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**Westinghouse  
Astronuclear  
Laboratory**



WANL-PR-(LL)-034

AUGUST 1970

FINAL PROGRESS REPORT

Contract No. NAS-8-24919

Control No. DCN 1 - X - 80 - 00056

**NUCLEAR ROCKET SHIELDING METHODS , MODIFICATION ,  
UPDATING , AND INPUT DATA PREPARATION**

**VOLUME 1**

SYNOPSIS OF METHODS AND RESULTS OF ANALYSIS

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SYNOPSIS OF METHODS AND RESULTS OF ANALYSIS

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## FOREWORD

This report is Volume 1 of six volumes of the final report on "Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation". This work was performed for the George C. Marshall Space Flight Center (MSFC), Huntsville, Alabama, under Contract No. NAS-8-24919, Control No. DCN 1-X-80-00056. The technical monitor of this contract was Mr. Henry E. Stern, Deputy Manager of the Nuclear and Plasma Physics Division of the Space Sciences Laboratory, MSFC. A summary of the work performed under this contract is presented in this volume.

In summary, the six volumes of the final report are as follows:

- Volume 1: "Synopsis of Methods and Results of Analyses" - A summary of the work performed under this contract,
- Volume 2: "Compilation of Neutron and Photon Cross Section Data" - A description of the six Master Libraries of neutron and photon, cross section data,
- Volume 3: "Cross Section Generation and Data Processing Techniques" - A description of the GAMLEG-W, APPROPOS, NAGS, and SATURN codes,
- Volume 4: "One-Dimensional, Discrete Ordinates Transport Technique" - A description of the ANISN-W code,
- Volume 5: "Two-Dimensional, Discrete Ordinates Transport Techniques" - A description of DOT-IIW, DOQ, ADOQ, and MAP codes, and
- Volume 6: "Point Kernel Techniques" - A description of the KAP-VI and SCAP codes.

## ACKNOWLEDGEMENT

The writers wish to acknowledge the technical guidance provided by Mr. Henry E. Stern, Deputy Manager of the Nuclear and Plasma Physics Division of the Space Sciences Laboratory, George C. Marshall Space Flight Center, the technical monitor of this contract.

The writers also appreciate the able assistance provided by Mr. John Watts of the Nuclear and Plasma Physics Division of the Space Sciences Laboratory in aiding in the completion of portions of this contract effort.

The hand calculations, plotting, and programming assistance provided by Mrs. Melissa Kinsey are gratefully acknowledged. The consultation provided by Miss Mary Ann Capo and Dr. Gordon Gibson is also appreciated. Thanks is also extended to the engineers and scientists who have provided invaluable suggestions in developing this package of computer codes.

## ABSTRACT

The report is Volume I of six volumes of the final report on "Nuclear Rocket Shielding Methods, Modifications, Updating, and Input Data Preparation." This volume summarizes the work performed under this contract.

This volume is subdivided into the following sections:

- 1) Use of the MSFC complete code package - a description of the preliminary or parametric and the detailed design method,
- 2) Nuclear shielding methods and master library data - a description of the computer codes and master library files provided under this contract, and
- 3) Evaluation and calculations - a presentation of selected results using the code package provided under this contract.

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## 1.0 INTRODUCTION

This report is Volume 1 of six volumes of the final report on "Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation." Presented in this volume is a synopsis of methods and results of analyses.

The nuclear rocket shielding methods and master library data are an integral part of both the preliminary or parametric and the detailed design radiation analysis methods provided for the Marshall Space Flight Center under this contract and the previous contractual work (NAS-8-20414).<sup>(1-3)</sup> A simplified, schematic diagram of each method is shown in Figures 1-1 and 1-2. Both methods are fully described in this volume.

In the preliminary or parametric design method (Figure 1-1), the APPROPOS code (Volume 3) is used to prepare neutron and photon cross sections and other basic data for use in the transport and data processing codes. These cross sections are input to the ANISN-W code (Volume 4). The ANISN-W code computes one-dimensional neutron and photon fluxes in the reactor geometry. From the neutron and photon fluxes, neutron and photon energy sources and distributions or heat generation rates are obtained using the NAGS data processing code (Volume 3). These sources and distributions are used as input to the KAP-VI point kernel code (Volume 6). The KAP-VI code provides gamma ray and fast neutron radiation levels at locations external to the reactor. Radiation sources, heat generation rates, and radiation environment, both internal and external to the reactor as well as shield effectiveness can be computed using the preliminary or parametric design method.

In the detailed design method (Figure 1-2), the neutron and photon cross sections prepared by the APPROPOS code (Volume 3) are used as input data to the DOT-IIW, two-dimensional, discrete ordinates transport code. The DOT-IIW code (Volume 5) computes the two-dimensional neutron and photon fluxes throughout the reactor geometry. The NAGS data processing code (Volume 3) processes these fluxes and calculates neutron and photon energy deposition and neutron and photon energy sources and distributions within the reactor system. These sources and distributions are used as input to the KAP-VI point kernel code (Volume 6). The KAP-VI code provides gamma ray and fast neutron radiation levels at locations external to the reactor. In addition, the surface leakage fluxes from the DOT-IIW

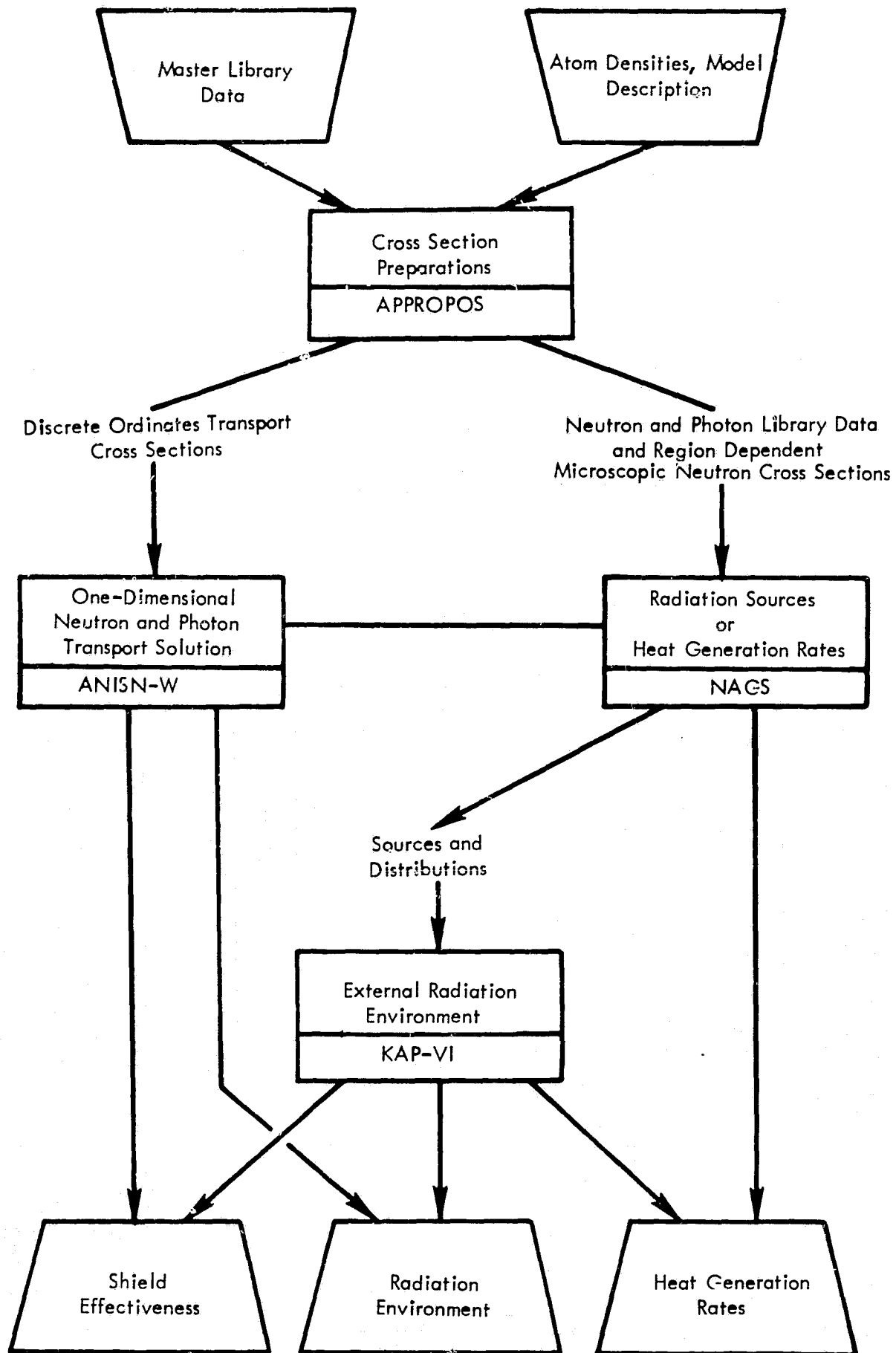


Figure 1-1. Flow Chart for Preliminary or Parametric Radiation Analysis

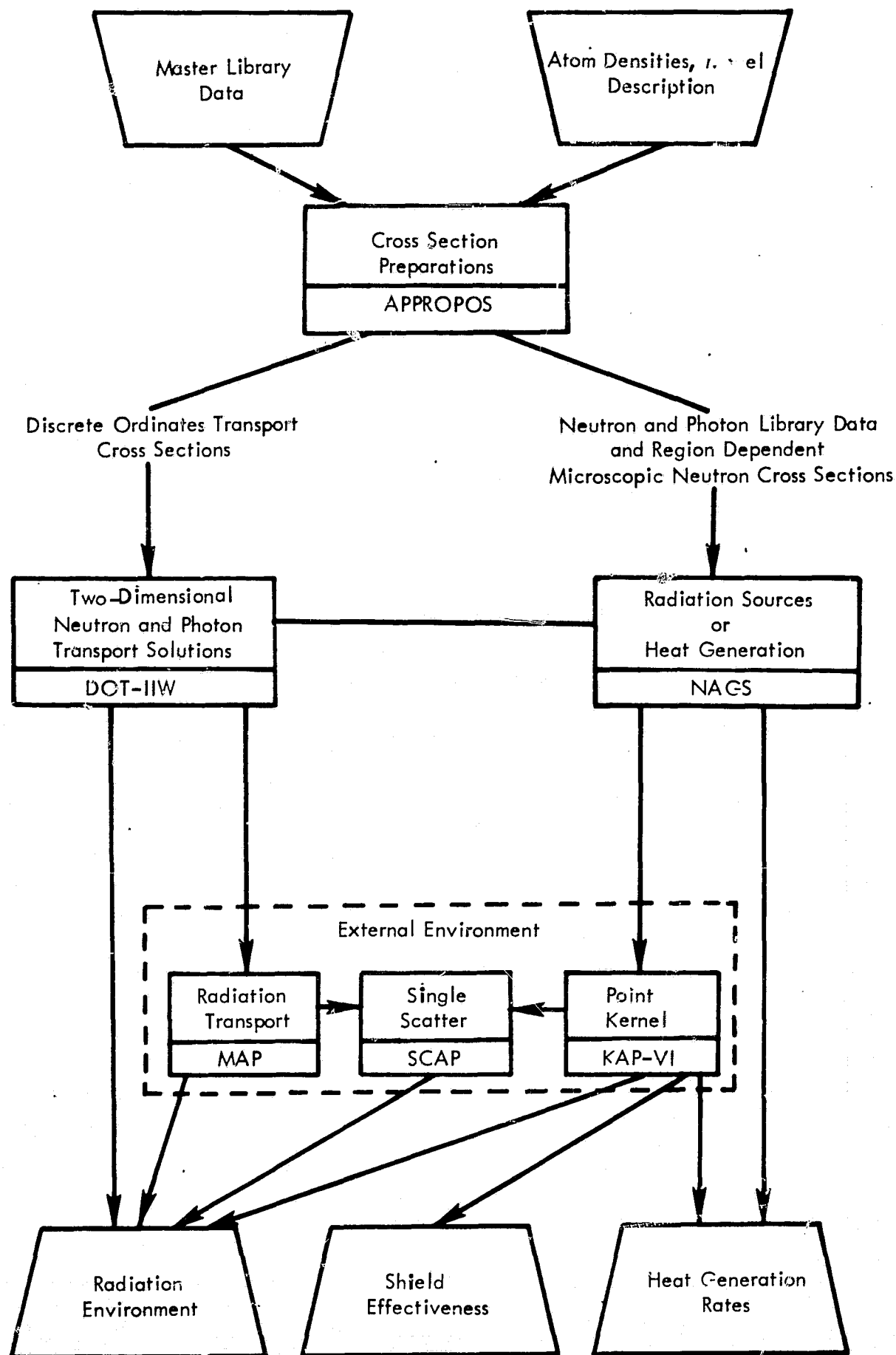


Figure 1-2. Flow Chart for Detailed Radiation Analysis

problem geometry are used as input to the MAP radiation transport code (Volume 5). The MAP code computes the radiation environment at selected surfaces or points external to the DOT-IIW geometry and includes provision for last-flight transport using optional point kernel techniques. The SCAP single- or albedo-scatter code (Volume 6) is used to compute external radiation environment using, as source input data, the output from either the KAP-VI or the MAP codes. Radiation sources, heat generation rates, and radiation environment, both internal and external to the reactor as well as shield effectiveness can be computed using the detailed design method.

The SATURN (Volume 3), DOQ (Volume 5), and ADOQ (Volume 5) codes are additional data preparation and handling codes. These codes are provided as convenient tools for manipulating large quantities of data or providing selected input data.

## 2.0 SUMMARY, CONCLUSIONS, AND RECOMMENDATIONS

Under this contract, the Westinghouse Astronuclear Laboratory has updated and expanded the current MSFC code package and has performed investigations with this augmented code package on a typical nuclear rocket engine. All computer codes developed or revised for the MSFC code package and documented by code manuals (Volumes 3 - 6) have been placed on an operational basis on the MSFC UNIVAC-1108 computer in the standard, USASI FORTRAN-IV (FORmula TRANslation) coding language. Because the identical computer codes are also operational on WANL's IBM 360/75 and CDC-6600 computers, the continuing operational status of these codes is further assured. These codes are also being sent to the Radiation and Shielding Information Center (RSIC), Oak Ridge, Tennessee, for dissemination to the shielding community.

WANL has performed the contracted work through the execution of the following seven tasks:

### Task 1    Compilation of Neutron and Photon Cross Section Data

The compilation of neutron and photon cross section data for MSFC has generated five Master Library Tapes and a deck of basic nuclear data as shown in Table 2-1 and described in detail in Volume 2 of this report.

Master Library Tape No. 1 contains 135 microscopic, transport corrected, neutron cross section sets for 36 elements for use in the ANISN-W, DOT-IIW, and APPROPOS codes. These data are in a 52 energy group structure. Master Library Tape No. 2 contains 413 microscopic,  $P_\ell$  ( $\ell \leq 3$ ) neutron cross section sets for 36 elements for use in the ANISN-W, DOT-IIW, and APPROPOS codes. These data are also in a 52 energy group structure. Master Library Tape No. 3 contains 134 sets of reaction rate cross section data for use in the APPROPOS code. The data on Master Library Tapes No. 1, 2, and 3 are averaged over four representative, spatially dependent, neutron spectra obtained from current NERVA R-1 nuclear subsystem design work. These first three tapes are obtained in a manner consistent with each other and with the nuclear and radiation analysis procedures used in the radiation analysis of reactors at the Westinghouse Astronuclear Laboratory (WANL).

TABLE 2-1  
SUMMARY OF MASTER LIBRARIES

| <u>MSFC Master<br/>Library Number</u> | <u>Description</u>                                                                                                                   |
|---------------------------------------|--------------------------------------------------------------------------------------------------------------------------------------|
| 1                                     | Microscopic, 52 Group, Transport Corrected Neutron Cross Section Sets for Use in the ANISN-W, DOT-IIW, and APPROPOS Codes            |
| 2                                     | Microscopic, 52 Group, $P_0$ , Neutron Cross Section Sets for Use in the ANISN-W, DOT-IIW, and APPROPOS Codes                        |
| 3                                     | Microscopic, 52 Group, Reaction Rate Cross Section Sets for Use in the APPROPOS Code                                                 |
| 4                                     | Microscopic, 13 Group, $P_0$ , Gamma Ray Cross Section Sets for Use in the ANISN-W, DOT-IIW, and APPROPOS Codes                      |
| 5                                     | Pair-Production and Photo-Electric Gamma Ray Cross Sections for Use in the KAP-VI, SCAP, MAP, and GAMLEG-W Codes                     |
| 6*                                    | Gamma Ray Production Data Due to Thermal Neutron Capture and Inelastic Scatter, Selected Nuclear Parameters, and Other Related Data. |

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\*On Cards

Master Library Tape No. 4 contains 510 microscopic,  $P_g$ , gamma ray cross section sets for 51 elements for use in the ANISN-W, DOT-IIW, and APPROPOS codes. These data, which are in a 13 energy group structure, are averaged over a source spectrum of prompt gamma ray energy emitted from the fission of  $^{235}\text{Uranium}$ . Master Library Tape No. 5 contains the pair-production and photo-electric gamma ray cross section data used in the KAP-VI, SCAP, MAP, and GAMLEG-W codes. Again, the data on Tapes 4 and 5 are obtained in a manner consistent with each other and with the radiation analysis procedures used in the radiation analysis of reactors at WANL.

Finally, a basic set of nuclear data is generated for use in the complete package of codes.

## Task 2    Auxiliary Data Preparation Techniques

WANL has developed, coded, and provided an expanded version of the POINT code,<sup>(4)</sup> called APPROPOS, for processing multigroup, neutron cross section data and multigroup, neutron reaction cross section data (e. g., radiative capture, fission, inelastic scatter, and elastic scatter) to provide spectral-weighted, neutron cross section data in a reduced number of energy groups. These data are then processed with input specified elemental atom densities, gamma ray production data due to neutron reactions, and photon cross section data to provide macroscopic, coupled, neutron-photon cross section data, neutron cross section data, and photon cross section data.

In addition, the SATURN data handling code has been added to the MSFC code package to prepare transport cross section data for the ANISN-W and DOT-IIW, discrete ordinate transport codes. SATURN provides the capability to analyze complex problems on the MSFC computer system with a minimum of computer memory use.

Both the APPROPOS and SATURN codes are described in detail in Volume 3 of this report.

Task 3 Asymmetric Quadrature Capability in the Discrete Ordinate Transport Code, DOT-IIW

WANL has provided in the MSFC code package the WANL version of the discrete ordinate transport code, DOT-II<sup>(5)</sup> called DOT-IIW. The DOT-IIW code includes an asymmetric quadrature capability as well as other desirable features. In addition, symmetric and asymmetric quadrature data and the capability to generate these data are provided with the DOQ and ADOQ codes.

The DOT-IIW, DOQ, and ADOQ codes are described in detail in Volume 5 of this report.

Task 4 Combined Point Kernel, Single Scatter Capability

WANL has provided the combined point kernel, single- or albedo-scatter technique in the SCAP computer code as part of the MSFC code package. The SCAP code is described in detail in Volume 6 of this report.

A systematic evaluation of the SCAP code has been performed to verify the use of the code as a shield design and parametric survey technique. Section 5.1 of this volume presents the results of this study. These results demonstrate that the SCAP code is a useful design and parametric survey tool provided the limitations of the method are not exceeded.

Task 5 Computer Code Compatibility with the MSFC Code Package

WANL has provided the necessary computer codes or revisions to the current MSFC code package to provide automatic data interfaces between the NAGS code and (a) the KAP-VI point kernel code (i. e., neutron and photon source distribution data); (b) the ANISN-W, DOT-IIW, and APPROPOS codes (cross section and flux data); and (c) the GAMLEG-W, SCAP, and KAP-VI codes. In addition, the MAP code, which treats the transport of radiation across a void external to a DOT-IIW calculation and provides a coupling technique for DOT-IIW calculations, has been provided in the MSFC code package.

Task 6    Provide Computer Codes on an Operational Basis on the MSFC UNIVAC-1108 Computer

WANL has provided all computer codes in standard, USASI FORTRAN-IV coding language compatible with the MSFC UNIVAC-1108 computer system. All codes have been compiled, checked out, and documented for this computer system. A series of calculations on a realistic nuclear rocket model (agreed upon by WANL and the MSFC technical monitor) has been set up. These calculations have not been completed on the MSFC UNIVAC-1108 computer for the following three reasons:

- 1)        The conversion of the Master Library Tapes from WANL's CDC-6600 computer to MSFC's UNIVAC-1108 computer delayed the efforts on Task 6 for a period of two and one-half months. The data tapes, as written by WANL under the direction of CSC\* personnel, could not be converted by the UNIVAC-1108 utility routines; UNIVAC-1108 personnel were also consulted during this period.
- 2)        Two specific instances of errors in the software of the UNIVAC-1108 computer delayed code conversion to the MSFC's computer for a total period of one month. The MSFC compiler improperly translated portions of the KAP-VI and ANISN-W codes. These two codes are written in standard, USASI (United States of America Standards Institute) FORTRAN-IV and have successfully compiled, loaded, and executed on WANL's IBM 360/75 and CDC-6600 computers with identical results (within the expected precision due to different word length). Each of WANL's two computer systems have completely separate Monitor Systems, and because of the standard, FORTRAN-IV coding philosophy at WANL, this delay at MSFC was not anticipated.
- 3)        Excessively long computer turnaround time at MSFC has delayed conversion, checkout, and execution of the code package throughout the period of performance. A time period of three to five working days have been

\* Computer Science Corporation, computer service support contractor to MSFC.

required to obtain a three-minute computer problem. This severely limited the number of debugging and checkout runs required for a computer code (this problem is magnified when the compiler has malfunctioned or a magnetic tape has failed). In a sequential design procedure, the next code cannot be run until the preceding results are obtained.

All computer codes in the MSFC package of nuclear codes, however, have been separately checked out on other sample problems. Results from MSFC's UNIVAC-1108 computer have been compared to the identical calculation on WANL's CDC-6600 or IBM 360/75 (whichever was convenient) and have yielded identical results within the precision limits of the respective computers. In one instance, the accuracy of the UNIVAC-1108 results was inadequate when compared to the CDC-6600 results. Consequently, the ADOQ code was converted to double precision (twice as many significant digits per calculation). With this modification, the desired precision was achieved.

All problem decks for the realistic model analysis have been checked out at WANL and transmitted to the Marshall Space Flight Center for computer submittal.

#### Task 7 LH<sub>2</sub> Radiation Transport Simulation with Polyethylene

An investigation of the simulation of LH<sub>2</sub> radiation transport with polyethylene has been carried out using the simultaneous, neutron-photon data and the ANISN-W discrete ordinate transport computer code. By the proper choice of CH<sub>2</sub> density, LH<sub>2</sub> radiation transport can be separately simulated for neutron and gamma ray sources.

#### Recommendations

Based on the development effort and analysis performed during this contract, the following recommendations are made:

- 1) Expand the MSFC code package of nuclear codes to include the MORSE, multigroup Monte Carlo code. The MORSE code uses the same multigroup cross sections as used in the ANISN-W and DOT-IIW codes. These required cross sections have already been provided in this contract as the six Master Library data files described in Volume 2. The MORSE code is

a self-contained code capable of neutron and/or photon, forward or adjoint computations, incorporates built-in analysis routines, and uses optional biasing techniques. This code is a logical extension to the capabilities of the MSFC code package provided by this contract.

- 2) Expand and improve the library of gamma ray production data in Master Library No. 6 to
  - a) include the effect of neutron energy in the neutron capture, gamma ray spectral data;
  - b) include additional element data and improve existing data based on new evaluations or models;
  - c) expand the neutron energy group definition to 52 energy groups to match the fine-group neutron library data.
- 3) Modify the APPROPOS code to perform a group-collapse and spectral-weighting of the Master Library No. 6 data developed in 2c above. With this modification, any broad-group, neutron-energy group structure can be obtained in the coupling calculation in the APPROPOS code.
- 4) Develop a coupling technique (or data link) between the DOT-IIW discrete ordinates transport code, the MAP radiation transport code, and the MORSE multigroup Monte Carlo code. Such a technique can significantly make the most advantageous use of each technique.
- 5) Investigate the use of adjoint kernel techniques in the analysis of external radiation environment. The use of adjoint kernel techniques to solve for the angular dependent, adjoint boundary source for use in a DOT-IIW problem is one possible application. In addition, the alternate technique of using an adjoint kernel technique to calculate the adjoint solution to each mesh cell within an  $r,z$  problem in combination with the use of a similar forward solution DOT-IIW problem is another area of interest. The use of adjoint kernels to the mesh cells provides a solution to the external

environment of the radiation source distribution obtained from the integration of the forward solution.

- 6) Investigate the use of adjoint flux calculations as a variance reduction technique in Monte Carlo analyses. A simplified, single scatter, adjoint technique to provide the polar-, spatial-, and energy-importance function can significantly improve the efficiency of a Monte Carlo code. This technique should be an integral part of the MSFC package of nuclear codes.
- 7) Provide the point kernel techniques to facilitate rapid, parametric analyses. Improve the automated transfer of data between point kernel codes; recode the KAP-VI code using the flexible dimension feature of FORTRAN-IV; and expand the capabilities of the SCAP code in the areas of exclusion radii, albedo model, and numerical methods.
- 8) Improve turnaround time at MSFC by reducing the number of physical tapes required for a particular calculation. Many of these codes require less than three minutes of central processor unit (CPU) time, but because as many as six data or library tapes are requested, poor turnaround time has been experienced. By combining data files on tape and assigning these data to separate FASTRAND disk files at the beginning of a calculation, this situation can be alleviated.
- 9) Investigate the discrepancy between MAP and KAP-VI results at locations where top leakage is predominant.

