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**Westinghouse
Astronuclear
Laboratory**

**WANL-TME-1912
SEPTEMBER 1970**

DATA ITEM T-119

**NUCLEAR SUBSYSTEM
SHIELDING PROVISION STUDY
EXPERIMENTS
FINAL REPORT**

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ABSTRACT

A series of experiments was performed with heterogeneous slab mockups of two types of R-1 internal shield designs that were considered for the Nuclear Subsystem Shielding Provision Study. These experiments, which were placed on the low power critical PAX-G0C reactor, provided information on the internal and external radiation environments for 69, 89, and 109 gm/cm² Boron Carbide-Aluminum-Titanium-Hydride (BATH) shields and for a 117 gm/cm² Borated Stainless Steel Liquid Hydrogen shield. Measurements were made to determine the shield mockup internal radiation environments using TLD and foil passive dosimetry to measure fast neutron fluxes, thermal neutron fluxes, and gamma ray-absorbed dose rates. Measurements of the external radiation environment were made with both active and passive dosimetry to determine fast neutron fluxes and dose rates, and gamma ray-absorbed dose rates. Test cell background for the external measurements was determined by similar measurements obtained on an auxiliary reactor shadow shield configuration. Direct leakage fluxes and dose rates outside the shield assemblies were thus obtained by subtraction of the test cell background to permit direct comparisons with analytical calculations.

An in-depth analysis of the experiments was performed utilizing the WANL Integrated Nuclear, Radiation and Shielding Standard Design Method (WISDM). Both one-dimensional analysis with the ANISN-W discrete ordinates code and two-dimensional analysis with the DOT-IIW discrete ordinates code were performed to provide neutron and photon radiation levels internal to the shield mockups. In the neutron analysis, 16 group (11 fast and 5 thermal) P_0 transport corrected cross sections were employed. In the photon problem, 13 group P_1 cross sections were utilized. An S_8 quadrature was employed throughout the analysis. Typically 8000 mesh cells were used to describe the reactor-shield geometry in the two-dimensional discrete ordinates analysis. This was accomplished by utilizing the boundary source option in the DOT-IIW code. Analytical results external to the experimental configurations were obtained both with the discrete ordinates and point kernel methods. The external discrete ordinates analysis was accomplished by the MAP code which utilizes as input the DOT-IIW surface angular leakage fluxes. Sources internal to the reactor-shield assemblies from the discrete

ordinates method were used as input to the KAP-V point kernel code to calculate external fast neutron and photon environment. Extensive comparisons between the experimental data and the various types of calculations are presented and discussed.

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1.0 INTRODUCTION

1.1 TEST IDENTIFICATION

1.1.1 Type of Report

This report is a complete final report.

1.1.2 Scope of Test Document

The applicable Scope of Test Document Number is 62DW-268.-3.

1.1.3 Functional Category of Test

This is a Development test.

1.1.4 Contractor

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

1.1.5 Test Facility

These tests were performed at the Westinghouse Astronuclear Experimental Facility located at the Westinghouse Waltz Mill Site, during February 21 through April 7, 1969.

1.2 PURPOSE

The purposes of these tests are to: Experimentally determine the shielding effectiveness of an R-1 internal shield mockup for some of the shield designs that are being considered for the Nuclear Subsystem Shielding Provisions Systems Study, provide reactivity measurements of the shield mockups, provide experimental data for comparison with analytical predictions, and provide input to data item S-100.

2.0 REFERENCES AND ASSUMPTIONS

2.1 REFERENCES

2.1.1 Specification

CP 677562 - Part I. Performance/Design and Qualification Requirements
Internal Shield for NERVA Nuclear Subsystem 677555A, 6/1/68.

2.1.2 Data Item Description

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

2.1.3 Scope of Test Document

62DW-268-3 - Nuclear Subsystem Shielding Provision Study

2.1.4 Data Release Memorandum

51157 - Calibration of a Fast Fission Detector (W Model No. WX-30748,
S/N 680401)

51158 - Calibration of a CO₂ Ionization Chamber RSG-8A Model, S/N G-929

51159 - PAX-G0 Detailed Power Measurements

51161 - Reactivity Worth of the Pressure Vessel & Shield Adapter Ring in the
PAX-G0C Reactor

51168 - Reactivity Measurements of Shield Mockups Used in the NSS Shielding
Provision Study Experiment

51169 - Nuclear Subsystem Shielding Provision Study Experiment Preliminary
Evaluation of BATH Versus BSS-LH₂ Shields

51173 - Calibration and Data Reduction Technique of a Hurst Fast Neutron Dosimeter

51176A - Calibration of Thermoluminescent Dosimeters

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

51179 - Bragg-Gray Detectors for Heating Rate Measurements

51181 - A Description of the Polar Angle Indexing System Used With the WANEF
(ρ, θ, ϕ) Traverse System for the PAX Reactor.

- 51183 - Active Dosimetry Data From the NSS Shield Provision Trade Study Experiment
- 51184 - NSS Shield Provision Trade Study Experiment Shield Material Composition

Tables Used on PAX-G0C Reactor at WANEF

- 51152A - PAX-G0 Shim Wire Inventory
- 51152 - PAX-G0B Shim Wire Inventory
- 51150A - Parts Inventory in the PAX-G0-A, -B, and -C Reactors
- 51150 - Parts Inventory in the PAX-G0 Reactor
- 51185 - Passive Dosimetry Data From NSS Shield Provision Trade Study Experiment
- 51105C - Flow Charts for an Integrated Nuclear and Radiation, Thermal and Structural Analysis Procedure and Identification and Description of Computer Codes
- 51198A - Hurst Data Reduction Code For a Hurst Fast Neutron Proportional Counter

2.1.5 Drawing

- 711J496 - Reactor/Vessel Assembly, R-1 PAX
- 100E153 - Pressure Vessel Ring

2.1.6 TME

WANL-TME-267, Revision H - Westinghouse Astronuclear Experimental Facility Reactor Operations Manual

WANL-TME-1852 - R-1 Parametric Shield Measurements Conducted on the NERVA Permanent Assembly Experiment Reactor

WANL-TME-1915 - Radiation Environment on CHESH Components

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

WANL-TME-1910 - PAX-G0 Control Drum Spans and Reactivity Coefficient

Measurements

WANL-TME-527 - Design of Neutron Detectors for NERVA

WANL-TME-1952, Rev A. - PARX2 Data System Users Manual

2.2 ASSUMPTIONS

2.2.1 General

The PAX-G0C reactor provides a suitable radiation leakage environment at the aft end of the reactor which approximates the R-1 reactor shield radiation environment such that the experiments performed can be used by the Radiation and Shielding Design group for checks on analytical calculations.

The test cell background can be adequately measured and accounted for and thus measurements of the direct radiation environment can be obtained.

The radiation environments do not change appreciably after a level reactor power has been obtained within the time scale of these experiments.

2.2.2 Power Calibrations and Fission Distributions

The digital reactor power monitor along with irradiated gold wires provide an adequate inter-run reactor power normalization system.

The centerline axial fission distribution measurements are sufficient to show the true effects of the shield configurations on the axial fission profile such that these data can be incorporated into the necessary power calibrations to determine "true" reactor power.

2.2.3 Passive Dosimetry and Active Dosimetry

Measurements made on the $\phi = 270^\circ$ radial from core centerline are more representative of the R-1 radiation environment due to a better mockup of fuel loading in this portion of the reactor and the fact that the radiation environment of the reactor for $\phi = 270^\circ$ is influenced least by the effects of the test cell walls.

The presence of a rectangular shield configuration rather than a circular shape does not significantly alter the measured radiation distribution.

The lack of sufficient gamma side shielding for the shield configuration used to obtain background measurements does not significantly affect the measured direct radiation environment.

Measurements made with the three gamma ray absorbed dose rate detectors operated simultaneously, are individually correct with no interference effects between detectors. Also there is no change in the radiation environment and the position of the center detector can be reported as the location point for all three detectors.

3.0 TEST ARTICLE CONFIGURATION

3.1 DESCRIPTION OF TEST CONFIGURATION

Experiments were performed at WANEF on the PAX-GOC reactor in order to fulfill the basic objectives outlined in Section 1.2. This section describes the test article configuration used during the Nuclear Subsystem Shielding Provision Study experiments. It describes the WANEF test facility, the PAX-GOC reactor, shield mockup configuration, the passive dosimetry systems, the active dosimetry systems, the traverse system, the nuclear instrumentation and the Boral enclosure used to reduce the "room return" thermal neutron background above the reactor.

3.1.1 WANEF Test Facility

The facility building houses two reactor assemblies: the PAX (Permanent Assembly Experiment) with which the reported measurements were performed and the FCX (Flexible Critical Experiment) which was inactive during the performance of these measurements. The test cell is constructed such that the height of the roof above the ground level is 52 feet and the basement depth is 22 feet 6 inches. Above ground level, the north and west concrete walls are 4 feet 6 inches thick to a height of 20 feet. Above this, the west wall is cut back to a thickness of 2 feet and the north wall to 1 foot for the remaining height of the test cell.

The east and south concrete walls are 2 feet thick to a height of 20 feet above ground level and are then 1 foot thick above this level. A 12 inch thick poured concrete slab on the north wall serves as a storage shelf for reactor fuel and miscellaneous experimental hardware for the WANEF facility. This storage shelf is 12 feet wide and extends the entire east-west length of the test cell. The reactor is mounted in a steel test stand which raises the reactor centerline approximately 9 feet above the test cell floor. A steel deck working platform surrounds the reactor to allow work to be done on the aft end of the reactor. A more detailed description of the test cell geometry has been presented in Section 4.1 of WANL-TME-1852, "R-1 Parametric Shield Measurements Conducted on the NERVA Permanent Assembly Experiment (PAX) Reactor".

To reduce the test cell "room return" radiation background, caused in part by the presence of the massive north and west concrete walls near the PAX assembly, wall liner material was placed on the surface of the walls in the immediate vicinity of the PAX reactor.

These walls were covered with borated polyethylene (2 w/o boron)* to a total of 5.080 cm thickness in a 2.44 meters x 4.88 meters area above the test stand and 1.22 meters x 1.22 meters area below the test stand centered on the reactor centerline. A general view of the wall covering and the PAX-GOC reactor is shown in Figures 3-1 and 3-2. Part of the polyethylene lined area in the vicinity of the reactor centerline was also covered with 0.3175 cm thick Pb sheets that are 1.52 meters x 0.91 meters and 1.22 meters x 0.91 meters.

The wall liner was constructed of sixteen (1.22 meters x 2.44 meters) x 2.54 cm and four (1.22 meters x 1.22 meters) x 2.54 cm borated polyethylene (2 w/o boron) sheets, six (1.52 meters x 0.91 meters) x 0.3175 cm lead sheets, plus welded frames made from Unistrut (light weight U-shaped steel channel) to secure the borated polyethylene next to the concrete walls. Further details of the construction and procedures used to assemble the wall liner material are contained in the experimental procedures in Appendix A.

3.1.2 PAX-GOC Reactor

The PAX-GOC reactor is an NRX type assembly which was modified to be an early nuclear mockup of the R-1 Reference Design. Basically, it is a beryllium reflected reactor with a core consisting of highly enriched uranium dicarbide fuel contained in a graphite matrix. Control is obtained through the rotation of 12 beryllium control drums which contain neutron absorbing plates on one side. These control drums are contained in the reflector. A permanent aluminum shield is bolted to the core support plate located on the inlet end of the PAX assembly.

* Manufacturer's quoted content.

Surrounding the reactor subassembly and shield is an aluminum pressure vessel which is comprised of three sections: the dome which contains the shield assembly, the cylinder body which contains the reactor subassembly, and a removable section at the nozzle end of the pressure vessel. This section was removed and a pressure vessel ring substituted to support the materials for the shielding provision study experiments. The PAX-GOC reactor with the pressure vessel ring in place is shown in Figure 3-1.

The PAX-GOC reactor build (including shim wire inventory, parts inventory, fuel element listings, and/or representative as-built weights) are detailed in Drawings 711J496, 100E153; and DRM's 51152, 51152A, 51150 and 51150A.

3.1.3 Shield Mockup Configurations

It was necessary, in order to fulfill the Scope of Test Document, to determine the nuclear and radiation environment of three different thicknesses of BATH* shield mockup and a BSS-LH₂ (Borated Stainless Steel - Liquid Hydrogen) shield mockup. In addition, measurements were needed, for comparison purposes, on a bare PAX-GOC reactor (i.e., no material covering the aft end of the core) and a measurement of the "room return" background was needed to enable the determination of the direct radiation environment coming from the shield mockup. (The term "direct" used throughout this report means that the measurement of the background has been subtracted from the "total" measured environment.) For ease of data reporting and analysis, each of the above measurements was assigned a shield mockup configuration number continuing a sequence reported in WANL-TME-1852. A brief discussion of each of the shield mockup configurations follows. A detailed discussion of the chemical compositions of the materials used in these shield mockup configurations is found in DRM 51184.

* Boron carbide - Aluminum - Titanium Hydride

3.1.3.1 Configuration 15 - Bare PAX-G0C Reactor

The PAX-G0C reactor, described in Section 3.1.2, with the following material addition was used for the shield configuration 15 measurements. The addition was in the form of a layer of Boral which was placed completely around the PAX-G0C pressure vessel. The Boral wrapper was fabricated from 3 sheets of 0.3175 cm thick by 132.08 cm wide by 132.08 cm long, 35 weight percent B_4C Boral**. Six sheets 0.3175 cm thick, 66.04 cm wide and 132.08 cm long of material were rolled to fit the curvature of the PAX pressure vessel surface. The wrapper was placed so that a continuous sheet was formed between elevation -66.04 ± 0.60 cm and elevation $+66.04 \pm 0.6$ cm completely around the pressure vessel where elevation is measured positively with reference zero at the reactor core midplane.

3.1.3.2 Configuration 16A - 20.32 Cm BATH Mockup

The shield configuration 16A mockup attempted to simulate, in a gross way, the nuclear and radiation attenuation properties of a 20.32 cm thick BATH shield at the dome end region of the PAX-G0C reactor.

The materials were added in the form of a heterogeneous array of alternating thin sheets, with a fixed sequence, nominal 132.08 cm x 132.08 cm in lateral extent, positioned on the aft (nozzle) end of the PAX-G0C reactor. The BATH shield mockup was composed of sheets of Ti, Boral, Al, and polyethylene in alternating layers adjusted as nearly as possible to the volume fraction of the BATH shield material. The alternating layers were kept thin (less than 0.635 cm) in an effort to approximate a homogeneous shield region.

The material composition was selected in order to mockup the particular atom number densities of the reference shield material. Due to the relatively large thickness of the shield, separate well defined regions of material were fabricated that individually mocked up the atom density composition of the reference shield material. These individual

* Boral is a trade name for material composed of a layer of B_4C -Al powder sandwiched between layers of aluminum sheets. Nominal 35 w/o B_4C of the core or 20.55 w/o B_4C of the total cross section. The reference with respect to the core weight percent of B_4C will be used throughout the text.

regions are defined as modules. The modules consisted of a vertical array of the following thickness of material: 0.686 cm titanium, 0.661 cm aluminum, 0.337 cm Boral (35 w/o B_4C), 0.493 cm aluminum, 0.625 cm polyethylene, 0.493 cm aluminum and 0.646 cm Boral (35 w/o B_4C) sheets of material. Aluminum spacers, 0.635 cm thick, were inserted at selected locations in order to provide access for passive detectors. The sides of the shield mockup were covered by four 34.0 cm x 132.08 cm x 0.337 cm Boral (35 w/o B_4C) sheets.

A photograph of configuration 16A, placed in position on top of the PAX-G0C reactor, is shown in Figure 3-3. The materials for configuration 16A were placed on the pressure vessel ring with the orientation shown in Figure 3-4. A cross-sectional diagram of configuration 16A is given in Figure 3-5. The total areal density of configuration 16A was $\sim 69.0 \text{ gm/cm}^2$.

3.1.3.3 Configuration 16B and 16C – 30.48 Cm and 38.10 Cm BATH Mockups

Shield mockup configuration 16B consisted of shield mockup configuration 16A plus the addition of two more modules of BATH mockup. Shield mockup configuration 16C was an extension of configuration 16A with the addition of four BATH mockup modules. Figure 3-5 shows cross sectional diagrams of both shield mockup configuration 16B and 16C. The material orientation was not changed from configuration 16A. The total areal density of configuration 16B and configuration 16C was $\sim 89.0 \text{ gm/cm}^2$ and $\sim 109.0 \text{ gm/cm}^2$, respectively.

3.1.3.4 Configuration 17 – BSS-LH₂ Mockup

Shield mockup configuration 17 attempted to simulate, in a gross way, the nuclear and radiation attenuation properties of a borated stainless steel – liquid hydrogen shield at the dome end region of the PAX-G0 reactor.

The materials were added in the form of a heterogeneous array of alternating slabs nominal 132.08 cm x 132.08 cm in lateral extent positioned on the aft (nozzle) end of the PAX-G0C reactor. The BSS-LH₂ shield was composed of sheets of borated stainless steel or stainless steel and polyethylene foam in alternating layers. The slabs were assembled such that the polyethylene foam sheets were sandwiched between sheets of borated stainless steel or stainless steel in successive sandwiches. Each sandwich section was approximately

10 cm thick. The material composition was selected in order to mockup the particular atom number densities specified in the Scope of Test document 62DW-268-3. Due to the large thickness of the shield, the passive detectors were placed directly in the polyethylene sheets in appropriately cut slots; thus, no spacers were used in this mockup. The sides of the shield mockup were covered by four 132.08 cm x 101.60 cm x 0.3175 cm Boral (35 w/o B_4C) sheets.

A cross-sectional diagram of configuration 17 is given in Figure 3-6. A photograph of configuration 17 in the PAX-G0C reactor with the Boral sides in place is shown in Figure 3-7. The materials for configuration 17 were placed on the pressure vessel ring with the same orientation as shown in Figure 3-4. The total areal density of Configuration 17 was $\sim 117.0 \text{ gm/cm}^2$.

3.1.3.5 Configuration 14 - Background Shadow Shield

Configuration 14 was used to obtain test cell background. Due to the material arrangement and composition, the configuration was denoted as the "thick" shield. A cross-sectional view of the top portion of configuration 14 is shown in Figure 3-8. The following sequence of material arrangements comprised the top portion of configuration 14: 4.078 cm aluminum (2219), 0.337 cm of Boral (35 w/o B_4C), two 2.540 cm thick borated polyethylene sheets (2 w/o boron), 2.540 cm of polyethylene, 0.337 cm thick Boral (35 w/o B_4C), 5.080 cm of lead bricks, 5.080 cm of borated polyethylene sheets (2 w/o Boron), 0.646 cm thick Boral (35 w/o B_4C), 7.620 cm of polyethylene, 0.646 cm thick Boral (35 w/o B_4C), 5.08 cm of lead bricks, 10.16 cm of polyethylene, 0.646 cm thick Boral (35 w/o B_4C), and 5.08 cm of lead bricks. The cross section was 132.08 cm x 132.08 cm. The edge of the second 5.08 cm layer of Pb bricks were recessed approximately 0.635 cm on two sides from the first layer to prevent gamma streaming through the joints of the assembly. The materials for the top portion of configuration 14 were placed on the pressure vessel ring with the same orientation as shown in Figure 3-4.

A cross-sectional view of the entire configuration 14 is given in Figure 3-9.

The side shield extension of configuration 14 was constructed with the addition of 20.32 cm of polyethylene side shield plus 5.08 cm of borated polyethylene sandwiched between two 0.635 cm thick Boral sheets (132.08 cm square x 0.3175 cm thick 35 w/o B₄C) which were permitted to rest diagonally against the PAX pressure vessel (at the top) and on the PAX test stand deck (at the bottom) and secured in position with steel tie-down cables. To prevent or minimize radiation streaming between the side shield and the top shield, the gap between the two shields was filled with a piece of polyethylene 5.08 cm x 10.16 cm x 121.92 cm and a 10.16 cm x 20.32 cm x 121.92 cm piece of polyethylene. The areal density of the top shield was 203.5 gm/cm². The areal density of the Boral-polyethylene combination side shield was 26.0 gm/cm².

3.2 ADDITIONAL TEST EQUIPMENT COMMON TO ALL TEST CONFIGURATIONS

3.2.1 Passive Dosimetry Systems

The passive dosimeters were chosen to provide information on the fast and thermal neutron flux densities and gamma dose rates. The following dosimeters were used throughout the experiments: dysprosium foils, exposed bare and cadmium-covered*, to obtain thermal neutron flux information; U-238 (190 ppm U-235) foils, cadmium covered to minimize the neutron fission of U-235, to obtain the fast neutron flux information for $E \approx 1.5$ MeV; sulfur pellets to obtain the fast neutron flux information for $E \approx 2.9$ MeV; and CaF₂:Mn hot pressed chip thermoluminescent dosimeters, exposed inside 0.0508 cm thick Tygon tubing, to obtain gamma dose.

Descriptions of the dosimeters, the reaction mechanism of the dosimeter and the methods of dosimeter readout are contained in Appendix A of WANL-TME-1915.

* All cadmium covers used were 0.0508 cm thick.

3.2.2 Active Dosimetry Systems

The active dosimeters were chosen to provide information on the fast neutron fluxes and absorbed doses, and gamma-ray absorbed dose rates. The following dosimeters were used throughout the experiments:

A WX-30748 fast fission detector, containing approximately 2 grams of U-238 having a U-235 content of 11 parts per million, was used to measure fast neutron flux densities for $E > 1.5$ MeV. This detector was operated with a wrapper of 0.3175 cm thick Boral (35 w/o B_4C) and 0.05 cm thick cadmium to minimize the response to U-235 fission.

A RSN-48 Hurst proportional counter, patterned after an Oak Ridge design was used to measure fast neutron absorbed dose for $E > 0.2$ MeV.

A RSG-8A carbon wall, CO_2 filled, ionization chamber was used to measure gamma absorbed dose rate in carbon.

Two Bragg-Gray type radiation detectors, sensitive volume of 1.3 cc ionization chambers and constructed of aluminum and beryllium walls were used to measure gamma ray absorbed dose rate in aluminum and beryllium.

Block diagrams of the instrumentation along with the chamber drawing for each of the above detectors are found in Appendix D.

3.2.3 (ρ, θ, ϕ) Traverse System

The (ρ, θ, ϕ) traverse system was designed as a remote controlled detector positioning system to obtain environmental radiation data as a function of spherical radius relative to the core center and as a function of polar angle.

Two sections of track permitted detectors to be traversed from the pressure vessel surface to 6.10 meters from the core center. The track was attached to a yoke arrangement mounted on the test stand which could pivot about the centerline of the reactor, and this provided the capability to vary the polar angle position of the "wishbone" A-frame by the remote control test cell crane.

The detectors for the particular traverse were mounted to an oak block (7.620 cm x 7.620 cm x 45.72 cm long) in turn mounted to a 0.3175 cm thick aluminum plate. The plate was mounted on a series of rack and pinion gears which provided the drive mechanism for the detector trolley. A Bodine motor of 1/50 horsepower and a speed of 7.2 rpm, coupled through a rack and pinion gear, provided the power. The Bodine motor was remotely controlled from the console. The rate of travel of the detectors was 5.08 cm/min. A steel tape was fastened to the inside rail of the I-beam from which a position indicator was mounted on the trolley system to determine the distance from core center. A closed-circuit TV camera was used to determine the position of the detectors with respect to the core center.

To determine the polar angle position of the track, a closed-circuit TV camera was mounted on the PAX reactor test stand. A protractor was mounted to the A-frame and pivot mounting plate. This provided remote determination of the angle with respect to the axial centerline of the PAX reactor. In addition to this system, which only measured the polar angle of the traversing track at a point close to the pivot point of the traversing mechanism, a means of measuring the polar angle of the detector at any location in the detector traverse was included in the system. This additional system was based on the use of a predetermined geometrical configuration of measuring tapes from which the polar angle could be determined using a geometrical relationship based on the Law of Cosines, a set of previously determined distance measurements, and the changing distances determined by the measuring tapes remotely monitored during the traversing of the detection system.

A further discussion of this system and its operation is contained in Section 4.7.2 of WANL-TME-1852 and DRM 51172.

3.2.4 Boral Enclosures

A Boral enclosure was used for the Dy-Al foil passive dosimetry measurements made in close proximity to the PAX reactor. This enclosure was used to reduce the test cell "room return" thermal background for measurements of the thermal neutron transmission through all shield configurations except 14. The Boral enclosure was fabricated from five Boral sheets (0.635 cm x 132.08 cm x 132.08 cm, 35 w/o B₄C) bolted to 0.159 cm x

4. 128 cm x 5.40 cm "Equipto" slotted steel angle. The open end of the cube was placed on top of the shield configurations after the dosimeters were placed into position. Due to a limited number of Boral sheets, the thickness was restricted to 0.635 cm. This enclosure is shown in Figure 3-10.

3.2.5 Nuclear Measurement Systems - Reactivity System

The reactivity measurements made in support of the Shield Provision Trade Study Experiments employ the use of positive period techniques. In using this method, the reactor period is determined for a given reactor configuration and then after a configuration change is made, a second period measurement is obtained at the same drum positions as the first. The two period measurements are then converted to reactivity using the "in-hour" relationship and the difference in the two reactivity values constitutes the effect of the configuration change. A more detailed description of the procedure and components used in the positive period system is given in Section 3.1.1 of WANL-TME-1910 while the "in-hour" equations and values of the constants used therein are found in Appendix D of WANL-TME-1910.

3.2.6 Nuclear Measurement Systems - Axial Fission Density and Power Calibration Measurement Systems

3.2.6.1 Uranium-Aluminum Wires

The axial fission density distribution within the PAX reactor was measured by irradiating 2.54 cm long, 0.102 cm in diameter uranium-aluminum wires containing 20 weight percent uranium. The U-235 enrichment is similar to the reactor fuel ($93.15 \pm 0.15\%$). A more detailed description of the wires, wire holders and radiation counting system used with the wires is contained in Section 3 of WANL-TME-1911.

3.2.6.2 Gold Wires

Au wires were used to monitor the PAX-GOC reactor power by exposing bare wires at a fixed location. The wire dimensions were 0.076 cm diameter by 2.540 cm in length with the material consisting of 24K gold. Additional details on the use of gold wires is found in WANL-TME-1915, Appendix B.

3.2.6.3 Fuel Element Slices

Details on the absolute power calibration techniques involving the use of fuel element slices and radiochemistry are contained in Appendix E.2 and Appendix B of WANL-TME-1915.

Figure 3-1. Photograph of Wall Liner Material in Place in the Vicinity of the PAX Reactor (Top View)

Figure 3-2. Photograph of Wall Liner Material in Place in the Vicinity of the PAX Reactor (Elevation View)

Figure 3-3. Photographs of Configuration 16A in Position on PAX-G0C Reactor

Figure 3-4. Plan View of Shield Position During Experimental Program

Figure 3-5. Cross Section of Configuration 16A, 16B and 16C--20.32, 30.48 and 38.10 cm BATH Mockup

NOTE:

- 1) All dimensions in centimeters.
- 2) Lateral dimensions: $\sim 132,0 \times 132,0$.
- 3) Letters A, B, and C refer to dosimeter holders.

Figure 3-6. Cross Section of Configuration 17 -- Borated Stainless Steel-Liquid Hydrogen Mockup

Figure 3-7. Photograph of Configuration 17 in Position on the PAX-GOC Reactor

NOTE:

- 1) All Boral 35 w/o B_4C .
- 2) All Borated-Polyethylene 2 w/o Boron.
- 3) Lateral dimensions: ~ 132.0 by 132.0 .
- 4) All dimensions in centimeters.

Figure 3-8. Cross Section of the Top Shield of Configuration 14 -- The Background Shadow Shield

- NOTE: 1) All Boral 35 w/o B_4C .
2) All dimensions in centimeters.

Figure 3-9. Cross Section of Configuration 14 -- Background Shadow Shield.

Figure 3-10. Photograph of Boral Enclosure

4.0 TEST RESULTS

4.1 REACTIVITY MEASUREMENTS

4.1.1 Narrative

The reactivity worths of the various shield configurations, added to the nozzle end of the PAX-G0C reactor, were measured with the period-pickoff system previously mentioned in Section 3.2.5. In accordance with WANEF operating procedures (WANL-TME-267, Revision H), each shield configuration was assembled in several steps in order to insure that no single reactivity change would be greater than + 50¢; thus, a large number of reactor runs were required to gather the reactivity data. DRM-51168 contains summaries of the changes made to the PAX-G0C reactor and shield configurations, the associated drum bank angles, excess reactivity and the associated methods involved.

The reactivity measurements and procedures were carried out in accordance with procedures set forth in Appendix A. All of the measurements were performed with the critical reactor at room temperature.

4.1.2 Test Results

The measured reactivity changes in cents due to the addition of shield configurations 14, 16A, 16B, 16C and 17 to the nozzle end of the PAX-G0C reactor are shown in Table 4-1. The worth of the 3.81 cm thick aluminum support plate mockup and the 0.3175 cm thick Boral (35 w/o B₄C) poison plate mockup are included in the total worth of each shield configuration.

TABLE 4-1. REACTIVITY WORTHS FOR VARIOUS SHIELD CONFIGURATIONS
USED ON THE PAX-G0C REACTOR

<u>Configuration</u>	<u>Type</u>	<u>Approximate Areal Density (gms/cm²)</u>	<u>Reactivity Change* (%)</u>
14	"Thick" Top Side	219 26	38.20 } 4.38 } 42.6 ± 0.8
16A	8" BATH	~ 69	47.8 ± 0.6
16B	12" BATH	~ 89	48.1 ± 0.7
16C	15" BATH	~ 109	48.1 ± 0.8
17	LH ₂ -BSS	~ 117	56.5 ± 0.9

* Uncertainties are ± 1σ.

The individual uncertainties associated with these measurements are reported in Appendix B, Section B. 1.

4.2 SHIELD INTERNAL AND EXTERNAL RADIATION ENVIRONMENTAL MEASUREMENTS WITH PASSIVE DOSIMETRY

4.2.1 Narrative

The radiation measurements reported in this section were obtained with the following passive dosimetry:

Sulfur pellets and U-238 (190 ppm U-235) foils for fast neutron flux density data, CaF₂:Mn thermoluminescent dosimeters (TLD's) for gamma ray-absorbed dose rate data, and

Bare and cadmium-covered dysprosium foils for thermal neutron flux density, saturated activity, and cadmium ratio data.

Additional details on each of the above mentioned dosimeter systems, thresholds, reactions, etc., are contained in Appendix A of WANL-TME-1915. The measurements were made on the reactor axial centerline and along radial traverses from core center along an azimuth located at $\phi = 270^\circ$ at various elevations from core midplane.

The results of the various passive dosimetry reactor runs were inter-related by normalizing all of the reported results to a watt of reactor power. The details involved in determining the PAX-GOC absolute power calibration, the effect of the various shield configurations on the axial fission distribution and inter-run reactor calibrations are all contained in Appendix E. Approximately 1100 data points were obtained in support of these measurements and all of the data have been reported in tabular, as well as plotted form in DRM 51185. The measurements were carried out in accordance with the procedures set forth in Appendix A.

All of the tests were performed with the critical reactor at room temperature and all of the measurements were performed at the projected R-1 cold critical drum angle of $115.0^\circ \pm 1.0^\circ$.

The passive dosimeters used for measurements made on shield configuration 15 (bare core) and external to shield configurations 16A, 16B, 16C and 17 were mounted in position using an aluminum "Christmas Tree" arrangement which was positioned on the core axial centerline. This arrangement is similar to the system utilized in WANL-TME-1852, Section 5.3.1, for the same purpose. Because of the physical arrangement of the passive dosimetry, all of the dosimeters were located at slightly off axis positions (approximately 5.0 cm radius from the core centerline) to avoid dosimeter self-shielding.

The passive dosimeter measurements made internal to shield configurations 16A, 16B, and 16C were located between the aluminum spacer bars used to separate the BATH shield mockup modules. The dosimetry was mounted on 5.08 cm wide by 66.04 cm long strips of manila cardboard and held in position with tape. This system allowed the positioning of the dosimetry, at selected radial locations ($\phi = 270^\circ$) as well as near the core centerline, between each of the BATH shield mockup modules making up the particular shield configuration (i.e., 16A, 16B, or 16C).

The passive dosimetry measurements made internal to shield configuration 17 were mounted in recessed sections of the polyethylene shield material as shown in EDC 67 in Appendix A. Measurements were obtained at selected radial locations ($\phi = 270^\circ$) as well as near the core centerline.

All of the data were reduced using the data reduction techniques described in Appendix C.

4.2.2 Test Results

The passive dosimetry measurements are summarized by the representative axial centerline plots contained in this section. In addition to these plots, radial plots and non-centerline axial plots along with tables of all of the data obtained have been reported in DRM-51185.

No attempt was made to determine the background radiation environment with the passive dosimetry due primarily to the low fluxes/doses and resulting low activities or dose rates. Thus, direct comparisons of passive dosimetry from configuration 15 data or data at the top of shield configurations 16A, 16B, 16C and 17 with active dosimetry from Section 4.3.2 is not recommended. All elevations are measured from core center. The plots include a one-sigma error bar on each data point; however, the individual components which contribute to the overall uncertainty of each measurement are described in Appendix B, Section B.3.

4.2.2.1 Fast Neutron Flux Density Measurements

Figures 4-1 and 4-2 show the axial centerline distributions of fast neutron flux density obtained with U-238 foil and sulfur pellet dosimetry, respectively, external to shield configuration 15 and internal to shield configurations 16A, 16B, 16C and 17. Figures 4-3, 4-4, and 4-5 show cross plots of both the U-238 and sulfur dosimetry data together for shield configurations 15, 16C, and 17, respectively.

4.2.2.2 Gamma Ray-Absorbed Dose Rate Measurements

Figure 4-6 shows the axial centerline distribution of absorbed dose rate obtained with $\text{CaF}_2\text{:Mn}$ TLD's external to shield configuration 15 and internal to shield configurations 16A, 16B, 16C and 17. Good agreement is obtained between the CaF_2 TLD data and the carbon wall, CO_2 filled detector for the case, where the background environment is small compared to the direct radiation, of configuration 15, bare core.

4.2.2.3 Thermal Neutron Flux Density Measurements

Figure 4-7 shows the axial centerline distribution of the thermal neutron flux density (bare foil minus cadmium-covered dysprosium foil activity) internal and external to shield configurations 15, 16A, 16B, 16C and 17 and internal to shield configurations 16A, 16B, 16C and 17. All of the above data were obtained within a Boral enclosure, described in Section 3.2.4, located above the reactor or shield surface and placed over the top of the dosimeter support stand. Figure 4-8 shows the plot of the cadmium ratio (bare foil data divided by cadmium-covered foil data).

4.3 DIRECT EXTERNAL RADIATION MEASUREMENTS WITH ACTIVE DOSIMETRY

4.3.1 Narrative

The radiation measurements reported in this section were obtained with the following active dosimetry:

U-238 (11 ppm U-235) fast fission detector, shielded from thermal neutrons with a 0.3175 cm thick Boral (35 w/o B₄C) and 0.05 cm thick cadmium wrapper, for fast neutron flux density data,

Hurst proportional counter for fast neutron absorbed dose rate data, and

Three "Bragg-Gray" type detectors with carbon, aluminum, and beryllium walls for gamma ray absorbed dose rate data.

Additional details on each of the above mentioned dosimeter systems are contained in Appendix D. The measurements were made on the reactor axial centerline and radially from core center along an azimuth located at $\theta = 270^\circ$ at various elevations from core midplane.

The results of the active dosimetry reactor runs were inter-related by normalizing all of the reported results to a watt of reactor power. The details involved in determining the inter-run reactor calibration through the use of the digital reactor power monitor and the actual absolute power calibration are contained in Appendix E. Approximately 1500

data points were obtained in support of these measurements and the data have been reported in tabular, as well as plotted form in DRM 51183. The measurements were carried out in accordance with the procedures set forth in Appendix A.

All of the tests were performed with the critical reactor at room temperature and all of the measurements were performed at the projected R-1 cold critical drum angle of $115.0^\circ \pm 1.0^\circ$.

All of the active dosimetry measuring devices were mounted on the (ρ, θ, ϕ) traverse system which remotely positioned the detector systems at both the ρ and θ positions. The ϕ coordinate was permanently fixed at $\phi = 270^\circ$ for all of the measurements. This system is described in detail in Section 3.2.3.

Active dosimetry measurements were obtained on the axial centerline ($\theta = 0^\circ$) and at $\theta = 10^\circ$, 20° , and 30° for a core ray distance from close to the configuration surface to a distance of approximately 600 cm from core center for all configurations. Measurements were also obtained for a fixed elevation of 474.0 cm (common plane height) as a function of polar angle from $\theta = 0^\circ$ to $\theta = 30^\circ$ for all configurations. The boral enclosure was removed for active dosimetry measurements.

All of the data were reduced using either the CDC-6600 computer on the programmable WANG desk calculator in conjunction with the data reduction techniques described in Appendix C.

4.3.2 Test Results

The direct active dosimetry measurements are summarized by the representative plots contained in this report. All of the active dosimetry measurements have been reported in DRM-51183. The data have been plotted for four polar angle traverses (0° —the axial centerline data, 10° , 20° , and 30°), and a traverse at the common plane elevation (474 cm) versus polar angle. In addition, a traverse at elevation 148 cm versus polar angle was made for the three gamma ray-absorbed dose rate detectors only.

The direct data are obtained by subtracting the measured background data, obtained from the use of shield configuration 14 and described in Section 4.4, from the data measured above shield configurations 15, 16A, 16B, 16C and 17.

The plots include a one-sigma error bar on representative data points; however, the individual components which contribute to the overall uncertainty of each measurement are described in Appendix B, Section B.4.

4.3.2.1 Direct Fast Neutron Flux Density Measurements

Figures 4-9 through 4-13 show the polar angle traverses and common plane distributions of the direct fast neutron flux density data obtained with the U-238 fission detector. These figures are composite plots of the data from each shield configuration (i.e., 15, 16A, 16B, 16C and 17).

4.3.2.2 Direct Fast Neutron Absorbed Dose Rate Measurements

Figures 4-14 through 4-18 show the polar angle traverses and common plane distributions of the direct fast neutron absorbed dose rate data obtained with the Hurst proportional counter. These figures are composite plots of the data from each shield configuration, except shield configuration 16C. No acceptable data were obtained for configuration 16C due to failure of the detector system during this measurement series.

4.3.2.3 Direct Gamma Ray Absorbed Dose Rate Measurements

Figures 4-19 through 4-24 show the polar angle traverses, 148 cm elevation traverse and common plane distributions of the direct gamma ray absorbed dose rate data obtained with the carbon wall, CO₂ filled detector. These figures are composite plots of the data from each shield configuration.

