RN-66250 (Vol. 1)

REON
SACRAMENTO PLANT

PROPOSAL NO. RN-66250

SYNTHESIS OF CALCULATION METHODS FOR THE DESIGN AND ANALYSIS OF RADIATION SHIELDS FOR NUCLEAR ROCKET SYSTEMS

SUBMITTED TO
MARSHALL SPACE FLIGHT CENTER
VOLUME I - TECHNICAL

MARCH 1966

MASTER

AEROJET-GENERAL CORPORATION
SACRAMENTO, CALIFORNIA

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Gentlemen:

Aerojet-General Corporation welcomes the opportunity to submit this proposal on the Synthesis of Methods for Design Analysis on Radiation Shields for Nuclear Rocket and Nuclear Electric Systems.

Since 1956, Aerojet has been engaged in nuclear rocket activities in both Government and Company-initiated programs. Within the past two months these efforts have culminated in the successful operation and restart of the NERVA Engine Systems at full power. These NRX/EST tests exceeded all test objectives.

Our achievements in NERVA and other major programs reflect the capability of our personnel, facilities, management and resources which will be applied toward accomplishment of the proposed program.

We at Aerojet approach this program with enthusiasm and with a genuine desire for the further advancement of national prestige in the field of nuclear rocket technology. The Corporation pledges to support the program with all available resources to attain for the Nation still another advance in nuclear rocketry. The pages which follow describe how we plan to make this effort a significant step in the achievement of that goal.

C. C. Ross

C. C. Ross

Vice President - REON

AEROJET
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ROCKET ENGINE OPERATIONS – NUCLEAR

VOLUME I - TECHNICAL
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AEROJET-GENERAL CORPORATION
A SUBSIDIARY OF THE GENERAL TIRE & RUBBER COMPANY

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I. INTRODUCTION AND SUMMARY

This proposal is submitted by Aerojet-General Corporation to the Marshall Space Flight Center in response to Request for Proposal No. 1-6-28-00029, dated 11 February 1966, entitled: "Synthesis of Calculation Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems."

The current methods used to predict radiation distributions about nuclear rocket and nuclear electrical systems are actually various combinations of shielding codes, each having inherent advantages and disadvantages. These codes evolved and were combined as a result of the immediate and pressing contractual requirements of past and current reactor programs without sufficient emphasis directed toward optimization of the methods on the basis of validity, efficiency, and cost. As a result, use of these current prediction methods has resulted in numerous analytical difficulties.

Aerojet recognizes the innate deficiencies of these current analytical methods and responds to the Marshall Space Flight Center's requirements for a comparison and critical evaluation of these methods by proposing a program for objectively screening, evaluating, and selecting prediction methods which will ensure that future calculation of radiation levels will be made quickly and accurately.

A. PROGRAM OBJECTIVES

The proposed program has four principal objectives. The first is to compare and critically evaluate the applicability of existing computer programs to radiation analyses for nuclear rocket and nuclear electric systems. The second objective is to select one or two optimum combinations of analytical methods for use in an automated analysis system that minimizes data handling by the analyst. The third objective is to reprogram the selected methods in the FORTRAN-IV language. The fourth objective is to check out the selected methods on the Marshall Space Flight Center computer and instruct MSFC personnel in the use of the selected programs.

B. TECHNICAL APPROACH

Aerojet's approach to achieve these program objectives is to conduct a literature and field survey in order to identify and ferret out those existing analysis methods that are either of interest or contain desirable features. Simultaneously, criteria will be developed on the basis of the program operational characteristics and on considerations of the analysis requirements of nuclear rocket and nuclear electric systems in order to evaluate prospective analysis methods. Suitable experimental data also will be collected and used to validate the results of the analysis methods. This validation is an important element of the proposed program because it will reduce uncertainties associated with the evaluation of prediction accuracy.

Operational characteristics of the candidate analysis methods will be determined through use of controlled calculations on selected test problems. Quantitative specification of these characteristics are needed for use in the evaluation of candidate analysis methods. Test problems will include basic or "clean" configurations as well as actual configurations in order to distinguish weaknesses in the analytic techniques stemming from possible weaknesses in the basic nuclear data. The input data for these tests will be prepared by personnel thoroughly familiar with the methods used to ensure complete objectivity during the evaluations.

On the basis of the characteristics of the candidate methods and the evaluation criteria developed earlier, comparisons and a critical evaluation of candidate methods will be made to facilitate final selection of the preferred method. This approach assures MSFC that the data generated will result in selection of the best analysis method or combination of methods.

Then, the selected method or methods will be integrated into a program system and rewritten as necessary in FORTRAN IV language. This will provide an automated analysis system which can be modified conveniently or updated as needed for future applications.

Finally, the selected method or methods will be checked out at the MSFC computer facility and MSFC personnel will be instructed in its use to allow complete utilization of the selected system. This will ensure conveyance of a complete analysis package to MSFC for maximum utilization by its personnel.

C. KEY FEATURES OF THE PROGRAM

The proposed program has several key features that reassure Marshall Space Flight Center personnel that the program will result in the successful development of an optimum analysis method for reliable prediction of radiation levels.

Criteria to be used in the evaluation of candidate methods will be formulated around specific analysis requirements and actual problems of nuclear rocket and nuclear electric systems. These will include such considerations as the requirements for definition of the energy and angular distributions of the radiation environment to ensure selection of those analytical methods which offer the greatest potential for predicting nuclear rocket radiation distributions.

Experimental data will be used in test problems to evaluate the prediction accuracy of candidate analysis methods. These data are readily available as a result of Aerojet's participation in NERVA, ML-1 and SNAP-8 programs, and Aerojet has a thorough understanding of the dosimetry methods used as well as their limits of applicability.

An interim report will be submitted to MSFC upon completion of the comparison and evaluation task. This report will provide MSFC personnel with an early opportunity to review the data in detail before participating in the final selection of the preferred analysis method.

Radiation analysis techniques developed by Aerojet over the past four years under the NERVA program will benefit the selected program system. These techniques include the geometric configuration plotter, several data reduction codes, prediction of computer memory requirements of discrete ordinates codes, and appropriate cross-section and moments-method data.

Aerojet's use of the discrete ordinates methods for predicting the neutron environment, secondary gamma-ray source terms, and primary gamma-ray environment will be given specific consideration. Previous use of these methods in predicting neutron distributions has resulted in excellent agreement between pretest predictions and measured data.

D. ANTICIPATED PROBLEM AREAS

The anticipated problem areas include the requirement to define radiation levels at the payload of flight vehicles. This problem requires treatment of radiation transport through large void media followed by treatment of radiation scattering, thermalization, absorption, and leakage in non-void regions. Although accurate solutions to this problem require use of the costly Monte Carlo programs keeping detailed account of the energy and angular distributions, valid estimates are possible by use of suitable combinations of albedo methods, point kernel methods, and discrete ordinates methods. The proposed program will determine the validity of existing methods, including the Monte Carlo methods, for treating this difficult radiation transport problem.

A related problem is the difficulty encountered in some applications of the discrete ordinates methods. Negative fluxes and oscillations are often encountered in regions far removed from the source regions. Aerojet has developed several techniques to minimize these effects, and these will be applied to the analyses during the proposed program.

Another problem is the inadequacy of available neutron moments data for the point kernel analysis method for shield penetration depths greater than about

150 $gram/cm^2$. Efforts are currently being made in the NERVA program to extend the range of applicability for these data, and the results of these efforts will be used in the proposed program.

Programming problems are those related to integration of the selected methods into program systems. Since the various candidate programs were developed by different groups for different applications, their respective input and output data are incompatible. Specifically, the output data from some analysis methods apply for point locations while the data in others apply for volume regions. This problem will be resolved by generating specialized routines to modify the data for compatibility.

E. PROGRAM PLAN

The proposed program will be conducted sequentially in three phases over a 12-month period.

The initial phase will begin with a literature search and field survey, conducted concurrently with the development of evaluation criteria for the analysis methods. This phase will span the first two program months and will end with a joint review by MSFC and Aerojet personnel and selection of candidate methods and test problems. During this phase, operational criteria also will be identified, characteristic radiation problems defined, and experimental data suitable for the evaluation criteria collected.

The second phase of the program begins at the third program month and extends for five months. The milestone which marks the completion of this phase is delivery of the interim report to MSFC for review prior to the final selection of analysis methods. During this second phase, suitable cross-section data will be selected and/or developed, test-problem calculations will be made, and candidate analysis methods will be compared and evaluated.

The third program phase begins at the eighth program month and extends for the duration of the program. During this phase, the best analysis methods will be selected in conjunction with MSFC personnel, and the choosen method or methods will be reprogrammed in FORTRAN IV language and checked out at the MSFC computer facility. This phase ends with delivery of the final report to MSFC.

The Program Plan is fully responsive to the scope of work defined in the Request for Proposal. In addition, the plan provides formal reviews at the end of the first and second phases to enable cognizant MSFC personnel to participate in formal reviews with Aerojet program personnel in the selection of candidate analysis methods and test problems. Frequent informal briefings and communications with MSFC personnel further assure that cognizant personnel are kept continuously apprised of program progress and actively participate in the assessment of current problems and in the initiation of corrective actions.

F. MANAGEMENT PLAN

The program will be assigned to the Nuclear Analysis Department of the NERVA Staff Engineering Division. This arrangement will provide the program participants access to the resources that have been developed in the course of the NERVA program.

Mr. W. R. Butler, currently manager of Reactor Analysis, will be assigned as full-time project engineer for the proposed program. He has had more than 10 years of reactor and radiation analysis experience. Some of his recent work includes radiation environment, nuclear heating, and neutron induced activity predictions for the NERVA engine system as well as for the advanced versions of NERVA. These analyses required application of existing Monte Carlo, discrete ordinates, and point kernel methods as well as development of numerous auxiliary support programs which will be investigated and utilized in the course of the proposed program.

Other key personnel who will assist Mr. Butler include Mr. B. T. Kimura, who also has had more than 10 years experience in radiation analyses of flight-type nuclear propulsion systems including nuclear rockets and Mr. P. L. Redden who has had extensive programming experience related to the requirements of this proposed program.

G. APPLICABLE FACILITIES

The computer facilities of the Sacramento Plant Computer Sciences Laboratory are available for this program. The equipment includes an IBM 7094 computer (32K core storage) with a complete complement of support equipment.

These Aerojet computer facilities will serve the low cost, short turn-around requirements necessary to complete the program on schedule and within the budgeted costs. However, a majority of the work will be performed on the MSFC computer facilities. This program will also take advantage of existing data-link facilities between MSFC and Aerojet. Data cards prepared in Sacramento will be submitted to the Aerojet message center and transmitted to the MSFC message center where the data will automatically be punched out on cards, ready for immediate use on the MSFC computer.

H. TECHNICAL CAPABILITIES RELATING TO THE PROGRAM

Aerojet possesses an extensive theoretical and experimental background in the fields of radiation transport analysis, shielding design, and nuclear rocket and nuclear electric systems design. Aerojet has participated in a number of projects with NASA, the AEC, the Army, the Air Force, and the Navy, and its personnel have extensive experience in reactor and shielding technology as well as computer applications technology. These project include NERVA, ML-1, SNAP-8, SNAP-50/SPUR, and others. Most of the analytical methods required in the proposed program have been used in numerous current and previous contracts, and their characteristics and applications are well understood by Aerojet personnel.

Aerojet, as prime contractor for the NERVA program, has encountered many of the problem areas associated with radiation environment prediction on nuclear rocket systems. Many of these problem areas have been resolved and potential solutions to the others identified. These resources that have developed in the course of the NERVA work will be made available to this proposed program thereby assuring MSFC that Aerojet will conduct a straight-forward program with a minimum number of problem areas in order to achieve all program objectives successfully.

II. TECHNICAL DISCUSSION

Radiation analyses currently under way in nuclear rocket and nuclear electric programs require the use of numerous combinations of computer codes having varying degrees of technical sophistication, efficiency, validity, and cost. These combinations of codes have evolved largely as a product of the technical talents, background, and initiative of the personnel among the different contractors performing the radiation analyses.

The object of this proposed program is to critically evaluate promising analysis methods in current use in order to synthesize one or two methods best suited for nuclear rocket and nuclear electric applications. This effort would involve use of the candidate methods for performing calculations on selected test problems. The results of these calculations, properly interpreted, will form the basis for selection of the best methods which will then be modified to include the desireable features of those methods not selected.