### 3.0 USE OF MSFC COMPLETE CODE PACKAGE

This section describes in detail the use of the complete code package for preliminary or parametric and detailed design radiation analysis. Because of the modular nature of these codes, considerable flexibility and interchangeability of techniques are possible. In addition, parallel paths may be chosen in an analysis to provide results based on different analytical techniques. Wherever possible, automated data linkage and data compatibility are provided.

#### 3.1 Preliminary or Parametric Design Method

In the preliminary or parametric design method (Figure 1-1), the APPROPOS code (Volume 3) is used to prepare neutron and photon cross sections and other basic data for use in the transport and data processing codes. The cross sections are input to the ANISN-W code (Volume 4). The ANISN-W code computes one-dimensional neutron and photon fluxes in the reactor geometry. From the neutron fluxes, neutron and photon energy sources and distributions are obtained using the NAGS data processing code (Volume 3). The sources and distributions are used as input to the KAP-VI point kernel code (Volume 6). The KAP-VI code provides gamma ray and fast neutron radiation levels at locations external to the reactor. Radiation sources, heat generation rates, and radiation environment, both internal and external to the reactor as well as shield effectiveness, can be computed using the preliminary or parametric design method.

For preliminary or parametric calculations, several alternate calculational procedures are possible. The first procedure involves calculations where either a neutron or photon transport solution is desired. This calculational procedure begins with either Master Library Tape No. 1, 2, or 4. The SATURN code may be used to prepare a cross section tape for the ANISN-W code. Parametric calculations using these data may be performed with the ANISN-W code.

The second procedure involves calculations where the definition provided by the fine-group neutron library data is not required, or where computer running time is important. With this procedure, a single, fine-group calculation is performed with the ANISN-W code on the basic geometry model. From this ANISN-W calculation, fine-group weighting spectra by region are obtained for use in the APPROPOS code. Next, a single APPROPOS

code calculation is performed to generate all the desired parametric compositions under study. Finally, broad-group, simultaneous, neutron-photon calculations are performed with the ANISN-W code on each of the specific configurations. Scoping study results with the ANISN-W code can often be obtained in 60 seconds of central processor unit (CPU) time or less per problem with this technique.

The effects of secondary gamma rays by region and/or element on the total gamma ray dose rate may be readily obtained by alternate NAGS-ANISN-W calculations, for example.

### 3.2 Detailed Design Method

In the detailed design method (Figure 1-2), the neutron and photon cross sections prepared by the APPROPOS code (Volume 3) are used as input data to the DOT-IIW, two-dimensional, discrete ordinates transport code. The DOT-IIW code (Volume 5) computes the two-dimensional neutron and photon fluxes throughout the reactor geometry. The NAGS data processing code (Volume 3) processes these fluxes and calculates neutron and photon energy deposition and neutron and photon energy sources and distributions within the reactor system. These sources and distributions are used as input to the KAP-VI point kernel code (Volume 6). In addition, the surface leakage fluxes from the DOT-IIW problem geometry are used as input to the MAP radiation transport code (Volume 5). The MAP code computes the radiation environment at selected surfaces or points external to the DOT-IIW geometry and includes provision for last-flight transport using optional, point kernel techniques. The SCAP single- or albedo-scatter code (Volume 6) is used to compute external radiation environment using, as source input data, the output from either the KAP-VI or the MAP codes. Radiation sources, heat generation rates, and radiation environment, both internal and external to the reactor as well as shield effectiveness can be computed using the detailed design method.

For detailed design radiation analysis, several alternate calculational procedures are possible. The first procedure of calculating radiation environment with the DOT-IIW code using coupled, neutron-photon transport cross section data may be bypassed by calculating

the neutron and photon radiation environment separately. The APPROPOS code generates the required cross section data for either procedure on the same cross section tape and the NAGS code can be used to generate the secondary gamma ray source data.

As shown in Figure 1-2, several codes are capable of calculating external radiation environment. Again, alternate paths may be chosen to take advantage of the capability of a particular technique. For example, point kernel results can be obtained quite efficiently for the generation of isodose rate contours external to a nuclear system.

Another technique that is quite useful is the coupling of two successive DOT-IIW calculations by using a fixed boundary source from the first DOT-IIW calculation in the second DOT-IIW calculation. The angular fluxes at three contiguous, axial mesh intervals can be saved from the first DOT-IIW calculation. These data are written on a tape by DOT-IIW for direct input to the second DOT-IIW calculation. The primary requirement on the geometrical model of the first DOT-IIW calculation is that the net flow of particles (or energy) across the chosen mesh interval must be conserved. This requirement is met by partially overlapping the geometrical models in the two calculations.

Other variations of the calculational procedure are possible and are primarily dependent on the desired calculational output parameters. Because of the modular nature of these codes and the data compatibility between codes, the user is provided considerable flexibility in tailoring the analysis procedure to the particular problem under study.

## 4.0 NUCLEAR SHIELDING METHODS AND MASTER LIBRARY DATA

Presented in this section is a description of each of the nuclear computer codes developed or revised for the MSFC code package. Also described are the six Master Library files of neutron and photon cross section data and basic nuclear data.

### 4.1 Compilation of Neutron and Photon Cross Section Data

The compilation of neutron and photon cross section data for MSFC has generated five Master Library Tapes and a deck of basic nuclear data as shown in Table 2-1 and described in detail in Volume 2 of this report.

Master Library Tape No. 1 contains 135 microscopic, transport corrected, neutron cross section sets for 36 elements for use in the ANISN-W, DOT-IIW, and APPROPOS codes. These data are in a 52 energy group structure. Master Library Tape No. 2 contains 413 microscopic,  $P_\ell$  ( $\ell \leq 3$ ) neutron cross section sets for 36 elements for use in the ANISN-W, DOT-IIW, and APPROPOS codes. These data are also in a 52 energy group structure. Master Library Tape No. 3 contains 134 sets of reaction rate cross section data for use in the APPROPOS code. The data on Master Library Tapes No. 1, 2, and 3 are averaged over four representative, spatially dependent, neutron spectra obtained from current NERVA R-1 nuclear subsystem design work. These first three tapes are obtained in a manner consistent with each other and with the nuclear and radiation analysis procedures used in the radiation analysis of reactors at the Westinghouse Astronuclear Laboratory (WANL).

Master Library Tape No. 4 contains 510 microscopic,  $P_\ell$ , gamma ray cross section sets for 51 elements for use in the ANISN-W, DOT-IIW, and APPROPOS codes. These data, which are in a 13 energy group structure, are averaged over a source spectrum of prompt gamma ray energy emitted from the fission of  $^{235}\text{Uranium}$ . Master Library Tape No. 5 contains the pair-production and photo-electric gamma ray cross section data used in the KAP-VI, SCAP, MAP, and GAMLEG-W code. Again, the data on Tapes 4 and 5 are obtained in a manner consistent with each other and with the radiation analysis procedures used in the radiation analysis of reactors at WANL.

Finally, a basic set of nuclear data is generated for use in the complete package of codes.

#### 4.1.1 Neutron Cross Section Data

A multigroup, microscopic library of neutron cross sections has been prepared for use in the discrete ordinates transport codes, DOT-IIW, and ANISN-W. Table 4-1 lists the 35 elements (or isotopes) for which cross section sets have been supplied. Transport corrected and full  $P_1$  scattering-approximation cross sections were generated for the elements or isotopes noted in the table. Each microscopic cross section set was spectrally weighted over four representative flux and current spectra. These spectra, as shown in Table 4-2, are selected from the NERVA project analyses of the current R-1 nuclear subsystem design and represent the spatially dependent neutron spectra in the R-1 reactor system. Full  $P_3$  scattering-approximation cross sections for neutron energies greater than 1.86 eV are supplied for 16 elements, as specified in Table 4-1. These data are weighted over a  $1/E$  spectrum. In addition, a vane prescription cross section set weighted over the R-1 reflector spectrum is included to represent the NERVA reactor peripheral control drums containing Boron-10. In one-dimensional (ANISN-W) or  $r,z$  two-dimensional (DOT-IIW) calculations, an annular ring containing the vane prescription cross section set is employed to represent the control drums.

All neutron cross sections are obtained in a manner consistent with the nuclear and radiation analysis procedures used in the radiation analysis of reactors at WANL. This cross section library is most applicable for radiation analyses of nuclear rocket reactors; however, the 52 energy group data are considered to be adequate for parametric survey calculations of other nuclear systems, as well.

#### 4.1.2 Photon Cross Section Data

The point kernel and discrete ordinates transport codes of the MSFC code package use the same basic, photon, cross section, library data and techniques. A basic library of energy dependent, photo-electric and pair-production data <sup>(6)</sup> for 51 elements as shown in Table 4-3 is used in the KAP-VI point kernel code, and the GAMLEG-W multigroup, cross

TABLE 4-1

COMPILATION OF NEUTRON CROSS SECTION DATA BY ELEMENT OR ISOTOPE

| Number | Element or Isotope            | Cross Section Scattering<br>Approximation* |       |       |
|--------|-------------------------------|--------------------------------------------|-------|-------|
|        |                               | Transport<br>Corrected                     | P0/P1 | P2/P3 |
| 1      | Hydrogen (20.4°K, Para)       | x                                          | x     | x     |
| 2      | Hydrogen (20.4°K, Ortho)      | x                                          | x     | x     |
| 3      | Hydrogen (300°K, Para)        | x                                          | x     | x     |
| 4      | Hydrogen (300°K, Ortho)       | x                                          | x     | x     |
| 5      | Hydrogen (2500°K, Gas Kernel) | x                                          | x     | x     |
| 6      | Carbon (296°K)                | x                                          | x     | x     |
| 7      | Beryllium (296°K)             | x                                          | x     | x     |
| 8      | Boron                         | x                                          | x     | x     |
| 9      | <sup>10</sup> Boron           | x                                          | x     | ---   |
| 10     | Aluminum                      | x                                          | x     | x     |
| 11     | Titanium                      | x                                          | x     | x     |
| 12     | Chromium                      | x                                          | x     | x     |
| 13     | Iron                          | x                                          | x     | x     |
| 14     | Copper                        | x                                          | x     | x     |
| 15     | Niobium                       | x                                          | x     | x     |
| 16     | Molybdenum                    | x                                          | x     | x     |
| 17     | <sup>235</sup> Uranium        | x                                          | x     | x     |
| 18     | <sup>238</sup> Uranium        | x                                          | x     | x     |
| 19     | Nickel                        | x                                          | x     | x     |
| 20     | Manganese                     | x                                          | x     | x     |
| 21     | Tantalum                      | x                                          | x     | ---   |
| 22     | Gadolinium                    | x                                          | x     | ---   |
| 23     | Tungsten                      | x                                          | x     | x     |
| 24     | Silicon                       | x                                          | x     | ---   |
| 25     | Oxygen                        | x                                          | x     | ---   |
| 26     | Magnesium                     | x                                          | x     | ---   |
| 27     | Zirconium                     | x                                          | x     | ---   |
| 28     | Lead                          | x                                          | x     | ---   |
| 29     | <sup>6</sup> Lithium          | x                                          | x     | ---   |
| 30     | <sup>7</sup> Lithium          | x                                          | x     | ---   |
| 31     | Lithium                       | x                                          | x     | ---   |
| 32     | Indium                        | x                                          | x     | ---   |
| 33     | Gold                          | x                                          | x     | ---   |
| 34     | Cadmium                       | x                                          | x     | ---   |
| 35     | Nitrogen                      | x                                          | x     | ---   |
| 36     | Cobalt                        | x                                          | x     | ---   |

\* x indicates the data have been included; --- indicates the data have not been included.

TABLE 4-2

SPECTRUM IDENTIFICATION NUMBER

- 1 - Core Center Spectrum
- 2 - Core Edge Spectrum
- 3 - Reflector Spectrum
- 4 - Shield Spectrum



TABLE 4-3

GAMMA RAY CROSS SECTION LIBRARY DATA PLACED ON TAPE BY GAMLEG-W

| Library Identification Number | Atomic Number | Element Name | Library Identification Number | Atomic Number | Element Name |
|-------------------------------|---------------|--------------|-------------------------------|---------------|--------------|
| 1                             | 1             | Hydrogen     | 26                            | 39            | Yttrium      |
| 2                             | 2             | Helium       | 27                            | 40            | Zirconium    |
| 3                             | 3             | Lithium      | 28                            | 41            | Niobium      |
| 4                             | 4             | Beryllium    | 29                            | 42            | Molybdenum   |
| 5                             | 5             | Boron        | 30                            | 47            | Silver       |
| 6                             | 6             | Carbon       | 31                            | 48            | Cadmium      |
| 7                             | 7             | Nitrogen     | 32                            | 49            | Indium       |
| 8                             | 8             | Oxygen       | 33                            | 50            | Tin          |
| 9                             | 11            | Sodium       | 34                            | 55            | Cesium       |
| 10                            | 12            | Magnesium    | 35                            | 56            | Barium       |
| 11                            | 13            | Aluminum     | 36                            | 62            | Samarium     |
| 12                            | 14            | Silicon      | 37                            | 64            | Gadolinium   |
| 13                            | 15            | Phosphorus   | 38                            | 66            | Dysprosium   |
| 14                            | 16            | Sulfur       | 39                            | 70            | Ytterbium    |
| 15                            | 19            | Potassium    | 40                            | 72            | Hafnium      |
| 16                            | 20            | Calcium      | 41                            | 73            | Tantalum     |
| 17                            | 22            | Titanium     | 42                            | 74            | Tungsten     |
| 18                            | 23            | Vanadium     | 43                            | 79            | Gold         |
| 19                            | 24            | Chromium     | 44                            | 80            | Mercury      |
| 20                            | 25            | Manganese    | 45                            | 82            | Lead         |
| 21                            | 26            | Iron         | 46                            | 84            | Polonium     |
| 22                            | 27            | Cobalt       | 47                            | 90            | Thorium      |
| 23                            | 28            | Nickel       | 48                            | 91            | Protactinium |
| 24                            | 29            | Copper       | 49                            | 92            | Uranium      |
| 25                            | 30            | Zinc         | 50                            | 93            | Neptunium    |
|                               |               |              | 51                            | 94            | Plutonium    |

TABLE 4-4  
GAMMA RAY ENERGY GROUP STRUCTURE

| <u>Group Number</u> | <u>Energy Bounds (MeV)</u> |
|---------------------|----------------------------|
| 1                   | 7.50 - 9.50                |
| 2                   | 7.00 - 7.50                |
| 3                   | 6.00 - 7.00                |
| 4                   | 5.00 - 6.00                |
| 5                   | 4.00 - 5.00                |
| 6                   | 3.00 - 4.00                |
| 7                   | 2.60 - 3.00                |
| 8                   | 2.20 - 2.60                |
| 9                   | 1.80 - 2.20                |
| 10                  | 1.35 - 1.80                |
| 11                  | 0.90 - 1.35                |
| 12                  | 0.40 - 0.90                |
| 13                  | 0.10 - 0.40                |

section preparation code. These codes calculate Compton scattering cross sections from the Klein-Nishina equation<sup>(7)</sup> and absorption cross sections from the basic library data to provide total cross section data. The numerical techniques used in the KAP-VI, SCAP, MAP, and GAMLEG-W codes are completely consistent; the final form of the data, however, is dependent on the code's individual requirements.

A microscopic, multigroup, gamma ray transport code section tape is generated in 13 energy groups using the GAMLEG-W code for the 51 elements shown in Table 4-3 using a  $P_9$  Legendre polynomial expansion of the scattering cross section. These data are contained on Master Library Tape No. 4. The multigroup structure of this library tape is shown in Table 4-4. This tape format is compatible with the ANISN-W, DOT-IIW, and APPROPOS codes. The tape contains a lead title record of two words, followed by the  $P_0$  data for each element, the  $P_1$  data, etc. A 105 energy point,  $^{235}\text{Uranium}$ , prompt fission photon source spectrum is used for averaging these cross sections. This source spectrum is adequate for photon transport analyses of graphite-moderated, nuclear rocket reactors. When used in the DOT-IIW and ANISN-W discrete ordinates transport codes, these cross sections permit the calculation of energy fluxes in units of  $(\text{MeV}/\text{cm}^2\text{-sec})$  as opposed to particle fluxes in units of  $(\text{photons}/\text{cm}^2\text{-sec})$ . Energy deposition calculations are more conveniently carried out using energy flux data.

A set of gamma ray production, spectral data are included in Master Library No. 6 for the elements or isotopes given in Table 4-5. Data are included for neutron capture and inelastic scatter reactions.

#### 4.2 Cross Section Generation and Data Processing Techniques

Four computer codes for the MSFC code package have been developed or revised to generate or process cross section data. The need for efficient techniques for preparing and handling multigroup, cross section data is an important part of a well-designed, flexible, analysis procedure. The GAMLEG-W, APPROPOS, NAGS, and SATURN codes satisfy this need.

##### 4.2.1 The GAMLEG-W Code

GAMLEG-W, a revised version of the GAMLEG code,<sup>(8)</sup> provides multigroup, photon transport cross sections for use in multigroup, discrete ordinate transport or Monte Carlo transport codes. The code is designed to provide photon transport cross sections in a

TABLE 4-5  
COMPILATION OF GAMMA RAY ENERGY SPECTRAL DATA BY ELEMENT OR ISOTOPE

| <u>Element or Isotope</u> | <u>Neutron Capture<br/>Gamma Ray<br/>Energy Spectral Data</u> | <u>Neutron Inelastic Scatter<br/>Gamma Ray<br/>Energy Spectral Data</u> |
|---------------------------|---------------------------------------------------------------|-------------------------------------------------------------------------|
| Hydrogen                  | x                                                             | ---                                                                     |
| Carbon                    | x                                                             | x                                                                       |
| Beryllium                 | x                                                             | ---                                                                     |
| Boron                     | x                                                             | ---                                                                     |
| <sup>10</sup> Boron       | x                                                             | ---                                                                     |
| Aluminum                  | x                                                             | x                                                                       |
| Titanium                  | x                                                             | x                                                                       |
| Chromium                  | x                                                             | x                                                                       |
| Iron                      | x                                                             | x                                                                       |
| Copper                    | x                                                             | ---                                                                     |
| Niobium                   | x                                                             | x                                                                       |
| Molybdenum                | x                                                             | ---                                                                     |
| <sup>235</sup> Uranium    | x                                                             | x                                                                       |
| <sup>238</sup> Uranium    | x                                                             | x                                                                       |
| Nickel                    | x                                                             | x                                                                       |
| Gold                      | x                                                             | ---                                                                     |
| Manganese                 | x                                                             | ---                                                                     |
| Tantalum                  | x                                                             | ---                                                                     |
| Gadolinium                | x                                                             | ---                                                                     |
| Tungsten                  | x                                                             | x                                                                       |
| Silicon                   | x                                                             | x                                                                       |
| Oxygen                    | x                                                             | x                                                                       |
| Magnesium                 | x                                                             | x                                                                       |
| Zirconium                 | x                                                             | x                                                                       |
| Lead                      | x                                                             | x                                                                       |
| <sup>6</sup> Lithium      | x                                                             | x                                                                       |
| <sup>7</sup> Lithium      | x                                                             | x                                                                       |
| Lithium                   | x                                                             | x                                                                       |
| Silver                    | x                                                             | ---                                                                     |
| Indium                    | x                                                             | ---                                                                     |
| Cadmium                   | x                                                             | ---                                                                     |
| Nitrogen                  | x                                                             | x                                                                       |

maximum of 100 groups with scatter transfer cross sections represented as a Legendre expansion ( $P_\ell$ ) of arbitrary order. The code performs a numerical integration with a user specified weighting function to obtain multigroup absorption and scatter-transfer cross sections. Absorption cross sections are obtained from pointwise, photoelectric and pair-production data on punched data cards or magnetic tape, and Compton absorption from the Klein-Nishina equation for the inelastic scattering of a photon with a free electron.<sup>(9)</sup> Scatter-transfer cross sections are obtained from the differential form of the Klein-Nishina equation for the inelastic scattering of a photon with a free electron.

The code can, at option, provide photon transport cross section data for use in energy flux or particle flux calculations. Production of a pair of 0.511 MeV photons due to pair production annihilation can be included in the energy scatter-transfer or particle scatter-transfer cross section data.

Output data from the GAMLEG-W code are compatible with the ANISN-W or DOT-IIW computer codes and are placed on either punched data cards or magnetic tape.

The GAMLEG-W code, as it is in production use on the UNIVAC-1108 at MSFC, is fully documented in Volume 3 of this report. Detailed user information including multigroup specifications, group integration requirements and library data requirements is provided. Typical problem setup information is supplied as well as a description of the printed output. A sample problem card input and printout are also included.

#### 4.2.2 The APPROPOS Code

In the analysis of nuclear reactor systems, an efficient technique for preparing multigroup, cross section data in the required format for discrete ordinate transport as well as Monte Carlo computer codes is required. The APPROPOS code provides this needed technique. APPROPOS prepares multigroup, simultaneous, neutron-photon cross section data and neutron reaction cross section data for radiation transport codes.

The APPROPOS code processes multigroup, neutron cross section data and multigroup, neutron reaction cross section data (e.g., radiative capture, fission, inelastic scatter, and elastic scatter) to provide spectral-weighted, neutron cross section data in a reduced number

of energy groups. These data are then processed with input specified elemental atom densities, gamma ray production data due to neutron reactions, and photon cross section data to provide macroscopic, coupled, neutron-photon cross section data, neutron cross section data, and photon cross section data. Output data from the APPROPOS code are placed on punched cards and/or magnetic tape for use in the ANISN-W, DOT-IIW, or NAGS codes.

Neutron and photon cross sections may be prepared by the GAMBIT<sup>(10)</sup> and GAMLEG-W codes, respectively. Photon production data may be generated with the POPOP4<sup>(11)</sup> code. A complete set of required library data for the APPROPOS code are fully described in Volume 2 of this report. Output data from the APPROPOS code are admissible to the one-dimensional ANISN-W, and the two-dimensional DOT-IIW discrete ordinate transport codes. A punched card library of photon production data and selected nuclear parameters is prepared for the NAGS data processing code.

The APPROPOS code, as it is in production use on the UNIVAC-1108 at MSFC, is fully documented in Volume 3 of this report. Detailed user information including neutron and gamma ray group structure specifications, weighting spectra, group-collapsing and up-scatter-removal, master library tape data specifications, composition data, and core storage requirements is provided. Typical problem setup information is supplied as well as a description of the printed output. A sample problem card input and printout is also included.

#### 4.2.3 The NAGS Code

When the vast amount of data handling required in a complete, two-dimensional radiation analysis of a nuclear system is considered, the need for an automated, data processing link between discrete ordinates transport techniques and both point kernel and Monte Carlo techniques is evident. The NAGS code provides this needed link. NAGS processes multi-group, neutron and photon flux distributions for one-dimensional, slab or cylinder geometry models, or two-dimensional  $r,z$  or  $r,\theta$  geometry models. Flux input data to the NAGS code are obtained by magnetic tape or disk from the ANISN-W or DOT-IIW discrete ordinate transport codes described in Volumes 4 and 5 of this report. Additional input data required by the NAGS code are prepared by the APPROPOS code described in this volume. The NAGS code provides: 1) neutron and photon source data for use in point kernel, Monte Carlo, or

discrete ordinate transport analyses, 2) neutron and photon dose rates, and 3) nuclear energy deposition data (neutron and photon) for use in subsequent thermal and hydraulic design analyses. Source distribution data are generated, on punched data cards for input into the KAP-VI code (Volume 6) and on magnetic tape for use in the ANISN-W or DOT-IIW codes. Optional SC-4020 plotting of the source distribution or energy deposition data is also provided.

The NAGS computer code utilizes the flexible dimensioning capability of FORTRAN-IV to obtain efficient data storage allocation. Because of this technique, no size restriction is imposed on any given array but the sum of the length of all data arrays is restricted by the available computer core memory.

The NAGS code, as it is in production use on the UNIVAC-1108 at MSFC, is fully documented in Volume 3 of this report. Detailed user information including geometry description, flux normalization, library data, region data, and problem size determination is provided. Typical problem setup information is supplied as well as a description of the printed output. A sample problem card input and printout is also included.

#### 4.2.3 The SATURN Code.

In certain calculations using the ANISN-W and DOT-IIW discrete ordinates transport codes with a large number of input cross section data, execution of the calculation is sometimes terminated because of insufficient core storage to process the cross section data. To alleviate this situation, the SATURN code was developed. SATURN processes  $P_L$  or transport corrected cross section data using a minimum amount of core storage to produce multigroup cross section data tapes in two formats for use in the ANISN-W or DOT-IIW codes. The two formats are the normal, group-dependent data tape or the specially prepared, group-independent data tape.

The data processing functions and options performed by the SATURN code are as follows:

- 1) Preparation of group-dependent and/or group-independent cross section data on magnetic tape or disk file and/or punched cards,

- 2) Inclusion of transverse leakage corrections to the macroscopic cross section data,
- 3) Preparation of macroscopic cross sections using mixing table operations,
- 4) Adjoint reversal of the cross section data, and
- 5) Output tape labeling and relabeling.

These operations are performed in the SATURN code as a sequence of logical steps to minimize the core storage required to prepare cross section data.

The SATURN code is an auxiliary code and should only be used for those situations where the DOT-IIW or ANISN-W codes cannot process the cross section data or for problems that require special, transverse leakage corrections, or group-independent tapes.

The SATURN code, as it is in production use on the UNIVAC-1108 at MSFC, is fully documented in Volume 3 of this report. Detailed user information including mixing table calculations, transverse leakage calculations, labeling options, and problem size determination is provided. Typical problem setup information is supplied as well as a description of the printed output. A sample problem card input and printout is also included.

