEC.	1.01394E+00	1.20204E+00	1.25736E+00	1.17125E+00	9.58582E-01	7.06293E-01	5.41517E-01
3.154(7) SEC.	5.58087E-01	6.61258E-01	6.91603E-01	6.44354E-01	5.27669E-01	3.89169E-01	2.98393E-01

176 10/19/70

TABLE 2-62
(Continued)

Time	Polar Angle				
	140.0 DEG.	150.0 DEG.	160.0 DEG.	170.0 DEG.	180.0 DEG.
1.8(3) SEC.	2.94788E+05	4.10309E+05	5.76514E+05	6.75541E+05	7.04559E+05
2.4(3) SEC.	2.06632E+05	2.90156E+05	4.09283E+05	4.79997E+05	5.00771E+05
3.0(3) SEC.	1.53795E+05	2.17529E+05	3.07806E+05	3.61222E+05	3.76946E+05
3.6(3) SEC.	1.20063E+05	1.70804E+05	2.42292E+05	2.84477E+05	2.96915E+05
4.8(3) SEC.	8.06238E+04	1.15657E+05	1.64643E+05	1.93426E+05	2.01933E+05
6.0(3) SEC.	5.93479E+04	8.55764E+04	1.22085E+05	1.43476E+05	1.49807E+05
7.2(3) SEC.	4.63175E+04	6.70100E+04	9.57317E+04	1.12526E+05	1.17502E+05
1.8(4) SEC.	1.33578E+04	1.93060E+04	2.75797E+04	3.24178E+04	3.38539E+04
3.6(4) SEC.	5.22580E+03	7.65412E+03	1.10016E+04	1.29473E+04	1.35288E+04
7.2(4) SEC.	1.94164E+03	2.95280E+03	4.31576E+03	5.09471E+03	5.33128E+03
8.64(4) SEC.	1.48620E+03	2.27854E+03	3.34289E+03	3.94911E+03	4.13389E+03
1.728(5) SEC.	5.67892E+02	8.71126E+02	1.28129E+03	1.51467E+03	1.58611E+03
4.320(5) SEC.	2.36125E+02	3.38249E+02	4.84584E+02	5.70358E+02	5.96082E+02
6.050(5) SEC.	1.87211E+02	2.61146E+02	3.69762E+02	4.34276E+02	4.53405E+02
8.640(5) SEC.	1.44695E+02	1.97336E+02	2.76459E+02	3.24029E+02	3.37972E+02
1.728(6) SEC.	7.56563E+01	1.00741E+02	1.39438E+02	1.63027E+02	1.69839E+02
2.592(6) SEC.	4.51041E+01	6.03722E+01	8.37195E+01	9.79022E+01	1.02000E+02
5.184(6) SEC.	1.30521E+01	1.85957E+01	2.64889E+01	3.11184E+01	3.24880E+01
7.776(6) SEC.	5.94744E+00	9.09411E+00	1.33186E+01	1.57184E+01	1.64441E+01
1.037(7) SEC.	3.78147E+00	5.98625E+00	8.87613E+00	1.04957E+01	1.09897E+01
1.296(7) SEC.	2.73386E+00	4.37369E+00	6.50742E+00	7.69853E+00	8.06251E+00
1.555(7) SEC.	2.03738E+00	3.26723E+00	4.86433E+00	5.75505E+00	6.02728E+00
2.074(7) SEC.	1.14214E+00	1.83074E+00	2.72508E+00	3.22392E+00	3.37637E+00
2.592(7) SEC.	6.42918E-01	1.02865E+00	1.53068E+00	1.81090E+00	1.89660E+00
3.154(7) SEC.	3.52388E-01	5.62203E-01	8.36322E-01	9.89522E-01	1.03645E+00

176 10/19/70

TABLE 2-63

DELAYED GAMMA RAY DOSE RATE VERSUS TIME AFTER SHUTDOWN FOLLOWING
A 30-MINUTE RUN AT FULL POWER (1511 MW THERMAL POWER)--GRAPHITE CORE

	Polar Angle							
Time	0.0 DEG.	2.5 DEG.	5.0 DEG.	7.5 DEG.	10.0 DEG.	12.5 DEG.	15.0 DEG.	
1.8(3) SEC.	4.29575E+03	4.23957E+03	4.08652E+03	3.86747E+03	3.57636E+03	3.19347E+03	3.03341E+03	
2.4(3) SEC.	2.94165E+03	2.90301E+03	2.79764E+03	2.64683E+03	2.44692E+03	2.18398E+03	2.07498E+03	
3.0(3) SEC.	2.16073E+03	2.13233E+03	2.05465E+03	1.94351E+03	1.79645E+03	1.60307E+03	1.52339E+03	
3.6(3) SEC.	1.66155E+03	1.63973E+03	1.57984E+03	1.49419E+03	1.38100E+03	1.23223E+03	1.17125E+03	
4.8(3) SEC.	1.10092E+03	1.08654E+03	1.04674E+03	9.89900E+02	9.14879E+02	8.16381E+02	7.76258E+02	
6.0(3) SEC.	8.08267E+02	7.97756E+02	7.68500E+02	7.26756E+02	6.71679E+02	5.99437E+02	5.70109E+02	
7.2(3) SEC.	6.30296E+02	6.22130E+02	5.99308E+02	5.66763E+02	5.23820E+02	4.67537E+02	4.44723E+02	
1.8(4) SEC.	1.95644E+02	1.93095E+02	1.86015E+02	1.75904E+02	1.62556E+02	1.45060E+02	1.37942E+02	
3.6(4) SEC.	6.64050E+01	6.55280E+01	6.30867E+01	5.95908E+01	5.50114E+01	4.90195E+01	4.66349E+01	
7.2(4) SEC.	1.49472E+01	1.47495E+01	1.41896E+01	1.33431E+01	1.22815E+01	1.09097E+01	1.04135E+01	
8.64(4) SEC.	1.01479E+01	1.00158E+01	9.61675E+00	9.05121E+00	8.32730E+00	7.39692E+00	7.07028E+00	
1.728(5) SEC.	5.00436E+00	4.94758E+00	4.76216E+00	4.50399E+00	4.16630E+00	3.73433E+00	3.57172E+00	
4.320(5) SEC.	5.10833E+00	5.05222E+00	4.87695E+00	4.63062E+00	4.30210E+00	3.87596E+00	3.69426E+00	
6.050(5) SEC.	4.89554E+00	4.84189E+00	4.67543E+00	4.44113E+00	4.12800E+00	3.72109E+00	3.54512E+00	
8.640(5) SEC.	4.32125E+00	4.27401E+00	4.12790E+00	3.92208E+00	3.64462E+00	3.28831E+00	3.13205E+00	
1.728(6) SEC.	2.55500E+00	2.52716E+00	2.44119E+00	2.32001E+00	2.15764E+00	1.94630E+00	1.85351E+00	
2.592(6) SEC.	1.48445E+00	1.46831E+00	1.41833E+00	1.34793E+00	1.25358E+00	1.13082E+00	1.07699E+00	
5.184(6) SEC.	2.96305E-01	2.93112E-01	2.83002E-01	2.68859E-01	2.49917E-01	2.25376E-01	2.14858E-01	
7.776(6) SEC.	6.16670E-02	6.10253E-02	5.88148E-02	5.57964E-02	5.17643E-02	4.66236E-02	4.46150E-02	
1.037(7) SEC.	1.48248E-02	1.46868E-02	1.40774E-02	1.32970E-02	1.22616E-02	1.10007E-02	1.06488E-02	
1.296(7) SEC.	5.07023E-03	5.03253E-03	4.77604E-03	4.47507E-03	4.07968E-03	3.63196E-03	3.49148E-03	
1.555(7) SEC.	2.74255E-03	2.72536E-03	2.56771E-03	2.39133E-03	2.16072E-03	1.91104E-03	1.91947E-03	
2.074(7) SEC.	1.56289E-03	1.55202E-03	1.46216E-03	1.36123E-03	1.22877E-03	1.08440E-03	1.08710E-03	
2.592(7) SEC.	1.08150E-03	1.07227E-03	1.01471E-03	9.47830E-04	8.59383E-04	7.59549E-04	7.52201E-04	
3.154(7) SEC.	7.74194E-04	7.66364E-04	7.28560E-04	6.82864E-04	6.21942E-04	5.50590E-04	5.38624E-04	

Note: 3.6(3) sec = 1 hour
3.6(4) sec = 10 hours
8.64(4) sec = 1 day
2.592(6) sec = 1 month
3.154(7) sec = 1 year

NOTE: Units are Rads-C/hr
Meridian ring is located at 10 feet
from the core center.

473 10/16/70

TABLE 2-63
(Continued)

Time	Polar Angle						
	17.5 DEG.	20.0 DEG.	25.0 DEG.	30.0 DEG.	40.0 DEG.	50.0 DEG.	60.0 DEG.
1.8(3) SEC.	2.81978E+03	2.58454E+03	1.33424E+04	6.76206E+04	3.01488E+05	5.87034E+05	8.50952E+05
2.4(3) SEC.	1.92955E+03	1.76685E+03	9.31711E+03	4.81723E+04	2.18086E+05	4.27261E+05	6.22097E+05
3.0(3) SEC.	1.41708E+03	1.29686E+03	6.94250E+03	3.64319E+04	1.66857E+05	3.28430E+05	4.79823E+05
3.6(3) SEC.	1.08980E+03	9.97012E+02	5.39943E+03	2.86772E+04	1.32597E+05	2.62003E+05	3.83850E+05
4.8(3) SEC.	7.22564E+02	6.60917E+02	3.62973E+03	1.95939E+04	9.18111E+04	1.82401E+05	2.68290E+05
6.0(3) SEC.	5.30785E+02	4.85499E+02	2.68616E+03	1.46401E+04	6.91516E+04	1.37837E+05	2.03230E+05
7.2(3) SEC.	4.14088E+02	3.78784E+02	2.10310E+03	1.15243E+04	5.46915E+04	1.09230E+05	1.61285E+05
1.8(4) SEC.	1.28395E+02	1.17426E+02	6.49583E+02	3.54267E+03	1.67499E+04	3.34092E+04	4.92822E+04
3.6(4) SEC.	4.34356E+01	3.95990E+01	2.32934E+02	1.33538E+03	6.52757E+03	1.31844E+04	1.96160E+04
7.2(4) SEC.	9.73569E+00	8.79445E+00	6.38199E+01	4.27870E+02	2.30548E+03	4.82396E+03	7.35142E+03
8.64(4) SEC.	6.61971E+00	5.97125E+00	4.55737E+01	3.16808E+02	1.74477E+03	3.67949E+03	5.63746E+03
1.728(5) SEC.	3.34288E+00	3.06610E+00	1.94168E+01	1.25719E+02	6.80486E+02	1.43268E+03	2.19681E+03
4.320(5) SEC.	3.44368E+00	3.20116E+00	1.47659E+01	7.17533E+01	3.23618E+02	6.39146E+02	9.40924E+02
6.050(5) SEC.	3.30307E+00	3.07495E+00	1.35878E+01	6.23737E+01	2.67672E+02	5.17711E+02	7.51175E+02
8.640(5) SEC.	2.91736E+00	2.71836E+00	1.16925E+01	5.16357E+01	2.13503E+02	4.06074E+02	5.82141E+02
1.728(6) SEC.	1.72610E+00	1.60966E+00	6.76965E+00	2.89055E+01	1.15428E+02	2.15906E+02	3.05719E+02
2.592(6) SEC.	1.00302E+00	9.35331E-01	3.94788E+00	1.69662E+01	6.82085E+01	1.27974E+02	1.81617E+02
5.184(6) SEC.	2.00305E-01	1.86483E-01	8.42896E-01	4.02262E+00	1.78790E+01	3.50679E+01	5.13772E+01
7.776(6) SEC.	4.17526E-02	3.86230E-02	2.19365E-01	1.34623E+00	7.13221E+00	1.49164E+01	2.27910E+01
1.037(7) SEC.	1.00820E-02	9.14444E-03	8.47627E-02	6.94139E-01	4.19379E+00	9.11852E+00	1.42607E+01
1.296(7) SEC.	3.47168E-03	3.03615E-03	4.86545E-02	4.64598E-01	2.95243E+00	6.50420E+00	1.02485E+01
1.555(7) SEC.	1.88302E-03	1.60167E-03	3.39487E-02	3.39296E-01	2.18365E+00	4.82526E+00	7.61566E+00
2.074(7) SEC.	1.06479E-03	9.03191E-04	1.91609E-02	1.91058E-01	1.22784E+00	2.71194E+00	4.27872E+00
2.592(7) SEC.	7.28516E-04	6.26820E-04	1.12363E-02	1.08822E-01	6.93173E-01	1.52777E+00	2.40739E+00
3.154(7) SEC.	5.15662E-04	4.50338E-04	6.56451E-03	6.08314E-02	3.82225E-01	8.39785E-01	1.32087E+00

TABLE 2-63
(Continued)

Time	Polar Angle						
	70.0 DEG.	80.0 DEG.	90.0 DEG.	100.0 DEG.	110.0 DEG.	120.0 DEG.	130.0 DEG.
1.8(3) SEC.	1.03273E+06	1.17829E+06	1.22179E+06	1.15254E+06	9.81384E+05	7.73832E+05	6.20149E+05
2.4(3) SEC.	7.56606E+05	8.65137E+05	8.97556E+05	8.46042E+05	7.18778E+05	5.64742E+05	4.51634E+05
3.0(3) SEC.	5.84521E+05	6.69481E+05	6.94846E+05	6.54593E+05	5.55174E+05	4.35001E+05	3.47306E+05
3.6(3) SEC.	4.68235E+05	5.37027E+05	5.57553E+05	5.25009E+05	4.44447E+05	3.47605E+05	2.77150E+05
4.8(3) SEC.	3.27894E+05	3.76793E+05	3.91369E+05	3.68286E+05	3.11296E+05	2.42566E+05	1.93023E+05
6.0(3) SEC.	2.48667E+05	2.86083E+05	2.97229E+05	2.79590E+05	2.36044E+05	1.83562E+05	1.45894E+05
7.2(3) SEC.	1.97485E+05	2.27362E+05	2.36257E+05	2.22185E+05	1.87442E+05	1.45587E+05	1.15626E+05
1.8(4) SEC.	6.03233E+04	6.94261E+04	7.21356E+04	6.78474E+04	5.72590E+04	4.44478E+04	3.53473E+04
3.6(4) SEC.	2.41139E+04	2.78724E+04	2.89887E+04	2.72264E+04	2.28764E+04	1.76491E+04	1.39568E+04
7.2(4) SEC.	9.14275E+03	1.06900E+04	1.11472E+04	1.04299E+04	8.66072E+03	6.54799E+03	5.10806E+03
8.64(4) SEC.	7.02943E+03	8.24007E+03	8.59745E+03	8.03740E+03	6.65670E+03	5.00944E+03	3.89392E+03
1.728(5) SEC.	2.74145E+03	3.21672E+03	3.35687E+03	3.13699E+03	2.59609E+03	1.94935E+03	1.50929E+03
4.320(5) SEC.	1.15200E+03	1.32698E+03	1.37889E+03	1.29621E+03	1.09380E+03	8.48006E+02	6.68811E+02
6.050(5) SEC.	9.13046E+02	1.04416E+03	1.08316E+03	1.02062E+03	8.67779E+02	6.80974E+02	5.41025E+02
8.640(5) SEC.	7.03217E+02	7.99169E+02	8.27783E+02	7.81602E+02	6.68916E+02	5.30366E+02	4.24033E+02
1.728(6) SEC.	3.66891E+02	4.14149E+02	4.28286E+02	4.05301E+02	3.49303E+02	2.79986E+02	2.25370E+02
2.592(6) SEC.	2.18184E+02	2.46548E+02	2.55028E+02	2.41259E+02	2.07691E+02	1.66197E+02	1.33687E+02
5.184(6) SEC.	6.27038E+01	7.19929E+01	7.47516E+01	7.03467E+01	5.95555E+01	4.64278E+01	3.68027E+01
7.776(6) SEC.	2.83698E+01	3.32048E+01	3.46317E+01	3.23900E+01	2.68718E+01	2.02638E+01	1.57518E+01
1.037(7) SEC.	1.79363E+01	2.11995E+01	2.21599E+01	2.06616E+01	1.69649E+01	1.25693E+01	9.66793E+00
1.296(7) SEC.	1.29312E+01	1.53292E+01	1.60345E+01	1.49365E+01	1.22254E+01	9.00864E+00	6.90747E+00
1.555(7) SEC.	9.61568E+00	1.14058E+01	1.19322E+01	1.11131E+01	9.08990E+00	6.69067E+00	5.12748E+00
2.074(7) SEC.	5.40157E+00	6.40617E+00	6.70160E+00	6.24188E+00	5.10629E+00	3.75962E+00	2.88192E+00
2.592(7) SEC.	3.03779E+00	3.60128E+00	3.76700E+00	3.50906E+00	2.87193E+00	2.11612E+00	1.62246E+00
3.154(7) SEC.	1.66578E+00	1.97372E+00	2.06430E+00	1.92327E+00	1.57499E+00	1.16159E+00	8.90632E+00

473 10/16/70

TABLE 2-63
(Continued)

Time	Polar Angle				
	140.0 DEG.	150.0 DEG.	160.0 DEG.	170.0 DEG.	180.0 DEG.
1.8(3) SEC.	6.54944E+05	9.17552E+05	1.29294E+06	1.51598E+06	1.58146E+06
2.4(3) SEC.	4.80078E+05	6.77966E+05	9.58675E+05	1.12487E+06	1.17378E+06
3.0(3) SEC.	3.71029E+05	5.27154E+05	7.47374E+05	8.77397E+05	9.15719E+05
3.6(3) SEC.	2.97321E+05	4.24541E+05	6.03176E+05	7.08389E+05	7.39442E+05
4.8(3) SEC.	2.08319E+05	2.99560E+05	4.26870E+05	5.01575E+05	5.23668E+05
6.0(3) SEC.	1.58038E+05	2.28228E+05	3.25800E+05	3.82915E+05	3.99828E+05
7.2(3) SEC.	1.25514E+05	1.81720E+05	2.59692E+05	3.05261E+05	3.18769E+05
1.8(4) SEC.	3.82619E+04	5.52852E+04	7.89717E+04	9.28244E+04	9.69365E+04
3.6(4) SEC.	1.52634E+04	2.23658E+04	3.21547E+04	3.78432E+04	3.95440E+04
7.2(4) SEC.	5.75066E+03	8.74925E+03	1.27902E+04	1.50993E+04	1.58007E+04
8.64(4) SEC.	4.40780E+03	6.75926E+03	9.91788E+03	1.17167E+04	1.22651E+04
1.728(5) SEC.	1.69850E+03	2.60509E+03	3.83143E+03	4.52921E+03	4.74280E+03
4.320(5) SEC.	7.08060E+02	1.01416E+03	1.45281E+03	1.70994E+03	1.78705E+03
6.050(5) SEC.	5.61558E+02	7.83313E+02	1.10909E+03	1.30260E+03	1.35997E+03
8.640(5) SEC.	4.33312E+02	5.90827E+02	8.27642E+02	9.70035E+02	1.01177E+03
1.728(6) SEC.	2.26918E+02	3.02110E+02	4.18126E+02	4.88855E+02	5.09280E+02
2.592(6) SEC.	1.35281E+02	1.81074E+02	2.51098E+02	2.93635E+02	3.05926E+02
5.184(6) SEC.	3.91362E+01	5.57614E+01	7.94318E+01	9.33146E+01	9.74220E+01
7.776(6) SEC.	1.78446E+01	2.72860E+01	3.99614E+01	4.71619E+01	4.93395E+01
1.037(7) SEC.	1.13458E+01	1.79605E+01	2.66307E+01	3.14899E+01	3.29719E+01
1.296(7) SEC.	8.20357E+00	1.31245E+01	1.95275E+01	2.31018E+01	2.41940E+01
1.555(7) SEC.	6.10856E+00	9.79592E+00	1.45842E+01	1.72547E+01	1.80709E+01
2.074(7) SEC.	3.43192E+00	5.50101E+00	8.18832E+00	9.68722E+00	1.01453E+01
2.592(7) SEC.	1.92622E+00	3.08180E+00	4.58577E+00	5.42526E+00	5.68200E+00
3.154(7) SEC.	1.05171E+00	1.67786E+00	2.49597E+00	2.95321E+00	3.09327E+00

473 10/16/70

TABLE 2-64

DELAYED GAMMA RAY DOSE RATE VERSUS TIME AFTER SHUTDOWN FOLLOWING
A 60-MINUTE RUN AT FULL POWER (1511 MW THERMAL POWER)--GRAPHITE CORE

	Polar Angle							
Time	0.0 DEG.	2.5 DEG.	5.0 DEG.	7.5 DEG.	10.0 DEG.	12.5 DEG.	15.0 DEG.	
1.8(3) SEC.	5.95545E+03	5.87748E+03	5.66461E+03	5.36001E+03	4.95584E+03	4.42436E+03	4.20341E+03	
2.4(3) SEC.	4.27487E+03	4.21878E+03	4.06527E+03	3.84567E+03	3.55493E+03	3.17265E+03	3.01490E+03	
3.0(3) SEC.	3.25938E+03	3.21662E+03	3.09922E+03	2.93134E+03	2.70941E+03	2.41772E+03	2.29801E+03	
3.6(3) SEC.	2.59343E+03	2.55947E+03	2.46585E+03	2.33206E+03	2.15537E+03	1.92326E+03	1.82840E+03	
4.8(3) SEC.	1.80910E+03	1.78553E+03	1.72009E+03	1.62668E+03	1.50341E+03	1.34165E+03	1.27587E+03	
6.0(3) SEC.	1.37383E+03	1.35600E+03	1.30627E+03	1.23534E+03	1.14173E+03	1.01902E+03	9.69238E+02	
7.2(3) SEC.	1.10089E+03	1.08664E+03	1.04677E+03	9.89931E+02	9.14923E+02	8.16639E+02	7.76814E+02	
1.8(4) SEC.	4.06790E+02	4.01435E+02	3.86723E+02	3.65751E+02	3.37950E+02	3.01510E+02	2.86542E+02	
3.6(4) SEC.	1.26933E+02	1.25257E+02	1.20585E+02	1.13895E+02	1.05135E+02	9.36774E+01	8.91265E+01	
7.2(4) SEC.	2.90986E+01	2.87140E+01	2.75836E+01	2.59732E+01	2.39054E+01	2.12343E+01	2.02703E+01	
8.64(4) SEC.	1.99309E+01	1.96718E+01	1.88877E+01	1.77768E+01	1.63550E+01	1.45282E+01	1.38877E+01	
1.728(5) SEC.	9.98789E+00	9.87460E+00	9.50475E+00	8.98916E+00	8.31602E+00	7.45416E+00	7.12948E+00	
4.320(5) SEC.	1.02105E+01	1.00983E+01	9.74801E+00	9.25566E+00	8.59903E+00	7.74726E+00	7.38403E+00	
6.050(5) SEC.	9.78711E+00	9.67988E+00	9.34710E+00	8.87872E+00	8.25270E+00	7.43921E+00	7.08742E+00	
8.640(5) SEC.	8.65122E+00	8.55666E+00	8.26415E+00	7.85211E+00	7.30065E+00	6.58333E+00	6.27049E+00	
1.728(6) SEC.	5.10136E+00	5.04578E+00	4.87412E+00	4.63216E+00	4.30796E+00	3.88596E+00	3.70069E+00	
2.592(6) SEC.	2.96802E+00	2.93573E+00	2.83579E+00	2.69503E+00	2.50640E+00	2.26094E+00	2.15331E+00	
5.184(6) SEC.	5.93278E-01	5.86884E-01	5.66643E-01	5.38322E-01	5.00393E-01	4.51252E-01	4.30193E-01	
7.776(6) SEC.	1.23295E-01	1.22012E-01	1.17592E-01	1.11557E-01	1.03496E-01	9.32179E-02	8.92023E-02	
1.037(7) SEC.	2.96542E-02	2.93781E-02	2.81594E-02	2.65982E-02	2.45272E-02	2.20050E-02	2.13012E-02	
1.296(7) SEC.	1.01482E-02	1.00727E-02	9.55949E-03	8.95719E-03	8.16592E-03	7.26981E-03	7.18856E-03	
1.555(7) SEC.	5.49376E-03	5.45926E-03	5.14364E-03	4.79046E-03	4.32863E-03	3.82846E-03	3.84499E-03	
2.074(7) SEC.	3.12560E-03	3.10385E-03	2.92415E-03	2.72229E-03	2.45739E-03	2.16867E-03	2.17408E-03	
2.592(7) SEC.	2.15952E-03	2.14113E-03	2.02609E-03	1.89246E-03	1.71576E-03	1.51641E-03	1.50196E-03	
3.154(7) SEC.	1.54809E-03	1.53245E-03	1.45682E-03	1.36543E-03	1.24458E-03	1.10091E-03	1.07704E-03	

Note: 3.6(3) sec = 1 hour
3.6(4) sec = 10 hours
8.64(4) sec = 1 day
2.592(6) sec = 1 month
3.154(7) sec = 1 year

NOTE: Units are Rads-C/hr
Meridian ring is located at 10 feet
from the core center.