The three gamma ray absorbed dose rate detectors were mounted as a unit on the (ρ, θ, ϕ) detector traversing mechanism and are shown in Figure 4-25. All detector positions were referenced to the center detector position since the maximum detector separation was only 2.0 cm.

Table 4-2 presents the direct data obtained with both the aluminum and beryllium Bragg-Gray detectors ratioed to the direct data obtained with the carbon wall, CO₂ filled detector. These data are only presented for shield configuration 15 due to the extremely low signals obtained from the aluminum and beryllium detectors which resulted in marginal data when shield configurations 16A, 16B, 16C and 17 were in place.

4.4 EXTERNAL BACKGROUND RADIATION MEASUREMENTS

4.4.1 Narrative

The radiation measurements reported in this section were used as background data and subtracted from the appropriate detector data to produce the direct data previously reported in Section 4.3. These data are presented here to indicate the magnitude of the background problem when compared to the various direct shield configuration data. These data were obtained with the shield configuration 14 on the reactor. The physical construction of this shield configuration has been discussed in detail in Section 3.1.3.5. This shield was designed to obtain the maximum attenuation for photons and neutrons above the aft reactor core within the existing weight restrictions placed on the reactor/test stand arrangement as well as preserving a close physical geometry to the BATH shield configuration 16C. In addition, a side shield for background measurements was also utilized. However, this shield was restricted primarily to a neutron attenuation shield with little photon attenuation due to the weight restrictions imposed on the reactor test stand and floor. All of the measurements were carried out in accordance with the procedures set forth in Appendix A and all of the measurements were performed under the same test conditions and were reduced in the same manner, as described in Section 4.3.

4.4.2 Test Results

The active dosimetry measurements of the external background data are summarized by the representative plots in this section. These plots are either composite plots of the 0° , 10° , 20° , and 30° polar angle traverses or the common plane traverse.

Figures 4-26 and 4-27 are plots of the fast neutron flux density background data, while Figures 4-28 and 4-29 are plots of the fast neutron absorbed dose rate background data. Examination of the data from Figures 4-26 and 4-28 shows the problems inherent with shielding only a limited portion of the reactor and trying to make an environmental background

measurement. Close to the shield the attenuation of the direct radiation is good for all polar angles; however, as the detector system is traversed away from the background shield, the scattered radiation from the unshielded reactor portions and test cell walls becomes predominant and thus follows a relatively " $1/r$ " fall-off as the distance is increased from the reactor/test cell wall scatter areas. The uniformity of the data from these figures indicates that both the top and side background shield appear to be effective for fast neutron attenuation at all polar angles.

Figures 4-30 and 4-31 present the gamma ray absorbed dose rate data obtained from the carbon wall, CO_2 -filled detector. From Figure 4-30 it can be seen that while the 0° and 10° polar angle traverses follow a similar curve shape, the 20° and 30° polar angle traverses do not, with the 30° traverse having almost a factor of 3 higher data near the background shield surface than the 0° or 10° traverses. The reasons presented in the preceding paragraph for the curve shapes of the 0° and 10° traverse are also valid for this data. However, it is seen, as was discussed in Section 4.4.1, that the side background shield is not effective in the reduction of direct photon radiation with the resulting factor of 3 increase in the measured background. Thus, the direct gamma ray absorbed dose rate data for 20° and 30° polar angle traverse are not believed to be nearly as accurate as the 0° and 10° traverses, and its use for analysis purposes should be questioned.

The aluminum and beryllium Bragg-Gray detector data are very similar in magnitude to the data presented in Figures 4-30 and 4-31 and thus will not be shown; however, the data are contained in DRM-51183 and the preceding discussion for the carbon wall, CO_2 -filled detector is valid for this data also.

The problem in making an accurate determination of the background radiation environment external to heavy shield mockups must be resolved. It is clear that background problems affect the data near the reactor axial centerline at large distances from shield configuration 16C for both the neutron and photon measurements. Serious problems also exist along the 30° polar angle traverse for the photon measurements for all configurations.

In addition, the considerable geometrical differences between the background shield configuration and the borated steel-liquid hydrogen shield mockup (configuration 17) contributed significantly to the problems of obtaining meaningful direct radiation data for nearly all photon and neutron measurements on this configuration.

It is of paramount importance that all approaches toward an accurate determination as well as reduction of the background radiation environment be thoroughly examined. Special emphasis on future measurements should be placed on the use of detector collimators for external radiation measurements to significantly reduce the background problem.

Figure 4-1

Axial Centerline Fast Neutron Flux Density ($E > 1.5$ MeV) Distribution Obtained With U-238 Foils External to Configuration 15 and Internal to Configurations 16A, 16B, 16C and 17

AXIAL SULFUR FAST FLUX DISTRIB. AT R = 0

Figure 4-2

Axial Centerline Fast Neutron Flux Density ($E \geq 2.9$ MeV) Distribution Obtained With Sulfur Pellets External to Configuration 15 and Internal to Configurations 16A, 16B, 16C and 17

Figure 4-3

Axial Centerline Comparison of Fast Neutron Flux Density Obtained
With U-238 Foil and Sulfur Pellet Dosimetry for Shield Configuration 15

AXIAL S AND U FAST FLUX AT R = 0 FOR 16C

Figure 4-4

Axial Centerline Comparison of Fast Neutron Flux Density Obtained With U-238 Foil and Sulfur Pellet Dosimetry for Shield Configuration 16C

Figure 4-5

Axial Centerline Comparison of Fast Neutron Flux Density Obtained
With U-238 Foil and Sulfur Pellet Dosimetry for Shield Configuration 17

AXIAL CaF_2 GAMMA DOSE RATE DISTRIB. $R = 0$ INCH

GRID NUMBER 79

Figure 4-6

Internal and External Axial Centerline Gamma Ray-Absorbed Dose Rate Distribution Obtained With $\text{CaF}_2\text{:Mn}$ TLD's for Configurations 15, 16A, 16B, 16C and 17

AXIAL THERMAL NEUTRON FLUX DISTRIB. AT R = 0

GRID NUMBER 59

Figure 4-7

Internal and External Axial Centerline Thermal Neutron Flux Density Distributions ($E < 0.4$ eV) Obtained With Dy-Al Foils for Configurations 15, 16A, 16B, 16C and 17

AXIAL DY CADMIUM RATIO DISTRIB. AT R = 0

Figure 4-8

Internal and External Axial Centerline Cadmium Ratio Distribution
Obtained With Dy-Al Foils for Configurations 15, 16A, 16B, 16C and 17

Figure 4-9
An Axial Centerline Distribution of the Direct U-238 Fast Neutron Flux
Density ($E > 1.5$ MeV) External to Configurations 15, 16A, 16B, 16C and 17

REP 1 PAX ANGLE 270000 E+03 MCSPA

Figure 4-10

A 10° Polar Angle Distribution of the Direct U-238 Fast Neutron Flux Density
(E > 1.5 MeV) External to Configurations 15, 16A, 16B, 16C and 17

Figure 4-11

A 20° Polar Angle Distribution of the Direct U-238 Fast Neutron Flux Density
(E > 1.5 MeV) External to Configurations 15, 16A, 16B, 16C and 17

REP 1 PAX ANGLE 270000 E+03 MCSPA

Figure 4-12

A 30° Polar Angle Distribution of the Direct U-238 Fast Neutron Flux Density
(E > 1.5 MeV) External to Configurations 15, 16A, 16B, 16C and 17

Figure 4-13

A Radial Distribution at the Common Plane of the Direct U-238 Fast Neutron Flux Density ($E > 1.5$ MeV) External to Configurations 15, 16A, 16B, 16C and 17

Figure 4-14

An Axial Centerline Distribution of the Direct Hurst Fast Neutron Absorbed Dose Rate ($E > 0.2$ MeV) External to Configurations 15, 16A, 16B, 16C and 17

Figure 4-15

A 10° Polar Angle Distribution of the Direct Hurst Fast Neutron Absorbed Dose Rate ($E > 0.2$ MeV) External to Configurations 15, 16A, 16B, 16C and 17

Figure 4-16

A 20° Polar Angle Distribution of the Direct Hurst Fast Neutron Absorbed Dose Rate ($E > 0.2$ MeV) External to Configurations 15, 16A, 16B, 16C and 17

Figure 4-17

A 30° Polar Angle Distribution of the Direct Hurst Fast Neutron Absorbed Dose Rate ($E > 0.2\text{MeV}$) External to Configurations 15, 16A, 16B, 16C and 17

Figure 4-18

A Radial Distribution at the Common Plane of the Direct Hurst Fast Neutron Absorbed Dose Rate ($E > 0.2$ MeV) External to Configurations 15, 16A, 16B, 16C and 17

Figure 4-19

An Axial Centerline Distribution of the Direct Gamma Absorbed Dose Rate Obtained With A Carbon Wall CO₂-Filled Ion Chamber External to Configurations 15, 16A, 16B, 16C and 17

REP 2 PAX ANGLE 270000 E+03 MCSPA

Figure 4-20

A 10° Polar Angle Distribution of the Direct Gamma Absorbed Dose Rate Obtained With A Carbon Wall CO₂-Filled Ion Chamber External to Configurations 15, 16A, 16B, 16C and 17

Figure 4-21

A 20° Polar Angle Distribution of the Direct Gamma Absorbed Dose Rate Obtained With A Carbon Wall CO₂-Filled Ion Chamber External to Configurations 15, 16A, 16B, 16C and 17

REP 2 PAX ANGLE 270000 E+03 MCSPA

Figure 4-22

A 30° Polar Angle Distribution of the Direct Gamma Absorbed Dose Rate Obtained With A Carbon Wall CO₂-Filled Ion Chamber External to Configurations 15, 16A, 16B, 16C and 17

Figure 4-23

A Radial Distribution of the Direct Gamma Absorbed Dose Rate Obtained With A Carbon Wall CO₂-Filled Ion Chamber Near the Top Surface of Configurations 15, 16A, 16B, and 16C

Figure 4-24

A Radial Distribution at the Common Plane of the Direct Gamma Absorbed Dose Rate
Obtained With A Carbon Wall CO₂-Filled Ion Chamber External to Configurations
15, 16A, 16B, 16C and 17

Figure 4-25
A View of the Three Gamma Ray Dose Rate Detectors
Mounted on the (ρ, θ, ϕ) Traverse System

Figure 4-26
Fast Neutron Flux Density Data for Configuration 14 for Polar Angles
0°, 10°, 20°, and 30° Traverses

Figure 4-27
Fast Neutron Flux Density Data for Configuration 14 for the Common Plane Traverse

Figure 4-28

Fast Neutron Dose Rate Data for Configuration 14 for
Polar Angles 0°, 10°, 20° and 30° Traverses

Figure 4-29

Fast Neutron Dose Rate Data for Configuration 14 for the Common Plane Traverse

Figure 4-30

Gamma Ray-Absorbed Dose Rate Data for Configuration 14 for
Polar Angles 0°, 10°, 20°, and 30° Traverses

Figure 4-31

Gamma Ray-Absorbed Dose Rate Data for Configuration 14 for the Common Plane Traverses

TABLE 4-2

RATIOS OF EXTERNAL ENVIRONMENTAL (DIRECT PLUS BACKGROUND) ABSORBED
GAMMA DOSE DATA OBTAINED WITH THE CARBON WALL, CO₂ FILLED DETECTOR AND
THE AL AND BE BRAGG-GRAY DETECTORS

PAX-G0 Run Number	Configuration	PAX Angle (Degrees)	Radius* (cm)	Elevation (cm)	Polar Angle (Degrees)	Core Ray** (cm)	Ratio	
							CO ₂ /Al Bragg-Gray	CO ₂ /Be Bragg-Gray
92 ↓	15 ↓	270 ↓	0 ↓	148.0	.2	148.0	1.045	1.027
				152.4	.2	152.4	1.038	1.026
				167.6	.2	167.6	1.028	1.073
				182.9	.1	182.9	1.029	1.092
				198.1	-0	198.1	1.042	1.138
92 ↓	15 ↓	270 ↓	0 ↓	213.4	-0	213.4	1.050	1.147
							$1.039 \pm .004$	$1.084 \pm .023$ Avg
92 ↓	15 ↓	270 ↓	26.9 ↓	150.0	10.1	152.4	1.047	1.072
				29.3	10.2	167.6	1.024	1.075
				31.9	10.1	182.9	1.030	1.081
				34.3	10.0	198.1	1.014	1.092
				210.2	9.9	213.4	1.034	1.140
92 ↓	15 ↓	270 ↓	52.9 ↓	142.9	20.3	152.4	1.039	1.048
				58.0	20.2	167.6	1.021	1.061
				63.0	20.2	182.9	1.026	1.076
				68.0	20.1	198.1	1.016	1.083
				200.4	20.1	213.4	1.019	1.071
92 ↓	15 ↓	270 ↓	73.2 ↓				$1.024 \pm .004$	$1.071 \pm .008$ Avg
92 ↓	15 ↓	270 ↓	85.2 ↓	144.4	30.6	167.6	1.025	1.110
				92.7	30.5	182.9	1.023	1.091
				100.2	30.4	198.1	1.027	1.108
				107.8	30.4	213.4	1.002	1.109
							$1.019 \pm .007$	$1.104 \pm .005$ Avg

NOTE: Ratio is the ratio between Rads(C)/Rads(Al) and Rads(C)/Rads(Be).

* Radial distance from core centerline.

** Distance from core center.

5.0 ANALYSIS OF PAX-GO SHIELD EXPERIMENTS

5.1 INTRODUCTION

The comparison of calculated results with good experimental data is an essential step in the development of an analytical technique for predicting radiation levels in and around a complex reactor-shield assembly. Only through comparisons of this sort, is it possible to accurately assess the capabilities of a given analytical technique; and, in turn, to confidently apply that technique to a design situation. Thus, in keeping with this philosophy, an in-depth analytical evaluation of the experimental data presented in Section 4.0 was performed using the WISDM⁽¹⁾ (Westinghouse Integrated Standard Design Method) system.

In Section 5.2, calculated neutron and photon radiation levels internal to the BATH as well as the borated steel-liquid hydrogen shield mockups are compared with the experimental data. Calculated results from both one and two-dimensional discrete ordinates techniques are presented.

A comparison of calculated and measured radiation levels external to the shield mockups is presented in Section 5.3. Analytical results external to the various shield mockups were generated using a discrete ordinates technique as well as the point kernel approach.

In Section 5.4, the comparisons of calculations and measurements both internal and external to the shield mockups are discussed; uncertainty factors to be applied to the various analytical approaches are developed.

5.2 ANALYSIS OF INTERNAL DATA

A schematic diagram of the PAX-GOC reactor assembly is shown in Figures 5-1 and 5-2. Figure 5-1 shows a sectional view through the reactor axis, whereas, Figure 5-2 depicts a top view of the system. As was stated in Section 3.0, experimental data were

obtained internal to four shield mockups placed in the position shown in Figure 5-1. Schematic drawings of the BATH (configs. 16A, 16B, and 16C) and the borated steel - liquid hydrogen (config. 17) shield mockups are shown in Figures 5-3 and 5-4, respectively. With the exception of the reactor core, atom densities for each of the reactor regions shown in Figure 5-1 are presented in Table 5-1. Atom densities for the various shield regions depicted in Figures 5-3 and 5-4 are given in Table 5-2. Core atom densities are not presented since this is an unclassified report. It may be noted, in Table 5-2, that atom densities are presented for a "smeared" BATH module. The reason for considering the BATH module as both a homogeneous and a heterogeneous region will be discussed later in Section 5.0.

The experimental data taken internal to the BATH shield mockups were presented in DRM51185, in the form of radial and axial distributions of equivalent neutron flux and photon dose rate at various locations within the mockup. These axial locations are defined by the ten dosimeter slots illustrated in Figure 5-3. Data obtained internal to the borated steel - liquid hydrogen mockup were presented in a similar fashion in DRM51185. The twelve axial data locations in the borated steel - liquid hydrogen mockup are indicated in Figure 5-4. From Figures 5-3 and 5-4, it may be noted that, while the axial dosimeter locations in the BATH mockups are defined by discrete void spaces between shield modules, those in the borated steel - liquid hydrogen mockup appear to be imbedded within the polyethylene slabs. The dosimeters utilized in the borated steel - liquid hydrogen shield were, in fact, located in recesses cut in the polyethylene. For a more detailed description of these dosimeter locations, the reader is referred to Appendix A of this report.

In order to make a direct quantitative comparison with all of the internal data presented in DRM51185, it is, of course, necessary to perform a two-dimensional analysis of the various shield mockups. However, for many design considerations, cost limitations prohibit the use of the two-dimensional technique and the analyst, therefore, must rely on the less expensive and less accurate one-dimensional approach. Thus, it is extremely important to compare results of not only the two-dimensional technique, but also the one-dimensional method, with the experimental data. The one-dimensional analysis is described in Section 5.2.2.

5.2.1 Two-Dimensional Discrete Ordinates Analysis

The DOT-IIW⁽²⁾ two-dimensional S_n transport code was employed to analyze two of the BATH shield mockups (configs. 16A and 16C) as well as the borated steel-liquid hydrogen shield mockup (config. 17). The intermediate thickness BATH shield mockup (config. 16B) was omitted from the two-dimensional phase of the analysis because it was concluded that sufficient information was garnered from the analysis of the two extreme thicknesses of BATH mockup; therefore, the additional time required to analyze a third BATH configuration was unwarranted.

5.2.1.1 Method of Analysis

The analytical procedure employed to calculate the neutron and photon radiation levels internal to the various shield mockups is shown schematically in Figure 5-5. In addition to depicting the flow of information throughout the calculation, Figure 5-5 illustrates the type of information derived from each computation as well as the sequence of calculations. The sequence numbers shown in Figure 5-5 will be employed later to facilitate discussion of the interconnection between various phases of the internal calculation and the external calculation.

In discussing the analytical procedure outlined in Figure 5-5, it is appropriate to describe the neutron analysis and photon analysis as separate entities. Although the basic approach described in the following paragraphs was applied to the analysis of both the neutron and photon internal environment, there are enough differences in the calculational procedure that a complete discussion of each approach is desirable.

Basically, the neutron transport analysis was carried out in two parts. The first part of the procedure was an eigenvalue calculation on the basic reactor assembly. The basic reactor is defined in this study as that portion of the assembly which extends from the pressure vessel dome ($z = 0.00$) to an axial plane located a short distance into the shield mockup (z dependent on the shield mockup being analyzed). From the eigenvalue problem, the neutron flux distributions throughout the basic reactor assembly were obtained, as well as an angular and energy dependent boundary source, based on the angular flux distribu-

tions at the midplane of the shield support plate ($z = 221.536$). Using this boundary source as a coupling mechanism, a second DOT calculation was employed to obtain flux distributions throughout the shield geometry. By performing the analysis in two parts, a rather large computer cost savings was realized. The eigenvalue calculations on the basic reactor required multiple (typically 5-10) outer iterations due to the criteria for source convergence. However, since the analysis of the shield geometry employed a fixed boundary source, only a single outer iteration was needed. Had the shield configurations been analyzed along with the basic reactor in a single problem, the multiple outer iterations necessary for eigenvalue convergence would, of course, have encompassed a much larger number of spatial mesh points; and the running time would have been increased accordingly. In addition, since two of the shield mockups analyzed were merely different thicknesses of the same material, it was possible to employ one boundary source for both shield geometries; and, by eliminating one reactor calculation, to effect a further cost savings. The analytical procedure outlined above will be discussed in more detail in connection with the analysis of the various shield mockups.

As can be seen from Figure 5-5, the GAMB1T⁽³⁾ code was employed to prepare cross-sections for the internal neutron environment analysis. Based on the atom densities presented in Tables 5-1 and 5-2, and in the reactor core, region dependent, 16 group, macroscopic P_0 transport corrected neutron cross-sections were generated for use in the subsequent transport calculations. These cross-sections were weighted over region dependent 155 fine group spectra which had been generated for use in prior PAX-G0 analysis. Since the magnetic tape containing these spectra is not generally available, it would be necessary for an analyst attempting to reproduce this study to employ the GAMB1T-ANISN⁽⁴⁾ - ANISIG⁽⁵⁾ linkage, described in Reference 1, in his cross-section generation procedure. Although Figure 5-5 indicates that the macroscopic neutron cross-sections flow through the APPROP0S⁽⁶⁾ code, this step was unnecessary for the two-dimensional analysis. Macroscopic neutron cross-sections are indeed required as input to the APPROP0S code. However, these cross-sections are only used to generate data for use in simultaneous

neutron-photon calculations. Analyses of this type will be discussed later in connection with the one-dimensional analysis of the PAX-GOC shield experiments. Thus, in the case of the two-dimensional analysis, the macroscopic neutron cross section output from APPROPOS was identical to the input.

Prior to being employed in the two-dimensional S_n analysis, the macroscopic neutron cross-section output from APPROPOS was input to two ANISN one-dimensional S_n transport calculations. The purpose of the one-dimensional calculations was twofold. First, by running a radial calculation characteristic of the reactor regions centered about the core midplane and an axial calculation including all regions located on the reactor axis, an initial flux guess for use in the two-dimensional S_n calculations was obtained. Second, the ANISN calculations were employed to effect upscatter removal from the neutron cross-sections. By removing upscatter from the cross-sections, a large cost savings is realized in the subsequent two-dimensional calculations. Furthermore, since the upscatter removed cross-sections were weighted over the region dependent flux calculated by ANISN, it is believed that no significant loss of accuracy was incurred by employing the upscatter removed cross sections in the DOT calculations.

The spatial mesh diagrams employed to analyze the two basic reactor configurations are shown in Figures 5-6 and 5-7. The mesh diagram shown in Figure 5-6, which was utilized in connection with the BATH shield mockups (configs. 16A and 16C), extended axially from the pressure vessel dome to a plane located approximately midway through the second BATH module. This problem geometry contained 3550 spatial mesh arranged in an array of 50 radial by 71 axial mesh cells. The dimensions of the radial and axial mesh lines shown in Figure 5-6 are presented in Tables 5-3 and 5-4. The mesh cell description of the basic reactor geometry employed in conjunction with the borated steel - liquid hydrogen shield mockup (config. 17) is given in Figure 5-7. The axial dimensions of the basic reactor, in this case, extended from the pressure vessel dome up to and including the third borated steel plate ($z = 246.616$). This problem encompassed 3850 spatial mesh arranged in a 50 radial by 77 axial array of cells. The radial and axial mesh line dimensions for this geometry are given in Tables 5-3 and 5-5,

respectively. A description of the material numbers shown in Figures 5-6 and 5-7 is given in Table 5-6.

In regard to the mesh cell diagrams presented in Tables 5-6 and 5-7, a few words of explanation are needed. From Figures 5-1 and 5-2, it can be noted that, while the PAX reactor is cylindrical in shape, the various shield mockups consisted of square slabs stacked atop the reactor. Thus, the shield mockups overhang the reactor assembly. The amount of overhang ranges from 2.007 cm. on the side of the shield to 29.493 cm. on the corners. It should be noted that the experimental data were taken along radial traverses which extended to the side of the shield mockup. Thus, in mocking up the reactor - shield geometry in an R-Z DOT problem, it was assumed that the system could be described as a cylinder of outer radius equal to one half the length of the side of the shield mockup ($r = 66.358$ cm). In mocking up the system in this manner, it was, of course, necessary to include a small void region external to the side boron liner and below the shield mockup. More important, however, was the necessity to delete the corner regions of the shield mockups from the analysis. It may be argued that neglecting the corners of the shield would tend to lower the calculated radiation levels internal to the shield mockup due to the loss of a scattering medium at the shield periphery; and, in the case of photons, to the omission of secondary photons born in those corners. Nevertheless, for the analysis, it was assumed that the omission of the corners of the shield mockup produced a negligible effect on calculated radiation levels internal to the various mockups.

As was stated earlier, the neutron transport analysis employed P_0 transport corrected cross-sections in a 16 energy group problem. The energy group structure used in the analysis is given in Table 5-7. The final eigenvalue calculation on both of the basic reactor configurations utilized S_8 angular quadrature. However, as indicated by Figure 5-5, the S_8 calculations were preceded by calculations which employed S_2 angular quadrature. It has been recommended⁽²⁾ that a computer cost savings is achieved by running a low order quadrature DOT problem to obtain a good flux guess for the calculation employing the higher order quadrature. Using this S_2 flux guess in the S_8 problem, allows eigenvalue and point-wise flux convergence in a minimum of computer time.

The boundary source to be used as input to the S_n calculations on the shield geometry was obtained from the basic reactor problem by saving, on magnetic tape, the angular flux for each of the three axial mesh intervals describing the shield support plate (see Figures 5-6 and 5-7). The angular flux obtained in the central axial interval was then input from tape to the appropriate shield calculation. In order to obtain a valid boundary source for input to subsequent calculations it is imperative for the angular flux at the boundary source location to accurately reflect the presence of the particular shield configuration under consideration. For this reason, the basic reactor geometry was extended beyond the axial plane at which the boundary source was obtained. Likewise, the change in shield material from BATH to borated steel - liquid hydrogen necessitated the calculation of an additional basic reactor configuration; and, thus two distinct boundary sources were obtained for use in the analysis of the shield mockups.

In applying the boundary source option of the WANL version of the DOT code, the boundary source itself must be input at the top of the problem geometry. In this analysis, therefore, it was necessary to invert the shield geometry with respect to the basic reactor. This manipulation of geometries presents no serious problems other than those of the "book-keeping" variety.

The mesh cell diagrams employed in the neutron analysis of configurations 16C, and 16A are given in Figures 5-8 and 5-9, respectively. The reader will recall, from Figure 5-3, that each BATH module was constructed by laminating a series of titanium, aluminum, polyethylene, and boral plates. In the case of configuration 16C, the mesh sizing requirements (see Reference 2) coupled with the large number of physical regions resulted in a problem whose size is beyond the capabilities of the computer system that is available to WANL (CDC6600 with 91,000 data locations available after DOT program is loaded). Thus, it was necessary either to relax the mesh sizing requirements or to homogenize the individual BATH modules. Of these two alternatives, the latter was chosen for the analysis. Therefore, the mesh diagram presented in Figure 5-8 reflects the homogenization of the BATH modules and, thus, depicts the shield mockup as nine distinct modules separated by dosimeter slots.

The configuration 16A mockup, it may be noted, was made up of only five modules and, thus, contained 28 fewer regions than the configuration 16C mockup. Due to the smaller number of physical regions and the corresponding decrease in the number of axial mesh cells required, it was possible to model the shield in an explicit fashion. Thus, the mesh cell diagram shown in Figure 5-9 depicts the configuration 16A mockup in the as-built laminated condition.

A comparison of the results of the analysis of the homogeneous mockup of configuration 16C with those of the heterogeneous mockup of configuration 16A affords an opportunity to investigate several interesting points. One obvious point of interest is centered on a comparison of calculated results in dosimeter slots 1 through 4. In these dosimeter locations, one would expect little variation between the calculated results for the two shield configurations. Indeed, the experimental results support this thesis. Thus, it may be concluded that any significant differences between the calculated results are due to the differences in the analytical model, i.e., the homogeneous versus heterogeneous mockup. A second point of interest is the answer to the question, "What is the effect of homogenization on the magnitude and distribution of secondary photons born in the BATH modules?" Boron is present in the BATH material as a poison. By capturing a relatively large number of the available neutrons with the accompanying release of low energy (0.5 Mev) photons, the boron acts to suppress the production of high energy secondary photons, particularly in the metallic elements. When the shield module is homogenized, the boron atoms are dispersed throughout the shield. In this diluted state, the effectiveness of the boron should be increased over that observed for the heterogeneous case, wherein the boron exists in discrete plates and as such is subjected to a rather large self-shielding effect.

The axial mesh line dimensions corresponding to Figures 5-8 and 5-9 are presented in Tables 5-8 and 5-9, respectively. The radial mesh line dimensions are, of course, identical to those presented in Table 5-3 for the basic reactor geometry. Since the mesh line dimensions presented in Tables 5-8 and 5-9 reflect the inverted shield geometry, it might be helpful to relate these dimensions to the reactor coordinate system employed in Figure 5-3. Table 5-12 presents a comparison of the normal axial dimensions with the inverted

dimensions for shield configurations 16C and 16A. When viewed in conjunction with Figure 5-3, the comparison of axial dimensions given in Table 5-12 should clarify the relationship between the normal shield geometry and the inverted shield geometry.

The mesh diagram utilized in the analysis of the configuration 17 shield mockup is depicted in Figure 5-10. The axial mesh line dimensions corresponding to Figure 5-10 are presented in Table 5-10. Again, the radial mesh line dimensions have previously been presented in Table 5-3. Unlike the BATH shield mockups which, due to the high boron content and large number of distinct physical regions, taxed the capabilities of the computer system, the explicit modeling of the borated steel - liquid hydrogen mockup was accomplished with relative ease. As in the case of the BATH shield mockups, it is informative to put the inverted model of Configuration 17 into perspective with the normal shield geometry shown in Figure 5-4. Thus, in Table 5-13, a comparison of the normal and inverted coordinate system for the borated steel - liquid hydrogen shield mockup is presented.

As in the case of the basic reactor calculation, the boundary source analysis of the three shield configurations employed P_0 transport corrected 16 group neutron cross-sections and S_8 angular quadrature. One point to note in regard to the mesh diagrams presented in Figures 5-8, 5-9, and 5-10 is that all three of the problems are rather large, encompassing 5350, 5400, and 4200 mesh cells, respectively. The reason for the large physical size of the problems was the strict adherence to mesh sizing requirements established in Reference 2. In many design analysis problems, there are rather strong economic pressures which promote a desire to relax mesh requirements in order to reduce the size of a given problem. In fact, in Reference 2, a discussion pertaining to the recommended procedure to be followed in relaxing mesh sizing requirements is presented. Nevertheless, for the purposes of this analysis, wherein the stated objective is the assessment of the capabilities of the technique, the philosophy was adopted that no mesh cell relaxation in the primary dimension of interest (axial dimension) would be permitted. There was, of course, some relaxation of radial mesh sizing in the borated regions so as to keep the problem size within bounds.

Both the experimental and analytical results internal to the various shield mockups are presented in this report in terms of an "equivalent neutron flux" based on the response of the following detectors; sulfur pellets ($E \gtrsim 2.9$ Mev), cadmium covered U-238 foils ($E \gtrsim 1.5$ Mev), and bare minus cadmium covered dysprosium foils ($E \lesssim 0.4$ ev). Briefly, the experimentally determined equivalent neutron flux is obtained by dividing the measured saturated activity of the given detector by a constant, C , based on an average cross-section for the neutron interaction being considered. The analytical equivalent neutron flux is obtained by calculating the saturated activity and then dividing by the same constant, C . Thus, the comparisons presented in this report are essentially based on measured and calculated activities and, for relative comparison purposes, the effective neutron interaction cross-section used to determine the constant, C , is rather immaterial. A detailed discussion of the generation and use of activation cross-sections for analysis of this type is given in Reference 7 and will not be repeated here. The actual 16 energy group response functions used to convert calculated neutron flux distributions to "equivalent neutron flux" response of Sulfur, U-238, and bare minus cadmium covered dysprosium detectors are listed in Table 5-15.

The analysis of the photon environment internal to the various shield mockups followed a procedure which was similar and, through the source terms, intimately related to internal neutron environmental analysis. As in the case of neutrons, the overall problem was divided into two parts; a basic reactor calculation and a shield calculation. However, unlike the neutron analysis, a single boundary source calculation on the shield geometry was not sufficient. In addition to the boundary source calculation of leakage through the shield mockup, it was necessary to perform a fixed volume source calculation in order to account for secondary photons produced via neutron interactions in the shield itself.

The region dependent photon cross-sections employed in the internal analysis were generated by the APPROPOS code using microscopic cross-section data from the GAMLEG⁽⁸⁾

code as input. Atom densities employed in the cross-section generation routine were, of course, compatible with those presented in Tables 5-1 and 5-2 and the classified appendix. The cross-sections utilized in the photon analysis contained a full P_1 expansion in the scattering matrix.

As can be seen from Figure 5-5, fixed volume distributed source terms for use in the analysis of the basic reactor as well as the shield mockups were calculated via the NAGS⁽⁹⁾ data processing code. Briefly, NAGS combines neutron scalar flux distributions obtained from a prior neutron analysis of the geometry of interest with neutron interaction cross-sections from GAMBIT and a photon production library from APPROPOS to produce a spatial and energy dependent photon source throughout the problem geometry.

All of the transport calculations performed to determine the photon environment internal to the basic reactor as well as the various shield mockups employed a P_1 scattering cross-section approximation with 13 energy groups and S_8 angular quadrature. The photon energy group structure is presented in Table 5-16.

The mesh diagrams utilized in the photon analysis were identical to those used in the analysis of the internal neutron environment. The mesh diagrams used to analyze the basic reactors are given in Figure 5-6 (BATH mockups) and Figure 5-7 (BSS-LH₂ mockup). The mesh diagrams used in the shield analysis are shown in Figures 5-8, 5-9, and 5-10 for configurations 16C, 16A, and 17, respectively.

The experimentally determined photon dose rates internal to the shield mockups were reported as Mrad(C)/Hr-watt. The data were obtained by calibrating the response of a CaF₂ thermoluminescent detector (TLD) with a known source of Co-60 radiation. The analytical response was obtained by converting the calculated energy flux distributions to dose rate by means of standard energy dependent flux to dose conversion factors. The response function used in this analysis is given in Table 5-17.

As was stated previously in this Section, there are several distinct differences between the neutron and photon internal analysis which may be summarized as follows. There were, of course, basic differences in cross-sections used in the analysis. The neutron

analysis employed P_0 transport corrected cross-sections, whereas, a full P_1 cross-section set was incorporated into the photon analysis. The neutron analysis of the basic reactor was an eigenvalue problem, while the photon calculation utilized a fixed distributed source based on converged neutron fluxes from the eigenvalue case. Finally, although a single boundary source computation was sufficient to obtain neutron flux distributions throughout the shield mockups, for photons it was necessary to employ, in addition to the boundary source calculation, a fixed distributed source problem to account for secondary photons born in the shield regions. Only by summing the results of the boundary source problem and the fixed source problem was it possible to obtain the total photon radiation levels internal to the shield mockups.

5.2.1.2 Results of Analysis of Internal Neutron Environment

In addition to discussing a comparison of the results of the single and the coupled calculation, it is also of interest to examine the effects of the shield mockups on the power distribution of the reactor. Figure 5-11 presents a comparison of the measured and calculated relative axial fission distribution with the configuration 16C shield mockup in place. As can be seen from Figure 5-11, the analytical results are in good agreement with the experimental data. Similar agreement was obtained for each of the shield mockups examined.

As was discussed in Section 5.2.1.1, two of the BATH shield mockups were analyzed using the two-dimensional discrete ordinates technique. Configuration 16C was analyzed by treating the BATH modules as nine homogeneous regions, whereas, configuration 16A was analyzed by treating the individual BATH modules as explicit laminations. The internal neutron environmental comparisons for both of these BATH shield mockups are presented in Figures 5-12 through 5-39. In addition, the shielding effectiveness of a borated steel - liquid hydrogen mockup was examined both experimentally and analytically. As in the case of configuration 16A, the analytical model used to represent configuration 17 employed explicit descriptions of the borated steel, steel, and polyethylene laminates. Comparisons of calculated and measured neutron radiation levels internal to the configuration 17 mockup are presented in Figures 5-40 through 5-58.

Before discussing in detail the comparisons presented in Figures 5-12 through 5-58, it is of interest to consider some data which demonstrate that, in the area of overlap, the eigenvalue calculation on the basic reactor geometry and the boundary source calculation on the shield geometry yield equivalent results. The reader will recall that the model of the basic reactor employed in conjunction with the BATH shield mockups extended to the midplane of the second BATH module. Thus, both the reactor calculation and the shield calculation encompassed, in part, an area which included a portion of the shield support plate and a 1/8 inch boral plate as well as the aforementioned amount of the BATH mockup. It may also be recalled that dosimeter slots 1 and 2 bracket the first BATH module and, as such, are also included in the area of overlap.

A comparison of calculated results from the basic reactor calculation with those from the configuration 16C coupled calculation is presented in Table 5-18. The choice of dosimeter slot 1 as the point of comparison was based solely on convenience. The ratios presented are typical of those observed from the boundary source location up to some distance into the first BATH module. As one might expect, the presence of subsequent BATH modules in the coupled calculation is reflected in results obtained within dosimeter slot 2 and beyond. Thus, comparisons of the type presented in Table 5-18 tend to diverge in the vicinity of the second BATH module. In general, the comparisons presented in Table 5-18 are quite favorable for both neutrons and photons and, thus, lend credence to the employment of the coupled calculation as a technique for analyzing the various shield mockups. The differences observed in Table 5-18 are, at least in part, due to differences in the levels of pointwise flux convergence achieved in the two problems. It should be pointed out that, since the data presented in Table 5-18 are intended solely as a comparison of coupled calculations, the CaF_2 response does not include the effects of secondary photons born in the shield materials. Rather, only those photons born within the basic reactor geometry which leak through the boundary source plane are included.

The comparisons presented in Figures 5-12 through 5-58 are, for convenience, subdivided by shield configurations rather than by data type. Since each of the shield configurations analyzed had its own peculiarities which, in turn, had some bearing on the analytical results, it is appropriate to discuss all of the data (Sulfur, U-238, Dy) for a given shield mockup prior to considering another mockup. The various shield mockups are, of course, intercompared periodically where appropriate.

Configuration 16C (Homogeneous BATH)

Comparisons of calculated and measured response of sulfur pellets as a function of radial position at several axial locations within the configuration 16C mockup are presented in Figures 5-12 through 5-15. Configuration 16C, which had a mass thickness of 96 gm/cm^2 is an equivalent mockup of a 15 inch thick BATH shield. A comparison of calculations with measurements is shown in Figure 5-16 as a function of axial position on the shield centerline. In general, the agreement between the measured and calculated sulfur response is excellent. Significant differences occur only towards the periphery of the shield.

There are several hypotheses that possibly explain the differences observed in Figures 5-12 through 5-15. The first, and probably most obvious, explanation is that the observed differences are due to extraneous scattering interactions from a medium that was not included in the analytical model. That is, there may have been some room return background induced by scattering from test cell walls, test stand etc.; or, as stated in Section 5.2.1.1, the omission of the shield corners from the calculation may have decreased the analytical result. Both of these phenomena have some influence on the comparisons presented. However, it is not believed that these extraneous scattering effects, especially in the case of 2.9 Mev neutrons, would be of sufficient magnitude to be the sole cause of the observed differences.