#### 4.3 One-Dimensional, Discrete Ordinates, Transport Technique

In the analysis of nuclear systems, a one-dimensional, transport theory code serves as the basis for several types of calculations such as criticality, reactivity coefficients, critical size, preliminary shield design, flux attenuation, cross section group-collapsing and upscatter-removal. The WANL version of the ANISN computer code, ANISN-W, was developed to provide a flexible, efficient code to facilitate these calculations. The ANISN-W code is an integral part of both the preliminary or parametric and the detailed design radiation analysis methods provided for the Marshall Space Flight Center under this contract and the previous contractual work (NAS-8-20414).

##### 4.3.1 The ANISN-W Code

ANISN-W is a multigroup, discrete ordinate transport code that solves the energy dependent, one-dimensional, Boltzmann transport equation with general anisotropic scattering for slab, cylindrical or spherical geometries. The code solves forward or adjoint, homogeneous or inhomogeneous problems. The method is applicable to both neutron and gamma transport problems. Inhomogeneous problems may have a fixed volume distributed source, or a specified angular-dependent shell source at any mesh interval; fissions may be included for a subcritical system. Vacuum, reflective, periodic, white, or albedo boundary conditions may be specified. Time absorption calculations, concentration searches, outer radius searches, buckling searches, or zone thickness searches are also solved. Cross sections may be input from a library tape and/or from punched cards; multigroup flux distributions may be used to perform a group reduction of the cross section data. The code also includes space point scaling to accelerate the flux solution on inner iterations.

The ANISN-W code utilizes the variable dimensioning technique of FORTRAN-IV to facilitate efficient core data storage allocation. With this technique, no size restriction is imposed on any given data array; a limitation is only placed on the sum of the length of all the data arrays. The total core storage required for a given problem may be exactly computed.

To provide a more efficient tool, the ANISN-W code has been improved over the original ANISN code<sup>(12)</sup> to provide for the inclusion of automated tape linkage of input and



output data and for the capability to input a distributed or shell source from magnetic tape and/or punched cards. Microscopic and/or macroscopic cross section data may be input to ANISN-W from cards and/or a library tape. Cross sections from tape can be input from a microscopic library data tape (the mixing table option in ANISN-W can be employed to calculate macroscopic cross sections) or from a SATURN prepared, group-dependent or group-independent, cross section tape. Volume 3 of this report describes the SATURN cross section preparation code.

The ANISN-W computer code, as it is in production use on the UNIVAC-1108 at MSFC, is fully documented in Volume 4 of this report. Detailed user information including problem setup, running time, boundary conditions, quadrature data and mesh spacing requirements is provided. Requirements for cross section weighting, search calculations, activities and stacked problems are given. Typical problem setup information is supplied as well as a description of the printed output. A sample problem card input and printout are also included.

#### 4.4 Two-Dimensional, Discrete Ordinate, Transport Techniques

The detailed design of a nuclear system requires a two-dimensional technique for predicting the transport of neutrons and/or photons through matter. A two-dimensional, transport theory technique serves as the basis for many different types of calculations such as criticality, radiation environment, shield effectiveness, and flux attenuation. The DOT-IIW and MAP codes provide flexible, efficient techniques to facilitate these calculations.

The WAPL version of the DOT-II computer code, DOT-IIW, solves the two-dimensional, energy dependent, linear Boltzmann transport equation with general anisotropic scattering; the MAP computer code solves for the angular, spatial, and energy dependent flux external to a nuclear system. The DOQ and ADOQ computer codes generate the symmetric and asymmetric discrete ordinates quadrature coefficients (direction cosines and weights) required as input to the DOT-IIW and MAP codes. The DOT-IIW and MAP codes are an integral part of both the preliminary or parametric and the detailed design radiation analysis methods provided for the Marshall Space Flight Center (MSFC) under this contract and the previous contractual work (NAS-8-20414).

#### 4.4.1 The DOT-IIW Code

DOT-IIW is an implementation of the discrete ordinates transport technique. This code solves the two-dimensional, energy dependent, linear Boltzmann transport equation with general anisotropic scattering for  $x, y; r, z$  and  $r, \theta$  geometries. The code solves forward or adjoint, homogeneous or inhomogeneous problems. The method is applicable to both neutron and gamma ray transport problems. Inhomogeneous problems may have a fixed volume distributed source, or a specified angular dependent boundary source at the right or top boundaries; fissions may be included for a subcritical system. Vacuum, reflective, periodic, white, or albedo boundary conditions may be specified. Time absorption calculations, concentration searches, or zone thickness searches are also solved. Cross sections may be input from a library tape and/or from punched cards. Asymmetric or symmetric quadrature calculations may be performed. The code includes a choice of Gaussian Iteration, Successive Overrelaxation, Space Point Scaling, or Chebyshev Acceleration to achieve a flux solution on inner iterations.

DOT-IIW is written in standard, USASI FORTRAN-IV. The code utilizes the variable dimensioning technique of FORTRAN-IV to facilitate efficient core data storage allocation. With this technique, no size restriction is imposed on any given data array; a limitation is only placed on the sum of the length of all the data arrays. The total core storage required for a given problem may be exactly computed.

The version of the DOT-IIW code provided to Marshall Space Flight Center under this contract has been changed from the original version of DOT-II to be consistent with the nuclear and radiation analysis methods provided MSFC, and to incorporate state-of-the-art features of the current DOT-II code as released by RSIC (with the exception of a polynomial source option). The major improvements in DOT-IIW are the inclusion of acceleration techniques on the group flux solution, improved convergence logic, asymmetric quadrature capability, and improved tape operations.

Microscopic and/or macroscopic cross section data may be input to DOT-IIW from magnetic tape and/or punched cards. Cross sections from tape can be input from a microscopic library data tape (the mixing table option in DOT-IIW can be employed to calculate macroscopic cross sections) or from a SATURN prepared, group-dependent or group-independent,

cross section tape. Volume 3 of this report describes the SATURN cross section preparation code. Because DOT-IIW contains no provision to directly input a transverse leakage term into the calculation (e.g., as required by an  $r, \theta$  problem), a correction factor of the  $DB^2$  form can be included in the cross sections with the SATURN code.

The DOT-IIW computer code, as it is in production use on the UNIVAC-1108 at MSFC, is fully documented in Volume 5 of this report. Detailed user information involving problem setup, running time, quadrature data, and mesh spacing requirements is provided. Requirements for eigenvalue, fixed volume distributed source, boundary source, and search calculations are given. Typical problem setup information is supplied as well as a description of the printed output. A sample problem card input and printout are also included.

#### 4.4.2 The MAP Code

The MAP code solves for the energy-dependent radiation transport external to a nuclear radiation source. The code uses as energy- and angular-dependent, surface source data the multigroup, angular-dependent surface leakage fluxes, as calculated by an  $r, z$  geometry discrete ordinate transport calculation. The capabilities of the MAP code include:

- 1) Two techniques of treating symmetric quadrature, discrete ordinate transport, angular-dependent, leakage fluxes to provide a continuous variation of flux with angle;
- 2) Techniques to use generalized quadrature, discrete ordinate transport, angular-dependent leakage fluxes (including asymmetric sets) at both the source surface and detector surface;
- 3) Techniques to produce magnetic data tapes of angular-dependent flux data for use as boundary-source input into subsequent, discrete ordinate transport problems; and
- 4) Techniques to calculate the uncollided radiation as well as estimates of the collided radiation (using point kernel techniques) reaching detector surfaces in a geometry described by intersecting, quadratic surfaces.

Cross section data are obtained from 1) internal calculations of point value data from Klein-Nishina relationships and data tables for photons, or 2) input values of macroscopic cross sections for materials in zones. This code utilizes the variable dimension technique of FORTRAN-IV and the input data capability of the FIDO subroutine as used in the ANISN-W, DOT-IIW and SCAP codes.

The MAP code is an integral part of the automated, radiation transport, calculation capability provided to MSFC under this contract. This code provides techniques which circumvent the use of discrete ordinate transport codes in calculating radiation transport through voids or near voids. MAP provides accurate results in areas where "ray effects"<sup>(13)</sup> (due to discrete direction angular flux solutions) produce anomalous results and provides for extended use of the discrete ordinate transport technique. MAP provides this capability without using high orders of angular quadrature or conventional symmetric quadrature techniques in discrete ordinate transport codes.

The MAP radiation transport code, as it is in production use on the UNIVAC-1108 at MSFC, is fully documented in Volume 5 of this final report. Detailed user information including problem setup, running time, required input data, and recommended use is provided. Problem setup information including input tapes as well as a description of printed, punched, and tape output is described. A sample problem card input and printout for the MAP code, as well as a companion DOT-IIW problem card input and printout are also included.

#### 4.4.3 The DOQ Code

The ANISN-W one-dimensional and DOT-IIW two-dimensional discrete ordinates transport codes require as input data the discrete directions for which solutions are desired. These discrete directions cannot be chosen arbitrarily. Because the energy-angle relationship of the scatter cross sections used by these transport codes are represented by Legendre polynomials, quadrature sets that correctly integrate Legendre polynomials are required for particle conservation. In addition, symmetry conditions on discrete directions are important. The ANISN-W and DOT-IIW codes perform specific checks on these quadrature data to insure consistency. At least seven significant digits of precision for the input quadrature coefficients are required.

The DOQ Discrete Ordinates Quadrature code\* was developed to assist the user in the preparation of these input coefficients to the transport codes. The DOQ code computes the discrete ordinates quadrature coefficients (direction cosines and weights) given the point coordinates and symmetry conditions. A generalized technique using the method of moments is used. The code prints out and punches on cards the quadrature coefficients in a format

\* Originally developed as the SNAFU code by R. G. Rodgers, F. R. Mynatt, and W. W. Engle, Jr., of the Computer Technology Center, Oak Ridge, Tennessee.

appropriate for direct input to the ANISN-W and DOT-IIW codes. Output from the DOQ code may also be used as input data to the ADOQ asymmetric quadrature code. The DOQ code utilizes the variable dimensioning technique of FORTRAN-IV to allow the calculation of high orders of quadrature by efficient use of core memory storage.

The DOQ computer code, as it is in production use on the UNIVAC-1108 at MSFC, is documented in Volume 5, Appendix A, of this report. Detailed user information, typical problem setup information and a description of printed and punched output are supplied. A sample problem card input and printout are also included.

#### 4.4.4 The ADOQ Code

The importance of accurately predicting the nuclear radiation environment and radiation transport external to a nuclear reactor system has necessitated the development of asymmetric quadrature techniques for use in discrete ordinate transport analyses. These two-dimensional, transport theory techniques have significantly reduced "ray effects" and have provided increased angular resolution of the spatially attenuated nuclear subsystem leakage spectrum. When conventional symmetric quadrature transport techniques are used to calculate the radiation transport through low-scattering media or voids, "ray effects" are observed with low order quadratures. These "ray effects," or anomalous computational results, are due to the discrete representation of the angular variable in discrete ordinate transport theory approximations and predict preferential radiation streaming along the discrete directions of angular quadrature chosen.

Accurate angular resolution of the spatially attenuated leakage spectrum at locations far removed from the nuclear system is also important in the prediction of neutron thermalization in local media for the calculation of secondary gamma ray sources.

Analytical radiation transport techniques have been developed to satisfy both of these requirements. The method consists of 1) the development<sup>(5)</sup> of asymmetric quadrature data using the DOQ and ADOQ computer codes to achieve a high degree of angular resolution along the axis and in the solution plane of interest; 2) the modification of the DOT two-dimensional, discrete ordinate transport code to use these data; and 3) the development and coding of the MAP code to reconstruct the angular and energy dependence of the neutron

or photon flux as a function of position external to the nuclear system using the reactor surface angular fluxes from the DOT code as input data.

In addition, techniques in the MAP code for processing asymmetric quadrature data at the reactor surface and providing asymmetric quadrature angular flux at the surface of a succeeding DOT problem of a medium separated from the reactor have been developed.

The ADOQ code is a technique for 1) combining asymmetric quadrature data with symmetric quadrature data in one hemisphere of the unit  $S_n$  sphere, 2) adjusting the level weights of the last asymmetric level to match the symmetric level, and 3) verifying the various relationships these data must satisfy.

ADOQ prints out and punches on cards the quadrature coefficients (direction cosines and weights) in a format appropriate for direct input to the DOT-IIW code. Up to 200 discrete directions may be input in the symmetric hemisphere data; up to 100 discrete directions may be input in the asymmetric portion of the asymmetric hemisphere; and up to a total of 500 discrete directions may be computed. If these maximum values become limiting, this FORTRAN program can be easily changed. A set of asymmetric quadrature data, as generated by the ADOQ code, has been successfully used in DOT-IIW.

The ADOQ computer code, as it is in production use on the UNIVAC-1108 at MSFC, is documented in Volume 5, Appendix B, of this report. Detailed user information, typical problem setup information and a description of printed and punched output are supplied. A sample problem card input and printout are also included.

#### 4.5 Point Kernel Techniques

In the analysis of nuclear systems, point kernel technique codes serve as the basis of calculations such as parametric analyses of nuclear rocket or space power shield designs, space nuclear propulsion system configurations, propellant management studies, and radiation environment and effects studies. The KAP-VI point kernel code and the SCAP single-scan albedo-scatter code provide a flexible point kernel method with consistent geometry and cross section data. Punched data and output from the KAP-VI code are used as input data to the SCAP code. The KAP-VI and SCAP codes are an integral part of both the preliminary or parametric and

the detailed design radiation analysis methods provided for the Marshall Space Flight Center under this contract and the previous contract, NAS-8-20414.

#### 4.5.1 The KAP-VI Code

KAP-VI is a point kernel code designed to calculate the radiation level at detector points located within or outside a complex radiation source geometry. This geometry can be described by a combination of quadratic surfaces with a maximum of six boundary surfaces per zone. The code evaluates the material thicknesses intercepted along the line-of-sight from the source point to the detector point. These material thicknesses (or path lengths) are then employed in attenuation functions to calculate the flux, dose rate, or heating rate at the detector. The attenuation function for gamma rays employs exponential attenuation with a buildup factor. Three optional neutron attenuation functions are included: (1) a modified Albert-Welton function for calculating fast neutron dose rate using removal cross sections; (2) a bivariate polynomial expression for computing neutron spectra using infinite media moments data; and (3) a monovariant polynomial for computing neutron spectra using infinite media moments data.

The KAP-VI code handles cylindrical, spherical, disc, line, or point sources. Separate source distributions may be employed for neutrons and gamma rays with a variety of available options for describing the source distributions. Radiation source distributions are assumed to be separable energy- and spatial-distributions. Internally calculated normalization is included to provide a volumetric source in cylindrical and spherical coordinates. An option is also provided to describe an azimuthal source density variation by specifying input data for discrete point sources.

The code is limited to 30 gamma ray and 30 neutron groups. A total of 100 surfaces and 100 zones are possible; the volumetric source zone can be described by a maximum of 4000 finite source volumes each representing a point source. Gamma ray absorption coefficients (cross sections) can be supplied by magnetic tape from the GAMLEG-W code (Volume 3) and neutron and gamma ray source distribution data can be supplied on punched cards from the NAGS Code (Volume 3). The KAP-VI code provides, on punched cards, detector flux data for use as effective anisotropic source data in SCAP code calculations (Volume 6).

#### 4.5.2 The SCAP Code

SCAP is a point kernel code employing energy-dependent single- or albedo-scatter methods to calculate the radiation level at a detector point located within or outside a complex scattering geometry. This geometry can be described by a combination of quadratic surfaces. The code evaluates the material thickness and scatter points in materials intercepted along the lines-of-sight from an anisotropic energy-dependent, source point. The material thicknesses (or path lengths) to each scatter point are used in an exponential attenuation function to calculate the radiation level at each scatter point.

Three options exist in the treatment of particle scattering to a detector point. Photon scattering from the scatter point to the detector point can be treated as a Compton scatter event using the differential form of the Klein-Nishina equation for the inelastic scattering of a photon with a free electron, or as a simple, albedo-scatter event at the surface of a geometric zone. Neutron scattering from the scatter point to the detector point can be treated only as an albedo-scatter event at the surface of the geometric zone. The attenuation function for the scattered particle (photon or neutron) on the scatter leg to the detector point is an exponential attenuation function with a buildup factor for photons only.

The code handles a series of anisotropic energy-dependent point sources. These anisotropic point sources are described with pointwise, energy-dependent distributions as a function of polar angle on a meridian.

SCAP is written using the flexible dimensioning technique of FORTRAN-IV so that no limits are imposed on the number of energy groups or geometric zones; only the total data array is restricted to the dimension of blank common. The geometric zone description is restricted to a maximum of six boundary surfaces per zone where each surface is defined by the general quadratic equation (or one of its degenerate forms). The SCAP running time is essentially independent of the number of energy groups and is only dependent upon the calculation of geometry-dependent data. Gamma ray absorption coefficients (cross sections) can be supplied by magnetic tape from the GAMLEG-W code (Volume 3) and neutron and gamma ray point source distribution data can be supplied on punched cards from the KAP-VI (Volume 6) and MAP (Volume 5) codes as anisotropic, energy-dependent source data.

The SCAP code, as it is in production use on the UNIVAC-1108 computer system at MSFC, is fully documented in Volume 6 of this report. Detailed user information, including input data for specified arrays, and a description of output data is provided. Problem setup information, including tape assignments and running time, is supplied. A description of the code logic and of the numerical method of solution, and a complete sample problem are also included.

## 5.0 EVALUATIONS AND CALCULATIONS

Included in this section are the evaluations and calculations performed with the code package supplied to the Marshall Space Flight Center. These results reflect the capability and the flexibility of the techniques provided MSFC and provide insight into methods of application.

### 5.1 Investigation of the Applicability of the SCAP Code to Nuclear Rocket Engine Shielding Analysis

A systematic evaluation of the SCAP program is reported in this section. Such an evaluation is required for the code to be logically used as a shield design and parametric survey technique. This evaluation, which was conducted during the development of the code, is included to demonstrate the validity of the technique in nuclear rocket shielding analyses. The evaluation of SCAP was performed to attempt to answer several questions requiring resolution prior to initiating a parametric analysis. These questions included:

- 1) What number of scatter points in the propellant tank is required for accurate single scattering analysis?
- 2) How sensitive is the scattering to the number of polar angle rays and the spacing of points along these rays?
- 3) What radius of meridian ring should be used to obtain input data for SCAP; i. e., how sensitive is SCAP to various radii meridian ring data?
- 4) How sensitive is SCAP to the spatial distribution of gamma energy fluxes on the meridian ring; i. e., what is the difference between a flat meridian ring?
- 5) How sensitive is SCAP to the location of the center of the meridian ring; i. e., how sensitive is it to the location of the effective point source of incident radiation?

- 6) How significant is the difference in scattered radiation obtained if MAP meridian ring data are used instead of KAP-VI meridian ring data; i. e., how sensitive to spectral changes?

These questions and the study performed to attempt to answer them are discussed in greater detail in the following sections.

#### 5.1.1 Number of Scattering Points

The number of scattering points to be placed within the liquid hydrogen, gaseous hydrogen, tank wall aluminum, and insulation of the propellant tank is determined by three input parameters to the SCAP code. The first parameter is the number of polar angles into which a specified polar angle range is to be subdivided. For example, if one specifies the minimum polar angle to be zero degrees and the maximum to be 25 degrees (a polar angle which encompassed all of the sources in the scattered region), and if 25 angular intervals are specified, then radiation would be directed into the tank along 25 rays spaced one degree apart, the first ray being at an angle of 0.5 degrees. The second parameter in determining the number of scatter points is the number of azimuthal intervals to be employed. This factor has significance when the configuration lacks symmetry. This condition occurs, for example, when either the point source of incident radiation or the detector point is not located on the axis of symmetry. The third input parameter in determining the total number of scatter points is a quantity specifying at what interval along each ray the scatter points are spaced. The scatter point locations along each ray are calculated internally based on an electron path length per scatter point (electrons/cm<sup>2</sup>), a predetermined input quantity. This calculation is done for each zone crossed by the ray. At least one scatter point is placed in each zone regardless of the electron density in the zone. The above can best be explained by the following example: If a value of  $6 \times 10^{24}$  electrons/cm<sup>2</sup> is input and the electron density of the material in the zone of interest is  $4.2 \times 10^{22}$  electrons/cm<sup>3</sup>, then the scatter point spacing will be approximately  $6 \times 10^{24} / 4.2 \times 10^{22}$  or 143 cm along the ray.

A study was performed in which the number of polar angle source rays and the number specifying the path length per scatter were varied. The configuration consists of the nuclear subsystem and a 60-degree propellant tank with an LH<sub>2</sub> propellant depth of 570 cm.

The results of this study are summarized in Table 5-1. From Table 5-1, the scattered dose rate at the top of the 60-degree tank is relatively insensitive to the number of polar angle source rays and path length per scatter for the cases considered. Although the effects are small, the scattered dose rate increases with increasing path length per scatter for a given number of source rays and decreases as the number of polar angle source rays is increased for a given path length per scatter. As a result of this study, 25 source rays between polar angles of  $0^\circ$  and  $20.88^\circ$ , and  $2 \times 10^{24}$  electrons/cm<sup>2</sup> path length per scatter were used in the scattering analysis for the 75-degree propellant tank study. The use of the above numbers in calculating the scattering from the nuclear subsystem sources results in typical numbers of scatter points of 274 for the case in which no LH<sub>2</sub> is present and 1,042 for the case in which the tank is completely filled with LH<sub>2</sub>. The nuclear subsystem sources used throughout this analysis were obtained from concurrent NERVA program design analyses using the Common Radiation Analysis Model (CRAM). The total number of scatter points is printed out as part of the SCAP output.

#### 5.1.2 Radius of Meridian Ring

Because the basis of all SCAP calculations is a multigroup, gamma ray flux distribution calculated on a meridian ring by the KAP-VI point kernel code (or by other suitable methods), the question arises as to what the effect of different radii meridian rings is on the results of the SCAP code. For the SCAP calculations to be independent of the radius of the meridian ring, the radiation levels on various meridian rings would have to differ only as a function of distance from an effective point source located at the center of the meridian ring. Stated another way, the radiation must actually originate at the point source for the SCAP results to be independent of the meridian ring data. This, of course, is not the case because of the volume distribution of sources in the reactor. A meridian ring whose radius is too small might, for example, include radiation directed such as to miss the propellant tank or may exclude radiation which hits the propellant tank.

The KAP-VI program was used to calculate data on meridian rings having radii of 472, 1554, 2636 cm. These radii correspond roughly to the bottom, center, and top of a

TABLE 5-1

EFFECT OF VARYING THE NUMBER OF SCATTER POINTS IN SCAP SCATTERING ANALYSIS CODE  
(GAMMA RAY DOSE RATE, Rem/hr)

| <u>Number of<br/>Source Rays</u> | <u>Path Length<br/>Per Scatter</u> | <u>LH<sub>2</sub><br/>Scatter</u> | <u>GH<sub>2</sub><br/>Scatter</u> | <u>Insulation<br/>Scatter</u> | <u>Aluminum<br/>Scatter</u> | <u>Total<br/>Scatter</u> | <u>Direct</u> | <u>Direct ±<br/>Scatter</u> |
|----------------------------------|------------------------------------|-----------------------------------|-----------------------------------|-------------------------------|-----------------------------|--------------------------|---------------|-----------------------------|
| 12                               | 1                                  | 4.154(3)                          | 9.650(1)                          | 2.507(2)                      | 7.772(2)                    | 5.279(3)                 | 2.696(3)      | 7.975(3)                    |
| 12                               | 2                                  | 4.155(3)                          | 9.650(1)                          | 2.507(2)                      | 7.766(2)                    | 5.279(3)                 | 2.696(3)      | 7.975(3)                    |
| 12                               | 6                                  | 4.159(3)                          | 9.650(1)                          | 2.507(2)                      | 7.738(2)                    | 5.280(3)                 | 2.696(3)      | 7.976(3)                    |
| 23                               | 1                                  | 4.057(3)                          | 9.529(1)                          | 2.566(2)                      | 8.178(2)                    | 5.226(3)                 | 2.696(3)      | 7.922(3)                    |
| 23                               | 2                                  | 4.058(3)                          | 9.529(1)                          | 2.566(2)                      | 8.169(2)                    | 5.227(3)                 | 2.696(3)      | 7.922(3)                    |
| 23                               | 6                                  | 4.063(3)                          | 9.529(1)                          | 2.566(2)                      | 8.140(2)                    | 5.229(3)                 | 2.696(3)      | 7.924(3)                    |
| 46                               | 1                                  | 4.040(3)                          | 9.147(1)                          | 2.988(2)                      | 6.735(2)                    | 5.103(3)                 | 2.696(3)      | 7.799(3)                    |
| 46                               | 2                                  | 4.040(3)                          | 9.147(1)                          | 2.988(2)                      | 6.730(2)                    | 5.104(3)                 | 2.696(3)      | 7.800(3)                    |
| 46                               | 6                                  | 4.045(3)                          | 9.147(1)                          | 2.988(2)                      | 6.712(2)                    | 5.106(3)                 | 2.696(3)      | 7.802(3)                    |

TABLE 5-1

Notes: 1) Detector point located at the top of the 60° tank.

2) LH<sub>2</sub> propellant depth = 570 cm.

3) 30 foot meridian ring gamma ray fluxes used in SCAP.

4) Numbers in parentheses refer to power of ten.



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60-degree propellant tank. SCAP calculations were then made using these meridian ring data to calculate the dose rate at the top of the 60-degree propellant tank containing 570 cm of  $\text{LH}_2$  propellant. The difference in direct radiation in Table 5-2 is about 15 percent for the smallest and largest meridian rings. The direct radiation calculated from the largest meridian ring is believed most correct. In SCAP, the transport of direct radiation from the meridian ring to the detector point is based on an inverse " $R^2$ " spatial divergence. This approximation is more nearly valid at large distances from the reactor (i. e., the farther one is from the reactor, the more the reactor approaches a point source). The difference in the total scatter shown in Table 5-2 is about 30 percent for the small versus large meridian rings. The difference in total scatter for the 1500 cm meridian ring versus the 2600 cm meridian ring is only 5 percent. With respect to the total (direct plus scattered) radiation, the difference for the extreme cases of meridian ring radii is about 19 percent.