476 10/16/70

TABLE 2-64
(Continued)

Time	Polar Angle						
	17.5 DEG.	20.0 DEG.	25.0 DEG.	30.0 DEG.	40.0 DEG.	50.0 DEG.	60.0 DEG.
1.8(3) SEC.	3.90844E+03	3.58051E+03	1.87374E+04	9.62835E+04	4.34058E+05	8.49015E+05	1.23480E+06
2.4(3) SEC.	2.80431E+03	2.56703E+03	1.36844E+04	7.15749E+04	3.27086E+05	6.43280E+05	9.39268E+05
3.0(3) SEC.	2.13812E+03	1.95636E+03	1.05661E+04	5.59997E+04	2.58565E+05	5.10640E+05	7.47847E+05
3.6(3) SEC.	1.70158E+03	1.55656E+03	8.48836E+03	4.54555E+04	2.11614E+05	4.19310E+05	6.15575E+05
4.8(3) SEC.	1.18775E+03	1.08643E+03	5.98942E+03	3.24987E+04	1.52947E+05	3.04412E+05	4.48351E+05
6.0(3) SEC.	9.02431E+02	8.25487E+02	4.57514E+03	2.50088E+04	1.18442E+05	2.36351E+05	3.48773E+05
7.2(3) SEC.	7.23311E+02	6.61649E+02	3.67757E+03	2.01855E+04	9.59360E+04	1.91723E+05	2.83220E+05
1.8(4) SEC.	2.66548E+02	2.43770E+02	1.33273E+03	7.18014E+03	3.36221E+04	6.68151E+04	9.82856E+04
3.6(4) SEC.	8.30184E+01	7.56704E+01	4.47300E+02	2.57520E+03	1.26247E+04	2.55276E+04	3.80095E+04
7.2(4) SEC.	1.89530E+01	1.71176E+01	1.24777E+02	8.39043E+02	4.52929E+03	9.48301E+03	1.44576E+04
8.64(4) SEC.	1.30036E+01	1.17297E+01	8.97025E+01	6.24577E+02	3.44324E+03	7.26402E+03	1.11322E+04
1.728(5) SEC.	6.67256E+00	6.12073E+00	3.86942E+01	2.50283E+02	1.35412E+03	2.85061E+03	4.37071E+03
4.320(5) SEC.	6.88312E+00	6.39845E+00	2.95026E+01	1.43291E+02	6.45992E+02	1.27563E+03	1.87772E+03
6.050(5) SEC.	6.60348E+00	6.14744E+00	2.71612E+01	1.24659E+02	5.34879E+02	1.03445E+03	1.50086E+03
8.640(5) SEC.	5.84069E+00	5.44233E+00	2.34075E+01	1.03366E+02	4.27384E+02	8.12855E+02	1.16529E+03
1.728(6) SEC.	3.44629E+00	3.21377E+00	1.35191E+01	5.77424E+01	2.30645E+02	4.31469E+02	6.11004E+02
2.592(6) SEC.	2.00543E+00	1.87008E+00	7.89398E+00	3.39287E+01	1.36416E+02	2.55959E+02	3.63260E+02
5.184(6) SEC.	4.01052E-01	3.73368E-01	1.68808E+00	8.05858E+00	3.58248E+01	7.02725E+01	1.02960E+02
7.776(6) SEC.	8.34796E-02	7.72224E-02	4.38616E-01	2.69189E+00	1.42618E+01	2.98275E+01	4.55743E+01
1.037(7) SEC.	2.01672E-02	1.82919E-02	1.69533E-01	1.38829E+00	8.38754E+00	1.82369E+01	2.85212E+01
1.296(7) SEC.	6.94857E-03	6.07714E-03	9.73293E-02	9.29279E-01	5.90516E+00	1.30090E+01	2.04977E+01
1.555(7) SEC.	3.77162E-03	3.20843E-03	6.79239E-02	6.78751E-01	4.36824E+00	9.65265E+00	1.52348E+01
2.074(7) SEC.	2.12946E-03	1.80628E-03	3.83210E-02	3.82107E-01	2.45561E+00	5.42369E+00	8.55711E+00
2.592(7) SEC.	1.45487E-03	1.25154E-03	2.24866E-02	2.17874E-01	1.38798E+00	3.05925E+00	4.82069E+00
3.154(7) SEC.	1.03118E-03	9.00486E-04	1.31401E-02	1.21795E-01	7.65330E-01	1.68152E+00	2.64482E+00

476 10/16/70

TABLE 2-64
(Continued)

Time	Polar Angle						
	70.0 DEG.	80.0 DEG.	90.0 DEG.	100.0 DEG.	110.0 DEG.	120.0 DEG.	130.0 DEG.
1.8(3) SEC.	1.50099E+06	1.71537E+06	1.77940E+06	1.67759E+06	1.42605E+06	1.12143E+06	8.97276E+05
2.4(3) SEC.	1.14391E+06	1.30981E+06	1.35934E+06	1.28072E+06	1.08652E+06	8.51711E+05	6.80190E+05
3.0(3) SEC.	9.12098E+05	1.04592E+06	1.08585E+06	1.02253E+06	8.66168E+05	6.77323E+05	5.40124E+05
3.6(3) SEC.	7.51645E+05	8.62937E+05	8.96127E+05	8.43537E+05	7.13685E+05	5.56984E+05	4.43641E+05
4.8(3) SEC.	5.48310E+05	6.30487E+05	6.54974E+05	6.16211E+05	5.20511E+05	4.05138E+05	3.22175E+05
6.0(3) SEC.	4.26926E+05	4.91367E+05	5.10557E+05	4.80194E+05	4.05232E+05	3.14907E+05	2.50178E+05
7.2(3) SEC.	3.46866E+05	3.99432E+05	4.15080E+05	3.90328E+05	3.29217E+05	2.55603E+05	2.02951E+05
1.8(4) SEC.	1.20150E+05	1.38099E+05	1.43439E+05	1.34975E+05	1.14065E+05	8.88420E+04	7.06678E+04
3.6(4) SEC.	4.67426E+04	5.40482E+04	5.62177E+04	5.27936E+04	4.43416E+04	3.41874E+04	2.70242E+04
7.2(4) SEC.	1.79843E+04	2.10319E+04	2.19325E+04	2.05197E+04	1.70356E+04	1.28752E+04	1.00412E+04
8.64(4) SEC.	1.38826E+04	1.62755E+04	1.69818E+04	1.58750E+04	1.31463E+04	9.89097E+03	7.68711E+03
1.728(5) SEC.	5.45417E+03	6.39956E+03	6.67833E+03	6.24096E+03	5.16499E+03	3.87845E+03	3.00294E+03
4.320(5) SEC.	2.29882E+03	2.64788E+03	2.75141E+03	2.58648E+03	2.18271E+03	1.69236E+03	1.33479E+03
6.050(5) SEC.	1.82423E+03	2.08615E+03	2.16405E+03	2.03911E+03	1.73380E+03	1.36062E+03	1.08103E+03
8.640(5) SEC.	1.40765E+03	1.59971E+03	1.65698E+03	1.56454E+03	1.33898E+03	1.06165E+03	8.48807E+02
1.728(6) SEC.	7.33290E+02	8.27777E+02	8.56040E+02	8.10090E+02	6.98134E+02	5.59557E+02	4.50394E+02
2.592(6) SEC.	4.36408E+02	4.93149E+02	5.10113E+02	4.82568E+02	4.15420E+02	3.32415E+02	2.67387E+02
5.184(6) SEC.	1.25661E+02	1.44280E+02	1.49809E+02	1.40981E+02	1.19351E+02	9.30396E+01	7.37503E+01
7.776(6) SEC.	5.67301E+01	6.63983E+01	6.92517E+01	6.47691E+01	5.37345E+01	4.05207E+01	3.14982E+01
1.037(7) SEC.	3.58723E+01	4.23985E+01	4.43193E+01	4.13227E+01	3.39295E+01	2.51385E+01	1.93357E+01
1.296(7) SEC.	2.58634E+01	3.06595E+01	3.20700E+01	2.98740E+01	2.44517E+01	1.80180E+01	1.38155E+01
1.555(7) SEC.	1.92358E+01	2.28171E+01	2.38702E+01	2.22314E+01	1.81840E+01	1.33843E+01	1.02570E+01
2.074(7) SEC.	1.08027E+01	1.28118E+01	1.34026E+01	1.24832E+01	1.02122E+01	7.51897E+00	5.76365E+00
2.592(7) SEC.	6.08309E+00	7.21150E+00	7.54337E+00	7.02683E+00	5.75096E+00	4.23742E+00	3.24889E+00
3.154(7) SEC.	3.33545E+00	3.95205E+00	4.13342E+00	3.85103E+00	3.15366E+00	2.32590E+00	1.78336E+00

476 10/16/70

TABLE 2-64
(Continued)

Time	Polar Angle				
	140.0 DEG.	150.0 DEG.	160.0 DEG.	170.0 DEG.	180.0 DEG.
1.8(3) SEC.	9.52283E+05	1.34219E+06	1.89630E+06	2.22458E+06	2.32113E+06
2.4(3) SEC.	7.26075E+05	1.03058E+06	1.46049E+06	1.71438E+06	1.78920E+06
3.0(3) SEC.	5.79149E+05	8.26452E+05	1.17389E+06	1.37856E+06	1.43895E+06
3.6(3) SEC.	4.77415E+05	6.84201E+05	9.73597E+05	1.14371E+06	1.19397E+06
4.8(3) SEC.	3.48404E+05	5.02185E+05	7.16319E+05	8.41797E+05	8.78935E+05
6.0(3) SEC.	2.71321E+05	3.92396E+05	5.60508E+05	6.58823E+05	6.87955E+05
7.2(3) SEC.	2.20443E+05	3.19413E+05	4.56623E+05	5.36771E+05	5.60538E+05
1.8(4) SEC.	7.61848E+04	1.09564E+05	1.56195E+05	1.83508E+05	1.91613E+05
3.6(4) SEC.	2.95832E+04	4.34035E+04	6.24347E+04	7.34873E+04	7.67937E+04
7.2(4) SEC.	1.13097E+04	1.72178E+04	2.51773E+04	2.97243E+04	3.11058E+04
8.64(4) SEC.	8.70393E+03	1.33523E+04	1.95952E+04	2.31500E+04	2.42339E+04
1.728(5) SEC.	3.37895E+03	5.18195E+03	7.62113E+03	9.00908E+03	9.43391E+03
4.320(5) SEC.	1.41281E+03	2.02313E+03	2.89799E+03	3.41087E+03	3.56467E+03
6.050(5) SEC.	1.12199E+03	1.56491E+03	2.21566E+03	2.60221E+03	2.71681E+03
8.640(5) SEC.	8.67381E+02	1.18268E+03	1.65670E+03	1.94173E+03	2.02527E+03
1.728(6) SEC.	4.53564E+02	6.03973E+02	8.35972E+02	9.77390E+02	1.01823E+03
2.592(6) SEC.	2.70590E+02	3.62211E+02	5.02298E+02	5.87394E+02	6.11983E+02
5.184(6) SEC.	7.84346E+01	1.11765E+02	1.59214E+02	1.87041E+02	1.95274E+02
7.776(6) SEC.	3.56837E+01	5.45640E+01	7.99113E+01	9.43101E+01	9.86647E+01
1.037(7) SEC.	2.26913E+01	3.59205E+01	5.32607E+01	6.29789E+01	6.59429E+01
1.296(7) SEC.	1.64077E+01	2.62497E+01	3.90559E+01	4.62047E+01	4.83892E+01
1.555(7) SEC.	1.22190E+01	1.95949E+01	2.91735E+01	3.45155E+01	3.61482E+01
2.074(7) SEC.	6.86367E+00	1.10017E+01	1.63761E+01	1.93738E+01	2.02899E+01
2.592(7) SEC.	3.85733E+00	6.17157E+00	9.18343E+00	1.08646E+01	1.13787E+01
3.154(7) SEC.	2.10597E+00	3.35985E+00	4.99806E+00	5.91363E+00	6.19410E+00

476 10/16/70

TABLE 2-65

**R-1 REFERENCE DESIGN (W&NL DWG. 939J723) HEATING RATE BY REGION AT FULL POWER
(1511 Mw THERMAL) WITH GRAPHITE FUEL IN THE CORE**

Region Number	Region Description	Material	Weight (lbs)	Density (lb/in. ³)	HEATING RATE				
					BTU/HOUR				Mw
					Neutron	Prompt Gamma	Delayed Gamma	TOTAL	
1-15	CORE	Fueled Element	3.27 (3)	-	4.43 (9)	2.55 (8)	2.85 (8)*	4.97 (9)	1.46 (3)
		Unfueled Element	3.57 (2)	0.0738	1.04 (7)	1.98 (7)	9.13 (6)	3.93 (7)	1.15 (1)
		Pyro Sleeve	8.77 (1)	0.0694	2.70 (6)	4.80 (6)	2.21 (6)	9.71 (6)	2.85 (0)
		A-286	1.71 (2)	0.286	2.43 (5)	9.60 (6)	4.36 (6)	1.42 (7)	4.17 (0)
		SS-304	1.24 (1)	0.287	1.70 (4)	6.89 (5)	3.14 (5)	1.02 (6)	2.99(-1)
		Nbc/c Composite	4.02 (1)	0.150	9.38 (4)	4.91 (5)	1.84 (5)	7.69 (5)	2.25(-1)
		Tip	1.18 (0)	-	1.47 (6)	1.22(5)	5.83 (4)	1.65 (6)	4.85(-1)
		Hydrogen	3.94 (3)	-	4.44 (9)	2.91 (8)	3.01 (8)	5.04 (9)	1.48 (3)
		TOTAL							
16	CORE PERIPHERY	Graphite	2.09 (2)	0.0658	3.10 (6)	5.30 (6)	2.30 (6)	1.07 (7)	3.13 (0)
		Pyrofoil	2.20 (1)	0.0405	3.27 (5)	5.60 (5)	2.43 (5)	1.13 (6)	3.31(-1)
		ZrC (60% Dense)	1.92 (2)	0.146	4.06 (5)	5.54 (6)	2.31 (6)	8.26 (6)	2.42 (0)
		TZM Moly	1.50 (1)	0.369	6.27 (3)	4.53 (5)	1.88 (5)	6.47 (5)	1.90(-1)
		Hydrogen	8.81(-3)	-	5.38 (3)	4.39 (2)	1.91 (2)	6.01 (3)	1.76(-3)
		TOTAL	4.38 (2)	-	3.84 (6)	1.19 (7)	5.04 (6)	2.07 (7)	6.07 (0)
17	LATERAL SUPPORT	P 03 Graphite	6.19 (0)	0.0668	6.48 (4)	1.08 (5)	4.43 (4)	2.17 (5)	6.35(-2)
		ZTA Graphite	3.85 (2)	0.0703	4.03 (6)	6.72 (6)	2.75 (6)	1.35 (7)	3.95 (0)
		Pyrofoil	7.90 (0)	0.0405	8.27 (4)	1.38 (5)	5.65 (4)	2.77 (5)	8.11(-2)
		Hydrogen	3.75(-1)	-	1.65 (5)	1.27 (4)	5.28 (3)	1.83 (5)	5.36(-2)
		TOTAL	3.99 (2)	-	4.34 (6)	6.98 (6)	2.86 (6)	1.42 (7)	4.15 (0)
18	STRUCTURE	P 03 Graphite	8.64 (0)	0.0668	6.03 (4)	1.07 (5)	3.97 (4)	2.07 (5)	6.07(-2)
		AL-6061	7.51 (2)	0.0975	1.10 (6)	9.31 (6)	3.39 (6)	1.38 (7)	4.04 (3)
		A-286	5.12 (1)	0.286	1.72 (4)	6.80 (5)	2.40 (5)	9.37 (5)	2.75(-1)
		Hydrogen	4.95(-1)	-	1.47 (5)	1.25 (4)	4.47 (3)	1.64 (5)	4.80(-2)
		TOTAL	8.11 (2)	-	1.32 (6)	1.01 (7)	3.67 (6)	1.51 (7)	4.42 (0)
19-21	REFLECTOR	P 03 Graphite	2.33 (1)	0.0668	6.38 (4)	1.72 (5)	5.52 (4)	2.91 (5)	8.51(-2)
		Pyrofoil	8.42 (0)	0.0405	2.30 (4)	6.21 (4)	1.99 (4)	1.05 (5)	3.07(-2)
		Beryllium	1.70 (3)	0.0665	8.18 (6)	1.31 (7)	3.57 (6)	2.49 (7)	7.29 (0)
		AL-6061	3.95 (2)	0.0975	2.28 (5)	2.94 (6)	9.20 (5)	4.09 (6)	1.20 (0)
		A-286	3.28 (2)	0.286	4.37 (4)	2.66 (6)	7.96 (5)	3.50 (6)	1.02 (0)
		Control Vane	1.17 (2)	0.306	1.04 (7)	9.46 (5)	2.84 (5)	1.16 (7)	3.41 (0)
		Hydrogen	3.52 (0)	-	4.15 (5)	4.96 (4)	1.64 (4)	4.81 (5)	1.41(-1)
		TOTAL	2.55 (3)	-	1.94 (7)	1.99 (7)	5.81 (6)	4.50 (7)	1.32 (1)
22	PRESSURE VESSEL SIDE A	AL-7039	9.07 (2)	0.0988	1.03 (5)	2.82 (6)	8.02 (5)	3.72 (6)	1.09 (0)
		Hydrogen	1.07 (0)	-	2.40 (4)	6.21 (3)	1.89 (3)	3.21 (4)	9.42(-3)
		TOTAL	9.08 (2)	-	1.27 (5)	2.83 (6)	8.04 (5)	3.75 (6)	1.10 (0)
23	CHESH	Pyrographite	8.01 (0)	0.0789	3.05 (4)	5.74 (4)	2.01 (4)	1.08 (5)	3.16(-2)
		Pyrofoil	5.90(-2)	0.0405	2.25 (2)	4.22 (2)	1.48 (2)	7.95 (2)	2.33(-4)
		Nbc/c Composite	2.22 (1)	0.144	3.71 (4)	1.73 (5)	5.82 (4)	2.68 (5)	7.87(-2)
		W-ThO ₂	1.01 (2)	0.675	2.68 (3)	1.28 (6)	4.34 (5)	1.72 (6)	5.04(-1)
		A-286	4.05 (1)	0.286	7.10 (3)	3.08 (5)	1.03 (5)	4.18 (5)	1.22(-1)
		SS-304	1.36 (0)	0.287	2.31 (2)	1.02 (4)	3.42 (3)	1.39 (4)	4.08(-3)
		SS-316	9.24(-2)	0.288	1.55 (1)	7.00 (2)	2.33 (2)	9.48 (2)	2.78(-4)
		Hydrogen	1.34(-1)	-	2.06 (4)	1.84 (3)	6.58 (2)	2.31 (4)	6.76(-3)
		TOTAL	1.73 (2)	-	9.85 (4)	1.83 (6)	6.20 (5)	2.55 (6)	7.48(-1)
24	NOZZLE CHAMBER	Hydrogen	1.52(-1)	-	1.15 (4)	9.54 (2)	3.46 (2)	1.28 (4)	3.76(-3)
		TOTAL	1.52(-1)	-	1.15 (4)	9.54 (2)	3.46 (2)	1.28 (4)	3.76(-3)
25	NOZZLE BARREL	SS-347	3.42 (2)	0.285	1.45 (4)	6.62 (5)	1.46 (5)	8.22 (5)	2.41(-1)
		TOTAL	3.42 (2)	-	1.45 (4)	6.62 (5)	1.46 (5)	8.22 (5)	2.41(-1)

TABLE 2-65
(Continued)

Region Number	Region Description	Material	Weight (lbs)	Density (lb/in. 3)	HEATING RATE				
					BTU/HOUR				Mhw
					Neutron	Prompt Gamma	Delayed Gamma	TOTAL	
26	AFT REFLECTOR HARDWARE	AL-6061 A-286 SS-440C Hydrogen TOTAL	5.37 (1) 1.18 (1) 9.06 (0) 5.10(-1) 7.50 (1)	0.0975 0.286 0.280 - -	6.46 (3) 3.23 (2) 2.77 (2) 1.23 (4) 1.94 (4)	1.12 (5) 2.67 (4) 2.02 (4) 1.91 (3) 1.61 (5)	3.09 (4) 7.08 (3) 5.37 (3) 5.86 (2) 4.39 (4)	1.49 (5) 3.41 (4) 2.58 (4) 1.48 (4) 2.24 (5)	4.36(-2) 9.99(-3) 7.57(-3) 4.35(-3) 6.55(-2)
27	AFT REFLECTOR PLENUM	Hydrogen TOTAL	4.31 (0) 4.31 (0)	- -	7.61 (4) 7.61 (4)	1.05 (4) 1.05 (4)	2.81 (3) 2.81 (3)	8.94 (4) 8.94 (4)	2.62(-2) 2.62(-2)
28	CORE PLENUM	TZM Moly Copper-Baron A-286 SS-302 SS-304 Hydrogen TOTAL	5.71 (1) 1.06 (1) 3.04 (1) 3.49 (1) 1.07 (2) 7.73(-1) 2.41 (2)	0.369 0.271 0.286 0.286 0.287 - -	1.17 (4) 8.82 (5) 1.02 (4) 1.13 (4) 3.49 (4) 2.29 (5) 1.18 (6)	1.26 (6) 1.99 (5) 5.97 (5) 6.70 (5) 2.05 (6) 2.60 (4) 4.80 (6)	3.00 (5) 4.93 (4) 1.43 (5) 1.63 (5) 5.01 (5) 7.02 (3) 1.16 (6)	1.57 (6) 1.13 (6) 7.41 (5) 8.44 (5) 2.59 (6) 2.62 (5) 7.14 (6)	4.59(-1) 3.33(-1) 2.17(-1) 2.47(-1) 7.61(-1) 7.69(-2) 2.09 (0)
29	SUPPORT PLATE	Pyrofoil AL-6061 A-286 SS-302 SS-304 Hydrogen TOTAL	2.08 (0) 4.20 (2) 1.68 (1) 8.01(-1) 2.34 (2) 3.52 (0) 6.78 (2)	0.0405 0.0975 0.286 0.286 0.287 - -	5.34 (3) 2.25 (5) 2.03 (3) 9.28 (1) 2.73 (4) 3.74 (5) 6.34 (5)	1.44 (4) 2.97 (6) 1.31 (5) 6.18 (3) 1.81 (6) 4.60 (4) 4.98 (6)	2.40 (3) 4.75 (5) 1.98 (4) 9.35 (2) 2.73 (5) 7.95 (3) 7.79 (5)	2.21 (4) 3.67 (6) 1.53 (5) 7.21 (3) 2.11 (6) 4.28 (5) 6.39 (6)	6.47(-3) 1.08 (0) 4.49(-2) 2.11(-3) 6.18(-1) 1.25(-1) 1.88 (0)
30	LATERAL SUPPORT - FORWARD	AL-6061 A-286 Hydrogen TOTAL	1.44 (1) 3.80 (1) 1.99(-1) 5.26 (1)	0.0975 0.286 - -	1.04 (4) 6.23 (3) 2.87 (4) 4.53 (4)	1.23 (5) 3.53 (5) 3.17 (3) 4.79 (5)	3.00 (4) 8.25 (4) 8.30 (2) 1.13 (5)	1.63 (5) 4.42 (5) 3.27 (4) 6.38 (5)	4.78(-2) 1.30(-1) 9.59(-3) 1.87(-1)
31	FORWARD REFLECTOR HARDWARE I	AL-6061 A-286 SS-304 SS-440C Hydrogen TOTAL	4.23 (1) 3.53 (1) 4.42 (1) 9.06 (0) 2.46 (0) 1.33 (2)	0.0975 0.286 0.287 0.280 - -	7.56 (3) 1.43 (3) 1.72 (3) 4.09 (2) 8.73 (4) 9.84 (4)	1.38 (5) 1.28 (5) 1.58 (5) 3.21 (4) 1.46 (4) 4.71 (5)	2.63 (4) 2.29 (4) 2.83 (4) 5.79 (3) 3.05 (3) 8.63 (4)	1.72 (5) 1.52 (5) 1.88 (5) 3.83 (4) 1.05 (5) 6.55 (5)	5.03(-2) 4.44(-2) 5.50(-2) 1.12(-2) 3.08(-2) 1.92(-1)
32	SUPPORT PLATE PLENUM	A-286 SS-304 Hydrogen TOTAL	5.33(-1) 5.37 (1) 4.79(-1) 5.47 (1)	0.286 0.287 - -	3.28 (1) 3.18 (3) 2.60 (4) 2.92 (4)	2.55 (3) 2.54 (5) 3.75 (3) 2.60 (5)	2.60 (2) 2.60 (4) 4.48 (2) 2.67 (4)	2.84 (3) 2.83 (5) 3.02 (4) 3.16 (5)	8.32(-4) 8.29(-2) 8.85(-3) 9.26(-2)
33	INSTRUMENTATION RING	AL-6061 SS-304 Hydrogen TOTAL	3.20 (1) 5.99 (0) 1.16(-1) 3.81 (1)	0.0975 0.287 - -	4.18 (3) 1.70 (2) 3.01 (3) 7.36 (3)	7.08 (4) 1.46 (4) 4.75 (2) 8.59 (4)	7.63 (3) 1.48 (3) 5.53 (1) 9.17 (3)	8.26 (4) 1.62 (4) 3.54 (3) 1.02 (5)	2.42(-2) 4.76(-3) 1.04(-3) 3.00(-2)
34	FORWARD REFLECTOR HARDWARE II	AL-6061 A-286 SS-304 Hydrogen TOTAL	1.85 (1) 4.25 (0) 3.14 (0) 3.61(-1) 2.63 (1)	0.0975 0.286 0.287 - -	1.63 (3) 8.42 (1) 6.01 (1) 6.27 (3) 8.04 (3)	3.22 (4) 8.17 (3) 5.96 (3) 1.16 (3) 4.75 (4)	4.31 (3) 1.03 (3) 7.55 (2) 1.68 (2) 6.26 (3)	3.81 (4) 9.28 (3) 6.78 (3) 7.60 (3) 6.18 (4)	1.12(-2) 2.72(-3) 1.99(-3) 2.23(-3) 1.81(-2)
35	AFT CENTRAL SHIELD PLATE	AL-6061 Hydrogen TOTAL	9.78 (1) 4.67(-2) 9.79 (1)	0.0975 - -	2.27 (4) 2.16 (3) 2.49 (4)	3.50 (5) 3.05 (2) 3.50 (5)	3.70 (4) 3.52 (1) 3.70 (4)	4.10 (5) 2.50 (3) 4.13 (5)	1.20(-1) 7.33(-4) 1.21(-1)
36	BATH CENTRAL SHIELD	BATH AL-6061 Hydrogen TOTAL	1.24 (3) 3.35 (0) 2.93(-1) 1.24 (3)	0.103 0.0975 - -	2.09 (6) 1.66 (7) 2.86 (3) 2.09 (6)	1.33 (6) 3.53 (3) 5.84 (2) 1.33 (6)	1.21 (5) 3.24 (2) 5.65 (1) 1.21 (5)	3.54 (6) 4.02 (3) 3.50 (3) 3.55 (6)	1.04 (0) 1.18(-3) 1.03(-3) 1.04 (0)

TABLE 2-65
(Continued)