In support of this contention, consider the following argument. The overall axial length of configuration 16C is only about 50 cm. If a large number of neutrons were to be scattered in random fashion around the test cell, it would be expected that, due to its relatively small size with respect to test cell dimensions, the lateral surface of the shield

mockup would not experience severe gradients of scattered flux along its axial length. In fact, it would not be unreasonable to assume that the scattered neutron flux impinging on the side of the shield mockup would be roughly constant with axial position. Now, an examination of Figure 5-16 indicates that between dosimeter slots 1 and 10 the neutrons penetrating the shield are attenuated by roughly two decades. Thus it follows that if the neutron flux at the shield periphery is made up of a "direct" component and a "scattered" component, the "scattered" portion, since it remains constant with axial position, should become a relatively more important contributor to the total flux as the shield penetration increases. One would, therefore, expect the calculations and measurements to progressively diverge along an axial traverse. This divergence is not observed. Rather, with the exception of dosimeter slot 1, the measurements remain approximately 40% greater than the analytical results over the entire axial traverse. Thus, one is led to the conclusion that the "scattered" component is a minor contributor to the total neutron flux at the shield periphery. Additional insight into the importance of room return background may be learned from Figure 5-15. Figure 5-15 depicts a radial traverse of sulfur response at the top of the configuration 16C mockup. The agreement between experimental data and analytical results is excellent. It is, therefore, clearly demonstrated that, at least on the top surface of the shield, the room return background is negligible.

In lieu of the extraneous scattering phenomenon which appears to be insufficient in magnitude to cause the observed differences in sulfur response, consider the following suggestion. The sulfur pellets employed in the experiments were placed in discrete void slots which were each slightly less than 1 cm. in axial extent. It may also be noted that the radial falloff of sulfur response varies from about a factor of 10 in dosimeter slot 1 to approximately a factor of 6.5 in dosimeter location 10. If it is assumed that lateral void streaming of neutrons occurs along the various dosimeter slots, it may then be possible to explain the differences between measured results and calculations at the shield periphery. From a comparison of the difference in calculated and measured neutron flux levels at the shield periphery with flux levels, either measured or calculated, near the shield centerline,

it is observed that the ratio of the flux difference at the periphery to the flux at the centerline is on the order of 0.05 to 0.07. Thus, it would not require an extremely large fraction of the neutrons near the axis of the shield to stream towards the periphery and create the observed differences. Furthermore, from a calculational standpoint, the large radius to thickness ratio of the slot coupled with the order of angular quadrature employed in the transport analysis makes it virtually impossible to calculate a streaming effect from the centerline to the shield periphery. Thus, it may be concluded that if a lateral void streaming situation exists, its effect would be reflected only in the measured data; and, therefore, differences between experimental and analytical results would be expected. Unfortunately, the verification of a streaming effect in the dosimeter slots is beyond the scope of this study.

Comparisons of measured and calculated response of cadmium covered U-238 foils are presented in Figures 5-17 through 5-20. Figures 5-17, 5-18, and 5-19 depict the data as a function of radial position within dosimeter slots 1, 3, and 5, respectively. An axial traverse located at a radius of 3.17 centimeters is shown in Figure 5-20. The agreement between measurements and calculations is quite good. As in the case of the sulfur pellets, there appears to be some divergence between experimental and analytical results near the shield periphery. However, due to the rather large standard deviations associated with the experimental data it becomes difficult to draw any definite conclusions in this area. Again, in the axial traverse shown in Figure 5-20, standard deviations are large and the experimental data becomes erratic for deep penetration. However, the calculated results agree quite favorably with the general trend of the experimental data.

Comparisons of measured and calculated response of bare minus cadmium covered dysprosium foils are presented in Figures 5-21 through 5-26. Figures 5-21 through 5-25 present the comparisons as a function of radius in various dosimeter slots, whereas, Figure 5-26 depicts an axial traverse at a radius of 1.57 centimeters. With few exceptions, the agreement between the experimental and analytical results appears to be rather poor. It is of interest to note that within dosimeter slot 1 the analysis underpredicts the experimental results, whereas, at other locations within the shield mockup the tendency is to overpredict.

Both of these apparent breakdowns in the calculation are explainable.

Consider first the comparison presented in Figure 5-21 wherein the analytical results appear to be too low. From the BATH shield mockup depicted in Figure 5-3 it can be observed that thermal neutrons may arrive at dosimeter slot 1 via two basic mechanisms: 1) thermalization of fast and intermediate neutrons in the BATH mockup beyond the dosimeter location, and 2) transmission of thermal neutrons from the basic reactor geometry through the boral plate to the dosimeter slot. The radial variation of thermal neutron flux produced via the thermalization process in the BATH modules should resemble the shape of the fast and intermediate flux incident on those modules. Thus, it would be expected that the greatest effect of neutron thermalization in BATH would occur near the shield centerline. The thermal flux produced in this manner would be a factor of 7 to 10 less at the shield periphery than at the centerline. It may be noted, from Figures 5-22 through 5-25, that in regions in which the thermalization process is paramount this dependence on fast flux is readily apparent. In contrast to the thermalization process, the transmission of thermal neutrons through the boral plate should be least important near the centerline and gradually increase in importance until a peak is reached near the periphery. This radial variation is, of course, due to the presence of the beryllium reflector which generates the rather high thermal flux near the periphery of the system. From both the experimental and the calculated radial variation of thermal flux within dosimeter slot 1, it is evident that the transmission process is a significant contributor to the flux in that slot. The experimental data, in fact, indicate that the thermal flux peaking caused by the reflector is still present following transmission through the boral. A distinct peak is not evident in the analytical data. It is further apparent that the discrepancies between calculations and measurements are largest at radii at which a high thermal flux was incident on the boral plate and smallest near the shield centerline where the thermalization process was perhaps the dominant mechanism. It seems feasible to conclude, therefore, that the differences noted in dosimeter slot 1 are due primarily to an inaccurate calculation of the thermal flux transmission through the boral plate.

In the analytical treatment of the transmission of neutrons through boral, the plate was considered to be a homogeneous region consisting of B_4C and aluminum. In the physical situation, the B_4C exists in discrete chunks dispersed throughout the aluminum matrix. This dispersion of discrete B_4C particles leads to a phenomenon termed "channeling"⁽⁸⁾ which tends to reduce the thermal neutron attenuation of the boral plate. Thus, the discrepancies noted in Figure 5-21 may have been due to this channeling phenomenon rather than to a breakdown in the analytical technique. It may have been possible to account for the channeling by a suitable adjustment to the boral cross-sections utilized in the transport analysis.

Although the channeling phenomenon may explain the underprediction of thermal flux in dosimeter slot 1, it does not account for the overpredictions evident throughout the remainder of the shield mockup. Once the radiation leaking from the basic reactor penetrates a short distance into the BATH mockup, incident thermal neutrons have been attenuated to the point of insignificance; and, the thermalization of fast and intermediate neutrons has become the dominant mechanism of thermal flux production. A great deal of the thermalization in the BATH mockups is due to the presence of polyethylene. Furthermore, it will be recalled that in the analysis of configuration 16C the BATH modules were treated as homogeneous regions. Thus, the thermalizing media (polyethylene) was dispersed throughout the module. It should also be noted that in the experimental configuration, the polyethylene was separated from the dosimeter slots and sandwiched between boral plates. Thus, from a calculational standpoint, the homogenization process, which not only moves thermalizing material close to the detector location but also disperses the boron poison, appears to have caused an overprediction of thermal flux at the detector locations. It will be shown in the discussion of configuration 16A, that the inclusion of the heterogeneous nature of the BATH mockup in the analytical model results in a much more accurate estimation of the thermal flux throughout the shield.

Configuration 16A (Heterogeneous BATH)

Comparisons of measured and calculated response of sulfur pellets are presented in Figures 5-27 through 5-30 as a function of radial position within several dosimeter slots of the configuration 16A mockup. Configuration 16A has a mass thickness of 60 gm/cm^2 and is a mockup of an equivalent BATH shield which is 8 inches thick. Similar data are presented as a function of axial location along the shield centerline in Figure 5-31. As in the case of configuration 16C, the agreement between experimental and analytical results is rather good. Again, discrepancies are observed in the vicinity of the shield periphery. It may be noted, from a comparison of Figures 5-27 and 5-29 with Figures 5-12 and 5-13 respectively, that the effect of homogenization of the shield modules on calculated sulfur threshold flux is slight.

Comparisons of measured and calculated response of cadmium covered U-238 response internal to the configuration 16A mockup are presented in Figures 5-32 through 5-34. Figures 5-32 and 5-33 depict the U-238 response as a function of radius in dosimeter slots 3 and 5 respectively, whereas, Figure 5-34 shows an axial traverse through the mockup at a radius of 3.17 cm. The agreement between the analytical results and the nominal measurements is encouraging. However, the magnitude of the experimental error makes it difficult to draw definite conclusions. A comparison of Figures 5-32 through 5-34 with the corresponding analytical data from the configuration 16C analysis indicates that for the U-238 threshold neutron flux the differences between the homogeneous and heterogeneous calculations are small.

In Figures 5-35 through 5-38, comparisons of measured and calculated response of bare minus cadmium covered dysprosium foils are presented as a function of radial position at various axial locations within the configuration 16A mockup. Similar data are presented in Figure 5-39 as a function of axial location at a radius of 1.57 cm. With the exception of dosimeter slot 1, the agreement between experimental and analytical results is much better than that observed in the homogeneous analysis of the configuration 16C mockup. It is quite clear that the improved agreement was due to the explicit modeling of the

configuration 16A mockup in contrast to the homogeneous calculational model used in the configuration 16C analysis.

It may be noted that in all dosimeter locations, including slot 1, the calculated results for configuration 16A are lower than those for configuration 16C. The reason for this difference in magnitude is that in the explicit mockup the principal thermalizing medium (polyethylene) has been isolated from the dosimeter location by two boral plates (see Figure 5-3). Thus, the thermal flux at dosimeter locations 2 through 5 is produced primarily from the slowing down of fast and intermediate neutrons in the titanium and aluminum plates adjacent to the dosimeter slot. Within dosimeter slot 1, the thermal flux component produced via transmission through the boral plate remains unchanged. However, a comparison of Figures 5-21 and 5-35 indicates that the thermal flux component induced via the thermalization process in the BATH modules has been decreased substantially in the heterogeneous case.

More insight into the importance of backscattering of thermal neutrons from subsequent BATH modules into a given dosimeter slot may be gained through an examination of Figure 5-39. From Figure 5-39, it is seen that the thermal flux gradient across the last BATH module is markedly steeper than those observed for any other module. If it is assumed that the observed change in slope is due solely to the lack of a BATH module above dosimeter slot 6, then the magnitude of the thermal flux produced via this backscattering phenomenon can be inferred. It can be shown that the backscattering process yields roughly 70% of the thermal flux in dosimeter slots 2 through 5. The remaining 30% is due, of course, to thermalization in polyethylene with subsequent transmission through a boral plate. The component of thermal flux which reaches the dosimeter location after passing through a boral plate is subjected to the channeling process previously described. However, since the thermal flux produced in this manner represents only 30% of the total thermal flux, the net effect of the channeling is relatively small.

Although the comparisons of measured and calculated thermal flux response are, in general, quite good for the configuration 16A mockup, some areas of rather large

disagreement still exist near the top and near the periphery of the mockup. The disagreement near the periphery is probably attributable to the return of intermediate energy neutrons from the test cell which, upon thermalizing, tend to increase the measured results. The disparity near the top of the system is most likely due partly to test cell background, but, also, to problems associated with channeling through the final boral plate.

From the comparisons just noted, it was seen that by homogenizing the individual BATH modules no significant error was introduced into the calculation of fast neutron flux internal to the assembly. However, a rather large error was observed in the calculation of thermal flux distributions throughout the mockup. The next question that comes to mind is; "What is the effect of the error in the thermal flux calculation on the analysis of the magnitude and distribution of secondary photons produced within the shield mockup?"

A comparison of the calculated secondary photon source within the first four modules of the Configuration 16A mockup with similar data from the Configuration 16C analysis is presented in Table 5-18. Table 5-18 illustrates that the total photon source produced in the heterogeneous mockup is essentially unchanged from that observed in the homogeneous case. There is, however, an obvious shift in the secondary photon spectra. By locating the boron poison in discrete plates, a rather large self-shielding effect is introduced into the heterogeneous calculation. Thus, the production of 0.5 Mev photons (group 12) via thermal neutron capture in boron is somewhat reduced. On the other hand, since the self-shielded boron cannot compete for neutrons as effectively as smeared boron, the number of captures and, hence, the secondary source in titanium, aluminum, and polyethylene is increased in the heterogeneous analysis. The amount of this increase is reflected in the larger source in groups 1 through 11 and group 13. As can be seen from Table 5-18, the amount of increase varies from 2% to 15%. It will be demonstrated, in the next section, that this small error in the secondary source in the homogeneous analysis has no significant effect on the photon radiation levels calculated internal to the shield mockup. The impact on calculated photon radiation levels external to the configuration 16C mockup is likewise negligible.

Configuration 17 (Borated Steel - Liquid Hydrogen)

Comparisons of measured and calculated response of sulfur pellets within the configuration 17 shield mockup are presented in Figures 5-40 through 5-47. Figures 5-40 through 5-46 present the comparisons as a function of radial position at several axial locations within the mockup. Figure 5-47 depicts an axial traverse through the shield taken at the centerline of the assembly. In general, the agreement between the analytical results and the measurements is good. However, for deep penetration, a striking anomaly is noted in the vicinity of the shield centerline. Figures 5-44, 5-45, and 5-46 indicate that, although the agreement between measurements and calculations is excellent for radii greater than 25 cm., a disagreement which progressively worsens with penetration, is evident near the centerline of the assembly. Again, Figure 5-47 demonstrates that beyond an axial location of about $z = 280$ cm. the slope of the analytical curve changes and the calculation tends to overpredict the experiment. Behavior of this nature is not explainable on physical grounds, but may be explained in terms of a calculational deficiency.

A close examination of the calculations performed on the configuration 17 mockup showed that this anomaly was observed only for the highest neutron energy group in the calculation. It was further observed that the radius over which the anomaly occurred was governed by only the first tier of polar angles in the S_8 quadrature. Thus the calculation near the top of the mockup and near the centerline involved an essentially uncollided transport of the forward directed flux from the boundary source plane to the top of the shield. Due to large physical size of the mockup in question, a calculation of this type represents a severe test of the analytical approach. It is quite possible that in the area of the calculational anomaly, the angular flux solutions became oscillatory in nature and the negative flux fix-up routine built into the DOT code resulted in the observed slope change and corresponding overpredictions.

Also of interest is the excellent agreement observed beyond dosimeter slot 1 and near the periphery of the mockup. It should be noted that the dosimeters used in the configuration 17 experiment were not placed in discrete void slots as was the case for the

BATH shield experiments. Thus, if the radial void streaming discussed in connection with the analysis of configuration 16C does indeed occur, it would be expected that, for configuration 17, the effect would be lessened or completely eliminated. There is, in Figures 5-41 through 5-46 evidence to support this contention.

Comparisons of calculated and measured response of cadmium covered U-238 foils are presented in Figures 5-48 and 5-49 as a function of radial position at two axial locations within the configuration 17 shield mockup. Similar data are depicted as a function of axial position within the shield in Figure 5-50. The axial traverse was taken at a radius of 3.17 cm. Although the experimental error estimates are large and the data become erratic for deep penetration, it could be concluded that the comparisons of experimental and analytical results are quite favorable. It may also be noted, from Figure 5-50, that the calculational anomaly observed in connection with the sulfur data ($E \geq 2.9$ MeV) is not evidenced in the calculations of cadmium covered U-238 response ($E \geq 1.5$ MeV).

Comparisons of measured and calculated response of bare minus cadmium covered dysprosium foils throughout the configuration 17 mockup are presented in Figures 5-51 through 5-58. Figures 5-51 through 5-57 present the comparisons as a function of radial position at several axial locations within the mockup. The data are cross-plotted in Figure 5-58. The axial traverse was taken at a radius of 3.18 cm. On the whole, the comparisons of experimental and analytical results are very good. There is a discrepancy noted within dosimeter location 1 which, as in the case of the BATH mockups, is probably due to an inaccurate calculation of the thermal neutron transmission through the boral plate adjacent to the dosimeter slot. Other significant differences are noted near the periphery of the shield as well as near the top of the mockup. These differences are most likely the result of room return background as discussed in connection with the configuration 16A analysis.

In addition to the discrepancies just cited, there appears to be a somewhat consistent difference of about 10% - 30% between calculations and measurements within dosimeter locations 6, 7, and 8. The reason for these differences in this portion of the shield is not known at this time. However, it might be suggested that the discrepancy is caused by an overestimate of the thermal neutron attenuation of steel in the calculational model. If the thermal neutron capture cross-sections of one or more constituents of steel were in error,

one would not expect the error to be manifest for the borated steel case due to the overriding effect of the extremely large capture cross-section of boron. It is, therefore, interesting to note that the differences mentioned above occur in the vicinity of unborated steel plates and not borated steel plates. The foregoing discussion is a supposition, the proof of which is beyond the scope of this work.

5.2.1.3 Results of Analysis of Internal Photon Environment

The reader will recall that in the discussion of the internal neutron environment the data were separated in terms of shield configuration and presented in the following order: Configuration 16C, Configuration 16A, and Configuration 17. That same format will be followed in this section.

Configuration 16C (Homogeneous BATH)

Comparisons of measured and calculated response of CaF_2 thermoluminescent detectors within the configuration 16C mockup are presented in Figures 5-59 through 5-63. Figures 5-59 through 5-62 depict the data as a function of radius at several axial locations within the mockup. Figure 5-63 presents an axial traverse through the shield at a radius of 3.63 cm. The agreement between the measurements and calculations is excellent at all locations internal to the shield mockup.

On Figures 5-59 through 5-63, the analytical photon data are presented as two curves; total and total minus shield secondaries. The dashed curve refers to photons which originate in the basic reactor assembly and are subsequently transmitted through the shield mockup. These data were obtained directly from the boundary source calculation on the mockup. The difference between the solid curve and the dashed curve is due to the transport of secondary photons which were born within the shield mockup itself. Information regarding this component of the internal photon radiation levels was obtained from the fixed distributed source calculation which was performed on the shield mockup. It is interesting to note, from Figure 5-63, that within dosimeter slot 1 secondary photons born in the shield comprise about 20% of the total photon dose rate. The contribution from the shield

secondaries then increases to a maximum of about 40% in dosimeter slot 5 before falling back to roughly 30% at the top of the shield.

Since the secondary photons born in the shield represent a maximum of 40% of the total dose rate within the mockup; and in view of the fact that the homogenization of the BATH modules in the configuration 16C analytical model introduced a relatively small error in the calculation of the secondary photon source, one would expect no more than a 5 - 10% error in the calculation of the overall photon radiation levels throughout the assembly. Hence, the excellent agreement between the experimental and analytical results is not surprising.

Configuration 16A (Heterogeneous BATH)

Comparisons of measured and calculated response of CaF_2 thermoluminescent detectors are presented in Figures 5-64 through 5-68 as a function of radius at several axial locations within the configuration 16A shield mockup. Similar data is presented as a function of axial location through the shield in Figure 5-68. The axial traverse was taken at a radius of 3.63 cm. As in the case of configuration 16C, the comparisons between analytical and experimental results are excellent.

It should be noted that, subsequent to the commencement of the analysis of configuration 16A, a translational error of dosimeter axial position was detected in the configuration 16A data; however, the DRM 51185, Rev. A with the corrected positions was not issued in time for the analysis. Thus, the comparisons presented in Figures 5-64 through 5-68 employ configuration 16A analytical data with configuration 16B experimental data. An examination of the construction of configurations 16A and 16B leads one to the conclusion that there is no physical reason to expect the measured results in dosimeter slots 1 through 5 to differ between the two configurations. Hence, the substitution of configuration 16B data into the analytical comparison is valid.

As seen in the case of configuration 16C, the configuration 16A analytical data are presented in terms of total dose rate and total minus shield secondaries. It can be seen, from Figure 5-68, that the secondary photons born in the shield contribute about 20% of the total dose rate in dosimeter slot 1 and roughly 40% in dosimeter slot 4. Similar

results were observed for configuration 16C. The excellent comparison between the analytical results obtained for configuration 16A and those observed for configuration 16C further support the contention that the effect of homogenizing the BATH modules in the configuration 16C analytical model did not significantly effect the determination of photon radiation levels internal to the shield mockup.

Configuration 17 (Borated Steel - Liquid Hydrogen)

Comparisons of measured and calculated response of CaF_2 thermoluminescent detectors internal to the configuration 17 shield mockup are presented in Figures 5-69 through 5-74. Figures 5-69 through 5-73 present the comparisons as a function of radial position at various axial locations within the mockup, whereas Figure 5-74 depicts an axial traverse through the shield at a radius of 0.00 cm. The agreement between analytical and experimental results is somewhat less than desirable. At present these differences are unresolved, and further examination of both the analytical approach and the experimental data is warranted.

5.2.2 One-Dimensional Discrete Ordinates Analysis

Economics and scheduler considerations often necessitate the employment of the less accurate one-dimensional approach to perform analyses. For example, in the design of a minimum weight shield system, the need for extensive parametric studies is obvious. Yet, to perform analyses of this type with the two-dimensional discrete ordinates technique is often not feasible. In addition, situations do occur in which the employment of the two-dimensional approach is not necessary and the one-dimensional technique will suffice. Thus, it is imperative, from a practical standpoint, that the shield analyst be well aware of the capability of the one-dimensional approach for analyzing complex shield configurations. Without this knowledge, the analyst can neither apply the technique to applicable situations with any degree of confidence; nor, when forced to employ the approach, realize the possible margin of error in the analytical results. Therefore, in order to gain some insight into the applicability of the one-dimensional discrete ordinates technique to NERVA shield design, extensive one-dimensional analyses of the shield configurations described in Section 5.2.1 were performed as a part of this study.

5.2.2.1 Method of Analysis

Each of the three BATH shield mockups as well as the borated steel - liquid hydrogen shield mockup was analyzed using the ANISN one-dimensional discrete ordinates transport code. Figure 5-75 depicts a schematic diagram of the mesh employed in the analysis of the three BATH shield mockups (configurations 16A, 16B, and 16C); Figure 5-76 shows similar information for the borated steel - liquid hydrogen shield mockup (configuration 17). Both Figures 5-75 and 5-76 describe each physical on-axis region of the reactor-shield geometry; and, in addition, define the boundaries of each region in terms of mesh lines and axial dimensions. A more detailed listing of the mesh line dimensions utilized in the analysis of the shield geometries depicted in Figures 5-75 and 5-76 is presented in Tables 5-19 and 5-20, respectively.

The analysis of the shield mockups depicted in Figures 5-75 and 5-76

was carried out via the simultaneous neutron-photon calculational technique incorporated into the APPROPOS - ANISN system. A schematic diagram of the overall analytical procedure is shown in Figure 5-77.

As can be seen from Figure 5-77, the basic neutron cross-sections employed in the analysis were obtained from the GAMB1T code, while basic photon cross-section data were generated by the GAMLEG code. Region atom densities employed in the cross-section generation routines were, of course, compatible with those discussed in Section 5.2.1 in connection with the two-dimensional discrete ordinates analysis of the reactor-shield systems. With the exception of the reactor core, there is a one-to-one correspondence between the axial region atom densities used in this section and those previously presented in Tables 5-1 and 5-2. Atom densities for the core region were obtained by volume averaging the atom densities in the 15 radial core zones shown in Figure 5-6.

As shown in Figure 5-77, the respective neutron and photon basic cross-sections from GAMB1T and GAMLEG were input to the APPROPOS code, wherein, they were combined with photon production data to yield a single transfer matrix for use in the simultaneous neutron-photon analysis. Briefly, this technique employed 29 energy groups (16 neutron groups and 13 photon groups) with the photon source generation process treated as a downscattering interaction from one of the neutron groups to one of the photon groups. For example, if, in the calculation, a hypothetical particle scatters from energy group 16 to energy group 24, the physical analog would be the capture of a group 16 neutron with the corresponding generation of a photon with energy characteristic of group 24. The energy group structure employed in the 29 group calculation is shown in Table 5-21.

It should be noted that the cross-sections utilized in the one-dimensional discrete ordinates analysis included a P_1 expansion in the scattering matrix. However, the values of the P_1 component existed only for groups 17 through 29. For the neutron groups (1 through 16), the P_1 component was input as zero. Thus, in effect, the neutron analysis employed P_0 transport corrected cross-sections, whereas, the photon calculations

employed P_1 cross-sections. Thus, from a cross-section standpoint, the one-dimensional analyses were entirely compatible with the two-dimensional calculations.

As in the case of the two-dimensional analysis, the ANISN calculations employed S_8 angular quadrature and solutions were based on both eigenvalue and pointwise flux convergence. Likewise, both neutron and photon response functions utilized for the experimental comparisons were identical to those presented in Tables 5-15 and 5-16, respectively. Unlike the two-dimensional calculations, however, the ANISN analyses for configurations 16A and 16B were not terminated at the physical top of the shield mockups. Rather, the problem geometry was extended a short distance above the mockup. By employing this simple expedient, it was possible to employ a single geometric model to analyze all three BATH shield mockups. Thus, the sole difference in the analytical models employed to analyze configurations 16A, 16B, and 16C was the location of the BATH modules within the basic problem geometry. The BATH modules employed throughout the one-dimensional analysis were homogeneous as in the case of the two-dimensional analysis of configuration 16C. No attempt was made to explicitly model the BATH modules in the ANISN analysis.

When using the ANISN code to perform a one-dimensional transport analysis, it is customary to include a DB^2 correction to the absorption cross-sections in order to estimate the effect of transverse leakage from the slab system on the overall attenuation properties of the medium in question. In addition to this transverse leakage correction, the WANL version of the ANISN code contains an option whereby a void streaming correction may be employed to account for transverse leakage in regions which contain no material.

In this study, both the DB^2 correction and the void streaming correction were employed in the analysis of the three BATH shield mockups. The lateral dimensions of the slabs of material which comprised the shield mockups were utilized in the generation of these transverse leakage terms. In the analysis of the borated steel-liquid hydrogen mockup, the employment of a void streaming correction was not necessary. However, the DB^2 correction was, at first, applied. Unfortunately, when the DB^2 correction was applied in the analysis of the configuration 17 shield, the analytical results were obviously in error. Thus, the calculation was repeated with the DB^2 correction removed. The results of this second analysis were more reasonable, but still not satisfactory. The

comparison of the calculations with and without the transverse leakage correction (to be presented in the next section) point out a rather serious problem area which must be taken into account when applying the one-dimensional discrete ordinates technique to the R-1 type reactor, particularly in the areas of hydrogen plena.

5.2.2.2 Results of Analysis of Internal Neutron Environment

The results of the one-dimensional transport analysis of the neutron environment internal to the configurations 16A, 16B, 16C, and 17 shield mockups are summarized in Figures 5-78 through 5-89. Figures 5-78 through 5-86 pertain to the three BATH shield mockups, whereas, Figures 5-87 through 5-89 depict the neutron radiation levels internal to the borated steel - liquid hydrogen shield mockup.

In contrast to the two-dimensional analysis, wherein configuration 16A was modeled explicitly while the configuration 16C mockup was homogenized, the one-dimensional analysis was performed using homogeneous BATH modules throughout. Thus, in the discussion of the results of the analysis, it is convenient to treat all three of the BATH mockups in a single section. On the other hand, the borated steel - liquid hydrogen mockup, due to its unique characteristics, will be discussed separately.

It should be pointed out that, although the one-dimensional calculations theoretically represent radiation levels which are averaged over the transverse dimensions of the mockup, the measurements employed in the analytical - experimental comparisons were taken at or near the centerline of the shield geometry. The reasons for this apparent anomaly are twofold. First, it is often the case in the design of a shield system that the parameter of interest is the maximum radiation level behind the shield. Thus, the test of an analytical approach may lie in its ability to predict that peak radiation level. For the particular set of experiments considered in this report, the maximum radiation levels obviously occur at the centerline of the system. Second, as will be illustrated in connection with the analysis of the borated steel - liquid hydrogen shield, the employment of the buckling correction to account for transverse leakage may, under certain circumstances, produce results which are not realistic; hence, it becomes senseless to discuss results as being

averaged over some physical plane of finite dimensions.

Configurations 16A, 16B, and 16C (Homogeneous BATH)

Comparisons of measured and calculated response of sulfur pellets are presented in Figures 5-78, 5-79, and 5-80 as a function of axial position within the configuration 16A, 16B, and 16C shield mockups, respectively. The comparisons presented in Figures 5-78 through 5-80 are rather good. Qualitatively speaking the falloff of the analytical results through the BATH mockups matches the experimental falloff quite favorably in all cases. There appears to be a factor of about 1.8 difference between the analytical and experimental results which remains fairly constant over the entire range of interest. A factor of this magnitude is in keeping with peak to average ratios observed in the two-dimensional analysis.

Comparisons of measured and calculated response of Cd covered U-238 foils are presented as a function of axial location within the configuration 16A, 16B, and 16C shield mockups in Figures 5-81, 5-82, and 5-83, respectively. Although much of the experimental data exhibit large error bars, there is a trend indicating that experimental - analytical comparisons are quite similar to those observed in the case of sulfur pellets. That is, the slope through the BATH material is predicted by ANISN with a rather good degree of accuracy, but the analytical results are low by a factor of about 1.8.

Comparisons of measured and calculated response of bare minus cadmium covered dysprosium foils are presented in Figures 5-84, 5-85, and 5-86 as a function of axial location internal to the configuration 16A, 16B, and 16C shield mockups, respectively. It would appear, from Figures 5-84 through 5-86, that the experimental - analytical thermal flux comparisons are excellent. Unfortunately, this agreement is entirely fortuitous. There were two competing processes occurring which ultimately negated one another to produce the favorable results. First, the one-dimensional approach inherently produces results which, in the slab configuration, are lower than the actual centerline value. This trait was observed not only in the fast flux comparisons just presented, but, also in the thermal flux comparisons to be discussed in connection with the borated steel - liquid hydrogen shield mockup. Thus, based on this fact alone, it would be expected that the calculated thermal

flux distributions in the BATH shield mockups would be less than the experimental distributions. The reader will recall, however, from the two-dimensional analysis of configuration 16C, that when the BATH modules were treated as homogeneous regions, an overprediction of the thermal flux was observed. The same mechanisms which resulted in the overprediction in the two-dimensional calculations were, of course, present in the one-dimensional analysis, i.e., homogeneous modules. Thus, it is evident that the overprediction induced by the homogenization process acted to cancel the inherent underprediction exhibited by the one-dimensional technique; the net result being the fortuitous agreement depicted in Figures 5-84 through 5-86.

Configuration 17 (Borated Steel - Liquid Hydrogen)

Comparisons of measured and calculated response of sulfur pellets are presented in Figure 5-87 as a function of axial position through the configuration 17 shield mockup. In Figure 5-87, the analytical results are presented for calculations both with and without the DB^2 adjustment to the absorption cross-sections. From Figure 5-87, it may be noted that when the buckling correction is employed, the slope through the shield mockup is too steep, whereas, when no buckling correction is used, the slope is too shallow. Thus, it is evident that the form of the transverse leakage correction presently incorporated into the ANISN code may, in some circumstances, lead to erroneous results. It is further apparent from Figure 5-87, however, that there should be some form of transverse leakage correction that will result in the proper slope through the mockup. The proper transverse leakage correction might be arrived at either by making an adjustment to the transverse dimension that is used to calculate the leakage terms or by employing some form of region dependent buckling correction. However, both of these improvements might prove to be impractical. In order to apply either an adjustment to the transverse dimension or a region dependent buckling, it is necessary to know a priori the leakage characteristics of the given system. Information of this type may be obtained only from a detailed two-dimensional analysis of the system in question. Thus, in lieu of an improved approach to the transverse leakage approximation, it appears that extreme caution must be exercised when results of a one-dimensional trans-

port analysis are being interpreted, particularly if the system being analyzed is much different than any configurations for which a two-dimensional analysis is available.

Comparisons of calculated and measured response of Cd covered U-238 foils are presented in Figure 5-88 as a function of axial location internal to the configuration 17 shield mockup. Again, results of analyses with and without a transverse leakage correction are presented. The comparisons are quite similar to those observed with the sulfur pellets. That is, the use of the buckling correction results in an overprediction of the attenuation characteristics of the medium, while the lack of the correction results in an underprediction.

Comparisons of measured and calculated response of bare minus cadmium covered dysprosium foils are presented as a function of axial location within the configuration 17 mockup in Figure 5-89. It is again evident that the use of the buckling correction results in an underprediction of the thermal flux gradients within the mockup. For the case in which no buckling correction was employed the comparisons of experimental and analytical results are not unreasonable.

5.2.2.3 Results of Analysis of Internal Photon Environment

As was the case in the discussion of the internal neutron environment, the analysis of the photon radiation levels internal to the three BATH shield mockups will be discussed in one section with the environment internal to the borated steel - liquid hydrogen mockup treated as a separate entity.

Configurations 16A, 16B, and 16C (Homogeneous BATH)

Comparisons of measured and calculated response of CaF_2 thermoluminescent dosimeters are presented in Figures 5-90, 5-91, and 5-92 as a function of axial position internal to the configuration 16A, 16B, and 16C BATH shield mockups, respectively. As in the case of fast neutrons, the one-dimensional analysis employing the transverse leakage correction reproduces the experimental gradient through each of the mockups with a good deal of accuracy. The analytical results are, however, consistently low by a factor of 1.3. Again, as in the case of fast neutrons, the factor of 1.3 is in keeping with the peak to

average ratios observed in the two-dimensional analysis of configurations 16A and 16C.

Configuration 17 (Borated Steel - Liquid Hydrogen)

Comparisons of measured and calculated response of CaF_2 thermoluminescent dosimeters as a function of axial location internal to the configuration 17 shield mockup are presented in Figure 5-93. As in the case of the neutron environmental analysis, results are presented in Figure 5-93 with and without the transverse leakage correction. Although the analytical results obtained with no buckling correction are reasonable, those which were obtained from the calculations wherein the buckling correction was employed are atrocious. It is interesting to note, from Figure 5-93, that the DB^2 correction in the polyethylene regions reduces the direct radiation component to such an extent that the secondary photons in the steel plates are able to manifest themselves in an increase in the photon dose rate in and around these plates. Clearly, the buckling correction used in ANISN should not be used for photon transport in low density media.

5.3 ANALYSIS OF EXTERNAL DATA

In addition to the measurement of the neutron and photon radiation levels internal to the various shield mockups, experimental data were also taken at a large number of locations external to each of the assemblies. The experimental data were presented in Section 4.0 in terms of equivalent fast neutron flux and photon dose rate as a function of polar angle on several meridian rings of varying radius. Figure 5-94 shows the relationship between the data locations and the configuration 16C mockup. The comparisons of analytical results with this set of data provide insight into the ability to calculate radiation transport around as well as through a shadow shield; and, demonstrate the capability, or lack of it, of the analytical techniques to calculate radiation levels as a function of distance away from the reactor assembly.

5.3.1 Method of Analysis

As was stated earlier in the text, two distinct methods were employed to calcu-

late the neutron and photon radiation levels external to the various shield mockups. The first technique, using the MAPIII⁽¹¹⁾ code, was based on the angular flux distributions at the surface of the reactor-shield assembly which were obtained from the two-dimensional discrete ordinates analysis of the various configurations. The second approach, employing the KAP-V⁽¹²⁾ code, was a straightforward point kernel calculation which was linked to the S_n transport analysis only through the source terms.

5.3.1.1 Discrete Ordinates Technique

The MAP code was employed to perform the analysis of the neutron and photon external environment. MAP is a code which performs a ray tracing calculation from the surface of a reactor, across a void, to a detector location. The source of the ray trace is the angular flux leakage at the surface of the reactor assembly as obtained from a preceding S_n transport calculation. Thus, the MAP calculation is dependent not only on the order of angular quadrature utilized in the S_n problem, but, also, on the number of mesh cells which define the surface of the reactor assembly.

For the purposes of this study, the MAP code was applied to the analysis of two of the BATH shield mockups (configurations 16A and 16C) as well as the borated steel - liquid hydrogen mockup (configuration 17). Due to the split nature of the transport analysis performed on these three reactor-shield assemblies, the MAP calculations are somewhat more complicated than might be suspected.

Consider, for example, the calculation of the neutron flux external to the configuration 16C mockup. From Figure 5-5, it will be recalled that two neutron DOT problems were run on that particular assembly; the basic reactor eigenvalue calculation and the boundary source problem on the shield geometry. These two calculations are labeled as sequence numbers 5 and 6 on Figure 5-5. In order to obtain a neutron flux solution at some point external to the assembly it was necessary to perform two MAP calculations and sum the results. The first MAP calculation utilized the leakage fluxes from the

basic reactor problem and computed external fluxes induced by neutrons which leaked from the lateral surface of the reactor between planes located at $Z = 0.00$ and $Z = 221.536$. The upper limit was determined by the location of the boundary source plane which coupled the basic reactor to the shield problem. The second MAP calculation utilized the total leakage from the boundary source problem to calculate fluxes at the point of interest. The sum of the results of these two MAP calculations yields the neutron flux at the point of interest that is induced by leakage from the entire reactor-shield assembly.

The procedure outlined in the preceding paragraph was followed to obtain the neutron radiation levels external to configurations 16A, 16C, and 17. In order to obtain the external photon environment, a third MAP calculation was added to the procedure. The reader will recall, from Figure 5-5, that, in addition to the basic reactor calculation and the boundary source calculation, it was necessary to perform a fixed volume distributed source calculation to account for secondary photons born in the shield regions. This final calculation was, of course, carried over into the external analysis, resulting in the additional MAP calculation.

In order to obtain a good definition of the neutron and photon environment external to the shield mockups, data points were calculated at 2.5° intervals in the polar angle range 0° to 35° for each of the meridian rings specified in Section 4.0. This particular pattern of analytical results not only permitted a direct comparison at all of the experimental data point locations, but, also, supplied sufficient data to produce complete meridian ring plots in the $0^\circ - 35^\circ$ polar angle range.

From Section 4.0, it will be noted that the neutron flux at the external locations was measured with a cadmium covered U-238 fission chamber which contained 11 PPM U-235. The low concentration of U-235 is in contrast to the 190 PPM that were present in the fission foils that were used to measure the internal neutron environment. Thus, the response function applied to the MAP results to convert to equivalent neutron flux was somewhat different than that applied to either the DOT or ANISN results internal to the shield mockups. The actual 16 energy group response functions used in conjunction with the MAP results are presented in Table 5-22.

Although the external photon dose rate measurements were obtained with carbon wall- CO_2 -gas ionization chambers as opposed to CaF_2 thermoluminescent dosimeters which were used for the internal measurements, the experimental data are reported, in Section 4.0, in terms of $\text{Rad(C)}/\text{Hr-Watt}$. Thus, the photon response function which was applied to the MAP results was identical to that used during the one and two-dimensional internal analysis. The photon response function was previously presented in Table 5-16.