This section provides a detailed discussion on the technical aspects of the proposed program. The considerations include a review of existing analysis methods, development of evaluation criteria, selection of test problems, the calculations to be performed, selection of candidate methods, and reprogramming of the selected methods to Fortran IV as required.

A. REVIEW OF EXISTING METHODS

Existing methods for radiation analysis consist of different combinations of special purpose codes that have been developed to meet the immediate requirements of different shielding analysis and reactor analysis groups. Varying degrees of automation have been developed by those groups which must process large numbers of problems requiring combinations of existing codes. This automation involves preparation of input data, transfer of output data for input to another succeeding chain, and processing of the final output into forms that are readily interpreted by the users. However, the existing automated methods have been developed for specific projects and consist of those combinations of methods that are popular among the particular working groups without any directed effort at reviewing and evaluating alternative methods in current use.

In view of the giant strides made in computer hardware development, it seems appropriate that some effort be directed at reviewing existing computer programs to determine whether new combinations or modifications to old combinations, would make better use of recent computer hardware improvements. Among the new developments are parallel processing of programs that do not require use of the entire memory of the computer.

The following subsections provide a discussion of the more popular of existing shielding codes. As is usual, the codes will be grouped according to the degree of sophistication in their basic theory.

In the first subsection, the point kernel methods are described including such codes as the Los Alamos QAD-P5, the General Electric Programs 14-0 and 14-1, and the General Dynamics Program C-17. The second subsection discusses existing Monte Carlo programs including the General Electric Program 18-1, the Los Alamos Program MCS, and the Oak Ridge Program O5R. The discrete ordinates methods are then discussed with consideration of their general use as well as some specific applications. The fourth and fifth subsections consist of discussions on the diffusion theory method as well as the auxiliary methods that are needed for data handling. These auxiliary methods consist of codes for gamma ray scattering, for plotting geometric [illegible] [illegible], and for generating nuclear cross section data, for data reduction. These various methods are summarized in Table I.

Table I

A Brief List of Existing Methods

Methods	Remarks
Point Kernel	
QAD-P5	General Geometry; FORTRAN
14-0; 14-1	Azimuthal Symmetry; FAP
C-17	Gamma Spectra; SAP-FAP
Monte Carlo	
FMC-N/FMC-G	Flexible Monte Carlo; FAP and FORTRAN
18-1	Monte Carlo for Cylindrical systems; FAP
MCS/MCG	Generalized Monte Carlo; FLOCO
05R	Generalized Neutron Monte Carlo; FORTRAN
Discrete Ordinates	
DSN/DTK	One-Dimensional; FLOCO
DTF-2/DTF-4	One-Dimensional; FORTRAN
TDC/DDK	Two-Dimensional; FLOCO
2DF/TDC (FORTRAN)	Two-Dimensional; FORTRAN
Diffusion Theory	
Auxiliary Methods	
GGG	Gamma scattering; FORTRAN
GECOP	Plots Configuration from input data; FORTRAN
RTDCO/RQADO	Data Reduction Codes for TDC and QAD-P5; FORTRAN
Cross Section Codes	Fast and Thermal neutron constants Gamma ray data.

1. Point Kernel Methods

The term "point kernel methods" is used to denote those methods which evaluate radiation response X (e.g., energy deposition and flux) according to an integral over source region V_s:

$$X = \int_{E_{min}}^{E_{max}} dE \int_{V_s} S(\vec{r}_1, E)\ K\ (\vec{r}_1 - \vec{r}_2) dV$$

where

$S(\vec{r}_1, \vec{r}_2)$ = Radiation emission per unit time within energy and volume increments dE and dV at energy E and position $\vec{r}_1$

$K(\vec{r}_1 - \vec{r}_2, E)$ = attenuation kernel or the response at the field point $\vec{r}_2$ due to a source of unit strength emitting particles (neutrons or gamma photons) of energy E.

The kernel K $(\vec{r}_1 - \vec{r}_2$, E) can either be a function or a functional but the feature that distinguishes this method is that the kernel for given source energy depends only on the distance $\vec{r}_1 - \vec{r}_2$ and on the material distribution along $\vec{r}_1 - \vec{r}_2$.

Despite the apparently gross assumption implied by the form of the kernel $K(\vec{r}_1 - \vec{r}_2$, E) and the comparative simplicity of the theory, the point kernel method has established itself as a basic method in central station reactor shielding analysis. In the radiation analysis of nuclear rockets and nuclear electrical systems, the method still retains its indispensability; however, the method must now be supplemented by additional, more sophisticated, procedures because of a more severe radiation environment in terms of intensity or integrated exposure, and because of a greater penalizing effect of shield weight.

The comparisons contained in Reference 1 should be extended to include a wider class of shielding situations. The study should include the evaluation of Los Alamos' QAD-P5 (Ref. 2) and the General Electric 14-0 and 14-1 code package (Ref. 3). Detailed consideration should also be given to the General Dynamics C-17 code (Ref. 4) to establish the accuracy of the unique procedure it uses to predict gamma ray spectra and penetration through multi-layer arrangement consisting of different materials. Any comparison of shielding analysis methods must include these three programs since they essentially represent all of the point kernel methods that are in widespread use in high-speed computer techniques. A brief description of these programs and the distinguishing features of each are identified in the following sections.

a. Los Alamos - Program QAD-P5 (Ref. 2)

The output of Program QAD-P5 includes neutron flux, neutron spectra, neutron dose rate, gamma energy flux, gamma dose rate, uncollided flux, and gamma energy deposition in up to four materials chosen by the user. The neutron spectrum and dose rate kernels are the same as those used by 14-0 and 14-1, but here removal cross sections are used in making thickness transformations to "reference" material thickness. In general, the radiation penetration theories and assumptions employed are similar to those of 14-0 and 14-1, with the exception that QAD calculates no gamma ray spectrum.

A single source region (in contrast to 6 in 14-1) with a single spectrum is treated by QAD. Spatial distribution of the source is represented by separable functions of r and z in the cylindrical coordinate system. These r and z functions may either be represented by point values or constants of fits to cosines. Source normalization is conveniently performed by the code. Unlike 14-0 the equivalent source points are placed at the center between integration mesh points and a tendency towards under estimation may occur for a given calculation of the integrand if it is concaved upwards significantly.

One of the most important features that distinguishes QAD from the other codes is in the geometric description of the source-shield assembly. The use of the three dimensional quadric equation permits geometric description to an almost arbitrary degree of correlation with actual configurations.

b. General Electric Computer Program 14-0 and 14-1 (Ref. 3)

Program 14-0 and 14-1 are written in the FAP system for use on the IBM 7090/7094. These codes evaluate neutron and gamma fluxes, spectra, and energy deposition rates (dose or heating rates) in and around shield systems containing multiple sources specified in a cylindrical coordinate system. Any or all of these radiation quantities may be calculated as the user wishes. 14-1 differs from 14-0 primarily in that it can accommodate a more general description of the source.

Practically all point kernel, gamma ray calculation methods utilize the penetration data of Goldstein and Wilkins (Ref. 5). Gamma buildup factors are input in terms of curve fit coefficients to a cubic polynominal. These data are then used, depending on the problem and user option, to evaluate single medium buildup factors or two-layer build-up factors. The two-layer buildup factors are based on Kalos' empirical formulas (Ref. 6) for a configuration of a heavy material followed by a light or vice-versa. In addition, these codes also calculate gamma spectra based on the assumption of a single homogeneous medium. Spectral data again are those of Ref. 5 and are defined in terms of coefficients to bi-variant (in energy and penetration depth) curve fit.

The trapezodal integration rule is used in 14-0 and 14-1. Since the integrand in question is mathematically concave upward with respect to spatial coordinates, experience at Aerojet has shown that poorly chosen integration mesh spacings can lead to significant over-estimations, especially when the detector is situated close to the source. This problem is aggravated further when the detector is in the source region iteself, for now an improper integral results; i.e., one whose integrand has a singularity. This source of numerical error is minimized depending on the user's judgement in selecting integration mesh spacing; it can be eliminated entirely by making a separate analytic estimate of the contribution due to a small region surrounding the singularity.

Dose rate for neutrons is calculated with the modified Albert-Welton point kernel (Ref. 7). The original expression of the kernel derived by Albert and Welton has been modified to agree with experimental measurements obtained at the GE-ANPD Source Plate Facility at Battelle Memorial Institute.

Neutron spectrum may also be obtained with use of spectrum kernels such as those generated by Nuclear Development Corporation of America (Ref. 8, 9, and 10) for a point fission source in an infinite homogeneous medium. The spectrum is normalized within the program to yield the Albert-Welton dose rate when integrated over energy with tissue response. The spectrum data are accepted by the codes in terms of coefficients of a bivariant polynominal curve fit. Since the system for which spectra are sought must be defined in terms of a single material, the user has to decide the single material which best represents the system. Defining this material as "reference", neutron path lengths in all other materials are assigned "effective" densities and are thereby converted to equivalent thicknesses of the "reference" material. C-17 and QAD on the other hand specify the use of fast neutron removal cross section as bases for material substitution. In either case, "effective" density or removal cross section are input quantities and may be adjusted. Therefore, the theories used by the programs 14-0, 14-1, C-17 and QAD in calculating neutron spectrum are in essence equivalent and the investigation of relative advantages between these codes must be sought in terms of user convenience, generality of output, and computer running time, rather than accuracy. The spectrum is normalized with the program so that it yields the Albert-Welton dose rate when integrated over energy with the tissue response.

The regions external to the source are described geometrically by a combination of sub-regions formed by rotation of rectangles and trapezoids and by translations of quadrilaterals. The material description of the entire system is accomplished by defining the composition of each material for each region.

The source regions are described in cylindrical coordinates, whose z-axis must correspond to the reactor axis. All source distributions are required to have cylindrical symmetry about the z-axis. In 14-0, the source distribution is represented by separable functions of z and r and for a given problem only a single source spectrum is allowed. A source distribution which is not separable in z and r is permitted in 14-1 and this requires the specifying of the strengths of each z, r source ring which compose the distributed source. Up to 6 different gamma spectra may be used to characterize the source. This feature, therefore, permits the calculation, in a single, run of dose rates from a multi-source assembly with each source region having its characteristic spectrum.

b. General Dynamics Program C-17 (Ref. 4)

Program C-17 differs from most programs, including 14-0 and 14-1 and QAD P5, in one important respect, namely in the calculation of gamma ray penetration. Since the C-17 method is comparatively more complicated in theory, with a corresponding increased running time and complexity in operation, its use must primarily be justified by the accuracy that might be obtained for the problems of interest. Should superior accuracy of C-17 gamma ray calculations be borne out in comparison with measurements obtained from certain key experiments or reliable Monte Carlo results, this unique procedure of C-17 should be included at least as an option in a generalized radiation analysis program.

The essence of the C-17 procedure can be understood by the following steps of the code in the penetration calculation for a point source preceding a multi-slab configuration: The spectrum at the first interface is calculated on the basis of the material just preceding using the moments method data of Goldstein and Wilkins (Ref. 5). This spectrum is then used as the source to calculate penetration into the following interface, now on the basis of the material of the second medium. This procedure is repeated till the detector is reached. The usual geometric inverse distance square factor of the attenuation kernel is based on the total distance between source point and detector; this is an assumption equivalent to saying that the interfaces (as opposed to the different regions themselves) encountered along the gamma ray path has no effect on penetration. The energy absorption rate is obtained by the integral over the energy of the spectrum with the appropriate response function.

Program C-17 treats a distributed source by summing the effects of an equivalent array of point sources. The strength (e.g., photon/sec) of each point source is specified along with its corresponding coordinates. This results in a rather tedious preparation of input which can be greatly alleviated by the auxiliary code, R-29 (Ref. 11).

The basic gamma penetration data (Ref. 5) are input in terms of energy dependent curve fit coefficients to functions defining spectra as a function of penetration and atomic number of the medium. Such coefficients must be included for all materials composing the system. Extensive tabulations for a wide range of materials have already been generated (Ref. 11) and should be adequate for analysis of nuclear rocket and nuclear electric systems.

Program C-17 allows an option to apply a void interface correction factor for detectors located in a void. These factors are energy dependent and are based on existing data obtained by the Monte Carlo Method.