Region Number	Region Description	Material	Weight (lbs)	Density (lbs/in. ³)	HEATING RATE				
					BTU/HOUR				Mw
					Neutron	Prompt Gamma	Delayed Gamma	TOTAL	
37	FLOW BAFFLE I	AL-6061	2.36 (2)	0.0975	5.61 (3)	1.04 (5)	9.86 (3)	1.15 (5)	3.50(-2)
		SS-304	2.25 (1)	0.287	1.16 (2)	1.08 (4)	9.73 (2)	1.19 (4)	3.48(-3)
		Hydrogen	1.83(-2)	-	8.58 (1)	1.47 (1)	1.52 (0)	1.02 (2)	3.00(-5)
		TOTAL	2.59 (2)	-	5.81 (3)	1.15 (5)	1.08 (4)	1.31 (5)	3.85(-2)
38	BATH PERIPHERAL SHIELD I	BATH	1.84 (2)	0.103	2.65 (5)	1.29 (5)	1.19 (4)	4.06 (5)	1.19(-1)
		AL-6061	1.99 (1)	0.0975	6.24 (2)	1.37 (4)	1.27 (3)	1.56 (4)	4.56(-3)
		Hydrogen	5.02(-1)	-	3.11 (3)	6.46 (2)	6.38 (1)	3.82 (3)	1.12(-3)
		TOTAL	2.05 (2)	-	2.69 (5)	1.43 (5)	1.32 (4)	4.25 (5)	1.25(-1)
39	BATH PERIPHERAL SHIELD II	BATH	2.48 (2)	0.103	2.19 (5)	1.46 (5)	1.42 (4)	3.79 (5)	1.11(-1)
		Hydrogen	4.85(-2)	-	2.28 (2)	5.25 (1)	5.48 (0)	2.86 (2)	8.39(-5)
		TOTAL	2.48 (2)	-	2.19 (5)	1.46 (5)	1.42 (4)	3.79 (5)	1.11(-1)
40	BATH PERIPHERAL SHIELD III	BATH	2.46 (2)	0.103	1.78 (5)	1.20 (5)	1.30 (4)	3.11 (5)	9.12(-2)
		AL-6061	1.24 (1)	0.0975	2.56 (2)	5.97 (3)	6.50 (2)	6.88 (3)	2.02(-3)
		A-286	3.12 (1)	0.286	1.44 (2)	1.64 (4)	1.71 (3)	1.83 (4)	5.37(-3)
		SS-304	2.19 (1)	0.287	9.77 (1)	1.17 (4)	1.19 (3)	1.28 (4)	3.74(-3)
		Hydrogen	1.14(-1)	-	4.62 (2)	1.02 (2)	1.19 (1)	5.76 (2)	1.69(-4)
		TOTAL	3.12 (2)	-	1.79 (5)	1.54 (5)	1.66 (4)	3.50 (5)	1.02(-1)
41	BATH PERIPHERAL SHIELD IV	BATH	2.39 (2)	0.103	1.31 (5)	1.09 (5)	1.32 (4)	2.53 (5)	7.41(-2)
		AL-6061	1.95 (1)	0.0975	3.75 (2)	8.76 (3)	1.07 (3)	1.02 (4)	2.97(-3)
		Hydrogen	2.12(-2)	-	7.99 (1)	1.76 (1)	2.32 (0)	9.98 (1)	2.93(-5)
		TOTAL	2.58 (2)	-	1.31 (5)	1.18 (5)	1.43 (4)	2.63 (5)	7.71(-2)
42	PRESSURE VESSEL SIDE B	AL-7039	3.43 (2)	0.0988	3.11 (3)	7.02 (4)	9.90 (3)	8.32 (4)	2.44(-2)
		Hydrogen	4.04(-1)	-	7.12 (2)	1.52 (2)	2.32 (1)	8.87 (2)	2.60(-4)
		TOTAL	3.43 (2)	-	3.82 (3)	7.04 (4)	9.92 (3)	8.41 (4)	2.47(-2)
43	LEAD CENTRAL SHIELD	Lead Alloy	5.30 (2)	0.408	7.41 (1)	6.57 (4)	4.91 (3)	7.07 (4)	2.07(-2)
		AL-6061	3.07(-1)	0.0975	1.37 (0)	2.02 (1)	1.65 (0)	2.32 (1)	6.79(-6)
		Hydrogen	6.12(-2)	-	5.32 (1)	7.26 (0)	6.43(-1)	6.11 (1)	1.79(-5)
		TOTAL	5.30 (2)	-	1.29 (2)	6.57 (4)	4.91 (3)	7.08 (4)	2.07(-2)
44	LEAD PERIPHERAL SHIELD I	Lead Alloy	7.84 (1)	0.408	9.06 (0)	8.32 (3)	6.32 (2)	8.96 (3)	2.63(-3)
		AL-6061	6.74(-1)	0.0975	2.47 (0)	3.71 (1)	3.07 (0)	4.26 (1)	1.25(-5)
		Hydrogen	7.13(-2)	-	5.15 (1)	6.96 (0)	6.36(-1)	5.91 (1)	1.73(-5)
		TOTAL	7.91 (1)	-	6.30 (1)	8.36 (3)	6.36 (2)	9.06 (3)	2.66(-3)
45	LEAD PERIPHERAL SHIELD II	Lead Alloy	1.09 (2)	0.408	1.02 (1)	7.89 (3)	6.27 (2)	8.53 (3)	2.50(-3)
		Hydrogen	1.09(-2)	-	6.37 (0)	7.48(-1)	7.17(-2)	7.19 (0)	2.11(-6)
		TOTAL	1.09 (2)	-	1.66 (1)	7.89 (3)	6.27 (2)	8.54 (3)	2.50(-3)
46	LEAD PERIPHERAL SHIELD III	Lead Alloy	1.08 (2)	0.408	7.30 (0)	6.22 (3)	5.01 (2)	6.73 (3)	1.97(-3)
		AL-6061	1.46 (0)	0.0975	3.11 (0)	4.46 (1)	3.94 (0)	5.16 (1)	1.51(-5)
		A-286	3.54 (0)	0.286	1.69 (0)	1.21 (2)	1.00 (1)	1.33 (2)	3.90(-5)
		SS-304	2.57 (0)	0.287	1.19 (0)	8.69 (1)	7.20 (0)	9.53 (1)	2.79(-5)
		Hydrogen	1.85(-2)	-	7.75 (0)	1.02 (0)	9.77(-2)	8.87 (0)	2.60(-6)
		TOTAL	1.16 (2)	-	2.10 (1)	6.47 (3)	5.22 (2)	7.02 (3)	2.05(-3)
47	LEAD PERIPHERAL SHIELD IV	Lead Alloy	1.25 (2)	0.408	9.13 (0)	8.99 (3)	7.38 (2)	9.74 (3)	2.85(-3)
		AL-6061	2.64 (0)	0.0975	6.11 (0)	1.00 (2)	8.89 (0)	1.15 (2)	3.37(-5)
		Hydrogen	5.39(-3)	-	2.43 (0)	3.65(-1)	3.54(-2)	2.83 (0)	8.29(-7)
		TOTAL	1.28 (2)	-	1.77 (1)	9.09 (3)	7.47 (2)	9.86 (3)	2.88(-3)

TABLE 2-65
(Continued)

Region Number	Region Description	Material	Weight (lbs)	Density (lbs/in. ³)	HEATING RATE				
					BTU/HOUR				M
					Neutron	Prompt Gamma	Delayed Gamma	TOTAL	
48	PERIPHERAL SHIELD PLATE	AL-6061	4.35 (1)	0.0975	9.08 (1)	9.55 (2)	7.45 (1)	1.12 (3)	3.28(-4)
		A-286	2.13 (0)	0.286	9.95(-1)	5.19 (1)	3.80 (0)	5.67 (1)	1.66(-5)
		SS-304	1.48 (0)	0.287	6.73(-1)	3.58 (1)	2.63 (0)	3.91 (1)	1.15(-5)
		Hydrogen	5.24(-2)	-	2.15 (1)	2.02 (0)	1.75(-1)	2.37 (1)	6.96(-6)
		TOTAL	4.71 (1)	-	1.14 (2)	1.04 (3)	8.11 (1)	1.24 (3)	3.63(-4)
49	SHIELD PLENUM	AL-6061	1.88 (1)	0.0975	5.65 (1)	5.16 (2)	3.95 (1)	6.12 (2)	1.79(-4)
		SS-304	3.03(-1)	0.287	1.98(-1)	9.14 (0)	6.58(-1)	1.00 (1)	2.94(-6)
		Hydrogen	1.59 (0)	-	9.38 (2)	7.55 (1)	6.51 (0)	1.02 (3)	3.00(-4)
		TOTAL	2.07 (1)	-	9.95 (2)	6.01 (2)	4.67 (1)	1.64 (3)	4.82(-4)
50	FLOW BAFFLE II	AL-6061	1.30 (2)	0.0975	2.87 (2)	3.14 (3)	2.28 (2)	3.65 (3)	1.07(-3)
		TOTAL	1.30 (2)	-	2.87 (2)	3.14 (3)	2.28 (2)	3.65 (3)	1.07(-3)
51	CENTRAL DOME PLENUM	Hydrogen	7.77(-1)	-	2.68 (2)	3.24 (1)	2.64 (0)	3.03 (2)	8.88(-5)
		TOTAL	7.77(-1)	-	2.68 (2)	3.24 (1)	2.64 (0)	3.03 (2)	8.88(-5)
52	PERIPHERAL DOME PLENUM	AL-6061	2.58 (1)	0.0975	2.71 (1)	4.35 (2)	3.26 (1)	4.95 (2)	1.45(-4)
		A-286	1.43 (1)	0.286	3.38 (0)	2.69 (2)	1.89 (1)	2.91 (2)	8.54(-5)
		SS-304	8.24 (0)	0.287	1.91 (0)	1.54 (2)	1.08 (1)	1.67 (2)	4.89(-5)
		Hydrogen	8.04 (0)	-	1.67 (3)	2.50 (2)	1.98 (1)	1.94 (3)	5.67(-4)
		TOTAL	5.63 (1)	-	1.70 (3)	1.11 (3)	8.21 (1)	2.89 (3)	8.46(-4)
53	PRESSURE VESSEL DOME	AL-7039	2.17 (2)	0.0988	1.77 (2)	2.94 (3)	2.19 (2)	3.34 (3)	9.80(-4)
		TOTAL	2.17 (2)	-	1.77 (2)	2.94 (3)	2.19 (2)	3.34 (3)	9.80(-4)
	NERVA NUCLEAR SUBSYSTEM	-	1.38 (4)	-	4.48 (9)	3.59 (8)	3.22 (8)	5.16 (9)	1.51(3)

NOTE: Numbers in parentheses refer to powers of ten.

Refer to Figure 2-12 for region locations.

*Includes betas

TABLE 2-66
TIME INTEGRATED RADIATION ENVIRONMENT AT SELECTED LOCATIONS
(GRAPHITE CORE)

Page 1 of 31

LOCATION - REACTOR CENTERLINE, AFT SURFACE OF CHESH
RADIUS (IN) 0.0 STATION (IN) 54.13

Point 1

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-NBC/C COMPOSITE	6.44E 00	1.32	2.43E 02	1.32	2.44E 02	1.32	2.56E 02	1.32
BTU/IN3-W-THO2	3.89E 01	1.37	1.49E 03	1.37	1.50E 03	1.37	1.57E 03	1.37
BTU/IN3-A-286	1.07E 01	1.36	4.08E 02	1.36	4.10E 02	1.36	4.30E 02	1.36
NEUTRONS/CM2, E GT 1.0MEV	5.46E 15	1.04	1.89E 17	1.04	1.90E 17	1.04	1.94E 17	1.04
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.84E 16	1.04	6.37E 17	1.04	6.40E 17	1.04	6.69E 17	1.04
NEUTRONS/CM2, E LT 0.4EV	7.76E 13	1.87	2.64E 15	1.87	2.70E 15	1.87	2.82E 15	1.87
GAMMA RAY DOSE, RADS (CARBON)	8.07E 06	1.37	3.10E 08	1.37	3.12E 08	1.37	3.27E 08	1.37

7423 11/20/70

LOCATION - REACTOR CENTERLINE, CORE EXIT
RADIUS (IN) 0.0 STATION (IN) 52.00

Point 2

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-NBC/C COMPOSITE	1.41E 01	1.11	5.40E 02	1.11	5.43E 02	1.11	5.69E 02	1.11
BTU/IN3-W-THO2	1.02E 02	1.12	3.94E 03	1.12	3.96E 03	1.12	4.16E 03	1.12
BTU/IN3-A-286	2.37E 01	1.12	9.17E 02	1.12	9.22E 02	1.12	9.67E 02	1.12
NEUTRONS/CM2, E GT 1.0MEV	9.46E 15	1.02	3.27E 17	1.02	3.29E 17	1.02	3.44E 17	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.35E 16	1.02	1.51E 18	1.02	1.51E 18	1.02	1.58E 18	1.02
NEUTRONS/CM2, E LT 0.4EV	4.80E 13	1.29	1.66E 15	1.29	1.67E 15	1.29	1.74E 15	1.29
GAMMA RAY DOSE, RADS (CARBON)	1.83E 07	1.12	7.11E 08	1.12	7.15E 08	1.12	7.50E 08	1.12

TABLE 2-66
(Continued)

LOCATION - REACTOR CENTERLINE, CORE MIDPLANE
RADIUS (IN) 0.0 STATION (IN) 26.00

Point 3

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-A-286	1.64E 02	1.12	6.49E 03	1.12	6.52E 03	1.12	6.84E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	7.42E 16	1.02	2.57E 18	1.02	2.59E 18	1.02	2.70E 18	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.78E 17	1.02	1.31E 19	1.02	1.31E 19	1.02	1.37E 19	1.02
NEUTRONS/CM2, E LT 0.4EV	3.00E 14	1.29	1.04E 16	1.29	1.04E 16	1.29	1.09E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	1.29E 08	1.12	5.12E 09	1.12	5.15E 09	1.12	5.40E 09	1.12

LOCATION - REACTOR CENTERLINE, CORE INLET
RADIUS (IN) 0.0 STATION (IN) 0.0

Point 4

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-T2M MOLY	7.50E 01	1.12	2.86E 03	1.12	2.88E 03	1.12	3.01E 03	1.12
BTU/IN3-CU-18V/O BN	1.62E 02	1.50	5.79E 03	1.50	5.82E 03	1.50	6.08E 03	1.50
BTU/IN3-A-286	5.17E 01	1.12	1.97E 03	1.12	1.98E 03	1.12	2.08E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.82E 16	1.02	6.29E 17	1.02	6.33E 17	1.02	6.61E 17	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	9.65E 16	1.02	3.34E 18	1.02	3.35E 18	1.02	3.51E 18	1.02
NEUTRONS/CM2, E LT 0.4EV	4.42E 14	1.29	1.53E 16	1.29	1.54E 16	1.29	1.61E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	3.96E 07	1.12	1.52E 09	1.12	1.53E 09	1.12	1.60E 09	1.12

TABLE 2-66
(Continued)

Page 3 of 31

LOCATION - REACTOR CENTERLINE, AFT FACE OF SUPPORT PLATE
 RADIUS (IN) 0.0 STATION (IN) -3.50

Point 5

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-TZM MULD	4.41E 01	1.07	1.61E 03	1.07	1.62E 03	1.07	1.69E 03	1.07
BTU/IN3-CU-18V/O BN	2.46E 02	1.64	8.55E 03	1.64	8.59E 03	1.64	8.98E 03	1.64
BTU/IN3-AL-6061	1.00E 01	1.07	3.65E 02	1.07	3.67E 02	1.07	3.84E 02	1.07
BTU/IN3-A-286	3.02E 01	1.07	1.10E 03	1.07	1.11E 03	1.07	1.16E 03	1.07
NEUTRONS/CM2, E GT 1.0MEV	9.88E 15	1.59	3.42E 17	1.59	3.44E 17	1.59	3.59E 17	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.37E 16	1.59	2.20E 18	1.59	2.21E 18	1.59	2.32E 18	1.59
NEUTRONS/CM2, E LT 0.4EV	2.34E 15	1.73	8.10E 16	1.73	8.15E 16	1.73	8.51E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	2.22E 07	1.07	8.17E 08	1.07	8.21E 08	1.07	8.59E 08	1.07

LOCATION - REACTOR CENTERLINE, FORWARD FACE OF SUPPORT PLATE
 RADIUS (IN) 0.0 STATION (IN) -9.50

Point 6

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	3.39E 00	1.07	1.21E 02	1.07	1.21E 02	1.07	1.27E 02	1.07
BTU/IN3-A-286	1.06E 01	1.07	3.75E 02	1.07	3.77E 02	1.07	3.94E 02	1.07
NEUTRONS/CM2, E GT 1.0MEV	2.78E 15	1.59	9.63E 16	1.59	9.68E 16	1.59	1.01E 17	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.31E 16	1.59	8.00E 17	1.59	8.04E 17	1.59	8.41E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	2.54E 15	1.73	8.78E 16	1.73	8.83E 16	1.73	9.23E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	7.52E 06	1.07	2.68E 08	1.07	2.69E 08	1.07	2.82E 08	1.07

TABLE 2-66
(Continued)

LOCATION - REACTOR CENTERLINE, AFT FACE OF CENTRAL SHIELD PLATE
 RADIUS (IN) 0.0 STATION (IN) -10.83

Point 7

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	2.87E 00	1.07	1.07E 02	1.07	1.02E 02	1.07	1.07E 02	1.07
BTU/IN3-A-286	8.89E 00	1.07	3.15E 02	1.07	3.16E 02	1.07	3.31E 02	1.07
NEUTRONS/CM2, E GT 1.0MEV	2.43E 15	1.59	8.41E 16	1.59	8.46E 16	1.59	8.84E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.85E 16	1.59	6.18E 17	1.59	6.42E 17	1.59	6.71E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	1.48E 15	1.73	5.13E 16	1.73	5.16E 16	1.73	5.39E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	6.36E 06	1.07	2.26E 08	1.07	2.28E 08	1.07	2.38E 08	1.07

LOCATION - REACTOR CENTERLINE, AFT FACE OF BATH CENTRAL SHIELD
 RADIUS (IN) 0.0 STATION (IN) -11.83

Point 8

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	1.47E 01	1.61	5.12E 02	1.61	5.15E 02	1.61	5.38E 02	1.61
BTU/IN3-AL-6061	2.41E 00	1.07	8.52E 01	1.07	8.56E 01	1.07	8.95E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.87E 15	1.59	6.47E 16	1.59	6.50E 16	1.59	6.80E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.47E 16	1.59	5.08E 17	1.59	5.11E 17	1.59	5.34E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	6.21E 14	1.73	2.15E 16	1.73	2.16E 16	1.73	2.26E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	5.43E 06	1.07	1.92E 08	1.07	1.93E 08	1.07	2.02E 08	1.07

TABLE 2-66
(Continued)

LOCATION - REACTOR CENTERLINE, BATH CENTRAL SHIELD MIDPLANE
 RADIUS (IN) 0.0 STATION (IN) -17.38

Point 9

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	1.05E 00	1.37	3.79E 01	1.37	3.80E 01	1.37	3.98E 01	1.37
BTU/IN3-AL-6061	5.04E-01	1.07	1.79E 01	1.07	1.80E 01	1.07	1.88E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	2.91E 14	1.59	1.01E 16	1.59	1.01E 16	1.59	1.06E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.33E 15	1.59	4.59E 16	1.59	4.62E 16	1.59	4.83E 16	1.59
NEUTRONS/CM2, E LT 0.4EV	1.99E 12	1.73	6.88E 13	1.73	6.91E 13	1.73	7.22E 13	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.17E 06	1.07	4.13E 07	1.07	4.15E 07	1.07	4.34E 07	1.07

LOCATION - REACTOR CENTERLINE, FORWARD FACE OF BATH CENTRAL SHIELD
 RADIUS (IN) 0.0 STATION (IN) -22.93

Point 10

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	2.55E-01	1.44	8.89E 00	1.44	8.93E 00	1.44	9.34E 00	1.44
BTU/IN3-LEAD ALLOY	9.15E-01	1.07	3.21E 01	1.07	3.23E 01	1.07	3.37E 01	1.07
BTU/IN3-AL-6061	4.76E-02	1.07	3.43E 00	1.07	3.45E 00	1.07	3.60E 00	1.07
NEUTRONS/CM2, E GT 1.0MEV	5.31E 13	1.59	1.84E 15	1.59	1.85E 15	1.59	1.93E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.99E 14	1.59	6.90E 15	1.59	6.94E 15	1.59	7.25E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	3.01E 12	1.73	1.04E 14	1.73	1.05E 14	1.73	1.09E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	2.21E 05	1.07	7.78E 06	1.07	7.81E 06	1.07	8.17E 06	1.07

TABLE 2-66
(Continued)

LOCATION - REACTOR CENTERLINE, FORWARD FACE OF LEAD CENTRAL SHIELD
 RADIUS (IN) 0.0 STATION (IN) -24.23

Point 11

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-LEAD ALLOY	1.36E-01	1.07	4.76E 00	1.07	4.78E 00	1.07	5.00E 00	1.07
BTU/IN3-AL-6061	2.00E-02	1.07	7.03E-01	1.07	7.06E-01	1.07	7.38E-01	1.07
NEUTRONS/CM2, E GT 1.0MEV	3.48E 13	1.59	1.20E 15	1.59	1.21E 15	1.59	1.26E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.35E 14	1.59	4.66E 15	1.59	4.69E 15	1.59	4.93E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.13E 13	1.73	3.93E 14	1.73	3.95E 14	1.73	4.12E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	4.14E 04	1.07	1.46E 06	1.07	1.46E 06	1.07	1.53E 06	1.07

LOCATION - REACTOR CENTERLINE, AFT FACE OF FLOW BAFFLE 11
 RADIUS (IN) 0.0 STATION (IN) -29.63

Point 12

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	1.73E-02	1.41	6.06E-01	1.41	6.09E-01	1.41	6.37E-01	1.41
NEUTRONS/CM2, E GT 1.0MEV	2.55E 13	3.10	8.84E 14	3.10	8.84E 14	3.10	9.28E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.02E 14	3.10	3.53E 15	3.10	3.55E 15	3.10	3.71E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.43E 13	3.10	4.96E 14	3.10	4.98E 14	3.10	5.21E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	3.65E 04	1.40	1.28E 06	1.40	1.29E 06	1.40	1.35E 06	1.40

TABLE 2-66
(Continued)

LOCATION - REACTOR CENTERLINE, FORWARD FACE OF FLOW BARRIER II
 RADIUS (IN) 0.0 STATION (IN) -30.75

Point 13

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	1.48E-02	1.40	5.20E-01	1.40	5.22E-01	1.40	5.46E-01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.94E 13	3.10	6.72E 14	3.10	6.75E 14	3.10	7.06E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.10E 13	3.10	2.80E 15	3.10	2.82E 15	3.10	2.74E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.43E 13	3.10	4.96E 14	3.10	4.99E 14	3.10	5.21E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	3.17E 04	1.40	1.11E 06	1.40	1.12E 06	1.40	1.17E 06	1.40

LOCATION - REACTOR CENTERLINE, AFT SURFACE OF PRESSURE VESSEL DOME
 RADIUS (IN) 0.0 STATION (IN) -30.04

Point 14

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-7039	1.13E-02	1.40	3.96E-01	1.40	3.98E-01	1.40	4.16E-01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.33E 13	3.10	4.60E 14	3.10	4.63E 14	3.10	4.84E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.49E 13	3.10	1.55E 15	3.10	1.56E 15	3.10	1.63E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	7.77E 12	3.10	2.69E 14	3.10	2.70E 14	3.10	2.82E 14	3.10
GAMMA RAY DOSE, RADS (CAPS)	2.41E 04	1.40	8.47E 05	1.40	8.51E 05	1.40	8.89E 05	1.40

TABLE 2-66
(Continued)

LOCATION - REACTOR CENTERLINE, TANGENT PLANE
 RADIUS (IN) 0.0 STATION (IN) -37.00

Point 15

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-7039	9.22E-03	1.40	3.24E-01	1.40	3.25E-01	1.40	3.40E-01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.03E 13	3.10	3.56E 14	3.10	3.58E 14	3.10	3.74E 14	3.10
NEUTRONS/CM2, 0.4EV LT - LT 1.0MEV	2.91E 13	3.10	1.01E 15	3.10	1.01E 15	3.10	1.06E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	4.78E 12	3.10	1.65E 14	3.10	1.66E 14	3.10	1.74E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	1.98E 04	1.40	6.97E 05	1.40	7.00E 05	1.40	7.32E 05	1.40

LOCATION - CORE EDGE, AFT SURFACE OF CHESH
 RADIUS (IN) 18.66 STATION (IN) 54.13

Point 16

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-NBC/C COMPOSITE	3.28E 00	1.32	1.22E 02	1.32	1.23E 02	1.32	1.29E 02	1.32
BTU/IN3-W-TH02	2.19E 01	1.37	8.29E 02	1.37	8.33E 02	1.37	8.73E 02	1.37
BTU/IN3-A-286	5.42E 00	1.36	2.04E 02	1.36	2.05E 02	1.36	2.15E 02	1.36
NEUTRONS/CM2, E GT 1.0MEV	2.65E 15	1.04	9.19E 16	1.04	9.23E 16	1.04	9.65E 16	1.04
NEUTRONS/CM2, 0.4EV LT - LT 1.0MEV	1.19E 16	1.04	4.11E 17	1.04	4.14E 17	1.04	4.32E 17	1.04
NEUTRONS/CM2, E LT 0.4EV	4.72E 14	1.87	1.63E 16	1.87	1.64E 16	1.87	1.72E 16	1.87
GAMMA RAY DOSE, RADS (CARBON)	4.03E 06	1.37	1.53E 08	1.37	1.54E 08	1.37	1.62E 08	1.37

TABLE 2-66
(Continued)

LOCATION - CORE EDGE, CORE EXIT
 RADIUS (IN) 18.66 STATION (IN) 52.00

Point 17

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-GRAPHITITE G	2.99E 00	1.11	1.12E 02	1.11	1.13E 02	1.11	1.18E 02	1.11
BTU/IN3-NBC/C COMPOSITE	5.96E 00	1.11	2.27E 02	1.11	2.28E 02	1.11	2.39E 02	1.11
BTU/IN3-ZRC(0.60DENSE)	5.61E 00	1.12	2.15E 02	1.12	2.16E 02	1.12	2.27E 02	1.12
BTU/IN3-W-THO2	4.33E 01	1.12	1.67E 03	1.12	1.68E 03	1.12	1.76E 03	1.12
BTU/IN3-A-286	9.91E 00	1.12	3.82E 02	1.12	3.84E 02	1.12	4.03E 02	1.12
NEUTRONS/CM2, E GT 1.0MEV	4.22E 15	1.02	1.46E 17	1.02	1.47E 17	1.02	1.53E 17	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.99E 16	1.02	6.89E 17	1.02	6.93E 17	1.02	7.24E 17	1.02
NEUTRONS/CM2, E LT 0.4EV	5.14E 14	1.29	1.78E 16	1.29	1.79E 16	1.29	1.87E 16	1.29
GAMMA RAY DOSE, RADS (CARB N)	7.57E 06	1.12	2.94E 08	1.12	2.95E 08	1.12	3.10E 08	1.12

LOCATION - CORE EDGE, CORE MIDPLANE
 RADIUS (IN) 18.66 STATION (IN) 26.00

Point 18

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-GRAPHITITE G	2.23E 01	1.11	8.49E 02	1.11	8.53E 02	1.11	8.94E 02	1.11
BTU/IN3-ZRC(0.60DENSE)	4.02E 01	1.12	1.58E 03	1.12	1.59E 03	1.12	1.67E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	3.34E 16	1.02	1.15E 18	1.02	1.16E 18	1.02	1.21E 18	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.78E 17	1.02	6.16E 18	1.02	6.19E 18	1.02	6.47E 18	1.02
NEUTRONS/CM2, E LT 0.4EV	3.65E 15	1.29	1.26E 17	1.29	1.27E 17	1.29	1.33E 17	1.29
GAMMA RAY DOSE, RADS (CARBON)	5.45E 07	1.12	2.16E 09	1.12	2.18E 09	1.12	2.28E 09	1.12

TABLE 2-66
(Continued)

LOCATION - CORE EDGE, CORE INLET
RADIUS (IN) 18.06

STATION (IN) 0.0

Point 19

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-GRAPHITITE G	6.38E 00	1.11	2.38E 02	1.11	2.39E 02	1.11	2.51E 02	1.11
BTU/IN3-ZRC(0.6DENSE)	1.25E 01	1.12	4.73E 02	1.12	4.76E 02	1.12	4.99E 02	1.12
NEUTRONS/CM2, E GT 1.0MEV	8.25E 15	1.02	2.86E 17	1.02	2.87E 17	1.02	3.03E 17	1.02
NEUTRONS/CM2, 0.4EV LT L LT 1.0MEV	4.56E 16	1.02	1.58E 18	1.02	1.59E 18	1.02	1.66E 18	1.02
NEUTRONS/CM2, E LT 0.4EV	1.27E 15	1.29	4.38E 16	1.29	4.40E 16	1.29	4.60E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	1.66E 07	1.12	6.36E 08	1.12	6.40E 08	1.12	6.70E 08	1.12