5.3.1.2 Point Kernel Technique

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 configuration 16A, 16C, and 17 shield mockups. The geometric model utilized in the point kernel analysis is shown in Figure 5-95. The material compositions corresponding to the regions shown in Figure 5-95 were, of course, compatible with the materials described in Tables 5-1 and 5-2 and in the classified appendix to this report.

There are a few salient points regarding the point kernel model shown in Figure 5-95 that should be discussed. First, in keeping with the format that is normally followed in point kernel design studies, the reactor core was treated as a single homogeneous region. As in the case of the one-dimensional transport analysis, material densities for the homogeneous core were obtained by volume averaging the densities of the 15 radial core zones described in the classified appendix. Second, preliminary studies indicated that the shield mockups could be treated as homogeneous regions with no significant loss in accuracy. This property of the various mockups held true not only for radiation transport through the shield, but, also, for the transport of secondary photons which were born in the shield itself. Thus, in order to reduce the number of geometric regions and, in turn, the computation time of the problem, all of the shield mockups analyzed were treated as single homogeneous regions.

Due to the generalized geometry routine incorporated into the KAP-V code, it was possible to model the reactor-shield assemblies in the as-built configuration, that is,

the square shield modules were placed on the cylindrical reactor. This configuration is reflected in Figure 5-95. Preliminary studies were carried out to ascertain the effect of removing the corners of the shield mockup on the external environment. When the reactor shield assembly was mocked up as a continuous cylinder identical to that employed in the discrete ordinates analysis, the KAP-V results were essentially identical to the results obtained with the as-built configuration.

Neutron and photon source terms for use in the point kernel analysis 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 7 and 8 on Figure 5-5. 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 reactor shield assembly. It should be noted that, since the source terms generated by NAGS were employed in both the discrete ordinates analysis and the point kernel calculations, the two methods had a consistent starting point; thus, differences in analytical results may be attributed to technique rather than to basic data.

The point kernel neutron analysis was carried out using the moments method data with carbon as the reference material. A bivariate polynomial expression was employed to calculate the differential neutron spectra based on infinite medium carbon data. The bivariate expression was solved in terms of initial neutron energy and of material penetration. Coefficients for use in the polynomial expression have previously been listed in Reference 13, whereas neutron removal cross-sections for use in the material attenuation functions were obtained from Reference 14. The 27 energy group response function employed to convert KAP-V results to equivalent fast neutron flux is presented in Table 5-23.

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 medium water dose buildup factors were applied to all of the photon calcu-

lations. As in the case of the attenuation coefficients, the infinite medium buildup factors were generated internal to the KAP-V code. The response function to be applied to the calculational results was presented in Table 5-16.

As with the discrete ordinates approach, calculations using the point kernel technique were carried out at 2.5° intervals within the 0° to 35° polar angle range upon each of eleven rings which were used for experimental traverses.

5.3.2 Results of Analysis of External Neutron Environment

Comparisons of measured and calculated response of a cadmium covered U-238 fission chamber at various locations external to the BATH shield mockups are presented in Figures 5-96 through 5-125. Figures 5-126 through 5-134 depict the analytical results external to the configuration 17 mockup. Due to the considerable geometrical differences between the "background" shield configuration and the borated steel-liquid hydrogen configuration, meaningful direct (with test cell background removed) data could not be obtained, thus preventing a comparison of experimental and analytical data for the borated steel-liquid hydrogen mockup. Comparisons on a purely analytical basis of radiation levels external to the configuration 17 mockup with those forward of the two BATH shield mockups are useful. Thus, the configuration 17 analytical data will be discussed. As was the case in the discussion of the two-dimensional analysis of the internal radiation environment, each of the three shield mockups (configurations 16C, 16A, and 17) will be treated separately.

Configuration 16C

Comparisons of measured and calculated response of Cd covered U-238 fission chambers external to the configuration 16C mockup are presented in Figures 5-96 through 5-110. Figures 5-96 through 5-106 depict the data as a function of polar angle on meridian rings of varying distance from the reactor core. Figures 5-107 through 5-110 depict the data as a function of distance from core center along radials located at polar angles of 0° , 10° , 20° , and 30° , respectively.

In general, both the point kernel and the discrete ordinates results agree fairly well with the experimental data on the meridian rings which are in close proximity to the

shield. However, the agreement progressively worsens with increasing distance from the reactor. Clearly, using the point kernel technique, the agreement between the experimental and analytical results is most favorable at a polar angle of 0° , becomes rather poor in the polar angle range of 10° to 20° , and then improves somewhat at the 30° data location. The results of the discrete ordinates analysis exhibit much the same behavior close to the reactor assembly. However, for meridian ring radii of about 275 cm and greater, the 0° polar angle location becomes the point of poorest agreement.

Consider, for a moment, Figures 5-107 through 5-110. At polar angles of 0° and 10° , the agreement between calculations and measurements is quite good up to about 275 cm, but gradually worsens over the remainder of the traverse. It is quite probable that the disagreement between the point kernel and experimental data observed at large distances from the reactor is due primarily to an anomaly in determining the experimental background. This background flux when subtracted from the absolute measurements presented in Section 4.0 yields the experimental data points shown in Figures 5-107 through 5-110. As can be seen from Figure 5-107, the experimental data do become somewhat erratic at radii greater than 275 cm. The rather large disagreement between the MAP results and the measurements at large distances from the reactor is due, in part, to the aforementioned problem with the background data; but also, to a large degree, is enhanced by an insufficient order of angular quadrature employed in the transport analysis.

From Figures 5-109 and 5-110, it can be seen that, at polar angles of 20° and 30° , the agreement between experimental and analytical results is rather poor. The suppression of the point kernel results are quite probably due to the classic inability of the technique to account for "short circuit" paths through and around the shield. On the 20° traverse, the discrete ordinates analysis markedly changes slope at about 250 cm. This change in slope is due to a significant contribution to the flux at the detector location of particles which leak through the side of the reactor and, thus, circumvent the shield. Clearly, this slope change is not evident in the point kernel results. Again, background problems may be a significant contributor to the observed differences.

Configuration 16A

Comparisons of measured and calculated response of cadmium covered U-238 fission chambers are presented in Figures 5-111 through 5-121 as a function of polar angle on several meridian rings external to the configuration 16A shield mockup. Similar data are presented in Figures 5-122 through 5-125 as a function of distance from core center along radial traverses taken at 0°, 10°, 20°, and 30°, respectively.

As might be suspected, both the point kernel and discrete ordinates results are improved over those observed for configuration 16C. It appears, from Figure 5-122, that the effect of employing S_8 angular quadrature in the discrete ordinates analysis is much less than was the case with configuration 16C. Results of this type should be expected since neutron leakage from the lighter weight shield is less forward peaked, thus negating the necessity of employing higher order quadratures. The point kernel results in the polar angle range of 10° to 30° are likewise improved due to the reduced mass of the configuration 16A mockup.

Configuration 17

Comparisons of discrete ordinates and point kernel calculated response of Cd covered U-238 fission chambers as a function of polar angle on various meridian rings external to the configuration 17 shield mockup are presented in Figures 5-126 through 5-134. As stated earlier, background radiation in the experimental data prevented a comparison of measurements with calculations. Comparisons on a purely analytical basis, are of interest. These data will be employed in Section 6 to effect a comparison of the BATH and BSS-liquid hydrogen shields.

5.3.3 Results of Analysis of External Photon Environment

Configuration 16C

Comparisons of measured and calculated response of Co_2 - ionization chambers are presented in Figures 5-135 through 5-145 as a function of polar angle on several meridian rings external to the configuration 16C shield mockup. Similar data are

presented as a function of distance from core center along traverses taken at polar angles of 0° , 10° , 20° , and 30° in Figures 5-146 through 5-149, respectively.

In general, the comparisons presented in Figures 5-135 through 5-149 are similar to those observed for fast neutrons. However, the disagreement between calculations and measurements at the 10° data location has been mollified somewhat and at the 20° location has essentially been eliminated. In addition, both the point kernel and discrete ordinates techniques seem to overpredict the photon dose rate along the 30° traverse (see Figure 5-149). However, it is believed that the measurements are in actuality an underestimate of the true photon dose rate along the traverse. Recall, from Section 3.0, that the test cell background was determined by placing a "black" shield on the assembly and measuring the neutron flux and photon dose rate at the points of interest; the "black" shield configuration included slabs of borated polyethylene at the side of the reactor. While being a very effective shield against neutron leakage from the lateral surface of the reactor, this particular configuration was not opaque to photons. Thus, the background measurements taken along the 30° traverse contain a response due to direct photon leakage from the lateral surface of the reactor; and is, therefore, overestimated. This overestimate of the photon background measurement leads to a corresponding underestimate of the true photon dose rate along the 30° traverse.

Configuration 16A

Comparisons of measured and calculated response of Co_2 - ionization chambers at various locations external to the configuration 16A shield mockup are presented in Figures 5-150 through 5-164. Figures 5-150 through 5-160 depict the comparisons as a function of polar angle on several meridian rings external to the mockup, whereas, Figures 5-161 through 5-164 show traverses from the core center along radials located at 0° , 10° , 20° , and 30° . The comparisons between the discrete ordinates results and the experimental data is excellent. The point kernel results, in general, tend to overpredict the measurement. Again, it appears reasonable to assume that the experimental data along the 30° traverse have underestimated the true photon dose rate at those locations.

Configuration 17

The analytically determined photon dose rate external to the borated steel-liquid hydrogen shield mockup is presented in Figures 5-165 through 5-173 as a function of polar angle on several meridian rings of increasing distance from core center. Again, there are no direct measurements with which to validly compare analytical data. The calculated results are presented to effect an analytical comparison with the results of the analysis of the two BATH shield mockups.

5.4 STATISTICAL ANALYSIS OF THE DATA

This section provides the analysis of the difference between calculated and experimental data leading to the development of uncertainty factors. These uncertainty factors can then be applied to calculated radiation environment to estimate possible variations in calculated results. Section 5.5 of WANL-TME-1914 "Radiation Environment in Various NERVA Components" derives the need for using uncertainty factors to discuss possible variations in the calculated data and contains a detailed discussion of the manner in which the statistical analysis was performed to obtain the uncertainty factors to be applied to the calculated environment data internal to the shield mockups. The manner employed to derive the uncertainty factors applicable to the external radiation environment is discussed in this section of this report.

The development of the uncertainty factors is provided in two subsections. The first subsection deals with the internal environment while the second section deals with the external environment. These sections report the nominal uncertainty factor between the calculated and experimental data and the one standard deviation limits on this nominal uncertainty factor.

5.4.1 Internal Radiation Environment

Table 5-24 provides the nominal uncertainty factor between the calculated and experimental data internal to the shield mockups. Also provided are the one sigma limits on this nominal uncertainty factor. The uncertainty factors are provided for both the neutron and photon environment determined internal to shield configuration 16C, the ~ 15 in.

BATH shield mockup, and shield configuration 17, the borated steel - LH₂ shield mockup. As an example of the use of these data, Table 5-24 indicates that the nominal uncertainty between the calculated and experimental sulfur data in shield configuration 16C is 1.08. Further, the experimental data lay between the product of 1.16 and 1.01 times the calculated data for 68% of the observations.

An examination of the nominal uncertainty factor between calculated and experimental data indicates that the difference in the nominal uncertainty factor for the neutron data between the two shield configurations is within two standard deviations of one another. The difference in the nominal uncertainty for the photon data between the two shield configurations is not within three standard deviations of one another. This implies that the ability to calculate the neutron environment in both shield configurations is similar; however, the ability to calculate the photon environment in the borated steel - LH₂ shield mockup appears to be significantly different.

It should be noted that an average ratio of calculated to experimental results not equal to unity may be due to causes other than a discrepancy between the calculated and actual fluxes. Therefore, the uncertainty in the calculated environment may be more accurately represented by just the one standard deviation in the ratio of calculated to experimental data. The one standard deviation in the data with nominal differences between calculated and experimental information removed is obtained from the results in Table 5-24 by dividing the one standard deviation upper limit in the ratio of experimental to calculated data by the average ratio of experimental to calculated data. The following equation provides the formulation for determining other than one standard deviation uncertainty factors, F, from this information:

$$F(m) = \sigma^m \quad \text{where}$$

F = the uncertainty factor

m = the desired level of uncertainty (i.e., 1.0 sigma, 2.3 sigma, 3.0 sigma, etc.)

σ = the one standard deviation uncertainty factor
derived above

$$\sigma = \frac{\text{One Standard Deviation Upper limit of Ratio}}{\text{Average Ratio of Experimental to Calculated Data}}$$

5.4.2 External Radiation Environment

One of the major purposes of the study presented in this report 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 subsystem.

The current R-1 reactor design* includes approximately 180 gm/cm^2 of material forward of the reactor core. Of this mass thickness, about 115 gm/cm^2 is the BATH 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. Recall that the largest BATH shield examined during this study 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 180 gm/cm^2 . Extrapolations of this sort are presented in Figures 5-174 through 5-177. Figures 5-174 and 5-175 depict the fast neutron flux as a function of shield mass thickness on a plane located roughly 66 inches forward of the reactor core center. Figure 5-174 shows centerline data, whereas, Figure 5-175 presents data taken near the periphery of the system. Similar data describing the variation of photon dose rate with shield mass thickness are presented in Figures 5-176 and 5-177. It should be pointed out that Figures 5-174 through 5-177 present extrapolations not only of the experimental data,

* R-1 Reference Design per WANL drawing 939J685E.

but also of the discrete ordinates calculated as well as the point kernel calculated data. The data locations chosen are typical of those positioned forward of the various experimental shield mockups.

Based on a comparison of the extrapolations of experimental and analytical data presented in Figures 5-174 through 5-177, it is possible to estimate uncertainty factors to be applied to the analysis of a 180 gm/cm^2 shield. The uncertainty factor is defined as:

$$F = \frac{\{\phi_{\text{exp}}\}_{180}}{\{\phi_{\text{calc}}\}_{180}} \quad (\text{Eq. 1})$$

where:

$\{\phi_{\text{exp}}\}_{180}$ = Estimated experimental fast neutron flux (photon dose rate) forward of a 180 gm/cm^2 shield

$\{\phi_{\text{calc}}\}_{180}$ = Estimated analytical fast neutron flux (photon dose rate) forward of a 180 gm/cm^2 shield

The above ratio is, of course, applicable to the results of both discrete ordinates and point kernel analyses.

Uncertainty factors determined via the method outlined above are summarized in Table 5-25. From Table 5-25, it is seen that on or close to the PVARA tangent plane, uncertainty factors applicable to fast neutron calculations vary 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.

It might be argued that the extrapolation described above is inadequate in view of the relatively large variation in uncertainty factors along the plane. Nevertheless, in lieu of experimental data directly applicable to a 180 gm/cm^2 configuration, these extrapolations are the only recourse. It is recommended that, until measurements are available for an assembly which more closely approximates the R-1 axial configuration, all design studies be based on the more conservative uncertainty factors; i.e., those obtained near the periphery of the shield assembly, given in Table 5-25.

5.5 References

1. WANL-TME-2689, "The Evaluation of the WANL Integrated Nuclear, Radiation and Shielding Standard Design Method (WISDM), " to be published.
2. WANL-TME-1982, "User's Manual for the DOT-IIW Discrete Ordinates Transport Computer Code, " R. G. Soltesz and R. K. Disney, December 1969.
3. WANL-TME-1969, "Second Version of the GAMB1T Code," G. Collier, G. Gibson, L. L. Moran, R. K. Disney and R. S. Kaiser, November 1969.
4. WANL-TMI-1967, "Revised WANL ANISN Program User's Manual," R. G. Soltesz, April 1969.
5. ANISIG User's Manual - to be published.
6. APPROPOS User's Manual - to be published
7. American Nuclear Society Paper Presented at the 1969 Winter Meeting, "Neutron Flux Spectra Measurements with Energy Dependent Activation Foils and Comparison with Multigroup Reaction Rate Calculations," S. Salah, W. D. Rankin, and V. S. Oblock.
8. GAMLEG-W User's Manual - to be published.
9. WANL-PR(LL)-010, "Synthesis of Computational Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems," Volume 7, R. K. Disney, R. G. Soltesz, and S. L. Zeigler, June 1967.
10. ORNL-2528, "Radiation Transmission through Boral and Similar Heterogeneous Materials Consisting of Randomly Distributed Absorbing Chunks," W. R. Burrus, January 1960.
11. WANL-TME-2706, "User's Manual for the MAP Radiation Transport Code," R. K. Disney, R. G. Soltesz, J. Jedruch, and S. L. Zeigler, - to be published.
12. WANL-PR(LL)-010, "Synthesis of Computational Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems, Volume 4, R. K. Disney and M. A. Capo, June 1967.
13. Unpublished data.
14. E. P. Blizard and L. S. Abbott Eds., Reactor Handbook - Volume III, Part B, Shielding, Interscience Publishers, 1962.

TABLE 5-1

ATOM DENSITIES OF REACTOR REGIONS USED IN PAX-GOC EXPERIMENTS

$$\left\{ \frac{\text{NUCLEI}}{\text{CM}^3 \times 10^{-24}} \right\}$$

REGION NAME	Be	B	C	Fe	Ni	Cr	Mn	Nb	Ti	Al	Cu	Gd
CLUSTER PLATE HARDWARE				.006846	.0008773	.001980	.0001975			.001945		
CLUSTER PLATE				.03714	.004757	.01074	.001071			.001945		.0001541
SUPPORT PLATE				.002894	.0003708	.0008371	.00008347			.02998	.0008559	
ALUMINUM SHIELD										.05507	.001491	
DRUM DRIVE				.0007906	.0002320	.0002319	.00002300		.001767	.03911	.001059	
DRIVE SHAFTS				.0008637	.0001786	.0002533	.00002538		.004143	.0004493		
PRESSURE VESSEL										.05998		
FILLER STRIP			.09300									
LATERAL SUPPORT			.06442									
REFLECTOR	.09603								.0006325	.001856		
SUPPORT BLOCKS			.05722	.001093	.0001400	.0003161	.00003152					
AFT LATERAL SUPPORT			.09609	.0004394	.00005494	.0001249				.0009255		
REFLECTOR SUPPORT				.001584	.0001285	.0004782	.00004595	.00005719	.008862	.02242		
TIE ROD ENDS				.001016	.0001302	.0002940	.00002932					
BORAL		.02196	.005490							.04223		

TABLE 5-2
 ATOM DENSITIES OF SHIELD REGIONS USED IN PAX-G0C EXPERIMENTS
 (NUCLEI/CM³ X 10⁻²⁴)

REGION NAME	H	B	C	Al	Ti	Fe	Cr	Ni	Mn
SHIELD SUPPORT PLATE				.05715					
BORAL		.02196	.005490	.04233					
BATH MOCKUP									
TITANIUM					.05683				
ALUMINUM				.06094					
POLYETHYLENE	.07996		.03998						
1/8 IN. BORAL		.02196	.005490	.04233					
1/4 IN. BORAL		.02314	.005784	.04463					
"SMEARED" BATH MODULE	.01332	.005683	.008080	.03638	.009467				
BSS-LH ₂ MOCKUP									
BORATED STEEL		.005811				.06132	.01262	.005518	.001684
STEEL						.06249	.01541	.005622	.001716
POLYETHYLENE	.01168		.005839						

TABLE 5-3

RADIAL MESH LINE DIMENSIONS FOR USE IN ALL DOT CALCULATIONS

Line No.	R (cm)	Line No.	R(cm)	Line No.	R(cm)	Line No.	R(cm)
1	0.0	16	27.519	31	46.210	46	64.351
2	4.0165	17	28.475	32	47.388	47	65.195
3	8.033	18	29.431	33	49.122	48	66.040
4	10.041	19	30.900	34	51.118	49	66.145
5	12.049	20	32.368	35	53.107	50	66.250
6	14.058	21	34.384	36	55.096	51	66.358
7	15.954	22	36.400	37	57.085		
8	17.850	23	38.152	38	59.0738		
9	18.967	24	39.903	39	59.322		
10	20.084	25	40.611	40	60.935		
11	21.359	26	41.318	41	62.548		
12	22.633	27	42.038	42	63.185		
13	23.989	28	42.757	43	64.033		
14	25.344	29	43.895	44	64.138		
15	26.432	30	45.032	45	64.240		

TABLE 5-4

AXIAL MESH LINE DIMENSIONS FOR BASIC REACTOR GEOMETRY
WITH BATH SHIELD MOCKUPS

Line No.	Z(cm)	Line No.	Z(cm)	Line No.	Z(cm)	Line No.	Z(cm)
1	0.00	19	73.446	37	187.847	55	225.026
2	0.848	20	73.921	38	194.566	56	225.420
3	5.239	21	74.500	39	202.285	57	225.814
4	11.172	22	75.500	40	206.001	58	226.208
5	17.105	23	77.500	41	207.906	59	226.602
6	23.038	24	82.500	42	209.811	60	226.997
7	28.971	25	90.219	43	211.716	61	227.391
8	34.904	26	97.938	44	212.850	62	227.785
9	40.835	27	105.657	45	213.984	63	228.179
10	55.349	28	113.376	46	219.661	64	228.573
11	59.159	29	121.095	47	220.911	65	228.967
12	62.969	30	128.814	48	222.161	66	229.920
13	66.779	31	136.533	49	223.739	67	230.314
14	70.589	32	144.252	50	223.806	68	230.708
15	71.859	33	151.971	51	223.873	69	231.102
16	73.129	34	159.690	52	223.939	70	231.496
17	73.229	35	168.409	53	224.006	71	231.891
18	73.339	36	178.128	54	224.073	72	232.285

TABLE 5-5

AXIAL MESH LINE DIMENSIONS FOR BASIC REACTOR GEOMETRY
WITH BSS-LH₂ SHIELD MOCKUP

Line No.	Z(cm)	Line No.	Z(cm)	Line No.	Z(cm)	Line No.	Z(cm)
1	0.00	19	73.446	37	187.847	55	225.026
2	0.848	20	73.921	38	194.566	56	225.280
3	5.239	21	74.500	39	202.285	57	225.534
4	11.172	22	75.500	40	206.001	58	225.788
5	17.105	23	77.500	41	207.906	59	226.042
6	23.038	24	82.500	42	209.811	60	226.296
7	28.971	25	90.219	43	211.716	61	228.519
8	34.904	26	97.938	44	212.850	62	230.741
9	40.835	27	105.657	45	213.984	63	232.964
10	55.349	28	113.376	46	219.661	64	235.186
11	59.159	29	121.095	47	220.911	65	235.440
12	62.969	30	128.814	48	222.161	66	235.694
13	66.779	31	136.533	49	223.739	67	235.948
14	70.589	32	144.252	50	223.806	68	236.202
15	71.859	33	151.971	51	223.873	69	236.456
16	73.129	34	159.690	52	223.939	70	238.678
17	73.229	35	168.409	53	224.006	71	240.901
18	73.339	36	178.128	54	224.073	72	243.123
						73	245.346
						74	245.599
						75	245.853
						76	246.107
						77	246.361
						78	246.616

TABLE 5-6
MATERIAL DESCRIPTION FOR BASIC
REACTOR DOT CALCULATIONS

ZONE NO.	DESCRIPTION	
	BATH MOCKUPS	BSS-LH ₂ MOCKUP
1	Core Region 1	Core Region 1
2	Core Region 2	Core Region 2
3	Core Region 3	Core Region 3
4	Core Region 4	Core Region 4
5	Core Region 5	Core Region 5
6	Core Region 6	Core Region 6
7	Core Region 7	Core Region 7
8	Core Region 8	Core Region 8
9	Core Region 9	Core Region 9
10	Core Region 10	Core Region 10
11	Core Region 11	Core Region 11
12	Core Region 12	Core Region 12
13	Core Region 13	Core Region 13
14	Core Region 14	Core Region 14
15	Core Region 15	Core Region 15
16	Filler Strips	Filler Strips
17	Lateral Support	Lateral Support
18	Reflector	Reflector
19	Control Vane	Control Vane
20	Void	Void
21	Pressure Vessel	Pressure Vessel
22	Boral (.125 in.)	Boral (.125 in.)
23	Aluminum Shield	Aluminum Shield
24	Support Plate	Support Plate
25	C. P. Hardware	C. P. Hardware
26	Cluster Plate	Cluster Plate
27	Drum Drive Near Dome	Drum Drive Near Dome
28	Drum Drive	Drum Drive
29	Support Blocks	Support Blocks
30	Tie Rod Ends	Tie Rod Ends
31	Aft Lateral Support	Aft Lateral Support
32	Aft Reflector Support	Aft Reflector Support
33	Shield Support Plate	Shield Support Plate
34	Boral (.125 in.)	Boral (.125 in.)
35	BATH Module	BATH Module
36		Borated Steel
37		Polyethylene

TABLE 5-7

NEUTRON ENERGY GROUP STRUCTURE

GROUP NO.	ENERGY LIMITS (MEV)
1	2.87 (0) - 10.0 (0)
2	1.35 (0) - 2.87 (0)
3	0.821 (0) - 1.35 (0)
4	0.388 (0) - 0.821 (0)
5	0.111 (0) - 0.388 (0)
6	0.015 (0) - 0.111 (0)
7	5.53 (-3) - 0.015 (0)
8	5.83 (-4) - 5.53 (-3)
9	7.89 (-5) - 5.83 (-4)
10	1.068 (-5) - 7.89 (-5)
11	1.86 (-6) - 1.068 (-5)
12	3.00 (-7) - 1.86 (-6)
13	1.20 (-7) - 3.00 (-7)
14	6.00 (-8) - 1.20 (-7)
15	2.00 (-8) - 6.00 (-8)
16	0.00 - 2.00 (-8)

NOTE: NUMBERS IN PARENTHESES REFER TO POWERS OF TEN.

TABLE 5-8

AXIAL MESH LINE DIMENSIONS FOR CONFIGURATION 16C SHIELD GEOMETRY

Line No.	Z(cm)	Line No.	Z(cm)	Line No.	Z(cm)	Line No.	Z(cm)
1	0.00	28	12.317	55	24.076	82	36.393
2	0.953	29	12.711	56	24.470	83	36.787
3	1.348	30	13.106	57	25.423	84	37.181
4	1.742	31	13.500	58	25.817	85	37.576
5	2.136	32	13.894	59	26.211	86	37.970
6	2.530	33	14.288	60	26.605	87	38.364
7	2.924	34	14.682	61	26.999	88	38.758
8	3.299	35	15.635	62	27.393	89	39.152
9	3.713	36	16.029	63	27.788	90	40.105
10	4.107	37	16.423	64	28.182	91	40.499
11	4.501	38	16.817	65	28.576	92	40.893
12	4.895	39	17.211	66	28.970	93	41.287
13	5.847	40	17.605	67	29.364	94	41.681
14	6.241	41	18.000	68	30.317	95	42.075
15	6.635	42	18.394	69	30.711	96	42.470
16	7.029	43	18.788	70	31.105	97	42.864
17	7.423	44	19.182	71	31.499	98	43.258
18	7.817	45	19.576	72	31.893	99	43.652
19	8.212	46	20.529	73	32.287	100	44.046
20	8.606	47	20.923	74	32.682	101	44.999
21	9.000	48	21.317	75	33.076	102	45.006
22	9.394	49	21.711	76	33.470	103	45.133
23	9.788	50	22.105	77	33.864	104	45.199
24	10.741	51	22.499	78	34.258	105	45.266
25	11.135	52	22.894	79	35.211	106	45.333
26	11.529	53	23.288	80	35.605	107	46.911
27	11.923	54	23.682	81	35.999	108	47.536

TABLE 5-9

AXIAL MESH LINE DIMENSIONS
FOR CONFIGURATION 16A GEOMETRY

Line No.	Z(cm)	Line No.	Z(cm)	Line No.	Z(cm)	Line No.	Z(cm)
1	0.000	29	6.341	57	13.157	85	20.741
2	0.953	30	6.419	58	13.220	86	20.811
3	1.024	31	6.482	59	13.284	87	20.882
4	1.095	32	6.960	60	13.919	88	20.953
5	1.165	33	7.595	61	14.554	89	21.023
6	1.236	34	8.072	62	15.635	90	21.094
7	1.306	35	8.136	63	15.706	91	21.164
8	1.377	36	8.199	64	15.776	92	21.642
9	1.448	37	8.263	65	15.847	93	22.277
10	1.518	38	8.326	66	15.917	94	22.754
11	1.589	39	8.390	67	15.988	95	22.818
12	2.066	40	9.025	68	16.059	96	22.881
13	2.701	41	9.660	69	16.129	97	22.945
14	3.179	42	10.741	70	16.200	98	23.008
15	3.243	43	10.812	71	16.270	99	23.072
16	3.306	44	10.882	72	16.748	100	23.707
17	3.370	45	10.953	73	17.383	101	24.342
18	3.433	46	11.023	74	17.860	102	25.423
19	3.497	47	11.094	75	17.924	103	25.490
20	4.132	48	11.165	76	17.987	104	25.557
21	4.767	49	11.235	77	18.051	105	25.624
22	5.847	50	11.306	78	18.114	106	25.691
23	5.918	51	11.376	79	18.178	107	25.758
24	5.988	52	11.854	80	18.813	108	27.336
25	6.059	53	12.489	81	19.448	109	27.961
26	6.129	54	12.966	82	20.529		
27	6.200	55	13.030	83	20.600		
28	6.271	56	13.093	84	20.670		

TABLE 5-10

AXIAL MESH LINE DIMENSIONS FOR CONFIGURATION 17 SHIELD GEOMETRY

Line No.	Z(cm)	Line No.	Z(cm)	Line No.	Z(cm)	Line No.	Z(cm)
1	0.0	21	20.828	41	46.355	61	67.945
2	0.254	22	21.082	42	46.990	62	70.168
3	0.508	23	21.336	43	47.625	63	72.390
4	0.762	24	21.590	44	49.847	64	72.644
5	1.016	25	23.812	45	52.070	65	72.898
6	1.270	26	26.035	46	52.317	66	73.152
7	3.492	27	28.257	47	52.580	67	73.406
8	5.715	28	30.480	48	52.834	68	73.660
9	7.939	29	30.734	49	53.088	69	75.882
10	10.160	30	30.988	50	53.340	70	78.105
11	10.414	31	31.242	51	55.564	71	80.327
12	10.668	32	31.496	52	57.786	72	82.550
13	10.922	33	31.750	53	60.008	73	82.804
14	11.176	34	33.972	54	62.230	74	83.058
15	11.430	35	36.195	55	62.485	75	83.312
16	13.652	36	38.417	56	62.738	76	83.566
17	15.875	37	40.640	57	62.993	77	83.820
18	18.097	38	41.275	58	63.247	78	84.773
19	20.320	39	41.910	59	63.500	79	84.840
20	20.574	40	44.132	60	65.723	80	84.907
						81	84.973
						82	85.040
						83	85.107
						84	86.685
						85	87.310

TABLE 5-11

MATERIAL DESCRIPTION FOR
SHIELD MOCKUP DOT CALCULATIONS

ZONE NO.	DESCRIPTION		
	CONFIG. 16C	CONFIG. 16A	CONFIG. 17
1	Void	Void	Void
2	BATH Module	Boral (.250 in.)	Steel
3	Boral (.125 in.)	Aluminum	Borated Steel
4	Shield Support Plate	Polyethylene	Polyethylene
5		Boral (.125 in.)	Boral (.125 in.)
6		Titanium	Boral (.125 in.)
7		Shield Support Plate	Shield Support Plate

TABLE 5-12
COMPARISON OF NORMAL AND INVERTED
COORDINATE SYSTEMS FOR BATH SHIELD MOCKUPS

AXIAL DIMENSIONS (CM)		
NORMAL GEOMETRY	CONFIG. 16C INVERTED GEOMETRY	CONFIG. 16A INVERTED GEOMETRY
221.536	47.536	27.961
223.739	45.333	25.758
224.073	44.999	25.423
225.026	44.046	24.342
228.967	40.105	20.529
229.920	39.152	19.448
233.861	35.211	15.635
234.814	34.258	14.554
238.755	30.317	10.741
239.708	29.364	9.660
243.649	25.423	5.847
244.602	24.470	4.767
248.543	20.529	0.953
249.496	19.576	0.000
253.437	15.635	
254.390	14.682	
258.331	10.741	
259.284	9.788	
263.225	5.847	
264.177	4.895	
268.119	0.953	
269.072	0.000	

NOTE: NORMAL GEOMETRY DIMENSIONS ARE COMPATIBLE
WITH FIGURE 5-3.

TABLE 5-13

COMPARISON OF NORMAL AND INVERTED COORDINATE SYSTEM
FOR BSS-LH₂ SHIELD MOCKUP

AXIAL DIMENSIONS (CM)	
NORMAL GEOMETRY	CONFIG. 17 INVERTED GEOMETRY
221.536	87.310
223.739	85.107
224.073	84.773
225.026	83.820
226.296	82.550
235.186	73.660
236.456	72.390
245.346	63.500
246.616	62.230
255.506	53.340
256.776	52.070
261.221	47.625
262.491	46.355
266.936	41.910
268.206	40.640
277.096	31.750
278.366	30.480
287.256	21.590
288.526	20.320
297.416	11.430
298.686	10.160
307.576	1.270
308.846	0.000

NOTE: Normal geometry dimensions are compatible
with Figure 5-4.

TABLE 5-14

PHOTON ENERGY GROUP STRUCTURE

Group Number	Energy Limits (Mev)		
1	7.5	-	---
2	7.0	-	7.5
3	6.0	-	7.0
4	5.0	-	6.0
5	4.0	-	5.0
6	3.0	-	4.0
7	2.6	-	3.0
8	2.2	-	2.6
9	1.8	-	2.2
10	1.35	-	1.8
11	0.9	-	1.35
12	0.4	-	0.9
13	-	-	0.4

TABLE 5-15

NEUTRON RESPONSE FUNCTIONS FOR USE IN NEUTRON ENVIRONMENT ANALYSIS

Group Number	$\left(\begin{array}{c} \text{Response Function} \\ \text{Equivalent Flux / n/cm}^2\text{-sec} \end{array} \right)$		
	Sulfur	Cd Covered U-238	Bare - Cd Covered Dysprosium
1	0.7859	1.087	3.817 (-7)
2	0.1018	0.8349	7.227 (-7)
3	0.00	0.04554	1.175 (-6)
4	0.00	4.073 (-4)	1.305 (-6)
5	0.00	4.792 (-4)	2.124 (-6)
6	0.00	7.711 (-4)	3.002 (-6)
7	0.00	1.436 (-3)	5.225 (-6)
8	0.00	2.359 (-3)	7.560 (-6)
9	0.00	6.125 (-3)	4.043 (-4)
10	0.00	0.01415	7.523 (-6)
11	0.00	0.01202	3.984 (-4)
12	0.00	0.01937	0.03032
13	0.00	0.06289	0.2640
14	0.00	0.08923	0.4457
15	0.00	0.1214	0.6876
16	0.00	0.2998	1.339

NOTE: 1) Numbers in parentheses refer to powers of ten.
 2) U-238 foils contained 190 ppm U-235.

TABLE 5-16

RESPONSE FUNCTION FOR USE IN PHOTON ENVIRONMENT ANALYSIS

Group Number	Photon Response Rad(c)/hr / Mev/cm ² -sec
1	8.957 (-7)
2	9.148 (-7)
3	9.475 (-7)
4	9.954 (-7)
5	1.066 (-6)
6	1.155 (-6)
7	1.232 (-6)
8	1.295 (-6)
9	1.368 (-6)
10	1.465 (-6)
11	1.580 (-6)
12	1.851 (-6)
13	3.311 (-6)

TABLE 5-17
COMPARISON OF RESULTS OF SINGLE CALCULATION WITH RESULTS OF
COUPLED CALCULATION IN DOSIMETER SLOT 1
 CONFIGURATION 16C

	SINGLE CALCULATION	COUPLED CALCULATION	RATIO: COUPLED/SINGLE
<u>SULFUR RESPONSE</u>			
R = 0 cm	3.324 (4)	3.397 (4)	1.022
R = 15 cm	3.152 (4)	3.219 (4)	1.021
R = 30 cm	2.551 (4)	2.606 (4)	1.022
R = 45 cm	1.510 (4)	1.545 (4)	1.023
R = 60 cm	5.764 (3)	5.899 (3)	1.023
<u>U-238 RESPONSE</u>			
R = 0 cm	1.306 (5)	1.345 (5)	1.030
R = 15 cm	1.240 (5)	1.274 (5)	1.027
R = 30 cm	1.007 (5)	1.035 (5)	1.028
R = 45 cm	6.001 (4)	6.184 (4)	1.030
R = 60 cm	2.344 (4)	2.416 (4)	1.031
<u>Dy RESPONSE</u>			
R = 0 cm	2.191 (2)	2.313 (2)	1.056
R = 15 cm	2.063 (2)	2.179 (2)	1.056
R = 30 cm	1.846 (2)	1.933 (2)	1.047
R = 45 cm	1.625 (2)	1.672 (2)	1.029
R = 60 cm	1.090 (2)	1.091 (2)	1.001
<u>CaF₂ RESPONSE</u>			
R = 0 cm	1.018 (0)	9.928 (-1)	0.9752
R = 15 cm	9.479 (-1)	9.237 (-1)	0.9745
R = 30 cm	7.679 (-1)	7.475 (-1)	0.9734
R = 45 cm	4.966 (-1)	4.848 (-1)	0.9762
R = 60 cm	2.542 (-1)	2.495 (-1)	0.9815

TABLE 5-18

COMPARISON OF SECONDARY PHOTON SOURCE PRODUCED
IN HOMOGENIZED AND EXPLICIT BATH MODULES

PHOTON GROUP	PHOTON SOURCE (MEV/SEC)							
	MODULE 1		MODULE 2		MODULE 3		MODULE 4	
	SMEARED	EXPLICIT	SMEARED	EXPLICIT	SMEARED	EXPLICIT	SMEARED	EXPLICIT
1	3.79 (6)	4.17 (6)	2.04 (6)	2.30 (6)	1.12 (6)	1.28 (6)	6.31 (5)	6.86 (5)
2	1.55 (6)	1.70 (6)	8.36 (5)	9.40 (5)	4.60 (5)	5.21 (5)	2.59 (5)	2.82 (5)
3	4.89 (7)	5.47 (7)	2.78 (7)	3.02 (7)	1.53 (7)	1.66 (7)	8.44 (6)	8.92 (6)
4	5.36 (6)	5.89 (6)	3.02 (6)	3.27 (6)	1.67 (6)	1.82 (6)	9.39 (5)	9.93 (5)
5	1.53 (7)	1.66 (7)	8.67 (6)	9.27 (6)	4.85 (6)	5.21 (6)	2.74 (6)	2.85 (6)
6	2.29 (7)	2.41 (7)	1.32 (7)	1.37 (7)	7.53 (6)	7.89 (6)	4.36 (6)	4.16 (6)
7	1.48 (7)	1.51 (7)	8.58 (6)	8.77 (6)	5.01 (6)	5.14 (6)	2.97 (6)	2.98 (6)
8	1.41 (7)	1.60 (7)	8.30 (6)	9.38 (6)	4.86 (6)	5.49 (6)	2.87 (6)	3.14 (6)
9	5.31 (6)	5.90 (6)	2.98 (6)	3.26 (6)	1.64 (6)	1.81 (6)	9.15 (5)	9.74 (5)
10	1.87 (7)	2.08 (7)	1.06 (7)	1.15 (7)	5.84 (6)	6.35 (6)	3.22 (6)	3.41 (6)
11	8.79 (7)	1.00 (8)	5.15 (7)	5.86 (7)	3.06 (7)	3.51 (7)	1.84 (7)	2.09 (7)
12	3.70 (8)	3.40 (8)	2.25 (8)	1.98 (8)	1.26 (8)	1.10 (8)	6.97 (7)	5.74 (7)
13	1.44 (6)	1.60 (6)	8.09 (5)	8.87 (5)	4.44 (5)	4.90 (5)	2.46 (5)	2.62 (5)
TOTAL	6.10 (8)	6.07 (8)	3.64 (8)	3.50 (8)	2.06 (8)	1.98 (8)	1.16 (8)	1.06 (8)

NOTE: NUMBERS IN PARENTHESES REFER TO POWERS OF TEN.