The geometry routines of the C-17 are approximately equivalent to those of the 14-0, 14-1 package and therefore need no elaboration.

2. Monte Carlo Methods

A Monte Carlo analysis in its most elementary form, (i.e. one which takes account of the various operative physical events in an exact simulation) is an experiment very much in the sense of noting the distribution of spots that appear in an academic experiment involving a series of throws with a pair of dice. In actual practice, there is departure from this strict analog approach. To reduce statistical variance the rules of the "game" are relaxed in part to permit some deterministic calculations, or may even be altered from an exact simulation with a subsequent corrective calculation. The various devices that are used to reduce variance are well established (Ref. 12) but their efficient application to a large extent remains a matter of individual judgement and experience.

Because nuclear events are probabilistic in nature, it is natural to turn to Monte Carlo methods in radiation transport problems that are intractable to deterministic methods. The capability of the Monte Carlo method is limited only by uncertainties in the input data and computer requirements. It cannot be overemphasized that this limitation is a real one, a fact that is not often appreciated by the inexperienced and one which leads to nullification of all the hoped-for advantage of the method.

The following sections provide brief descriptions of the existing Monte Carlo programs that should be included in the comparison. These programs are described in an order not governed by preference but rather in an order that facilitate their description.

a. General Electric Program FMC-N and FMC-G (Ref. 13 and 14)

Program FMC-N and FMC-G (Flexible Monte Carlo Program-Neutron and -Gamma) are, respectively, a generalized neutron and gamma transport program designed to accommodate a large variety of source-shield problems. The description of geometric surfaces in terms of the three dimensional quadric and the various source options available permit a close simulation of most configurations encountered in practice. Its capability to simulate, to a high degree of correspondence, the configuration and source characteristics of a nuclear rocket engine sytem has been verified at Aerojet in certain gamma heating and leakage problems.

For both gammas and neutrons, the outputs of FMC include tallies of absorption or energy deposition, entrance, leakage (applying to external regions only), as well as flux and history of particles reaching selected regions. The flux tally is actually a quantity that is proportional to flux; it is a tally of total track length which, when divided by region volume, yields a region-averaged flux. The history tally consists of information on angle, spatial and energy coordinates at entry point, the identification of the region from which each particle originated, record of prior collisions, and the statistical weight of the particle entering the region in question.

The angular, spatial and energy coordinates of a source particle may be entered into the program as input from a tape, cards, or the source generator routine of the program. Using the source generator routine, four to seven sets of tables, depending on user option, must be supplied to define source characteristics. Spatial and energy coordinate information must always be supplied whether or not a uniform distribution applies to either coordinate. No angular coordinate data need be input for an isotropic emission of particles. Angular asymmetry with its corresponding sets of data may be introduced with respect to either one or two direction cosines. The source characteristic tables from which initial coordinates are selected consist of sets of probability-coordinate pairs: $(1, X_o, \ldots P_i, X_i \ldots 0, P_n)$ where the P_o, X_i represents the probability P_i with which coordinate X occur with magnitude X_i or less.

The source-shield configuration is described by a composite of subregions whose surfaces are defineable in terms of the general quadric,

$$AX^2 + X_oX + BY^2 + YoY + CZ^2 + Z_oZ - K = 0$$

where X, Y, and Z are cartesian coordinate variables and the remaining quantities, input constants. Five other reduced forms of the general quadric are allowed for efficiency in computation and data preparation. Input of the concentration of the various materials completes the physical description of the source-shield assembly.

The microscopic cross sections required by FMC-N are for (1) elastic scattering, (2) inelastic scattering, (3) fission (including the N, 2n reaction) and (4) total reaction. For FMC-G the cross sections required are for (1) Compton scattering, (2) photoelectric effect, and (3) pair production. The absorption cross section for netrons is given by the difference between the total and the sum of the three remaining cross section types. These cross sections are represented as a function of energy in the program as straight line segments connecting values at the energy group bounds of the bins defined for tally purposes. This connection between the energy mesh points of the cross sections and energy group bounds is one of the disadvantages of FMC as it limits the flexibility desired for a more detailed description of the variation of cross section with energy.

In FMC, and probably in all generalized codes of this class, a particle history is not terminated by the absorption event. Rather, as a standard variance reduction scheme, the prevailing neutron weight is reduced by the non-absorption probability at each collision. In FMC the history of a particle is terminated through any of the following processes: (1) degradation of energy to a specified cut-off, (2) reduction of weight to a specified cut-off, and (3) particle leakage out of the system.

The angular distribution of particles scattered elastically or inelastically may either be isotropic or anistropic. For anistropic scattering, tables of probability versus cosine of scattering angle are required.

Secondary gamma rays from electron-positron recombinations may also be processed as desired in FMC-G. In FMC-N the information in secondary gammas arising from neutron absorption and inelastic scattering may be stored on tape at the user's option and subsequently processed in FMC-G. If this option is requested, the necessary spectra and intensity information must be provided.

The fission event is considered in FMC-N both as a source of local energy deposition and additional neutrons. The n, 2n reaction is treated as a fission event without local energy deposition. The coordinates at birth of fission neutrons are stored in memory and are processed after all the histories of the initial neutrons have been followed to completion.

A number of variance reduction techniques are available in FMC as options. So-called acceleration factors may be used to perform systematic sampling on the source spectrum. The other purpose of the acceleration factors, that of performing importance sampling on the source spectrum, is nullified by an incorrect weight correction by the program (see Page 154, Ref. 13).

Splitting and Russion Roulette may be used at boundary crossings or on energy change due to collision by the appropriate assignment of importances to regions and energy groups. Splitting on change in location, direction, and energy at collision, called "fine-splitting" is also available as an option.

b. General Electric Program 18-1 (Ref. 15 and 16)

The General Electric Program 18-1 is designed to calculate energy deposition and fluxes in each shield region; energy-angle leakage distribution for neutrons and gammas for a point source equivalent to the assembly, or, optionally, a tape record of the escaping particles; and a collison parameter tape for later processing to obtain point flux and heating values through the use of an auxiliary code 20-9 (Ref. 17).

The shield region description of 18-1 is not as general as that of FMC, being limited to regions formed by a rotation of a class of simply connected quadrilaterals about the reactor-shield assembly axis. This limitation is not so serious as to nullify its ability to treat configurations typical of nuclear rocket or nuclear electric systems.

The spatial and energy coordinates of source neutrons and gamma rays whose histories are to be followed are generated by an auxiliary code 20-0 (Reference 18). This information is stored on tape and used as input to 18-1. The angular coordinates, however, are still generated in 18-1.

Neutron reactions treated by the program are (1) elastic scattering, (2) inelastic scattering, (3) radiative capture, (4) n, a reaction (5) n, 2n reaction and (6) absorption with no secondary emission. Gamma ray events treated are, (1) Compton scattering, (2) absorption (photoelectric and pair production) and (3) the photoneutron reaction. The cross sections for all nuclear events are input in the same way as in FMC.

The angular distribution of scattered neutrons may include anisotropy in which case the necessary input distribution data must be provided. The angular distribution of a gamma following a Compton scattering is obtained by the rejection technique (in contrast to tabular distributions required by FMC) from the Klein-Nishina distribution.

Aside from the use of the non-absorption probability at each collision, 18-1 provides for the use of splitting and Russian roulette on energy and region for neutrons and on region for gamma rays. There is no provision for the use of an exponential transformation.

c. Los Alamos General Monte Carlo Code, MCS (Ref. 19)

The code MCS is a general purpose Monte Carlo neutron transport code, written in the FLOCO coding system for the IBM 7090 computer. Its geometry description capability is as general as that of the FMC program.

All reactions of importance are considered. The cross section data routine has a desirable feature (not found in FMC or 18-1) in that the cross section energy mesh points are completely arbitrary and independent from the energy mesh points selected for tally purposes. Different energy points may be selected as desired for the different nuclei. The cross sections between energy points are interpolated according to a $\sqrt{E}$ variation above thermal energies. In the thermal energy range the scattering cross section is assumed to be a constant, the absorption cross section to vary as $1/\sqrt{E}$ and the total cross section either, according to user option, to vary as $1/\sqrt{E}$ or to be a constant.

A desirable feature of MCS is its automatic treatment of the inelastic scattering event, which is defined to include the fission and the n, 2n reaction. A very detailed description is possible with options available to accommodate all of the usual interaction models. The initial/final energy and angle relations may be input either in the form of tabular values for the n, n^1 event or be treated within the code by the evaporation model or level excitation model. The n, 2n reaction can similarly be treated by any of the number of possible models.

The variance reduction schemes available in MCS are splitting, Russian roulette, and exponential transformation. In addition there is the built-in use of the statistical estimation for non-absorption at each collision.

The prime disadvantage to the use of MCS is that it is not a self contained package. Its use will require the coding of auxiliary routines to provide source data and to facilitate the reduction of the history information generated by MCS.

d. Oak Ridge Program 05R (Ref. 20)

Program 05R, written for both the CDC-1604-A and IBM-7090/7094 computer, is a versatile Monte Carlo neutron program suitable for a wide variety of reactor physics and shielding problems. It is distingusihed by its capability of a detailed representation of cross sections covering the range of 77.13 Mev to 0.07×10^{-3} ev. The utilization of so much information is made possible by a sequential processing of batches of neutrons through energy spans containing cross section data points of manageable numbers, thus requiring storage in machine memory of only those cross sections that are in immediate need. In other respects, 05R appears to be approximately equivalent in capability to MCS.

The output of the program is essentially a history tally tape. The versatility of the code is obtained in part at the expense of requirements imposed on the user to write his particular routines to initiate the program and to analyze the history information generated. These include routines for source generation, description of inelastic reactions, and history tape analysis. It is evident that the use of 05R would require a significant programming capability on the part of the user not often available to organizations whose activities do not extend into detailed shielding research.

3. Discrete S_n Transport Method

The discrete ordinates, or S_n method, is a numerical technique developed by Carlson (Ref. 21) of Los Alamos to solve the neutron transport equation. This technique accounts for all the necessary variables; energy dependence is in the multigroup approximation, angular dependence by discrete ordinates of equal weight, and spatial coordinates by variable spacing mesh points.

At present there are a number of discrete S_n codes, DSN, DTK DTF, GAPLSN, (one-dimensional), and DDK and 2DF (two-dimensional, XY, RZ, RΘ). (Ref. 21-27). These codes can be used to solve both the homogeneous problems where an eigenvalue is to be determined and the inhomogeneous problems where source terms are present. Among the homogeneous problems are determination of the effective multiplication factor, inverse period, material concentration, or geometrical size. Among the inhomogeneous problems are the determination of flux distributions in configurations which have surface sources or volume distributed sources. These problems are solved by source iterations involving the overall mesh sweep on all variables. Outputs from programs include the eigenvalue, neutron flux as a function of energy and position, source as a function of position, components of the neutron balance, and volume-integrated sources and reaction rates.

The discrete ordinates methods are receiving considerable interest in regard to their applicability for prediction of both neutron and gamma ray distributors. As a result, FORTRAN versions have been developed for the one-dimensional as well as the two-dimensional program The one dimensional FORTRAN programs include DTF, DTF-2, and DTF-4 as programmed by United Nuclear Corporation, Atomics International Inc., and Los Alamos Scientific Laboratory, respectively. The two dimensional FORTRAN programs include 2DF and TDC as programmed by United Nuclear Corporation and Pratt and Whitney Aircraft - CANEL, respectively.

The DSN, DTK, DTF, DDK, and 2DF programs will all allow anisotropy of scattering up through a P_1 component of the Legendre expansion in the laboratory system. GAPLSN, a one-dimensional program, will allow treatment of higher orders of anisotropy. But use of the anisotropic matrices in two-dimensional codes will result in a significant reduction in capability to define the problem geometry. However, this limitation may not be severe when larger capacity (64-K) computers are used.

a. General Use of the Discrete S_n Method

The discrete S_n method is the basic criticality calculational tool for neutronics systems in which the angular distribution of neutron flux is highly anisotropic. However, there are certain limitations even in these calculations. Due to the separate discrete angular equations, the computing machine time required greatly exceeds that for the usual diffusion calculations. Also, for the regions of low material density, machine time can can increase greatly, and numerical instabilities have occasionally arisen. The limitations on treatment of anisotropic scattering result in limited accuracy on systems in which the materials have large anisotropic differential scattering cross sections, e.g., hydrogenous systems.