LOCATION - OUTER RADIUS OF CORE PERIPHERY, CORE EXIT
RADIUS (IN) 19.48

STATION (IN) 52.00

Point 20

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-GRAPHITITE G	2.50E 00	1.11	9.40E 01	1.11	9.45E 01	1.11	9.90E 01	1.11
BTU/IN3-ZRC(0.6DENSE)	4.75E 00	1.12	1.83E 02	1.12	1.84E 02	1.12	1.93E 02	1.12
NEUTRONS/CM2, E GT 1.0MEV	3.40E 15	1.13	1.18E 17	1.13	1.18E 17	1.13	1.24E 17	1.13
NEUTRONS/CM2, 0.4EV LT L LT 1.0MEV	1.83E 16	1.13	6.32E 17	1.13	6.35E 17	1.13	6.54E 17	1.13
NEUTRONS/CM2, E LT 0.4EV	7.71E 14	1.29	2.67E 16	1.29	2.68E 16	1.29	2.80E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	6.38E 06	1.12	2.48E 08	1.12	2.49E 08	1.12	2.61E 08	1.12

TABLE 2-66
(Continued)

LOCATION - OUTER RADIUS OF CORE PERIPHERY, CORE MIDPLANE
 RADIUS (IN) 19.48 STATION (IN) 25.00

Point 21

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-GRAPHITITE G	1.64E 01	1.11	6.22E 02	1.11	6.26E 02	1.11	6.56E 02	1.11
BTU/IN3-ZRC(0.6DENSE)	2.94E 01	1.11	1.15E 03	1.11	1.16E 03	1.11	1.21E 03	1.11
NEUTRONS/CM2, E GT 1.0MEV	2.49E 16	1.13	8.63E 17	1.13	8.67E 17	1.13	9.06E 17	1.13
NEUTRONS/CM2, 0.4EV LT F LT 1.0MEV	1.62E 17	1.13	5.62E 18	1.13	5.65E 18	1.13	5.91E 18	1.13
NEUTRONS/CM2, E LT 0.4EV	6.14E 15	1.29	2.12E 17	1.29	2.13E 17	1.29	2.23E 17	1.29
GAMMA RAY DOSE, RADS (CARBON)	3.94E 07	1.12	1.56E 09	1.12	1.56E 09	1.12	1.64E 09	1.12

LOCATION - OUTER RADIUS OF CORE PERIPHERY, CORE INLET
 RADIUS (IN) 19.48 STATION (IN) 0.0

Point 22

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-GRAPHITITE G	5.07E 00	1.11	1.48E 02	1.11	1.49E 02	1.11	1.93E 02	1.11
BTU/IN3-ZRC(0.6DENSE)	9.99E 00	1.12	3.77E 02	1.12	3.79E 02	1.12	3.77E 02	1.12
NEUTRONS/CM2, E GT 1.0MEV	6.49E 15	1.13	2.25E 17	1.13	2.26E 17	1.13	2.36E 17	1.13
NEUTRONS/CM2, 0.4EV LT F LT 1.0MEV	4.22E 16	1.13	1.46E 18	1.13	1.47E 18	1.13	1.53E 18	1.13
NEUTRONS/CM2, E LT 0.4EV	1.93E 15	1.29	6.69E 16	1.29	6.73E 16	1.29	7.03E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	1.31E 07	1.12	5.00E 08	1.12	5.02E 08	1.12	5.26E 08	1.12

TABLE 2-66
(Continued)

LOCATION - INNER RADIUS OF STRUCTURE, AFT END OF STRUCTURE
 RADIUS (IN) 21.11 STATION (IN) 55.20

Point 23

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	1.71E 00	1.11	6.46E 01	1.11	6.50E 01	1.11	6.81E 01	1.11
BTU/IN3-A-286	5.05E 00	1.12	1.91E 02	1.12	1.92E 02	1.12	2.02E 02	1.12
NEUTRONS/CM2, E GT 1.0MEV	2.08E 15	1.13	7.21E 16	1.13	7.25E 16	1.13	7.54E 16	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.09E 16	1.13	3.78E 17	1.13	3.80E 17	1.13	3.97E 17	1.13
NEUTRONS/CM2, E LT 0.4EV	5.59E 14	1.29	1.94E 16	1.29	1.95E 16	1.29	2.03E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	3.77E 06	1.12	1.44E 08	1.12	1.45E 08	1.12	1.52E 08	1.12

LOCATION - INNER RADIUS OF STRUCTURE, CORE MIDPLANE
 RADIUS (IN) 21.11 STATION (IN) 26.00

Point 24

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	1.19E 01	1.11	4.61E 02	1.11	4.64E 02	1.11	4.86E 02	1.11
BTU/IN3-A-286	3.45E 01	1.12	1.34E 03	1.12	1.35E 03	1.12	1.41E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.56E 16	1.13	5.43E 17	1.13	5.43E 17	1.13	5.68E 17	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.37E 17	1.13	4.74E 18	1.13	4.75E 18	1.13	4.93E 18	1.13
NEUTRONS/CM2, E LT 0.4EV	8.45E 15	1.29	2.92E 17	1.29	2.94E 17	1.29	3.07E 17	1.29
GAMMA RAY DOSE, RADS (CARBON)	2.63E 07	1.12	1.03E 09	1.12	1.04E 09	1.12	1.09E 09	1.12

TABLE 2-66
(Continued)

LOCATION - INNER RADIUS OF STRUCTURE, FORWARD END OF STRUCTURE
 RADIUS (IN) 21.11 STATION (IN) -3.50

Point 25

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	3.07E 00	1.11	1.12E 02	1.11	1.13E 02	1.11	1.18E 02	1.11
BTU/IN3-A-286	9.37E 00	1.12	3.42E 02	1.12	3.44E 02	1.12	3.60E 02	1.12
NEUTRONS/CM2, E GT 1.0MEV	2.96E 15	1.13	1.02E 17	1.13	1.03E 17	1.13	1.07E 17	1.13
NEUTRONS/CM2, 0.4EV LT F LT 1.0MEV	2.14E 16	1.13	7.40E 17	1.13	7.44E 17	1.13	7.77E 17	1.13
NEUTRONS/CM2, E LT 0.4EV	2.15E 15	1.29	7.46E 16	1.29	7.49E 16	1.29	7.83E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	6.78E 06	1.12	2.49E 08	1.12	2.51E 08	1.12	2.62E 08	1.12

LOCATION - OUTER RADIUS OF STRUCTURE, AFT END OF STRUCTURE
 RADIUS (IN) 22.21 STATION (IN) 55.20

Point 26

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	1.23E 00	1.11	4.59E 01	1.11	4.61E 01	1.11	4.83E 01	1.11
BTU/IN3-A-286	3.69E 00	1.12	1.38E 02	1.12	1.39E 02	1.12	1.45E 02	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.28E 15	1.13	4.45E 16	1.13	4.47E 16	1.13	4.67E 16	1.13
NEUTRONS/CM2, 0.4EV LT F LT 1.0MEV	8.19E 15	1.13	2.84E 17	1.13	2.85E 17	1.13	2.98E 17	1.13
NEUTRONS/CM2, E LT 0.4EV	8.75E 14	1.29	3.03E 16	1.29	3.04E 16	1.29	3.18E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	2.73E 06	1.12	1.03E 08	1.12	1.04E 08	1.12	1.08E 08	1.12

2-235

TABLE 2-66
(Continued)

LOCATION - INNER RADIUS OF REFLECTOR, AFT END OF REFLECTOR
RADIUS (IN) 22.21 STATION (IN) 52.00

Point 27

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BERYLLIUM	1.69E 00	1.12	6.21E 01	1.12	6.24E 01	1.12	6.54E 01	1.12
BTU/IN3-AL-6061	1.76E 00	1.11	6.70E 01	1.11	6.73E 01	1.11	7.06E 01	1.11
BTU/IN3-A-286	5.25E 00	1.12	2.00E 02	1.12	2.01E 02	1.12	2.10E 02	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.82E 15	1.13	6.30E 16	1.13	6.34E 16	1.13	6.62E 16	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.52E 16	1.13	5.25E 17	1.13	5.28E 17	1.13	5.52E 17	1.13
NEUTRONS/CM2, E LT 0.4EV	1.22E 15	1.29	4.23E 16	1.29	4.25E 16	1.29	4.44E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	3.94E 06	1.12	1.51E 08	1.12	1.52E 08	1.12	1.59E 08	1.12

LOCATION - INNER RADIUS OF REFLECTOR, CORE MIDPLANE
RADIUS (IN) 22.21 STATION (IN) 26.00

Point 28

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BERYLLIUM	9.69E 00	1.14	3.55E 02	1.14	3.57E 02	1.14	3.74E 02	1.14
BTU/IN3-AL-6061	9.03E 00	1.11	3.45E 02	1.11	3.47E 02	1.11	3.64E 02	1.11
BTU/IN3-A-286	2.63E 01	1.12	1.01E 03	1.12	1.02E 03	1.12	1.07E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.13E 16	1.13	3.90E 17	1.13	3.92E 17	1.13	4.10E 17	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.21E 17	1.13	4.17E 18	1.13	4.19E 18	1.13	4.38E 18	1.13
NEUTRONS/CM2, E LT 0.4EV	8.99E 15	1.29	3.11E 17	1.29	3.13E 17	1.29	3.27E 17	1.29
GAMMA RAY DOSE, RADS (CARBON)	1.98E 07	1.12	7.68E 08	1.12	7.72E 08	1.12	8.69E 08	1.12

TABLE 2-66
(Continued)

LOCATION - INNER RADIUS OF REFLECTOR, FORWARD END OF REFLECTOR
 RADIUS (IN) 22.21 STATION (IN) 0.0

Point 29

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BERYLLIUM	3.07E 00	1.12	1.12E 02	1.12	1.12E 02	1.12	1.17E 02	1.12
BTU/IN3-AL-6061	3.26E 00	1.11	1.22E 02	1.11	1.22E 02	1.11	1.28E 02	1.11
BTU/IN3-A-286	9.82E 00	1.12	3.66E 02	1.12	3.68E 02	1.12	3.86E 02	1.12
NEUTRONS/CM2, E GT 1.0MEV	3.16E 15	1.13	1.09E 17	1.13	1.10E 17	1.13	1.15E 17	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.96E 16	1.13	1.02E 18	1.13	1.03E 18	1.13	1.08E 18	1.13
NEUTRONS/CM2, E LT 0.4EV	2.72E 15	1.29	9.42E 16	1.29	9.47E 16	1.29	9.89E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	7.25E 06	1.12	2.73E 08	1.12	2.74E 08	1.12	2.87E 08	1.12

LOCATION - OUTER RADIUS OF REFLECTOR, AFT END OF REFLECTOR
 RADIUS (IN) 27.00 STATION (IN) 52.00

Point 30

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BERYLLIUM	5.53E-01	1.11	2.05E 01	1.11	2.06E 01	1.11	2.16E 01	1.11
BTU/IN3-AL-7039	7.70E-01	1.12	2.87E 01	1.12	2.90E 01	1.12	3.04E 01	1.12
BTU/IN3-A-286	2.37E 00	1.12	8.88E 01	1.12	8.93E 01	1.12	9.35E 01	1.12
NEUTRONS/CM2, E GT 1.0MEV	3.21E 14	1.13	1.11E 16	1.13	1.12E 16	1.13	1.17E 16	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.64E 15	1.13	9.14E 16	1.13	9.18E 16	1.13	9.30E 16	1.13
NEUTRONS/CM2, E LT 0.4EV	5.10E 14	1.29	1.76E 16	1.29	1.77E 16	1.29	1.85E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	1.75E 06	1.12	6.62E 07	1.12	6.65E 07	1.12	6.97E 07	1.12

TABLE 2-66
(Continued)

LOCATION - OUTER RADIUS OF REFLECTOR, CORE MIDPLANE
 RADIUS (IN) 27.00 STATION (IN) 26.00

Point 31

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BERYLLIUM	2.32E 00	1.11	8.65E 01	1.11	8.67E 01	1.11	9.10E 01	1.11
BTU/IN3-AL-7039	3.12E 00	1.12	1.18E 02	1.12	1.19E 02	1.12	1.24E 02	1.12
BTU/IN3-A-286	9.55E 00	1.12	3.61E 02	1.12	3.67E 02	1.12	3.83E 02	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.50E 15	1.13	5.19E 16	1.13	5.21E 16	1.13	5.45E 16	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.50E 16	1.13	5.18E 17	1.13	5.21E 17	1.13	5.44E 17	1.13
NEUTRONS/CM2, E LT 0.4EV	3.06E 15	1.29	1.06E 17	1.29	1.06E 17	1.29	1.11E 17	1.29
GAMMA RAY DOSE, RADS (CARBON)	7.10E 06	1.12	2.71E 08	1.12	2.72E 08	1.12	2.85E 08	1.12

LOCATION - OUTER RADIUS OF REFLECTOR, FORWARD END OF REFLECTOR
 RADIUS (IN) 27.00 STATION (IN) 0.00

Point 32

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BERYLLIUM	1.04E 00	1.11	3.81E 01	1.11	3.83E 01	1.11	4.01E 01	1.11
BTU/IN3-AL-7039	1.42E 00	1.12	5.24E 01	1.12	5.26E 01	1.12	5.51E 01	1.12
BTU/IN3-A-286	4.40E 00	1.12	1.62E 02	1.12	1.63E 02	1.12	1.70E 02	1.12
NEUTRONS/CM2, E GT 1.0MEV	6.52E 14	1.13	2.26E 16	1.13	2.27E 16	1.13	2.37E 16	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.66E 15	1.13	1.94E 17	1.13	1.95E 17	1.13	2.03E 17	1.13
NEUTRONS/CM2, E LT 0.4EV	1.01E 15	1.29	3.49E 16	1.29	3.51E 16	1.29	3.60E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	3.21E 06	1.12	1.19E 08	1.12	1.19E 08	1.12	1.25E 08	1.12

TABLE 2-66
(Continued)

LOCATION - PRESSURE VESSEL SIDE SURFACE, PRESSURE VESSEL - NOZZLE MATING FLANGE
 RADIUS (IN) 27.89 STATION (IN) 63.24

Point 33

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-7039	2.38E-01	1.39	8.87E-00	1.39	8.91E-00	1.39	9.33E-00	1.39
NEUTRONS/CM2, E GT 1.0MEV	2.48E-14	3.10	8.58E-15	3.10	8.62E-15	3.10	9.01E-15	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.43E-14	3.10	2.22E-16	3.10	2.23E-16	3.10	2.34E-16	3.10
NEUTRONS/CM2, E LT 0.4EV	7.16E-13	3.10	2.48E-15	3.10	2.49E-15	3.10	2.60E-15	3.10
GAMMA RAY DOSE, RADS (CARBON)	5.19E-05	1.40	1.95E-07	1.40	1.96E-07	1.40	2.06E-07	1.40

LOCATION - PRESSURE VESSEL SIDE SURFACE, COPE MIDPLANE
 RADIUS (IN) 27.85 STATION (IN) 26.00

Point 34

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-7039	2.22E-00	1.39	8.41E-01	1.39	8.45E-01	1.39	8.86E-01	1.39
NEUTRONS/CM2, E GT 1.0MEV	9.77E-14	3.10	3.38E-16	3.10	3.40E-16	3.10	3.55E-16	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.95E-15	3.10	2.75E-17	3.10	2.77E-17	3.10	2.89E-17	3.10
NEUTRONS/CM2, E LT 0.4EV	1.63E-15	3.10	5.62E-16	3.10	5.65E-16	3.10	5.91E-16	3.10
GAMMA RAY DOSE, RADS (CARBON)	5.07E-06	1.40	1.93E-08	1.40	1.94E-08	1.40	2.04E-08	1.40

TABLE 2-66
(Continued)

LOCATION - PRESSURE VESSEL SIDE SURFACE, TANGENT PLANE
 RADIUS (IN) 27.89 STATION (IN) -37.00

Point 35

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-7039	2.57E-03	1.39	9.04E-02	1.39	9.08E-02	1.39	9.49E-02	1.39
NEUTRONS/CM2, E GT 1.0MEV	1.70E 12	3.10	5.89E 13	3.10	5.92E 13	3.10	6.19E 13	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MLV	4.12E 12	3.10	1.42E 14	3.10	1.40E 14	3.10	1.50E 14	3.10
NEUTRONS/CM2, E LT 0.4EV	7.08E 11	3.10	2.45E 13	3.10	2.46E 13	3.10	2.57E 13	3.10
GAMMA RAY DOSE, RADS (CAPD N)	5.63E 03	1.40	1.99E 05	1.40	2.00E 05	1.40	2.09E 05	1.40

LOCATION - OUTER RADIUS OF SUPPORT PLATE, AFT FACE OF SUPPORT PLATE
 RADIUS (IN) 22.21 STATION (IN) -3.00

Point 36

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (F)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	2.39E 00	1.07	8.71E 01	1.07	8.75E 01	1.07	9.16E 01	1.07
BTU/IN3-A-286	7.39E 00	1.07	2.68E 02	1.07	2.70E 02	1.07	2.82E 02	1.07
NEUTRONS/CM2, E GT 1.0MEV	2.04E 15	1.59	7.06E 16	1.59	7.10E 16	1.59	7.42E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.64E 16	1.59	5.69E 17	1.59	5.72E 17	1.59	5.98E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	2.20E 15	1.73	7.61E 16	1.73	7.65E 16	1.73	7.99E 16	1.73
GAMMA RAY DOSE, RADS (CAPD N)	5.32E 06	1.07	1.95E 08	1.07	1.96E 08	1.07	2.05E 08	1.07

TABLE 2-66
(Continued)

LOCATION - OUTER RADIUS OF SUPPORT PLATE, FORWARD FACE OF SUPPORT PLATE
RADIUS (IN) 22.21 STATION (IN) -9.50

Point 37

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	1.07E 00	1.07	3.79E 01	1.07	3.81E 01	1.07	3.99E 01	1.07
BTU/IN3-A-286	3.35E 00	1.07	1.19E 02	1.07	1.20E 02	1.07	1.25E 02	1.07
NEUTRONS/CM2, E GT 1.0MEV	7.45E 14	1.59	2.58E 16	1.59	2.59E 16	1.59	2.71E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.19E 15	1.59	2.14E 17	1.59	2.15E 17	1.59	2.25E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	9.66E 14	1.73	3.34E 16	1.73	3.35E 16	1.73	3.51E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	2.39E 06	1.07	8.52E 07	1.07	8.56E 07	1.07	8.95E 07	1.07

LOCATION - OUTER RADIUS OF BATH CENTRAL SHIELD, AFT FACE OF CENTRAL SHIELD PLATE
RADIUS (IN) 19.48 STATION (IN) -10.43

Point 38

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	1.25E 00	1.07	4.43E 01	1.07	4.45E 01	1.07	4.65E 01	1.07
BTU/IN3-A-286	3.49E 00	1.07	1.37E 02	1.07	1.38E 02	1.07	1.44E 02	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.03E 15	1.59	3.57E 16	1.59	3.59E 16	1.59	3.75E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.66E 15	1.59	2.65E 17	1.59	2.66E 17	1.59	2.78E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	7.10E 14	1.73	2.46E 16	1.73	2.47E 16	1.73	2.58E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	2.78E 06	1.07	9.86E 07	1.07	9.91E 07	1.07	1.04E 08	1.07

TABLE 2-66
(Continued)

LOCATION - OUTER RADIUS OF BATH CENTRAL SHIELD, AFT FACE OF BATH CENTRAL SHIELD
 RADIUS (IN) 19.48 STATION (IN) -11.83

Point 39

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	8.51E 00	1.64	2.95E 02	1.64	2.97E 02	1.64	3.10E 02	1.64
BTU/IN3-AL-0061	1.03E 00	1.07	3.64E 01	1.07	3.66E 01	1.07	3.83E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	4.12E 14	1.59	2.81E 16	1.59	2.82E 16	1.59	2.95E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.09E 15	1.59	2.11E 17	1.59	2.12E 17	1.59	2.21E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	5.24E 14	1.73	1.12E 16	1.73	1.13E 16	1.73	1.18E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	2.31E 06	1.07	8.20E 07	1.07	8.24E 07	1.07	8.61E 07	1.07

LOCATION - OUTER RADIUS OF BATH CENTRAL SHIELD, BATH CENTRAL SHIELD MIDPLANE
 RADIUS (IN) 19.48 STATION (IN) -17.38

Point 40

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	5.80E-01	1.42	2.02E 01	1.42	2.03E 01	1.42	2.12E 01	1.42
BTU/IN3-AL-0061	2.34E-01	1.07	8.23E 00	1.07	8.27E 00	1.07	8.64E 00	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.43E 14	1.59	4.94E 15	1.59	4.96E 15	1.59	5.19E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.44E 14	1.59	2.57E 16	1.59	2.59E 16	1.59	2.70E 16	1.59
NEUTRONS/CM2, E LT 0.4EV	2.53E 12	1.73	8.76E 13	1.73	8.80E 13	1.73	9.20E 13	1.73
GAMMA RAY DOSE, RADS (CARBON)	5.34E 05	1.07	1.88E 07	1.07	1.89E 07	1.07	1.98E 07	1.07

TABLE 2-66
(Continued)

LOCATION - OUTER RADIUS OF BATH CENTRAL SHIELD, FORWARD FACE OF BATH CENTRAL SHIELD
 RADIUS (IN) 19.48 STATION (IN) -27.93

Point 41

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	1.45E-01	1.47	5.03E 00	1.47	5.05E 00	1.47	5.28E 00	1.47
BTU/IN3-LEAD ALLOY	4.57E-01	1.07	1.60E 01	1.07	1.61E 01	1.07	1.68E 01	1.07
BTU/IN3-AL-6061	4.94E-02	1.37	1.73E 00	1.07	1.74E 00	1.07	1.82E 00	1.07
NEUTRONS/CM2, E GT 1.0MEV	2.84E 13	1.59	9.82E 14	1.59	9.87E 14	1.59	1.03E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.28E 14	1.59	4.42E 15	1.59	4.45E 15	1.59	4.65E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	3.11E 12	1.73	1.08E 14	1.73	1.08E 14	1.73	1.13E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.11E 05	1.77	3.90E 06	1.07	3.92E 06	1.07	4.09E 06	1.07

LOCATION - OUTER RADIUS OF BATH CENTRAL SHIELD, FORWARD FACE OF LEAD CENTRAL SHIELD
 RADIUS (IN) 19.48 STATION (IN) -24.23

Point 42

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-LEAD ALLOY	1.34E-01	1.07	4.70E 00	1.07	4.72E 00	1.07	4.94E 00	1.07
BTU/IN3-AL-6061	1.70E-02	1.07	5.96E-01	1.07	5.99E-01	1.07	6.26E-01	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.95E 13	1.59	6.75E 14	1.59	6.78E 14	1.59	7.09E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	9.22E 13	1.59	3.19E 15	1.59	3.21E 15	1.59	3.35E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	7.41E 12	1.73	2.57E 14	1.73	2.58E 14	1.73	2.70E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	3.65E 04	1.07	1.28E 06	1.07	1.29E 06	1.07	1.34E 06	1.07

TABLE 2-66
(Continued)

LOCATION - OUTER RADIUS OF BATH CENTRAL SHIELD, AFT FACE OF FLOW BAFFLE II
 RADIUS (IN) 19.48 STATION (IN) -29.63

Point 43

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	9.61E-03	1.40	3.35E-01	1.40	3.37E-01	1.40	3.52E-01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.20E 13	3.10	4.16E 14	3.10	4.18E 14	3.10	4.37E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.56E 13	3.10	1.92E 15	3.10	1.93E 15	3.10	2.02E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.02E 13	3.10	3.51E 14	3.10	3.53E 14	3.10	3.69E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.05E 04	1.40	7.18E 05	1.40	7.22E 05	1.40	7.54E 05	1.40

LOCATION - OUTER RADIUS OF BATH CENTRAL SHIELD, FORWARD FACE OF FLOW BAFFLE II
 RADIUS (IN) 19.48 STATION (IN) -30.75

Point 44

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-AL-6061	8.15E-03	1.40	2.84E-01	1.40	2.86E-01	1.40	2.98E-01	1.40
NEUTRONS/CM2, E GT 1.0MEV	8.84E 12	3.10	3.06E 14	3.10	3.07E 14	3.10	3.21E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.34E 13	3.10	1.50E 15	3.10	1.51E 15	3.10	1.58E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.04E 13	3.10	3.61E 14	3.10	3.62E 14	3.10	3.79E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	1.76E 04	1.40	6.16E 05	1.40	6.19E 05	1.40	6.47E 05	1.40

TABLE 2-66
(Continued)

LOCATION - I.R. OF BATH PERIPHERAL SHIELD 1, AFT FACE OF PERIPHERAL SHIELD
 RADIUS (IN) 20.50 STATION (IN) -11.04

Point 45

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-PATH	1.50E 01	1.68	5.20E 02	1.68	5.23E 02	1.68	5.47E 02	1.68
BTU/IN3-AL-6061	1.08E 00	1.07	3.82E 01	1.07	3.84E 01	1.07	4.01E 01	1.07
BTU/IN3-SS-304	3.34E 00	1.07	1.18E 02	1.07	1.18E 02	1.07	1.24E 02	1.07
NEUTRONS/CM2, E GT 1.0MEV	8.14E 14	1.59	2.82E 16	1.59	2.83E 16	1.59	2.96E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.31E 15	1.59	2.18E 17	1.59	2.20E 17	1.59	2.30E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	5.27E 14	1.73	1.82E 16	1.73	1.83E 16	1.73	1.92E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	2.42E 06	1.07	8.57E 07	1.07	8.57E 07	1.07	9.01E 07	1.07

LOCATION - I.R. OF BATH PERIPHERAL SHIELD 1, FOR. FACE OF BATH PER. SHIELD 1
 RADIUS (IN) 20.50 STATION (IN) -22.14

Point 46

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	2.17E-01	1.53	7.53E 00	1.53	7.57E 00	1.53	7.91E 00	1.53
BTU/IN3-LEAD ALLOY	5.56E-01	1.07	1.95E 01	1.07	1.96E 01	1.07	2.05E 01	1.07
BTU/IN3-AL-6061	5.77E-02	1.07	2.03E 00	1.07	2.04E 00	1.07	2.13E 00	1.07
BTU/IN3-SS-304	1.81E-01	1.07	6.36E 00	1.07	6.39E 00	1.07	6.68E 00	1.07
NEUTRONS/CM2, E GT 1.0MEV	3.32E 13	1.59	1.15E 15	1.59	1.16E 15	1.59	1.21E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.53E 14	1.59	5.31E 15	1.59	5.33E 15	1.59	5.57E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	2.30E 12	1.73	7.94E 13	1.73	7.99E 13	1.73	8.35E 13	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.30E 05	1.07	4.58E 06	1.07	4.60E 06	1.07	4.81E 06	1.07

TABLE 2-66
(Continued)