TABLE 5-19
MESH LINE DIMENSIONS FOR ANISN ANALYSIS OF
PAX-GO BATH SHIELD MOCKUPS (CONFIG. 16A, 16B, AND 16C)

LINE NO.	Z (CM)	LINE NO.	Z (CM)	LINE NO.	Z (CM)	LINE NO.	Z (CM)
1	0.00	41	207.906	81	236.390	121	254.390
2	0.848	42	209.811	82	236.785	122	254.784
3	5.239	43	211.716	83	237.179	123	255.178
4	11.172	44	212.850	84	237.573	124	255.572
5	17.105	45	213.984	85	237.967	125	255.966
6	23.038	46	219.661	86	238.361	126	256.361
7	28.971	47	220.911	87	238.755	127	256.753
8	34.904	48	222.161	88	239.708	128	257.149
9	40.835	49	223.739	89	240.102	129	257.543
10	55.349	50	223.806	90	240.496	130	257.937
11	59.159	51	223.873	91	240.890	131	258.331
12	62.969	52	223.939	92	241.284	132	259.284
13	66.779	53	224.006	93	241.679	133	259.678
14	70.589	54	224.073	94	242.073	134	260.072
15	71.859	55	225.026	95	242.467	135	260.466
16	73.129	56	225.420	96	242.861	136	260.860
17	73.229	57	225.814	97	243.255	137	261.255
18	73.339	58	226.208	98	243.649	138	261.649
19	73.446	59	226.602	99	244.602	139	262.043
20	73.921	60	226.997	100	244.996	140	262.437
21	74.500	61	227.391	101	245.390	141	262.831
22	75.500	62	227.785	102	245.784	142	263.225
23	77.500	63	228.179	103	246.178	143	264.177
24	82.500	64	228.573	104	246.573	144	264.571
25	90.219	65	228.967	105	246.967	145	264.965
26	97.938	66	229.920	106	247.361	146	265.359
27	105.657	67	230.314	107	247.755	147	265.753
28	113.376	68	230.708	108	248.149	148	266.148
29	121.095	69	231.102	109	248.543	149	266.542
30	128.814	70	231.496	110	249.496	150	266.936
31	136.533	71	231.891	111	249.890	151	267.330
32	144.252	72	232.285	112	250.284	152	267.724
33	151.971	73	232.679	113	250.678	153	268.119
34	159.690	74	233.073	114	251.072	154	269.072
35	168.409	75	233.467	115	251.467		
36	178.128	76	233.861	116	251.861		
37	187.847	77	234.814	117	252.255		
38	194.566	78	235.208	118	252.649		
39	202.285	79	235.602	119	253.043		
40	206.001	80	235.996	120	253.437		

TABLE 5-20

AXIAL MESH LINE DIMENSIONS FOR 1D TRANSPORT ANALYSIS OF CONFIGURATION 17
(BORATED STEEL - LIQUID HYDROGEN)

Mesh No.	Mesh Line (cm)	Mesh No.	Mesh Line (cm)	Mesh No.	Mesh Line (cm)	Mesh No.	Mesh Line (cm)	Mesh No.	Mesh Line (cm)
1	0.0	27	105.657	53	224.006	79	248.838	105	280.589
2	0.848	28	113.376	54	224.073	80	251.060	106	282.811
3	5.239	29	121.095	55	225.026	81	253.282	107	285.034
4	11.172	30	128.814	56	225.280	82	255.506	108	287.256
5	17.105	31	136.533	57	225.534	83	255.758	109	287.510
6	23.038	32	144.252	58	225.788	84	245.012	110	287.764
7	28.971	33	151.971	59	226.042	85	256.266	111	288.018
8	34.904	34	159.690	60	226.296	86	256.520	112	288.272
9	40.835	35	168.409	61	228.519	87	256.776	113	288.526
10	55.349	36	178.128	62	230.741	88	258.999	114	290.749
11	59.159	37	187.847	63	232.964	89	261.221	115	292.971
12	62.969	38	194.566	64	235.186	90	261.856	116	295.194
13	66.779	39	202.285	65	235.440	91	262.491	117	297.416
14	70.589	40	206.001	66	235.694	92	264.714	118	297.670
15	71.859	41	207.906	67	235.948	93	266.936	119	297.924
16	73.129	42	209.811	68	236.202	94	267.571	120	298.178
17	73.229	43	211.716	69	236.456	95	268.206	121	298.432
18	73.339	44	212.850	70	238.678	96	270.429	122	298.686
19	73.446	45	213.984	71	240.901	97	272.651	123	300.909
20	73.921	46	219.661	72	243.123	98	274.874	124	303.131
21	74.500	47	220.911	73	245.346	99	277.096	125	305.354
22	75.500	48	222.161	74	245.599	100	277.350	126	307.576
23	77.500	49	223.739	75	245.853	101	277.604	127	307.830
24	82.500	50	223.806	76	246.107	102	277.858	128	308.084
25	90.219	51	223.873	77	246.361	103	278.112	129	308.338
26	97.938	52	223.939	78	246.616	104	278.366	130	308.592
								131	308.846

TABLE 5-21

ENERGY GROUP STRUCTURE EMPLOYED IN ANISN ANALYSIS
OF PAX-GO SHIELD MOCKUPS

Group	Neutron Energy (MeV)	Photon Energy (MeV)
1	2.87 - 10.0	
2	1.35 - 2.87	
3	0.821 - 1.35	
4	0.388 - 0.821	
5	0.111 - 0.388	
6	0.015 - 0.111	
7	5.53 (-3) - 0.015	
8	5.83 (-4) - 5.53 (-3)	
9	7.89 (-5) - 5.83 (-4)	
10	1.068 (-5) - 7.89 (-5)	
11	1.86 (-6) - 1.068(-5)	
12	3.00 (-7) - 1.86 (-6)	
13	1.20 (-7) - 3.00 (-7)	
14	6.00 (-8) - 1.20 (-7)	
15	2.00 (-8) - 6.00 (-8)	
16	0.00 - 2.00 (-8)	
17		7.5 -
18		7.0 - 7.5
19		6.0 - 7.0
20		5.0 - 6.0
21		4.0 - 5.0
22		3.0 - 4.0
23		2.6 - 3.0
24		2.2 - 2.6
25		1.8 - 2.2
26		1.35 - 1.8
27		0.9 - 1.35
28		0.4 - 0.9
29		- 0.4

NOTE: Numbers in parentheses refer to powers of ten.

TABLE 5-22

RESPONSE FUNCTION FOR USE WITH DISCRETE ORDINATES ANALYSIS
OF RESPONSE OF A Cd COVERED U-238 (11 ppm U-235) FISSION CHAMBER

Group	Response Function {Equivalent Flux/n/cm ² -sec}
1	1.087
2	0.8347
3	0.04524
4	0.00002355
5	0.00002773
6	0.00004462
7	0.00008317
8	0.0001336
9	0.0003547
10	0.0008195
11	0.0006960
12	0.001122
13	0.003641
14	0.005166
15	0.007031
16	0.01736

TABLE 5-23

RESPONSE FUNCTION FOR USE WITH POINT KERNEL ANALYSIS
OF THE RESPONSE OF A Cd COVERED U-238 (11 ppm U-235) FISSION CHAMBER

(Equivalent Flux/Fiss/cm²-sec-MeV)

Group	Response Function	Group	Response Function
1	1.342	15	5.817(-6)
2	2.685	16	6.849(-6)
3	2.685	17	6.849(-6)
4	2.685	18	6.059(-6)
5	2.685	19	1.102(-6)
6	2.685	20	1.102(-6)
7	2.685	21	1.102(-6)
8	1.611	22	1.102(-6)
9	1.649	23	1.102(-6)
10	1.237	24	1.102(-6)
11	4.473(-2)	25	1.102(-6)
12	3.352(-2)	26	1.102(-6)
13	1.163(-5)	27	1.027(-6)
14	8.725(-6)		

NOTE: Numbers in parentheses refer to powers of ten.

TABLE 5-24
UNCERTAINTY FACTORS FOR 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
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
Shield Configuration 17			
Sulfur Data ⁽¹⁾	0.91	1.04	0.79
U-238 Data ⁽²⁾	1.25	1.74	0.91
Dy Thermal Flux Data ⁽³⁾	1.12	1.27	0.99
CaF ₂ TLD Data ⁽⁴⁾	0.62	0.67	0.58

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 5-25

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.

* Note that shield mockup was located on the aft end of the reactor for these experiments.

FIGURE 5-1

PAX-G0C REACTOR MOCKUP FOR SHIELD EXPERIMENTS
(SECTIONAL VIEW THROUGH THE REACTOR AXIS)

NOTE: 1) Dimensions are in cm.
2) Drawing is not to scale.

FIGURE 5-2
PAX-G0C REACTOR MOCKUP FOR SHIELD EXPERIMENTS
(TOP VIEW)

NOTE: 1) Dimensions are in cm.
2) Drawing is not to scale.

FIGURE 5-3. BATH SHIELD MOCKUPS USED FOR PAX-G0C EXPERIMENTS

DATA SLOT

NOTE: 1) Dimensions are in cm.
2) Drawing is not to scale.

FIGURE 5-4
BORATED STEEL-LIQUID HYDROGEN SHIELD MOCKUP
USED FOR PAX-GOC EXPERIMENTS

NOTE: (N) Indicates sequence of calculations.

FIGURE 5-5

FLOW CHART FOLLOWED TO CALCULATE NEUTRON AND PHOTON ENVIRONMENT
INTERNAL TO THE PAX-GOC SHIELD MOCKUPS

NOTE: 1) Drawing is not to scale.
 2) (N) Indicates material number.
 3) See Table 5-6 for material description.

FIGURE 5-6
 MESH GEOMETRY FOR BASIC REACTOR DOT CALCULATIONS
 FOR CONFIGURATIONS 16A and 16C

NOTE: 1) Drawing is not to scale.
2) (N) Indicates material number.
3) See Table 5-6 for material description.

FIGURE 5-7
MESH GEOMETRY FOR BASIC DOT CALCULATIONS FOR CONFIGURATION 17

NOTE: 1) Drawing is not to scale.
 2) N Indicates material number.
 3) See Table 5-11 for material description.

FIGURE 5-8
 MESH GEOMETRY FOR SHIELD DOT CALCULATIONS CONFIGURATION 16C

NOTE: 1) Drawing is not to scale.
2) N Indicates material number.
3) See Table 5-11 for material description.

FIGURE 5-9
MESH GEOMETRY FOR SHIELD DOT CALCULATIONS CONFIGURATION 16A

NOTE: 1) Drawing is not to scale.
 2) N Indicates material number.
 3) See Table 5-11 for material description.

FIGURE 5-10
 MESH GEOMETRY FOR SHIELD DOT CALCULATIONS CONFIGURATION 17

FIGURE 5-11
COMPARISON OF CALCULATED AND MEASURED AXIAL FISSION DISTRIBUTION
WITH BATH SHIELD MOCKUPS IN PLACE

FIGURE 5-12
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION
IN DOSIMETER SLOT NO. 1

FIGURE 5-13
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION
IN DOSIMETER SLOT NO. 4

FIGURE 5-14
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION
IN DOSIMETER SLOT NO. 7

FIGURE 5-15
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION
IN DOSIMETER SLOT NO. 10

FIGURE 5-16. RESPONSE OF SULFUR PELLETS AS A FUNCTION OF AXIAL POSITION THROUGH THE PAX-G0 CONFIGURATION 16C SHIELD MOCKUP

FIGURE 5-17
RESPONSE OF CADMIUM COVERED U-238 (190 PPM U-235) FOILS
AS A FUNCTION OF RADIAL POSITION IN DOSIMETER SLOT NO. 1

FIGURE 5-18

RESPONSE OF CADMIUM COVERED U-238 (190 PPM U-235) FOILS
AS A FUNCTION OF RADIAL POSITION IN DOSIMETER SLOT NO. 3

FIGURE 5-19

RESPONSE OF CADMIUM COVERED U-238 (190 PPM U-235) FOILS
AS A FUNCTION OF RADIAL POSITION IN DOSIMETER SLOT NO. 5

FIGURE 5-20. RESPONSE OF Cd COVERED U-238 (190 PPM U-235) FOILS AS A FUNCTION OF AXIAL POSITION THROUGH THE PAX-G0 CONFIGURATION 16C SHIELD MOCKUP

FIGURE 5-21

RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS
AS A FUNCTION OF RADIAL POSITION IN DOSIMETER SLOT NO. 1

FIGURE 5-22

RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS
AS A FUNCTION OF RADIAL POSITION IN DOSIMETER SLOT NO. 2

FIGURE 5-23

RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS
AS A FUNCTION OF RADIAL POSITION IN DOSIMETER SLOT NO. 3

FIGURE 5-24

RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS
AS A FUNCTION OF RADIAL POSITION IN DOSIMETER SLOT NO. 4

FIGURE 5-25

RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS
AS A FUNCTION OF RADIAL POSITION IN DOSIMETER SLOT NO. 7

FIGURE 5-26. BARE Cd COVERED DYSPROSIUM RESPONSE AS A FUNCTION OF AXIAL POSITION THROUGH THE PAX-G0 CONFIGURATION 16C SHIELD MOCKUP

FIGURE 5-27

RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION
IN DOSIMETER SLOT NO. 1

FIGURE 5-28
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION
IN DOSIMETER SLOT NO. 2

FIGURE 5-29
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION
IN DOSIMETER SLOT NO. 4

FIGURE 5-30
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION
IN DOSIMETER SLOT NO. 6

FIGURE 5-31
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF AXIAL POSITION
THROUGH THE PAX-G0 CONFIGURATION 16A SHIELD MOCKUP

FIGURE 5-32
RESPONSE OF CADMIUM COVERED U-238 (190 PPM U-235) FOILS
AS A FUNCTION OF RADIAL POSITION IN DOSIMETER SLOT NO. 3

FIGURE 5-33
RESPONSE OF CADMIUM COVERED U-238 (190 PPM U-235) FOILS
AS A FUNCTION OF RADIAL POSITION IN DOSIMETER SLOT NO. 5

FIGURE 5-34
RESPONSE OF CADMIUM COVERED U-238 (190 PPM U-235) FOILS
AS A FUNCTION OF AXIAL POSITION THROUGH THE PAX-G0

FIGURE 5-35
RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS
AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER SLOT NO. 1

FIGURE 5-36
RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS
AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER SLOT NO. 2

FIGURE 5-37
RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS
AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER SLOT NO. 3

FIGURE 5-38
RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS
AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER SLOT NO. 4

FIGURE 5-39
RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS
AS A FUNCTION OF AXIAL POSITION THROUGH THE PAX-G0

FIGURE 5-40
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION
IN DOSIMETER SLOT NO. 1

FIGURE 5-41
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION IN DOSIMETER
SLOT NO. 3

FIGURE 5-42
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION IN DOSIMETER
SLOT NO. 5

FIGURE 5-43
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION IN DOSIMETER
SLOT NO. 7

FIGURE 5-44
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION IN DOSIMETER
SLOT NO. 8

FIGURE 5-45
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION IN DOSIMETER
SLOT NO. 10

FIGURE 5-46
RESPONSE OF SULFUR PELLETS AS A FUNCTION OF RADIAL POSITION IN DOSIMETER
SLOT NO. 12

FIGURE 5-47. RESPONSE OF SULFUR PELLETS AS A FUNCTION OF AXIAL POSITION THROUGH THE PAX-G0

FIGURE 5-48
RESPONSE OF CADMIUM COVERED U-238 (190 PPM U-235) FOILS AS A FUNCTION OF
RADIAL POSITION IN DOSIMETER SLOT NO. 1

FIGURE 5-49
RESPONSE OF CADMIUM COVERED U-238 (190 PPM U-235) FOILS AS A FUNCTION OF
RADIAL POSITION IN DOSIMETER SLOT NO. 5

FIGURE 5-50. RESPONSE OF Cd COVERED U-238 (190 PPM U-235) FOILS AS A FUNCTION OF AXIAL POSITION THROUGH THE PAX-G0 CONFIGURATION 17 SHIELD MOCKUP

FIGURE 5-51.
RESPONSE OF BARD MINUS CADMIUM COVERED DYSPROSIUM FOILS AS A FUNCTION
OF RADIAL POSITION IN DOSIMETER SLOT NO. 1

FIGURE 5-52
RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS AS A FUNCTION
OF RADIAL POSITION IN DOSIMETER SLOT NO. 2

FIGURE 5-53
RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS AS A FUNCTION
OF RADIAL POSITION IN DOSIMETER SLOT NO. 4

FIGURE 5-54
RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS AS A FUNCTION
OF RADIAL POSITION IN DOSIMETER SLOT NO. 6

FIGURE 5-55
RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS AS A FUNCTION
OF RADIAL POSITION IN DOSIMETER SLOT NO. 7

FIGURE 5-56
RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS AS A FUNCTION
OF RADIAL POSITION IN DOSIMETER SLOT NO. 9

FIGURE 5-57
RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM FOILS AS A FUNCTION
OF RADIAL POSITION IN DOSIMETER SLOT NO. 11

FIGURE 5-58. BARE-Cd COVERED DYSPROSIUM RESPONSE AS A FUNCTION OF AXIAL P POSITION THROUGH THE PAX-G0 CONFIGURATION 17 SHIELD MOCKUP

FIGURE 5-59
RESPONSE OF CaF_2 TLDs AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER
SLOT NO. 1

FIGURE 5-60
RESPONSE OF CaF_2 TLDs AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER
SLOT NO. 4

FIGURE 5-61
RESPONSE OF CaF_2 TLDs AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER
SLOT NO. 7

FIGURE 5-62
RESPONSE OF CaF_2 TLDs AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER
SLOT NO. 9

FIGURE 5-63. CdF_2 TLD RESPONSE AS A FUNCTION OF AXIAL POSITION THROUGH THE PAX-G0 CONFIGURATION 16C SHIELD MOCKUP

FIGURE 5-64
RESPONSE OF CaF_2 TLDs AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER
SLOT NO. 1

FIGURE 5-65
RESPONSE OF CaF_2 TLDs AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER
SLOT NO. 2

FIGURE 5-66
 RESPONSE OF CaF_2 TLDs AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER
 SLOT NO. 4

FIGURE 5-67
RESPONSE OF CaF_2 TLDs AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER
SLOT NO. 6

FIGURE 5-68
 RESPONSE OF CaF_2 TLDs AS A FUNCTION OF AXIAL POSITION THROUGH THE PAX-G0
 CONFIGURATION 16A SHIELD MOCKUP

FIGURE 5-69
RESPONSE OF CaF_2 TLDs AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER
SLOT NO. 1

FIGURE 5-70
RESPONSE OF CaF_2 TLDs AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER
SLOT NO. 4

FIGURE 5-71
RESPONSE OF CaF_2 TLDs AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER
SLOT NO. 7

FIGURE 5-72
RESPONSE OF CaF_2 TLDs AS A FUNCTION OF RADIAL POSITION THROUGH DOSIMETER
SLOT NO. 10

FIGURE 5-73

RESPONSE OF CaF_2 TLD'S AS A FUNCTION OF RADIAL POSITION
THROUGH DOSIMETER SLOT NO. 12

FIGURE 5-74. RESPONSE OF CaF_2 TLD'S AS A FUNCTION OF AXIAL POSITION THROUGH THE PAX-G0 CONFIGURATION 17 MOCKUP

MESH LINE NO.		AXIAL DISTANCE (CM)	
154	DOSIMETER SLOT 10	269.072
153	BATH	268.119	
143	DOSIMETER SLOT 9	264.177	
142	BATH	263.225	
132	DOSIMETER SLOT 8	259.284	
131	BATH	258.331	
121	DOSIMETER SLOT 7	254.390	
120	BATH	253.437	
110	DOSIMETER SLOT 6	249.496	
109	BATH	248.543	
99	DOSIMETER SLOT 5	244.602
98	BATH	243.649	
88	DOSIMETER SLOT 4	239.708	
87	BATH	238.755	
77	DOSIMETER SLOT 3	234.814	
76	BATH	233.861	
66	DOSIMETER SLOT 2	229.920	
65	BATH	228.967	
55	DOSIMETER SLOT 1	225.026	
54	BORAL	224.073	
49	SHIELD SUPPORT PLATE	223.739
46	VOID	219.661	
45	TIE ROD ENDS	213.984	
43	SUPPORT BLOCKS	211.716	
40	CORE	206.001	
20	C. P. HARDWARE	73.921	
19	CLUSTER PLATE	73.446	
16	C. P. HARDWARE	73.129	
14	SUPPORT PLATE	70.589	
10	VOID	55.349	
9	ALUMINUM SHIELD	40.835
3	VOID	5.239	
2	PRESSURE VESSEL	0.848	
1		0.000	

FIGURE 5-75

AXIAL GEOMETRY FOR ANISN ANALYSIS OF PAX-G0 BATH SHIELD MOCKUPS

MESH LINE NO.		AXIAL DISTANCE (CM)	
131		308.846	
126	BORATED STEEL	307.576	
	POLYETHYLENE		
122	BORATED STEEL	298.686	
117	POLYETHYLENE	297.416	
	POLYETHYLENE		
113	BORATED STEEL	288.526	
108	POLYETHYLENE	287.256	
	POLYETHYLENE		
104	BORATED STEEL	278.366	
99	POLYETHYLENE	277.096	
	POLYETHYLENE		
95	STEEL	268.206	
93	POLYETHYLENE	266.936	
	POLYETHYLENE		
91	STEEL	262.491	
89	POLYETHYLENE	261.221	
	POLYETHYLENE		
87	BORATED STEEL	256.776	
82	POLYETHYLENE	255.506	
	POLYETHYLENE		
78	BORATED STEEL	246.616	
73	POLYETHYLENE	245.346	
	POLYETHYLENE		
69	BORATED STEEL	236.456	
64	POLYETHYLENE	235.186	
	POLYETHYLENE		
60	BORATED STEEL	226.296	
55	VOID	225.026	
54	BORAL	224.073	
49		223.739	
	SHIELD SUPPORT PLATE		
46	VOID	219.661	
45	TIE ROD ENDS	213.984	
43	SUPPORT BLOCKS	211.716	
40		206.001	
	CORE		
20	C. P. HARDWARE	73.921	
19	CLUSTER PLATE	73.446	
16	C. P. HARDWARE	73.129	
14	SUPPORT PLATE	70.589	
	VOID		
10	ALUMINUM SHIELD	55.349	
9	VOID	40.835	
3	VOID	5.239	
2	PRESSURE VESSEL	0.848	
1		0.000	

FIGURE 5-76
AXIAL GEOMETRY FOR ANISN ANALYSIS OF PAX-G0
POLYETHYLENE-STEEL SHIELD MOCKUP

FIGURE 5-77
FLOW CHART FOLLOWED FOR ONE-DIMENSIONAL ANALYSIS
OF PAX-G0 SHIELD MOCKUPS

FIGURE 5-78. RESPONSE OF SULFUR PELLETS THROUGH THE PAX-G0 CONFIGURATION 16A SHIELD MOCKUP

FIGURE 5-79. RESPONSE OF SULFUR PELLETS THROUGH THE PAX-G0 CONFIGURATION 16B SHIELD MOCKUP

FIGURE 5-80. RESPONSE OF SULFUR PELLETS THROUGH THE PAX-G0 CONFIGURATION 16C SHIELD MOCKUP

FIGURE 5-81. RESPONSE OF Cd COVERED U-238 (190 PPM U-235) FOILS THROUGH THE PAX-G0 CONFIGURATION 16A SHIELD MOCKUP

FIGURE 5-82. RESPONSE OF Cd COVERED U-238 (190 PPM U-235) FOILS THROUGH THE PAX-G0 CONFIGURATION 16B SHIELD MOCKUP

FIGURE 5-83. RESPONSE OF Cd COVERED U-238 (190 PPM U-235) FOILS THROUGH THE PAX-G0 CONFIGURATION 16C SHIELD MOCKUP

FIGURE 5-84. RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM THROUGH THE PAX-G0 CONFIGURATION 16A SHIELD MOCKUP

FIGURE 5-85. BARE-Cd DYSPROSIUM RESPONSE THROUGH THE PAX-G0 CONFIGURATION 16B SHIELD MOCKUP

FIGURE 5-86. BARE-Cd COVERED DYSPROSIUM RESPONSE THROUGH THE PAX-GO CONFIGURATION 16C SHIELD MOCKUP

FIGURE 5-87. RESPONSE OF SULFUR PELLETS THROUGH THE PAX-G0 CONFIGURATION 17 SHIELD MOCKUP

FIGURE 5-88. RESPONSE OF Cd COVERED U-238 (190 PPM U-235) FOILS THROUGH THE PAX-G0 CONFIGURATION 17 SHIELD MOCKUP

FIGURE 5-89. RESPONSE OF BARE MINUS CADMIUM COVERED DYSPROSIUM THROUGH THE PAX-G0 CONFIGURATION 17 SHIELD MOCKUP

FIGURE 5-90. RESPONSE OF CaF_2 TLD THROUGH THE PAX-G0 CONFIGURATION 16A SHIELD MOCKUP

FIGURE 5-91. RESPONSE OF CaF_2 TLD THROUGH THE PAX-G0 CONFIGURATION 16B SHIELD MOCKUP

FIGURE 5-92. RESPONSE OF CaF_2 TLD THROUGH THE PAX-G0 CONFIGURATION 16C SHIELD MOCKUP

FIGURE 5-93. RESPONSE OF CaF_2 TLD THROUGH THE PAX-G0 CONFIGURATION 17 SHIELD MOCKUP

FIGURE 5-95. GEOMETRY FOR PAX-G0, CONFIGURATION 16C
SHIELD MOCKUP - KAP-V CALCULATIONS

FIGURE 5-96

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER AS A FUNCTION OF POLAR ANGLE ON A 152.4 CM MERIDIAN RING

FIGURE 5-97

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER AS A
FUNCTION OF POLAR ANGLE ON A 167.6 CM MERIDIAN RING

FIGURE 5-98

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER AS A FUNCTION OF POLAR ANGLE ON A 182.9 CM MERIDIAN RING

FIGURE 5-99

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER AS A
FUNCTION OF POLAR ANGLE ON A 198.1 CM MERIDIAN RING

FIGURE 5-100

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER AS A FUNCTION OF POLAR ANGLE ON A 213.4 CM MERIDIAN RING

FIGURE 5-101

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 274.3 CM MERIDIAN RING

FIGURE 5-102
 RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
 AS A FUNCTION OF POLAR ANGLE ON A 335.3 CM MERIDIAN RING

FIGURE 5-103

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 396.2 CM MERIDIAN RING

FIGURE 5-104

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 457.2 CM MERIDIAN RING

FIGURE 5-105

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 518.2 CM MERIDIAN RING

FIGURE 5-106

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 574.0 CM MERIDIAN RING

FIGURE 5-107

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF DISTANCE FROM CORE CENTER AT A POLAR ANGLE
OF 0 DEGREES

FIGURE 5-108
 RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
 AS A FUNCTION OF DISTANCE FROM CORE CENTER AT A POLAR ANGLE
 OF 10 DEGREES

FIGURE 5-109
RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF DISTANCE FROM CORE CENTER AT A POLAR ANGLE
OF 20 DEGREES

FIGURE 5-110
 RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
 AS A FUNCTION OF DISTANCE FROM CORE CENTER AT A POLAR ANGLE
 OF 30 DEGREES

FIGURE 5-111
RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 152.4 CM MERIDIAN RING

FIGURE 5-112
 RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
 AS A FUNCTION OF POLAR ANGLE ON A 167.6 CM MERIDIAN RING

FIGURE 5-113

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 182.9 CM MERIDIAN RING

FIGURE 5-114

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 198.1 CM MERIDIAN RING

FIGURE 5-115

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 213.4 CM MERIDIAN RING

FIGURE 5-116

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 274.3 CM MERIDIAN RING

FIGURE 5-117

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 335.3 CM MERIDIAN RING

FIGURE 5-118

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 396.2 CM MERIDIAN RING

FIGURE 5-119

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 457.2 CM MERIDIAN RING

FIGURE 5-120

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 518.2 CM MERIDIAN RING

FIGURE 5-121

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF POLAR ANGLE ON A 574.0 CM MERIDIAN RING

FIGURE 5-122

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF DISTANCE FROM CORE CENTER AT A POLAR ANGLE
OF 0 DEGREES

FIGURE 5-123
RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF DISTANCE FROM CORE CENTER AT A POLAR ANGLE
OF 10 DEGREES

FIGURE 5-124
RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF DISTANCE FROM CORE CENTER AT A POLAR ANGLE
OF 20 DEGREES

FIGURE 5-125
RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER
AS A FUNCTION OF DISTANCE FROM CORE CENTER AT A POLAR ANGLE
OF 30 DEGREES

FIGURE 5-126

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER AS A FUNCTION OF POLAR ANGLE ON A 182.9 CM MERIDIAN RING

FIGURE 5-127

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER AS A
FUNCTION OF POLAR ANGLE ON A 198.1 CM MERIDIAN RING

FIGURE 5-128

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER AS A
 FUNCTION OF POLAR ANGLE ON A 213.4 CM MERIDIAN RING

FIGURE 5-129

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER AS A
FUNCTION OF POLAR ANGLE ON A 274.3 CM MERIDIAN RING

FIGURE 5-130

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER AS A
FUNCTION OF POLAR ANGLE ON A 335.3 CM MERIDIAN RING

FIGURE 5-131

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER AS A
FUNCTION OF POLAR ANGLE ON A 396.2 CM MERIDIAN RING

FIGURE 5-132

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER AS A FUNCTION OF POLAR ANGLE ON A 457.2 CM MERIDIAN RING

FIGURE 5-133

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER AS A
FUNCTION OF POLAR ANGLE ON A 518.2 CM MERIDIAN RING

FIGURE 5-134

RESPONSE OF CADMIUM COVERED U-238 (11 PPM U-235) FISSION CHAMBER AS A FUNCTION OF POLAR ANGLE ON A 574.0 CM MERIDIAN RING

FIGURE 5-135

RESPONSE OF A CARBON WALL- CO_2 - GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 152.4 CM MERIDIAN RING

FIGURE 5-136
RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 167.6 CM MERIDIAN RING

FIGURE 5-137

RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 182.9 CM MERIDIAN RING

FIGURE 5-138
 RESPONSE OF A CARBON WALL- CO_2 - GAS ION CHAMBER AS A FUNCTION OF
 POLAR ANGLE ON A 198.1 CM MERIDIAN RING

FIGURE 5-139
RESPONSE OF A CARBON WALL- CO_2 - GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 213.4 CM MERIDIAN RING

FIGURE 5-140
RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 274.3 CM MERIDIAN RING

FIGURE 5-141

RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 335.3 CM MERIDIAN RING

FIGURE 5-142
RESPONSE OF A CARBON WALL-CO₂ - GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 396.2 CM MERIDIAN RING

FIGURE 5-143
RESPONSE OF A CARBON WALL-CO₂ - GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 457.2 CM MERIDIAN RING

FIGURE 5-144

RESPONSE OF A CARBON WALL- CO_2 - GAS ION CHAMBER AS A FUNCTION OF POLAR ANGLE ON A 518.2 CM MERIDIAN RING

FIGURE 5-145
RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 574.0 CM MERIDIAN RING

FIGURE 5-146

RESPONSE OF A CARBON WALL- CO_2 -GAS ION CHAMBER AS A FUNCTION OF DISTANCE FROM CORE CENTER AT A POLAR ANGLE OF 0 DEGREES

FIGURE 5-147
RESPONSE OF CARBON WALL-CO₂ - GAS ION CHAMBER AS A FUNCTION OF
DISTANCE FROM CORE CENTER AT A POLAR ANGLE OF 10 DEGREES

FIGURE 5-148

RESPONSE OF CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF DISTANCE FROM CORE CENTER AT A POLAR ANGLE OF 20 DEGREES

FIGURE 5-149
RESPONSE OF CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
DISTANCE FROM CORE CENTER AT A POLAR ANGLE OF 30 DEGREES

FIGURE 5-150

RESPONSE OF A CARBON WALL- CO_2 - GAS ION CHAMBER AS A FUNCTION OF
 POLAR ANGLE ON A 152.4 CM MERIDIAN RING

FIGURE 5-151
RESPONSE OF A CARBON WALL-CO₂-GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 167.6 CM MERIDIAN RING

FIGURE 5-152
RESPONSE OF A CARBON WALL-CO₂ - GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 182.9 CM MERIDIAN RING

FIGURE 5-153
RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 198.1 CM MERIDIAN RING

FIGURE 5-154
RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 213.4 CM MERIDIAN RING

FIGURE 5-155
RESPONSE OF A CARBON WALL- CO_2 - GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 274.3 CM MERIDIAN RING

FIGURE 5-156
RESPONSE OF A CARBON WALL- CO_2 - GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 335.3 CM MERIDIAN RING

FIGURE 5-157
RESPONSE OF A CARBON WALL-CO₂-GAS ION CHAMBER AS A FUNCTION OF POLAR
ANGLE ON A 396.2 CM MERIDIAN RING

FIGURE 5-158
RESPONSE OF A CARBON WALL-CO₂-GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 457.2 CM MERIDIAN RING

FIGURE 5-159
 RESPONSE OF A CARBON WALL- CO_2 - GAS ION CHAMBER AS A FUNCTION OF
 POLAR ANGLE ON A 518.2 CM MERIDIAN RING

FIGURE 5-160
RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 574.0 CM MERIDIAN RING

FIGURE 5-161
RESPONSE OF CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
DISTANCE FROM CORE CENTER AT A POLAR ANGLE OF 0 DEGREES

FIGURE 5-162
RESPONSE OF CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
DISTANCE FROM CORE CENTER AT A POLAR ANGLE OF 10 DEGREES

FIGURE 5-163
RESPONSE OF CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
DISTANCE FROM CORE CENTER AT A POLAR ANGLE OF 20 DEGREES

FIGURE 5-164
RESPONSE OF CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
DISTANCE FROM CORE CENTER AT A POLAR ANGLE OF 30 DEGREES

FIGURE 5-165
RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 182.9 CM MERIDIAN RING

FIGURE 5-166
 RESPONSE OF A CARBON WALL- CO_2 - GAS ION CHAMBER AS A FUNCTION OF
 POLAR ANGLE ON A 198.1 CM MERIDIAN RING

FIGURE 5-167
RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 213.4 CM MERIDIAN RING

FIGURE 5-168
RESPONSE OF A CARBON WALL-CO₂-GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 274.3 CM MERIDIAN RING

FIGURE 5-169
RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 335.3 CM MERIDIAN RING

FIGURE 5-170
RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 396.2 CM MERIDIAN RING

FIGURE 5-171
RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 457.2 CM MERIDIAN RING

FIGURE 5-172
RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 518.2 CM MERIDIAN RING

FIGURE 5-173

RESPONSE OF A CARBON WALL-CO₂- GAS ION CHAMBER AS A FUNCTION OF
POLAR ANGLE ON A 574.0 CM MERIDIAN RING

FIGURE 5-174
 RESPONSE OF U-238 (11 PPM U-235) FISSION CHAMBER AS A FUNCTION OF BATH
 MASS THICKNESS ON THE REACTOR CENTERLINE APPROXIMATELY 66"
 FORWARD OF PAX-G0 CORE CENTER

FIGURE 5-175
RESPONSE OF U-238 (11 PPM U-235) FISSION CHAMBER AS A FUNCTION OF BATH
MASS THICKNESS NEAR THE SHIELD PERIPHERY APPROXIMATELY 66" FORWARD
OF THE PAX-GO CORE CENTER

FIGURE 5-176
 RESPONSE OF A CARBON WALL-CO₂-GAS ION CHAMBER AS A FUNCTION
 OF BATH MASS THICKNESS ON THE REACTOR CENTERLINE APPROXIMATELY
 66" FORWARD OF THE PAX-G0 CORE CENTER

FIGURE 5-177

RESPONSE OF A CARBON WALL-CO₂-GAS ION CHAMBER AS A FUNCTION
OF BATH MASS THICKNESS NEAR THE SHIELD PERIPHERY APPROXIMATELY
66" FORWARD OF THE PAX-G0 CORE CENTER

6.0 CONCLUSIONS

Based on the comparisons between the measurements and calculations presented in the preceding section, the following conclusions are 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 mock-ups 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) The one-dimensional technique, at least in slab geometry, should not be employed to estimate the absolute radiation environment internal to shield configurations of the type examined in this report. In fact, as indicated by the results of the analysis of the borated steel-liquid hydrogen shield, the use of the ANISN code to perform analysis on a relative basis may, under certain circumstances, be highly questionable. The problem associated with the one-dimensional analysis is attributed to the transverse leakage approximation utilized in the calculation. The error induced by employing the transverse leakage approximation is largest for regions in which the actual leakage is relatively large; and, as might be expected in regions with either large transverse dimensions or high density material these effects are somewhat mollified. Since NERVA type reactors include axial regions which contain material of widely varying density, it is imperative that a thorough understanding of the effects of transverse leakage on analytical results be gained in order to sensibly apply the technique to the design situation.
- 4) Problems associated with the cornering effect of the shadow shields on external environment 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.

5) The discrete ordinates technique (S_8) produced anomalous results at large distances from the reactor on the centerline of the system. This anomalous behavior was intensified with increasing shield opacity.

6) Uncertainty factors applicable to internal and external environment data were developed.