The discrete S_n calculations have not been widely used in shielding analysis for following relatively simple reasons. For deep penetration problems, the anisotropy of the flux requires a high order S_n approximation and hence long computing times, and regions of low material density increases the machine time even more. In addition, the existing limitations on the anisotropy of scattering cause the flux of particles reaching a large distance from the source to be underestimated Another reason for the infrequent use of these discrete S_n calculations in shielding analysis is that three separate calculations have to be prepared for each shielding problem. These are for the primary neutrons and gamma rays coming from the reactor and the secondary gamma rays resulting from absorption, inelastic scattering, etc. This increases the overall time from start to finish of a specific shielding problem.

b. Specific Application of the S_n Methods

A large fraction of the radiation analysis problems in the nuclear rocket program do not involve deep penetration problems. But, these problems do require accurate definition of the angular and energy distributions of both neutrons and gamma rays. The S_n approach has provided satisfactory treatment for most of these problems. Figures 1 and 2, respectively, show some of the TDC and DDK predicted spatial and angular distributions for fast neutrons. The streaming patterns around the internal shield indicate that adequate treatment is required for neutron angular distributions in peripheral locations of the reactor. The Aerojet copy of the DDK code provides an edit of the three-dimensional angular fluxes at the configuration boundaries. Figure 3 shows the discrete angular directions for which neutron currents are listed. An interpretation of those results were presented at the 1963 Winter Meeting of the American Nuclear Society (Ref. 27A). A brief summary of the angular current data follows:

In the S-6 approximation of the discrete ordinates method, 24 discrete angular directions are used. Two angular segment sizes are used including solid angles of 0.28797906 steradians 0.2356192 steradians. The angular distribution is symmetrical about the RZ-plane. Definition of the 24 direction angles is given in Figure 3 as obtained from the DDK output listing. The symbol represents the vertical angle with its zero position normal to the Z-axis. The symbol ϕ represents the aximuthal angle with its zero position normal to the RZ-plane.

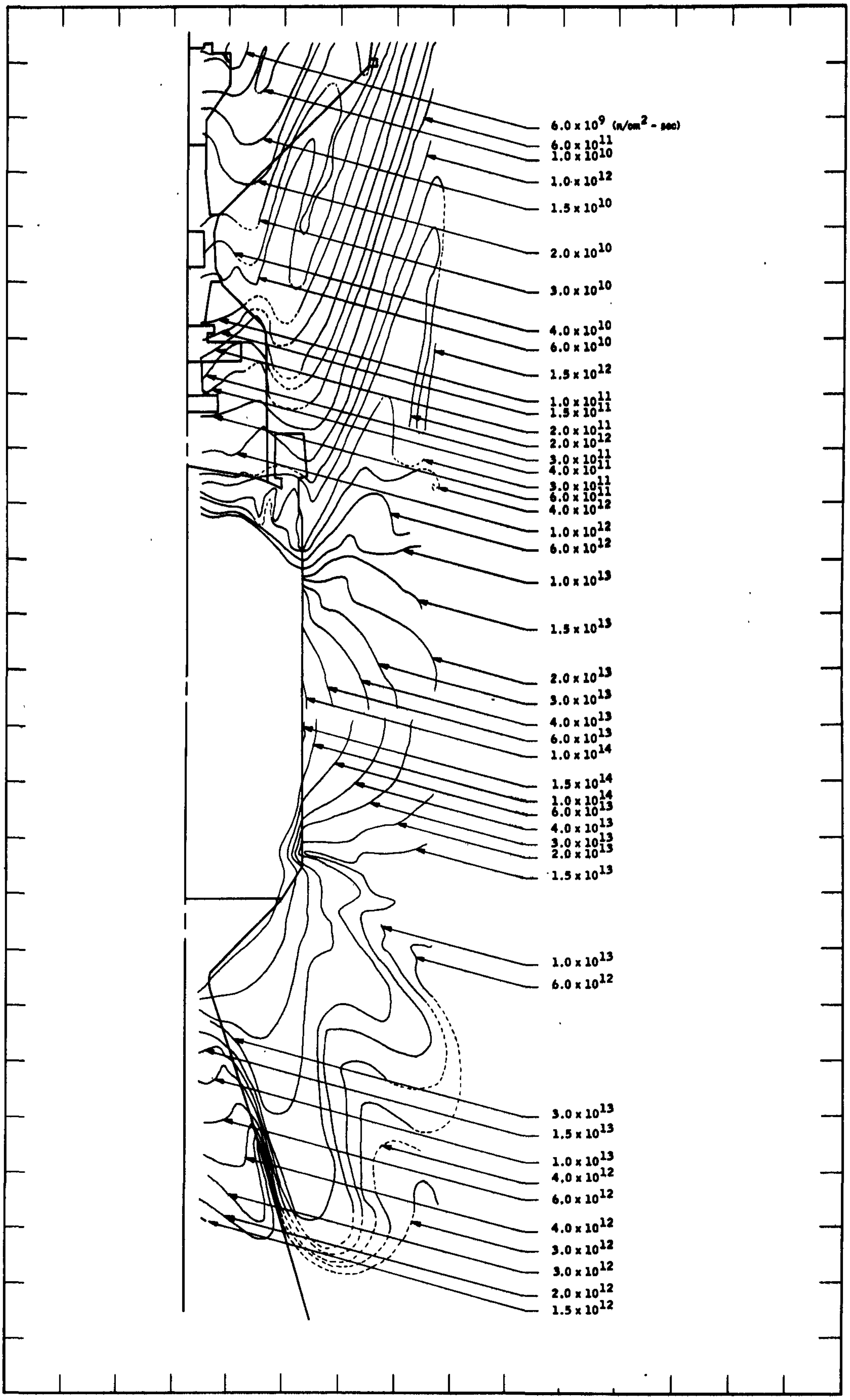


FIGURE 1

S_N PREDICTION OF THE NERVA FAST NEUTRON ENVIRONMENT

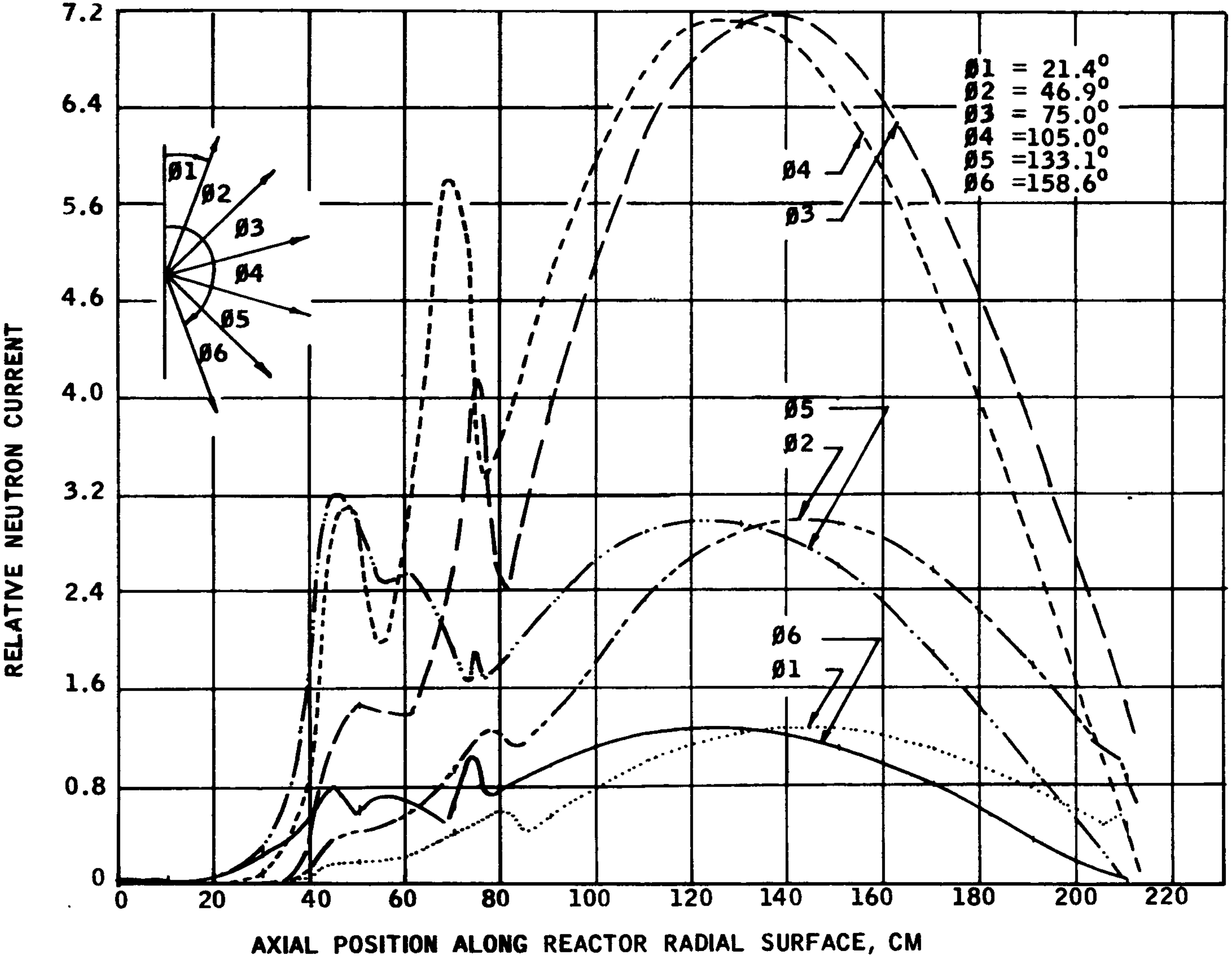
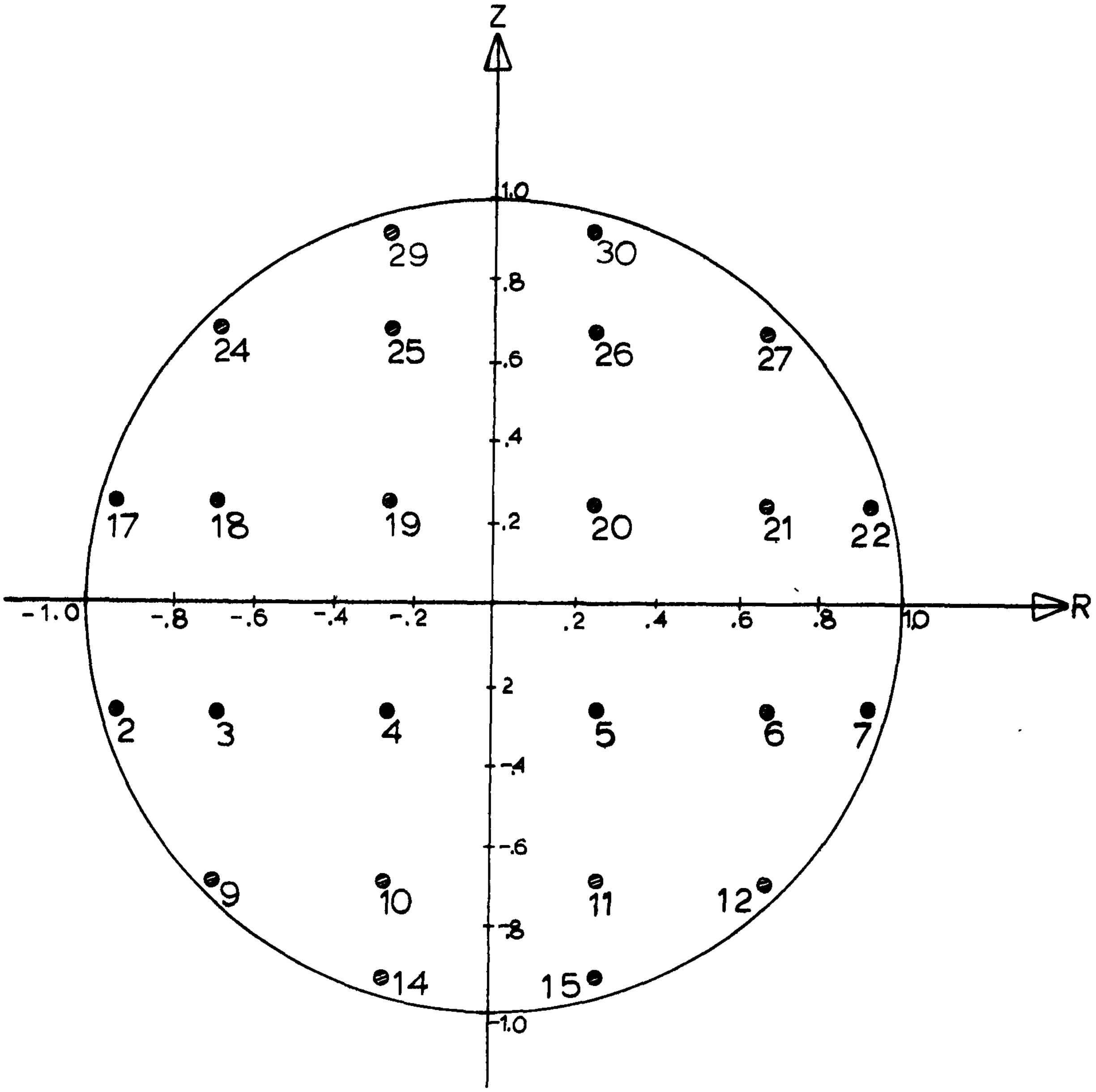