As a result of this study, KAP-VI data on a 30-foot meridian ring were generated for input to the SCAP code in all subsequent analyses.

### 5.1.3 Spatially Averaged Meridian Ring Fluxes

Meridian ring gamma ray energy fluxes for typical shielded NERVA reactors have a spatial distribution characterized by low values on or near the axis at small polar angles and higher values (which result from radiation bypassing internal central shielding) at somewhat larger polar angles. To determine what effect a redistribution of the gamma ray energy flux on the meridian ring would have on the scattered dose rate at the top of the tank, meridian ring data was spatially averaged in each of 13 energy groups. The actual and averaged distributions were then input into the SCAP code to calculate the dose rate at the top of the 60-degree tank containing a 570 cm depth of liquid hydrogen propellant. The reactor geometry for this study included a 15-inch BATH central shield. Interestingly, the total gamma ray energy flux in each energy group in the polar angle range 0 to 25 degrees was exactly the same for each distribution. Also, averaging the spatial distribution had the effect of reducing the flux at large polar angles and increasing the flux at small polar angles.

TABLE 5-2

## EFFECT OF USING VARIOUS RADII MERIDIAN RING GAMMA RAY FLUXES

IN SCAP SCATTERING ANALYSIS CODE  
(GAMMA RAY DOSE RATE, Rem/hr)

| Radius of<br>Meridian Ring<br>(cm) | LH <sub>2</sub> |  | GH <sub>2</sub> |  | Insulation     |  | Aluminum       |  | Total          |  | Direct        |  | Direct+<br>Scatter |
|------------------------------------|-----------------|--|-----------------|--|----------------|--|----------------|--|----------------|--|---------------|--|--------------------|
|                                    | <u>Scatter</u>  |  | <u>Scatter</u>  |  | <u>Scatter</u> |  | <u>Scatter</u> |  | <u>Scatter</u> |  | <u>Direct</u> |  |                    |
| 472                                | 3.518(3)        |  | 9.395(1)        |  | 1.704(2)       |  | 5.596(2)       |  | 4.342(3)       |  | 2.984(3)      |  | 7.038(3)           |
| 1554                               | 4.230(3)        |  | 9.775(1)        |  | 2.423(2)       |  | 7.888(2)       |  | 5.358(3)       |  | 2.652(3)      |  | 8.054(3)           |
| 2636                               | 4.448(3)        |  | 1.001(2)        |  | 2.601(2)       |  | 8.456(2)       |  | 5.654(3)       |  | 2.589(3)      |  | 8.350(3)           |

Notes: 1) Detector point located at the top of the 60° tank.

2) LH<sub>2</sub> propellant depth = 570 cm.

3) Numbers in parentheses refer to powers of ten.

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Results of this study are presented in Table 5-3. Because averaging the distribution increases the gamma ray energy flux on the axis (polar angle of zero degrees), the increase in direct radiation resulting from spatial averaging is not surprising, nor is the decrease in scattered radiation originating in the aluminum tank wall and insulation. However, the scattering in  $\text{LH}_2$  and  $\text{GH}_2$  increases as a result of flattening the spatial distribution. Thus, small-angle scatter occurring from gamma rays leaking from the top of the reactor may be more important than the scatter of gamma rays hitting the tank at large polar angles in the particular configuration examined. However, the importance of leakage from various parts of the reactor with respect to scattering depends on tank geometry, reactor and shield configuration, and propellant depth. Hence, conclusions reached here should not be too broadly generalized. One can conclude an explicit representation of the meridian ring spatial distribution is needed. Fortunately, the modification to the KAP-VI program permitting punched output to be obtained in a form required by SCAP eliminates any need to perform spatial averaging of meridian ring data.

#### 5.1.4 Location of Point Source of Incident Radiation

A special problem arises when attempting to apply SCAP to the calculation of direct and scattered dose rates from nozzle sources. These sources, defined in the October 1969 CRAM, consist of 22 separate point sources with locations specified by a value of  $R$  and  $Z$ , corresponding to positions along the nozzle. It was desirable, from the standpoint of computer running economy, to take advantage of the symmetry of the nozzle sources by defining the point source of incident radiation on the axis of the nozzle. However, this point does not lie within the nozzle source volume. The validity of the "effective" point source approximation is subject to some question under these circumstances. If one assumes that the effective point source representing nozzle sources can be placed on the nozzle axis, the question of where to place the source then arises. Consider that the nozzle is about 400 cm long with several decades of variation in point source strength along the nozzle.

An attempt to resolve these questions was made by first employing the KAP-VI program to generate data both on a meridian ring centered at the core center and on a

TABLE 5-3  
EFFECT OF USING SPATIALLY AVERAGED MERIDIAN RING GAMMA RAY FLUXES

| IN SCAP SCATTERING ANALYSIS CODE<br>(GAMMA RAY DOSE RATE, Rem/hr) |                            |                            |                       |                     |                  |                   |
|-------------------------------------------------------------------|----------------------------|----------------------------|-----------------------|---------------------|------------------|-------------------|
| Type of<br>Meridian Ring<br>Data                                  | LH <sub>2</sub><br>Scatter | GH <sub>2</sub><br>Scatter | Insulation<br>Scatter | Aluminum<br>Scatter | Total<br>Scatter | Direct<br>Scatter |
| Explicit<br>(from KAP-V)                                          | 4.063(3)                   | 9.529(1)                   | 2.566(2)              | 8.140(2)            | 5.229(3)         | 2.696(3)          |
| Spatially<br>Averaged in Each<br>Energy Group                     | 5.170(3)                   | 1.504(2)                   | 1.909(2)              | 6.230(2)            | 6.134(3)         | 5.332(3)          |
|                                                                   |                            |                            |                       |                     |                  | 1.760(4)          |

- Notes: 1) Detector point located at the top of the 60° tank.  
 2) LH<sub>2</sub> propellant depth = 570 cm.  
 3) 30 foot meridian ring gamma ray fluxes used in SCAP.  
 4) Numbers in parentheses refer to powers of ten.

meridian ring centered at  $Z = -205.34$  cm, i. e., at the geometrical center of the nozzle. These meridian ring data were then employed in the SCAP code to calculate scattered and direct radiation at the top of the 75-degree propellant tank. Another approach consisted of utilizing the asymmetry (azimuthal angle incrementation) option available in SCAP to compute dose rates from the nozzle sources directly. Explicit representation of nozzle sources in SCAP was accomplished by inputting the point source data as meridian ring data on a 1 cm radius meridian ring. Instead of multiplying each source strength by  $1/4\pi$ , this scale factor was input and the calculation performed internal to SCAP.

Large variations in the results obtained by the several methods are shown in Table 5-4. In Table 5-4, the SCAP code explicit calculation and the KAP-VI code calculation of direct dose rate are within a few percent of each other. However, the direct radiation obtained from the use of axis-centered, meridian ring data in SCAP is much lower. These direct dose rates are incorrect. This will become apparent if one considers that SCAP uses the gamma ray flux at a polar angle of zero degrees (in the completely symmetric condition) and extrapolates these data to the top of the propellant tank. However, because the nozzle sources are displaced from the axis of the system, direct radiation between the nozzle and a detector point at the top of the tank is recorded on the meridian ring at some angle other than zero degrees. A large error is incurred in utilizing the zero degree data because of the severe depression in meridian ring fluxes caused by the shielding in the reactor.

Differences in scattered radiation resulting from using either set of meridian ring data or an explicit representation of the nozzle sources directly in SCAP were somewhat surprisingly small. One might intuitively expect that an explicit representation of the nozzle sources in SCAP would yield a higher value of total scattered radiation than the use of meridian ring data in SCAP. That this is not the case may possibly be attributed to use of fewer source polar angle rays in any given azimuthal plane compared to the number of rays used in the single plane employed where symmetry exists. The SCAP code is considerably more costly to run for the nonsymmetric condition which exists in explicit representation of the nozzle sources. In view of this and the results of the study summarized in Table 5-4, all calculations for the nozzle sources include a SCAP calculation of scattered dose rate based

TABLE 5-4  
COMPARISON OF SEVERAL METHODS FOR CALCULATING TANK TOP DOSE RATE FROM NOZZLE SOURCES

|                                                                                                                | Rem/Hr                 |               |                                |
|----------------------------------------------------------------------------------------------------------------|------------------------|---------------|--------------------------------|
|                                                                                                                | <u>Total Scattered</u> | <u>Direct</u> | <u>Direct<br/>Plus Scatter</u> |
| A. Explicit representation of nozzle sources* in SCAP                                                          | 1.35(3)                | 8.71(2)       | 2.22(3)                        |
| B. KAP-SCAP, 30 foot meridian ring centered at the center of the core                                          | 1.95(3)                | 1.42(2)       | 2.09(3)                        |
| C. KAP-SCAP, 30 foot meridian ring centered at $Z = -205.34$ cm, i.e., at the geometrical center of the nozzle | 1.96(3)                | 4.20(1)       | 2.00(3)                        |
| D. KAP-SCAP (as in C) calculation of scatter, KAP calculation of direct                                        | 1.96(3)                | 9.30(2)       | 2.89(3)                        |

\*Nozzle sources from October CRAM.

Notes: 1) Detector point located at the top of the 75° tank.

2) Numbers in parentheses refer to powers of ten.

on a 30-foot meridian ring centered at  $Z = -205.34$  and a KAP-VI calculation of the direct radiation.

#### 5.1.5 KAP-SCAP versus MAP-SCAP

For several years, the discrete ordinate ( $S_n$ ) technique has been recognized as a method for performing reactor nuclear and shielding calculations when greater accuracy (over more conventional methods) is required. The two-dimensional, discrete ordinate code, DOT-IIW, together with the data processing code, NAGS, forms the basis for this analysis technique. These codes were used to calculate neutron and gamma ray sources within the nuclear subsystem for subsequent use in the KAP-VI point kernel code. The discrete ordinate method is the method used to calculate the angular distributions of neutron and gamma ray fluxes. The method is capable of meeting the severe accuracy requirements of neutron energy spectrum distributions essential to an accurate calculation of gamma ray source strengths and distributions.

The application of the discrete ordinate method to deep penetration shielding problems and, in particular, problems involving the transport of radiation over large distance (such as might be encountered in a reactor and tank geometrical configuration) has been the subject of considerable interest at WANL. Methods development effort in this area has resulted in the development of the linked DOT-MAP codes. The MAP code processes the surface, angular-leakage fluxes calculated by DOT-IIW and transports this radiation to arbitrarily selected points in space. One technique consists of employing MAP to calculate the angular- and energy-distribution of neutrons and gamma radiation at the bottom surface of the propellant tank. These data are subsequently input into other DOT-IIW problems in which the radiation is transported to the top of the propellant tank.

Another interesting procedure consists of calculating meridian ring data using the DOT-MAP technique and inputting the scalar fluxes thus obtained into the SCAP code for subsequent transport to the top of the tank. A comparison of gamma ray energy fluxes calculated on the meridian ring by the KAP-VI code and the DOT-MAP code is shown in Figure 5-1. A versatile feature of the DOT-MAP technique is apparent here in that the radiation on the meridian ring may be separated into a reactor top leakage and reactor side

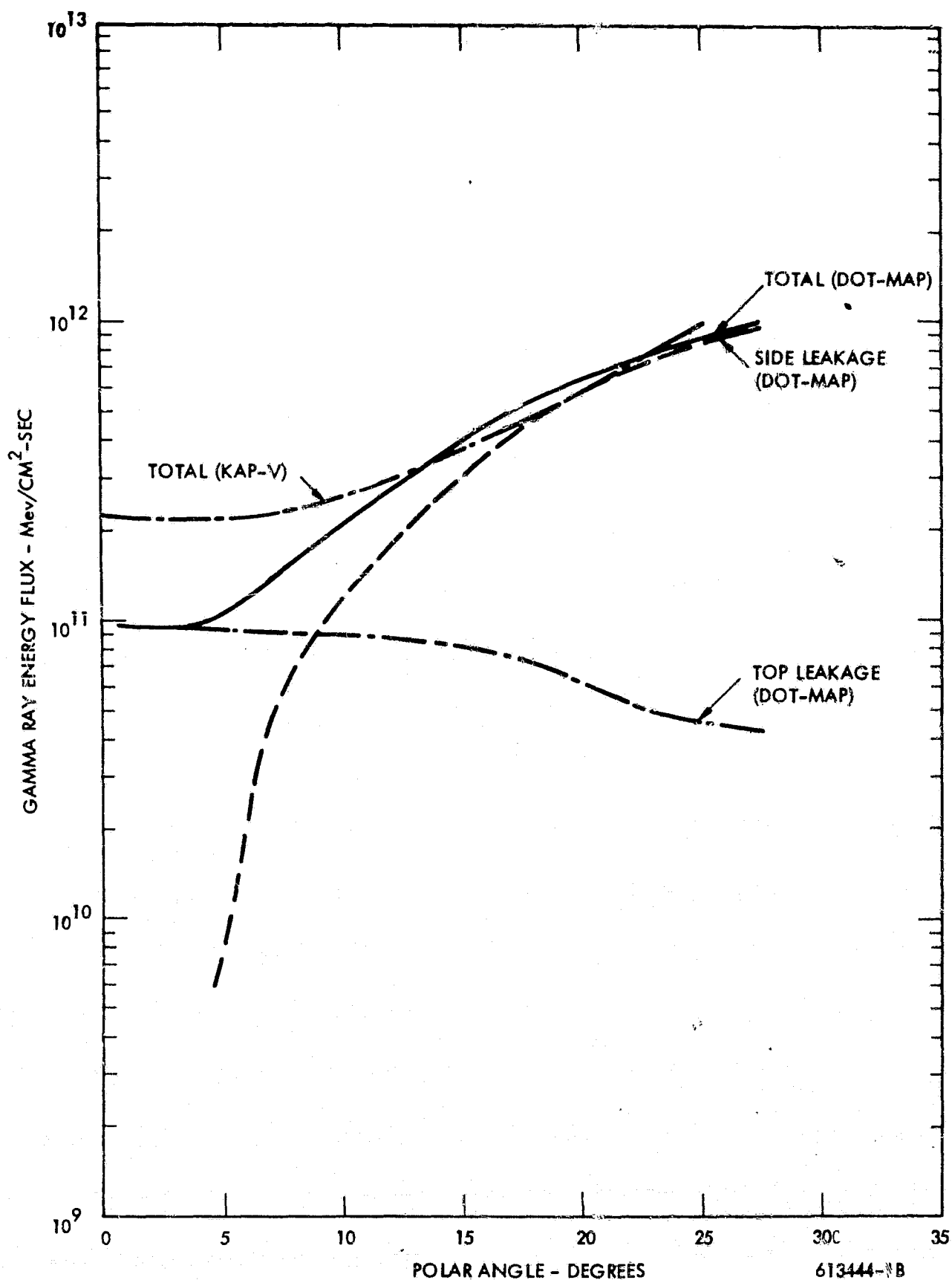


Figure 5-1. Comparison of DOT-MAP vs KAP-VI Gamma Ray Flux on a 30-Foot Meridian Ring

leakage component. Close agreement between the DOT-MAP data and the KAP data is obtained at large polar angles. The DOT-MAP results indicate that at large polar angles, side leakage constitutes most of the radiation. The DOT-MAP results indicate that at large polar angles, side leakage constitutes most of the radiation. The DOT-MAP data on the axis (zero degree polar angle) are more than a factor of 2 lower than the KAP-VI data.

The difference in the average energy spectra calculation by DOT-MAP versus KAP-VI (averaged over the polar angle of 20.884 degrees) is shown in Figure 5-2. The KAP-VI data are recognized as not being a true spectrum because the buildup of low energy gamma rays through energy degradation from higher groups is taken into account by applying a buildup factor to the original energy group. Retention of the degraded energy in the original group through use of a dose buildup factor is a reasonable approximation for calculating the gamma ray dose. However, interpretation of this gamma ray response data as an energy spectrum distribution is not valid. What is obtained is, at best, described as a pseudo-spectrum.

On the other hand, the MAP code yields a gamma ray energy spectrum which may be interpreted as a true spectrum. In view of the foregoing discussion, the difference in average gamma distributions shown in Figure 5-2 appears to be reasonable.

The result of the SCAP calculations of scattered dose rate at the top of the 75-degree propellant tank using both spectra are shown in Figure 5-3. At small propellant depths, the SCAP-MAP data are up to 30 percent higher than the KAP-SCAP data. However, the curves cross at about 340 cm propellant depth with the result that for a full tank, the KAP-SCAP data are about a factor of 2 greater than the MAP-SCAP totals. The relatively large amount of gamma ray energy in the lower energy groups from MAP as compared to KAP may be responsible in part for the MAP-SCAP data being greater than KAP-SCAP data at the low propellant depth. At greater propellant depths, the increased thickness of hydrogen selectively attenuates the low energy portion of the spectrum. Hence, it appears likely that the scattering of high energy gamma rays is relatively more important than low energy gamma rays at large propellant depths. (We saw that the KAP-VI pseudo-spectrum includes more high energy gamma rays than does the MAP spectrum.) The low values of gamma ray energy flux predicted by

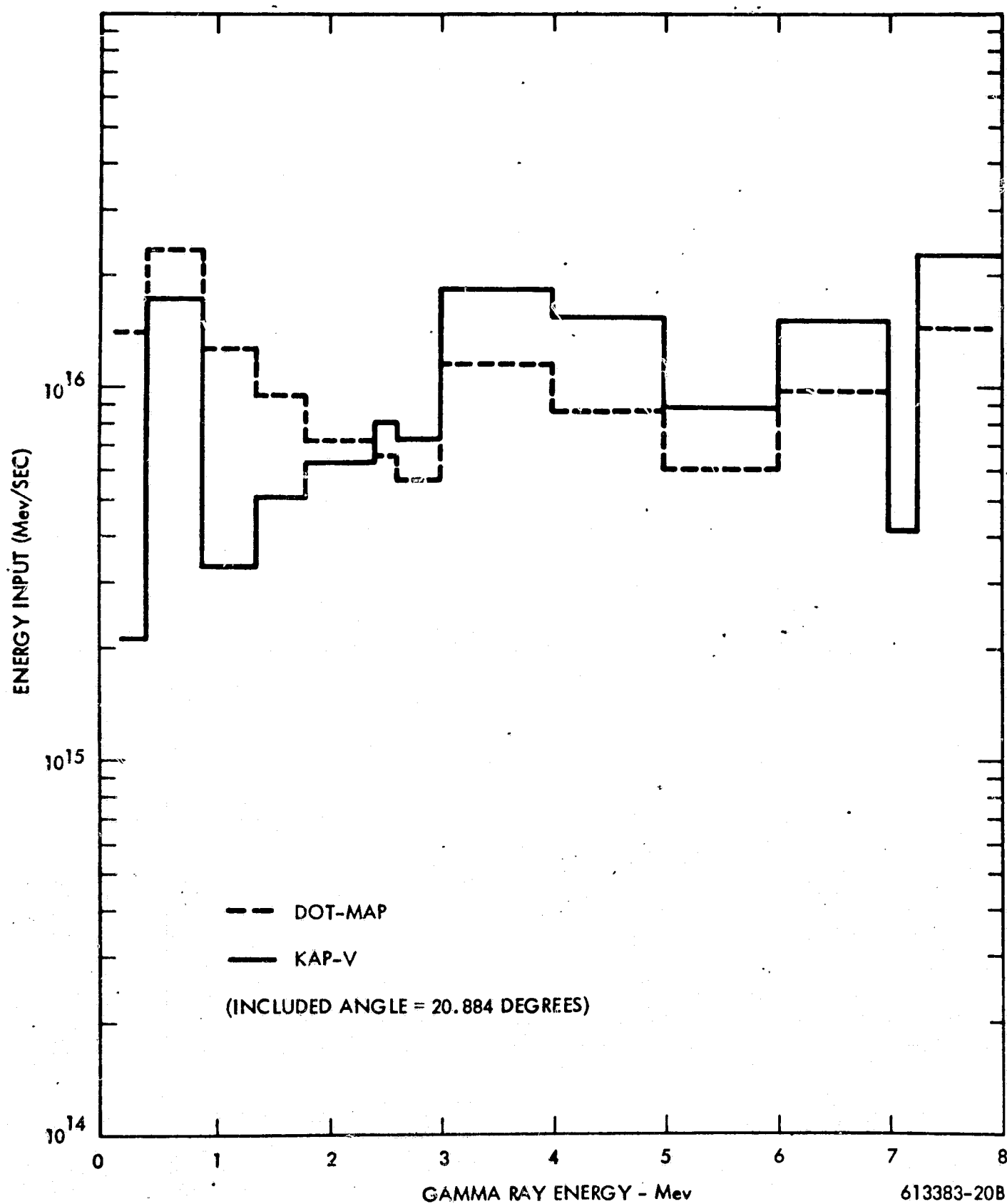


Figure 5-2 Comparison of Total Energy Input KAP-VI vs DOT-MAP

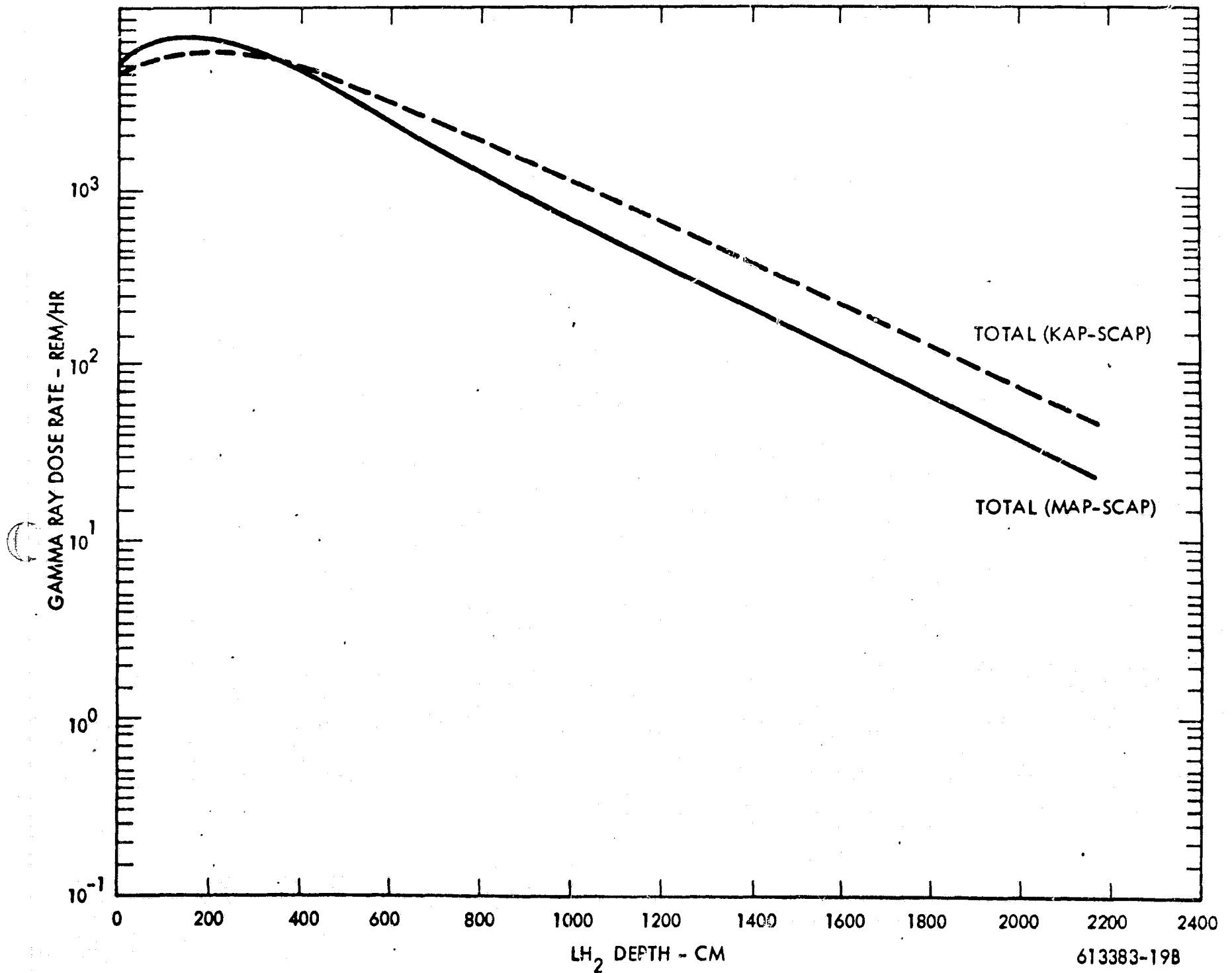


Figure 5-3 Comparison of 75 Degree Tank Top Scattered Gamma Ray Dose Rate vs. LH<sub>2</sub> Depth Obtained Using KAP-VI and DOT-MAP Meridian Ring Fluxes



MAP at locations near the reactor axis is undoubtedly partly responsible for the low values of MAP-SCAP data at propellant depths greater than 340 cm.

## 5.2 Checkout of the Revised MSFC Code Package

Under the work scope of Task 6 of this contract, a neutron and photon cross section library (described in Volume 2) and a revised MSFC code package (described in Volumes 3 through 6) are provided as a complete package to analyze the radiation environment around a nuclear powered rocket reactor. As a final checkout of the cross section Master Libraries and the complete MSFC code package, WANL has performed calculations on an  $r,z$  geometry, realistic, nuclear rocket model. This model includes a  $B_4C$ -Ti  $H_2$ -Al (BATH) shielded, beryllium reflected reactor and a 75-degree conical propellant tank. The detailed model and atom densities for this calculation were mutually agreed upon by WANL and Mr. H. E. Stern, of MSFC, the technical monitor of this contract.