LOCATION - I.R. OF BATH PERIPHERAL SHIELD I, FOR. FACE OF LEAD PER. SHIELD I
 RADIUS (IN) 20.50 STATION (IN) -23.44

Point 47

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-LEAD ALLOY	2.34E-01	1.07	8.29E 00	1.07	8.24E 00	1.07	8.61E 00	1.07
BTU/IN3-AL-6061	2.87E-02	1.07	1.00E 00	1.07	1.01E 00	1.07	1.06E 00	1.07
BTU/IN3-SS-304	8.97E-02	1.07	3.14E 00	1.07	3.16E 00	1.07	3.20E 00	1.07
NEUTRONS/CM2, E GT 1.0MEV	2.34E 13	1.59	8.09E 14	1.59	8.13E 14	1.59	8.50E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.04E 14	1.59	3.75E 15	1.59	3.77E 15	1.59	3.94E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	5.61E 12	1.73	1.94E 14	1.73	1.95E 14	1.73	2.04E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	6.27E 04	1.07	2.20E 06	1.07	2.22E 06	1.07	2.32E 06	1.07

LOCATION - O.R. OF BATH PERIPHERAL SHIELD I, AFT FACE OF PERIPHERAL SHIELD
 RADIUS (IN) 22.04 STATION (IN) -11.04

Point 48

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	8.03E 00	1.64	2.79E 02	1.64	2.80E 02	1.64	2.93E 02	1.64
BTU/IN3-AL-6061	9.32E-01	1.07	3.30E 01	1.07	3.31E 01	1.07	3.46E 01	1.07
BTU/IN3-SS-304	2.89E 00	1.07	1.02E 02	1.07	1.03E 02	1.07	1.07E 02	1.07
NEUTRONS/CM2, E GT 1.0MEV	6.00E 14	1.59	2.08E 16	1.59	2.09E 16	1.59	2.18E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.83E 15	1.59	1.67E 17	1.59	1.68E 17	1.59	1.76E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	4.23E 14	1.73	1.46E 16	1.73	1.47E 16	1.73	1.54E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	2.11E 06	1.07	7.49E 07	1.07	7.52E 07	1.07	7.87E 07	1.07

TABLE 2-66
(Continued)

LOCATION - O.R. OF BATH PERIPHERAL SHIELD I, FOR. FACE OF BATH PER. SHIELD I
 RADIUS (IN) 22.04 STATION (IN) -22.14

Point 49

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	1.30E-01	1.45	4.51E 00	1.45	4.53E 00	1.45	4.74E 00	1.45
BTU/IN3-LEAD ALLOY	4.39E-01	1.07	1.54E 01	1.07	1.55E 01	1.07	1.62E 01	1.07
BTU/IN3-AL-6061	4.80E-02	1.07	1.69E 00	1.07	1.70E 00	1.07	1.77E 00	1.07
BTU/IN3-SS-304	1.51E-01	1.07	5.29E 00	1.07	5.32E 00	1.07	5.56E 00	1.07
NEUTRONS/CM2, E GT 1.0MEV	2.86E 13	1.59	9.88E 14	1.59	9.93E 14	1.59	1.04E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.16E 14	1.59	4.01E 15	1.59	4.03E 15	1.59	4.22E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.99E 12	1.73	6.84E 13	1.73	6.92E 13	1.73	7.23E 13	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.08E 05	1.07	3.80E 06	1.07	3.82E 06	1.07	3.99E 06	1.07

LOCATION - O.R. OF BATH PERIPHERAL SHIELD I, FOR. FACE OF LEAD PER. SHIELD I
 RADIUS (IN) 22.04 STATION (IN) -23.44

Point 50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-LEAD ALLOY	8.69E-02	1.07	3.05E 00	1.07	3.07E 00	1.07	3.20E 00	1.07
BTU/IN3-AL-6061	1.25E-02	1.07	4.39E-01	1.07	4.41E-01	1.07	4.61E-01	1.07
BTU/IN3-SS-304	3.70E-02	1.07	1.30E 00	1.07	1.31E 00	1.07	1.37E 00	1.07
NEUTRONS/CM2, E GT 1.0MEV	2.10E 13	1.59	7.26E 14	1.59	7.30E 14	1.59	7.63E 14	1.59
NEUTRONS/CM2, 0.4EV LT L LT 1.0MEV	9.66E 13	1.59	3.00E 15	1.59	3.01E 15	1.59	3.15E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	5.72E 12	1.73	1.98E 14	1.73	1.99E 14	1.73	2.08E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	2.60E 04	1.07	9.14E 05	1.07	9.21E 05	1.07	9.62E 05	1.07

TABLE 2-66
(Continued)

LOCATION - O.R. OF BATH PERIPHERAL SHIELD II, AFT FACE OF PERIPHERAL SHIELD
 RADIUS (IN) 23.63 STATION (IN) -11.04

Point 51

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	7.36E 00	1.64	7.56E 02	1.64	2.57E 02	1.64	2.68E 02	1.64
BTU/IN3-AL-6061	8.76E-01	1.07	3.11E 01	1.07	3.12E 01	1.07	3.27E 01	1.07
BTU/IN3-SS-304	2.72E 00	1.07	9.63E 01	1.07	9.64E 01	1.07	1.01E 02	1.07
NEUTRONS/CM2, E GT 1.0MEV	5.46E 14	1.59	1.89E 16	1.59	1.90E 16	1.59	1.99E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.18E 15	1.59	1.45E 17	1.59	1.45E 17	1.59	1.52E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	3.94E 14	1.73	1.36E 16	1.73	1.37E 16	1.73	1.43E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.99E 06	1.07	7.07E 07	1.07	7.11E 07	1.07	7.43E 07	1.07

LOCATION - O.R. OF BATH PERIPHERAL SHIELD II, FOR. FACE OF BATH PER. SHIELD II
 RADIUS (IN) 23.63 STATION (IN) -22.15

Point 52

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	1.02E-01	1.45	3.56E 00	1.45	3.4E 00	1.45	3.74E 00	1.45
BTU/IN3-LEAD ALLOY	3.38E-01	1.07	1.19E 01	1.07	1.4E 01	1.07	1.74E 01	1.07
BTU/IN3-AL-6061	3.76E-02	1.07	1.32E 00	1.07	1.43E 00	1.07	1.79E 00	1.07
BTU/IN3-SS-304	1.18E-01	1.07	4.16E 00	1.07	4.19E 00	1.07	4.4E 00	1.07
NEUTRONS/CM2, E GT 1.0MEV	2.18E 13	1.59	7.46E 14	1.59	7.7E 14	1.59	7.94E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.36E 13	1.59	2.89E 15	1.59	2.1E 15	1.59	3.24E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.56E 12	1.73	5.40E 13	1.73	5.12E 13	1.73	5.77E 13	1.73
GAMMA RAY DOSE, RADS (CARBON)	8.45E 04	1.07	2.98E 05	1.07	2.77E 05	1.07	3.13E 05	1.07

TABLE 2-66
(Continued)

LOCATION - O.R. OF BATH PERIPHERAL SHIELD II, FOR. FACE OF LEAD PER. SHIELD II
 RADIUS (IN) 23.63 STATION (IN) -23.44

Point 53

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-LEAD ALLOY	5.77E-02	1.07	2.02E-00	1.07	2.03E-00	1.07	2.13E-00	1.07
BTU/IN3-AL-6061	8.44E-03	1.07	2.96E-01	1.07	2.99E-01	1.07	3.11E-01	1.07
BTU/IN3-SS-304	2.48E-02	1.07	8.69E-01	1.07	8.73E-01	1.07	9.13E-01	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.59E 13	1.59	5.48E 14	1.59	5.51E 14	1.59	5.76E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.40E 13	1.59	2.22E 15	1.59	2.23E 15	1.59	2.33E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	4.85E 12	1.73	1.68E 14	1.73	1.69E 14	1.73	1.76E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.73E 04	1.07	6.08E 05	1.07	6.11E 05	1.07	6.39E 05	1.07

LOCATION - O.R. OF BATH PERIPHERAL SHIELD III, AFT FACE OF PERIPHERAL SHIELD
 RADIUS (IN) 25.38 STATION (IN) -11.04

Point 54

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	6.19E 00	1.63	2.15E 02	1.63	2.16E 02	1.63	2.46E 02	1.63
BTU/IN3-AL-6061	7.35E-01	1.07	2.85E 01	1.07	2.86E 01	1.07	2.99E 01	1.07
BTU/IN3-SS-304	2.45E 00	1.07	8.77E 01	1.07	8.81E 01	1.07	9.21E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	5.46E 14	1.59	1.89E 16	1.59	1.90E 16	1.59	1.98E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.44E 15	1.59	1.20E 17	1.59	1.21E 17	1.59	1.26E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	3.10E 14	1.73	1.07E 16	1.73	1.08E 16	1.73	1.13E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.80E 06	1.07	6.47E 07	1.07	6.50E 07	1.07	6.80E 07	1.07

TABLE 2-66
(Continued)

LOCATION - O.R. OF PATH PERIPHERAL SHIELD III, FOR. FACE OF BATH PER. SHIELD III
 RADIUS (IN) 25.38 STATION (IN) -22.14

Point 55

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-BATH	6.96E-02	1.47	2.42E 00	1.47	2.43E 00	1.47	2.54E 00	1.47
BTU/IN3-LEAD ALLOY	2.00E-01	1.07	7.01E 00	1.07	7.04E 00	1.07	7.36E 00	1.07
BTU/IN3-AL-6061	2.38E-02	1.07	8.37E-01	1.07	8.41E-01	1.07	8.79E-01	1.07
BTU/IN3-SS-304	7.51E-02	1.07	2.64E 00	1.07	2.65E 00	1.07	2.77E 00	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.57E 13	1.59	5.43E 14	1.59	5.46E 14	1.59	5.71E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.29E 13	1.59	1.83E 15	1.59	1.84E 15	1.59	1.92E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.57E 12	1.73	5.42E 13	1.73	5.45E 13	1.73	5.69E 13	1.73
GAMMA RAY DOSE, RADS (CARBON)	5.28E 04	1.07	1.86E 06	1.07	1.87E 06	1.07	1.95E 06	1.07

LOCATION - O.R. OF BATH PERIPHERAL SHIELD III, FOR. FACE OF LEAD PER. SHIELD III
 RADIUS (IN) 25.38 STATION (IN) -23.44

Point 56

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

BTU/IN3-LEAD ALLOY	6.13E-02	1.07	2.15E 00	1.07	2.16E 00	1.07	2.26E 00	1.07
BTU/IN3-AL-6061	8.39E-03	1.07	2.95E-01	1.07	2.96E-01	1.07	3.10E-01	1.07
BTU/IN3-SS-304	2.56E-02	1.07	8.97E-01	1.07	9.04E-01	1.07	9.45E-01	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.13E 13	1.59	3.91E 14	1.59	3.93E 14	1.59	4.11E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.19E 13	1.59	1.45E 15	1.59	1.46E 15	1.59	1.52E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	3.84E 12	1.73	1.33E 14	1.73	1.33E 14	1.73	1.39E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.77E 04	1.07	6.25E 05	1.07	6.28E 05	1.07	6.56E 05	1.07

TABLE 2-66
(Continued)

LOCATION - O.R. OF BATH PERIPHERAL SHIELD IV, AFT FACE OF PERIPHERAL SHIELD
 RADIUS (IN) 27.00 STATION (IN) -11.4

Point 57

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-BATH	5.14E 00	1.63	1.79E 02	1.63	1.40E 02	1.63	1.88E 02	1.63
BTU/IN3-AL-6061	6.56E-01	1.07	2.37E 01	1.07	2.38E 01	1.07	2.49E 01	1.07
BTU/IN3-SS-304	2.03E 00	1.07	7.31E 01	1.07	7.45E 01	1.07	7.68E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	4.70E 14	1.59	1.63E 16	1.59	1.93E 16	1.59	1.71E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.33E 15	1.59	8.06E 16	1.59	8.11E 16	1.59	8.47E 16	1.59
NEUTRONS/CM2, E LT 0.4EV	2.12E 14	1.73	7.33E 15	1.73	7.37E 15	1.73	7.70E 15	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.48E 06	1.07	5.37E 07	1.07	5.40E 07	1.07	5.65E 07	1.07

LOCATION - O.R. OF BATH PERIPHERAL SHIELD IV, FOR. FACE OF BATH PER. SHIELD IV
 RADIUS (IN) 27.00 STATION (IN) -20.64

Point 58

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-BATH	7.01E-02	1.38	2.45E 00	1.38	2.44E 00	1.38	2.57E 00	1.38
BTU/IN3-LEAD ALLOY	2.83E-01	1.07	9.98E 00	1.07	1.00E 01	1.07	1.04E 01	1.07
BTU/IN3-AL-6061	3.24E-02	1.07	1.14E 00	1.07	1.15E 00	1.07	1.20E 00	1.07
BTU/IN3-SS-304	1.03E-01	1.07	3.64E 00	1.07	3.65E 00	1.07	3.82E 00	1.07
NEUTRONS/CM2, E LT 1.0MEV	1.77E 13	1.59	6.11E 14	1.59	6.14E 14	1.59	6.42E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.62E 13	1.59	1.60E 15	1.59	1.61E 15	1.59	1.68E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	4.87E 11	1.73	1.68E 13	1.73	1.69E 13	1.73	1.77E 13	1.73
GAMMA RAY DOSE, RADS (CARBON)	7.26E 04	1.07	2.56E 06	1.07	2.58E 06	1.07	2.69E 06	1.07

TABLE 2-66
(Continued)

LOCATION - O.R. OF BATH PERIPHERAL SHIELD IV, FOR. FACE OF LEAD PER. SHIELD IV
 RADIUS (IN) 27.00 STATION (IN) -21.94

Point 59

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-LEAD ALLOY	7.51E-02	1.07	2.64E 00	1.07	2.66E 00	1.07	2.78E 00	1.07
BTU/IN3-AL-6061	9.66E-03	1.07	3.40E-01	1.07	3.42E-01	1.07	3.57E-01	1.07
BTU/IN3-SS-304	2.96E-02	1.07	1.04E 00	1.07	1.05E 00	1.07	1.09E 00	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.24E 13	1.59	4.29E 14	1.59	4.31E 14	1.59	4.51E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.49E 13	1.59	1.21E 15	1.59	1.21E 15	1.59	1.27E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.53E 12	1.73	5.30E 13	1.73	5.33E 13	1.73	5.57E 13	1.73
GAMMA RAY DOSE, RADS (CARBON)	2.07E 04	1.07	7.30E 05	1.07	7.34E 05	1.07	7.67E 05	1.07

LOCATION - REACTOR CENTERLINE, FORWARD FACE OF UNFUELED TIPS
 RADIUS (IN) 0.0 STATION (IN) 51.63

Point 60

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-A-286	2.68E 01	1.12	1.04E 03	1.12	1.05E 03	1.12	1.10E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.08E 16	1.02	3.73E 17	1.02	3.75E 17	1.02	3.92E 17	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.02E 16	1.02	1.74E 18	1.02	1.75E 18	1.02	1.83E 18	1.02
NEUTRONS/CM2, E LT 0.4EV	4.85E 13	1.29	1.68E 15	1.29	1.69E 15	1.29	1.76E 15	1.29
GAMMA RAY DOSE, RADS (CARBON)	2.08E 07	1.12	8.12E 08	1.12	8.16E 08	1.12	9.56E 08	1.12

TABLE 2-66
(Continued)

LOCATION - CORE EDGE, FORWARD FACE OF UNFUELED TIPS
 RADIUS (IN) 18.66 STATION (IN) 51.63

Point 61

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
BTU/IN3-A-286	1.12E 01	1.12	4.34E 02	1.12	4.36E 02	1.12	4.57E 02	1.12
NEUTRONS/CM2, E GT 1.0MEV	4.84E 15	1.02	1.67E 17	1.02	1.68E 17	1.02	1.76E 17	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.31E 16	1.02	7.98E 17	1.02	8.02E 17	1.02	8.38E 17	1.02
NEUTRONS/CM2, E LT 0.4EV	5.33E 14	1.29	1.84E 16	1.29	1.85E 16	1.29	1.94E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	8.60E 06	1.12	3.35E 08	1.12	3.37E 08	1.12	3.54E 08	1.12

TABLE 2-67

GAMMA RAY SOURCE TERMS FOR GRAPHITE CORE AT TRANSIENT STATE POINTS

Gamma Group	Energy (MeV)	Source (MeV/sec) at State Points				
		B	C	D	E	F
1	7.5 - 10.	2.36(18)	2.28(18)	1.44(18)	1.47(18)	0
2	7.0 - 7.5	4.77(18)	4.62(18)	2.92(18)	2.97(18)	0
3	6.0 - 7.0	1.34(19)	1.30(19)	8.19(18)	8.33(18)	0
4	5.0 - 6.0	1.58(19)	1.64(19)	1.09(19)	1.08(19)	1.25(18)
5	4.0 - 5.0	3.16(19)	3.62(19)	2.58(19)	2.47(19)	6.70(18)
6	3.0 - 4.0	6.46(19)	7.67(19)	5.60(19)	5.29(19)	1.68(19)
7	2.6 - 3.0	3.25(19)	4.04(19)	3.03(19)	2.83(19)	1.06(19)
8	2.2 - 2.6	5.93(19)	6.95(19)	5.03(19)	4.78(19)	1.43(19)
9	1.8 - 2.2	6.93(19)	8.13(19)	5.89(19)	5.59(19)	1.68(19)
10	1.35 - 1.8	8.71(19)	1.05(20)	7.72(19)	7.27(19)	2.44(19)
11	0.9 - 1.35	1.19(20)	1.39(20)	1.01(20)	9.60(19)	2.89(19)
12	0.4 - 0.9	1.62(20)	1.84(20)	1.31(20)	1.26(20)	3.30(19)
13	0.0 - 0.4	8.44(19)	8.82(19)	5.92(19)	5.84(19)	7.63(18)
	TOTAL	7.46(20)	8.57(20)	6.13(20)	5.86(20)	1.60(20)

NOTE: Numbers in parentheses indicate powers of ten.

086-10/14/70

TABLE 2-68
AFTER SHUTDOWN GAMMA DOSE RATE 10 FEET FROM CORE CENTER
FOR R-1 REFERENCE DESIGN (939J723) WITH A COMPOSITE CORE
(REACTOR RUN = 60 MIN. SHUTDOWN TIME = 30 MIN.)

Source Region *	On Reactor Axis		On Core Midplane	
	(Rad-C/hr)	% of Total	(Rad-C/hr)	% of Total
Core	4.12 (3)	87.13	1.34 (6)	83.25
Core Periphery	7.41 (-2)	.0016	4.06 (2)	.0252
Structure	2.87 (0)	.0607	6.63 (3)	.411
Reflector	7.14 (1)	1.51	2.50 (5)	15.53
Core Plenum	4.85 (1)	1.03	1.23 (3)	.0764
Lateral Support Forward	9.31 (0)	.197	1.19 (3)	.0739
Support Plate	2.58 (2)	5.46	5.36 (3)	.333
Forward Reflector Hardware	4.96 (1)	1.05	4.13 (3)	.256
Support Plate Plenum	6.10 (1)	1.29	3.96 (2)	.0246
Instrumentation Ring	3.54 (0)	.0748	1.25 (2)	.00777
Aft Central Shield Plate	8.00 (0)	.169	4.43 (1)	.00275
BATH Central Shield	1.26 (1)	.266	1.01 (1)	.00063
Lead Central Shield	4.70 (-2)	.0010	5.59 (-5)	<.00001
BATH Peripheral Shield-1	1.71 (0)	.0361	4.68 (0)	.00029
BATH Peripheral Shield-2	1.09 (0)	.0231	4.43 (0)	.00028
BATH Peripheral Shield-3	2.79 (0)	.0590	2.16 (1)	.00134
BATH Peripheral Shield-4	5.06 (-1)	.0107	7.94 (0)	.00094
Lead Peripheral Shield-1	1.46 (-3)	.00003	6.93 (-5)	<.00001
Lead Peripheral Shield-2	1.60 (-3)	.00003	6.67 (-6)	<.00001
Lead Peripheral Shield-3	1.81 (0)	.038	8.92 (-2)	<.00001
Lead Peripheral Shield-4	5.19 (-2)	.0011	7.76 (-3)	<.00001
Peripheral Shield Plate	5.58 (-0)	.118	2.30 (-1)	.00001
Shield Plenum	3.45 (0)	.0739	3.89 (-2)	<.00001
Flow Baffle - Side	1.12 (1)	.237	1.35 (1)	.00083
Flow Baffle - Dome	4.25 (1)	.899	1.42 (0)	.00009
Dome Plenum	1.31 (1)	.277	1.88 (0)	.00012
TOTAL	4.73 (3)	100.0	1.61 (6)	100.0

* Refer to Figure 2-13 and Table 2-36 for region location.

NOTE: Numbers in Parentheses refer to powers of ten.

474 - 10/16/70

670 - 12/17/70

162 - 12/18/70

TABLE 2-69

AFTER SHUTDOWN GAMMA DOSE RATE 10 FEET FROM CORE CENTER
 FOR R-1 REFERENCE DESIGN (939J723) WITH A COMPOSITE CORE
 (REACTOR RUN = 60 MIN. SHUTDOWN TIME = 1 MONTH)

Source Region *	On Reactor Axis		On Core Midplane	
	(Rad-C/hr)	% of Total	(Rad-C/hr)	% of Total
Core	2.04 (0)	99.53	3.81 (2)	98.86
Core Periphery	1.69 (-6)	.00008	1.00 (0)	.259
Structure	1.41 (-5)	.00069	2.84 (-1)	.0737
Reflector	5.97 (-5)	.00291	2.25 (0)	.584
Core Plenum	5.20 (-4)	.0254	1.20 (-1)	.0310
Lateral Support Forward	9.35 (-5)	.00456	1.79 (-1)	.0464
Support Plate	1.86 (-3)	.0907	2.73 (-1)	.0708
Forward Reflector Hardware	6.26 (-4)	.0305	2.65 (-1)	.0688
Support Plate Plenum	6.93 (-4)	.0338	1.75(-2)	.00454
Instrumentation Ring	4.05 (-5)	.00198	5.67 (-3)	.00147
Aft Central Shield Plate	4.81 (-5)	.00235	3.05 (-4)	.00008
BATH Central Shield	6.33 (-4)	.0309	4.03 (-4)	.00010
Lead Central Shield	7.63 (-4)	.0372	1.57 (-6)	<.00001
BATH Peripheral Shield-1	5.71 (-5)	.00279	1.73 (-4)	.00004
BATH Peripheral Shield-2	4.89 (-5)	.00239	2.29 (-4)	.00006
BATH Peripheral Shield-3	1.59 (-4)	.00776	4.20 (-3)	.00109
BATH Peripheral Shield-4	2.06 (-5)	.00101	3.96 (-4)	.00010
Lead Peripheral Shield-1	4.03 (-5)	.00197	2.26 (-6)	<.00001
Lead Peripheral Shield-2	9.47 (-5)	.00462	1.66 (-6)	<.00001
Lead Peripheral Shield-3	3.48 (-4)	.0170	1.36 (-5)	<.00001
Lead Peripheral Shield-4	6.21 (-5)	.00303	9.59 (-6)	<.00001
Peripheral Shield Plate	3.94 (-4)	.0192	1.43 (-5)	<.00001
Shield Plenum	7.76 (-5)	.00379	7.36 (-7)	<.00001
Flow Baffle - Side	7.92 (-5)	.00386	8.71 (-5)	.00002
Flow Baffle - Dome	2.02 (-3)	.0985	5.95 (-5)	.00002
Dome Plenum	9.32 (-4)	.0455	1.31 (-4)	.00003
TOTAL	2.05 (0)	100.0	3.85 (2)	100.0

* Refer to Figure 2-13 and Table 2-36 for region location.

474 - 10/16/70

NOTE: Numbers in Parentheses refer to powers of ten.

161 - 12/18/70

240 - 12/18/70

TABLE 2-70
AFTER SHUTDOWN GAMMA DOSE RATE 10 FEET FROM CORE CENTER
FOR R-1 REFERENCE DESIGN (939J723) WITH A COMPOSITE CORE
(REACTOR RUN = 60 MIN. SHUTDOWN TIME = 1 YEAR)

Source Region *	On Reactor Axis		On Core Midplane	
	(Rad-C/hr)	% of Total	(Rad-C/hr)	% of Total
Core	1.06 (-3)	58.11	3.10 (0)	92.56
Core Periphery	8.66 (-8)	.00475	5.76 (-2)	1.72
Structure	3.04 (-6)	.166	2.06 (-2)	.615
Reflector	8.77 (-6)	.481	1.25 (-1)	3.73
Core Plenum	5.62 (-5)	3.08	7.60 (-3)	.227
Lateral Support Forward	1.60 (-5)	.877	1.00 (-2)	.299
Support Plate	2.23 (-4)	12.23	1.52 (-2)	.454
Forward Reflector Hardware	5.00 (-5)	2.74	1.17 (-2)	.349
Support Plate Plenum	3.65 (-5)	2.00	8.73 (-4)	.0261
Instrumentation Ring	4.00 (-6)	.219	3.13 (-4)	.00935
Aft Central Shield Plate	1.52 (-5)	.833	8.04 (-5)	.00240
BATH Central Shield	4.03 (-5)	2.21	2.57 (-5)	.000767
Lead Central Shield	4.33 (-5)	2.37	6.25 (-8)	<.00001
BATH Peripheral Shield-1	4.44 (-6)	.243	1.28 (-5)	.000382
BATH Peripheral Shield-2	3.04 (-6)	.167	1.43 (-5)	.000427
BATH Peripheral Shield-3	1.44 (-5)	.789	2.43 (-4)	.00726
BATH Peripheral Shield-4	1.48 (-6)	.0811	2.68 (-5)	.00080
Lead Peripheral Shield-1	2.13 (-6)	.117	1.08 (-7)	<.00001
Lead Peripheral Shield-2	5.43 (-6)	.298	7.35 (-8)	<.00001
Lead Peripheral Shield-3	1.98 (-5)	1.09	7.86 (-7)	.00002
Lead Peripheral Shield-4	3.66 (-6)	.201	5.42 (-7)	.00002
Peripheral Shield Plate	2.49 (-5)	1.37	9.52 (-7)	.00003
Shield Plenum	6.71 (-6)	.368	7.50 (-8)	<.00001
Flow Baffle - Side	2.02 (-5)	1.11	2.37 (-5)	.00071
Flow Baffle - Dome	1.17 (-4)	6.41	3.70 (-6)	.00011
Dome Plenum	4.45 (-5)	2.44	6.24 (-6)	.00019
TOTAL	1.82 (-3)	100.0	3.35 (0)	100.0

* Refer to Figure 2-13 and Table 2-36 for region location.

NOTE: Numbers in Parentheses refer to powers of ten.