FIGURE 2

RELATIVE NEUTRON CURRENT ALONG REACTOR RADIAL SURFACE



Column No.	Cos θ	Cos ϕ	Column No.	Cos θ	Cos ϕ
2	-.2581989	-.9309493	17	+.2581989	-.9309493
3	-.2581989	-.6831300	18	+.2581989	-.6831300
4	-.2581989	-.2581989	19	+.2581989	-.2581989
5	-.2581989	+.2581989	20	+.2581989	+.2581989
6	-.2581989	+.6831300	21	+.2581989	+.6831300
7	-.2581989	+.9309493	22	+.2581989	+.9309493
9	-.6831300	-.6831300	24	+.6831300	-.6831300
10	-.6831300	-.2581989	25	+.6831300	-.2581989
11	-.6831300	+.2581989	26	+.6831300	+.2581989
12	-.6831300	+.6831300	27	+.6831300	+.6831300
14	-.9309493	-.2581989	29	+.9309493	-.2581989
15	-.9309493	+.2581989	30	+.9309493	+.2581989

FIGURE 3

THE DISCRETE ANGULAR DIRECTION
FOR THE CYLINDRICAL S-6 APPROXIMATION

)

The three paired curves in Figure 3 having crossover points near the 135 cm axial position, define the neutron angular distribution in the RZ-plane. The leakage neutron distribution near 135 cm is symmetrical about a line normal to the reactor surface. But in any other position, the leakage currents are biased in either the upward or downward direction.

Another application is the S_n treatment of neutron thermalization and spectrum prediction in various media. This has been found to be particularly successful in predicting secondary gamma generation rates in shielding media. These secondary gamma generation rates are then input to a point kernel code such as QAD-Q5 to determine the resulting contributions from secondaries.

In addition, the discrete ordinates methods are useful in predicting gamma ray distributions. Recent work (Ref. 28) has shown them to be in excellent agreement with Monte Carlo predictions for gamma ray spectra.

One of the major problems in calculating large systems with the discrete ordinates method is the lack of sufficient memory in computing machines. The best results are obtained by using as many mesh intervals as possible to describe the analytical model without exceeding available memory. This is a trial-and-error scheme for many users of the discrete ordinates method.

At Aerojet, a technique has been devised for predicting the memory requirements as a function of all the input parameters. An example of this technique is shown in Figure 4. With this technique, the problem size can be adjusted as appropriate to make full use of available computer memory.

Will my TDC Problem Fit into the 709-7090-7094?

Factor	Parameter	Cond	No of Loc	Factor	Parameter	Condition	No of Loc
10	G01	U		3	M03	U	
1	G01	3		2	M04	U	
1	G01	5		1	M05	U	
3	G03	U		-1	M06	U	
2	I02	U		13	M07	U	
12	I03	U		1	S02	U	
18	I04	U		12	Rem Cards	U	
4	M02	U					
	Subtotal			→ Enter →			

Factor	Parameter	Condition	
1	G01 x G03	U	
1	G01 x M02	U	
1	G05 x I04	U	
1	I03 x G01	3	
1	I04 x G01	5	
3	I04 x I03	U	
3	I04 x M07	U	
1	M02 x G03	U	
1	M07 x I03	U	
1	G05 x G01 x M02	U	
1	I04 x I03 x G01	U	
1	I04 x I03 x S02	U	
	2D + 3D Constants		11
	Parameters (261)		177
	TDC Program (200)+(200)+(4250)+(100)		2536
	Floco (5500)		2880
	Elbow Room		100
Grand Total (Cannot Exceed Core Size)			

Conditions

3 S04 ≠ 0

5 S03 ≠ 0

U Unconditional

FIGURE 4. Recipe for Estimating Size of TDC Problems

4. Diffusion Theory Methods

The diffusion theory method for radiation analysis has a very limited application. It cannot treat problems containing void regions nor problems involving thick shields. Its principal application in shielding analysis is the prediction of fission power, neutron flux, and neutron capture distributions within the reactor proper and the prediction of infinite media spectra for flux weighting calculations in generating multi-group neutron cross sections.

It is unlikely that any of the diffusion theory codes will be selected as part of the recommended shielding package except as they might exist as an integral part of a cross section code. All of the diffusion theory calculations can be performed equally and better by the S_n transport method, though at some modest added expense. However, the additional expense is small when considering the speed of current computing machines and the frequency that diffusion theory methods can be used reliably.

5. Auxiliary Programs

A number of auxiliary programs are required to prepare and check input data and to process output data. Some of the more important ones currently in use at Aerojet are described in this section. These include the gamma scattering code GGG, the geometric configuation plotter GECOP, the data reduction codes RTDCO and RQADO, and cross section codes.

a. Los Alamos Program GGG (Ref. 29)

Program GGG is designed to evaluate gamma scattering from regions surrounding a point isotropic gamma source. The collision density is evaluated on the basis of the un-collided flux. The dose rate at the detector is evaluated by use of the Klein-Nishina scattering distribution and the application of a buildup factor in the attenuation along the second leg of the gamma path.

The code requires information on collision coordinates, the material distributions that interpose the rays that connect the source, scatter point, and detector. The geometry routine is identical with that of QAD-P5.

b. Geometric Configuration Plotter

The shielding codes currently in use at Aerojet require specification of geometric configurations by use of coefficients to the expression shown below.

$$A(X-X_o)^2 + B(Y-Y_o)^2 + C(Z-Z_o)^2 = K$$

One of the major problems of using this form of geometric specification is that errors are seldom immediately evident. These may be key punch errors, transcription errors, or engineering errors that define configurations significantly different from the desired configuration.

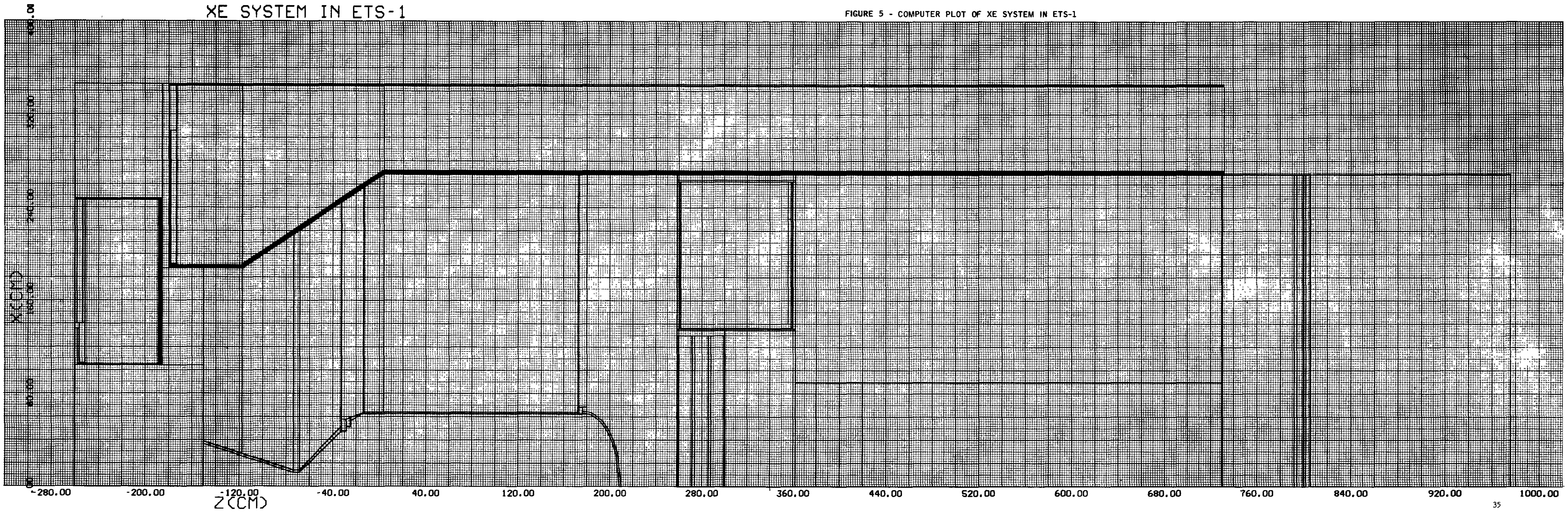
For assurance that the input coefficients actually describe the desired analytical model, a program (GECOP) has been written by Aerojet, which plots the configuration defined by the selected coefficients.

A photo-reduced copy of a typical plot is shown in Figure 5 which is the configuration defined by the abstract coefficients listed in Table II and III. Originals of Figure 5 are provided in a folder in the back cover of the copies of this proposal transmitted to MSFC. This GECOP program should be included as part of the selected analysis method or methods.

c. Data Reduction Programs

Determining the nominal radiation environment for all components of the NERVA engine requires the use of two-dimensional maps of the radiation distributions about the reactor. Data reduction programs, such as RTDCO and RQADO, are used to identify the location of the iso-flux or iso-dose contours on the basis of flux or dose data at selected points in the regions of interest.

FIGURE 5 - COMPUTER PLOT OF XE SYSTEM IN ETS-1



3	9	-28	15	-13	18	29	46			16
4	9	-13	18	30	45	14	46	-29	16	17
4	9	-12	19	30	45	13	17	-21	11	18
4	9	-2	29	22	44	12	18	-21	3	19

Table 2

Zone Description Data for Figure 4

Sheet 1 of 4

[redacted] [redacted]

4	12	-35	31	17	26	3	24	-16	22			23
4	12	-3	23	18	25	4	20	-16	22			24

4	9	-1	30	24	43	2	19	-23	27			29
4	9	-35	31	24	43	1	27	-19	28			30
4	12	-36	32	31	43	35	23	-16	22			31
4	12	-37	33	43	125	36	31	-42	126			32
4	12	-38	34	41	124	37	32	-40	35			33
4	12	-39	37	45	41	38	33	-44	36			34
3	5	-38	36	40	33	37	126					35
3	5	-39	37	44	34	38	35					36
3	13	-57	114	59	38	39	36					37
4	13	-46	64	73	65	39	41	-59	37			38
4	13	-46	40	88	112	67	65	-73	38			39
4	13	-57	114	88	133	46	39	-65	59			40
4	13	-39	37	73	65	70	42	-45	34			41
4	13	-70	41	89	75	38	124	-45	34			42
5	13	-36	125	89	75	2	127	-24	29	-31	31	43
4	13	-71	127	80	81	12	45	-22	19			44
4	13	-12	44	80	81	14	46	-30	17			45
4	13	-29	16	-14	17	80	81	95	87			46
4	13	-95	46	103	86	117	49	-118	98			47
4	13	-95	46	80	81	102	113	-109	88			48
3	13	-117	47	103	86	102	120					
4	9	-55	51	60	52	51	60	-59	37			50
4	9	-57	114	62	54	55	50	-59	37			51
4	14	-54	53	61	56	52	58	-60	50			52
4	14	-55	51	62	54	54	52	-60	50			53
4	9	-57	114	63	55	54	56	-62	51			54
4	9	-57	114	64	59	56	57	-63	54			55
4	15	-54	53	63	57	52	58	-61	52			56
4	15	-56	55	64	59	53	115	-63	54			57
4	14	-52	52	64	59	51	60	-60	50			58
4	9	-57	114	65	40	51	60	-64	55			59
4	9	-51	50	65	40	50	61	-59	37			60