### 5.2.1 Description of $r,z$ Geometry, Realistic, Nuclear Rocket Model

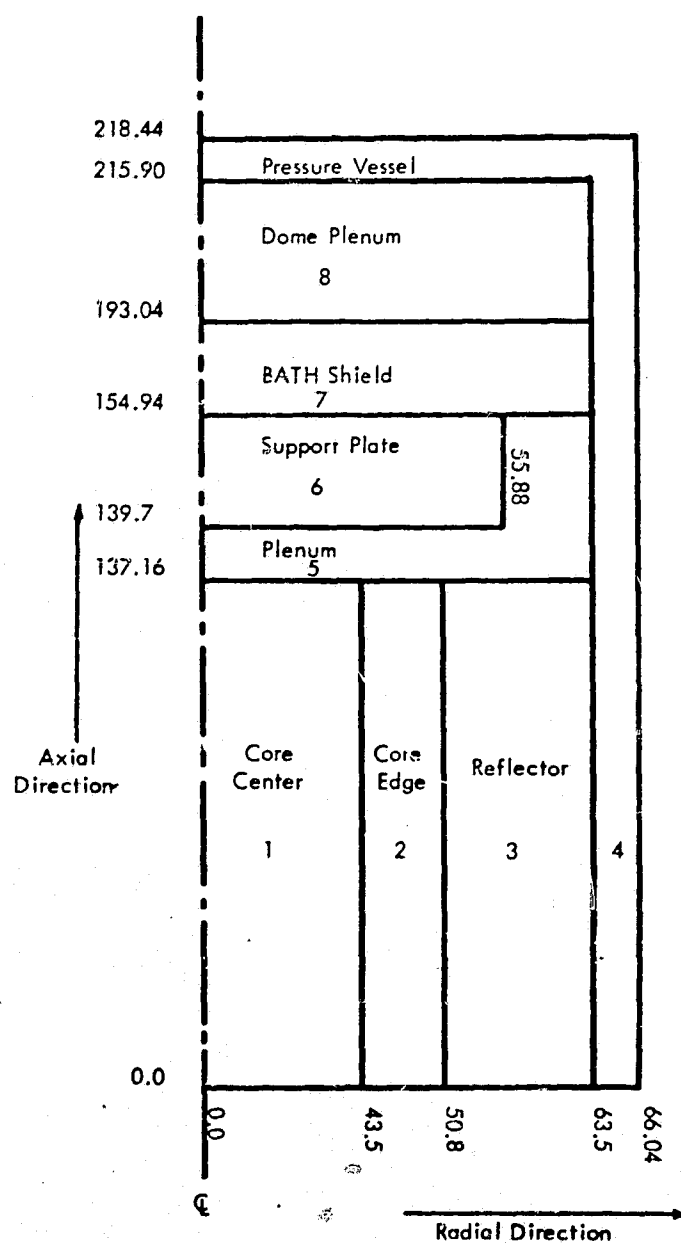
Figure 5-4 describes the basic geometry for the realistic, nuclear rocket model, which includes a 107 gram per  $cm^2$  BATH shield. The reactor core is separated into two regions which are designated as "core center" and "core edge." These core regions have the same material composition and differ only in the multigroup, spectrally weighted, cross section data used in each region. The radial and axial mesh coordinates for the MSFC sample problem are detailed in Tables 5-5 and 5-6, respectively. These mesh coordinates are used in the discrete ordinates transport codes.

Table 5-7 defines the element atom densities for each of the eight regions of the model. Hydrogen is assumed at para-ortho equilibrium at ambient temperature (25% para--75% ortho hydrogen at 300°K. <sup>(14)</sup>

### 5.2.2 Description of Method of Analysis

An integrated approach to nuclear radiation analysis requires an efficient and effective interchange of compatible data between computer codes. This compatibility requires the consistent use of neutron and photon cross section data throughout the code package and the preparation of output data by one computer code for direct input into succeeding codes. This interchange of data is most efficiently accomplished by use of

The Bottom of the Propellant  
Tank is at Z = 583.0 centimeters



- Notes:
- 1) All dimensions are in centimeters,
  - 2) Drawing is not to scale, and
  - 3) N- Indicates composition number.

Figure 5-4. Basic Geometry for MSFC Realistic Model Sample Problem

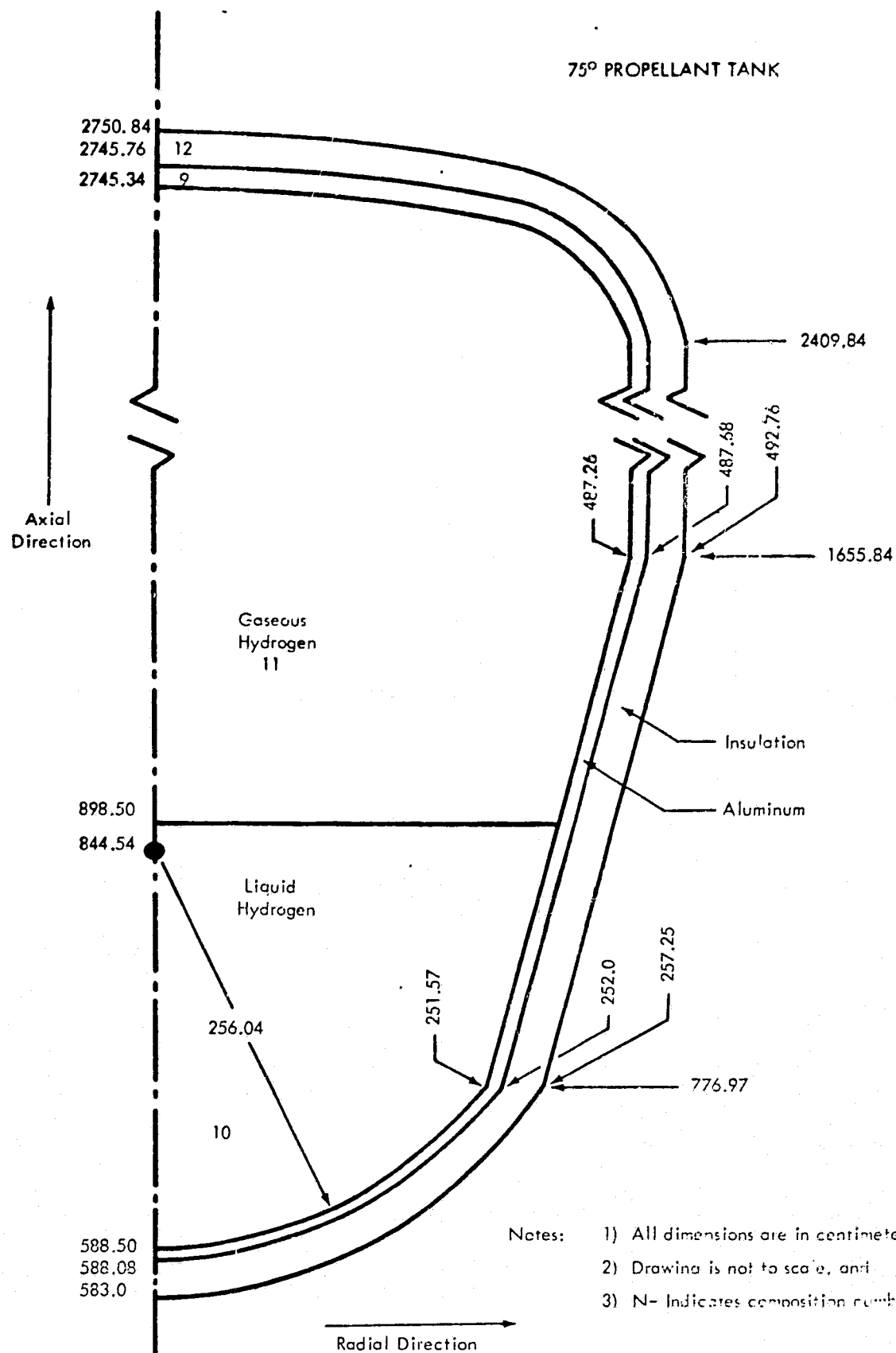


Figure 5-4. Basic Geometry for MSFC Realistic Model Sample Problem

TABLE 5-5RADIAL MESH COORDINATES FOR MSFC REALISTIC MODEL SAMPLE PROBLEM

| <u>Radial Mesh<br/>Line No.</u> | <u>R(cm.)</u> |
|---------------------------------|---------------|
| 1                               | 0.0           |
| 2                               | 7.0           |
| 3                               | 12.0          |
| 4                               | 16.0          |
| 5                               | 20.0          |
| 6                               | 24.0          |
| 7                               | 28.0          |
| 8                               | 32.0          |
| 9                               | 35.0          |
| 10                              | 38.0          |
| 11                              | 40.0          |
| 12                              | 42.0          |
| 13                              | 43.5          |
| 14                              | 45.0          |
| 15                              | 46.0          |
| 16                              | 47.0          |
| 17                              | 47.75         |
| 18                              | 48.50         |
| 19                              | 49.0          |
| 20                              | 49.5          |
| 21                              | 50.0          |
| 22                              | 50.4          |
| 23                              | 50.8          |
| 24                              | 51.3          |
| 25                              | 52.0          |
| 26                              | 53.0          |
| 27                              | 54.5          |
| 28                              | 55.88         |
| 29                              | 57.5          |
| 30                              | 59.0          |
| 31                              | 60.5          |
| 32                              | 62.0          |
| 33                              | 63.5          |
| 34                              | 65.0          |
| 35                              | 66.04         |

TABLE 5-6

AXIAL MESH COORDINATES FOR MSFC REALISTIC MODEL SAMPLE PROBLEM

| <u>Axial Mesh<br/>Line No.</u> | <u>Z (cm.)</u> | <u>Axial Mesh<br/>Line No.</u> | <u>Z (cm.)</u> |
|--------------------------------|----------------|--------------------------------|----------------|
| 1                              | 0.0            | 41                             | 155.59         |
| 2                              | 1.0            | 42                             | 155.72         |
| 3                              | 2.5            | 43                             | 155.85         |
| 4                              | 3.5            | 44                             | 156.04         |
| 5                              | 6.0            | 45                             | 156.04         |
| 6                              | 9.0            | 46                             | 157.04         |
| 7                              | 14.0           | 47                             | 158.14         |
| 8                              | 20.0           | 48                             | 159.24         |
| 9                              | 27.0           | 49                             | 160.74         |
| 10                             | 34.0           | 50                             | 162.24         |
| 11                             | 41.0           | 51                             | 163.74         |
| 12                             | 48.0           | 52                             | 165.24         |
| 13                             | 55.0           | 53                             | 167.24         |
| 14                             | 62.0           | 54                             | 169.24         |
| 15                             | 69.0           | 55                             | 171.24         |
| 16                             | 76.0           | 56                             | 173.24         |
| 17                             | 83.0           | 57                             | 175.24         |
| 18                             | 90.0           | 58                             | 177.24         |
| 19                             | 97.0           | 59                             | 179.24         |
| 20                             | 104.0          | 60                             | 181.24         |
| 21                             | 111.0          | 61                             | 183.24         |
| 22                             | 118.0          | 62                             | 185.24         |
| 23                             | 125.0          | 63                             | 186.74         |
| 24                             | 130.0          | 64                             | 188.24         |
| 25                             | 133.0          | 65                             | 189.24         |
| 26                             | 135.0          | 66                             | 190.24         |
| 27                             | 136.3          | 67                             | 190.94         |
| 28                             | 137.16         | 68                             | 191.64         |
| 29                             | 138.0          | 69                             | 192.04         |
| 30                             | 139.0          | 70                             | 192.24         |
| 31                             | 139.7          | 71                             | 192.44         |
| 32                             | 140.0          | 72                             | 192.64         |
| 33                             | 143.9          | 73                             | 192.84         |
| 34                             | 147.9          | 74                             | 192.94         |
| 35                             | 151.9          | 75                             | 193.04         |
| 36                             | 154.94         | 76                             | 198.0          |
| 37                             | 155.07         | 77                             | 204.0          |
| 38                             | 155.20         | 78                             | 210.0          |
| 39                             | 155.33         | 79                             | 215.9          |
| 40                             | 155.46         | 80                             | 217.5          |
|                                |                | 81                             | 218.44         |

**TABLE 5-7**
**ELEMENT ATOM DENSITIES FOR MSFC REALISTIC MODEL SAMPLE PROBLEM**

(atoms/barn-cm)

| Composition<br>Number | Composition<br>Description | Aluminum  | Beryllium | Natural<br>Boron | Carbon    | Chromium | Para<br>Hydrogen<br>(300°K)** | Ortho<br>Hydrogen<br>(300°K)** | Iron     | Nickel  | Niobium | Titanium  | <sup>235</sup> U | <sup>238</sup> U |
|-----------------------|----------------------------|-----------|-----------|------------------|-----------|----------|-------------------------------|--------------------------------|----------|---------|---------|-----------|------------------|------------------|
| 1                     | Core Center                | ---       | ---       | ---              | 7.54(-2)  | 4.5(-6)  | 8.025(-5)                     | 2.4075(-4)                     | 2.91(-5) | 5.4(-5) | 2.0(-4) | ---       | 1.74(-4)         | 1.4(-5)          |
| 2                     | Core Edge                  | ---       | ---       | ---              | 7.54(-2)  | 4.5(-6)  | 8.025(-5)                     | 2.4075(-4)                     | 2.91(-5) | 5.4(-5) | 2.0(-4) | ---       | 1.74(-4)         | 1.4(-5)          |
| 3                     | Reflector                  | ---       | 1.106(-1) | ---              | ---       | ---      | 2.09(-4)                      | 6.27(-4)                       | ---      | ---     | ---     | ---       | ---              | ---              |
| 4                     | Pressure Vessel            | 6.25(-2)* | ---       | ---              | ---       | ---      | ---                           | ---                            | ---      | ---     | ---     | ---       | ---              | ---              |
| 5                     | Plenum                     | ---       | ---       | ---              | ---       | ---      | 9.375(-3)                     | 2.8125(-2)                     | ---      | ---     | ---     | ---       | ---              | ---              |
| 6                     | Support Plate              | 5.0(-2)   | ---       | ---              | ---       | ---      | 1.8825(-3)                    | 5.6475(-3)                     | ---      | ---     | ---     | ---       | ---              | ---              |
| 7                     | BATH Shield                | 3.997(-2) | ---       | 6.125(-3)        | 1.530(-3) | ---      | 4.8725(-3)                    | 1.46175(-2)                    | ---      | ---     | ---     | 1.071(-2) | ---              | ---              |
| 8                     | Dome Plenum                | ---       | ---       | ---              | ---       | ---      | 1.7625(-3)                    | 5.2875(-3)                     | ---      | ---     | ---     | ---       | ---              | ---              |
| 9                     | Tank Wall                  | 6.006(-2) | ---       | ---              | ---       | ---      | ---                           | ---                            | ---      | ---     | ---     | ---       | ---              | ---              |
| 10                    | Liquid Hydrogen            | ---       | ---       | ---              | ---       | ---      | 4.168(-2) <sup>†</sup>        | ---                            | ---      | ---     | ---     | ---       | ---              | ---              |
| 11                    | Gaseous Hydrogen           | ---       | ---       | ---              | ---       | ---      | 5.24(-4) <sup>†</sup>         | ---                            | ---      | ---     | ---     | ---       | ---              | ---              |
| 12                    | Insulation                 | ---       | ---       | ---              | 2.748(-3) | ---      | 1.435(-3) <sup>†</sup>        | ---                            | ---      | ---     | ---     | ---       | ---              | ---              |

\*Numbers in parentheses indicate powers of 10.

\*\*Hydrogen is assumed to be at room temperature (300°K) and at equilibrium (25% para- 75% ortho-hydrogen).

†Natural hydrogen.

magnetic tapes and/or punched cards prepared in a format directly usable in succeeding analyses.

The flow chart shown in Figure 5-5 defines the calculational procedure followed in the analysis of the realistic model problem. This procedure illustrates the compatibility of codes provided to MSFC under this contract and, specifically, the maximum use of magnetic tape and punched data card output. The SATURN, ANISN-W, APPROPOS, DOT-IIW, NAGS, KAP-VI, SCAP and MAP codes, as modified and developed by WANL and supplied to MSFC, were used in this analysis. Table 5-8 tabulates the calculations used in this analysis.

The SATURN cross section preparation code (described in Volume 3) was used to prepare 52 group, full  $P_1$ , macroscopic neutron cross sections, with upscatter, for the radial and axial regions of the realistic model under analysis. The atom densities specified in Table 5-7 and the full  $P_1$  microscopic cross section data specified in Table 5-9 were input to the SATURN code. The resultant 52 group, macroscopic, cross section data is placed by SATURN on a normal, group-dependent data tape and a specially prepared, group-independent tape.

Using the SATURN-produced, cross section data and the reactor model described in Figure 5-4, radial and axial ANISN-W, one-dimensional, fixed distributed source,  $S_4$  transport calculations were performed. These calculations generated the 52 group neutron spectra by region for input to the APPROPOS code. The APPROPOS data processing technique was then used to produce spectrally weighted cross section data in a reduced number of energy groups.

The APPROPOS computer code requires as input:

- 1) 52 group, microscopic, neutron cross section data;
- 2) 52 group, neutron reaction rate, cross section data, (i. e., radiative capture, fission, inelastic scatter and elastic scatter cross sections);
- 3) gamma ray production data;
- 4) 13 group, photon cross sections, produced by the GAMLEG-W code;
- 5) 52 group, regionwise spectra as calculated by ANISN-W; and
- 6) element atom density data.

For each region of the model under analysis, the APPROPOS calculation produced, on magnetic tape, 29 energy-group, coupled, neutron-photon; 16 group neutron; and 13 group photon,

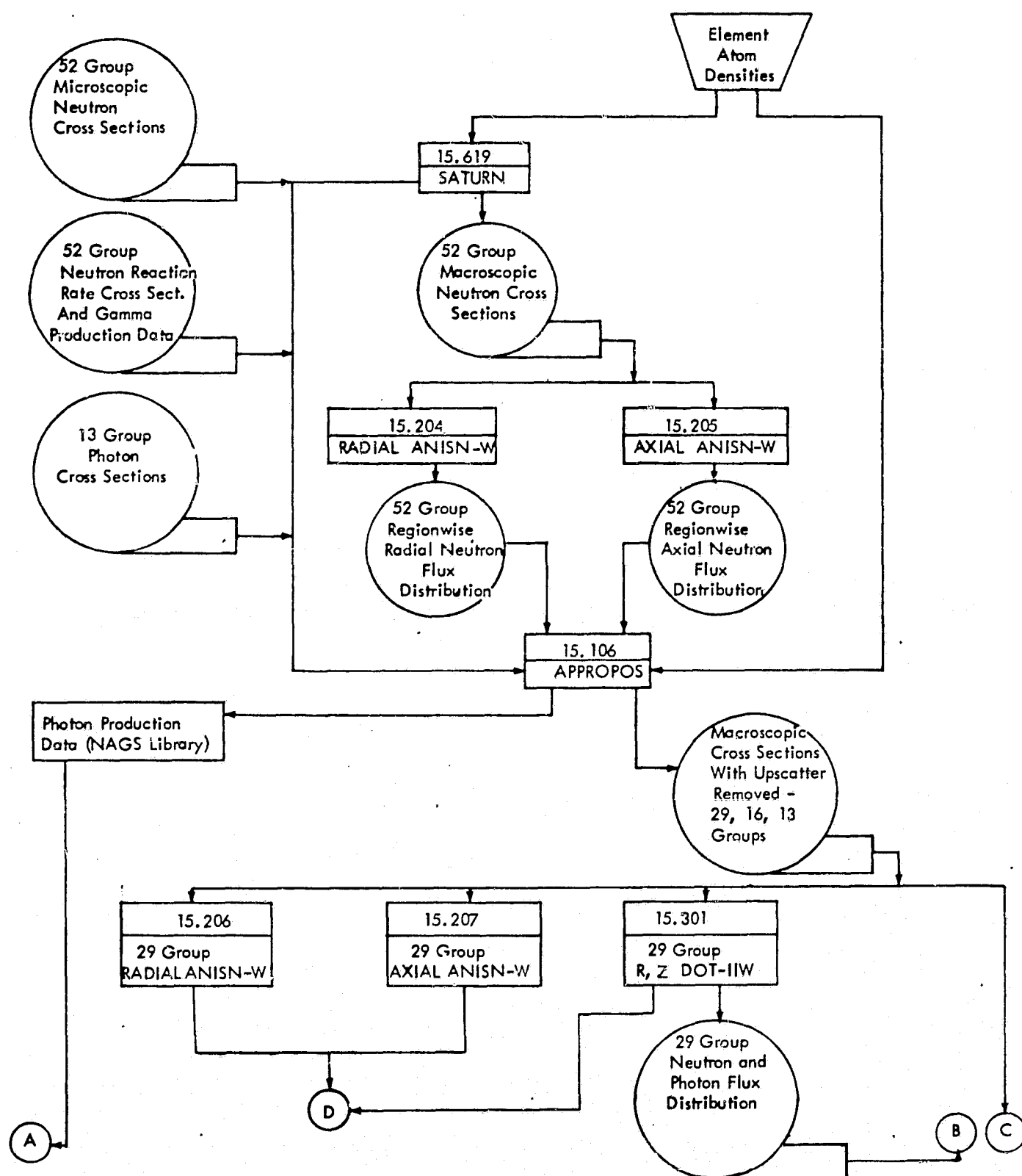


FIGURE 5-5. FLOW CHART FOR RADIATION ANALYSIS OF MSFC REALISTIC MODEL SAMPLE PROBLEM

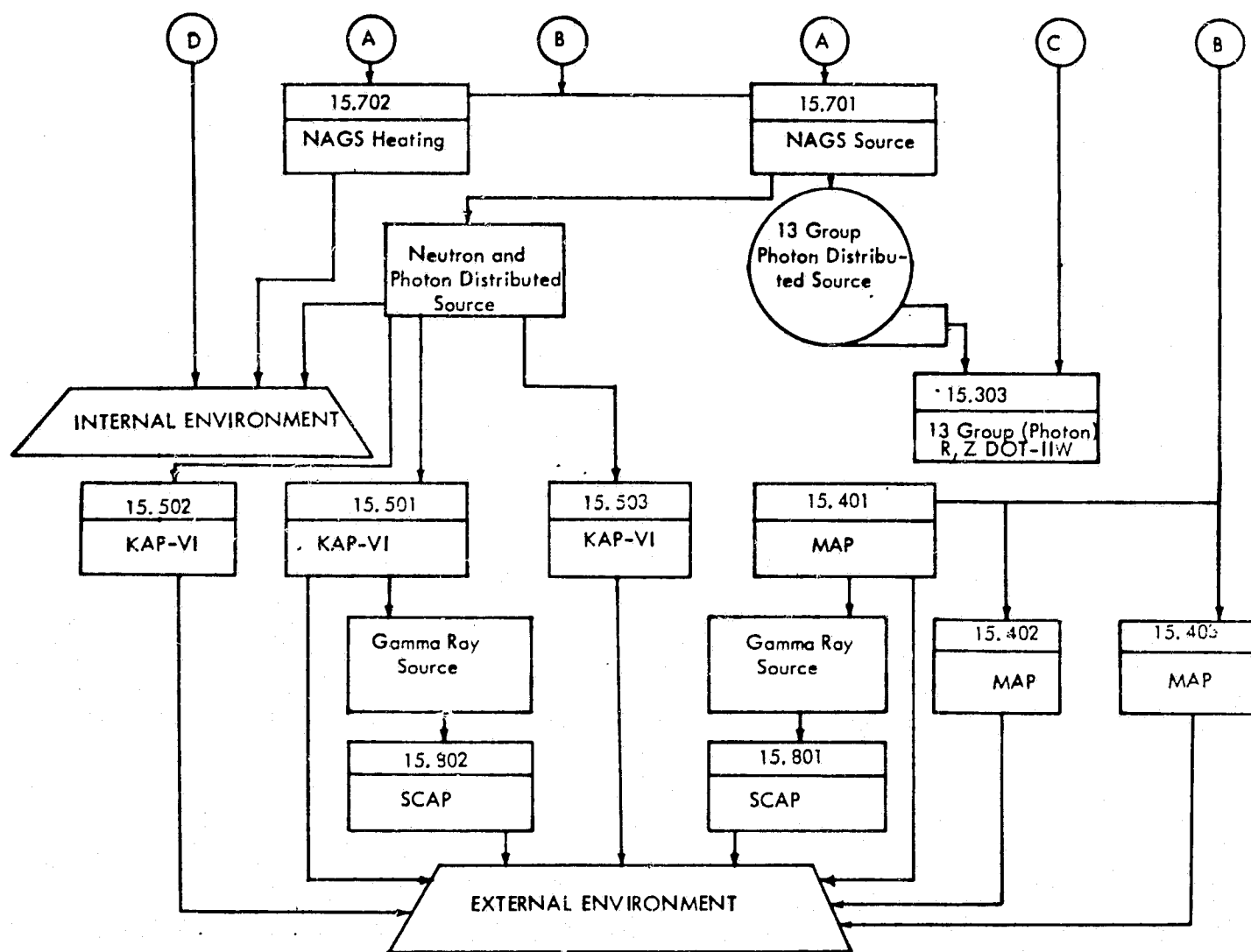


FIGURE 5-5. FLOW CHART FOR RADIATION ANALYSIS OF MSFC REALISTIC MODEL SAMPLE PROBLEM

TABLE 5-8

TABULATION OF THE CALCULATIONS IN THE REALISTIC MODEL ANALYSIS

| <u>Problem ID<br/>Number</u> | <u>Code</u> | <u>Description</u>                                                          |
|------------------------------|-------------|-----------------------------------------------------------------------------|
| 15. 619                      | SATURN      | Generate macroscopic, 52 group, $P_1$ cross sections                        |
| 15. 204                      | ANISN-W     | 52 group, radial calculation to generate spectra                            |
| 15. 205                      | ANISN-W     | 52 group, axial calculation to generate spectra                             |
| 15. 156                      | APPROPOS    | Place the Master Library No. 6 data behind<br>Master Library No. 3          |
| 15. 106                      | APPROPOS    | Generate coupled, neutron-photon, neutron,<br>and photon cross section data |
| 15. 206                      | ANISN-W     | Radial, 29 group, typical design calculation                                |
| 15. 207                      | ANISN-W     | Axial, 29 group, typical design calculation                                 |
| 15. 301                      | DOT-IIW     | 29 group, r,z geometry, fixed source calculation                            |
| 15. 701                      | NAGS        | Source calculation, and plotting                                            |
| 15. 702                      | NAGS        | Heating calculation, and plotting                                           |
| 15. 303                      | DOT-IIW     | 13 group, r,z geometry, gamma ray calculation                               |
| 15. 501                      | KAP-VI      | Meridian ring calculation                                                   |
| 15. 502                      | KAP-VI      | Common plane calculation                                                    |
| 15. 503                      | KAP-VI      | Tank top calculation                                                        |
| 15. 401                      | MAP         | Meridian ring calculation without interpolation                             |
| 15. 402                      | MAP         | Common plane calculation without interpolation                              |
| 15. 403                      | MAP         | Tank top calculation without interpolation                                  |
| 15. 404                      | MAP         | Meridian ring calculation with interpolation                                |
| 15. 801                      | SCAP        | Tank top calculation with MAP card input data                               |
| 15. 802                      | SCAP        | Tank top calculation with KAP-VI card input<br>data                         |