474 - 10/16/70

163 - 12/18/70

241 - 12/18/70

Figure 2-1 Frequency Distribution of Possible Radiation Levels

Figure 2-2 Flow Chart for Nuclear, Thermal, and Structural Analysis

Figure 2-3. Nuclear and Radiation Design NSS Analysis (Block 11)

Figure 2-4 Flow Chart to Calculate Neutron and Photon Environment and Heating Rates Internal to the Nuclear Subsystem (Block 11-7, 11-8, and 11-9)

Figure 2-5. Flow Chart to Calculate Neutron and Photon Environment External to the Nuclear Subsystem (Block 11-10)

Figure 2-6. Radiation Environment Drawing

THIS PAGE INTENTIONALLY BLANK

Figure 2-7. Basic Reactor Geometry Model for the Neutron DOT Problem

Figure 2-8. Shield Geometry Model for the Neutron DOT Problem

Figure 2-9. Basic Reactor Geometry Model for the Photon Dot Problem

Figure 2-10. Shield Geometry Model for the Photon DOT Problem

Figure 2-11. Heating Rate Model for the Composite Core

Figure 2-12. Heating Rate Model for the Graphite Core

Figure 2-13. Common Radiation Analysis Model for the NSS Employing Composite Fuel

Figure 2-14. Common Radiation Analysis Model for the NSS Employing Graphite Fuel

Figure 2-15 Gamma Ray Dose Rate as a Function of Axial Location on the Reactor Centerline (Composite Core)

Figure 2-14 Gamma Ray Dose Rate as a Function of Axial Location at the Core Mid-Radius (Composite Core)

Figure 2-17. Gamma Ray Dose Rate as a Function of Axial Location at the Core Edge (Composite Core)

Figure 2-18 Gamma Ray Dose Rate as a Function of Axial Location at the Outer Radius of the Core Periphery (Composite Core)

ASTRONUCLEAR
Laboratory

Figure 2-20. Gamma Ray Dose Rate as a Function of Axial Location at the Inner Radius of the Reflector (Composite Core)

Figure 2-21. Gamma Ray Dose Rate as a Function of Axial Location on the Control Drum Centerline (Composite Core)

Figure 2-22 Gamma Ray Dose Rate as a Function of Axial Location at the Outer Radius of the Reflector (Composite Core)

Figure 2-23. Gamma Ray Dose Rate as a Function of Axial Location at the Pressure Vessel Side Surface (Composite Core)

Figure 2-24. Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the CHESH (Composite Core)

Figure 2-25 Gamma Ray Dose Rate as a Function of Radial Location at the Core Exit
(Composite Core)

Figure 2-26. Gamma Ray Dose Rate as a Function of Radial Location at the Core Midplane (Composite Core)

Figure 2-27. Gamma Ray Dose Rate as a Function of Radial Location at the Core Inlet (Composite Core)

Figure 2-28 Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the Support Plate (Composite Core)

Figure 2-29. Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the Support Plate (Composite Core)

Figure 2-30. Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the Shield Plate (Composite Core)

Figure 2-31. Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the BATH Central Shield (Composite Core)

Figure 2-32. Gamma Ray Dose Rate as a Function of Radial Location at the BATH Central Shield Midplane (Composite Core)

Figure 2-34. Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Composite Core)

Figure 2-35. Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the Flow Baffle (Composite Core)

Figure 2-36. Gamma Ray Dose Rate as a Function of Radial Location at the Pressure Vessel Dome (Composite Core)

Figure 2-37. Fast Neutron Flux as a Function of Axial Location on the Reactor Centerline (Composite Core)

Figure 2-38. Fast Neutron Flux as a Function of Axial Location at the Core Mid-Radius (Composite Core)

Figure 2-39. Fast Neutron Flux as a Function of Axial Location at the Core Edge
(Composite Core)

Figure 2-40. Fast Neutron Flux as a Function of Axial Location at the Outer Radius of the Core Periphery (Composite Core)

Figure 2-41. Fast Neutron Flux as a Function of Axial Location at the Outer Radius of the Lateral Support (Composite Core)

Figure 2-42. Fast Neutron Flux as a Function of Axial Location at the Inner Radius of the Reflector (Composite Core)

Figure 2-43. Fast Neutron Flux as a Function of Axial Location on the Control Drum Centerline (Composite Core)

Figure 2-44. Fast Neutron Flux as a Function of Axial Location at the Outer Radius of the Reflector (Composite Core)

Figure 2-45. Fast Neutron Flux as a Function of Axial Location at the Pressure Vessel Side Surface (Composite Core)

Figure 2-46. Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the CHESH (Composite Core)

Figure 2-48. Fast Neutron Flux as a Function of Radial Location at the Core Midplane (Composite Core)

Figure 2-49. Fast Neutron Flux as a Function of Radial Location at the Core Inlet (Composite Core)

Figure 2-50. Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the Support Plate (Composite Core)

Figure 2-51. Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the Support Plate (Composite Core)

Figure 2-52. Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the Shield Plate (Composite Core)

Figure 2-53. Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the BATH Central Shield (Composite Core)

Figure 2-54. Fast Neutron Flux as a Function of Radial Location at the BATH Central Shield Midplane (Composite Core)

Figure 2-55. Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the BATH Central Shield (Composite Core)

Figure 2-56. Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Composite Core)

Figure 2-57. Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the Flow Baffle (Composite Core)

Figure 2-58. Fast Neutron Flux as a Function of Radial Location at the Pressure Vessel Dome
(Composite Core)

Figure 2-59. Intermediate Neutron Flux as a Function of Axial Location on the Reactor Centerline (Composite Core)

Figure 2-60 Intermediate Neutron Flux as a Function of Axial Location at the Core Mid-Radius (Composite Core)

Figure 2-61. Intermediate Neutron Flux as a Function of Axial Location at the Core Edge (Composite Core)

Figure 2-62. Intermediate Neutron Flux as a Function of Axial Location at the Outer Radius of the Core Periphery (Composite Core)

Figure 2-63. Intermediate Neutron Flux as a Function of Axial Location at the Outer Radius of the Lateral Support (Composite Core)

Figure 2-64. Intermediate Neutron Flux as a Function of Axial Location at the Inner Radius of the Reflector (Composite Core)

Figure 2-65. Intermediate Neutron Flux as a Function of Axial Location on the Control Drum Centerline (Composite Core)

Figure 2-66. Intermediate Neutron Flux as a Function of Axial Location at the Outer Radius of the Reflector (Composite Core)

Figure 2-67. Intermediate Neutron Flux as a Function of Axial Location at the Pressure Vessel Side Surface (Composite Core)

Figure 2-68. Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the CHESH (Composite Core)

Figure 2-69. Intermediate Neutron Flux as a Function of Radial Location at the Core Exit
(Composite Core)

Figure 2-70 Intermediate Neutron Flux as a Function of Radial Location at the Core Midplane (Composite Core)

Figure 2-71. Intermediate Neutron Flux as a Function of Radial Location at the Core Inlet (Composite Core)

Figure 2-72. Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the Support Plate (Composite Core)

Figure 2-73. Intermediate Neutron Flux as a Function of Radial Location at the Forward Surface of the Support Plate (Composite Core)

Figure 2-74 Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the Shield Plate (Composite Core)

Figure 2-75 Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the BATH Central Shield (Composite Core)

Figure 2-76 Intermediate Neutron Flux as a Function of Radial Location at the BATH Central Shield Midplane (Composite Core)

**Astronuclear
Laboratory**

Figure 2-78 Intermediate Neutron Flux as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Composite Core)

Figure 2-79. Intermediate Neutron Flux as a Function of Radial Location at the Forward Surface of the Flow Baffle (Composite Core)

Figure 2-80 Intermediate Neutron Flux as a Function of Radial Location at the Pressure Vessel Dome (Composite Core)

Figure 2-81. Thermal Neutron Flux as a Function of Axial Location on the Reactor Centerline (Composite Core)

Figure 2-82 Thermal Neutron Flux as a Function of Axial Location at the Core Mid-Radius (Composite Core)

Figure 2-83 Thermal Neutron Flux as a Function of Axial Location at the Core Edge (Composite Core)

Figure 2-84 Thermal Neutron Flux as a Function of Axial Location at the Outer Radius of the Core Periphery (Composite Core)

Figure 2-85. Thermal Neutron Flux as a Function of Axial Location at the Outer Radius of the Lateral Support (Composite Core)

Figure 2-86 Thermal Neutron Flux as a Function of Axial Location at the Inner Radius of the Reflector (Composite Core)

Figure 2-87. Thermal Neutron Flux as a Function of Axial Location on the Control Drum Centerline (Composite Core)

Figure 2-88 Thermal Neutron Flux as a Function of Axial Location at the Outer Radius of the Reflector (Composite Core)

Figure 2-89. Thermal Neutron Flux as a Function of Axial Location at the Pressure Vessel Side Surface (Composite Core)

Figure 2-90 Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the CHESH (Composite Core)

Figure 2-91. Thermal Neutron Flux as a Function of Radial Location at the Core Exit
(Composite Core)

Figure 2-92. Thermal Neutron Flux as a Function of Radial Location at the Core Midplane (Composite Core)

Figure 2-93. Thermal Neutron Flux as a Function of Radial Location at the Core Inlet (Composite Core)

Figure 2-94. Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the Support Plate (Composite Core)

Figure 2-95 Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the Support Plate (Composite Core)

Figure 2-96 Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the Shield Plate (Composite Core)

Figure 2-97. Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the BATH Central Shield (Composite Core)

Figure 2-98. Thermal Neutron Flux as a Function of Radial Location at the BATH Central Shield Midplane (Composite Core)

Figure 2-99. Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the BATH Central Shield (Composite Core)

Figure 2-100. Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Composite Core)

Figure 2-101. Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the Flow Baffle (Composite Core)

Figure 2-102. Thermal Neutron Flux as a Function of Radial Location at the Pressure Vessel (Composite Core)

Figure 2-103. Average 155 Group Neutron Flux versus Neutron Energy in the Composite Core

Figure 2-104. Average 155 Group Neutron Flux versus Neutron Energy in the Outer Reflector Region

Figure 2-105. Average 155 Group Neutron Flux versus Neutron Energy in the BATH Central Shield

Figure 2-106. Average 155 Group Neutron Flux versus Neutron Energy in the Pressure Vessel Dome

Figure 2-107. Average 155 Group Neutron Flux versus Neutron Energy in the Side Pressure Vessel

Figure 2-108. Photon Energy Flux as a Function of Photon Energy at the Center of the Composite Core

Figure 2-109. Photon Energy Flux as a Function of Photon Energy at the Inner Radius of the Reflector on the Core Midplane

Figure 2-110. Photon Energy Flux as a Function of Photon Energy at the Outer Radius of the Pressure Vessel on the Core Midplane

Figure 2-111. Photon Energy Flux as a Function of Photon Energy at the Aft Face of the BATH Shield on the Centerline

Figure 2-112. Photon Energy Flux as a Function of Photon Energy at the Surface of the Pressure Vessel Dome on the Centerline

Figure 2-113. Photon Heating Rate in Aluminum 6061 ($\rho = 2.71 \text{ gm/cc}$) as a Function of Axial Location on the Reactor Centerline

Figure 2-14. Photon Heating Rate in Stainless Steel 304 ($\rho = 7.92 \text{ gm/cc}$) as a Function of Axial Location on the Reactor Centerline

Figure 2-115. Photon Heating Rate in Aluminum 6061 ($\rho = 2.71 \text{ gm/cc}$) as a Function of Axial Location on the Reactor Centerline

Figure 2-13. Photon Heating Rate in Stainless Steel 304 ($\rho = 7.92 \text{ gm/cc}$) as a Function of Axial Location on the Reactor Centerline

Figure 2-117. Relative Radial Distribution of Photon Source in Region 7, Pressure Vessel Side - 1

Figure 2-118 Relative Axial Distribution of Photon Source in Region 7, Pressure Vessel Side - 1

Figure 2-119. Relative Radial Distribution of Photon Source in Region 34, Pressure Vessel Dome

Figure 2-120. Relative Axial Distribution of Photon Source in Region 34, Pressure Vessel Dome

Figure 2-121. Relative Radial Distribution of Photon Source in Region 35, Pressure Vessel Side - 2

Figure 2-122. Relative Axial Distribution of Photon Source in Region 35, Pressure Vessel, Side-2

W59835-1

Figure 2-123. Photon Kerma Rate on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design (939J723) With a Composite Core

Figure 2-124. Fast ($E > 1.0$ Mev) Neutron Flux on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design (939J723) With a Composite Core

W50035-1

Figure 2-125. Intermediate ($0.4 \text{ ev} < E < 1.0 \text{ Mev}$) Neutron Flux on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design (939J723) With a Composite Core

Figure 2-126. Thermal ($E < 0.4$ eV) Neutron Flux on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design (939J723) With a Composite Core

W59035-1

Figure 2-127. Gamma Ray Kerma Rate Vs. Polar Angle for Meridian Rings of Various Radii for the R-1 Reference Design (939J723)

Figure 2-128. Fast ($E > 1.0 \text{ Mev}$) Neutron Flux Vs. Polar Angle For Meridian Rings of Various Radii For the R-1 Reference Design (939J723)

Figure 2-129. Intermediate ($0.4 \text{ ev} < E < 1 \text{ Mev}$) Neutron Flux Vs. Polar Angle for Meridian Rings of Various Radii for the R-1 Reference Design (939J723)

Figure 2-130. Thermal ($E < 0.4 \text{ ev}$) Neutron Flux Vs. Polar Angle For Meridian Rings of Various Radii for the R-1 Reference Design (939J723)

Figure 2-131. Gamma Ray Kerma Rate on the Reactor Axis (Forward End)
Vs. Distance from Core Center for R-1 Reference Design
(939J723)

W58941

Figure 2-132. Neutron Fluxes on the Reactor Axis (Forward End) Vs. Distance from Core Center for R-1 Reference Design (939J723)

Figure 2-133. Gamma Ray Isodose Rate (RADS-C/HR) Contours at Full Power (1511 MW Thermal) Around the R-1 Reference Design Reactor (939J723) with Test Car and Test Cell C Shield Removed (Composite Fuel)

Figure 2-134. Fast Neutron ($E < 1.0$ Mev) Isoflux (Neutrons/ CM^2 - Sec) Contours at Full Power (1511 MW Thermal) Around the R-1 Reference Design (939J723) with Test Car and Test Cell C Shield Removed (Composite or Graphite Fuel)

Figure 2-135. Intermediate Neutron ($0.4 \text{ EV} < E < 1.0 \text{ MeV}$) Isoflux (Neutrons / $\text{CM}^2 - \text{Sec}$) Countours at Full Power (1511 MW Thermal) Around the R-1 Reference Design Reactor (939J723) with Test Car and Test Cell C Shield Removed (Composite or Graphite Fuel)

Figure 2-136. Thermal Neutron ($E < 0.4$ eV) Isoflux (Neutrons/ CM^2 - Sec) Countours at Full Power (1511 MW Thermal) Around the R-1 Reference Design Reactor (939J723) with Test Car and Test Cell C Shield Removed (Composite or Graphite Fuel)

Figure 2-137. Relative Heating Rate As a Function of Reactor Power Level

Figure 2-138 Relative Radial Distribution of Heating in the Pressure Vessel Side, Region 22
(Composite Core)

Figure 2-139. Relative Axial Distribution of Heating in the Pressure Vessel Side, Region 22 (Composite Core)

Figure 2-140. Relative Radial Distribution of Heating in the Pressure Vessel Side-B,
Region 42

(Composite Core)

Figure 2-141. Relative Axial Distribution of Heating in the Pressure Vessel Side B, Region 42 (Composite Core)

Figure 2-14. Relative Radial Distribution of Heating in the Pressure Vessel Dome, Region 53
(Composite Core)

Figure 2-143. Relative Axial Distribution of Heating in the Pressure Vessel Dome, Region 53
(Composite Core)

Figure 2-144. Relative Increase Due to the Test Cell C Facility Shield on the Gamma Dose Rate as a Function of Radius on the Core Midplane

Figure 2-145. Relative Increase Due to the Test Cell C Facility Shield on the Fast Neutron Flux ($E > 1.0$ MeV) as a Function of Radius on the Core Midplane

Figure 2-146. Relative Increase Due to the Test Cell C Facility Shield on the Intermediate Neutron Flux ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) as a Function of Radius on the Core Midplane

Figure 2-147. Relative Increase Due to the Test Cell C Facility Shield on the Thermal Neutron Flux ($E < 0.4$ eV) as a Function of Radius on the Core Midplane

W59931-1

Figure 2-14B. Relative Increase Due to the Test Cell C Facility Shield on the Al-7039 Heating Rate as a Function of Radius on the Core Midplane

Figure 2-149. Relative Increase Due to the Test Cell C Facility Shield on the Al-6061 Heating Rate as a Function of Radius on the Core Midplane.

Figure 2-150. Relative Increase Due to the Test Cell C Facility Shield on the Beryllium Heating Rate as a Function of Radius on the Core Midplane

Figure 2-151. Relative Increase Due to the Test Cell C Facility Shield on the Carbon Heating Rate as a Function of Radius on the Core Midplane

W58931-1

Figure 2-152. Relative Increase Due to the Test Cell C Facility Shield on the A-286 Heating Rate as a Function of Radius on the Core Midplane

W59031-1

Figure 2-153. Relative Increase Due to the Test Cell C Facility Shield on the Thermal Neutron Flux ($E < 0.4$ eV) as a Function of Axial Height on the Core Centerline

Figure 2-154 Relative Increase Due to the Test Cell C Facility Shield on the Intermediate Neutron Flux ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) as a Function of Axial Height on the Core Centerline

Figure 2-155 Relative Increase Due to the Test Cell C Facility Shield on the Fast Neutron Flux ($E > 1.0$ MeV) as a Function of Axial Height on the Core Centerline

Figure 2-156 Relative Increase Due to the Test Cell C Facility Shield on the Gamma Dose Rate as a Function of Axial Height on the Core Centerline

Figure 2-157. Relative Increase Due to the Test Cell C Facility Shield on the Al-6061 Heating Rate as a Function of Axial Height on the Core Centerline

Figure 2-158. Relative Increase Due to the Test Cell C Facility Shield on the Al-7039 Heating Rate as a Function of Axial Height on the Core Centerline

Figure 2-159 Relative Increase Due to the Test Cell C Facility Shield on the A-286 Heating Rate as a Function of Axial Height on the Core Centerline

Figure 2-160. Relative Increase Due to the Test Cell C Facility Shield on the Carbon Heating Rate as a Function of Axial Height on the Core Centerline

Figure 2-161. R-1 Power Profile for Both a Composite or Graphite Core

M5932-1

Figure 2-162. Neutron Integrating Factor as a Function of Run Time

W59831-1

Figure 2-163. Prompt Integrating Factor as a Function of Run Time

Figure 2-164. Delayed Integrating Factor as a Function of Run Time

Figure 2-165. Prompt and Decay Power as a Function of Reactor Run Time

Figure 2-166.

Delayed Photon Plus Beta Power As a Function of Time After Shutdown Following a 60 Minute Full Power Run (1511 Mw)

Figure 2-167. Gamma Ray Dose Rate as a Function of Axial Location on the Reactor Centerline (Graphite Core)

Figure 2-168 Gamma Ray Dose Rate as a Function of Axial Location at the Core Mid-Radius (Graphite Core)

Figure 2-169. Gamma Ray Dose Rate as a Function of Axial Location at the Core Edge (Graphite Core)

Figure 2-170. Gamma Ray Dose Rate as a Function of Axial Location at the Outer Radius of the Core Periphery (Graphite Core)

Figure 2-171. Gamma Ray Dose Rate as a Function of Axial Location at the Outer Radius of the Lateral Support (Graphite Core)

Figure 2-172. Gamma Ray Dose Rate as a Function of Axial Location at the Inner Radius of the Reflector (Graphite Core)

Figure 2-173. Gamma Ray Dose Rate as a Function of Axial Location at the Control Drum Centerline (Graphite Core)

Figure 2-174. Gamma Ray Dose Rate as a Function of Axial Location at the Outer Radius of the Reflector (Graphite Core)

Figure 2-175. Gamma Ray Dose Rate as a Function of Axial Location at the Pressure Vessel Side Surface (Graphite Core)

Figure 2-176. Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the CHESH (Graphite Core)

Figure 2-177. Gamma Ray Dose Rate as a Function of Radial Location at the Core Exit
(Graphite Core)

Figure 2-178. Gamma Ray Dose Rate as a Function of Radial Location at the Core Midplane (Graphite Core)

Figure 2-179. Gamma Ray Dose Rate as a Function of Radial Location at the Core Inlet
(Graphite Core)

Figure 2-180. Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the Support Plate (Graphite Core)

Figure 2-181. Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the Support Plate (Graphite Core)

Figure 2-182. Gamma Ray Dose Rate as a Function of Radial Location at the Aft Surface of the Shield Plate (Graphite Core)

Figure 2-183. Gamma Ray Dose Rate as a Function of Radial Location at Aft Surface of the BATH Central Shield (Graphite Core)

Figure 2-184. Gamma Ray Dose Rate as a Function of Radial Location at the BATH Central Shield Midplane (Graphite Core)

Figure 2-185. Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the BATH Central Shield (Graphite Core)

Figure 2-186. Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Graphite Core)

Figure 2-187. Gamma Ray Dose Rate as a Function of Radial Location at the Forward Surface of the Flow Baffle (Graphite Core)

Figure 2-188. Gamma Ray Dose Rate as a Function of Radial Location at the Pressure Vessel Dome (Graphite Core)

Figure 2-189. Fast Neutron Flux as a Function of Axial Location on the Reactor Centerline (Graphite Core)

Figure 2-190. Fast Neutron Flux as a Function of Axial Location at the Core Mid-Radius (Graphite Core)

Figure 2-191. Fast Neutron Flux as a Function of Axial Location at the Core Edge (Graphite Core)

Figure 2-192. Fast Neutron Flux as a Function of Axial Location at the Outer Radius of the Core Periphery (Graphite Core)

Figure 2-193. Fast Neutron Flux as a Function of Axial Location at the Outer Radius of the Lateral Support (Graphite Core)

Figure 2-194 Fast Neutron Flux as a Function of Axial Location at the Inner Radius of the Reflector (Graphite Core)

Figure 2-195. Fast Neutron Flux as a Function of Axial Location on the Control Drum Centerline (Graphite Core)

Figure 2-196 Fast Neutron Flux as a Function of Axial Location at the Outer Radius of the Reflector (Graphite Core)

Figure 2-197. Fast Neutron Flux as a Function of Axial Location at the Pressure Vessel Side Surface (Graphite Core)

Figure 2-198. Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the CHESH (Graphite Core)

Figure 2-199. Fast Neutron Flux as a Function of Radial Location at the Core Exit (Graphite Core)

Figure 2-200. Fast Neutron Flux as a Function of Radial Location at the Core Midplane (Graphite Core)

Figure 2-201. Fast Neutron Flux as a Function of Radial Location at the Core Inlet (Graphite Core)

Figure 2-202. Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the Support Plate (Graphite Core)

Figure 2-203. Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the Support Plate (Graphite Core)

Figure 2-204 Fast Neutron Flux as a Function of Radial Location
at the Aft Surface of the Shield Plate (Graphite Core)

Figure 2-205. Fast Neutron Flux as a Function of Radial Location at the Aft Surface of the Bath Central Shield (Graphite Core)

Figure 2-206. Fast Neutron Flux as a Function of Radial Location at the Bath Central Shield Midplane (Graphite Core)

Figure 2-207 Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the Bath Central Shield (Graphite Core)

Figure 2-208. Fast Neutron Flux as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Graphite Core)

**Astronuclear
Laboratory**

Figure 2-21Q Fast Neutron Flux as a Function of Radial Location at the Pressure Vessel Dome (Graphite Core)

Figure 2-211. Intermediate Neutron Flux as a Function of Axial Location on the Reactor Centerline (Graphite Core)

Figure 2-212. Intermediate Neutron Flux as a Function of Axial Location at the Core Mid Radius (Graphite Core)

Figure 2-213. Intermediate Neutron Flux as a Function of Axial Location at the Core Edge (Graphite Core)

Figure 2-214 Intermediate Neutron Flux as a Function of Axial Location at the Outer Radius of the Core Periphery (Graphite Core)

**Astronuclear
Laboratory**

Figure 2-216 Intermediate Neutron Flux as a Function of Axial Location
at the Inner Radius of the Reflector (Graphite Core)

**Astronuclear
Laboratory**

Figure 2-218 Intermediate Neutron Flux as a Function of Axial Location at the Outer Radius of the Reflector (Graphite Core)

Figure 2-219. Intermediate Neutron Flux as a Function of Axial Location at the Pressure Vessel Side Surface (Graphite Core)

Figure 2-220. Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the CHESH (Graphite Core)

Figure 2-221. Intermediate Neutron Flux as a Function of Radial Location at the Core Exit (Graphite Core)

Figure 2-222. Intermediate Neutron Flux as a Function of Radial Location at the Core Midplane (Graphite Core)

Figure 2-223. Intermediate Neutron Flux as a Function of Radial Location at the Core Inlet (Graphite Core)

Figure 2-224. Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the Support Plate (Graphite Core)

**Astronuclear
Laboratory**

Figure 2-226. Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the Shield Plate (Graphite Core)

Figure 2-227. Intermediate Neutron Flux as a Function of Radial Location at the Aft Surface of the Bath Central Shield (Graphite Core)

Figure 2-228. Intermediate Neutron Flux as a Function of Radial Location at the Bath Central Shield Midplane (Graphite Core)

Figure 2-229. Intermediate Neutron Flux as a Function of Radial Location
at the Forward Surface of the BATH Central Shield (Graphite Core)

Figure 2-230. Intermediate Neutron Flux as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Graphite Core)

Figure 2-231 Intermediate Neutron Flux as a Function of Radial Location
at the Forward Surface of the Flow Baffle (Graphite Core)

Figure 2-232 Intermediate Neutron Flux as a Function of Radial Location
at the Pressure Vessel Dome (Graphite Core)

Figure 2-233. Thermal Neutron Flux as a Function of Axial Location on the Reactor Centerline (Graphite Core)

Figure 2-234. Thermal Neutron Flux as a Function of Axial Location at the Core Mid Radius (Graphite Core)

**Astronuclear
Laboratory**

Figure 2-236. Thermal Neutron Flux as a Function of Axial Location at the Outer Radius of the Core Periphery (Graphite Core)

Figure 2-237. Thermal Neutron Flux as a Function of Axial Location at the Outer Radius of the Lateral Support (Graphite Core)

Figure 2-238. Thermal Neutron Flux as a Function of Axial Location at the Inner Radius of the Reflector (Graphite Core)

Figure 2-239. Thermal Neutron Flux as a Function of Axial Location on the Control Drum Centerline (Graphite Core)

Figure 2-240. Thermal Neutron Flux as a Function of Axial Location at the Outer Radius of the Reflector (Graphite Core)

Figure 2-241. Thermal Neutron Flux as a Function of Axial Location at the Pressure Vessel Side Surface (Graphite Core)

Figure 2-242. Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the CHESH (Graphite Core)

Figure 2-243 Thermal Neutron Flux as a Function of Radial Location at the Core Exit (Graphite Core)

Figure 2-244. Thermal Neutron Flux as a Function of Radial Location at the Core Midplane (Graphite Core)

Figure 2-245. Thermal Neutron Flux as a Function of Radial Location at the Core Inlet (Graphite Core)

Figure 2-246. Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the Support Plate (Graphite Core)

Figure 2-247. Thermal Neutron Flux as a Function of Radial Location
at the Forward Surface of the Support Plate (Graphite Core)