One of the objectives of this experimental-analytical program was to compare shielding effectiveness of the BATH material with that of a borated steel-liquid hydrogen laminate. Unfortunately, differences in background environment for the BATH and borated steel-LH₂ data prevents the above comparison from being made on an experimental basis. It is possible, however, to compare the shielding effectiveness of the two materials strictly on an analytical basis. Using the point kernel calculated radiation environment external to the various shield mockups, it can be shown that, on a weight basis, the BATH and BSS-LH₂ shields are roughly equivalent for both neutron and photon attenuation for locations directly forward of the shield mockups. The BSS-LH₂ shield is, however, almost twice as thick as the corresponding BATH shield. As a result of its larger axial dimension and correspondingly greater transverse leakage the BSS-LH₂ shield is somewhat less effective than BATH at off-axis locations.

Based on the study reported herein the following recommendations are made:

1) 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. A thorough investigation of the basic approach to the problem may also be in order. 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. Again, some of the disparity in point kernel and discrete ordinates calculated photon radiation levels external to the shield mockups may be resolved through an investigation of the basic cross-section data employed in these analyses.

2) The problem in the experimental discipline of the accurate determination of the background radiation environment external to the shield mockups must be resolved. It is quite clear that background problems produced anomalous results at large distances forward of the configuration 16C mockup in both the neutron and photon measurements. Likewise, along the 30° traverse, the background appears to be too high for all of the various shield configurations. In addition, the anomalous behavior of all of the data pertaining to the borated steel-liquid hydrogen mockup may have been due entirely to an inadequate determination of the background environment. It is of paramount importance that all avenues of approach toward accurate determination as well as reduction of the background environment be thoroughly explored, especially the use of detector collimators for external radiation measurements.

3) The two-dimensional discrete ordinates analysis performed on the borated steel-liquid hydrogen shield mockup raised several questions regarding the calculation of radiative transport internal to shields which are composed of low density material and, thus, physically thick. In particular, the anomaly, on the reactor centerline, in the sulfur data comparisons points up a problem area which should be investigated further. Similarly, the apparent inability of the analytical approach to accurately predict the photon radiation levels internal to the borated steel-liquid hydrogen mockup gives some cause for concern and most certainly warrants deeper investigation.

APPENDIX A

THE NUCLEAR SUBSYSTEM SHIELDING PROVISION STUDIES EXPERIMENTAL PROCEDURE

In accordance with WANEF procedures*, four Experimental Details Checklists (EDC) were written to set forth the details of experimental procedures to be used in making the measurements associated with the nuclear subsystem shielding provision study. The experimental procedures applicable to shield configuration 15, and the wall lining procedure are found in the Experimental Details section of EDC 69. The experimental procedures applicable to shield configuration 14 are found in the Experimental Details section of EDC 71. The experimental procedures applicable to shield configurations 16A, 16B and 16C are found in the Experimental Details of EDC 72. The experimental procedures applicable to shield configuration 17 are found in the Experimental Details of EDC 73. The bodies of EDC's 69, 71, 72 and 73 are contained in this appendix.

* WANL-TME-267, Revision H, "Westinghouse Astronuclear Experimental Facility Reactor Operations Manual".

INFORMATION CATEGORY

Unclassified
J. L. Corcoran 2/26/69
Authorized Classifier Date

WANEF

EXPERIMENTAL DETAILS CHECKLIST*

No. 69

Date 2/26/69

TITLE OF EXPERIMENT: Nuclear Subsystem Shielding Provision Study Experiment
(Bare Core Experiments and Wall Lining Procedures)

OBJECTIVES:

- 1) To determine the radiation environment above the end of the PAX-G0 reactor with no shield mockup present, i. e. , Configuration 15.
- 2) To reduce the test cell background by the appropriate use of materials around the reactor.

EQUIPMENT REQUIREMENTS:

- 1) The PAX-G0 reactor shimmed to $115 \pm 1^\circ$ with shield adapter ring in place.
- 2) The (ρ, θ, ϕ) traversing system, remotely controlled, on which the active dosimeters can be mounted.
- 3) Detecting systems consisting of CO_2 ionization chambers and Li^7F and CaF TLD's for gamma ray dose rates; Hurst dosimeter for fast neutron dose rate; U-238 fast fission chamber wrapped in 20 mil Cd inside 1/4" thick Boral (35 w/o B_4C) and U-238 and/or thorium foils and sulfur pellets for fast neutron flux, bare and cadmium-covered dysprosium foils for thermal neutron flux, aluminum and beryllium Bragg-Gray chambers to determine the gamma ray heating rates; and one inch long 0.040" diameter U-Al wires for axial fission density measurements.
- 4) Three 52" x 52" x 1/8" Boral (35 w/o B_4C) sheets rolled to fit on the external surface of the pressure vessel. A 52" x 6" x 1/8" Boral (35 w/o B_4C) piece. The top of these sheets will be placed approximately even with the top of the reflector and will be securely fastened to the pressure vessel with the use of at least two steel bands to form a 360° shield.

* In partial response to the Scope of Test Document 62DW-268-1 for the Nuclear Subsystem Shielding Provision Study Experiments dated January 21, 1969.

- 5) A Boral enclosure as used previously in Addendum No. 4, EDC-62, which consists of a 52-inch³ box composed of five 1/4-inch thick and 52 x 52-inch square (35 w/o B₄C) Boral sheets bolted to a rigid frame made from Equipto angle.
- 6) The overhead test cell crane capable of remote movement from the control console.
- 7) Three closed circuit TV cameras and two 50-foot tapes to measure polar angle and distance of the center of detectors about the center of the core.
- 8) A fixture for centerline and one radial passive dosimetry traverse comprised of a 14-foot length of 1/2" O.D. x 0.028" wall thickness aluminum tubing with an aluminum adapter at one end. The adapter will be machined to fit in the tie tube cup of the central cluster for base support. Four guy wires (heavy duty string) will be securely attached to the tubing at a position of approximately eight feet from the adapter. These wires will be securely attached to the shield adapter ring to position the aluminum tube parallel to the reactor axis. An aluminum bar or tube (approximately 1/2 inch diameter) will be securely attached to the tubing for a radial traverse of passive dosimetry. It will be placed perpendicular to the tubing at a position that will permit the passive dosimeters to be placed at a distance of 83.85 cm above the core midplane out to a radius of 60 cm.
- 9) An aluminum tube that will be the same as item 8), except that it will be 50" long. This will be used to perform Dy-Al activation on the bare core inside the Boral enclosure defined in item 5).
- 10) A WX-5362 B-10 compensated ion chamber and associated equipment securely fastened under the dome end of the reactor for power level indication.
- 11) Sixteen 4' x 8' x 1" and four 4' x 4' x 1" borated polyethylene (2 w/o boron) sheets, six 5' x 3' x 1/8" and two 4' x 3' x 1/8" lead sheets to be placed in position against the test cell walls.
- 12) Welded frames made from Unistrut to secure the B-CH₂ next to the concrete walls.

HAZARDS CONSIDERATIONS:

A. Materials Lining Test Cell Walls

The north and west test cell walls are to be covered with borated polyethylene (2 w/o boron) to a total of 2 inch thickness in a 8' x 16' area above the

test stand and 4' x 4' area below the test stand centered on the reactor centerline. The general view of the area to be covered is depicted in Figure 1. The area in the general vicinity of the reactor centerline will be covered with 1/8 inch thick Pb sheets that are 5' x 3' and 4' x 3'.

The method of securing the borated polyethylene to the walls in order to prevent the sheets from falling on the test stand is as follows:

A frame is fabricated from 1-1/2 inch Unistrut steel and secured to the test cell walls by means of 3/8 inch bolts and concrete anchors. Figure 2 is an isometric view of the method of support of the frame to the test cell walls. The vertical pieces are single 8 foot length 1-1/2 inch Unistrut steel pices. Between each of the five vertical pieces are welded 4 foot pieces of Unistrut. This makes a unibody construction of the frame. Each vertical piece of Unistrut is securely anchored at the top and middle to the wall by means of small pieces of Unistrut, angle iron and 3/8 inch bolts and concrete anchors. At the base of the vertical member, steel angle is welded to the test stand deck. The frame is then bolted to the angle iron. The method of attaching the Unistrut frame to the wall at the top and middle is shown in the photo presented in Figure 3.

The borated polyethylene will be placed vertically against the frame and permitted to rest on the test stand deck. Figure 4 is an isometric view of the material in position. The inside (towards the wall) 1 inch layer will consist of 2 pieces that are 2 feet wide and 3 pieces that are 4 feet wide. The outside 1 inch layer will be four - 4 foot pieces. The joints between the inside and outside layers will be staggered as shown in the figure. On the outside of each 4 foot wide piece will be placed vertical pieces of 1-1/2 inch Unistrut. Two vertical pieces of Unistrut will be used for each four foot section as shown in Figure 4. The top will be bolted to the concrete ledge with 3/8 inch bolts and concrete anchors. The base will be bolted to steel angle that is welded to the test stand deck. Four pieces of angle attached to each vertical piece of Unistrut will be used to bolt the two one inch pieces of borated polyethylene together and to the inside frame that is attached to the wall. Cut-away A of Figure 4 depicts this method of attachment. These four pieces of angle will be attached to all of the outside vertical members.

The 1/8 inch thick Pb sheets will be secured to the outside surface of the borated polyethylene by means of horizontal pieces of aluminum secured to the outside vertical Unistrut members. The lead sheets will also be placed between the vertical Unistrut members and the borated polyethylene. At least two horizontal members will be used to support the Pb sheets (5' x 3' x 1/8").

The total mass of the borated polyethylene, Pb and Unistrut frame on the west wall above the stand is 2200 lbs. (1380 lbs. of B-CH₂, 420 lbs. of Pb and 400 lbs. of Unistrut). If the test stand deck should fail the frame is capable of supporting the liner material with a safety factor in excess of 5. This is based upon concrete anchors capability

of supporting a load of 450 lbs. (based upon the supplier's catalog of safe loads) with the force applied perpendicular to the wall. However, at least one house jack will be placed between the main test cell floor and the test stand deck at a suitable location near the lining material for additional support of the test stand deck.

The test cell floor has been load tested* to 14,000 lbs. in excess of the reactor assembly and test stand. Should the wall lining material fall to the test stand, the test cell floor will be capable of supporting the additional load.

The borated polyethylene placement on the north test cell wall will be similar to that on the west wall. An area of 8' x 16' will be covered above the test stand level and a 4' x 4' area below the test stand level. Figure 5 depicts the inside Unistrut frame which is mounted to the test cell walls. This frame will be mounted to the wall in the same fashion as that on the west wall. However, the presence of the steps to the fuel storage area prevents identical fabrication. The same type of materials will be used. Two 4' x 8' x 1" pieces will be placed next to the west wall. Two vertical pieces of Unistrut, one on each side, will be attached to the fuel storage shelf and test stand deck in the same manner as those on the west wall. The pieces that go above the stairs to the fuel storage shelf will be cut at the angle of the steps. These borated polyethylene pieces will rest on the steps. To the steps will be welded angle iron to which vertical pieces (two per four foot wide piece) of Unistrut will be bolted to these angles. The top of the Unistrut will be secured to the shelf in the same manner as the west wall supports.

The borated polyethylene below the steps on the north wall will also have two vertical pieces of Unistrut per four foot wide piece. The top of the Unistrut will be bolted to angle that is welded to the underside of the steps. The bottom of these vertical pieces will be anchored to the wall by concrete anchors and angles.

As on the west wall, the borated polyethylene and the inner frame of Unistrut will be bolted to the vertical outer pieces in four locations per vertical piece. The pieces above the stairs to the fuel storage shelf will have a minimum of two depending upon the position above or below the stairs.

The method of support of the 4' x 4' x 2" sections of B-CH₂ and one piece of 4' x 3' x 1/8" Pb under the test stand deck on the north and west wall will be to construct a 4' x 4' frame of Unistrut into which will be placed the B-CH₂ sheets. This assembly will be securely welded to the underside of the test stand deck on the west and north walls and centered about the reactor centerline.

* EDC-63, Addendum I, Appendix III.

When the borated polyethylene is being placed on the walls, only one piece at a time will be brought to the test stand deck and fixed to the Unistrut frame before the next piece is brought to the test stand deck. This is done in order to avoid possible violation of the minimum shutdown with three personnel** present near the reactor. Since the CH_2 is borated (2 w/o boron), this will be a conservative estimate of the maximum reactivity change due to a sheet of CH_2 falling during installation. The resulting configuration that is expected to yield the largest reactivity insertion would be for two sheets to pile up on top of the bare core and rest upon the shield adapter ring. The top of the adapter ring is 5.35 inches from the top of the core. Figure 6 shows the material coming to rest at an angle to the core; the dotted line shows a 2-inch thickness directly on the adapter ring.

The worth of the two-inch thick material coming to rest on the adapter ring is estimated from WANL-TNR-099. From Figure 5-A, the worth of a 2-inch thickness of polyethylene 35 inches in diameter at the top of the core is 1.85\$. The worth of the CH_2 on top of the adapter ring (5.35 inches from the top of the core) is obtained from Figure A-6 of WANL-TNR-099*. The worth of two inches of CH_2 5.35 inches from the top of the core is 0.37 (2.4/6.5) times the worth on top of the core. This makes 2 inches of CH_2 placed on the adapter ring worth approximately +69¢.

In addition to the 69¢ for the borated CH_2 placed horizontally on top of the adapter ring, an additional 1¢ could be added in the event that additional poly falls from the wall diagonally against the reactor as shown in Figure 6. The latter reactivity effects are based on measurements performed on the PAX-E5 reactor with Configuration 12 of the R-1 shielding parametric measurements as reported in WANL-TME-1852. The 1/8" Pb sheets on the CH_2 would not increase the 69¢ value since CH_2 would be considered as causing the maximum change.

* Analysis by Nuclear Design staff members indicates axial worth of CH_2 in WANL-TNR-099 are valid for the PAX/R-1 reactor.

** Present shutdown 4.44\$.

An Industrial Hygiene and Safety committee composed of Messrs. E. G. Troutman, R. A. Scarano and M. R. Beebe considered the fire hazards associated with two 1" thick 2 w/o boron CH_2 sheets and determined there were no undue hazards involved. Two more CO_2 fire extinguishers will be installed in the test cell as recommended by the committee.

B. Materials Around the PAX Pressure Vessel

A full 360° Boral shield (1/8" thick 35 w/o B_4C) will be securely banded directly outside the cylindrical PAX-G0 pressure vessel. The measured reactivity worth of a 120° Boral shield (1/8" thick 35 w/o B_4C) has been shown to be $+1.05\%$ *, thus the total worth of the 360° Boral shield is expected to be $+4.10\%$ **. The worth of the steel banding material holding the shields against the pressure vessel is negligible. If the banding materials should fail, the Boral shield will fall to a less reactive position.

C. Traverse Systems and Detectors

The various active dosimeters used in this experiment will be mounted upon and remotely moved by, a traverse system. The traverse system controls will be located in the control room and one or more TV systems will be used to monitor the location of the detectors with respect to the PAX core. The (ρ, θ, ϕ) traverse system will be utilized to obtain dosimetry data by remote rotation of the traverse system about the core center using the test cell crane.

* EDC-63, Addendum 1, April 16, 1968, Appendix III. These measurements were made on the PAX-E5 reactor.

** This value includes a 30% increase in reactivity worth as a safety margin for possible differences between the NRX PAX and PAX/R-1 reactors.

The (ρ, θ, ϕ) traverse system consists of two aluminum rails 16" apart. One end is attached to a yoke pivoted about the midplane of the PAX so that the rails can be elevated to some fixed polar angle. A small motor controlled from the control room will drive a platform on a rack and pinion gear drive along the rails which may be extended to 20'. This system weighs about 40 kgs. With a coefficient of .1¢/kg (WANL-TNR-099 Figure A-3) at 12" away from the pressure vessel, it is worth 4¢. Previous experiments using the (ρ, θ, ϕ) traverse system on the bare PAX-E5 reactor have shown that movement of the traverse system and the detectors about the core have caused negligible reactivity perturbations. Should the detector platform, detectors and rails accidentally fall onto the bare core, the expected reactivity insertion will be much less than 1\$. Therefore, no position interlocks will be required. However, mechanical and electrical stops will be placed at each end of the track to preclude the platform overrunning the end of the track.

D. Passive Dosimetry

The irradiation of virgin fuel slices will be performed in a central cluster for power level determination. Two Th-232 (no thermally fissionable isotopes present) and two U-238 (190 ppm U-235) foils will be irradiated on either side of the core midplane fuel slice. Passive dosimetry of U-238, Th-232, S and TLD's will also be performed. Since Th-232 has approximately the same fission threshold as U-238 and a fission cross section that is about a factor of four less than U-238, the Th-232 and U-238 foils in the center of the core will be worth less than 1¢. This is based upon previous experience with foil irradiation in the core. The foils located external to the bare core and the aluminum support will introduce a negligible reactivity insertion due to any support failure.

E. Configuration 15 - Bare Core

The presence of the (ρ, θ, ϕ) traverse system centered at $\phi = 270^\circ$ will preclude the use of at least one PASS system hose (the one near control drum 11). The remaining PASS system hose will be connected.

F. Boral Enclosure Above Bare Core

A Boral enclosure as used previously in Addendum No. 4, EDC-62, which consists of a hollow box composed of five 1/4 inch thick, 52 x 52 inch (35 w/o B₄C) Boral sheets bolted to a rigid frame constructed from Equipto angle will be suspended from the test cell crane above Configuration 15, as shown in Figure 7. While this positioning precludes the use of the PASS system, it is within the scope of SOP 5.5.8 since the presence of the box precludes extraneous material from falling onto or into the reactor.

If the support mechanism should fail, the Boral enclosure would slide over the pressure vessel with the top coming to rest on the shield adapter ring. The worth of a 3/8" thick Boral sheet placed on the tie tube cups has been measured on the PAX-E5 core as +8.67¢*. Therefore, the worth of the 1/4" thick piece on the shield adapter ring (5.3 inches further from the core) would be significantly less. The four pieces of Boral along the side of the vessel can be estimated as twice the worth of the 1/8 inch thick Boral wrapper analyzed in Section 3. This amounts to a value of approximately 8.20¢. Therefore, failure of the Boral enclosure support would result in a conservative reactivity insertion of approximately 17¢. Since this is less than 1.00\$, no position interlocks are required for this experiment. The Boral enclosure support has a safety factor in excess of 8.

G. Conclusions

The performance of the experiment presents no unevaluated hazard when performed as outlined in this EDC according to the provision of SOP 5.5.8 (PAX Experiments With a Nozzle or Plate Over the End of the Reactor) as well as those of SOP's 5.2 (Reactor Operating Procedure), 5.4 (Manual Reactivity Change Procedure), and 5.5.4 and 5.5.5 (PAX In-Core and PAX Out-of-Reactor Experiments).

* WANL-TME-1785.

EXPERIMENTAL DETAILS:

A. Configuration 15 - Bare Core

Bare Core Shimming

After the full 360° Boral shield has been placed around the PAX-G0 pressure vessel, the core should be shimmed to a delayed critical drum bank angle of $115^\circ \pm 1^\circ$. The shimming is to be performed by placing TaC elements in the core locations 2E4A and 6E4A. This TaC element change will be worth -7.5% *. The 360° Boral wrapper will be worth $+4.1\%$ resulting in a net change of -3.4% . The drum bank is worth $\sim 5\%/degree$ (near 115°). Since the delayed critical drum bank is 113.6° , a net change of -7.5% is desired. Therefore, four poison wires (worth $\sim 1\%$ each) will be dispersed uniformly about the center along with the TaC elements. The addition of the wires, TaC elements and Boral around the vessel will result in a net change of -7.5% and the change can be made directly.

Passive Dosimetry

The following passive dosimetry measurements will be performed.

- 1) Axial fission density distribution will be obtained with 1" long, 0.040" diameter U-Al wires placed axially in the central coolant channel of an element in the central cluster in 1" increments from 1/2" below core midplane to the top of the support blocks.
- 2) Radiochemistry will be performed on sections of a fuel element exposed at a central location in the reactor. Sandwiched between the slices of the fuel element used for radiochemical analysis will be U-238 and Th-232 foils.
- 3) The centerline axial distribution of subcadmium neutron flux will be obtained with bare and 20-mil thick Cd-covered Dy-Al foils in a Boral enclosure. Measurements will be made at 6" intervals from the top of the support blocks to a distance of 50 inches above the top of the support blocks on the axial centerline. At the same time, a radial traverse ($\phi = 270^\circ$) at a distance of 83.85 cm above core midplane will be made at radii of 0, 10, 15, 20 and 25 inches from the axial centerline. The Boral enclosure will be placed over the end of the reactor and suspended by the test cell crane.

Since the presence of the Boral enclosure precludes the use of the PASS system, the following operating procedure will be used to satisfy SOP 5.5.8.

- a) Operate reactor without Boral enclosure present to establish a basepoint.
- b) Install Dy passive dosimetry external to the core (negligible reactivity effect) and place Boral enclosure in position over the bare end of the core.

The maximum possible reactivity change expected is $+17\%$ (see Hazards Considerations, Section F); therefore, the change can be made directly.

*DRM-1057, "Nuclear Predictions for the PAX/R-1 Experimental Program", R. A. Doncals, Dec. 2, 1968.

c) Operate reactor to establish the reactivity worth of Boral enclosure and irradiate the dosimetry.

4) An axial centerline distribution of gamma dose rate will be obtained with Li^7F TLD chips placed at 6" intervals for the first two feet from the top of the support blocks, then in 2' intervals out to a distance of 14 feet above the top of the support blocks. Also, $\text{CaF}_2\text{:Mn}$ TLD chips will be placed in 6" intervals for the first two feet above the top of the support blocks, then in 4' intervals to a distance of 14 feet above the top of the support blocks.

A radial traverse ($\phi = 270^\circ$) at a distance of 83.85 cm above the core midplane will be made at radii of 0, 10, 15, 20 and 25 inches with Li^7F and $\text{CaF}_2\text{:Mn}$ TLD's.

5) An axial centerline distribution of the fast neutron flux density ($E > 2.9$ MeV) will be obtained with sulfur pellets. The pellets will be placed in 6" intervals for the first two feet above the top of the support blocks, then in 2' intervals out to a distance of 14 feet from the top of the support blocks.

A radial traverse ($\phi = 270^\circ$) at a distance of 83.85 cm above the core midplane will be made at radii of 0, 10, 15, 20, and 25 inches.

6) An axial centerline distribution of the fast neutron flux density ($E > 1.5$ MeV) will be obtained with 20-mil Cd-covered U-238 (190 ppm U-235) foils and bare and Cd-covered Th-232 foils. To provide counter calibration, bare U-238 and Th-232 foils will be irradiated near the center of the core for radiochemical determination of the number of fissions. The 20-mil Cd-covered Th-232 foils will be placed in 6" intervals for the first two feet above the support blocks, then in 2' increments out to a distance of 14 feet above the top of the support blocks. The Cd-covered U-238 foils and the bare Th-232 foils will be placed at the top of the support blocks and thereafter at every other location (as outlined above) with the Cd-covered Th-232 foils at the alternate locations.

A radial traverse ($\phi = 270^\circ$) at a distance of 83.85 cm above the core midplane at radii of 0, 10, 15, 20, and 25 inches midplane will be made with Cd-covered Th-232 foils.

Active Dosimetry

Traverses will be made using the (ρ, θ, ϕ) traverse system at polar angles (azimuthal angle of 270°) of 0, 10, 20 and 30 degrees at distances from the core center of 5, 5.5, 6, 6.5, 7, 9, 11, 13, 15, 17 and 19 feet for the following detectors.

- 1) Cd (20-mil thick)-covered U-238 (11 ppm U-235) fission chamber (WX-30748) in 1/4" thick Boral, to determine fast neutron flux density ($E \leq 1.5$ MeV).
- 2) Carbon wall CO₂ gas-filled ion chamber (RSG-8A) to determine gamma dose rate.
- 3) Hurst fast neutron proportional counter to determine fast neutron dose rate ($E \geq 0.2$ MeV).

As a minimum, data will be obtained with Be and Al Bragg-Gray chambers at polar angles and distances from the core center of 0° (58-1/4"), 10° (59-3/16"), 19° (61-9/16"), 27° (65-3/8"), 34° (70-1/4") and 41° (77-3/8").

Common Plane

There will not be any passive dosimetry measurements made on the common plane.

The active measurements on the common plane (13.2 feet above the support blocks) will be obtained at a minimum of five radial locations out to a radius of 8.5 feet from the axial centerline of the reactor. The (ρ, θ, ϕ) traverse system will be employed and basic trigonometric relationships used to obtain the data by varying the polar angle and distance from the core center. Measurements will be obtained with the U-238 fission chamber, carbon wall ion chamber, Hurst proportional counter, and the Be and Al Bragg-Gray chambers, respectively.

Dosimetry Power Levels

Passive dosimetry Items 2, 5, and 6 will require one high integrated power run of approximately 1000 watt-hours. Passive dosimetry Items 1 and 4 will require one power run of approximately 30 watt-hours. Passive dosimetry Item 3 will require one power run of approximately 50 watt-hours.

Active dosimetry will require 3 power runs at a power level suitable to provide adequate counting statistics. This will be determined during operations but will not exceed 750 watts.

Although the R-1 test sector will be centered at 225° ($\phi = 0$ center of drum number 1), the active and passive data will be obtained at $\phi = 270^\circ$ where background is expected to be a minimum.

DATA REQUIRED:

1. Photos of experimental setup.
2. Material composition of all mockups on the reactor.
3. Measured dosimeter positions on the core.
4. Power calibration of bare core configuration in terms of output current of WX-5362 located under dome end of reactor.
5. Delayed critical drum angle of all runs.
6. Saturated activity of foils used in dis/gm*-sec-w.
7. Thermal and fast neutron flux density per watt values with cross-section.
8. Gamma and neutron absorbed dose rates per watt.
9. Uncertainties in position and magnitudes of flux densities, foil activity dose rates.
10. Location of TaC elements and poison wires.
11. Data are to be obtained in the format of the attached sample data sheets for active and passive dosimetry.

WRITTEN BY: _____

 V. S. Oblock/J. R. Magaw

Date: _____

2/26/69

SIC: _____

 F. S. Frantz, Manager, WANEF

Date: _____

2/26/69

MANAGER, NUCLEAR & RADIATION DESIGN _____

 W. P. Kovacik

Date: _____

2/27/69

 * Grams of nuclide that is activated.

Figure 1 - Schematic of Relative Positioning of Borated CH₂ and Pb Wall Lining

613213-10B

Figure 2 -West Wall Inside FrameSupport for B-CH₂ Placement
Above Test Stand Level

Figure 3 - Photo of Method of Support
of Inside Frame to the Test Cell Wall

Figure 4 - West Wall B-CH₂ and Support Structure

Figure 5 - Inside Unistrut Frame to Which B-CH₂
Will Be Secured On The North Test Cell Wall

613212-8B

Figure 6 - Failure Configuration For Wall Liner Material

Figure 7 - Boral Enclosure in Position on the Bare Core

INFORMATION CATEGORY

Unclassified
J. C. Worden 3/5/69
Authorized Classifier Date

WANEF

EXPERIMENTAL DETAILS CHECKLIST*

No. 71

Date 3/4/69

TITLE OF EXPERIMENT: Nuclear Subsystem Shielding Provision Study Experiment - Background Measurements

OBJECTIVES:

- 1) To determine the influence of the test cell environment on the radiation level above the PAX-G0 reactor by the use of a shadow shield at the reactor, i. e., Configuration 14.

EQUIPMENT REQUIREMENTS:

- 1) The PAX-G0 reactor shimmed to $115 \pm 1^\circ$ with shield adapter ring in place and Configuration 14 on the core.
- 2) The Configuration 14 background shield mockup consisting of a top shield and a side shield, centered at $\theta = 270^\circ$ ($\theta = 0$ center of drum number 1).
 - a) The Configuration 14 top shield mockup consists of one 1-1/2 inch plate of 2219 aluminum 52 x 52 inches, four 1/4 inch Boral (35 w/o B₄C) sheets 52 x 52 inches and one 1/8 inch Boral sheet 52 x 52 inches, 254 - 2 x 4 x 8 inch lead bricks, two 1 inch thick borated (2 w/o boron) polyethylene sheets 52 x 52 inches and ten 1 inch thick polyethylene sheets 52 x 52 inches assembled as shown in Figure 1. The sides of the shield will be covered with four 52 x 26 x 1/8 inch Boral (35 w/o B₄C) sheets.
 - b) The Configuration 14 side shield mockup consists of two 1/4 inch thick Boral (35 w/o B₄C) sheets 52 x 52 inches, ten 1 inch thick polyethylene (clear or borated) sheets 48 x 48 inches, one 2 inch by 4 inch by 48 inch piece of polyethylene and one 4 inch by 8 inch by 48 inch piece of polyethylene assembled as shown in Figure 2. Also needed are two cables plus a piece of steel angle bolted to the test stand deck to hold the side shield securely in position.
- 3) A fixture for centerline passive dosimetry traverse comprised of a 14 foot length of 1/2" OD x 0.028" wall thickness aluminum tubing with an aluminum adapter at one end. The adapter will be a quadripod that will secure the base of the tubing to the top of the Configuration 14 shield. Four guy wires (heavy duty string) securely attached to the tubing at a position of approximately eight feet from the shield. These wires will be securely attached to the shield to position the aluminum tube parallel to the reactor axis.

* In partial response to the Scope of Test document 62DW-268-1 for the Nuclear Subsystem Shielding Provision Study Experiment dated January 21, 1969.

- 4) Fifty poison shipping wires.
- 5) The (ρ, θ, ϕ) traversing system, remotely controlled, on which the active dosimeters can be mounted.
- 6) Detecting systems consisting of CO_2 ionization chambers and Li^7F and CaF TLD's for gamma ray dose rates, Hurst dosimeter for fast neutron dose rate; U-238 fast fission chamber wrapped in 20 mil Cd inside 1/4" thick Boral (35 w/o B_4C) and U-238 and/or thorium foils and sulfur pellets for fast neutron flux, bare and cadmium covered dysprosium foils for thermal neutron flux, aluminum and beryllium Bragg-Gray chambers to determine the gamma ray heating rates.
- 7) Three 52" x 52" x 1/8" Boral (35 w/o B_4C) sheets rolled to fit on the external surface of the pressure vessel. A 52" x 6" x 1/8" Boral (35 w/o B_4C) piece. The top of these sheets will be placed approximately even with the top of the reflector and will be securely fastened to the pressure vessel with the use of at least two steel bands to form a 360° shield.
- 8) The overhead test cell crane capable of remote movement from the control console.
- 9) Three closed circuit T.V. cameras and one 20-foot measuring tape.
- 10) A WX-5362 B-10 compensated ion chamber and associated equipment securely fastened under the dome end of the reactor for power level indication.
- 11) Sixteen 4' x 8' x 1" and four 4' x 4' x 1" borated polyethylene (2 w/o boron) sheets, six 5' x 3' x 1/8" and two 4' x 3' x 1/8" lead sheets to be placed in position against the test cell walls.
- 12) Welded frames made from Unistrut to secure the B-CH₂ next to the concrete walls.

HAZARDS CONSIDERATIONS:A. Configuration 14 (Top Shield Only)

The Configuration 14 top shield mockup shown in Figure 1 consists of 52 x 52 inch layers of materials assembled in the following sequence: 1-1/2" of aluminum, 1/8" sheet of Boral (35 w/o B₄C), two 1" thick borated polyethylene sheets (2 w/o boron), one 1" thick polyethylene sheet, 1/4" thick sheet of Boral (35 w/o B₄C), one 2" layer of lead bricks, two 1" thick polyethylene sheets, 1/4" thick sheet of Boral (35 w/o B₄C), three 1" sheets of polyethylene, 1/4" thick Boral sheet (35 w/o B₄C), one 2" layer of lead bricks, four 1" sheets of polyethylene, 1/4" thick Boral (35 w/o B₄C) sheet, and one 2" layer of lead bricks. The sides of this configuration are enclosed by four 1/8" thick sheets of Boral (35 w/o B₄C) securely fastened to the shield mockup. These materials will be added onto the shield adapter ring incrementally so as to stay within the 50¢ limit on manual reactivity insertion.

The total weight of the top shield of the configuration is 8300 pounds. An analysis* of the failure of the adapter ring or pressure vessel was performed by the Stress Analysis Group at WANL. The smallest margin of safety was at the cylinder flange interface. The margin of safety for an 8500 pound load was determined to be 27* based upon a minimum yield strength of 35,000 psi. This analysis was performed with the pressure vessel bolted to the dome flange and test stand. Therefore, as long as the pressure vessel is bolted to the test stand, there is no hazard.

If the support should fail, the shield will produce a reactivity insertion that is <1.00\$. This is based upon the following analysis. The worth of the configuration 14 top shield is expected to be 48¢.** Failure of the support will result in a displacement of the materials from the adapter ring position (5.35 inches from top of core) to the top of the support blocks. From Figure A-6 of WANL-TNR-099, the worth of polyethylene at the top of the support block position is 2.7 (6.5/2.4) times the worth at the adapter ring position (5.35 inches from the top of the core). Thus, the maximum reactivity insertion due to failure of the shield support would be 2.7 times the worth of the polyethylene at the shield adapter ring position (this results in the worth of the polyethylene on top of the support blocks) minus the worth at the shield adapter ring position. This results in a net change of + 82¢ (2.7 x 48¢ - 48¢). Therefore, a decision on the need for position interlocks will be based upon the measured worth of the polyethylene on the shield adapter ring. If 1.7 times the measured worth of the 1-1/2 inch aluminum, 1/8 inch Boral and 3 inches of borated and clear polyethylene is less than 1\$, position interlocks will not be required.

* DRM-51135, "PAX Pressure Vessel for Shield Experiments." G. T. Downs, Feb. 25, 1969.

** See Experimental Details of this EDC.

B. Configuration 14 (Side Shield Only)

The CH₂ sheets sandwiched between Boral (see Figure 2) can be placed in position directly. From WANL-TNR-099, the worth of CH₂ placed completely around the full length of PAX reaches a maximum value of approximately 2.75\$. If the 10" thick slabs were placed directly against PAX, rather than slanted, they would "cover" approximately 15% of the surface area of the pressure vessel. This amounts to an effective worth of approximately +53¢** (41¢ x 1.3) and the change can be made directly. The use of the Boral and the angle of the CH₂ with respect to the pressure vessel will make the reactivity change significantly less than +53¢.

If the sheets should fall and come to rest against the pressure vessel and test stand floor, there will be a thickness of 10" of CH₂ "looking" at the core midplane. Figure 3 depicts the possible configurations due to failure of the side shield.

The worth of the 10" CH₂ slab "looking" at the core midplane can be estimated in the following manner. From WANL-TNR-099, the approximate worth of an infinite amount of CH₂ placed completely around the full length of PAX is approximately +2.75\$. The worth of a section wrapped around the midplane is estimated by the following:

$$W_{M.P.} = 2.75 \$ \frac{\int_a^b \sin^2 A x dx}{\int_0^L \sin^2 A x dx}$$

where: L = length of core

A = π/L

a = station of lower CH₂ level

b = station of upper CH₂ level

The calculated value for a 10" piece of CH₂ wrapped completely around PAX is approximately +74.1¢** (57¢ x 1.3).

Neglecting the difference in curvature of the pressure vessel and the sheets, the sheets "see" approximately 30% ($48/\pi D$) of the pressure vessel circumference. This makes the CH₂ worth approximately 22.2¢** (0.30×74.1) reduced by the worth of the Boral plus CH₂ in the angled position.

* EDC-63, Addendum 1, April 16, 1968, Appendix II

** This value includes a 30% increase in reactivity worth as a safety margin for possible differences between the NRX PAX and PAX/R-1 reactors.

The 4" x 12" x 48" CH₂ placed in the following position, CH₂ against the pressure vessel with the 48" direction of the CH₂ aligned with the core axis and the 4" thick (assumed to be infinite thickness) direction aligned parallel with a plane bisecting the core midplane, encompasses an effective area of approximately 3.5% of the PAX pressure vessel surface. This amounts to a change of +12.5¢** (0.035 x 2.75\$ x 1.3).

The 4" x 12" x 48" CH₂ placed with the 12" direction of the CH₂ aligned with the core axis and the 48" aligned tangent to, and horizontal to the pressure vessel surface encompasses an effective area of approximately 5.5% of the surface. This amounts to a change of + 19.6¢** (0.055 x 2.75\$ x 1.3).

Therefore, the maximum net change for the 10" of CH₂ plus the 4" x 12" x 48" piece of CH₂ would be approximately + 41.8¢** (22.2¢ + 19.6¢). No position interlocks are required since the net change due to failure of the experiment would be <1.00\$. However, as an added precaution, the side shield will be secured to the test stand deck to which steel cables pass over the side and top shield to the other side of the reactor and secured to the floor.

C. Weight on Test Cell Floor

The test cell floor has been load tested and the results reported in EDC No. 63, Addendum I, Appendix III. Table 1 presents the weight contributions to the test cell floor in the vicinity of the test stand support. The weight (4440 pounds) of all the materials and Unistrut support on the north and west test cell walls is included in case of failure of the supports on the north and west walls. This would put the weight on the test cell floor in the vicinity of the stand. It is seen from Table 1 that if the wall liner supports failed and 10 people were in the test stand area, the load tested weight on the floor would be exceeded by approximately 200 pounds*. Since this is very close to the load tested value, all storage cabinets and miscellaneous hardware will be cleared from the PAX test cell floor quadrant and the personnel limited to nine in that area (on the PAX test stand and test cell floor) when the Configuration 14 is on the core. Signs will be posted in clear view to limit the personnel to nine.

D. Conclusions

Due to the hazards evaluation of Configuration 14, the restrictions imposed will be the following: 1) the pressure vessel must be bolted to the dome flange and test stand, 2) no more than nine personnel may be present in the PAX sector of the test cell at any one time, and 3) all extraneous hardware will be removed from the PAX sector of the test cell.

* Assuming maximum uncertainty in PAX-G0 weight.

** This value includes a 30% increase in reactivity worth as a safety margin for possible differences between the NRX PAX and PAX/R-1 reactors.

The presence of the shield materials on top of PAX preclude the use of the PASS system. The performance of the experiments presents no unevaluated hazard when performed as outlined in this EDC according to the provisions of SOP 5.5.8 (PAX Experiments With a Nozzle or Plate Over the End of the Reactor) as well as those of SOP's 5.2 (Reactor Operating Procedure), 5.4 (Manual Reactivity Change Procedure), and 5.5.5 (PAX Out-of-Reactor Experiments).