TABLE 5-9  
IDENTIFICATION NUMBERS OF FULL - P1 MICROSCOPIC CROSS SECTION LIBRARY DATA  
FOR MSFC REALISTIC MODEL SAMPLE PROBLEM

| Composi-<br>tion<br>Number | Composition<br>Description | Aluminum | Beryllium | Natural<br>Boron | Carbon | Chromium | Para Hydro-<br>gen (300° K) | Ortho Hydro-<br>gen (300° K) | Iron   | Nickel | Niobium | Titanium | Uranium-<br>235 | Uranium-<br>238 |
|----------------------------|----------------------------|----------|-----------|------------------|--------|----------|-----------------------------|------------------------------|--------|--------|---------|----------|-----------------|-----------------|
| 1                          | Core Center                | ---      | ---       | ---              | 0610-1 | 0710-1   | 1410-1                      | 1510-1                       | 1810-1 | 2610-1 | 2710-1  | ---      | 3510-1          | 3610-1          |
| 2                          | Core Edge                  | ---      | ---       | ---              | 0820-1 | 0720-1   | 1420-1                      | 1520-1                       | 1820-1 | 2620-1 | 2720-1  | ---      | 3520-1          | 3620-1          |
| 3                          | Reflector                  | ---      | 0230-1    | ---              | ---    | ---      | 1430-1                      | 1530-1                       | ---    | ---    | ---     | ---      | ---             | ---             |
| 4                          | Pressure Vessel            | 0130-1   | ---       | ---              | ---    | ---      | ---                         | ---                          | ---    | ---    | ---     | ---      | ---             | ---             |
| 5                          | Plenum                     | ---      | ---       | ---              | ---    | ---      | 1430-1                      | 1530-1                       | ---    | ---    | ---     | ---      | ---             | ---             |
| 6                          | Support Plate              | 0130-1   | ---       | ---              | ---    | ---      | 1430-1                      | 1530-1                       | ---    | ---    | ---     | ---      | ---             | ---             |
| 7                          | BATH Shield                | 0140-1   | ---       | 0340-1           | 0640-1 | ---      | 1440-1                      | 1540-1                       | ---    | ---    | ---     | 3340-1   | ---             | ---             |
| 8                          | Dome Plenum                | ---      | ---       | ---              | ---    | ---      | 1430-1                      | 1530-1                       | ---    | ---    | ---     | ---      | ---             | ---             |

full  $P_1$ , macroscopic cross sections. Upscatter for the neutron data was removed by reaction rate balance. Photon production data, by element, and selected nuclear parameters were punched on data cards. These cards are required as input to the NAGS data processing code.

Using these 29 energy-group, full  $P_1$ , macroscopic, cross section data and the reactor model defined in Figure 5-4, radial and axial, simultaneous neutron-photon,  $S_4$ , fixed distributed source, ANISN calculations were performed. The same APPROPOS-produced 29 group cross section data were then used in a simultaneous, neutron-photon,  $r,z$ , DOT-IIW calculation in which the calculated two-dimensional, neutron and photon flux distributions were output on magnetic tape. The fixed-source input to the ANISN-W and DOT-IIW calculations, varied as the fission spectrum by energy group, with a uniform radial, and a cosine axial distribution in the core. The source was zero in all other regions.

The NAGS code was used in this sample analysis to provide neutron and photon source distributions for direct input into the KAP-VI point kernel code. As input, NAGS uses the two-dimensional  $r,z$  flux distribution calculated by DOT-IIW and the photon production data output on punched cards from APPROPOS. A 13 group, photon, DOT-IIW calculation was then performed, using the photon distributed source calculated by NAGS. The purpose of the photon DOT-IIW calculation was to verify the APPROPOS coupling calculation, the 29 group, DOT-IIW calculation and the NAGS source calculation. A sample NAGS heating calculation was also performed.

The external environment to the nuclear subsystem was calculated using the KAP-VI, MAP, and SCAP codes. For this analysis, calculations were performed along a meridian ring, whose radius is 914.4 cm (or 30 feet) centered about  $z = 69.0$  cm, the core center. Additional calculations were performed along a common radial plane separating the nuclear subsystem from the propellant tank. This plane is located at  $z = 583.0$  cm, the bottom of the propellant tank. Calculations were also performed along a radial plane at the top of the propellant tank at  $z = 2752$  cm.

Distributed neutron and gamma ray sources punched from the NAGS code were input to the three KAP-VI calculations. Both neutron and gamma ray dose rates (as well as other responses) were obtained. The tank geometry was not present in the meridian ring and common plane calculations.

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

Leakage fluxes from the 29-group, neutron-photon DOT-IIW calculation were input to the MAP code by magnetic tape. Again, the tank geometry was not present in the meridian ring and common plane calculations. The common plane calculations were run both with and without interpolation of the discrete angular flux data.

Gamma ray source data on the meridian ring (i. e., fluxes at detector points) from the MAP and KAP-VI codes were input to separate SCAP code calculations. The SCAP code calculations provided tank top dose rates on the centerline.

### 5.2.3 Results of Analyses

Radiation environment both internal and external to this nuclear subsystem have been obtained. In some portions of this analysis, detailed results have been obtained; other calculations have provided only one data point. In this section, a brief presentation and intercomparison of results is given.

Neutron isodose rate contours internal to the nuclear subsystem are shown in Figure 5-6. The regions shown in this figure are defined in Figure 5-4. This figure graphically illustrates the results obtained from the DOT-IIW code. Three decades of neutron dose rate attenuation in the BATH shield are shown. Smooth, symmetric distributions are also visible throughout the geometry.

External neutron dose rate environment on the common plane as computed by the MAP and KAP-VI codes is shown in Figure 5-7. Considerable disagreement at radii below

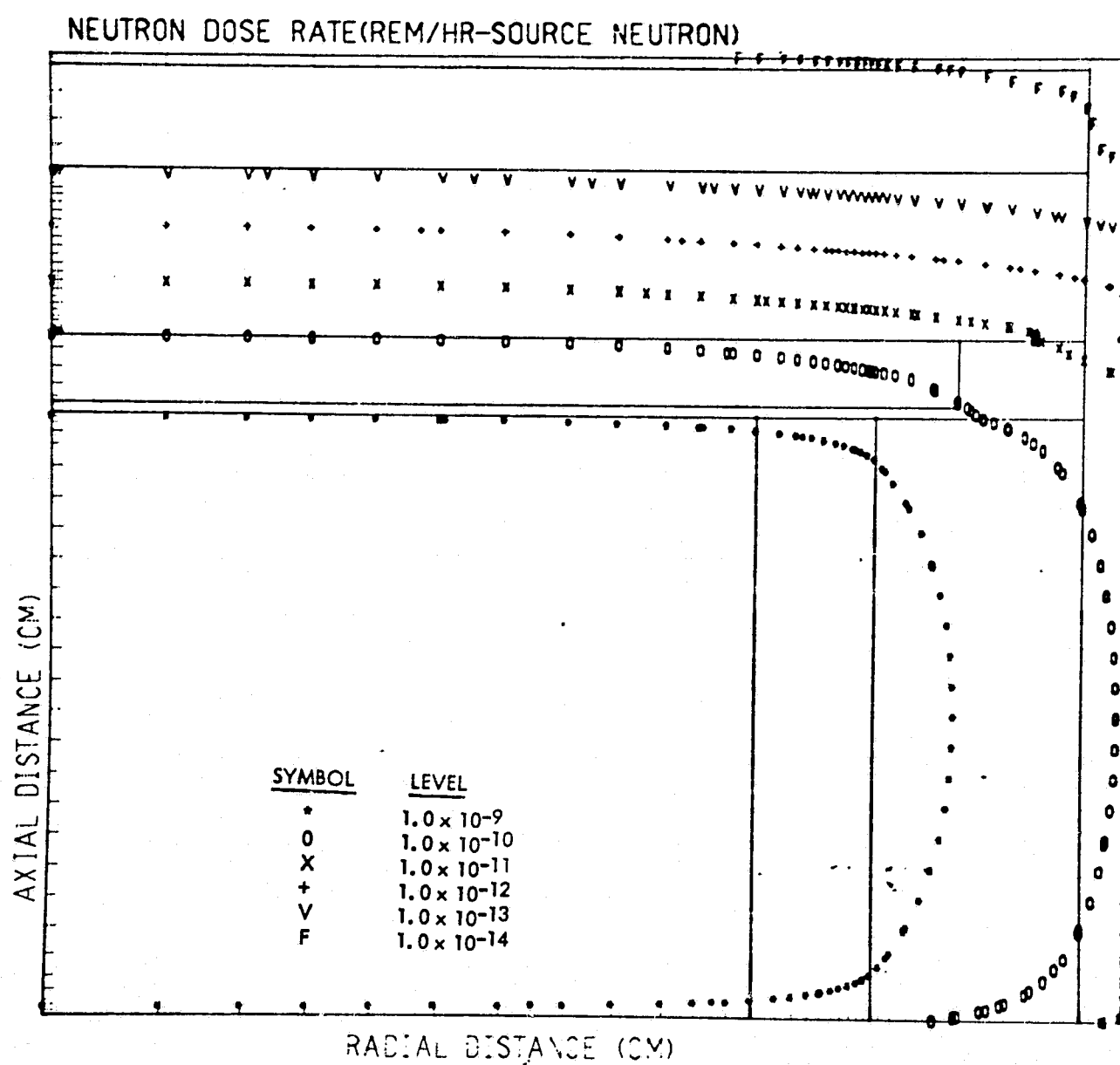


Figure 5-6. Isodose Rate Contours Internal to Nuclear Subsystem

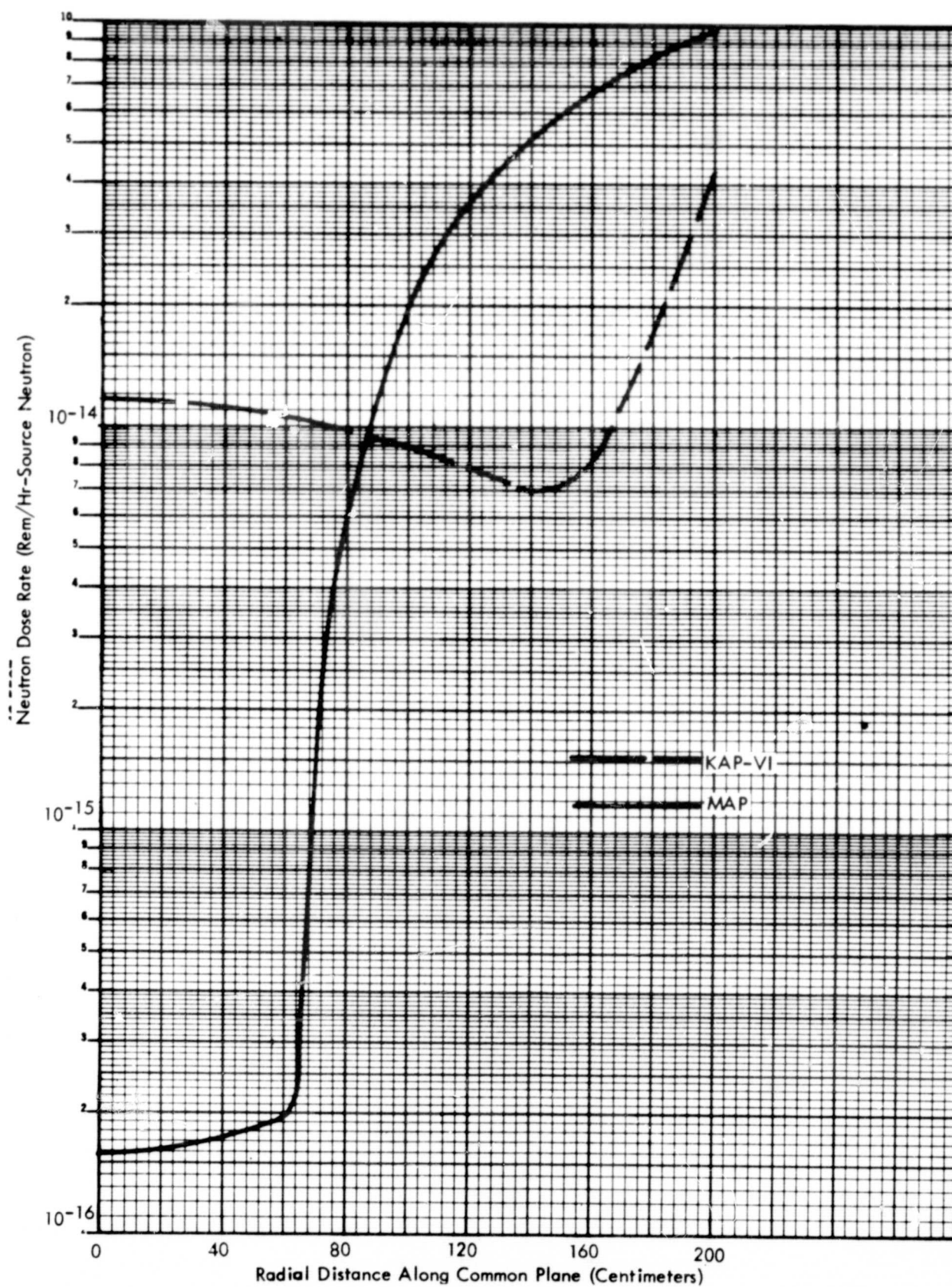


Figure 5-7. Comparison of MAP vs KAP-VI Neutron Dose Rate Along the Common Plane

200 centimeters is obtained. Beyond 200 centimeters, reasonable agreement is obtained. The KAP-VI results overemphasize the extent of the shield in the radial direction by excluding scattered radiation from around the shield. The "knee" in the curve, from an intuitive basis, should occur at about 60 to 70 centimeters. The MAP code correctly predicts this "knee." The MAP curve from 0.0 to 60 centimeters (the contribution from the top surface) is unexplainably low. This anomalous result propagates throughout the MAP results presented here. The gamma ray dose rate calculations on the common plane are shown in Figure 5-8. Again, the MAP results appear suspect at low radii.

External neutron dose rate environment on the 30-foot meridian ring as computed by the MAP and KAP-VI codes is shown in Figure 5-9. The MAP calculations were run both with and without polynomial interpolation of the discrete angular flux data. Beyond a polar angle of approximately 25 degrees, reasonable agreement is obtained. The gamma ray dose rate calculations on the meridian ring are shown in Figure 5-10. The effects observed on the common plane are also observed in the meridian ring calculations.

Gamma ray dose rate calculations at the top of the propellant tank as computed by the MAP, KAP-VI, and SCAP codes is shown in Figure 5-11. The 75-degree conical tank geometry shown in Figure 5-4 was included in all the tank top calculations. The SCAP code was run with spatial- and energy-dependent source data from both the MAP and KAP-VI calculations on the 30-foot meridian ring. The difference in input source spectra to the SCAP calculations is illustrated in Figure 5-12. Over most of the high energy range, a factor of 2 discrepancy in source intensity is obtained. This difference is reflected in the SCAP results at the tank top as shown in Figure 5-11. The SCAP results at the tank top appear consistent with the KAP-VI results shown. A thorough discussion of the SCAP code is presented in Section 5.1. The MAP code disagrees with the other results at radii below 150 centimeters but is consistent at larger radii.

These external environment results illustrate the flexibility of the techniques provided to the Marshall Space Flight Center. However, the recommendations outlined in Section 2.0 list specific areas where further development or improvements in the calculational technique are warranted.

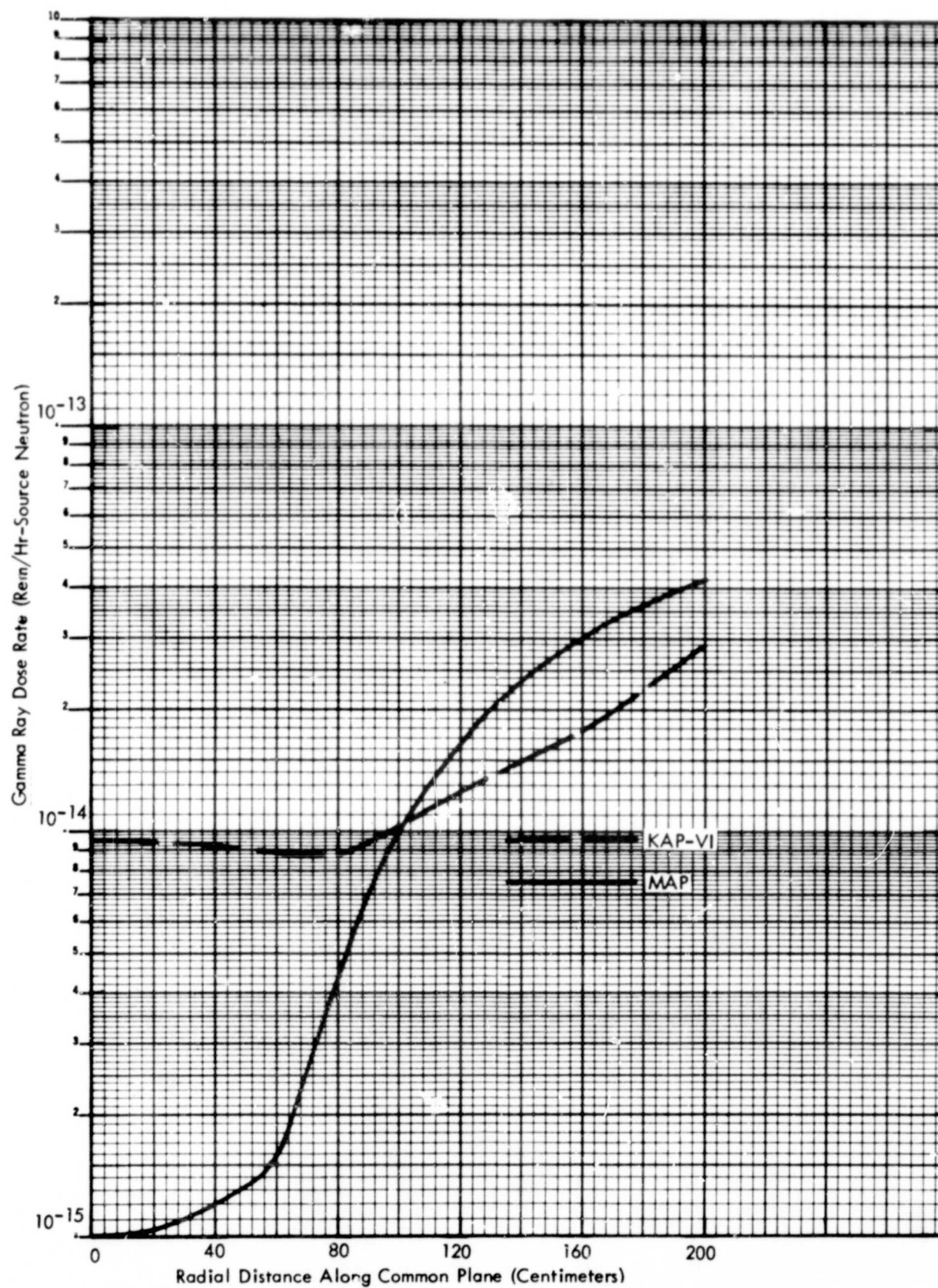


Figure 5-8. Comparison of MAP vs KAP-VI Gamma Ray Dose Rate Along the Common Plane

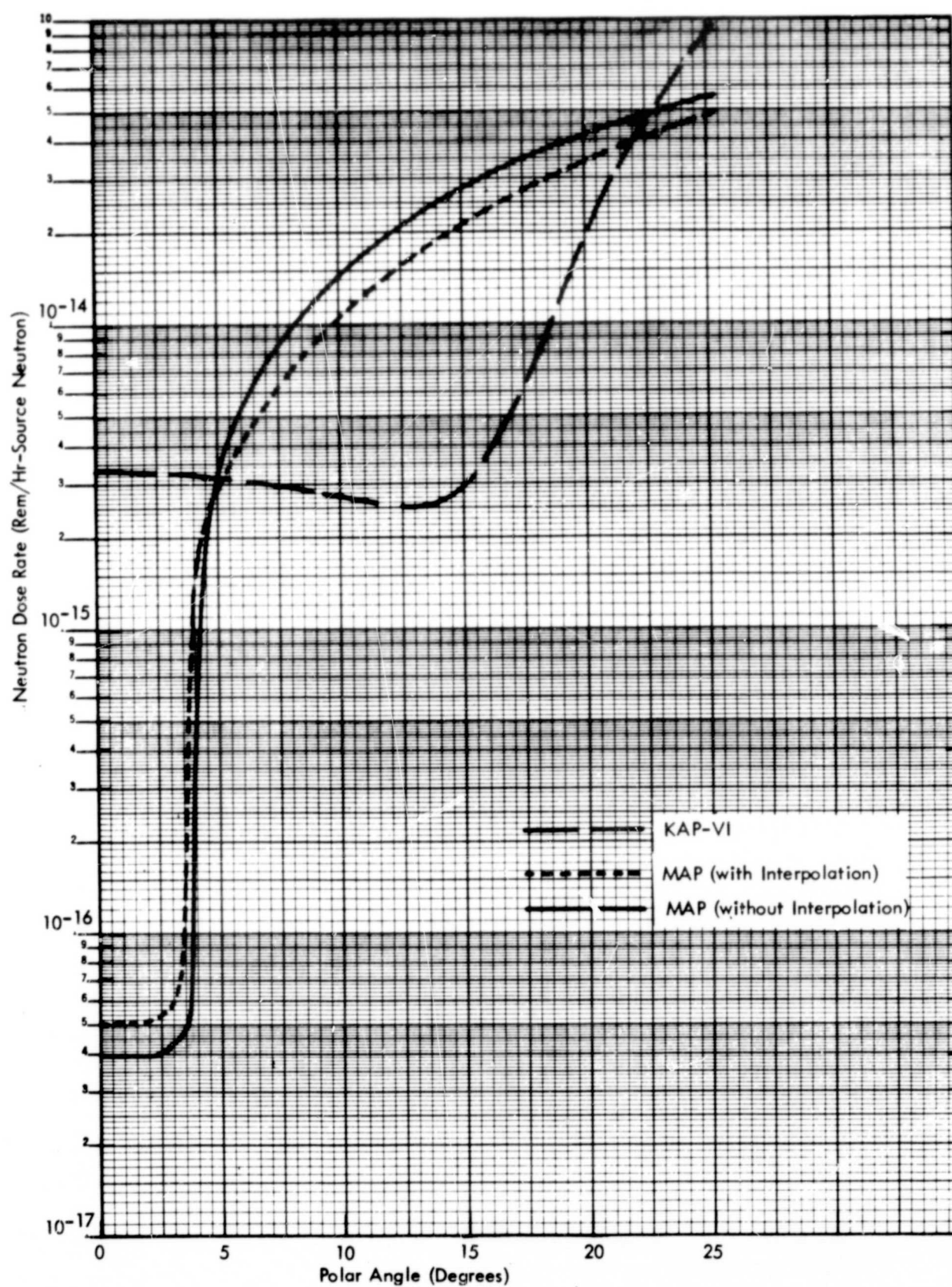


Figure 5-9. Comparison of MAP vs KAP-VI Neutron Dose Rate on a 30-Foot Meridian Ring