Figure 2-248. Thermal Neutron Flux as a Function of Radial Location at the Aft Surface of the Shield Plate (Graphite Core)

Figure 2-249 Thermal Neutron Flux as a Function of Radial Location
at the Aft Surface of the Bath Central Shield (Graphite Core)

Figure 2-250. Thermal Neutron Flux as a Function of Radial Location at the Bath Central Shield Midplane (Graphite Core)

Figure 2-251. Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the Bath Central Shield (Graphite Core)

Figure 2-252. Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the Lead Central Shield (Graphite Core)

Figure 2-253 Thermal Neutron Flux as a Function of Radial Location at the Forward Surface of the Flow Baffle (Graphite Core)

Figure 2-254. Thermal Neutron Flux as a Function of Radial Location at the Pressure Vessel Dome (Graphite Core)

Figure 2-255 Average 155 Group Neutron Flux versus Neutron Energy in the Outer Reflector Region

Figure 2-256. Average 155 Group Neutron Flux versus Neutron Energy in the Graphite Core

Figure 2-257. Average 155 Group Neutron Flux versus Neutron Energy in the BATH Central Shield

Figure 2-258. Average 155 Group Neutron Flux versus Neutron Energy in the Pressure Vessel Dome

Figure 2-259. Average 155 Group Neutron Flux versus Neutron Energy in the Pressure Vessel Side

Figure 2-260. Photon Energy Flux as a Function of Photon Energy at the Center of the Graphite Core

Figure 2-261. Photon Energy Flux as a Function of Photon Energy at the Inner Radius of the Reflector on the Core Midplane

Figure 2-262. Photon Energy Flux as a Function of Photon Energy at the Surface of the Pressure Vessel on the Core Midplane

Figure 2-263. Photon Energy Flux as a Function of Photon Energy at the Aft Face of the BATH Shield on the Centerline

Figure 2-264. Photon Energy Flux as a Function of Photon Energy at the Surface of the Pressure Vessel on the Centerline

Figure 2-265 Photon Heating Rate in Aluminum 6061 ($\rho = 2.71 \text{ gm/cc}$) as a Function of Axial Location on the Reactor Centerline

Figure 2-266 Photon Heating Rate in Aluminum 6061 ($\rho = 2.71 \text{ gm/cc}$) as a Function of Axial Location on the Reactor Centerline

Figure 2-267 Photon Heating Rate in Stainless Steel 304 ($\rho = 7.92 \text{ gm/cc}$) as a Function of Axial Location on the Reactor Centerline

Figure 2-268 Photon Heating Rate in Stainless Steel 304 ($\rho = 7.92 \text{ gm/cc}$) as a Function of Axial Location on the Reactor Centerline

Figure 2-269 Photon Kerma Rate on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design (939J723) With a Graphite Core

Figure 2-270 Fast ($E > 1.0$ Mev) Neutron Flux on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design with a Graphite Core

W59935-1

Figure 2-271 Intermediate ($0.4 \text{ ev} < E < 1.0 \text{ Mev}$) Neutron Flux on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design (939J723) With a Graphite Core

Figure 2-272 Thermal ($E < 0.4$ eV) Neutron Flux on a Plane 63 Inches Forward of the Core Midplane Vs. Radial Distance -- R-1 Reference Design (939J723) With a Graphite Core

Figure 2-273. Gamma Ray Isodose Rate (RAD-C/Hr) Contours at Full Power (1511 MW Thermal) Around the R-1 Reference Design Reactor (939J723) with Test Car and Test Cell C Shield Removed (Graphite Fuel)

THIS PAGE INTENTIONALLY BLANK

Figure 2-274 Relative Radial Distribution of Heating in the Pressure Vessel Side-A, Region 22
(Graphite Core)

Figure 2-275 Relative Axial Distribution of Heating in the Pressure Vessel Side, Region 22
(Graphite Core)

Figure 2-276 Relative Radial Distribution of Heating in the Pressure Vessel Side B, Region 42
(Graphite Core)

Figure 2-277 Relative Axial Distribution of Heating in the Pressure Vessel Side-B, Region 42
(Graphite Core)

Figure 2-278 Relative Radial Distribution of Heating in the Pressure Vessel Dome, Region 53
(Graphite Core)

Figure 2-279 Relative Axial Distribution of Heating in the Pressure Vessel Dome, Region 53
(Graphite Core)

Figure 2-280. Relative Increase Due to the Test Cell C Facility Shield on the Gamma Dose Rate as a Function of Radius on the Core Midplane

Figure 2-281 Relative Increase Due to the Test Cell C Facility Shield on the Fast Neutron Flux ($E > 1.0$ MeV) as a Function of Radius on the Core Midplane

Figure 2-282. Relative Increase Due to the Test Cell C Facility Shield on the Intermediate Neutron Flux ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) as a Function of Radius on the Core Midplane

Figure 2-283. Relative Increase Due to the Test Cell C Facility Shield on the Thermal Neutron Flux ($E < 0.4$ eV) as a Function of Radius on the Core Midplane

Figure 2-284. Relative Increase Due to the Test Cell C Facility Shield on the Al-7039 Heating Rate as a Function of Radius on the Core Midplane

Figure 2-285. Relative Increase Due to the Test Cell C Facility Shield on the A-286 Heating Rate as a Function of Radius on the Core Midplane

Figure 2-286. Relative Increase Due to the Test Cell C Facility Shield on the Beryllium Heating Rate as a Function of Radius on the Core Midplane

Figure 2-287. Relative Increase Due to the Test Cell C Facility Shield on the Carbon Heating Rate as a Function of Radius on the Core Midplane

Figure 2-288. Relative Increase Due to the Test Cell C Facility Shield on the Al-6061 Heating Rate as a Function of Radius on the Core Midplane

W58831-1

Figure 2-289. Relative Increase Due to the Test Cell C Facility Shield on the Thermal Neutron Flux ($E < 0.4$ eV) as a Function of Axial Height on the Core Centerline

W59931-1

Figure 2-290. Relative Increase Due to the Test Cell C Facility Shield on the Intermediate Neutron Flux ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) as a Function of Axial Height on the Core Centerline

W59931-1

Figure 2-291. Relative Increase Due to the Test Cell C Facility Shield on the Fast Neutron Flux ($E > 1.0$ MeV) as a Function of Axial Height on the Core Centerline

Figure 2-292. Relative Increase Due to the Test Cell C Facility Shield on the Gamma Dose Rate as a Function of Axial Height on the Core Centerline

Figure 2-293. Relative Increase Due to the Test Cell C Facility Shield on the Al-6061 Heating Rate as a Function of Axial Height on the Core Centerline

Figure 2-294. Relative Increase Due to the Test Cell C Facility Shield on the Al-7039 Heating Rate as a Function of Axial Height on the Core Centerline

W50031-1

Figure 2-295. Relative Increase Due to the Test Cell C Facility Shield on the A-286 Heating Rate as a Function of Axial Height on the Core Centerline

Figure 2-296. Relative Increase Due to the Test Cell C Facility Shield on the Carbon Heating Rate as a Function of Axial Height on the Core Centerline

3.0 FUEL ELEMENTS

3.1 INTRODUCTION

The Fuel Element design has not been selected; therefore, information will be presented in S-054 TS 772 – Fuel Element Trade Study, R-1 Selection.

3.2 SUMMARY OF RESULTS AND CONCLUSIONS

Portions of the radiation analyses for this ECC were documented in internal memoranda (DRM) for purposes of traceability or, as in the case of the radiation sources, because the information for all ECCs internal to the NSS was required for Engine PDR. The radiation sources in the composite and graphite fuels are presented, respectively, in DRM 52846 and DRM 52847. The radiation heating rate distributions in composite and graphite fuels are presented in DRM 52852.

4.0 CLUSTER HARDWARE

4.1 INTRODUCTION

The Cluster Hardware design has not been selected; therefore, information will be presented in S-054 TS 774 - NERVA Cluster and Hot End Hardware, R-1 Selection.

4.2 SUMMARY OF RESULTS AND CONCLUSIONS

Portions of the radiation analyses for this ECC were documented in internal memoranda (DRM) for purposes of traceability or, as in the case of the radiation sources, because the information for all ECCs internal to the NSS was required for Engine PDR. The radiation sources for the cluster hardware defined by WANL Drawing 940J433 G02 with either a composite or graphite fueled core are presented, respectively in DRM 52846 and DRM 52847. The radiation heating rate distributions in the Cluster Hardware with either a composite or graphite fueled core are presented in DRM 52852.

5.0 CORE PERIPHERY

5.1 INTRODUCTION

The Core Periphery design has not been selected; therefore, information will be presented in S-054, TS 778 - Core Periphery Trade Study for R-1 selection.

5.2 SUMMARY OF RESULTS AND CONCLUSIONS

Portions of the radiation analyses for this ECC were documented in internal memoranda (DRM) for purposes of traceability or, as in the case of the radiation sources, because the information for all ECCs internal to the NSS was required for Engine PDR. The radiation sources for the core periphery defined by WANL Drawing 940J422 with either a composite or graphite fueled core are presented, respectively, in DRM 52846 and DRM 52847. The radiation heating rate distributions in the Core Periphery with either a composite or graphite fueled core are presented in DRM 52852.

6.0 SUPPORT PLATE AND PLENA

6.1 INTRODUCTION

The SPP consists of:

- (a) A core support plate with three integral plena (an SSCA bypass flow plenum, a support stem inlet flow plenum and a plenum which combines the bypass flow with the support stem effluent).
- (b) A piping arrangement which distributes the SSCA outlet flow to the support plate.
- (c) Flow separation baffles and associated hardware which separate the flow to the turbine from the turbine exit flow.
- (d) A diffuser or baffle and associated hardware which distributes turbine exit flow to the central shield.

6.1.1 Description of Support Plate and Plena

6.1.1.1 Support Plate and Plena

The Core Support Plate and Plena consists of a 6 inch thick aluminum plate 44.5 inches in diameter which is internally machined to form three separate plena. The plena consists of cylindrical cells on three levels within the plate at each support stem location interconnected by radial holes.

The support plate transmits the core load to the reflector barrel and thence into the engine load path. Propellant flow to the core is accommodated by axial flow holes through the support plate.

6.1.1.2 Support Plate Plena, Feed and Bypass System

The SSCA feed flow enters the pressure vessel closure through a single pipe and enters the inlet torus. The flow is channeled from the torus to thirty sections of inlet tubing distributed around the torus.

The inlet tubing directs the support stem inlet flow through the internal shield, into the plenum closure ring and subsequently through the feed plenum.

The bypass flow enters the pressure vessel closure through a single pipe and enters the bypass torus. This torus directs the flow to six sections of bypass tubing uniformly spaced around the circumference.

The bypass tubing directs the flow through the internal shield through the plenum closure ring and subsequently into the aft or bypass plenum.

6.1.1.3 Piping Cylinder

The piping cylinder is an aluminum structure which keeps the flow to the turbine inlet separated from the turbine discharge flow. It serves an additional function of carrying the axial load imposed by the internal shield and transmitting this load to the support plate.

6.1.1.4 Central Dome

The central dome is also an aluminum structure which separates the turbine inlet from the turbine discharge flow and mates with the forward flange of the piping cylinder. The dome has a penetration and accommodates an interface seal through which the turbine discharge flow re-enters the NSS.

6.1.2 Content of Chapter

This chapter provides the radiation analysis information related to the Support Plate and Plena ECC. The radiation analysis reported herein was performed for the NSS containing a composite fueled core and for the NSS containing a graphite fueled core. The radiation data for the SPP in this chapter consists of the following information:

- 1) the maximum radiation environment in the SPP,
- 2) the maximum heating rate for each part in the SPP,
- 3) the heating rate distributions in the heating rate model (See Section 6.4.2) regions which comprise the SPP,

- 4) a summary of the maximum effects of Test Cell C on the SPP environment,
- 5) the photon source distribution in the CRAM model (See Section 6.4.2) regions which comprise the SPP, and
- 6) the transient radiation environment in the SPP at selected times during the reactor operating profile.

6.2 SUMMARY OF RESULTS AND CONCLUSIONS

This section presents a summary of the results and conclusions regarding the radiation data obtained for the SPP.

6.2.1 Composite Core

This Section presents the results obtained for the SPP with the NSS containing a composite fueled core.

6.2.1.1 Radiation Environment

The maximum radiation environment in the SPP ECC for the NSS with a composite fueled core occurs at the aft face of the support plate on the reactor centerline. At this location, the gamma ray dose rate is 4.49×10^9 Rads-C/hr and the fast neutron ($E > 1.0$ MeV) flux is 6.40×10^{14} n/cm²-sec. The corresponding dose and fluence for one hour of full power operation is, respectively, 4.49×10^9 Rads-C and 2.30×10^{18} n/cm² while the corresponding dose and fluence for 60 ten minute cycles of the reactor over the power profile defined by Figure 2-161 is, respectively, 4.49×10^{10} Rads-C and 2.30×10^{19} n/cm². To a 95 percent confidence level, the one standard deviation uncertainty factors associated with this data is 1.09 for gamma rays and 1.71 for fast neutrons.

A comparison of the maximum radiation environment in the SPP ECC for the NSS with a composite fueled core to the maximum radiation environment in the SPP ECC for the NSS with a graphite fueled core indicates the following differences:

- 1) the maximum gamma ray dose rate is 16 percent higher in the SPP for the NSS with a graphite fueled core,

- 2) the maximum fast neutron flux is seven percent lower in the SPP for the NSS with a graphite fueled core.
- 3) the maximum intermediate neutron flux is 10 percent lower in the SPP for the NSS with a graphite fueled core, and
- 4) the maximum thermal neutron flux is one percent higher in the SPP for the NSS with a graphite fueled core.

6.2.1.2 Radiation Heating Rates

A total of 3.9 Mw of heat is deposited in the SPP ECC heating rate model for the NSS containing a composite fueled core. Approximately 10 percent more heat is deposited in these regions for the NSS containing a graphite fueled core.

The maximum heating rate in the support plate is 2.02×10^3 Btu/hr-in³. This heating rate for the NSS containing a graphite fueled core is approximately 13 percent higher. To a 95 percent confidence level, the one standard deviation-uncertainty factor for the heating rate in the support plate is 1.09.

6.2.1.3 Radiation Sources

The total photon source in the SPP ECC photon source model regions for the NSS containing a composite fueled core is 1.88×10^{19} Mev/sec and is primarily due to the Fe-Ni-Cr alloys and TZM molybdenum. The photon source strength in these regions for the NSS containing a graphite fueled core is 3.8 percent lower. The one standard deviation uncertainty factor for the SPP photon source is 1.35. For the NSS containing a composite fueled core, the photon sources in the core plenum contribute 17 percent of the dose rate on a ten-foot meridian ring at the reactor centerline. Whereas, the photon sources in the support plate contribute 29 percent of the dose rate at this location. Therefore, the use of Fe-Ni-Cr alloys and TZM molybdenum in these regions should be minimized in order to obtain the minimum weight internal shield.

6.2.2 Graphite Core

This section presents the results obtained for the SPP with the NSS containing a graphite fueled core.

6.2.2.1 Radiation Environment

The maximum radiation environment in the SPP ECC for the NSS containing a graphite fueled core occurs at the same location as that for NSS containing a composite fueled core. At this location, the gamma ray dose rate is 5.22×10^9 Rads-C/Hr and the fast neutron flux is 5.98×10^{14} n/cm²-sec. The corresponding dose and fluence for one hour of full power operation is, respectively, 5.22×10^9 Rad-C and 2.15×10^{18} n/cm² while the corresponding dose and fluence for 60 ten-minute cycles of the reactor over the power profile defined by Figure 2-161 is, respectively, 5.22×10^{10} Rad-C and 2.15×10^9 n/cm². To a 95 percent confidence level, the one standard deviation uncertainty factor associated with this data is 1.07 for gamma rays and 1.59 for fast neutrons.

6.2.2.2 Radiation Heating Rates

A total of 4.3 Mw of heat is deposited in the SPP ECC heating rate model regions for the NSS containing a graphite fueled core. The maximum heating rate in the support plate is 2.29×10^3 Btu/hr-in³. To a 95 percent confidence level, the one standard deviation uncertainty factor for this heating rate in the support plate is 1.07.

6.2.2.3 Radiation Sources

The total photon source in the SPP ECC photon source model regions for the NSS containing a graphite fueled core is 1.81×10^{19} Mev/sec and is primarily due to Fe-Ni-Cr alloys and TZM molybdenum. The one standard deviation uncertainty factor for the photon sources is 1.35.

For the NSS containing a graphite fueled core, the photon sources in the core plenum contribute 15 percent of the dose rate on a ten-foot meridian ring at the reactor centerline. Whereas, the photon sources in the support plate contribute 25 percent of the dose rate at this location.

6.3 REQUIREMENTS

This Section provides SPP requirements which pertain to radiation data.

6.3.1 Specification Identification

The current governing specification for the Support Plate and Plena (SPP) is Specification No. EC-677561.

6.3.2 Applicable Requirements

The applicable requirements from EC-677561 for the SPP which are either evaluated from a radiation standpoint or require radiation information to be evaluated are as follows:

"3.1.1.1.3.1.5 Thermal-Hydraulic Characteristics. - The SPP shall accommodate the requirements of 3.1.1.1.5.1.2 (Startup Operations), 3.1.1.1.5.1.3 (Steady State Operations), 3.1.1.1.5.1.4 (Shutdown Operations), and 3.1.1.1.5.1.5 (Cooldown Operations) of CP-677555. The steady state and throttle hold thermal-hydraulic characteristics of the SPP pertinent to Normal Mode shall be in accordance with Tables 1 through 4.

3.1.1.1.4 Nuclear Radiation Shielding Function. - The SPP design shall limit radiation streaming through piping penetrations through the Internal Shield in conformance with the requirements of 3.1.1.1.4 (Nuclear Radiation Shielding Function) of EC-677562.

3.1.2.1 Reliability - The reliability allocation for the SPP is 0.999939.

3.1.2.4.2.1 Nuclear Environment. - The SPP shall be designed to withstand the induced nuclear environment specified in Table 10. "

6.4 METHOD OF ANALYSIS (DESIGN PRACTICES)

This section describes the methods and models employed in the radiation analysis of the SPP.

6.4.1 Analytical Procedures

Section 2.4.1 described 1) the overall framework of nuclear, thermal, and structural analyses, 2) the relationship of radiation and shielding computations to this overall framework, and 3) the details of the procedure which was employed to obtain radiation and shielding results for the NSS and all of the ECCs in the NSS which includes the SPP.

6.4.2 Analytical Models

Section 2.4.2 provides a detailed description of

- 1) the position at which radiation data at selected locations in the NSS are provided (Figure 2-6 and Table 2-3),
- 2) the computational model employed to perform two-dimensional, discrete ordinates transport calculations with the DOT code (Figures 2-7 through 2-10),
- 3) the radiation heating rate model employed to report heating rates (Figures 2-11 and 2-12), and
- 4) the CRAM model employed to report photon source distributions (Figures 2-13 and 2-14).

Table 2-4 provides a description of the regions contained in each model and indicates which model regions are associated with a particular ECC. Table 2-4, Figure 2-11 for composite fuel and Figure 2-12 for graphite fuel indicate that the following regions comprise the SPP ECC in the heating rate model:

- 1) Region 28, Core Plenum,
- 2) Region 29, Support Plate,
- 3) Region 30, Lateral Support-Forward,
- 4) Region 32, Support Plate Plenum,
- 5) Region 33, Instrumentation Ring,
- 6) Region 37, Flow Baffle I,
- 7) Region 50, Flow Baffle II,
- 8) Region 51, Central Dome Plenum, and
- 9) Region 52, Peripheral Dome Plenum.

Table 2-4, Figure 2-13 for composite fuel, and Figure 2-14 for graphite fuel indicate that the following regions comprise the SPP ECC in the CRAM, or photon source, model:

- 1) Region 9, Core Plenum,
- 2) Region 10, Lateral Support - Forward,
- 3) Region 11, Support Plate,
- 4) Region 13, Support Plate Plenum,
- 5) Region 14, Instrumentation Ring,
- 6) Region 28, Flow Baffle-1,
- 7) Region 29, Flow Baffle-2,
- 8) Region 30, Central Dome Plenum,
- 9) Region 31, Peripheral Dome Plenum-1,
- 10) Region 32, Peripheral Dome Plenum-2, and
- 11) Region 33, Peripheral Dome Plenum-3.

6.4.3 Status/Maturity of Input Data

Section 2.4.3 described the type of input information required to perform the radiation analyses of the NSS which, of course, includes the SPP ECC. This section then described the status/maturity of the input data and the effect on radiation analyses.

The input to radiation analysis for the SPP consisted of atom densities which were based on weights calculated from WANL Drawings 940J410 and 940J420. Subsequent to the time at which these weight calculations were performed, the drawings for the SPP were updated to WANL Drawings 940J449 and 939J725. Very little modification to the new drawings was made. The changes to these drawings consisted of a revised filler strip hanger configuration and the elimination of the internal shield Z-Ring Support.

The revision to the filler strip hanger constituted virtually no change in the mass of materials present. In eliminating the internal shield Z-Ring Support, the radius of the central shield was increased by approximately 0.1 inch. Since BATH and the aluminum Z-Ring have similar densities, the net mass change was negligible. The elimination of the Z-Ring represents an increase in the central shield material of only 18 pounds. Therefore, both revisions to the drawings have a negligible effect on the calculated radiation data for the SPP.

6.4.4 Validity of Analysis

This Section describes the comparison of analytical and experimental results which were employed to determine the validity of the analytical procedure and the uncertainties in the radiation data obtained from this procedure.

6.4.4.1 Comparison with Experimental Results

Section 2.4.4.1 describes all of the experiments which were performed on the PAX reactor so that calculated and experimental results could be compared. Section 2.4.4.1 also provides the major conclusions that were reached for each experiment. Of the experiments performed, the Nuclear Subsystem Shielding Provision Study Experiments ⁽¹⁾ provided results which applied to the SPP ECC.

6.4.4.2 Uncertainty Analysis

The results for the analysis of statistical variations between calculated and experimental data for the Nuclear Subsystem Shielding Provision Study Experiments are presented in Section 2.4.4.2. The derivation of uncertainty factors for radiation environment data

(fast, intermediate, and thermal neutron fluxes and gamma ray dose rates) and radiation heating rates (both neutron and photon) via the procedures outlined in NRP 600 and NRP 605 are also described in Section 2.4.4.2.

Uncertainty factors are derived for possible variations in the radiation data at a point (pointwise uncertainty factor, F_p) and for the possible variations in average radiation data over a region (i. e. , a regionwise uncertainty factor, F_R for the total neutron heating rate in the support plate). Table 2-20 summarizes and describes these uncertainty factors for those portions of the SPP ECC existing below the forward face of the internal shield, while Table 2-21 provides similar information for those portions of the SPP ECC above the forward face of the internal shield. The data in these tables are provided with the NSS containing a composite fueled core. Tables 2-28 and 2-29 provide similar information for the SPP ECC with the NSS containing a graphite fueled core.

6.5 RESULTS, ANALYSIS, AND INTERPRETATION

This Section presents the results of the radiation analysis of the SPP and interpretation of the significance of these results.

6.5.1 Composite Core

The radiation analysis of the SPP was performed for the NSS containing a composite fueled core and for the NSS containing a graphite fueled core. This Section presents radiation results for the SPP with the NSS containing a composite fueled core.

6.5.1.1 Steady State

This Section provides radiation data for the SPP with the NSS at steady state conditions which are defined to be 1) a composite fueled core in the SOL condition, 2) nuclear properties which exist at the end of 60 minutes of full power operation, and 3) thermal-hydraulic conditions which exist at EOL.

6.5.1.1.1 Radiation Environment

Table 6-1 presents the maximum radiation environment in the SPP ECC for the NSS

containing a composite fueled core. The maximum environment in the SPP ECC occurs at the aft face of the support plate on the reactor centerline. This table presents the fast, intermediate, and thermal neutron fluxes, the heating rate (w/gm-SS) and the gamma ray dose rate. Also presented in this table are the neutron fluences and gamma ray doses for one hour of full power (1511 Mw) reactor operation and 60 ten minute cycles of the reactor over the power profile defined by Figure 2-161. Section 6.5.1.2.1 presents the maximum environment integrated over the nominal reactor operating profile at several state points during operation.

The radiation environment (fast, intermediate, and thermal neutron fluxes and photon dose rate) along axial and radial traverses through the SPP is presented in Section 2.5.1.1.1. Section 2.5.1.2.1 presents radiation data at selected locations in the SPP integrated over the reactor operating profile.

The primary purpose of the internal radiation environment data is to provide support to thermal analyses so that thermal properties at the proper irradiated conditions are obtained. In addition, the internal radiation environment data is employed to evaluate the effects of radiation damage on the reactor materials. The S-110 Data Item provides a list of major NSS materials and the safe radiation level which can occur before significant radiation damage occurs. By comparing these levels with the preceding results and the radiation data to be presented in Section 6.5.2.1.1 for a graphite fueled core, the effect of radiation damage on SPP materials can be assessed.

6.5.1.1.2 Radiation Heating Rates

This Section provides the heating rate by region, heating rate distributions, and the effect of Test Cell C on radiation data for the SPP with the NSS containing a composite fueled core.

6.5.1.1.2.1 Total Heating Rates

Table 2-49 provides the heating rates for the heating rate model regions (See Section 6.4.2) associated with the SPP ECC for the NSS containing a composite fueled core. This table indicates that 3.9 Mw of heat are deposited in the regions comprising the SPP ECC. Section 2.5.1.1.4 describes data on this table in detail.

Table 6-2 provides the maximum heating rate in each component in the SPP ECC.

This table provides the following information:

- 1) The drawing numbers for the SPP ECC,
- 2) The item number of each component in the ECC as given by the drawings,
- 3) The part nomenclature,
- 4) The material of which the part is made, and
- 5) The maximum heating rate for the component (BTU/hr-in.³) for the NSS containing either a composite fueled core or a graphite fueled core.

This table indicates that the maximum heating rate in the support plate (item No. 1), is 2.02×10^3 BTU/hr-in.³ for the NSS containing a composite fueled core. The heating rate in the same component for the NSS containing a graphite fueled core is 13 percent higher.

6.5.1.1.2.2 Heating Rate Distributions

Figures 6-1 through 6-22 provide the relative radial and relative axial distribution of radiation heating for the heating rate model regions (See Section 6.4.2) which comprise the SPP ECC. This data is provided for the NSS containing a composite fueled core.

The heating rate at any location within the SPP regions is determined in the following manner:

1. Determine the value of the relative radial distribution of heating for the position of interest,
2. Determine the value of the relative axial distribution of heating for the position of interest,
3. Determine the average value of the heating rate from the appropriate curve (the average value of the heating rate appears on both the radial and axial distribution curves for each region), and
4. The heating rate at the position of interest is the product of the average heating rate, the value of the relative radial distribution, and the value of the relative axial distribution.

6.5.1.1.2.3 Effect of Test Cell C on Radiation Data

Section 2.5.1.1.5 presents the results, and a description of these results, which indicate the effect of Test Cell C on the radiation data internal to the NSS. The data presented represents the maximum effect of Test Cell C on the radiation data.