EXPERIMENTAL DETAILS:

The material will be placed on the shield adapter ring in the following manner:

a) Poison wires (50) will be added to the core prior to the placing of any material on the shield adapter ring. They will be placed uniformly about the core center--one per cluster with none in the center cluster.

b) The 1-1/2" thick aluminum plate and the 1/8" thick sheet of Boral will be placed on the shield adapter ring as shown in Figure 1. The change due to the addition of the 1-1/2" aluminum plate (neglecting the 1/8" thick Boral sheet) can be estimated from Figure A-3 of WANL-TNR-099. The net worth for this material is estimated to be +39¢. Thus, the net change produced will be approximately -11¢ with both the poison wires and the 1-1/2" aluminum plate present. This assumes that each poison wire is worth approximately 1¢.

c) The reactivity worth of the above materials will be measured along with the delayed critical drum bank angle.

d) The two 1" thick borated polyethylene sheets will be placed on top of the previous materials. The worth of this material is expected to be 7¢. This is based on the following analysis of WANEF run log data. The worth of the Boral cluster plate, 1-1/2" Al and 1" of borated polyethylene is found to be 30¢ from PAX-E5-107. This was obtained by measuring the difference between Configuration 7D and the Boral cluster plate, 1-1/2" aluminum and the 1" borated polyethylene. Previously, the worth of the 1-1/2" aluminum and Boral cluster plate had been determined to be 24¢. The net worth of the 1" of borated polyethylene was determined to be 6¢. Also from the PAX-E5 run log, the further addition of 1" of borated polyethylene (net 2" of borated polyethylene) induced a net change of <1¢. Thus, the total worth of two inches of borated polyethylene on top of the 1-1/2" Al plate has been determined to be <7¢. Therefore, because the Configuration 14 arrangement of 1-1/2" Al and 1/8" Boral is expected to behave similarly to the Boral cluster plate and 1-1/2" Al of the R-1 shield mockup series, the worth of two 1" thick borated polyethylene sheets is expected to be 7¢.

e) The reactivity worth of the above materials will be measured along with the delayed critical drum bank angle.

f) Next, the 1" thick polyethylene sheet will be placed on the previous materials. The worth of this material is expected to be less than 2¢ and is determined from the analysis in Section d above. This change will be significantly less than 50¢, thus the change can be made directly. Since the 3 inches of polyethylene can be considered to be an infinite thickness, the core should be reshimmed at this time with poison wires to a delayed critical drum angle of $115^{\circ} + 1^{\circ}$, if necessary. The shims should be placed uniformly about the core center, per cluster with none in the center cluster.

g) The reactivity worth of the above material is to be measured along with the delayed critical drum bank angle. If 1.7 times the worth of this material is $>1\%$, position interlocks are to be installed*. If not, position interlocks are not required.

h) Since the three inches of polyethylene is the most effective reflector**, the reactivity change due to the addition of the remaining materials will be negligible ($<5\%$) and all of the remaining materials can be added directly. No shimming of the core is required.

i) Measure the reactivity worth of configuration 14 top shield.

j) Add the configuration 14 side shield. Previous measurements on PAX-E5 have shown that the worth of the side shield is less than 1% and the change can be made directly. No core shimming is required.

k) Measure the reactivity worth of the configuration 14 side shield.

Passive Dosimetry

a) An axial centerline distribution of gamma dose rate will be obtained with Li^7F TLD chips placed at 6" intervals for the first two feet above the top surface of configuration 14, then in 2' intervals out to a distance of 14 feet above the top of the shield surface. Also, $\text{CaF}_2\text{:Mn}$ TLD chips will be placed in 6" intervals for the first two feet above the top of the shield surface then in 4' intervals to a distance of 14 feet above the top of the shield surface.

b) An axial centerline distribution of the fast neutron flux density ($E > 2.9 \text{ MeV}$) will be obtained with sulfur pellets. The pellets will be placed in 6" intervals for the first two feet above the top surface of configuration 14, then in 2' intervals out to a distance of 14 feet from the top of the shield surface.

c) The U-238 and Th-232 will not be irradiated due to lack of sufficient neutron intensity.

d) No passive dosimetry will be performed on the common plane.

Active Dosimetry

Traverses will be made using the (ρ, θ, ϕ) traverse system at polar angles (azimuthal angle of 270°) of 0, 10, 20 and 30 degrees at distances from the core center of 5, 5.5, 6, 6.5, 7, 9, 11, 13, 15, 17 and 19 feet for the following detectors:

* See Hazards Considerations - Section A.

** WANL-TNR-099, Revision B, pg. 8-13.

- 1) Cd (20-mil thick)-covered U-238 (11 ppm U-235) fission chamber (WX-30748) in 1/4" thick Boral, to determine fast neutron flux density ($E < 1.5$ MeV).
- 2) Carbon wall CO₂ gas-filled ion chamber (RSG-8A) to determine gamma dose rate.
- 3) Hurst fast neutron proportional counter to determine fast neutron dose rate ($E > 0.2$ MeV).

As a minimum, data will be obtained with Be and Al Bragg-Gray chambers at polar angles and distances from the core center of 0° (58-1/4"), 10° (59-3/16"), 19° (61-9/16"), 27° (65-3/8"), 34° (70-1/4") and 41° (77-3/8").

The active measurements on the common plane (13.2 feet above the support blocks) will be obtained at a minimum of five radial locations out to a radius of 8.5 feet from the axial centerline of the reactor. The (ρ, θ, ϕ) traverse system will be employed and basic trigonometric relationships used to obtain the data by varying the polar angle and distance from the core center. Measurements will be obtained with the U-238 fission chamber, carbon wall ion chamber, Hurst proportional counter, and the Be and Al Bragg-Gray chambers, respectively.

Dosimetry Power Levels

There will be one reactor run of 1000 watt-hours duration for irradiation of the passive dosimetry.

The active dosimetry will require 3 power runs at a power level suitable to provide adequate signal levels. This will be determined during operations but not to exceed 750 watts.

DATA REQUIRED:

1. Photos of experimental setup.
2. Material composition of all mockups on the reactor.
3. Measured dosimeter positions on the core.
4. Power calibration of bare core configuration in terms of output current of WX-5362 located under dome end of reactor.
5. Delayed critical drum angle of all runs.
6. Saturated activity of foils used in dis/gm*-sec-w.
7. Thermal and fast neutron flux density per watt values with cross-section.
8. Gamma and neutron absorbed dose rates per watt.
9. Uncertainties in position and magnitudes of flux densities, foil activity dose rates.
10. Location of TaC elements and poison wires.
11. Data are to be obtained in the format of the attached sample data sheets for active and passive dosimetry.
12. Material reactivity worths of configuration 14 as measured during procedure outlined in Experimental Details.

* Grams of nuclide that is activated.

WRITTEN BY: *V. S. Oblock / J. R. Magaw* DATE *3/5/69*
 SCIENTIST-IN-CHARGE: *F. S. Frantz* DATE *3/7/69*
 MANAGER, NUCLEAR & RADIATION DESIGN *W. P. Kovacic* DATE *3/10/69*

TABLE I
WEIGHT ANALYSIS ON TEST CELL FLOOR
IN VICINITY OF PAX TEST STAND

PAX Test Stand*	10,800
PAX Core Assembly*	12,000
Drive Units*	2,760
Configuration 14 (Top Shield)	8,300
Configuration 14 (Side Shield)	948
Boral Sheets on Pressure Vessel	96
Traverse System Plus Detector Systems	110
Materials on Test Cell Walls & Unistrut Supports	4,440
Personnel (10 Men at 200 pounds each)	<u>2,000</u>
TOTAL	41,454
Floor Load Tested To**	41,340

* Private Communication, J. R. Magaw and H. D. Kulikowski, PAX-G core and PAX-E weight equal to ± 100 pounds.

** EDC No. 63, Addendum 1, April 16, 1968, Appendix III.

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Figure 3 - Failure Configurations For Configuration 14
(Side Shield)

INFORMATION CATEGORY

Unclassified
F. Frantz 3/14/69
Authorized Classifier Date

WANEF
EXPERIMENTAL DETAILS CHECKLIST*No. 72Date 3/14/69

TITLE OF EXPERIMENT: Nuclear Subsystem Shielding Provision Study Experiment
(Borated Aluminum-Titanium Hydride** Shield Experiments)

OBJECTIVES:

To determine the direct external and internal radiation environment with mockups of 8, 12 and 15 inches of BATH placed above the end of the PAX-G0 reactor.

EQUIPMENT REQUIREMENTS:

- 1) The PAX-G0 reactor shimmed to delayed critical drum bank angle of $115^{\circ} \pm 1^{\circ}$.
- 2) A fixture for centerline passive dosimetry traverse comprised of a 4-foot length of 1/2" O. D. x 0.028" wall thickness aluminum tubing with an aluminum adapter at one end. The adapter will be a quadripod that will secure the base of the tubing to the top of the 8, 12 and 15 inch BATH shields. This fixture will be used to perform Dy-Al activation inside the Boral enclosure.
- 3) Thirty-six pieces of 3/8" x 3/4" aluminum bar cut to four foot lengths to be used as spacers for insertion of passive dosimeters within the BATH-shield mockups.
- 4) Material for the mockup of 8 inches of BATH, i. e., configuration 16A with an areal density of 53.24 gm/cm^2 of BATH, consisting of the following:
 - a) One 1-1/2 inch thick plate of 2219 aluminum 52" x 52" in cross section
 - b) Twenty 1/8 inch thick sheets of titanium 52" x 26" in cross section
 - c) Fifty 1/16 inch thick sheets of 2219 aluminum 52" x 52" in cross section

* In partial response to the Scope of Test Document 62DW-268-1 for the Nuclear Subsystem Shielding Provision Study Experiments dated January 21, 1969.

** Referred to as BATH.

d) Six 1/8-inch thick Boral (35 w/o B₄C) sheets 52" x 52" in cross section

e) Five 1/4-inch thick Boral (35 w/o B₄C) sheets 52" x 52" in cross section

f) Five 1/4-inch thick polyethylene 52" x 52" in cross section or a combination of 48" x 48" and 4" x 52" and 4" x 48" sheets to give sheets equivalent to 52" x 52" in cross section

g) Four 1/8-inch thick Boral (35 w/o B₄C) sheets 52" x 24" in cross section to cover the sides of the mockup.

The material will be arranged in the sequence shown in Figure 1.

5) Material for the mockup of 12 inches of BATH, i.e., configuration 16B with an areal density of 74.53 gm/cm² of BATH, consisting of the following:

a) One 1-1/2-inch thick plate of 2219 aluminum 52" x 52" in cross section

b) Twenty-eight 1/8-inch thick sheets of titanium 52" x 26" in cross section

c) Seventy 1/16-inch thick sheets of 2219 aluminum 52" x 52" in cross section

d) Eight 1/8-inch thick Boral (35 w/o B₄C) sheets 52" x 52" in cross section

e) Seven 1/4-inch thick Boral (35 w/o B₄C) sheets 52" x 52" in cross section

f) Seven 1/4-inch thick polyethylene 52" x 52" in cross section or a combination of 48" x 48" and 4" x 52" and 4" x 48" sheets to give an equivalent of 52" x 52" cross section

g) Four 1/8-inch thick Boral (35 w/o B₄C) sheets 52" x 24" in cross section to cover the exposed sides of the mockup.

The material will be arranged in the sequence shown in Figure 1.

6) Material for the mockup of 15 inches of BATH, i.e., configuration 16C with an areal density of 95.82 gm/cm^2 of BATH, consisting of the following:

- a) One 1-1/2-inch thick plate of 2219 aluminum 52" x 52" in cross section
- b) Thirty-two 1/8-inch thick sheets of titanium 52" x 26" in cross section and eight pieces of 26" x 26" x 1/8"
- c) Ninety 1/16-inch thick sheets of 2219 aluminum 52" x 52" in cross section
- d) Ten 1/8-inch thick Boral (35 w/o B_4C) sheets 52" x 52" in cross section
- e) Nine 1/4-inch thick Boral (35 w/o B_4C) sheets 52" x 52" in cross section
- f) Nine 1/4-inch thick polyethylene sheets 52" x 52" in cross section or a combination of 48" x 48" and 4" x 52" and 4" x 48" sheets to give a 52" x 52" cross section
- g) Four 1/8-inch thick Boral (35 w/o B_4C) sheets 52" x 24" in cross section to cover the exposed sides of the mockup.

The material will be arranged in the sequence shown in Figure 1.

- 7) The following items already described in EDC's 69 and/or 71:
 - a) Fifty poison shipping wires
 - b) The (ρ, θ, ϕ) traverse system
 - c) CO_2 ion chamber, U-238 fission chamber, Hurst dosimeter, Be and Al Bragg-Gray chambers
 - d) Boral enclosure (52 inch³)
 - e) Borated polyethylene and Pb on test cell walls
 - f) WX-5362 under the dome for power indication
 - g) Detector position indicators
 - h) Test cell crane (remotely controlled)
 - i) Boral on the pressure vessel surface
 - j) CaF_2 and Li^7F TLD's, Dy-Al, U-238 and Th-232 foils and sulfur pellets and U-Al wires.

HAZARDS CONSIDERATIONS:A. Active And Passive Dosimetry

Any hazards associated with the use of the traverse system external to the core and U-Al wires in the core have been analyzed in EDC 69.

B. Weight on Pressure Vessel and Test Cell Floor

The total weight on the shield adapter ring with configuration 16C (15 inch BATH mockup that gives a maximum weight) is 4200 pounds. This is half of the weight of the configuration 14 top shield*. Therefore, the safety factor on the pressure vessel and adapter ring is greater than 27**.

However, if the adapter ring or pressure vessel should fail, it is assumed that the materials will come to rest on top of the PAX support blocks. From the same failure analysis of configuration 14, it was determined that position interlocks on the shield would be required if the reactivity worth of configuration 14 times 1.7 was $> 1\$$. The factor 1.7 was determined from Figure A-6 of WANL-TNR-099 where the material was polyethylene. Therefore, applying this factor to configurations 16A, 16B and 16C will be conservative. Thus, if the reactivity worth of configurations 16A, 16B or 16C times 1.7 is greater than 1\$, position interlocks are required for that particular configuration.

Since the weight on the PAX test stand is reduced by 5100 pounds over that with configuration 14, the evaluation of the Hazards Considerations presented in EDC No. 71, Section C, Page 5, will apply. However, there will be no unusual limitations on the number of personnel on the PAX sector of the test cell floor and test stand.

C. Conclusions

Due to the hazards evaluation of configurations 16A, 16B and 16C, the restrictions imposed will be the following: 1) the pressure vessel must be bolted to the dome flange and test stand, 2) all extraneous hardware will be removed from the PAX sector of the test cell.

* See Section A of Hazards Considerations of EDC No. 71, Page 3.

** DRM-51135, "PAX Pressure Vessel for Shield Experiments", G. T. Downs, February 25, 1969.

The presence of the shield materials on top of PAX preclude the use of the PASS system. The performance of the experiments presents no unevaluated hazard when performed as outlined in this EDC according to the provisions of SOP 5.5.8 (PAX Experiments With a Nozzle of Plate Over the End of the Reactor) as well as those of SOP's 5.2 (Reactor Operating Procedure), 5.4 (Manual Reactivity Change Procedure), and 5.5.4 and 5.5.5 (PAX In-Core and Out-of-Reactor Experiments).

EXPERIMENTAL DETAILS:

The experimental arrangement present on the PAX reactor prior to the addition of the BATH mockup configurations will be configuration 14 described in EDC 71. Prior to the placement of any of the BATH mockup material on the end of the PAX reactor, all of the material comprising configuration 14 will be removed except the 1-1/2 inch thick aluminum plate on the adapter ring and the 1/8 inch thick Boral on the aluminum plate. This removal will result in a net negative reactivity change and can be made directly. The reactivity change due to the material removal will be -23.4% (4.4% for the side shield and 19% for the 2 inches of borated polyethylene and 1 inch of plain polyethylene) as measured on PAX-G0 according to the procedure outlined in EDC 71.

After the material removal of configuration 14 defined above, the BATH mockup material will be placed on the 1-1/2 inch aluminum plate and 1/8 inch Boral and required measurements performed in the following manner:

- 1) Place five BATH mockup modules on top of the 1/8 inch Boral sheet in the sequence shown in Figure 1. This is the 8 inch thick BATH mockup. The net reactivity change due to the removal of the borated polyethylene of configuration 14 material and the addition of five BATH modules will result in a net negative change because: 1) there are more boron atoms and less hydrogen atoms in the BATH mockup relative to the borated polyethylene, and 2) the side shield removal contributes a negative 4.4% change.

Since the removal of material from configuration 14 and the addition of the five BATH modules will result in a net change in reactivity that is $< +50\%$, the change can be made directly.

2) The reactivity worth of the five BATH modules will be measured along with the delayed critical drum angle.

3) Add four more BATH modules to the top of the five present. Since the five BATH modules will be worth less than $+50\%$, the additional four will be worth much less than $+50\%$ and the change can be made directly. The nine modules mockup the 15 inch BATH thickness. This will be designated as configuration 16C.

4) The reactivity worth of the additional four modules will be measured along with the delayed critical drum angle. If the total reactivity worth times 1.7 is $> 1\%$, position interlocks are required*.

5) Shim the core by the addition or removal of poison wires as required to attain a delayed critical drum bank angle of $115^\circ \pm 1^\circ$.

6) Perform the following dosimetry measurements on configuration 16C.

Active Dosimetry

Traverses will be made using the (r, θ, ϕ) traverse system at polar angles (azimuthal angle of 270°) of 0° , 10° , 20° and 30° degrees at distances from the core center of 5, 5.5, 6, 6.5, 7, 9, 11, 13, 15, 17 and 19 feet for the following detectors:

- a) Cd (20 mil thick) covered U-238 (11 ppm U-235) fission chamber (WX-30748) in 1/4" thick Boral, to determine fast neutron flux density ($E > 1.5$ MeV).
- b) Carbon wall CO_2 gas-filled ion chamber (RSG-8A) to determine gamma dose rate.
- c) Hurst fast neutron proportional counter to determine fast neutron dose rate ($E > 0.2$ MeV).

As a minimum, data will be obtained with Be and Al Bragg-Gray chambers at polar angles and distances from the core center of 0° (58-1/4"), 10° (59-3/16"), 19° (61-9/16"), 27° (65-3/8"), 34° (70-1/4") and 41° (77-3/8").

* See Hazards Considerations, Section B.

The active measurements on the common plane (13.2 feet above the support blocks) will be obtained in one foot intervals out to a radius of 9 feet from the axial centerline of the reactor. The (ρ, θ, ϕ) traverse system will be employed and basic trigonometric relationships used to obtain the data by varying the polar angle and distance from the core center. Measurements will be obtained with the U-238 fission chamber, carbon wall ion chamber, Hurst proportional counter, and the Be and Al Bragg-Gray chambers, respectively.

The active dosimetry will require 3 power runs at a power level suitable to provide adequate signal levels. This will be determined during operations but not to exceed 750 watts.

Passive Dosimetry

a) Passive dosimetry will be performed with bare and 20 mil Cd-covered Dy-Al foils, Li^7F and CaF_2 TLD's, S pellets and 20 mil Cd-covered U-238 and Th-232 foils at the internal shield positions 1 thru 10 listed in Table 1.

b) The external measurements will be performed with bare and Cd-covered Dy-Al foils inside the Boral enclosure. The positions are indicated in Table 2 and Figure 1.

The passive dosimetry will require 2 power runs: one run will be 1000 watt-hrs for the irradiation of S, Th-232 and U-238 and one run of 350 watt-hrs for the Dy-Al foils and TLD's.

7) Remove two BATH mockup modules from configuration 16C. This will result in a small negative reactivity change and it can be made directly. This will result in the 12 inch BATH mockup, configuration 16B, depicted in Figure 2. If the reactivity worth of configuration 16B times 1.7 is $>1\%$, position interlocks are required*.

8) Measure the delayed critical drum angle. If required, shim the core by the removal of poison wires to give a delayed critical drum angle of $115^\circ \pm 1^\circ$.

9) Perform the following dosimetry measurements on configuration 16B.

* See Hazards Considerations, Section B.

Active Dosimetry

Repeat all of the active dosimetry measurements described in Item 6).

Passive Dosimetry

a) Passive dosimetry will be performed with bare and 20 mil Cd-covered Dy-Al foils, Li^7F and CaF_2 TLD's, S pellets and 20 mil Cd-covered U-238 and Th-232 foils at the internal shield positions 1 thru 8 listed in Table 1 and Figure 1.

b) The external measurements will be performed with bare and Cd-covered Dy-Al foils inside the Boral enclosure. The positions are indicated in Table 2.

c) An axial fission distribution internal to the reactor core will be performed with 1 inch long 0.040 inch diameter U-Al wires from core midplane to the top of the support blocks. The location of the axial distribution will be made in the same location as that performed in the bare core measurement described in EDC 69.

The passive dosimetry will require 3 power runs: one run will be 1000 watt-hrs for irradiation of S, Th-232 and U-238 and one run of 300 watt-hrs for the Dy-Al foils and TLD's plus one run of 30 watt-hrs for the U-Al wires.

10) Remove two BATH mockup modules from configuration 16B. This will result in a small negative reactivity change and it can be made directly. If the reactivity worth of configuration 16A times 1.7 is $>1\%$, position interlocks are required*.

This will result in the 8 inch BATH mockup configuration 16A depicted in Figure 1.

11) The delayed critical drum angle will be known from the measurement performed in Item 2) above. If required, shim the core by the removal of poison wires to give a delayed critical drum angle of $115^\circ \pm 1^\circ$.

12) Perform the following dosimetry measurements on configuration 16A.

Active Dosimetry

Repeat all of the active dosimetry measurements described in Item 6).

Passive Dosimetry

a) Passive dosimetry will be performed with bare and 20 mil Cd-covered Dy-Al foils, Li^7F and CaF_2 TLD's, S pellets and 20 mil Cd-covered U-238 and Th-232 foils at the internal shield positions 1 thru 6 listed in Table 1 and Figure 1.

* See Hazards Considerations, Section B.

b) The external measurements will be performed with bare and Cd-covered Dy-Al foils inside the Boral enclosure. The positions are indicated in Table 2.

The passive dosimetry will require 2 power runs: one run will be 1000 watt-hrs for irradiation of S, Th-232 and U-238 and one run of 300 watt-hrs duration for the Dy and TLD's.

DATA REQUIRED:

1. Photos of experimental setup.
2. Material composition verified by analysis of all mockups on the reactor.
3. Measured dosimeter positions on the core.
4. Power calibration of configuration 16B in terms of output current of WX-5362 located under dome end of reactor.
5. Delayed critical drum angle of all runs.
6. Saturated activity of foils used in dis/gm*-sec-w.
7. Thermal and fast neutron flux density per watt values with cross section.
8. Gamma and Neutron absorbed dose rates per watt.
9. Uncertainties in position and magnitudes of flux densities, foil activity dose rates.
10. Location of TaC elements and poison wires.
11. Data are to be obtained in the format of the sample data sheets in EDC 69 for active and passive dosimetry.
12. Material reactivity worths of configurations 16A, 16B and 16C as measured during procedure outlined in Experimental Details.

* Grams of nuclide that is activated.

WRITTEN BY

V. S. Oblock
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DATE

3/14/69

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W. P. Kovacic
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DATE

3/17/69

TABLE 1 PASSIVE DOSIMETER PLACEMENT INTERNAL TO CONFIGURATION 16C- 15" BATH MOCKUP

Radius* (inches)	Dosimeter Axial Slot Number**	Bare and Cd-Covered Dy-Al	Li ⁷ F	CaF ₂	S	Cd-Covered	
						U-238 (190 ppm U-235)	Th-232
0, 10, 15, 20, 25	1	x, x, x, x, x	x, x, x, x, x	x, -, x, -, x	x, x, x, x, x	x, x, x, x, x	
<div>A-41</div>	2					x, -, -, -, -	x, -, -, -, -
	3					x, x, x, x, x	x, -, x, -, x
	4					x, -, -, -, -	x, -, -, -, -
	5					x, -, -, -, -	x, -, -, -, -
	6					x, -, -, -, -	x, -, -, -, -
	7					x, -, -, -, -	x, -, -, -, -
	8					x, -, -, -, -	x, -, -, -, -
	9					x, -, -, -, -	x, -, -, -, -
	10					x, -, -, -, -	x, -, -, -, -
	Total No. of Dosimeters (16C)		100	50	30	50	18
" " " (16B)		80	40	24	40	16	14
" " " (16A)		60	30	18	30	14	12

* Distance from axial centerline at azimuthal angle of 270° (Drum No. 1 = zero degrees).

** See Figure 1 for detailed description of axial placement of passive dosimeters within configuration 16C.

" " " " " " " " " " " 16B.

" " " " " " " " " " " 16A.

NOTE: x denotes dosimeter placement
- denotes no dosimeter placement

TABLE 2 - PASSIVE DOSIMETER PLACEMENT EXTERNAL TO THE BATH MOCKUPS

<u>Radius (Inches)</u>	<u>Distance Above Top of Shield Surface (Feet)</u>	<u>Bare and Cd-Covered Dy-Al*</u>
0	0.5	X
0	1.0	X
0	1.5	X
0	2.0	X
0	2.5	X
0	3.0	X
0	3.5	X
0	4.0	X

TOTAL NUMBER OF DOSIMETERS 16

* In 52 inch³ Boral enclosure.

NOTE: X denotes dosimeter placement

INFORMATION CATEGORY	
Unclassified	
F. J. Frank	3/24/69
AUTHORIZED CLASSIFIER	DATE

WANEF
EXPERIMENTAL DETAILS CHECKLIST*

No. 73
Date 3/20/69

TITLE OF EXPERIMENT: Nuclear Subsystem Shielding Provision Study Experiment
(Borated Steel - Liquid Hydrogen Shield Mockup Experiments)

OBJECTIVE:

To determine the direct external and internal radiation environment with a mockup of a borated stainless steel and liquid hydrogen shield (BSS-LH₂) placed above the end of the PAX-G0 reactor.

EQUIPMENT REQUIREMENTS:

- 1) The PAX-G0 reactor shimmed to a delayed critical drum bank angle of $115 \pm 1^\circ$.
- 2) The (ρ, θ, ϕ) traverse system with the yoke assembly extended 15 inches to clear the shield on the end of PAX, i. e., the support of the I-beam on the yoke of the traverse system must be lengthened such that this I-beam is 75" from the center of the core.
- 3) Four pieces of 3/8" x 3/4" aluminum bar cut to four foot lengths to be used as shield spacers.
- 4) Material for the mockup of the borated stainless steel₂ - liquid hydrogen shield, i. e., configuration 17 that has a total areal density of 110 gm/cm² of borated steel and polyethylene, consisting of the following:
 - a) One 1-1/2 inch thick plate of 2219 aluminum 52" x 52" cross section.
 - b) One 1/8 inch thick sheet of Boral (35 w/o B₄C) with 52" x 52" cross section.
 - c) Fourteen pieces of 1/4 inch thick borated 304 stainless steel (1.34 w/o boron) with 52" x 52" cross section, or equivalent.
 - d) Four pieces of 1/8 inch thick borated 304 stainless steel (1.34 w/o boron) with 52" x 52" cross section, or equivalent.

* In partial response to T/S 62DW-268-2 for the "Nuclear Subsystem Shielding Provision Study Experiment".

- e) Four pieces of 1/4 inch thick ordinary 304 stainless steel with 52" x 52" cross section, or equivalent.
 - f) Sheets of 0.14 gm/cc density polyethylene foam consisting of the following dimensions:
 - i) Forty-two pieces 1-3/4" x 13" x 52".
 - ii) Sixteen pieces 1-3/4" x 13" x 23-1/2".
 - iii) Four pieces 1-3/4" x 13" x 52" with a 5" x 1" slot centered on the 52" side as shown in Figure 1.
 - iv) Ten pieces 1-3/4" x 13" x 52" cut back 2" for a length of 27" on the 52" side as shown in Figure 2.
 - v) Five pieces 1-5/8" x 4" x 27" with dosimeter slots as shown in Figure 3.
 - vi) Three pieces 1-5/8" x 5" x 27" with dosimeter slots as shown in Figure 4.
 - vii) One piece 1-5/8" x 5" x 27" with dosimeter slots as shown in Figure 5.
- The material in Item 4) will be arranged in the sequence shown in Figure 6.
- 5) The following items already described in EDC's 69 and/or 71:
 - a) Fifty poison shipping wires.
 - b) CO₂ ion chamber, U-238 fission chamber, Hurst dosimeter, Be and Al Bragg-Gray chambers.
 - c) Boral enclosure (52 inch³).
 - d) Borated polyethylene and Pb on test cell walls.
 - e) WX-5362 under the dome for power indication.
 - f) Detector position indicators.
 - g) Test cell crane (remotely controlled)
 - h) Boral on the pressure vessel surface.

i) CaF_2 and Li^7F TLD's, Dy-Al, U-238 and Th-232 foils, sulfur pellets, U-Al wires, and 20 mil Cd-covers.

j) Four 1/8 inch thick Boral sheets (or equivalent) to cover exposed sides of the shield mockup.

HAZARDS CONSIDERATIONS:

A. Active and Passive Dosimetry

Any hazards associated with the use of the traverse system external to the core and the U-Al wires internal to the core have been analyzed in EDC 69.

B. Weight on Pressure Vessel and Test Cell Floor

The total weight on the shield adapter ring with configuration 17 is 4700 pounds. This is 56.5% of the weight of the configuration 14 top shield*. Therefore, the safety factor on the pressure vessel and adapter ring is greater than 27**.

However, if the adapter ring or pressure vessel should fail, it is assumed that the materials will come to rest on top of the PAX support blocks. From the same failure analysis of configuration 14, it was determined that position interlocks on the shield would be required if the reactivity worth of configuration 14 times 1.7 was $> 1\$$. The factor 1.7 was determined from Figure A-6 of WANL-TNR-099 where the material was polyethylene. Therefore, applying this factor to configuration 17 will be conservative. Thus, if the reactivity worth of configuration 17 times 1.7 is greater than 1\$, position interlocks are required for that particular configuration.

Since the weight on the PAX test stand is reduced by 4500 pounds over that with configuration 14, the evaluation of the Hazards Considerations presented in EDC 71, Section C, Page 5, will apply. However, there will be no unusual limitations on the number of personnel on the PAX sector of the test cell floor and test stand.

C. Conclusions

Due to the hazards evaluation of configuration 17, the restrictions imposed will be the following: 1) the pressure vessel must be bolted to the dome flange and test stand, 2) all extraneous hardware will be removed from the PAX sector of the test cell, and 3) if the reactivity worth of configuration 17 times 1.7 is greater than 1\$, position interlocks are required on the top surface of the mockup.

* See Section A of Hazards Considerations of EDC 71, Page 3.

** DRM-51135, "PAX Pressure Vessel for Shield Experiments", G. T. Downs, February 25, 1969.

The presence of the shield materials on top of PAX preclude the use of the PASS system. The performance of the experiments presents no unevaluated hazard when performed as outlined in this EDC according to the provisions of SOP 5.5.8 (PAX Experiments With a Nozzle or Plate Over the End of the Reactor) as well as those of SOP's 5.2 (Reactor Operating Procedure, 5.4 (Manual Reactivity Change Procedure), and 5.5.4 and 5.5.5 (PAX In-Core and Out-of-Reactor Experiments).

EXPERIMENTAL DETAILS:

The experimental arrangement present on the PAX reactor prior to the addition of the BSS-LH₂ mockup will be configuration 16A (8" BATH mockup) described in EDC 72. Prior to the placement of any of the materials for the BSS-LH₂ mockup on the end of the PAX reactor, all of the material comprising configuration 16A will be removed except the 1-1/2 inch thick aluminum plate on the adapter ring and the 1/8 inch thick Boral on the aluminum plate. This removal will result in a net negative reactivity change of -26¢ as measured on the PAX-G0 reactor according to the procedure outlined in EDC 72 and the change can be made directly.

After the removal of configuration 16A as defined above, the BSS-LH₂ mockup will be placed on the 1-1/2 inch aluminum plate and 1/8 inch Boral in the following manner:

- 1) Before placing any of the BSS-LH₂ mockup on the Boral, operate the reactor and establish a basepoint.
- 2) Place four 3/8" x 3/4" aluminum spacers on the 1/8" Boral. Two spacers will be placed parallel to and spaced one inch from the azimuthal angle of 270°. The other two spacers will also be parallel to $\theta = 270^\circ$ and will be placed approximately 3 inches from either side of the material.
- 3) Place a 1/2 inch thickness of borated stainless steel (BSS) on top of the spacers and a 1-3/4 inch thickness of 0.14 gm/cc polyethylene (equivalent to 0.268 inches of 0.915 gm/cc polyethylene) on top of the BSS. This should result in net increase in reactivity of less than 45¢. This is based upon previous measurements made on PAX according to EDC's 71 and 72. It was determined that 7-1/2 inches of BATH mockup placed on the 1-1/2 inch aluminum plate and 1/8 inch thick Boral was worth +26¢. It was also determined that two inches of borated and one inch of unborated polyethylene in the same position was worth +19¢. It can be safely assumed that the 1/2 inch of BSS will be worth less than the 8 inch BATH mockup. It can also be safely assumed that the 1-3/4 inch low density polyethylene will not be worth more than the two inches of borated polyethylene. This will be so since there is approximately the same number of boron atoms between the polyethylene and the core (0.18 gm/cm² homogeneous mixture of boron for the three inches of polyethylene versus 0.13 gm/cm² of boron between the 1-3/4 inch thick low density of polyethylene and the core); also, there is less than one-tenth the areal density of polyethylene present than for the three inch thick case of configuration 14 (0.623 gm/cm² of polyethylene for the 1-3/4 inch low density polyethylene versus 6.96 gm/cm² of polyethylene for the three inch thick case).

Since the change in reactivity is expected to be less than $+50\%$, i. e., $+45\%$ ($26\% + 19\%$), the 1/2 inch BSS and 1-3/4 inch thick low density polyethylene can be added directly.

- 4) Operate the reactor to obtain the reactivity change.
- 5) Add an additional 1-3/4 inch thick low density sheet. This should be worth less than $+19\%$ based upon the arguments presented in Item 3) above.
- 6) Operate the reactor to obtain the reactivity change.
- 7) Add the remaining material in the sequence shown in Figure 6, using the measured reactivity worth of the 1/2 inch BSS plus 3.5 inch low density polyethylene as an estimate of the worth of an additional like amount of material. This will assure that a $+50\%$ change in reactivity will not be exceeded nor the 3.40\$ shutdown requirement be violated.
- 8) Operate, as necessary, to obtain the total reactivity worth of the BSS-LH₂ mockup configuration and add position interlocks to the top of the configuration if the worth times 1.7 is greater than 1\$.
- 9) Shim the core by the addition or removal of poison shipping wires to obtain a delayed critical drum bank angle of $115^\circ \pm 1^\circ$.
- 10) Perform the following dosimetry measurements on configuration 17.

Active Dosimetry

Traverses will be made using the (ρ, θ, ϕ) traverse system at polar angles (azimuthal angle of 270°) of 0, 10, 20 and 30 degrees at distances from the core center of 5, 5.5, 6, 6.5, 7, 9, 11, 13, 15, 17 and 19 feet for the following detectors:

- a) Cd (20 mil thick) covered U-238 (11 ppm U-235) fission chamber (WX-30748) in 1/4" thick Boral, to determine fast neutron flux density ($E > 1.5$ MeV).
- b) Carbon wall CO₂ gas-filled ion chamber (RSG-8A) to determine gamma dose rate.
- c) Hurst fast neutron proportional counter to determine fast neutron dose rate ($E > 0.2$ MeV).

The active measurements on the common plane (13.2 feet above the support blocks) will be obtained in one foot intervals out to a radius of 9 feet from the axial centerline of the reactor. The (ρ, θ, ϕ) traverse system will be employed and basic trigonometric relationships used to obtain the data by varying the polar angle and distance from the core center. Measurements will be obtained with the U-238 fission chamber, carbon wall ion chamber, Hurst proportional counter, and the Be and Al Bragg-Gray chambers, respectively.

As a minimum, data will be obtained with Be and Al Bragg-Gray chambers at an elevation 4 feet above the support blocks at polar angles of 0, 5, 10, 15, 20, 25, 30, 35, 40 and 45 degrees.

The active dosimetry will require 3 power runs at a power level suitable to provide adequate signal levels. The power level will be determined during operations but it will not exceed 750 watts.

Passive Dosimetry

a) Passive dosimetry will be performed with bare and 20 mil thick Cd-covered Dy-Al foils, Li^7F and CaF_2 TLD's, S pellets, and 20 mil thick Cd-covered U-238 and Th-232 foils at the locations listed in Table 1 and Figure 6.

b) The external measurements will be performed with bare and Cd-covered Dy-Al foils inside the Boral enclosure. The positions for these measurements are indicated in Table 2.

c) An axial fission distribution internal to the reactor core will be performed with 1 inch long 0.040 inch diameter U-Al wires from core midplane to the top of the support blocks. The location of the axial distribution will be made in the same location as that performed in the bare core measurement described in EDC 69.

The passive dosimetry will require 3 power runs: one run will be 1000 watt-hrs for irradiation of S, Th-232 and U-238, one run will be 350 watt-hrs for the Dy-Al foils and TLD's, and one run of 30 watt-hrs will be required for the U-Al wires.

11) If at any time there is a need to change configurations (configurations described in EDC's 69, 71, 72 or this EDC) on the reactor, the following procedure is to be used:

From Configurations 14, 16A, 16B, 16C or 17 to Bare Core

a) Remove all of the material from the shield adapter ring. This will result in a net negative reactivity change that will be known from the reactivity measurements.

b) Remove all of the poison wires except the ones that were used to shim the bare core to a delayed critical drum angle of $115^{\circ} + 1^{\circ}$. This will result in a positive reactivity change. Since the core was shimmed to a delayed critical drum angle of $115^{\circ} + 1^{\circ}$ for all configurations (14, 16A, 16B, 16C and 17), the maximum net change in reactivity due to the removal of poison wires and material will be $< +5\%$ which corresponds to a 1° change in delayed critical drum angle.

c) Perform the desired measurements according to EDC 69 for the bare core.

From the Bare Core to Configurations 14, 16A, 16B, 16C or 17

a) Add the poison wires that were present in the core for the configuration that gave a delayed critical drum angle of $115^{\circ} + 1^{\circ}$ when the configuration was present on the adapter ring. This will result in a net negative reactivity change.

b) Add the material for the configuration desired. This will result in a net positive reactivity change that is known from the reactivity measurements made. Since the core was shimmed to a delayed critical drum angle of $115^{\circ} + 1^{\circ}$ for all configurations (14, 16A, 16B, 16C and 17), the maximum net change in reactivity due to the addition of poison wires and material will be $< +5\%$ which corresponds to a 1° change in delayed critical drum angle.

c) Perform the desired measurements according to EDC 71 for configuration 14, EDC 72 for configurations 16A, 16B and 16C, or this EDC for configuration 17.

DATA REQUIRED:

- 1) Photos of experimental setup.
- 2) Material composition verified by analysis of the BSS-LH₂ mockup on the reactor.
- 3) Measured dosimeter positions.
- 4) Power calibration of configuration 17 in terms of output current of WX-5362 located under dome end of reactor.
- 5) Delayed critical drum angle of all runs.
- 6) Saturated activity of foils used in dis/gm*-sec-w.
- 7) Thermal and fast neutron flux density per watt values with cross section.
- 8) Gamma and neutron absorbed dose rates per watt.