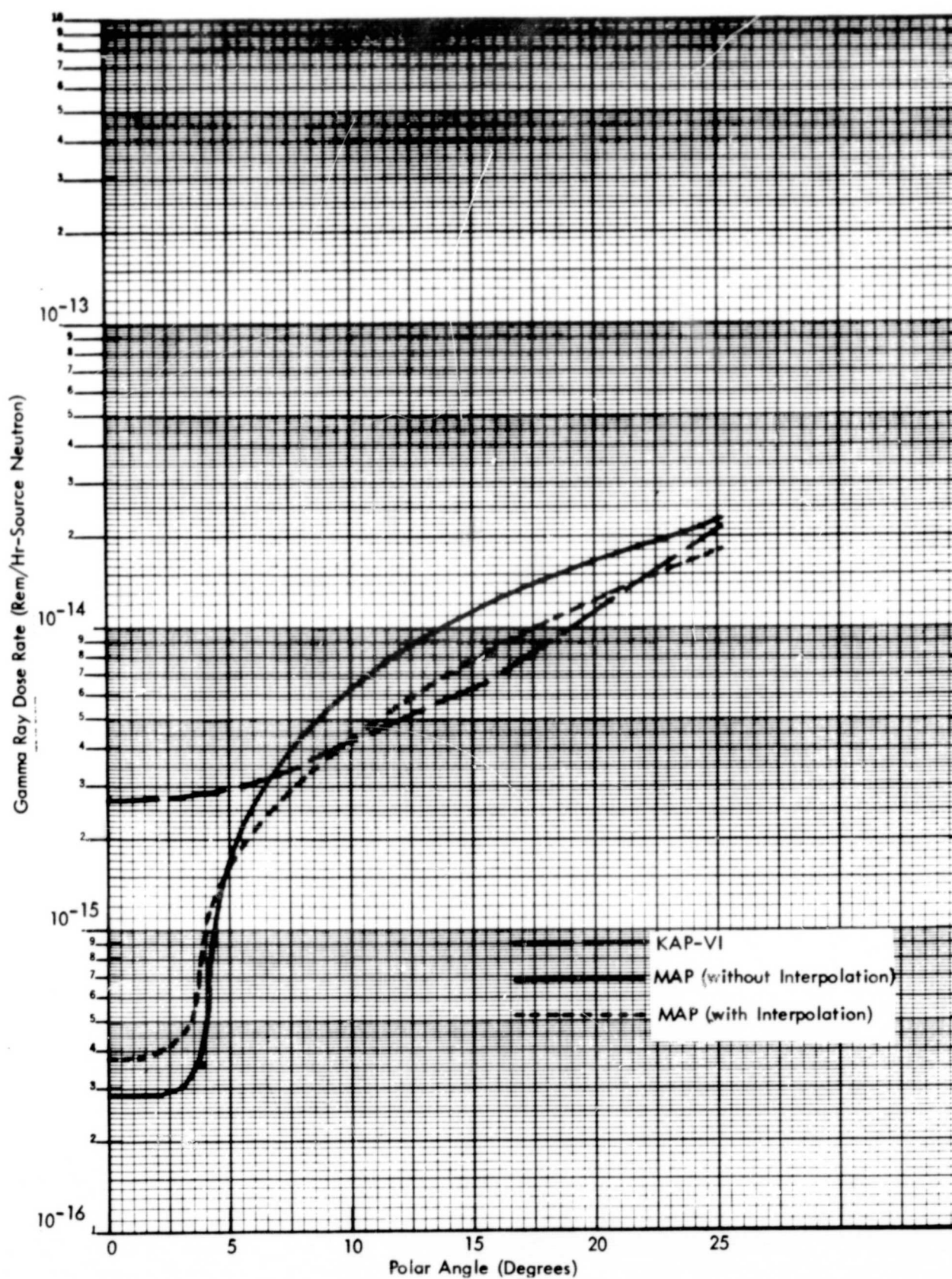


Figure 5-10. Comparison of MAP vs KAP-VI Gamma Ray Dose Rate on a 30-Foot Meridian Ring

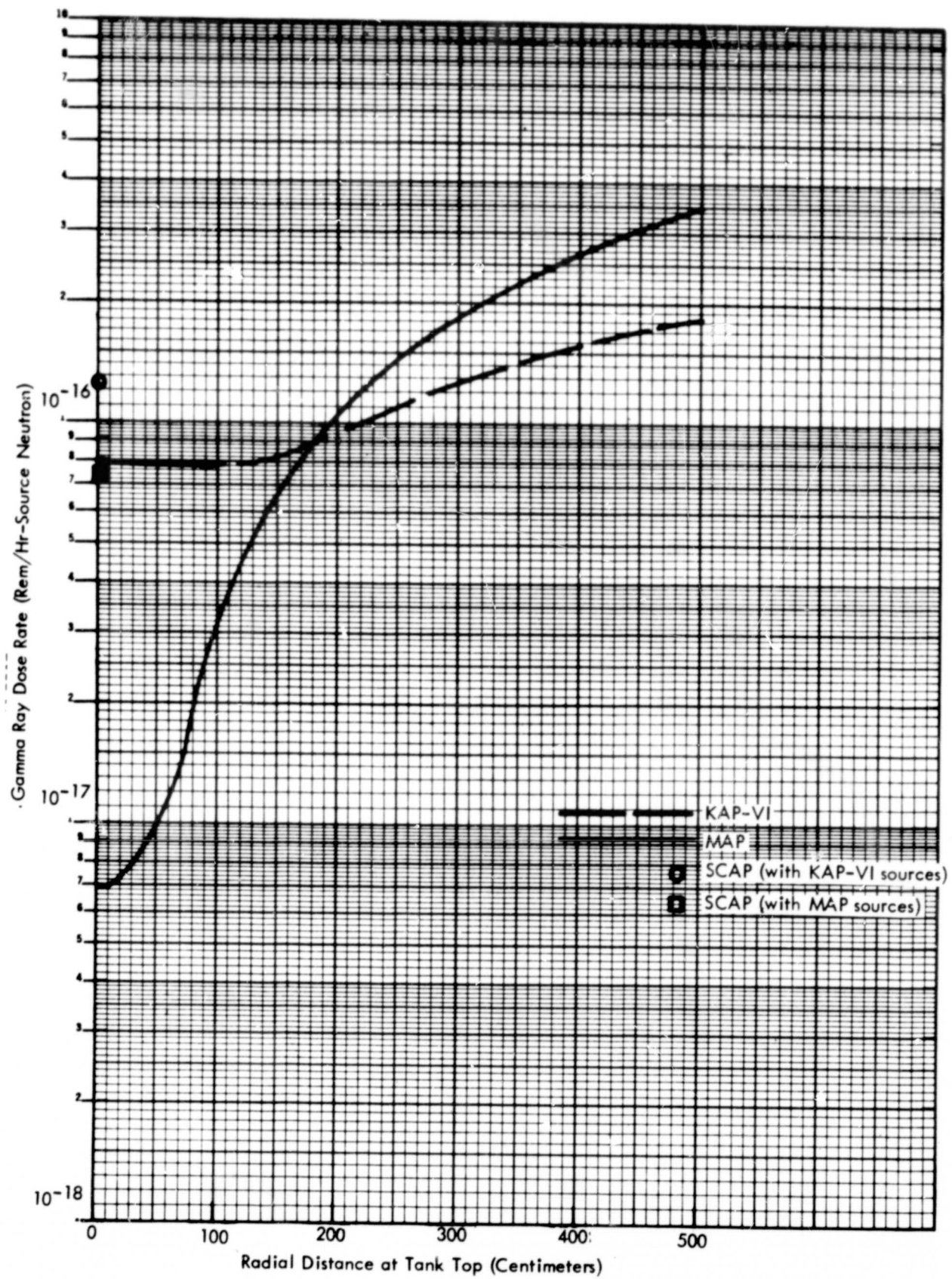


Figure 5-11. Gamma Ray Dose Rate at the Top of the Propellant Tank

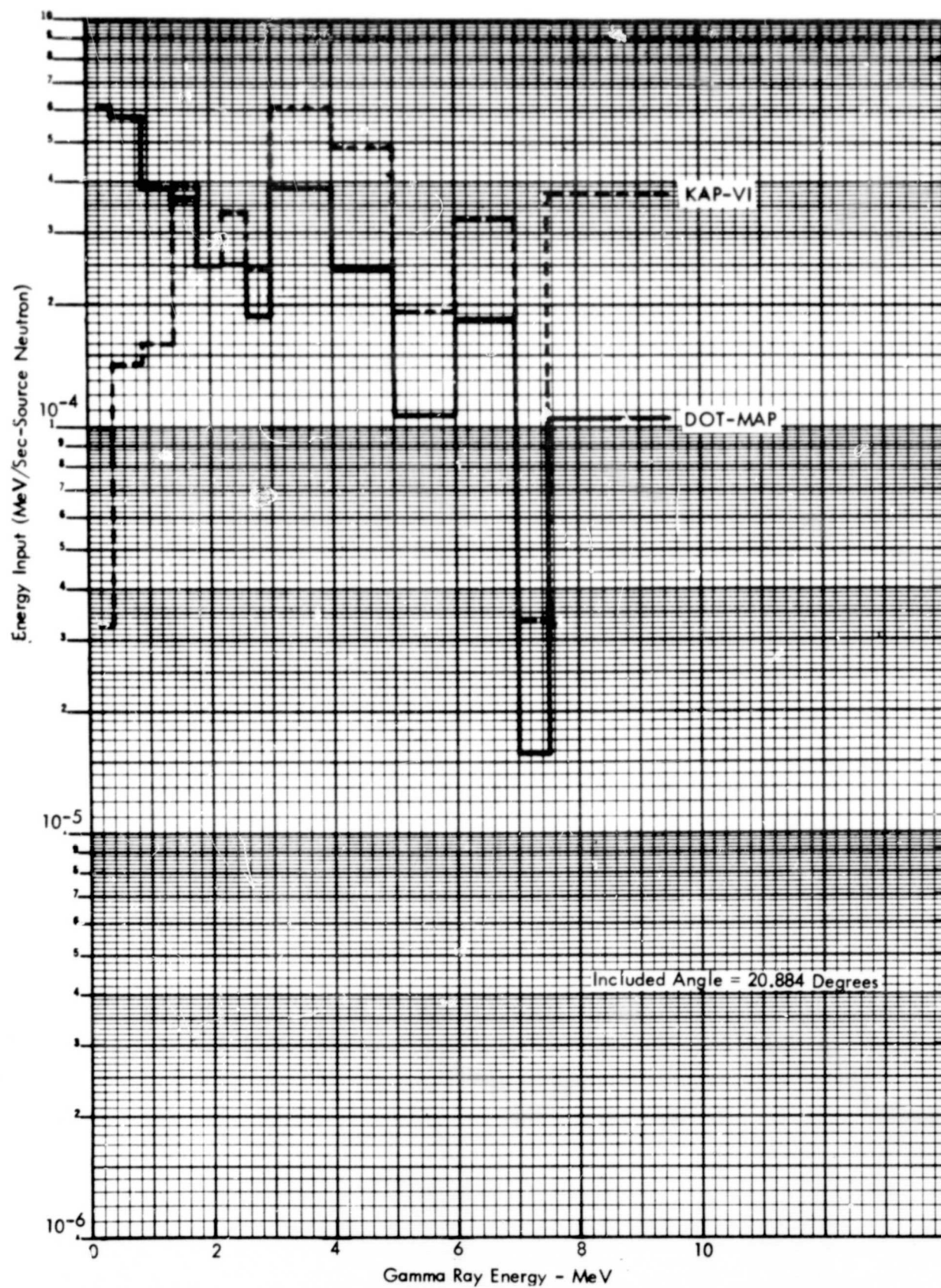


Figure 5-12. Comparison of MAP vs KAP-VI Total Energy Input to the SCAP Code

### 5.3 Investigation of LH<sub>2</sub> Radiation Transport Simulation Using CH<sub>2</sub>

This section presents an analytical investigation of the simulation of liquid parahydrogen radiation transport characteristics with polyethylene. In the recent PAX-E5/MSFC experiments conducted at the Westinghouse Astronuclear Laboratory Experimental Facility, the radiation transport characteristics of liquid parahydrogen were simulated using either high density (0.59 grams/cc) or low density (0.14 grams/cc) polyethylene for neutron and gamma ray transport, respectively. To verify this simulation and to illustrate the flexibility of the package of codes provided MSFC, APPROPOS and ANISN-W code calculations were performed. A representative top leakage spectrum from the DOT-IIW calculation (Problem Number 15,301) in Task 6 of the Contract was used as an input isotropic boundary source to the ANISN-W calculations. Slab geometry was chosen with transverse leakage parameters  $DY = DZ = 891.402$  centimeters and  $BF = 1.4209$ . Thus, a 33-foot diameter slab with a thickness of 500.0 centimeters (or 35.4 gms/cm<sup>2</sup> areal density LH<sub>2</sub>) was analyzed. P<sub>3</sub> neutron and photon cross sections were obtained as coupled data from the APPROPOS code and Master Libraries for simultaneous, neutron-photon calculations with the ANISN-W code. Full-range, Gauss-Legendre, S<sub>16</sub> quadrature coefficients were input to the ANISN-W calculations, as well. Neutron and gamma ray dose rates, threshold fluxes, and energy deposition responses were obtained. The following calculations were performed:

- (1) APPROPOS calculations to obtain LH<sub>2</sub> and CH<sub>2</sub> cross section data on a 1.0 gram/cc basis,
- (2) ANISN-W calculations with an LH<sub>2</sub> density of 0.0708 grams/cc and sources in the neutron energy groups,
- (3) ANISN-W calculations with an LH<sub>2</sub> density of 0.0708 grams/cc and sources in both the neutron and photon energy groups,
- (4) ANISN-W calculations with a CH<sub>2</sub> density of 0.14 grams/cc and sources in the neutron energy groups,
- (5) ANISN-W calculations with a CH<sub>2</sub> density of 0.14 grams/cc and sources in both the neutron and photon energy groups.

- (6) ANISN-W calculations with a CH<sub>2</sub> density of 0.35 grams/cc and sources in the neutron energy groups, and
- (7) ANISN-W calculations with a CH<sub>2</sub> density of 0.35 grams/cc and sources in both the neutron and photon energy groups.

The isotropic, energy-dependent source used as input to the ANISN-W calculations is shown in Table 5-10. For Problem 2 described above, a small, fictitious source in the 17th group was input to alleviate numerical problems (division by zero) in ANISN-W.

Upscatter in the LH<sub>2</sub> ANISN-W problems was a small effect; well-converged solutions were obtained after only four outer iterations. In the room-temperature CH<sub>2</sub> calculations, this was not the case. A time limit of 900 Central Processor Unit (CPU) seconds was imposed and resulted in upscatter convergence within 1 percent. In all cases, results obtained were of sufficient accuracy for this study.

The results of this study indicate that, by the proper choice of CH<sub>2</sub> density, LH<sub>2</sub> radiation transport can be separately simulated for neutron and gamma ray sources. Figure 5-13 illustrates the simulation of neutron radiation transport through LH<sub>2</sub> using CH<sub>2</sub>. A density of 0.35 to 0.40 grams/cc provides a good simulation of the neutron radiation transport properties of LH<sub>2</sub>.

By equating the electron density of polyethylene with that of liquid hydrogen resulting in a density of 0.14 grams/cc, a good simulation of the photon radiation transport properties of LH<sub>2</sub> is achieved. In this study, the secondary gamma rays in LH<sub>2</sub> or CH<sub>2</sub> contribute about 1 percent of the total gamma ray dose rate. The total gamma ray dose rate obtained in these calculations is shown in Figure 5-14; the gamma ray dose rate due to gamma rays generated in the LH<sub>2</sub> or CH<sub>2</sub> is shown in Figure 5-15. Because the reactor core is completely shielded in the realistic model calculations, and because no contribution from the reactor side leakage was included, the magnitude of the secondary gamma ray sources in the LH<sub>2</sub> or CH<sub>2</sub> is not surprising. A previous analysis of the simulation of LH<sub>2</sub> with CH<sub>2</sub> with no shield in the nuclear subsystem resulted in the same densities of CH<sub>2</sub> for simulation purposes, but the secondaries in LH<sub>2</sub> and CH<sub>2</sub> contributed approximately 10 percent of the total gamma ray dose rate.

TABLE 5-10  
INPUT SOURCE DATA FOR ANALYSIS OF LH<sub>2</sub> SIMULATION WITH CH<sub>2</sub>

|                     | <u>Energy Group</u> | <u>Normalized Source*</u> |
|---------------------|---------------------|---------------------------|
| Neutron<br>Groups   | 1                   | 4.964(-4)                 |
|                     | 2                   | 4.843(-4)                 |
|                     | 3                   | 3.363(-4)                 |
|                     | 4                   | 3.591(-4)                 |
|                     | 5                   | 3.152(-4)                 |
|                     | 6                   | 2.470(-4)                 |
|                     | 7                   | 1.096(-4)                 |
|                     | 8                   | 2.279(-4)                 |
|                     | 9                   | 1.961(-4)                 |
|                     | 10                  | 1.911(-4)                 |
|                     | 11                  | 1.561(-4)                 |
|                     | 12                  | 1.511(-4)                 |
|                     | 13                  | 8.277(-5)                 |
|                     | 14                  | 1.512(-4)                 |
|                     | 15                  | 2.898(-4)                 |
|                     | 16                  | 7.587(-4)                 |
| Gamma Ray<br>Groups | 17                  | 1.123(-1)                 |
|                     | 18                  | 1.730(-2)                 |
|                     | 19                  | 7.049(-2)                 |
|                     | 20                  | 6.877(-2)                 |
|                     | 21                  | 1.214(-1)                 |
|                     | 22                  | 1.413(-1)                 |
|                     | 23                  | 6.223(-2)                 |
|                     | 24                  | 7.119(-2)                 |
|                     | 25                  | 6.457(-2)                 |
|                     | 26                  | 7.173(-2)                 |
|                     | 27                  | 6.798(-2)                 |
|                     | 28                  | 7.655(-2)                 |
|                     | 29                  | 5.030(-2)                 |
|                     | Total               | 1.000(-0)                 |

\*Top leakage from DOT-IIW calculation (15.301)

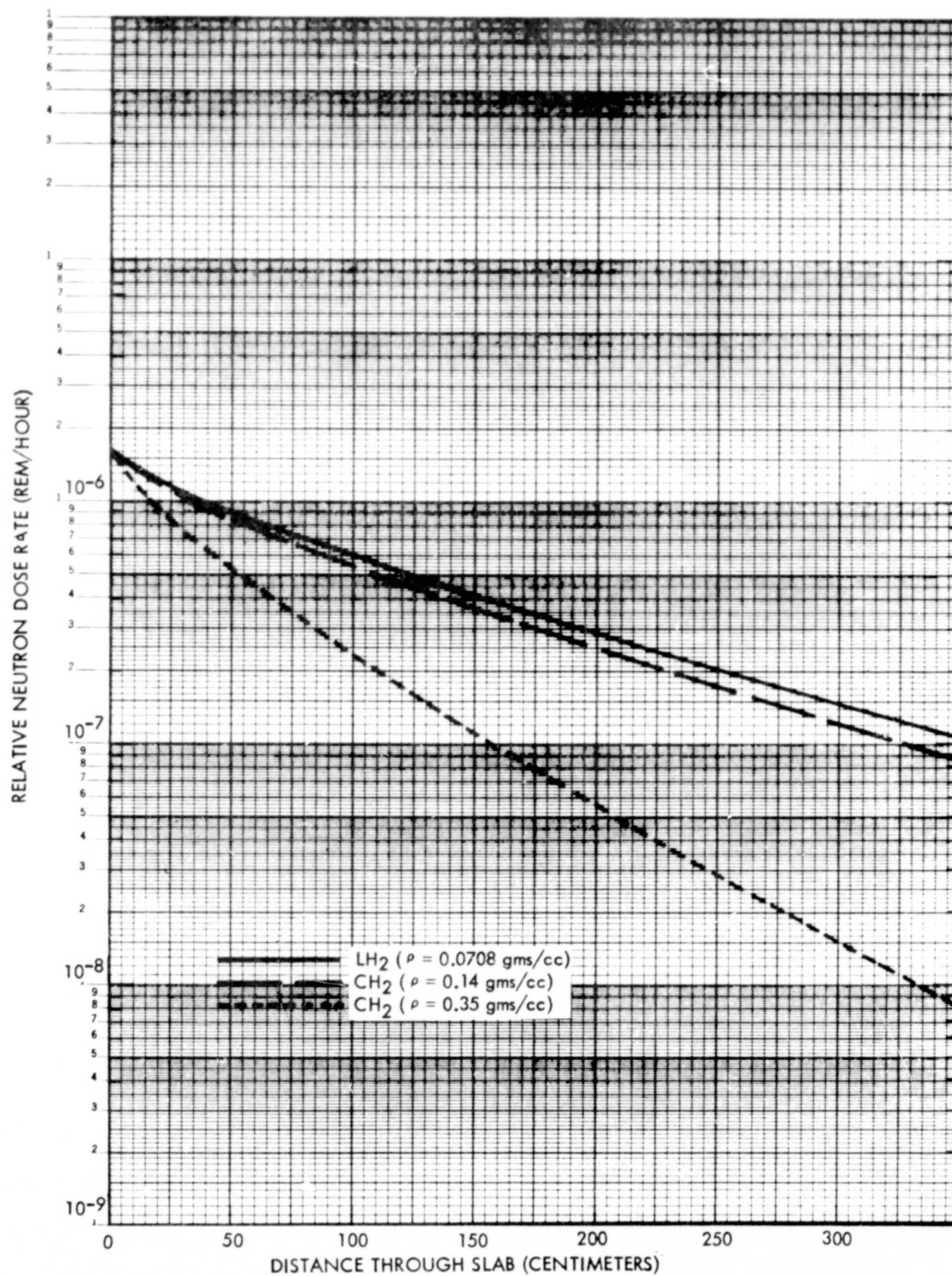


Figure 5-13. Comparison of Neutron Dose Rate Attenuation in Liquid Parahydrogen vs Room Temperature Polyethylene

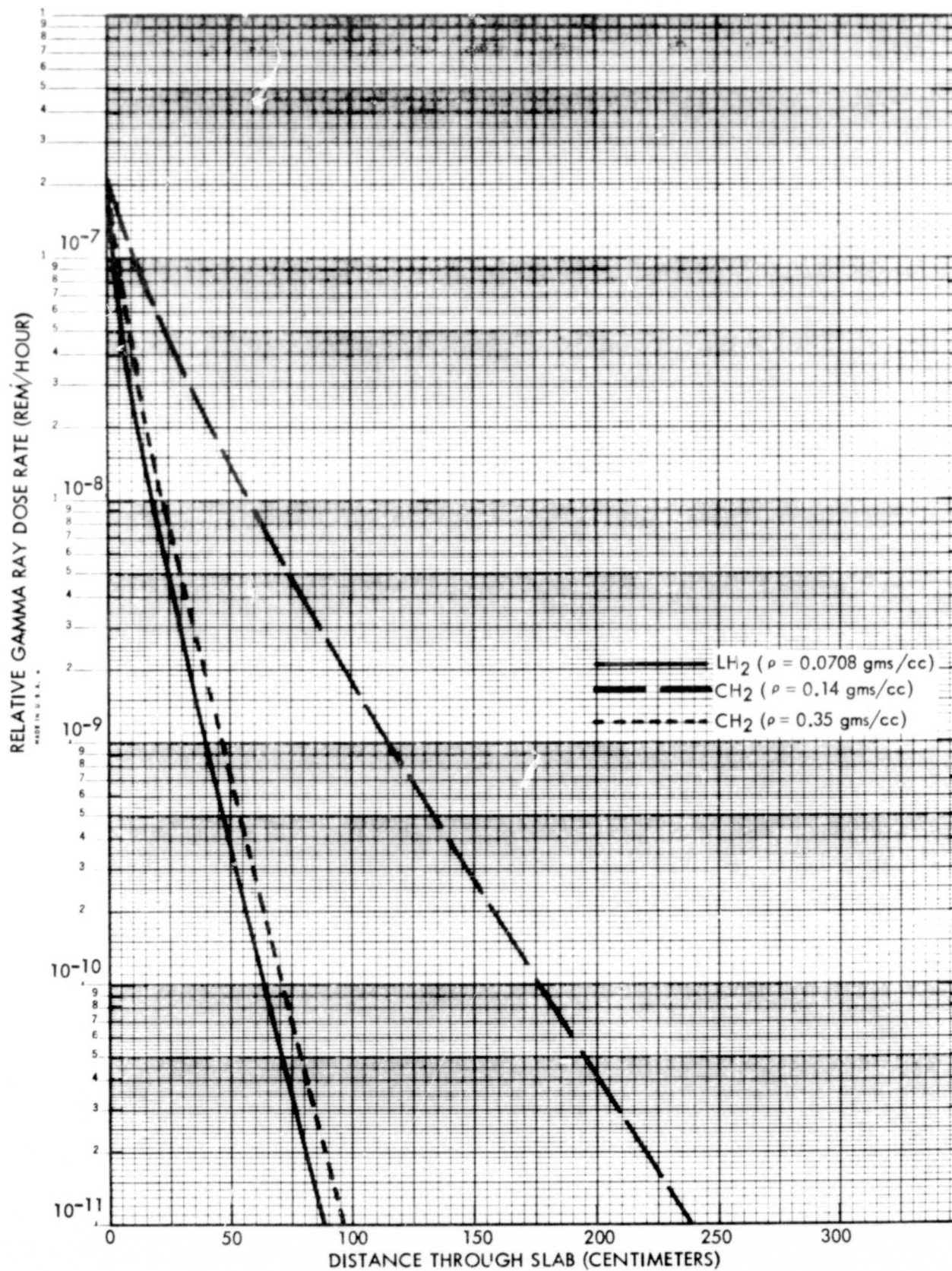


Figure 5-14. Comparison of Primary Plus Secondary Gamma Ray Dose Rate Attenuation in Liquid Parahydrogen vs Room Temperature Polyethylene

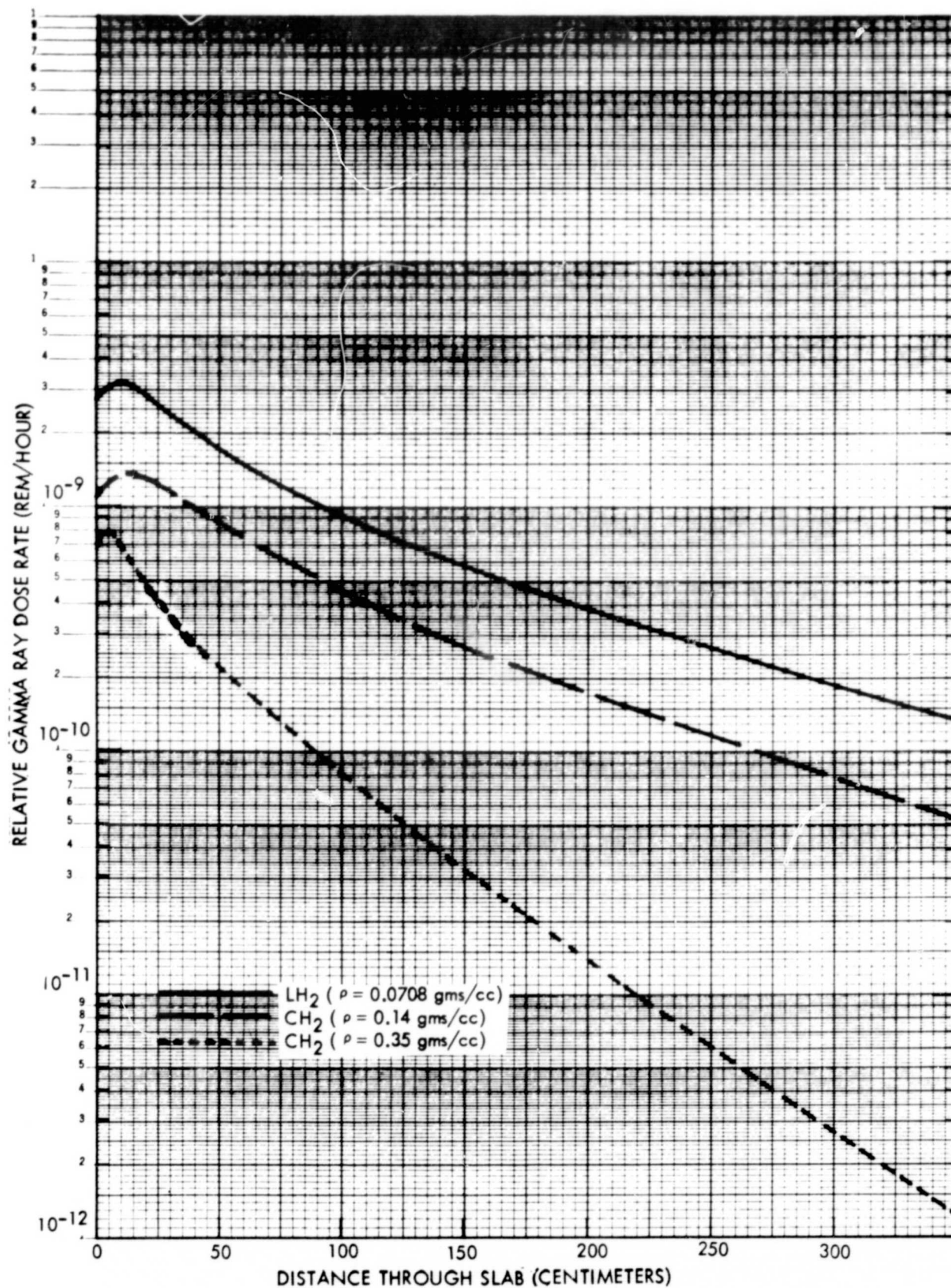


Figure 5-15. Comparison of Secondary Gamma Ray Dose Rate Attenuation in Liquid Parahydrogen vs Room Temperature Polyethylene

#### 5.4 Analysis of Shadow Cone Experimental Data

The ultimate worth of a shield design and radiation analysis technique must lie in its ability to accurately predict radiation levels about the radiation source. However, because of the complexity of the NERVA reactor sources and the many variables affecting test cell radiation levels, analysis of the experimental data for a NERVA type reactor is sometimes insufficient to separate these variables or to determine the worth of an analytical technique. The accuracy of a method should first be tested against simple, relatively "clean" experiments before proceeding to more complex systems.