Briefly, these data indicate that the maximum change in the SPP radiation information occur near the pressure vessel dome and can be quite substantial. However, the change in the SPP radiation data for those portions of the SPP ECC located below the internal shield is less than 0.1 percent.

6.5.1.1.3 Radiation Sources

Table 2-36 provides gamma ray source strengths for each of the regions in the CRAM or photon source, model which comprise the SPP ECC (See Section 6.4.2). Section 2.5.1.1.2 provides a description of the data presented in this table.

Figures 6-23 and 6-24 provide the relative radial and relative axial distribution of the photon source in the core plenum (Region 9) for the NSS containing a composite fueled core. Figures 6-25 through 6-44 provide the relative radial and relative axial distribution of the photon source for the remaining CRAM model regions which comprise the SPP ECC. The data in Figures 6-25 through 6-44 represents the distributions in these regions for the NSS containing either a composite fueled core or a graphite fueled core because no change in the distributions occur with the change in cores.

The photon source strength at any location and for any given energy group in the SPP ECC is determined in the following manner:

1. Determine the value of the relative radial distribution of photon source at the position of interest,
2. Determine the value of the relative axial distribution of the photon source at the position of interest,
3. Determine the source strength for the energy group and region of interest from Table 2-36, and
4. The source strength for the energy group and position of interest is the product of the source strength, the value of the relative radial distribution, and the value of the relative axial distribution.

6.5.1.2 Transient

Figure 2-161 displays the power profile, percent full power as a function of time, for the NSS which employs either a composite fueled core or a graphite fueled core. Section 2.5.1.2 describes this profile and how the transient data provided in this section relates to the State Points indicated on the profile.

6.5.1.2.1 Radiation Environment

Figures 2-162 through 2-164 provide the integrating factors to convert fluxes, dose rates and heating rates to fluences, doses, and heat. Section 2.5.1.2.1 describes the manner in which these integrating factors are applied to radiation data to obtain integrated data over the reactor operating profile.

Table 6-3 presents the maximum radiation data in the SPP ECC for the NSS with a composite fueled core integrated over the power profile to the point in time at which each State Point occurs.

Table 6-3 presents the following information for each State Point:

- 1) the gamma dose and associated uncertainty factor (See Section 2.4.4.2 for the details concerning the uncertainty factor),
- 2) the fast neutron fluence and associated uncertainty factor,
- 3) the intermediate neutron fluence and associated uncertainty factor, and
- 4) the thermal neutron fluence and associated uncertainty factor.

Section 2.5.1.2.1 describes this radiation data in more detail, and Table 2-51 provides similar information at selected locations in the SPP ECC.

6.5.1.2.2 Heating Rates

Figure 2-165 presents the prompt total reactor power and the delayed total reactor power as a function of run time for the power profile defined in Figure 2-161. Figure 2-166 presents the power after shutdown due to delayed photons and beta particles as a function of time after shutdown for the NSS with either a graphite or composite fueled core.

Figures 2-162 through 2-164 provided the integrating factors to convert the total heating rate for a region or material in a region, as supplied by Table 2-49, to the total heat

deposition at any point in time during the run profile defined by Figure 2-161. It should be noted that the neutron, prompt, and delayed integrating factors must be applied to the corresponding heating rate in Table 2-49.

6.5.2 Graphite Core

The radiation analysis of the SPP was performed for the NSS containing a composite fueled core and for the NSS containing a graphite fueled core. This Section presents radiation results for the SPP with the NSS containing a graphite fueled core.

6.5.2.1 Steady State

This Section provides radiation data for the SPP with the NSS at steady state conditions which are defined to be 1) a graphite fueled core in the SOL condition, 2) nuclear properties which exist at the end of 60 minutes of full power operation, and 3) thermal-hydraulic conditions which exist at EOL.

6.5.2.1.1 Radiation Environment

Table 6-1 presents the maximum radiation environment in the SPP ECC for the NSS containing a graphite fueled core. The maximum environment in the SPP ECC occurs at the aft face of the support plate on the reactor centerline. This table presents the fast, intermediate, and thermal neutron fluxes, the heating rate (w/gm-SS) and the gamma ray dose rate. Also presented in this table are the neutron fluences and gamma ray doses for one hour of full power (1511 Mw) reactor operation and 60 ten minute cycles of the reactor over the power profile defined by Figure 2-161. Section 6.5.1.2.1 presents the maximum environment integrated over the nominal reactor operating profile at several state points during operation.

A comparison of the maximum radiation environment in the SPP ECC for the NSS with a composite fueled core to the maximum radiation environment in the SPP ECC for the NSS with a graphite fueled core indicates the following differences:

- 1) the maximum gamma ray dose rate is 16 percent higher in the SPP for the NSS with a graphite fueled core,
- 2) the maximum fast neutron flux is seven percent lower in the SPP for the NSS with a graphite fueled core,

- 3) the maximum intermediate neutron flux is 10 percent lower in the SPP for the NSS with a graphite fueled core, and
- 4) the maximum thermal neutron flux is one percent higher in the SPP for the NSS with a graphite fueled core

The radiation environment (fast, intermediate, and thermal neutron fluxes and photon dose rate) along axial and radial traverses through the SPP are presented in Section 2.5.2.1.1. Section 2.5.2.2.1 presents radiation data at selected locations in the SPP integrated over the reactor operating profile.

The primary purpose of the internal radiation environment data is to provide support to thermal analyses so that thermal properties at the proper irradiated conditions are obtained. In addition, the internal radiation environment data is employed to evaluate the effects of radiation damage on the reactor materials. The S-110 Data Item provides a list of major NSS materials and the safe radiation level which can occur before significant radiation damage occurs. By comparing these levels with the preceding results, the effect of radiation damage in SPP materials can be assessed.

6.5.2.1.2 Radiation Heating Rates

This Section provides the heating rate by region, heating rate distributions, and the effect of Test Cell C on radiation data for the SPP with the NSS containing a graphite fueled core.

6.5.2.1.2.1 Total Heating Rates

Table 2-65 provides the heating rates for the heating rate model regions (See Section 6.4.2) associated with the SPP ECC for the NSS containing a graphite fueled core. This table indicates that 4.3 Mw of heat are deposited in the regions comprising the SPP ECC which is 10 percent more heat deposited in the SPP than was deposited for the SPP with the NSS containing a composite fueled core. Section 2.5.2.1.4 describes data on this table in detail.

Table 6-2 provides the maximum heating rate in each component in the SPP ECC. This table provides the following information:

- 1) The drawing numbers for the SPP ECC,
- 2) The item number of each component in the ECC as given by the drawings,
- 3) The part nomenclature,
- 4) The material of which the part is made, and
- 5) The maximum heating rate for the component (BTU/hr-in.³) for the NSS containing either a composite fueled core or a graphite fueled core.

This table indicates that the maximum heating rate in the support plate (item No. 1), is 2.29×10^3 BTU/hr-in.³ for the NSS containing a graphite fueled core. The heating rate in the same component for the NSS containing a composite fueled core is 13 percent lower.

6.5.2.1.2.2 Heating Rate Distributions

Figures 6-45 through 6-66 provide the relative radial and relative axial distribution of radiation heating for the heating rate model regions (See Section 6.4.2) which comprise the SPP ECC. This data is provided for the NSS containing a graphite fueled core. Section 6.1.1.2.2 describes how the heating rate at any position within the SPP is obtained from this data.

6.5.2.1.2.3 Effect of Test Cell C on Radiation Data

Section 2.5.2.1.5 presents the results, and a description of these results, which indicate the effect of Test Cell C on the radiation data internal to the NSS. The data presented represents the maximum effect of Test Cell C on the radiation data.

Briefly, these data indicate that the maximum change in the SPP radiation information occur near the pressure vessel dome and can be quite substantial. However, the change in the SPP radiation data for those portions of the SPP ECC located below the internal shield is less than 0.1 percent.

6.5.2.1.3 Radiation Sources

Table 2-55 provides gamma ray source strengths for each of the regions in the CRAM or photon source, model which comprise the SPP ECC (See Section 6.4.2). Section 2.5.2.1.2 provides a description of the data presented in this table.

Figures 6-67 and 6-68 provide the relative radial and relative axial distribution of the photon source in the core plenum (Region 9) for the NSS containing a graphite fueled core. Figures 6-25 through 6-44 provide the relative radial and relative axial distribution of the photon source for the remaining CRAM model regions which comprise the SPP ECC. The data in Figures 6-25 through 6-44 represents the distributions in these regions for the NSS containing either a composite fueled core or a graphite fueled core because no change in the distributions occur with the change in cores. Section 6.5.1.1.3 describes how the photon source strength at any position and for any given energy group is obtained from this data.

6.5.2.2 Transient

Figure 2-161 displays the power profile, percent full power as a function of time, for the NSS which employs either a composite fueled core or a graphite fueled core. Section 2.5.1.2 describes this profile and how the transient data provided in this section relates to the State Points indicated on the profile.

6.5.2.2.1 Radiation Environment

Figures 2-162 through 2-164 provide the integrating factors to convert fluxes, dose rates and heating rates to fluences, doses, and heat. Section 2.5.1.2.1 describes the manner in which these integrating factors are applied to radiation data to obtain integrated data over the reactor operating profile.

Table 6-4 presents the maximum radiation data in the SPP ECC for the NSS with a graphite fueled core integrated over the power profile to the point in time at which each State Point occurs.

Table 6-4 presents the following information for each State Point:

- 1) the gamma dose and associated uncertainty factor (See Section 2.4.4.2 for the details concerning the uncertainty factor),

- 2) the fast neutron fluence and associated uncertainty factor,
- 3) the intermediate neutron fluence and associated uncertainty factor, and
- 4) the thermal neutron fluence and associated uncertainty factor.

Section 2.5.1.2.1 describes this radiation data in more detail, and Table 2-66 provides similar information at selected locations in the SPP ECC.

6.5.2.2.2 Heating Rates

Figure 2-165 presents the prompt total reactor power and the delayed total reactor power as a function of run time for the power profile defined in Figure 2-161. Figure 2-166 presents the power after shutdown due to delayed photons and beta particles as a function of time after shutdown for the NSS with either a graphite or composite fueled core.

Figures 2-162 through 2-164 provided the integrating factors to convert the total heating rate for a region or material in a region, as supplied by Table 2-65, to the total heat deposition at any point in time during the run profile defined by Figure 2-161. It should be noted that the neutron, prompt, and delayed integrating factors must be applied to the corresponding heating rate in Table 2-65.

6.6 REFERENCES

1. S. L. Anderson and J. R. Magaw, "Nuclear Subsystem Shielding Provision Study Experiments," WANL-TME-1912 (September 1970)

TABLE 6-1

**MAXIMUM RADIATION ENVIRONMENT AT SUPPORT PLATE
COMPOSITE AND GRAPHITE FUEL IN THE REACTOR CORE**

Description	Location		Gamma Environment			
	Radius (in.)	Reactor Station (in.)	Heating Rate (w/gm-SS)	Dose Rate (Rads-C/hr)	1 Hour Dose (Rads-C)	10 Hour Dose (Rads-C)
<u>Composite Core</u>						
Aft Surface of Support Plate - Reactor Centerline	0.000	-3.500	1.35(1)	4.49(9)	4.49(9)	4.49(10)
<u>Graphite Core</u>						
Aft Surface of Support Plate - Reactor Centerline	0.000	-3.500	1.55(1)	5.22(9)	5.22(9)	5.22(10)

Description	Neutron Environment								
	Fast ($E > 1.0$ MeV)			Intermediate ($0.4 \text{ eV} \leq E \leq 1.0$ MeV)			Thermal ($E < 0.4 \text{ eV}$)		
	Flux ($n/cm^2\text{-sec}$)	1 Hour Fluence (n/cm^2)	10 Hour Fluence (n/cm^2)	Flux ($n/cm^2\text{-sec}$)	1 Hour Fluence (n/cm^2)	10 Hour Fluence (n/cm^2)	Flux ($n/cm^2\text{-sec}$)	1 Hour Fluence (n/cm^2)	10 Hour Fluence (n/cm^2)
<u>Composite Core</u>									
Aft Surface of Support Plate - Reactor Centerline	6.40(14)	2.30(18)	2.30(19)	4.30(15)	1.55(19)	1.55(20)	1.41(14)	5.07(17)	5.07(18)
<u>Graphite Core</u>									
Aft Surface of Support Plate - Reactor Centerline	5.98(14)	2.15(18)	2.15(19)	3.86(15)	1.39(19)	1.39(20)	1.42(14)	5.11(17)	5.11(18)

TABLE 6-2

MAXIMUM HEATING RATE BY COMPONENT FOR THE SUPPORT PLATE AND PLENA
(Defined by DWG. 940J410 and 940J420)

Item Number	Part Nomenclature	Material	Max. Heating Rate (BTU/hr-in. ³)	
			Composite Fuel	Graphite Fuel
	<u>(940J410)</u>			
1	Support Plate and Plena	Al	2. 02(3)	2. 29(3)
2	Bolt	CRES	1. 83(3)	2. 02(3)
3	Locking Cup	CRES	1. 43(3)	1. 59(3)
4	"O" Ring . 125 Tube Dia. . 010 Wall	CRES	1. 63(3)	1. 82(3)
5	"O" Ring . 125 Tube Dia. . 010 Wall	CRES	2. 20(3)	2. 48(3)
6	"O" Ring . 125 Tube Dia. . 010 Wall	CRES	1. 10(3)	1. 19(3)
7	"O" Ring . 125 Tube Dia. . 010 Wall	CRES	9. 87(2)	1. 02(3)
8	Ring, Plenum Closure	Al	4. 03(2)	4. 23(2)
9	Segment, Baffle	Al	6. 48(2)	7. 15(2)
10	Ring, Instrumentation	Al	2. 74(2)	2. 88(2)
11	Bolt	CRES	1. 08(3)	1. 14(3)
12	Locking Cup	CRES	7. 86(2)	8. 23(2)
13	Cap Screw, Socket HD	CRES	7. 94(2)	8. 33(2)
14	"O" Ring . 063 Tube Dia. . 010 Wall	CRES	7. 06(2)	7. 40(2)

NOTE: Numbers in parentheses indicate powers of ten.

TABLE 6-2 (Continued)

MAXIMUM HEATING RATE BY COMPONENT FOR THE SUPPORT PLATE AND PLENA
(Defined by DWG. 940J410 and 940J420)

Item Number	Part Nomenclature	Material	Max. Heating Rate (BTU/hr-in. ³)	
			Composite Fuel	Graphite Fuel
	<u>(940J420)</u>			
1	Baffle	Al	2.56(2)	2.66(2)
2	Tubing	Al	2.01(2)	2.08(2)
	.875 OD x			
	.058 Wall			
3	"O" Ring	CRES	8.04(0)	8.34(0)
	.125 Dia			
	.010 Wall - 39.5 OD			
4	Bolt	CRES	7.24(0)	7.50(0)
	.375 - 16 Unc - 2A			
5	Locking Device	CRES	8.65(0)	8.97(0)
6	Stud - .375 - 16	CRES	9.07(2)	9.51(2)
	Unc 2A			
7	Nut - .375 - 16	CRES	7.70(0)	7.99(0)
	Unc - 2B			
8	Locking Device	CRES	7.70(0)	7.99(0)
9	Waving Spring	CRES	9.10(0)	9.41(0)
10	Piston Ring Seal	CRES	9.09(0)	9.40(0)
11	Piston Ring Housing	Al	3.08(0)	3.18(0)
12	Sleeve	Al	3.81(0)	3.95(0)
13	Flow Diffuser	Al	3.85(0)	3.99(0)
14	Locking Device	CRES	1.15(1)	1.19(1)
15	Bolt	CRES	1.15(1)	1.19(1)
	.375 - 16			
	Unc - 2A			
16	Central Dome	Al	3.65(0)	3.82(0)
17	Elbow Union	Al	2.70(0)	2.81(0)
	(.875 OD Tubing)			
18	Tubing - .875 OD x	Al	2.33(0)	2.40(0)
	.058 Wall			

NOTE: Numbers in parentheses indicate powers of ten.

TABLE 6-2 (Continued)

MAXIMUM HEATING RATE BY COMPONENT FOR THE SUPPORT PLATE AND PLENA
(Defined by DWG. 940J410 and 940J420)

Item Number	Part Nomenclature	Material	Max. Heating Rate (BTU/hr-in. ³)	
			Composite Fuel	Graphite Fuel
	(940J420 Continued)			
19	Elbow Union (.875 OD Tubing)	Al	2.70(0)	2.81(0)
20	Tubing .875 OD x .058 Wall	Al	2.21(0)	2.27(0)
21	Inlet Torus	Al	2.55(0)	2.63(0)
22	Bypass Torus	Al	2.28(0)	2.34(0)
23	Interface Sleeve	CRES	6.25(0)	6.47(0)
24	Wave Spring	CRES	6.06(0)	6.54(0)
25	Piston Ring Seal	CRES	6.09(0)	6.56(0)
26	Screen No. 20 Mesh (.016 Dia Wire)	CRES	5.79(0)	5.99(0)
27	Ring, Retainer	CRES	5.05(0)	5.23(0)
28	Spirolox 3.875 OD, .120 Thk	CRES	5.02(0)	5.19(0)
29	Interface Seal Sleeve	CRES	5.75(0)	5.96(0)
30	Screen No. 20 Mesh (.016 Dia Wire)	CRES	5.58(0)	5.80(0)
31	Piston Ring Seal	CRES	5.44(0)	5.67(0)
32	Wave Spring	CRES	5.41(0)	5.64(0)
33	Ring, Retainer	CRES	4.51(0)	4.71(0)
34	Spirolox, 1.562 OD .068 Thk	CRES	4.48(0)	4.68(0)
35	Interface Seal	Al	2.95(0)	3.05(0)
36	Locking Device	CRES	6.07(0)	6.28(0)
37	Screw - .250 20 Unc - 2A	CRES	6.85(0)	7.06(0)
38	Tubing .875 OD x .058 Wall	Al	2.33(0)	2.40(0)

NOTE: Numbers in parentheses indicate powers of ten.

TABLE 6 - 3

MAXIMUM TIME INTEGRATED RADIATION ENVIRONMENT IN THE SUPPORT PLATE AND PLENA
(COMPOSITE CORE)

LOCATION - AFT SURFACE OF SUPPORT PLATE - REACTOR CENTERLINE
RADIUS (IN) O.C STATION (IN) -3.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

NEUTRONS/CM2, E GT 1.0MEV	9.86E 15	1.71	3.41E 17	1.71	3.43E 17	1.71	3.58E 17	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.36E 16	1.71	2.20E 18	1.71	2.21E 18	1.71	2.31E 18	1.71
NEUTRONS/CM2, E LT 0.4EV	2.34E 15	1.88	8.10E 16	1.88	8.14E 16	1.88	8.51E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.22E 07	1.09	8.16E 08	1.09	8.20E 08	1.09	8.58E 08	1.09

TABLE 6 - 4
 MAXIMUM TIME INTEGRATED RADIATION ENVIRONMENT IN THE SUPPORT PLATE AND PLENA
 (GRAPHITE CORE)

LOCATION - AFT SURFACE OF SUPPORT PLATE - REACTOR CENTERLINE RADIUS (IN) 0.0 STATION (IN) -3.50 *****									
		STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
		NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
NEUTRONS/CM2, E GT 1.0MEV		1.05E 16	1.59	3.65E 17	1.59	3.67E 17	1.59	3.83E 17	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV		7.09E 16	1.59	2.45E 18	1.59	2.46E 18	1.59	2.58E 18	1.59
NEUTRONS/CM2, E LT 0.4EV		2.32E 15	1.73	8.04E 16	1.73	8.08E 16	1.73	8.45E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)		1.95E 07	1.07	7.06E 08	1.07	7.09E 08	1.07	7.42E 08	1.07

Figure 6-1 Relative Radial Distribution of Heating in the Core Plenum, Region 28
(Composite Core)

Figure 6-2 Relative Radial Distribution of Heating in the Core Plenum, Region 28
(Composite Core)

Figure 6-3 Relative Axial Distribution of Heating in the Core Plenum, Region 28
(Composite Core)

Figure 6-4 Relative Axial Distribution of Heating in the Core Plenum, Region 28
(Composite Core)

Figure 6-5 Relative Radial Distribution of Heating in the Support Plate, Region 29
(Composite Core)

Figure 6-6 Relative Axial Distribution of Heating in the Support Plate, Region 29
(Composite Core)

Figure 6-7 Relative Radial Distribution of Heating in the Lateral Support - Forward, Region 30

(Composite Core)

Figure 6-8 Relative Axial Distribution of Heating in the Lateral Support-Forward,
Region 30 (Composite Core)

Figure 6-9 Relative Radial Distribution of Heating in the Support Plate Plenum, Region 32
(Composite Core)

Figure 6-10 Relative Axial Distribution of Heating in the Support Plate Plenum, Region 32
(Composite Core)

Figure 6-11 Relative Radial Distribution of Heating in the Instrumentation Ring, Region 33
(Composite Core)

Figure 6-12 Relative Axial Distribution of Heating in the Instrumentation Ring, Region 33
(Composite Core)

Figure 6-13 Relative Radial Distribution of Heating in the Flow Baffle I, Region 37
(Composite Core)

Figure 6-14 Relative Axial Distribution of Heating in the Flow Baffle I, Region 37
(Composite Core)

Figure 6-15 Relative Radial Distribution of Heating in the Flow Baffle II, Region 50

(Composite Core)

Figure 6-16 Relative Axial Distribution of Heating in the Flow Baffle II, Region 50
(Composite Core)

Figure 6-17 Relative Radial Distribution of Heating in the Peripheral Dome Plenum-A,
Region 52

(Composite Core)

Figure 6-18 Relative Axial Distribution of Heating in the Peripheral Dome Plenum-A,
Region 52

(Composite Core)

Figure 6-19 Relative Radial Distribution of Heating in the Peripheral Dome Plenum-B,
Region 52

(Composite Core)

Figure 6-20 Relative Axial Distribution of Heating in the Peripheral Dome Plenum-B,
Region 52

(Composite Core)

Figure 6-21 Relative Radial Distribution of Heating in the Peripheral Dome Plenum-C,
Region 52

(Composite Core)

Figure 6-22 Relative Axial Distribution of Heating in the Peripheral Dome Plenum-C,
Region 52

(Composite Core)

Figure 6-23. Relative Radial Distribution of Photon Source in Region 9, Core Plenum

Figure 6-24. Relative Axial Distribution of Photon Source in Region 9, Core Plenum

Figure 6-25. Relative Radial Distribution of Photon Source in Region 10, Lateral Support Forward

Figure 6-26. Relative Axial Distribution of Photon Source in Region 10, Lateral Support Forward

Figure 6-27. Relative Radial Distribution of Photon Source in Region 11 Support Plate

Figure 6-28. Relative Axial Distribution of Photon Source in Region 11, Support Plate

Figure 6-29. Relative Radial Distribution of Photon Source in Region 13, Support Plate Plenum

Figure 6-30. Relative Axial Distribution of Photon Source in Region 13, Support Plate Plenum

Figure 6-31. Relative Radial Distribution of Photon Source in Region 14, Instrumentation Ring

Figure 6-32. Relative Axial Distribution of Photon Source in Region 14, Instrumentation Ring

Figure 6-33. Relative Radial Distribution of Photon Source in Region 28, Flow Baffle - 1

Figure 6-34. Relative Axial Distribution of Photon Source in Region 28, Flow Baffle - 1

Figure 6-35. Relative Radial Distribution of Photon Source in Region 29, Flow Baffle - 2

Figure 6-36 Relative Axial Distribution of Photon Source in Region 29, Flow Baffle-2

Figure 6-37. Relative Radial Distribution of Photon Source in Region 30, Central Dome Plenum

Figure 6-38 Relative Axial Distribution of Photon Source in Region 30, Central Dome Plenum

Figure 6-39. Relative Radial Distribution of Photon Source in Region 31, Peripheral Dome Plenum - 1

Figure 6-40 Relative Axial Distribution of Photon Source in Region 31, Peripheral Dome Plenum-1

Figure 6-41. Relative Radial Distribution of Photon Source in Region 32, Peripheral Dome Plenum - 2

Figure 6-42. Relative Axial Distribution of Photon Source in Region 32, Peripheral Dome Plenum - 2

Figure 6-43. Relative Radial Distribution of Photon Source in Region 33, Peripheral Dome Plenum - 3

Figure 6-44. Relative Axial Distribution of Photon Source in Region 33, Peripheral Dome Plenum - 3

Figure 6-45 Relative Radial Distribution of Heating in the Core Plenum, Region 28
(Graphite Core)

Figure 6-46 Relative Axial Distribution of Heating in the Core Plenum, Region 28
(Graphite Core)

Figure 6-47 Relative Radial Distribution of Heating in the Core Plenum, Region 28
(Graphite Core)

Figure 6-48 Relative Axial Distribution of Heating in the Core Plenum, Region 28
(Graphite Core)

Figure 6-49 Relative Radial Distribution of Heating in the Support Plate, Region 29
(Graphite Core)

Figure 6-50 Relative Axial Distribution of Heating in the Support Plate, Region 29
(Graphite Core)

Figure 6-51 Relative Radial Distribution of Heating in the Lateral Support - Forward, Region 30
(Graphite Core)

Figure 6-52 Relative Axial Distribution of Heating in the Lateral Support - Forward,
Region 30

(Graphite Core)

Figure 6-53 Relative Radial Distribution of Heating in the Support Plate-Plenum, Region 32
(Graphite Core)

Figure 6-54 Relative Axial Distribution of Heating in the Support Plate Plenum, Region 32
(Graphite Core)

Figure 6-55 Relative Radial Distribution of Heating in the Instrumentation Ring, Region 33
(Graphite Core)

Figure 6-56 Relative Axial Distribution of Heating in the Instrumentation Ring, Region 33
(Graphite Core)

Figure 6-57 Relative Radial Distribution of Heating in the Flow Baffle I, Region 37
(Graphite Core)

Figure 6-58 Relative Axial Distribution of Heating in the Flow Baffle I, Region 37
(Graphite Core)

Figure 6-59 Relative Radial Distribution of Heating in the Flow Baffle II, Region 50
(Graphite Core)

Figure 6-60 Relative Axial Distribution of Heating in the Flow Baffle II, Region 50
(Graphite Core)

Figure 6-61 Relative Radial Distribution of Heating in the Peripheral Dome Plenum-A,
Region 52-A

(Graphite Core)

Figure 6-62 Relative Axial Distribution of Heating in the Peripheral Dome Plenum-A,
Region 52-A

(Graphite Core)

Figure 6-63 Relative Radial Distribution of Heating in the Peripheral Dome Plenum-B,
Region 52-B

(Graphite Core)

Figure 6-64 Relative Axial Distribution of Heating in the Peripheral Dome Plenum-B
Region 52-B

(Graphite Core)

Figure 6-65 Relative Radial Distribution of Heating in the Peripheral Dome Plenum-C,
Region 52-C

(Graphite Core)

Figure 6-66 Relative Axial Distribution of Heating in the Peripheral Dome Plenum-C,
Region 52-C

(Graphite Core)

Figure 6-67. Relative Radial Distribution of Photon Source in Region 9, Core Plenum

Figure 6-68. Relative Axial Distribution of Photon Source in Region 9, Core Plenum