Table 2

Zone Description Data for Figure 4

Sheet 2 of 4

46336

4	14	-50	60	65	40	49	62	-59	37	61
4	16	-49	61	65	40	48	63	-59	37	62
4	14	-48	62	65	40	47	64	-59	37	63
4	9	-47	63	65	40	46	38	-59	37	64
4	9	-67	39	74	66	70	75	-73	38	65
4	14	-67	39	75	67	70	76	-74	65	66
4	16	-67	39	76	68	70	77	-75	66	67
4	14	-67	39	77	69	70	78	-76	67	68
4	9	-67	39	78	70	70	78	-77	68	69
4	14	-69	71	37	80	70	123	-94	79	70
4	14	-68	72	79	74	69	70	-78	69	71
4	14	-67	39	86	73	68	71	-78	69	72
4	14	-67	39	87	80	69	70	-86	72	73
4	15	-68	72	86	73	69	70	-79	71	74
4	9	-70	65	90	76	71	81	-89	42	75
4	14	-70	66	91	77	71	82	-90	75	76
4	16	-70	67	92	78	71	83	-91	76	77
4	14	-70	68	93	79	71	84	-92	77	78
4	9	-70	69	94	123	71	85	-93	78	79
4	9	-67	40	88	112	72	116	-87	70	80
4	9	-71	75	81	82	72	116	-80	44	81
4	14	-71	76	82	83	72	116	-81	81	82
4	16	-71	77	83	84	72	116	-82	82	83
4	14	-71	78	84	85	72	116	-83	83	84
4	9	-71	79	85	122	72	116	-84	84	85
4	9	-95	46	104	87	102	113	-103	47	86
4	9	-95	46	108	88	96	91	-104	86	87
4	9	-95	46	109	48	102	113	-108	87	88
4	9	-101	93	108	88	102	113	-104	86	89
4	14	-97	91	105	93	101	89	-104	86	90
4	14	-96	87	108	88	97	97	-104	86	91
4	14	-98	97	108	88	99	94	-106	96	92
4	14	-100	94	108	88	101	89	-105	90	93
4	14	-99	92	108	88	100	93	-107	95	94
4	15	-99	92	107	94	100	93	-106	96	95
4	15	-98	97	106	92	100	93	-105	90	96
4	16	-97	91	108	88	98	92	-105	90	97
3	9	-110	46	118	47	111	99			98

Table 2

Zone Description Data for Figure 4

Sheet 3 of 4

3	14	-111	98	118	47	112	100										99
3	14	-112	99	118	47	113	101										100
3	17	-113	100	118	47	114	102										101
3	14	-114	101	118	47	115	103										102
3	17	-115	102	118	47	116	104										103
3	9	-116	103	118	47	117	49										104
3	9	-119	113	80	81	120	106										105
3	14	-120	105	80	81	121	109										106
3	14	-122	109	80	116	123	110										107
3	14	-124	110	80	116	125	111										108
3	15	-121	106	80	116	122	107										109
3	15	-123	107	80	116	124	108										110
3	9	-125	108	80	116	126	117										111
7	13	72	116	139	136	-46	133	-127	128	88	39	-129	136	130	136		
4	13	-102	48	80	81	119	105	-58	118								113
7	13	57	37	138	132	135	131	-132	130	-127	129	-140	134	141	135		
4	14	-53	57	64	59	52	58	-63	56								115
6	13	131	136	139	136	-72	81	-127	128	80	106	-129	136	130	136		
3	13	-126	111	80	116	131	136										117
3	13	-32	120	58	113	33	119									0	1118
3	13	-33	118	58	113	34	121									0	1119
3	13	-102	49	58	113	32	118									0	1120
3	13	-34	119	58	113	119	105									0	1121
4	14	-71	123	87	80	72	116	-85	85								122
4	14	-70	70	87	80	71	122	-94	79								123
4	13	-38	42	89	75	37	125	-41	33								124
4	13	-37	124	89	75	36	31	-43	32								125
3	5	-37	35	42	32	36	22										126
4	13	-2	43	89	75	71	44	-22	19								127
6	18	127	112	-46	133	-128	136	131	136	-129	136	130	136				
6	18	127	114	-132	130	-128	136	57	133	-129	136	130	136				
6	18	-128	136	-134	137	133	136	132	114	-129	136	130	136				
6	13	133	136	-132	130	-135	114	136	132	-129	136	130	136				
6	13	137	136	-136	131	-138	114	-139	133	-129	136	130	136				
7	18	-88	40	-57	114	139	132	46	112	-129	136	130	136	-128	136		
7	18	138	132	135	131	-132	130	-127	129	57	133	-142	136	140	114		
7	18	138	132	135	131	-132	130	-127	129	57	133	-141	114	130	136		
1	13																
6	18	134	130														

Table 2

Zone Description Data for Figure 4

Sheet 4 of 4

46338

```
 1   6                                                                      -24.65          2  1
 2   6                                                                      -13.15
 3   6                                                                      -6.75
 4   6                                                                      -4.95
 5   6                                                                      -3.85
 6   6                                                                      +0.0
 7   6                                                                      +132.8
 8   6                                                                      +136.71
 9   6                                                                      +148.61
10   6                                                                      +152.42
11   6                                                                      +155.51
12   6                                                                      +172.85
13   6                                                                      +176.25
14   6                                                                      +179.95
15   3                                                                      +2025.0
16   3                                                                      +2209.0
17   3                                                                      +2323.24
18   3                                                                      +2450.25
19   3                                                                      +2631.69
20   3                                                                      +3636.09
21   3                                                                      +3951.38
22   3                                                                      +4096.0
23   1+1.0     +1.0      -0.3723                                   -86.498  +5035.0
24   1+1.0     +1.0      -0.1443                                   -51.76   +4670.1
25   2  961.      961.     3833.155                                 122.052 3683437.1B    2025
26   2  961.      961.     3833.155                                 136.536 3683437.1B    2026
27   2  961.      961.     3833.155                                 172.100 3683437.1B    2027
28   2 987.216  987.216   3952.008                                  177.000 3901485.5B    2028
29   21097.067 1097.067   4168.897                                  177.000 4573559.3B    2029
30   3                                                                      +4651.24
31   3                                                                      +3314.30
32   6                                                                       380.0     B    2032
33   6                                                                       400.0     B    2033
34   6                                                                       420.0     B    2034
35   6                                                                      -28.45
36   6                                                                      -32.85
37   6                                                                      -68.45
```

Table 3

Boundary Equations Data for Figure 4

Sheet 1 of 4

38	6				-73.45		
39	6				-150.45		
40	3				+144.0		
41	3				+196.0		
42	2+1.0	+1.0	-0.9666	-80.655			
43	2+1.0	+1.0	-1.0978	-81.812			
44	2+1.0	+1.0	-0.1054	-36.49			
45	2+1.0	+1.0	-0.1054	-30.33			
46	6				-185.5		
47	6				-186.3		
48	6				-186.9		
49	6				-187.5		
50	6				-188.1		
51	6				-188.9		
52	6				-251.5		
53	6				-253.5		
54	6				-256.5		
55	6				-258.0		
56	6				-258.6		
57	6				-260.5		
58	3				17956.0	B	2058
59	3				+10816.0		
60	3				+10983.0		
61	3				+11236.0		
62	3				+18225.0		
63	3				+19600.0		
64	3				+60516.0		
65	3				+61009.0		
66	6				-180.9		
67	6				-179.5		
68	6				-178.5		
69	6				-173.35		
70	6				-117.5		
71	6				+3.55		
72	6				+732.0		
73	3				+34969.0		
74	3				+35193.8		
75	3				+35438.1		

Table 3

Boundary Equations Data for Figure 4

Sheet 2 of 4

46340

76	3					+35675.7
77	3					+35910.3
78	3					+36149.4
79	3					+36481.0
80	3					+71824.0
81	3					+72146.0
82	3					+72495.6
83	3					+72835.2
84	3					+73170.0
85	3					+73603.7
86	3					+93025.0
87	3					+118473.64
88	3					+119163.04
89	1+1.0	+1.0	-0.4474		-355.24	+70514.
90	1+1.0	+1.0	-0.4474		-356.26	+70923.
91	1+1.0	+1.0	-0.4474		-357.28	+71327.
92	1+1.0	+1.0	-0.4474		-358.30	+71734.
93	1+1.0	+1.0	-0.4474		-359.31	+72140.
94	1+1.0	+1.0	-0.4474		-360.34	+72554.
95	6					+258.5
96	6					+259.8
97	6					+260.5
98	6					+261.5
99	6					+355.5
100	6					+358.0
101	6					+359.0
102	6					+361.9
103	3					+17956.0
104	3					+18306.1
105	3					+18468.9
106	3					+19544.0
107	3					+52900.0
108	3					+68486.9
109	3					+69169.0
110	6					+258.5
111	6					+259.77
112	6					+271.2
113	6					+273.74

Table 3

Boundary Equations Data for Figure 4

Sheet 3 of 4

114	6							+285.17
115	6							+287.71
116	6							+299.14
117	6							+300.41
118	3							+16770.3
119	6							+729.5
120	6							+730.5
121	6							+791.5
122	6							+794.0
123	6							+799.1
124	6							+801.6
125	6							+805.0
126	6							+805.7
127	4							-579.6
128	4							-670.8
129	5							-1493.5
130	5							547.3
131	6							975.6
132	6							-1951.2
133	4							1706.4
134	6							-2030.4
135	4							1616.4
136	6							-585.6
137	2	1.0		1.0	1226.8		-676.66	241113.2
138	2	1.0		1.0	1226.8		-676.66	173099.3
139	4							1213.2
140	5							-457.2
141	5							457.2
142	5							-547.3

Table 3

Boundary Equations Data for Figure 4

Sheet 4 of 4

These data reduction codes perform scaling, summing, and interpolation functions using punched output data from the TDC, DDK, and QAD-5 programs. The multi-group flux dump from TDC, for example, is read back into the computer for scaling to the desired power level and units in order to sum the desired groups of fluxes and to determining the coordinates of the iso-flux contours. An example of the resulting two-dimensional contours is shown in Figure 1.

d. Nuclear data

Output results of any calculation depend, independent of the refinement of the theory used, on the accuracy of the input data. For shielding and reactor analysis, these input data consist of cross sections usually in a multi-group format. Since these cross sections are formulated from basic experimental and theoretical data, Aerojet maintains several libraries of basic nuclear data. This is a continuing process because new or revised data must be incorporated whenever available. The various basic data libraries are described below. Similar thermal neutron data is available at Aerojet, but since it is of secondary importance in this shielding program, it will not be discussed further.

(1) Differential Neutron Scattering Data Library

The Legendre expansion coefficients of the differential scattering cross sections are required in a variety of calculation techniques, e.g., determination of the multigroup scattering matrices, and Aerojet has written the SPADE code to generate the first six of these coefficients in both the L and CM systems, as well as the various lethargy moments.

All of R. Howerton's (Ref. 30) differential elastic scattering data have been reduced by the SPADE code, and other data are also reduced as they are published. At the present time, this library contains the specified data as a function of energy for about 50 elements.

(2) Fast Neutron Cross Section Library

The basic fast neutron cross section library, which has taken several man-years to develop to its present state, incorporates 75 subgroups of equal lethargy (zero lethargy at 10 Mev and subgroup lethargy width of 0.25). Data required for the library are subgroup averaged cross sections, neutrons per fission, resonance parameters, and angular data as obtained from the SPADE code.

At the present time the library contains about 200 materials with formats compatible with all fast constant generating codes in use at Aerojet.

(3) Cross Section Generation

Aerojet maintains the calculational tools to generate cross sections, in a completely flexible and general manner, for any reactor shielding or physics analysis. Thus, it is possible to generate group constants utilizing the most up-to-date theory for arbitrary group spacing and flux averaging.

The capability of generating general cross sections is important for a number of reasons. Calculated reaction rates for any specific region depend on the group constants which in themselves depend upon the intragroup flux distribution. Thus, to obtain correct reaction rates, the appropriate spectrum in the region must be used to average the group constants.