As a result, one of the radiation analysis codes, the SCAP single- or albedo-scatter code, has been used to analyze the experiments of R. E. Baker.<sup>(15)</sup>

##### 5.4.1 Description of the Experiment

Baker's shadow cone experiment included measurements of the gamma ray dose rates in water utilizing Lansverk, 0-200 mr pocket ionization chambers. The gamma source was a 1.0 centimeter diameter by 1.0 centimeter long encapsulated cylinder of  $^{60}\text{Cobalt}$ . Measurements were made behind conic, lead shadow-shields of sufficient thickness to reduce the "direct" radiation to a small fraction of the radiation scattering around the shadow shields. Additional details of the experiment are included in Reference .

##### 5.4.2 Description of the SCAP Code Calculations

Figure 5-16 shows the geometry used for the analysis of the Baker shadow cone experiments with the SCAP code. The actual geometry used in the SCAP code is shown in Figure 5-17. Calculations were made for no cone, a  $20^\circ$  shadow cone, and a  $60^\circ$  shadow cone. Detectors were located along the z axis behind the cone as well as along a traverse perpendicular to the z axis at  $z = 100.0$  centimeters.

##### 5.4.3 Results of the SCAP Code Calculations

Figure 5-18 presents a comparison of SCAP code calculations with experimental data along the z axis behind the  $60^\circ$  shadow cone. Very good agreement is apparent. In Figure 5-19, the  $20^\circ$  shadow cone results are shown. The SCAP results here appear higher than

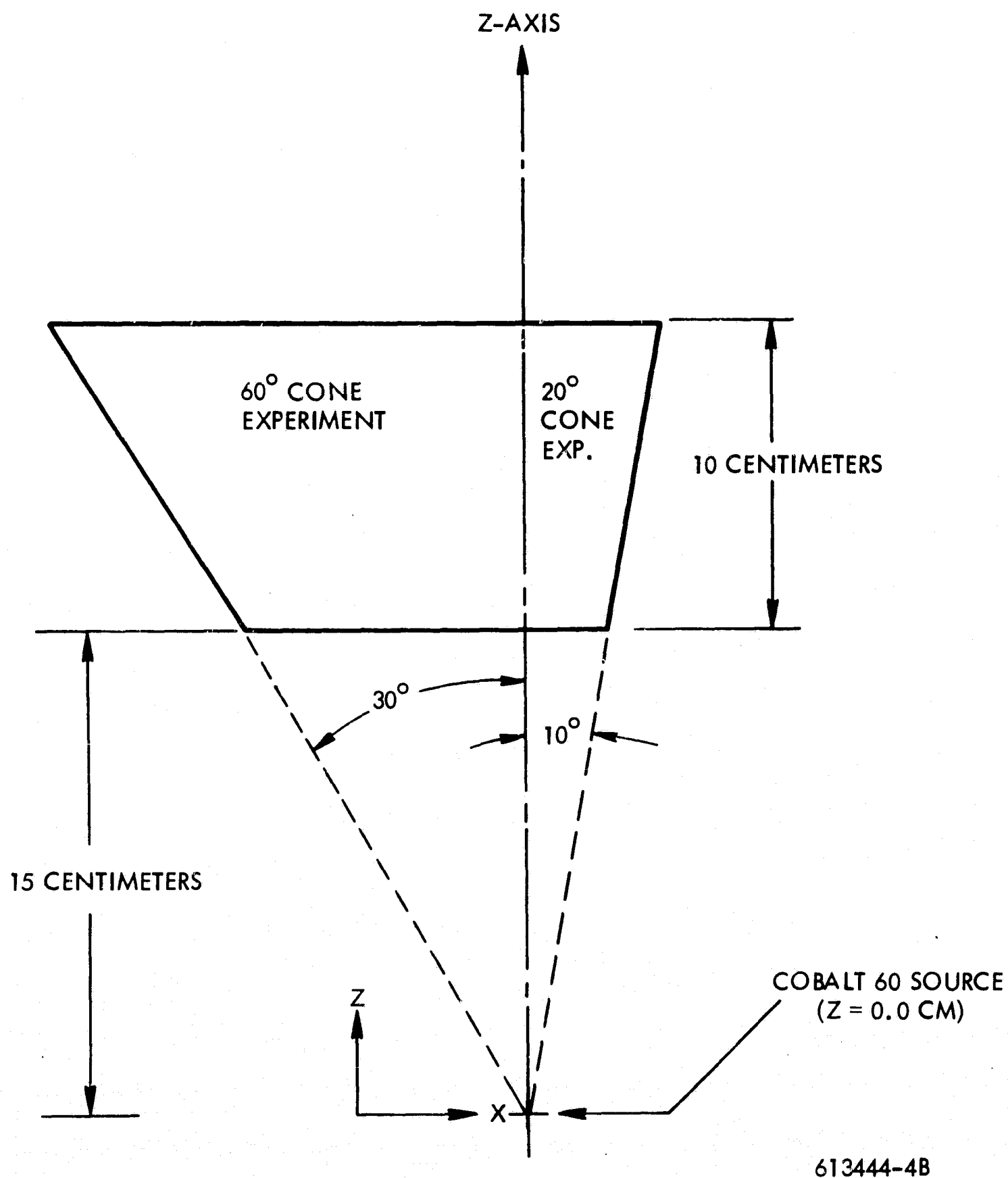
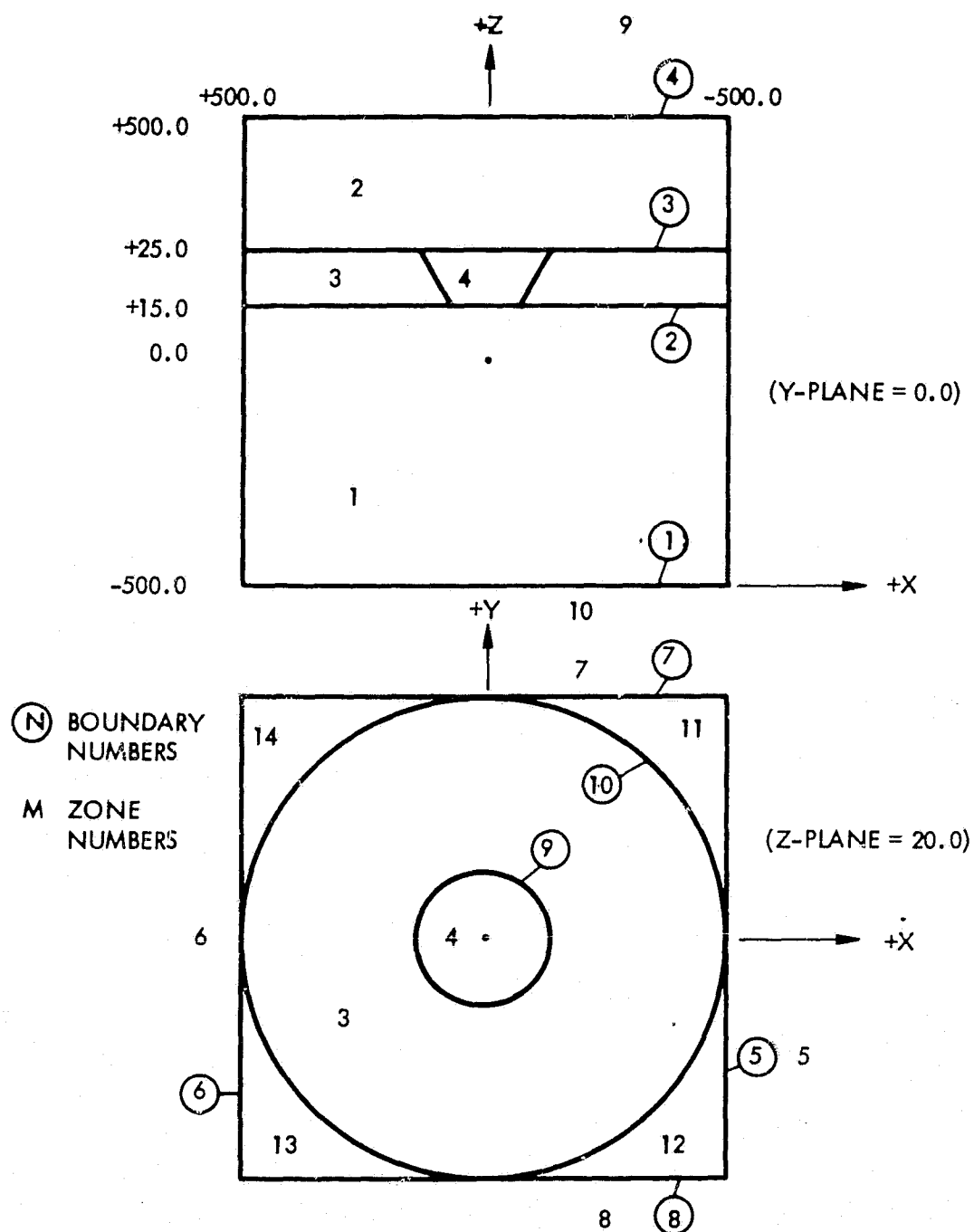


Figure 5-16. Geometry for Analysis of Baker Shadow Cone Experiments with the SCAP Code



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Figure 5-17. SCAP Code Input Geometry for Analysis of Shadow Cone Experiments

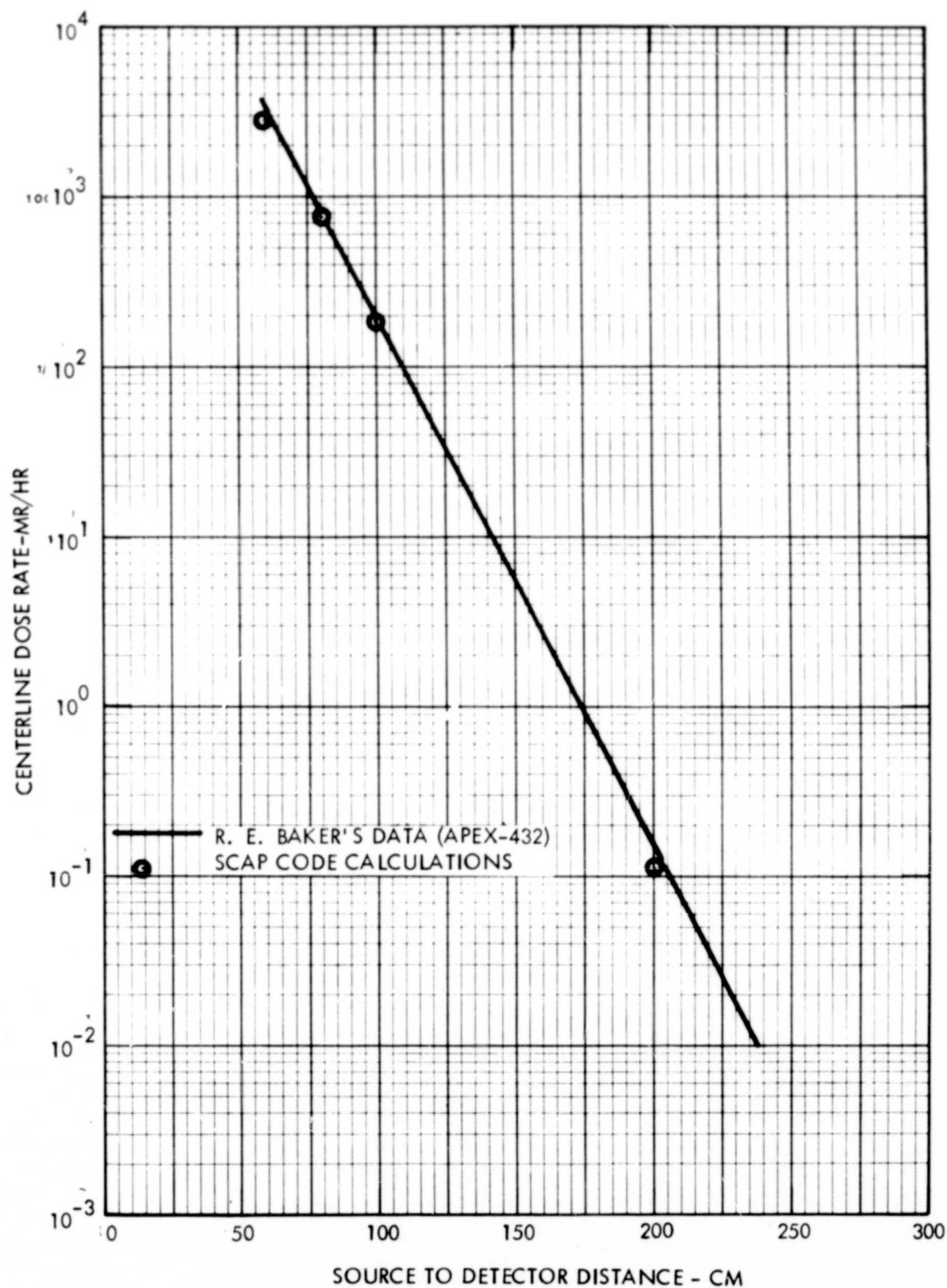


Figure 5-18. Comparison of SCAP Code Calculations vs Experiment  
for a 60-Degree Shadow Cone

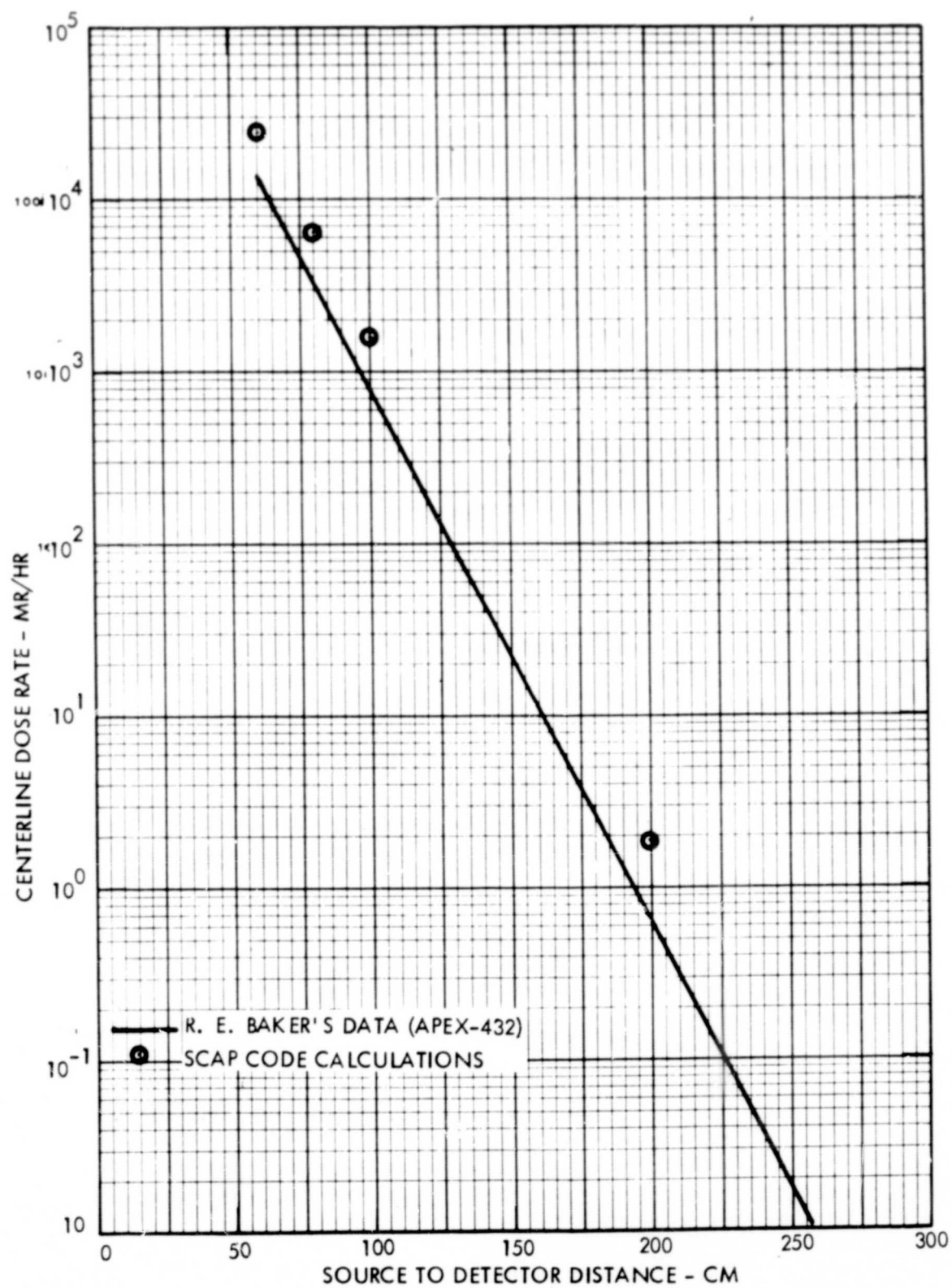


Figure 5-19. Comparison of SCAP Code Calculations vs Experiment for a 20-Degree Shadow Cone

the experimental results. To explain the discrepancy, a traverse perpendicular to the  $z$  axis, behind the cone at  $z = 100.0$  centimeters was analyzed. The results for the  $60^\circ$  shadow cone are shown in Figure 5-20. This figure shows an excessive increase in dose rate as the scatter medium is approached. In fact, in the early stages of this analysis, an exclusion radius about the scatter-point was included in the SCAP code to prevent the dose rate from "blowing up" when the detector is located within the scatter zone. Additional refinements to the code appear necessary for this case; caution in detector location and scatter ray placement should be exercised whenever this situation exists.

As a result of these calculations, the SCAP code can predict the dose rate when the application of a single-scatter technique is valid. For gamma rays in the energy range of approximately 200 KeV to 1.5 MeV, the predominant interaction of the gamma ray is Compton scattering.<sup>(16)</sup> Trubey has shown<sup>(17)</sup> that the increase from multiple scattering is offset somewhat by the photoelectric effect, and that even with no attenuation or buildup the simple approximation is generally adequate for unshielded detectors.

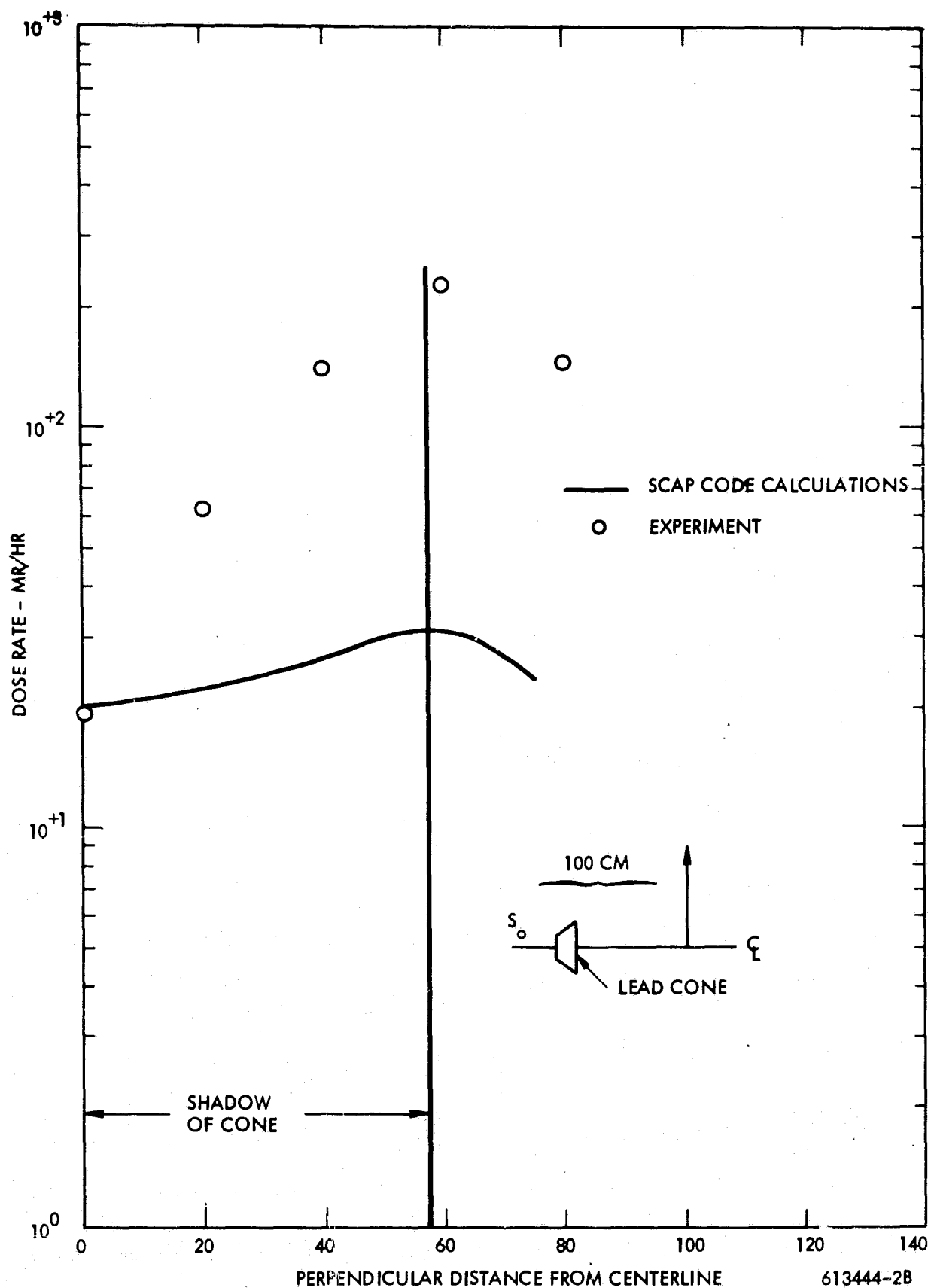


Figure 5-20. Comparison of SCAP Code Calculations vs Experiment for a 60-Degree Cone for a Traverse Perpendicular to the Cone Axis

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