* Grams of nuclide that is activated.

- 9) Uncertainties in position and magnitudes of flux densities, foil activity and dose rates.
- 10) Location of TaC elements and poison wires.
- 11) Data are to be obtained in the format of the sample data sheets in EDC 69 for active and passive dosimetry.
- 12) Material reactivity worth of configuration 17 as measured during procedure outlined in Experimental Details.

WRITTEN BY V. S. Oblock DATE 3/24/69
SCIENTIST-IN-CHARGE F. S. Frantz, Manager, WANEF DATE 3/24/69
MANAGER, NUCLEAR AND RADIATION DESIGN W. P. Kovacik DATE 3/24/69

TABLE 1 PASSIVE DOSIMETER PLACEMENT INTERNAL TO CONFIGURATION 17 - BSS-LH₂ MOCKUP

Radius* (inches)	Dosimeter Axial Slot Number**	Bare and Cd-Covered Dy-Al	Li ⁷ F	CaF ₂	S	Cd-Covered	
						U-238 (190 ppm U-235)	Th-232
<div>0, 10, 15, 20, 25</div> <div>A-51</div> <div></div> <div></div> <div></div> <div></div> <div></div> <div></div> <div></div> <div></div> <div></div> <div></div>	1	x,x,x,x,x	x,x,x,x,x	x,-,x,-,x	x,x,x,x,x	x,x,x,x,x	x,-,x,-,x
	2					x,-,-,-,-	x,-,-,-,-
	3						x,-,-,-,-
	4						-,-,-,-,-
	5					x,x,x,x,x	x,-,x,-,x
	6					x,-,-,-,-	x,-,-,-,-
	7					x,-,-,-,-	-,-,-,-,-
	8					x,x,x,x,x	x,-,x,-,x
	9					x,-,-,-,-	-,-,-,-,-
	10						x,-,-,-,-
	11						-,-,-,-,-
	12						x,-,-,-,-
Total No. of Dosimeters		120	60	36	60	24	14

* Distance from axial centerline at azimuthal angle of 270° (Drum No. 1 = zero degrees).

** See Figure 6 for detailed description of axial placement of passive dosimeters within configuration 17.

NOTE: x denotes dosimeter placement
- denotes no dosimeter placement

TABLE 2 - PASSIVE DOSIMETER PLACEMENT EXTERNAL TO THE BSS-LH₂ MOCKUP
CONFIGURATION 17

<u>Radius (Inches)</u>	<u>Distance Above Top of Shield Surface (Feet)</u>	<u>Bare and Cd-Covered Dy-Al*</u>
0	0.5	X
0	1.0	X
0	1.5	X
0	2.0	X
0	2.5	X
0	3.0	X
0	3.5	X
0	4.0	X

TOTAL NUMBER OF DOSIMETERS 16

* In 52 inch³ Boral enclosure.

NOTE: X denotes dosimeter placement

613212-1B

NOTE: All dimensions in inches

Material: 0.14 gm/cc polyethylene

Figure 1 - Low Density Polyethylene Sheet to Accept 5 Inch Dosimeter Slug

NOTE: ALL DIMENSIONS IN INCHES
MATERIAL: 0.14 GM/CC POLYETHYLENE NOT TO SCALE

613212-3B

Figure 2 - Low Density Polyethylene Sheet to Accept 4 Inch Dosimeter Slug

NOTE: ALL DIMENSIONS IN INCHES"

MATERIAL: 0.14 GM/CC POLYETHYLENE

NOT TO SCALE

613212-5B

Figure 3 - Dosimeter Holder - 4 Inches Wide

NOTE: ALL DIMENSIONS IN INCHES
MATERIAL: 0.14 GM/CC POLYETHYLENE

613212-4B

Figure 4 - Dosimeter Holder - 5 Inches Wide

1.625 DEPTH - IDENTICAL DOSIMETER SLOTS ON OPPOSITE SIDE

NOTE: ALL DIMENSIONS IN INCHES
MATERIAL: 0.14 GM/CC POLYETHYLENE

613212-6B

Figure 5 - Dosimeter Holder With Slots Placed on Both Faces of 5 Inch Side

APPENDIX B

UNCERTAINTY ANALYSIS FOR DATA CONTAINED IN THE TEST RESULTS SECTION

B.1 REACTIVITY MEASUREMENTS

The reactivity worth of each configuration was obtained by summing up the individual components as they were placed in sequence on top of the PAX-G0C core. The uncertainty of the worth of each shield configuration was similarly obtained by summing up the components uncertainty vectorially assuming independent contributions. Each error was assumed to be $\pm 0.39\%$ when measured by the period-pickoff technique and $\pm 1\%$ by the integral drum worth versus delayed critical drum bank technique.

B.2 FISSION DENSITY DISTRIBUTION

The components comprising the uncertainties for the fission density distribution are presented in Table B-1.

B.3 PASSIVE DOSIMETRY MEASUREMENTS

The counting data include an error analysis for each dosimeter based on counting statistics and the uncertainties of background rates, system resolving times, dosimeter weights and counting time. In addition to these quantities, the other possible sources of error have been estimated for each type of dosimeter and have been inserted into the final steps of the data reduction. Tables B-2 and B-3 list the significant sources of error and estimated magnitudes in terms of one-sigma standard deviations in percent. These uncertainties are in addition to those uncertainties due to counting statistics which are calculated separately for each data point.

The accuracy of the TLD measurements have also been estimated. Table B-4 summarizes the principal sources of uncertainty on the basis of one-sigma standard deviations in percent.

TABLE B-1
FISSION DENSITY UNCERTAINTY

Uncertainty Contribution	1 σ Uncertainty (Percent)
1. Counting Statistics. Counting rate limited by resolving time limitations	$\pm 0.2 - 0.8$
2. Resolving Time Uncertainty	$\pm 0.1 - 0.5$
3. Counting Equipment Stability	± 0.5
4. Computer Arithmetical Approximations	± 0.2
5. Relative Detector Sensitivities (3 channel system)	± 0.5
6. Self-Shielding of Wires. (based on PAX-GOB reactor measurements)	± 0.1
7. Relative content of U in U-Al wires	± 1.5
TOTAL (Root Mean Square of Listed Uncertainties)	$\pm 1.7 - 1.9$

The error bars appearing on the data points in the figures present the statistical sum of the counting statistics (except for the TLD data) and the above estimated statistical uncertainties on a one-sigma basis.

B.4 ACTIVE DOSIMETRY MEASUREMENTS

The accuracy of the active dosimetry measurements has been estimated for each of the detectors used. Table B-5 summarizes the contributing sources of uncertainty on the basis of one-sigma estimates given in percent. Since these are independent in nature, the total uncertainty is taken to be the square root of the sum of the squares of the individual uncertainties.

The uncertainties placed on the direct data, i. e., data with the background subtracted,* have been calculated from the formula presented in Table B-6 for each detector system. These formulae have been generated from the uncertainties listed previously in Table B-5.

The error bars appearing on the data points in the figures are obtained from the statistical uncertainty formula listed in Table B-6.

B.5 DOSIMETER POSITION

The accuracy of the position of the passive dosimetry measurements has been estimated for the dosimetry located external to the shield configurations and also for the dosimetry located internal to the shields. The position uncertainties for the external shield dosimetry are approximately ± 0.50 cm for both the radius and elevation. The position uncertainties for the internal shield dosimetry are approximately ± 0.50 cm for both radius and elevation. For all cases of non-centerline data, the azimuthal uncertainty is ± 1.0 degree.

The accuracy of the position of the active dosimetry measurements has been estimated for the (ρ, θ, ϕ) system to be approximately ± 1.0 cm in the ρ coordinate, ± 0.37 degree in the θ coordinate, and ± 0.5 degree in the ϕ coordinate.

*As noted in Section 2.2.1, it is assumed that the background measurements give a true measure of room background.

TABLE B-2
SOURCES OF SYSTEMATIC ERRORS IN ABSOLUTE ACTIVITY AND FLUX

Source	Dy	S	U-238
Isotopic Abundance	0.1	0.1	-
Alloy Content	7.0	-	-
Normalizer Count Rate	1.0	1.0	1.0
Saturation Exposure Calculation	0.1	1.0	-
Reactor Power	6.0	6.0	6.0
Absolute Counting Efficiency of Normalizer	6.0	10.0	10.0
Contribution of Unwanted Activities	-	1.0	-
Cross Section (For Flux Computation Only)	7.4	7.0	10.0
TOTAL	13.3	13.7	15.4

(Percent errors on a one-sigma basis.)

TABLE B-3
SOURCES OF RANDOM AND BIASING ERRORS

Source	Dy	S	U-238
Spectral Dependent Variations of Cross Section	1.0	7.0	5.0
Variations in Alloy Content	2.0	-	-
Counter Inter-Normalization	2.0	0.5	5.0
Variations in Dosimeter Weights, if uncorrected	-	0.5	-
Contributions From Unwanted Reactions	-	0.5	5.0
TOTAL	3.0	7.1	8.6

(Percent errors on a one-sigma basis.)

TABLE B-4

TLD MEASUREMENT UNCERTAINTIES

Source of Uncertainty	One-Sigma Percent
	TLD
Reproducibility	5
Uniformity	12
Power	6
Calibration	6.4**
Readout Accuracy	1 to 10
TOTAL	16 to 19 *

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

**See DRM 51176A.

TABLE B-5
ACTIVE MEASUREMENT UNCERTAINTIES

Source of Uncertainty	ONE-SIGMA UNCERTAINTIES IN PERCENT					
	DETECTOR					
	Fast Fission Chamber Report		CO ₂	Bragg-Gray Chambers		Hurst
	Fiss.	n		Al	Be	
	gm-sec-w	$\frac{2}{\text{cm}^2\text{-sec-w}}$				
Power Reading	2	2	2	2	2	2
Counting Statistics	$100/\sqrt{n}$	$100/\sqrt{n}$				
Chamber Reading			3	3	3	4 (max.)
Dark Current				5	5	
Power Calibration	5	5	5	5	5	5
Chamber Calibration	8.2	9.6	2	3.8	4.1	5.38
Conversion from R to Rads			3	3	3	
External Sigma*	$\sqrt{96.2 + 10000/n}$	$\sqrt{121.2 + 10000/n}$	7.14	$\sqrt{61.4 + (5i)^2}$	$\sqrt{63.8 + (5i)^2}$	8.60

* i = dark current/chamber reading.

n = number of counts

TABLE B-6

ACTIVE MEASUREMENT DIFFERENCE UNCERTAINTIES

Detector System	Uncertainty Formula in Percent
Fast Fission Chamber:	
Fiss/gm-sec-w	$\sqrt{92.24 + (A^2(4 + 10000/n_A) + B^2(4 + 10000/n_B)) / (A - B)^2}$
Neuts/cm ² -sec-w	$\sqrt{117.16 + (A^2(4 + 10000/n_A) + B^2(4 + 10000/n_B)) / (A - B)^2}$
Hurst Chamber	$\sqrt{54 + 20(A^2 + B^2) / (A - B)^2}$
Bragg-Gray Chambers:	
CO ₂	$\sqrt{38 + 13(A^2 + B^2) / (A - B)^2}$
Al	$\sqrt{3.8^2 + 34 + (13 + (5i)^2)(A^2 + B^2) / (A - B)^2}$
Be	$\sqrt{4.1^2 + 34 + (13 + (5i)^2)(A^2 + B^2) / (A - B)^2}$

A = Measured Data
B = Background Data

APPENDIX C

DATA REDUCTION TECHNIQUES

C.1 FISSION DENSITY DATA

Data from the NC-8010 counting system used to count the beta radiation emitted by fission product decay from the U-Al wires were reduced to units of fission density relative to 1.0 at the core centerline by the CDC-6600 computer using the WANEF passive data reduction program as reported in DRM 51105C.

Essentially, this data reduction program averages the count rate data from the NC-8010 system for each U-Al wire counted, decay corrects the results, sorts the data according to designation and then matches the data with the appropriate fuel element loading and approximate radial location, normalizes all of the data to unity at the core center location, and generates computer printed tables and plots of the data.

C.2 Dy, S AND U-238 PASSIVE FOIL DOSIMETRY DATA

The activity of the detector at any time during its irradiation can be calculated by an equation expressing the rate of change of a radioactive species. These equations have been programmed via the WANEF passive data reduction program (DRM-51105C) for the CDC-6600 computer to reduce the data to units of neutron flux density per watt or saturated activity per gram of detector per watt. Table C-1 presents the constants used in the data reduction program to obtain flux density for the various types of passive dosimeters.

TABLE C-1
CONSTANTS USED IN DATA REDUCTION

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

* Chart of Nuclides, July 1966.

** BNL 325, 2nd Edition, Supplement 2.

C.3 TLD DOSIMETRY DATA

The TLD data were reduced using the WANEF passive data reduction program found in DRM 51105C and the CDC-6600 computer. TLD calibration data used in the above program have been reported in DRM 51176A.

Although the data are reported as mrad-(C)/hr-w, the data are based upon the light output of the TLD in the EG&G reader when the dosimeter is exposed in a known Co-60 field for a specified period of time. Since the mass energy absorption coefficient of $\text{CaF}_2:\text{Mn}$ relative to carbon increases with decreasing gamma ray energy below approximately 0.5 MeV, the comparison of the results of TLD measurements with analytical predictions should include corrections for this effect.

C.4 ACTIVE DOSIMETRY DATA

The WANEF active data reduction program, RN-PARX2, found in WANL-TME-1952, Rev A, was programmed for the CDC-6600 computer to process data for the Bragg-Gray detectors and the U-238 fission chamber. The data were keypunched in the proper format and reduced with the use of appropriate conversion factors and sort programs.

The RSN-48 Hurst proportional counter data were reduced by processing the data on the CDC-6600 computer using the HDRC 2 program (DRM 51198, Rev A) and the WANG-360K programmable desk calculator for final computations.

APPENDIX D

ACTIVE DOSIMETRY SYSTEM & DETECTOR DESCRIPTIONS

Four different types of active detectors were employed to measure fast neutron dose, thermal neutron flux, fast neutron flux and gamma dose rate. The detectors employed for these measurements are described in the next sections. Details on calibration are given in DRM's 51157, 51158, 51173, 51177. The detectors, described in this section, were traversed remotely by the traverse systems described in Section 3.2.3.

D.1 FAST NEUTRON DOSE

The RSN-48 Hurst proportional counter made by Reuter Stokes and patterned after an Oak Ridge design was used to measure fast neutron dose ($E > 0.2$ Mev). Figure D-1 gives details of this detector, Table D-1 lists the equipment used, and Figure D-2 shows a block diagram of the instrumentation. A more detailed description of the counter operation is contained in WANL-TME-1852, Section 4.5.1.

The calibration of the Hurst proportional counter is reported in DRM-51173.

D.2 THERMAL NEUTRON POWER MONITOR

The WX-5362 (shown in Figure D-3) detector, manufactured by Westinghouse Electronic Tube Division, Elmira, is a B-10 lined compensated ion chamber originally developed* for NERVA applications and was used to monitor reactor power throughout the experiment. Figure D-4 shows the WX-5362 in position at the dome end of the PAX-G0 reactor, and Table D-1 lists the equipment used, while Figure D-5 contains a block diagram of the instrumentation.

D.3 FAST NEUTRON THRESHOLD FLUX

The WX-30748 fast fission detector, manufactured by Westinghouse Electronic Tube Division, Elmira, contains approximately 2 grams of U-238 having a U-235 content of 11 parts per million. Pertinent details of the detector are given in Figure D-6. Table D-1 lists the equipment used and Figure D-7 contains a block diagram showing the interconnections of this instrumentation.

*WANL-TME-527.

During the measurements in the external environment, the U-238 fission detector was operated with the detector wrapped in 0.0508 cm cadmium and then placed in the Boral enclosure to minimize the response to U-235 fission.

The calibration of the fast fission detector is reported in DRM 51157.

D.4 GAMMA DOSE RATE

The Reuter-Stokes RSG-8A is a carbon wall gamma sensitive ionization chamber. This particular one is filled with CO₂ gas and the design is based upon the Bragg-Gray principle. Actual calibration, therefore, is a measure of the absorbed dose in carbon. Details of the chamber are shown in Figure D-8. Figure D-9 contains a block diagram of the instrumentation, and Table D-1 lists the equipment used.

The calibration of the carbon walled CO₂ filled detector is reported in DRM 51158.

D.5 HEATING RATES

Two Bragg-Gray type radiation heating rate measuring detectors (sensitive volume of 1.3 cc ionization chambers) which obey the Bragg-Gray principle have been constructed at WANL following the successful design of C. I. McVey of General Electric Company. The detectors are shown in Figure D-10. Figure D-11 contains a block diagram of the instrumentation and Table D-1 lists the equipment used. Detectors of this type were constructed of aluminum and beryllium walls specifically for heating rate measurements in the NERVA reactor environment.

The calibration of these Bragg-Gray detectors is reported in DRM 51177.

Figure D-1. Cross Section of Standard Hurst Fast Neutron Dosimeter

Figure D-2. Block Diagram of the Hurst Data Acquisition System

Figure D-3. Cross Section of the WX-5362 Boron-Lined Compensated Ion Chamber

Figure D-4. PAX Power Monitor in Position Under Dome of the PAX-G0C Reactor

Figure D.5 Block Diagram of the WX-5362 Power Level Readout System

THE ACTIVE VOLUME IS COMPOSED OF A SERIES OF CONCENTRIC SHELLS UPON WHICH THE URANIUM IS PLATED. THE MAXIMUM O D OF THE ACTIVE VOLUME IS 1.4" WHILE THE MINIMUM I D IS 0.5".

Figure D-6. Cross Section of the U-238 Fast Fission Chamber (WX-30748)

Figure D-7. Block Diagram of the U-238 Fast Fission Chamber (WX-30748) Data Acquisition System

Figure D-8. Cross Section of the RSG-8A Carbon Wall Ion Chamber

Figure D-9. Block Diagram of the RSG-8A Data Acquisition System

Scale 2" = 1"

Figure D-10. Bragg-Gray Ion Chamber for External Measurements

Figure D-11. Block Diagram of the Bragg-Gray Data Acquisition System

TABLE D-1
ACTIVE DOSIMETRY EQUIPMENT LIST
U-238 FAST FISSION DETECTOR

<u>Equipment Types</u>	<u>Manufacturer</u>	<u>Model No.</u>	<u>Serial/Property No.</u>
U-238 Fission Chamber	Westinghouse	WX-30748	680401
Discriminator	Hamner	NC-10	MA 102282
Discriminator	Hamner	NC-10	MA 102284
Scaler	Hamner	NS-11	AEC 911375-1614
Timer	Hamner	NT-15	AEC 91137501130
H.V. Power	Burgess	U-200/300 Volt Battery	
Pre-Amplifier	ORTEC	109 pc	134
Amplifier	ORTEC	410	AEC 911375-1605

HURST FAST NEUTRON DOSE DETECTOR

Hurst Detector	Reuter Stokes	RSN-48	G-1497
Hurst Detector	Reuter Stokes	RSN-48	G-1499
Pre-Amplifier	ORTEC	109 pc	AEC MA 106878
Amplifier	ORTEC	410	AEC 911375-1605
H.V. Power	Power Designs	HV-1544	AEC 91137501257
Solenoid Power	NJE Corp.	TR-36-10	AEC MA 106869
Pre-Amp Power	ORTEC	115	283
Data Output Unit	Technical Measure- ments Corporation	220B	5830
Pulse Height Analyzer	TMC	CN-110	13679
Pulse Height Analyzer	TMC	Z10-1024	6030

TABLE D-1 (continued)

CO₂ BRAGG-GRAY DETECTOR

<u>Equipment Types</u>	<u>Manufacturer</u>	<u>Model No.</u>	<u>Serial/Property No.</u>
CO ₂ Bragg-Gray Detector	Reuter Stokes	RSG-8A	G-928
H.V. Power	Burgess	300 Volt Battery	
Picoammeter	Keithley	417	01332

POWER LEVEL DETECTOR

Thermal Neutron Detector	Westinghouse	WX-5362	108
H.V. Power	Burgess	300 Volt Battery	
Picoammeter	Keithley	410C	101571
10 Turn Potentiometer	Bechman	-	-
Digital Readout	Fairchild	7020	797

APPENDIX E

NUCLEAR MEASUREMENTS - FISSION DENSITIES AND POWER CALIBRATION MEASUREMENTS

E.1 AXIAL FISSION DISTRIBUTIONS

Measurements of the axial distribution of fission density in the PAX-G0C reactor core were obtained for the bare reactor configuration (15), as well as for shield configurations 16B and 17. The measurements were performed in the central coolant channel of fuel element Q11C, near the core centerline. The axial distribution was obtained by irradiating 2.54 cm long U-Al wires placed vertically in stainless steel wire holder in 2.54 cm increments from the core midplane up through the support blocks.

These tests were carried out in accordance with the procedures set forth in Appendix A.

All of the tests were performed with the cold critical reactor at room temperature and at the projected cold-critical drum angle of $115.0^\circ \pm 1.0^\circ$.

Data from the NC-8010 counting system, used to count the beta radiation emitted by the fission product from the irradiated U-Al wires, were reduced to units of fission density, normalized to unity, at the core midplane location by the CDC-6600 computer using the WANEF passive data reduction program as described in DRM 51105C.

The measured results, normalized to unity at the core midplane, are plotted as a function of elevation above core midplane in Figures E-1, E-2, and E-3 for shield configurations 15, 16B, and 17, respectively. Figure E-4 presents the results for the three shield configurations together. The effect of the shield configurations in reflecting neutrons back into the top of the core can be clearly seen in Figures E-5 and E-6. These figures show the ratio of the relative fission density at corresponding core elevations for shield configuration 16B ratioed to shield configuration 15 (bare core) and shield configuration 17 ratioed to shield configuration 15, respectively. It can be also noted that the difference between shield configurations 16B and 17 is negligible.

The measured results for shield configurations 15, 16B, and 17 are also presented as tabular data in Tables E-1, E-2, and E-3, respectively. The estimated standard deviation of each measurement in percent on a one-sigma basis is included in the tables. The individual components associated with the uncertainties are discussed in Section B.2 of Appendix B of this report.

E.2 POWER CALIBRATION OF PAX-G0C

An absolute power level determination was made on the PAX-G0C for shield configuration 15 (bare core). The determination of reactor power level was accomplished by means of a radiochemical analysis of the number of fissions per gram of uranium in a central fuel element. These data were combined with a previous measurement, made with U-Al wires, of the fission rate in that element relative to the average fission rate in the PAX-G0 reactor core in order to obtain a reactor power level.

Inter-run power level calibrations were accomplished with gold wires irradiated in reproducible reactor locations for selected runs. These data were combined with reactor neutronics instrumentation and the digital reactor power level monitor to determine the power level for all reactor runs where active dosimetry was employed.

E.2.1 Fission Density Ratio

Fission density distributions, reported in WANL-TME-1911, were used to calculate the ratio of fission density at the core center to the fission density averaged over the core test sector (at the core midplane). The ratio is assumed to be equal to the ratio of fission density in a central element, to the average fission density in the entire core.

E.2.2 Power Level Determination for Shield Configuration 15

To establish a calibration of reactor power level, a sectioned fuel element was inserted into the center cluster of the PAX-G0C core and irradiated. This element contained 13 quarter-inch slices of virgin fuel evenly spaced over the length of the element. These 13 slices were dissolved and analyzed for Mo-99 content using the procedure described in WANL-TME-1915. The analysis was reported in terms of total fissions in the element, and from this, the fission per gram of U-235 in the element. The reactor power during the run was then determined according to the following equation:

$$P = \frac{F_g W}{A T_e F_w}$$

where

- P = reactor power in watts
- F_g = fissions per gram of U-235 in the calibration element
- W = total number of grams of U-235 in the core
- A = ratio of fission density in the calibration element to average fission density in the core
- T_e = reactor run time in seconds
- F_w = total number of fissions per watt-sec

Using the determined reactor power level, it was possible to establish calibration factors for the reactor neutronic channel, digital reactor power level monitor and for the gold wires which were used for inter-run power level determination.

E. 2.3 Power Level Determination for Shield Configuration

The measurement of axial power shape reported in Section E. 1 shows an effect on the axial power shape due to reflection of neutrons from the shield back into the end of the core. Since this is only a local effect concentrated at one end surface of the core, it increases the total reactor power level without a corresponding increase in the power level indication on either the reactor instrumentation or the gold calibration wires used for inter-run calibration. The change in axial shape was, therefore, calculated from the data to increase the core power by the factor 1.0375 and this factor was used to correct the power calibration established for the bare core. Since there was no significant measured difference between the axial shapes for configurations 16B and 17, it is assumed that this factor is valid for all four shield configurations.

The measurements described in this section and the procedures involved were carried out in accordance with the procedures set forth in Appendix A. All measurements were performed with the cold critical reactor at room temperature and all measurements are performed at projected R-1 cold-critical drum angle of $115.0^\circ \pm 1.0^\circ$.

Data were reduced using the CDC-6600 computer and the WANEF passive data reduction program as reported in DRM-51105C.

The measurements of the power calibration for the PAX-G0C reactor resulted in the determination of the following values:

$$\text{Ratio} \quad \frac{\text{Fission Density at Core Center}}{\text{Fission Density Averaged Over Core Sector}} = 1.267$$

$$\text{Power Level (Section E. 2. 2)} = 831.1 \text{ watts}$$

Using the above information, it was possible to establish calibration factors for the reactor neutronics channels and for the gold wires which were used for inter-run power level determination. It was also possible to establish calibration factors for the digital reactor power level monitor used to determine the power level for all reactor runs where active dosimetry was employed. These values are given in Table E-4.

The statistical uncertainty of the power calibration factors is estimated at ± 6.0 percent on a one sigma basis.

E.2.4 Inter-Run Calibrations

The activation of gold wires placed on the pressure vessel was used as a standard by which the power level of the reactor was established for all passive dosimetry runs using the absolute calibration described above.

For each passive dosimetry run the power level was determined from the gold wires and a calibration factor was established for linear channel 1. These calibration factors were then averaged separately for each reactor configuration and the results for each configuration were used in order to determine the power level of the active dosimetry runs in that same configuration.

Figure E-1. Axial Power Density In O11C10 For Bare Core

Figure E-2. Axial Power Density in 011C10 For Shield 16B

Figure E-3. Axial Power Density in 011C10 For Shield 17

Figure E-4. Axial Power Density In 011C10

Figure E-5. A Ratio of the Relative Fission Density of 16B to the Relative Fission Density of 15 Versus Elevation

Figure E-6. A Ratio of the Relative Fission Density of 17
to the Relative Fission Density of 15 Versus Elevation

TABLE E-1.

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AXIAL DISTRIBUTION OF RELATIVE POWER DENSITY - BARE CORE

PAGE 1

LOCATION	RADIUS (CM.)	AZIMUTH (DEG.)	ELEVATION (CM.)	ACTIVITY	TOTAL STD.DEV.	CODE
Q11C08	0.00	0.00	0.00	9.9425E-01	.19	0
E08	0.00	0.00	0.00	9.9804E-01	.19	0
G08	0.00	0.00	0.00	1.0060E+00	.19	0
B12	0.00	0.00	0.00	9.9791E-01	.19	0
D12	0.00	0.00	0.00	1.0024E+00	.21	0
F12	0.00	0.00	0.00	9.9889E-01	.21	0
C12	0.00	0.00	0.00	1.0078E+00	.18	0
E12	0.00	0.00	0.00	9.9960E-01	.21	0
G12	0.00	0.00	0.00	9.9507E-01	.21	0
Q11C10	.75	285.00	71.12	1.0249E-01	1.53	0
Q11C10	.75	285.00	68.58	1.5450E-01	.85	0
Q11C10	.75	285.00	66.04	1.8885E-01	.67	0
Q11C10	.75	285.00	63.50	2.2199E-01	.63	0
Q11C10	.75	285.00	60.96	2.6615E-01	.45	0
Q11C10	.75	285.00	58.42	3.1434E-01	.39	0
Q11C10	.75	285.00	55.88	3.6255E-01	.33	0
Q11C10	.75	285.00	53.34	4.0896E-01	.35	0
Q11C10	.75	285.00	50.80	4.5037E-01	.32	0
Q11C10	.75	285.00	48.26	4.9488E-01	.26	0
Q11C10	.75	285.00	45.72	5.4020E-01	.29	0
Q11C10	.75	285.00	43.18	5.8328E-01	.28	0
Q11C10	.75	285.00	40.64	6.2277E-01	.23	0
Q11C10	.75	285.00	38.10	6.5914E-01	.23	0
Q11C10	.75	285.00	35.56	7.0278E-01	.25	0
Q11C10	.75	285.00	33.02	7.3807E-01	.24	0
Q11C10	.75	285.00	30.48	7.6410E-01	.21	0
Q11C10	.75	285.00	27.94	7.9741E-01	.21	0
Q11C10	.75	285.00	25.40	8.2941E-01	.23	0
Q11C10	.75	285.00	22.86	8.4474E-01	.23	0
Q11C10	.75	285.00	20.32	8.8505E-01	.20	0
Q11C10	.75	285.00	17.78	9.1212E-01	.21	0
Q11C10	.75	285.00	15.24	9.2487E-01	.20	0
Q11C10	.75	285.00	12.70	9.5071E-01	.22	0
Q11C10	.75	285.00	10.16	9.5829E-01	.21	0
Q11C10	.75	285.00	7.62	9.7099E-01	.22	0
Q11C10	.75	285.00	5.08	9.8312E-01	.22	0

E-12

TABLE E-1. (contd)

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AXIAL DISTRIBUTION OF RELATIVE POWER DENSITY - BARE CORE

PAGE 2

LOCATION	RADIUS (CM.)	AZIMUTH (DEG.)	ELEVATION (CM.)	ACTIVITY	TOTAL STD.DEV.	CODE
Q11C10	.75	285.00	2.54	9.9330E-01	.21	C
	0.00	0.00	0.00	9.9553E-01	.19	0

TABLE E-2.

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AXIAL DISTRIBUTION OF RELATIVE POWER DENSITY - 16B SHIELD

PAGE 1

LOCATION	RADIUS (CM.)	AZIMUTH (DEG.)	ELEVATION (CM.)	ACTIVITY	TOTAL STD. DEV.	CODE
CC8	C.CC	C.CC	0.00	9.9566E-01	.26	C
EO8	C.CC	C.CC	0.00	9.9960E-01	.19	C
GO8	C.CC	C.CC	0.00	1.0007E+00	.21	C
B12	C.CC	C.CC	0.00	9.9774E-01	.26	C
D12	C.CC	C.CC	0.00	9.9565E-01	.21	C
F12	C.CC	C.CC	0.00	9.9910E-01	.22	C
C12	C.CC	C.CC	0.00	1.0066E+00	.26	C
E12	C.CC	C.CC	0.00	1.0048E+00	.19	C
G12	C.CC	C.CC	0.00	1.0002E+00	.19	C
Q11C10	.75	285.00	71.12	1.8986E-01	.80	C
Q11C10	.75	285.00	68.58	2.4306E-01	.50	C
Q11C10	.75	285.00	66.04	2.7530E-01	.49	C
Q11C10	.75	285.00	63.50	2.9739E-01	.42	C
Q11C10	.75	285.00	60.96	3.3468E-01	.43	C
Q11C10	.75	285.00	58.42	3.7668E-01	.34	C
Q11C10	.75	285.00	55.88	4.1860E-01	.35	C
Q11C10	.75	285.00	53.34	4.5612E-01	.29	C
Q11C10	.75	285.00	50.80	5.0310E-01	.27	C
Q11C10	.75	285.00	48.26	5.4037E-01	.31	C
Q11C10	.75	285.00	45.72	5.7820E-01	.28	C
Q11C10	.75	285.00	43.18	6.2540E-01	.26	C
Q11C10	.75	285.00	40.64	6.6136E-01	.24	C
Q11C10	.75	285.00	38.10	6.8590E-01	.22	C
Q11C10	.75	285.00	35.56	7.2143E-01	.22	C
Q11C10	.75	285.00	33.02	7.7136E-01	.23	C
Q11C10	.75	285.00	30.48	7.9053E-01	.21	C
Q11C10	.75	285.00	27.94	8.2707E-01	.23	C
Q11C10	.75	285.00	25.40	8.5096E-01	.24	C
Q11C10	.75	285.00	22.86	8.7384E-01	.23	C
Q11C10	.75	285.00	20.32	8.9741E-01	.20	C
Q11C10	.75	285.00	17.78	9.2147E-01	.24	C
Q11C10	.75	285.00	15.24	9.3322E-01	.20	C
Q11C10	.75	285.00	12.70	9.5978E-01	.22	C
Q11C10	.75	285.00	10.16	9.7199E-01	.23	C
Q11C10	.75	285.00	7.62	9.7990E-01	.21	C
Q11C10	.75	285.00	5.08	9.8713E-01	.22	C

TABLE E-2. (contd)

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AXIAL DISTRIBUTION OF RELATIVE POWER DENSITY - 16B SHIELD

PAGE 2

LOCATION	RADIUS (CM.)	AZIMUTH (DEG.)	ELEVATION (CM.)	ACTIVITY	TOTAL STD.DEV.	CODE
Q11C10	.75	285.00	2.54	9.8745E-01	.21	C
	0.00	0.00	0.00	1.0014E+00	.15	0

TABLE E-3.

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AXIAL DISTRIBUTION OF RELATIVE POWER DENSITY - 17 SHIELD

PAGE 1

LOCATION	RADIUS (CM.)	AZIMUTH (DEG.)	ELEVATION (CM.)	ACTIVITY	TOTAL STD.DEV.	CODE
CC8	0.00	0.00	0.00	9.9571E-01	1.16	C
E08	0.00	0.00	0.00	1.0056E+00	1.16	C
G08	0.00	0.00	0.00	9.9903E-01	1.38	C
B12	0.00	0.00	0.00	1.0030E+00	1.16	C
D12	0.00	0.00	0.00	9.9590E-01	1.16	C
F12	0.00	0.00	0.00	1.0014E+00	1.18	C
C12	0.00	0.00	0.00	1.0013E+00	1.16	C
E12	0.00	0.00	0.00	9.9850E-01	1.16	C
G12	0.00	0.00	0.00	9.9955E-01	1.18	C
Q11C10	.75	285.00	71.12	1.8845E-01	1.40	C
Q11C10	.75	285.00	68.58	2.4291E-01	1.28	C
Q11C10	.75	285.00	66.04	2.7424E-01	1.27	C
Q11C10	.75	285.00	63.50	2.9094E-01	1.24	C
Q11C10	.75	285.00	60.96	3.3409E-01	1.21	C
Q11C10	.75	285.00	58.42	3.7316E-01	1.23	C
Q11C10	.75	285.00	55.88	4.1232E-01	1.19	C
Q11C10	.75	285.00	53.34	4.5041E-01	1.19	C
Q11C10	.75	285.00	50.80	4.9649E-01	1.20	C
Q11C10	.75	285.00	48.26	5.3667E-01	1.18	C
Q11C10	.75	285.00	45.72	5.8117E-01	1.17	C
Q11C10	.75	285.00	43.18	6.2108E-01	1.19	C
Q11C10	.75	285.00	40.64	6.4797E-01	1.17	C
Q11C10	.75	285.00	38.10	6.9527E-01	1.17	C
Q11C10	.75	285.00	35.56	7.3032E-01	1.19	C
Q11C10	.75	285.00	33.02	7.5709E-01	1.16	C
Q11C10	.75	285.00	30.48	7.9586E-01	1.16	C
Q11C10	.75	285.00	27.94	8.3011E-01	1.19	C
Q11C10	.75	285.00	25.40	8.5181E-01	1.16	C
Q11C10	.75	285.00	22.86	8.7359E-01	1.16	C
Q11C10	.75	285.00	20.32	9.0657E-01	1.19	C
Q11C10	.75	285.00	17.78	9.1929E-01	1.16	C
Q11C10	.75	285.00	15.24	9.4134E-01	1.16	C
Q11C10	.75	285.00	12.70	9.6230E-01	1.19	C
Q11C10	.75	285.00	10.16	9.6798E-01	1.16	C
Q11C10	.75	285.00	7.62	9.8472E-01	1.16	C
Q11C10	.75	285.00	5.08	9.9797E-01	1.18	C

TABLE E-3 (contd)

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AXIAL DISTRIBUTION OF RELATIVE POWER DENSITY - 17 SHIELD

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LOCATION	RADIUS (CM.)	AZIMUTH (DEG.)	ELEVATION (CM.)	ACTIVITY	TOTAL STD.DEV.	CODE
Q11C10	.75	285.00	2.54	9.9460E-01	1.16	C
	0.00	0.00	0.00	1.0087E+00	1.16	C

TABLE E-4
PAX-G0C POWER CALIBRATION FACTORS

PAX-G0C Run No.	Configuration	Power (watts)	Reactor Neutronics Calibration Factor (amps/watt)		Digital Reactor Power Level Calibration Factor (amps/watt)		
91	15	111.4	8.97	(-9)*	--		
94	15	831.8	9.02	(-9)	3.04	(-11)	
96	15	66.7	9.00	(-9)	3.03	(-11)	
			9.00	(-9)	3.035	(-11)	Average
115	16C	585.98	8.53	(-9)	2.95	(-11)	
118	16C	688.07	8.50	(-9)	2.90	(-11)	
			8.515	(-9)	2.925	(-11)	Average
120	16B	676.14	8.64	(-9)	--		
121	16B	675.21	8.65	(-9)	2.93	(-11)	
			8.645	(-9)	2.93	(-11)	Average
125	16A	719.20	8.13	(-9)	2.77	(-11)	
127	16A	695.13	8.42	(-9)	2.86	(-11)	
			8.275	(-9)	2.815	(-11)	Average
134	17	706.43	8.07	(-9)	2.75	(-11)	
136	17	709.65	8.24	(-9)	2.82	(-11)	
			8.155	(-9)	2.785	(-11)	Average
106, 107	14	--	--		2.864	(-11)	Average of 16A, 16B, 16C & 17

*(-9) indicates 10^{-9} , etc.

	TITLE Nuclear Subsystem Shielding Provision Study Experiments	
	DOCUMENT NO. WANL-TME-1912	DATE 090170
ABSTRACT An experiment was performed to measure the radiation environment internal and external to heterogeneous slab mockups of two proposed NERVA internal shield designs placed on the PAX-G00 reactor. Fast and thermal neutron fluxes, fast neutron dose rates, and gamma dose rates were obtained experimentally and compared with detailed analytical calculations. The WISDM system was used and included one and two-dimensional discrete ordinates calculations of neutron and photon internal radiation environment and discrete ordinates and point kernel methods for neutron and photon external environment.		
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