The ability to vary energy group boundaries is important, both for economical reasons and for accuracy. In multi-dimensional problems whose calculation time is a direct function of the number of groups, one tries to balance the economy of a few groups with the accuracy of many groups to obtain the desired number. Also, the energy regions of interest are not necessarily the same for criticality calculations as for shielding calculations, and therefore, one set of group specifications is usually not optimum for both.

For the shielding codes to be used in this study, MYSTIC (Ref. 31) and AN-GAM (Ref. 32) will be used to generate neutron cross sections as needed for the S_n codes and Monte Carlo calculations. The 20-2 to 20-5 (Ref. 33) series of computer programs will be used to generate neutron and gamma data for the Monte Carlo Calculations.

(4) Existing Multi-group Cross Section Data

Aerojet maintains several neutron cross section libraries for use in criticality and shielding calculations. There is a 16 group S_n transport theory set, based on G. Hansen's (Ref. 34) energy limits which includes cross sections for approximately 50 materials. This set has been used in an extensive calculation check for determining the accuracy of the fast cross section library.

There are a number of collections of gamma ray cross sections available: 20 group gamma constants (Ref. 35 and 36) have been collected and utilized in the S_n codes, and there are also the tabulated gamma data of R. A. Mann (Ref. 37) which are presented in the Monte Carlo format.

B. DEVELOPMENT OF EVALUATION CRITERIA

Comparison and critical evaluation of the candidate analysis methods will require definition of evaluation criteria. In defining such evaluation criteria, consideration must be given to determining the nature of nuclear rocket radiation analysis problems as well as to the availability of suitable experimental data. These considerations will contribute to the assignment of weights or importances for the items in the evaluation criteria.

1. Preliminary Evaluation Criteria

Some of the more important items in the evaluation criteria can be readily identified. These include 1) accuracy and cost of results, 2) adequacy as to format and as to the detail contained in the results, and 3) relative complexity of the input data.

Accuracy of results must be evaluated relative to the accuracy requirements. These accuracy requirements range from the 10-20% for component design up to factors of 5 and 10 for general radiation environment estimates.

Generally, analysis costs vary in some proportion to accuracy of results. A large number of variables have dominant effects on this proportionality so that it cannot be defined in any generally applicable terms. In Monte Carlo analysis methods, costs are affected by the degree to which the specific problem allows use of variable reduction techniques as well as the availability of schedular time during which these techniques may be applied. In the discrete ordinates method calculations, the choice of geometric mesh points, number of energy groups, order of the S_n level, and convergence criteria have dominant effects on analysis cost. And, in point kernal method calculations, analysis costs are affected by the number of receiver points, the number of source points, the geometric configuration, and the number of energy groups. Thus, establishing definitive relationships between accuracy and cost must set constraints on most of the above parameters and the resulting conclusions will apply to the specific case considered. However, if nuclear rocket radiation analysis problems can be characterized by certain degrees of accuracy requirements, these can affect the comparative "goodness" of the different candidate analysis methods and therefore should be a part of the evaluation criteria.

Adequacy of the candidate methods, as to format and to the detail contained in the results, is the second item in the evaluation criteria. This item refers to the convenience and direct usefulness of the output data as well as to whether energy and angular distributions are adequately defined.

In some cases data are given for point locations in space and in others data are given which are integrated over volume elements in space. Moreover, differences also exist in units of the output data as well as in the scaling to specific levels of operating power.

In some gamma ray point kernal calculations, output data are provided for dose rates from each of the source energy groups. These data which contain energy dependent build-up factors do not provide the gamma ray energy distributions required in many radiation analysis problems. Thus considerations such as these should form an important part of the evaluation criteria.

Relative complexity of the input data must also be considered in the evaluation criteria. Efficient use of many of the current computer programs often require an intimate familiarity with the computing logic. In some cases auxiliary computer codes are required for preparation of input data as well as interpretation of output data. Thus, while one group of analysis methods might be given a high rating on the accuracy criterion, it could very well have a low rating on the criterion for complexity of input data. Depending on the importances of these criteria for nuclear rocket and nuclear electric programs, suitable weighting factors can be assigned to each criterion. What is required, therefore, is a survey of the radiation analysis requirement so that these weighting factors can be defined.

2. Characteristic Radiation Transport Problems

As part of the effort to develop criteria for evaluation of analysis methods, it is necessary to identify the radiation transport problems which are characteristic of nuclear rocket and nuclear electric systems. This information will help define the requirements and hence the evaluation criteria for analytical methods. In addition, these characteristic problems will form the basis for selection of test problems to be used in evaluating the effectiveness of the more promising analysis methods.

a. Nuclear Rocket Systems

Nuclear rocket systems are characterized by high power and short duration operation. Radiation intensities are sufficiently high to create serious problems relative to rate effects as well as to integral effects. The various radiation shields that must be considered include the reactor internal shadow-shield, the LH_2 propellant tank, the multiple shields required in the ground test facility, and those engine components located between the reactor and the propellant tank.

In addition to shield design analysis, the nuclear rocket program requires a considerable amount of radiation analysis to define nuclear heating and dose rate distributions within the engine components. To minimize the requirements for reactor shielding, the effects of self-shielding in the engine components must also be considered. To date, these analyses have been performed by the combinations of shielding programs that have been chosen by Aerojet. (Ref. 38-44). A brief description of the Aerojet nuclear codes library is given in Section VIB.

The more difficult problems encountered in these analyses are described below. First is the problem of accounting for neutron spectra from the point of emergence out of the reactor to the point of absorption or leakage from the system boundaries. This requires accounting for the angular and energy distributions of neutrons through large void regions or low density media followed by treatment of the thermalization, absorption, and leakage problem in solid or liquid media. A specific example is the definition of the radiation distributions in the turbopump assembly (TPA). Radiation sensitivity of the TPA is spatially dependent within the TPA. The TPA envelope for NERVA is nominally a cylinder with 36-inch diameter and 48-inch length and its materials include LH_2, Inconel-718, titanium, and aluminum. Thus it is necessary to predict the transport of neutrons and gamma rays through void or low density regions into the TPA, to predict the thermalization and absorption of neutrons within the TPA, to define the resulting sources for secondary gamma rays, and finally to determine the resulting spatial distribution of total heating rates and total dose rates within the TPA.

Another difficult problem is definition of the radiation environment at the nuclear rocket payload. In single engine configurations, most of the integrated exposure at the payload occurs during the terminal 5 or 10 minutes of operation. (Ref. 43). The important contributors include direct gammas, LH_2 capture gammas, tank wall scattered gammas, and tank wall scattered neutrons. Adequate treatment of these dominant contributors requires the use of analysis methods that can handle both the energy and angular dependence of the scattering process. Albedo methods, which make use of empirically determined reflection coefficients, provide suitable estimates for these contributors in some cases.

The problems of radiation analysis became especially severe in the analysis of ground testing of nuclear rockets. Exposure of components in ground tests must be predicted for component design and shielding design where the expected exposure is too extreme. An example of such a problem is the development of the NRX/EST test of the NERVA technology engine. The radiation analysis required for this test is described in Figure 6 and documented in Ref. 44. Comparable analyses have been conducted by Aerojet to support the design of the Phoebus-2 nozzle. A description of this analysis is shown in Figure 7 and documented in Ref. 41. The contracting agency was sufficiently impressed with the Aerojet work that it requested the analysis be extended to

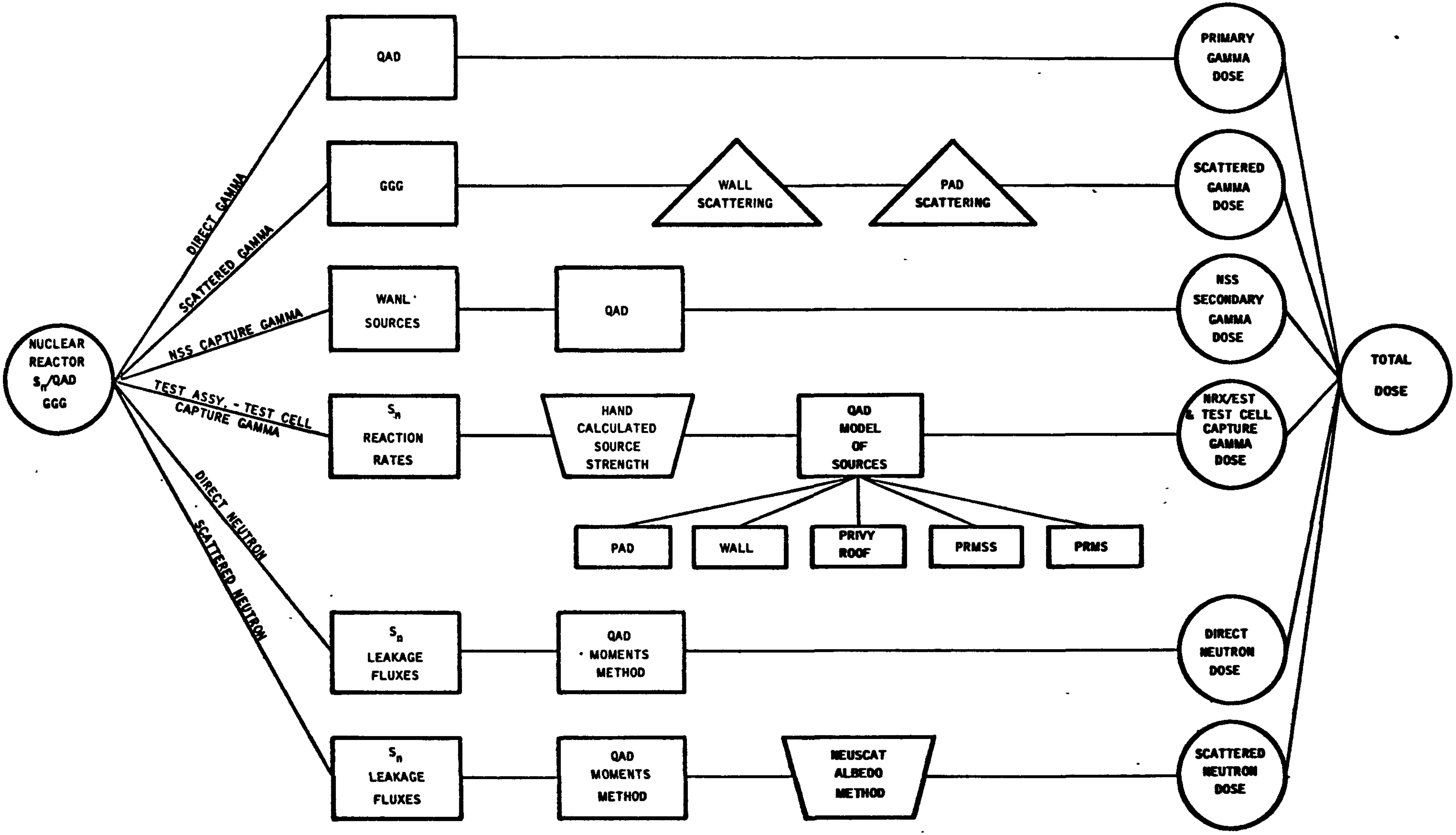


FIGURE 6

FLOW SCHEMATIC OF RADIATION ANALYSIS FOR NRX/EST

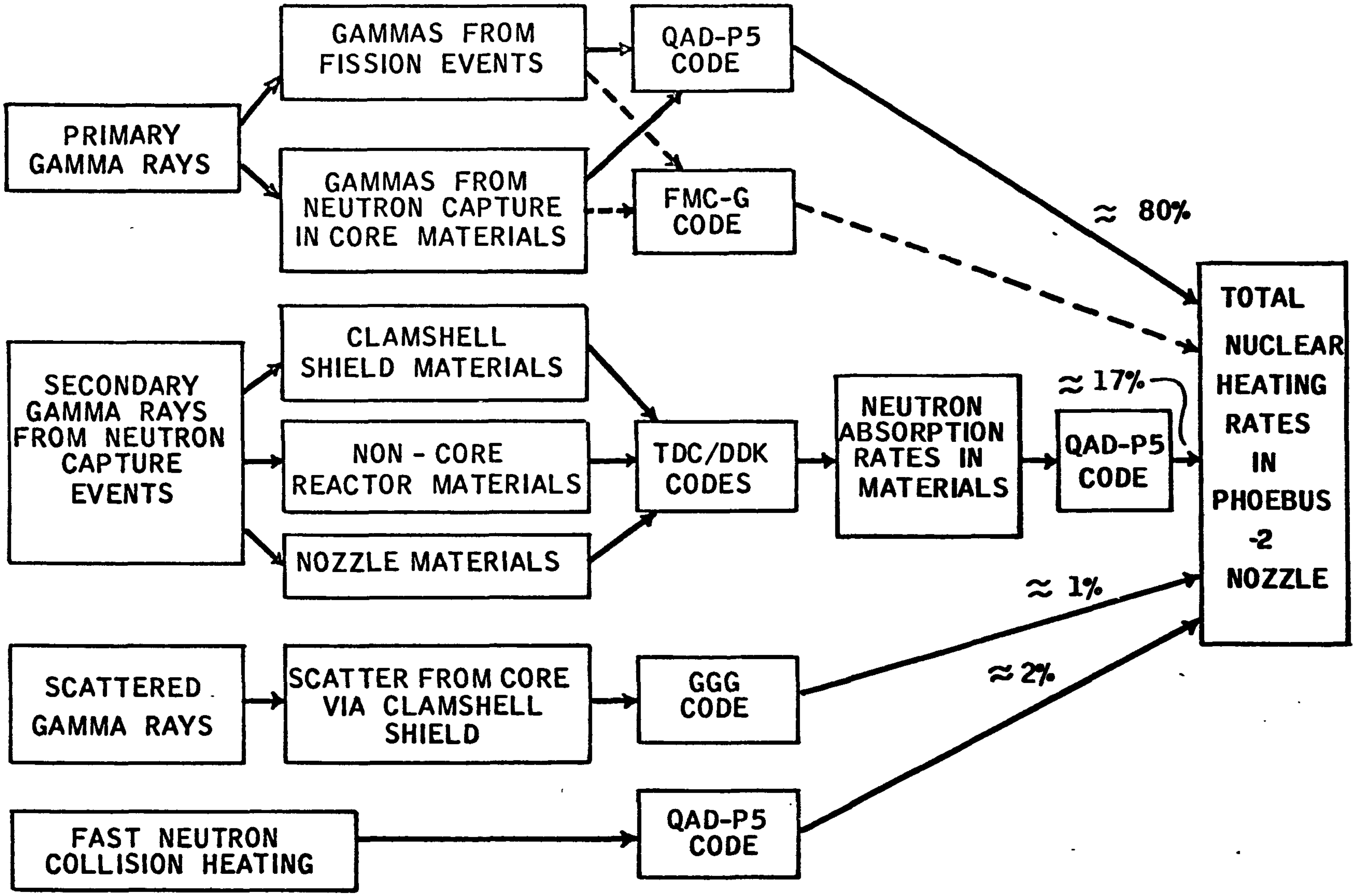


Figure 7. Elements of Nuclear Heating in the Phoebus-2 Nozzle

include the pressure vessel design requirements as well. Note that the methods used include various combinations of Monte Carlo, discrete ordinates, point kernel, as well as auxiliary methods.

b. Nuclear Electric Systems

Nuclear electric systems are characterized by significantly lower power levels than nuclear rocket systems, i.e., about three decades. Thus the radiation intensities are far less hostile than those in nuclear rocket systems. However, because of its significantly longer operating duration, i.e., three to four decades, the tolerable radiation intensities are far below those in nuclear rocket systems. For this reason, reliable prediction methods are equally important for nuclear electric systems.

One of the more difficult problems is related to the radiators of nuclear electric systems. Figure 8 provides a perspective of the SNAP-8 system. The large radiator, which is exposed to the reactor leakage flux, serves as a scattering medium for neutrons and gamma rays as well as a significant source of secondary gammas. These secondary gammas are produced with the absorption of leakage neutrons in the radiator materials. As with the nuclear rocket system problems, this radiator problem requires treatment of energy and angular distributions in media that are separated by large void regions.

3. Collection of Suitable Experimental Data

Experimental data will be needed for evaluation of the adequacy and accuracy of analysis results. Accuracy evaluation in many instances is performed by comparing the results from different analysis methods and/or from different analysis groups. Accuracy, in these comparisons, is considered adequate when there exists some general agreement in the different predictions. When good agreement does not exist, preference is usually given to the results generated by the more sophisticated analysis method. However, consideration must also be given to the

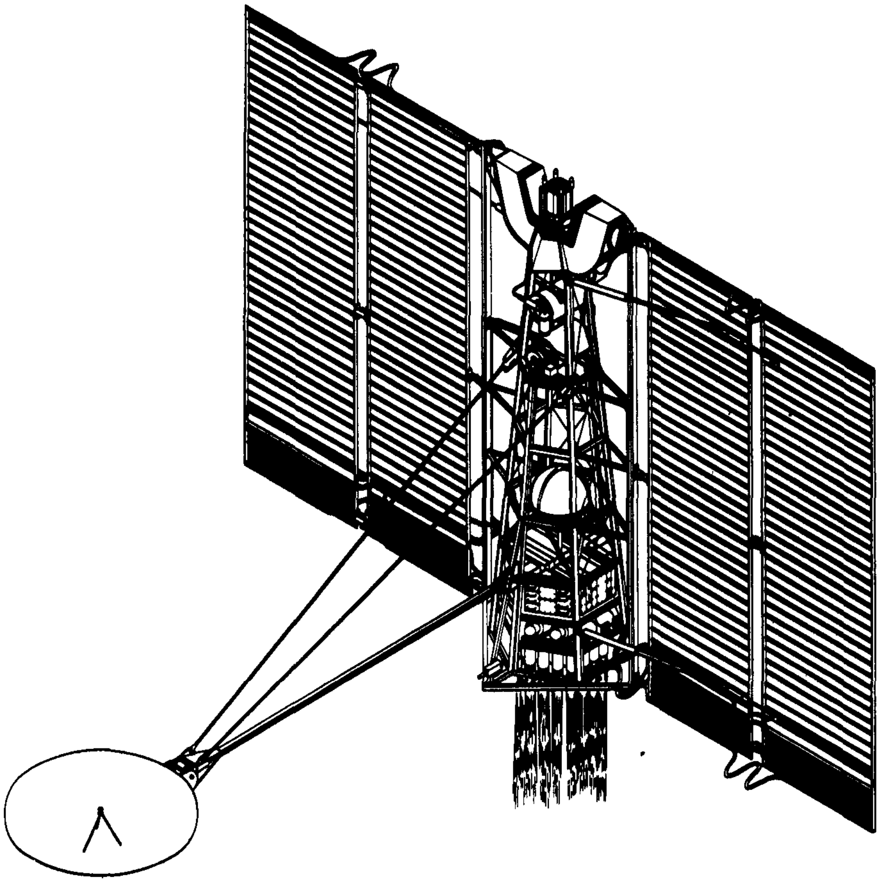


FIGURE 8. SNAP-8 INTERPLANETARY VEHICLE, EXTENDED CONFIGURATION

relative adequacy of the analytical models in both geometric configuration and material compositions, and to the prediction uncertainties associated with the degree of convergence or statistical variances.

The use of suitable experimental data provides considerable relief to this problem of accuracy evaluation. However, it is also important to understand the uncertainties associated with the measured data. The collection of suitable experimental data must include definition of the conditions under which the data were obtained as well as the various normalizations and adjustments made to the measured data. As prime contractor for the NERVA program, Aerojet has direct cognizance for much of the dosimetry work and therefore has the required familiarity with the measured data.

A considerable amount of useful data have been obtained in the course of the recent NRX/EST test conducted by Aerojet and its subcontractor, Westinghouse. This is the first test of a combined reactor and engine system. These measured data will allow evaluation of the methods for predicting energy and some angular distributions of radiation about the test article.

Additional data are expected from the Phoebus-2 firings which will more closely represent the flight configuration reactor and nozzle.

Past data taken in the course of the Kiwi-B and the NRX-A programs are also available. Appendix A provides some comparisons between predictions and experiment for gamma ray dose rates, neutron fluxes, and neutron spectra. These data indicate that the analysis methods and nuclear data selected to date by Aerojet have provided results that are in good agreement with measured data. Agreement is good in both the spatial and energy distributions of the leakage radiation. Data defining angular distribution, except as angular distributions affect spatial distribution, are not yet available for comparison with prediction.

Measured data will also be collected for basic configurations such as a point source in an infinite medium. These data are well documented in the open literature.

In addition, measurements obtained in the course of the SNAP-8 program and SNAP-50/SPUR program will be explored. If available, they will be used in the design of test problems as well as in the evaluation of prediction accuracy.

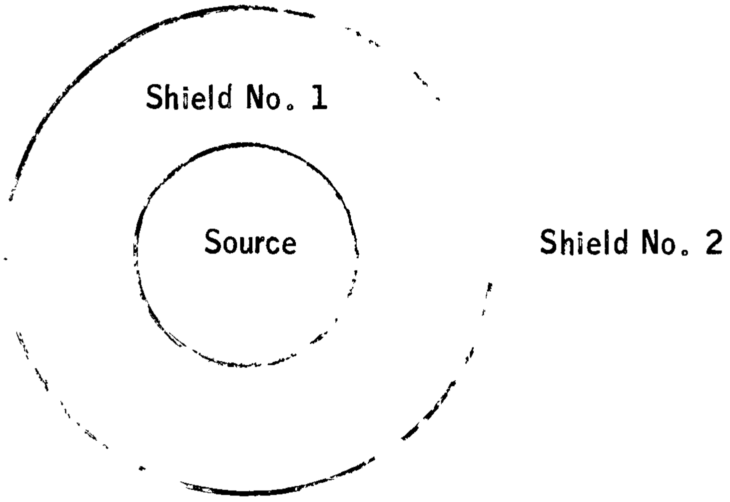
C. SELECTION OF TEST PROBLEMS

Test problems will be required for clean configurations and for actual configurations. Extensive amounts of reliable data exist on clean configurations which can serve as a basic check on the analytical method as well as on the nuclear constants used.

1. Basic Configuration

The final test of a calculation method for application in the analysis of nuclear rocket and nuclear electric systems must be based on how well it predicts radiation quantities in realistic configurations. The prediction for a realistic system, however, is made difficult by complexity in geometry, material distribution and sources of radiations. The discrepancy between calculation and measurements are therefore the net effect of various errors of various origin. In order to help isolate the specific causes of discrepancies between generalized calculation methods, on the one hand, and calculational results and measurements on the other, a comparison of results for "clean" basic configurations will be made.

The basic configuration that will be considered is a three-medium spherical system as shown below:



The inner source region is followed by two concentric shield regions. The special case of a point source will be included in the study in order that comparisons may be made with the experimental neutron point kernel for water (derived from measurements with fission source plates) and also kernels and spectra obtained through rigorous solutions (moments method data) of the Boltzman transport equation. For the above configurations the detector location will be varied within the shield regions.

The results of a recent study (Ref. 1) indicate that the greatest discrepancies exist in the results of different neutron calculation methods. No comparison of gamma ray spectrum prediction was reported in the above referenced report. The comparisons to be performed by Aerojet will therefore emphasize (1) the neutron transport problem with an objective to identifying conditions under which a given code and/or theory excel and (2) gamma ray spectra predictions.

2. Nuclear Rocket System

Test problems for actual nuclear rocket configurations will be significantly more complicated than the basic configuration of the previous section. This complication arises because the radiation environment in an actual configuration is composed of a number of dominant contributors. A typical configuration is that depicted in Figure 9.

Calculations on the test problems for the actual configuration should be performed by the use of a series of codes. Each leg of the calculation should be performed with all the candidate techniques selected for comparison purposes.

The actual problems to be selected should depend on the availability of suitable experimental data. The ideal test problem would be one that can be verified with reliable experimental data on such parameter as spatial, energy, and angular distributions. However, even if such detailed measurements are available, they would be of total values rather than values for each of the dominant contributors. Hence, a significant amount of interpretive work is involved.

A tentative selection for one of the test problems is the NRX/EST system with the reactor containing an internal aluminum replacement shield. A considerable amount of measured data exist for this configuration which is sufficiently complicated as to require use of a full complement of analysis methods.

The second test problem may be the Phoebus-2 reactor in a single engine flight configuration. This will allow treatment of the LH_2 tank and payload problem.

3. Nuclear Electric Systems

Selection of the test problem for an actual nuclear electric configuration will also be based on the availability of suitable experimental data. Surveys of the SNAP-8 and SNAP-50 programs should indicate the proper selection.

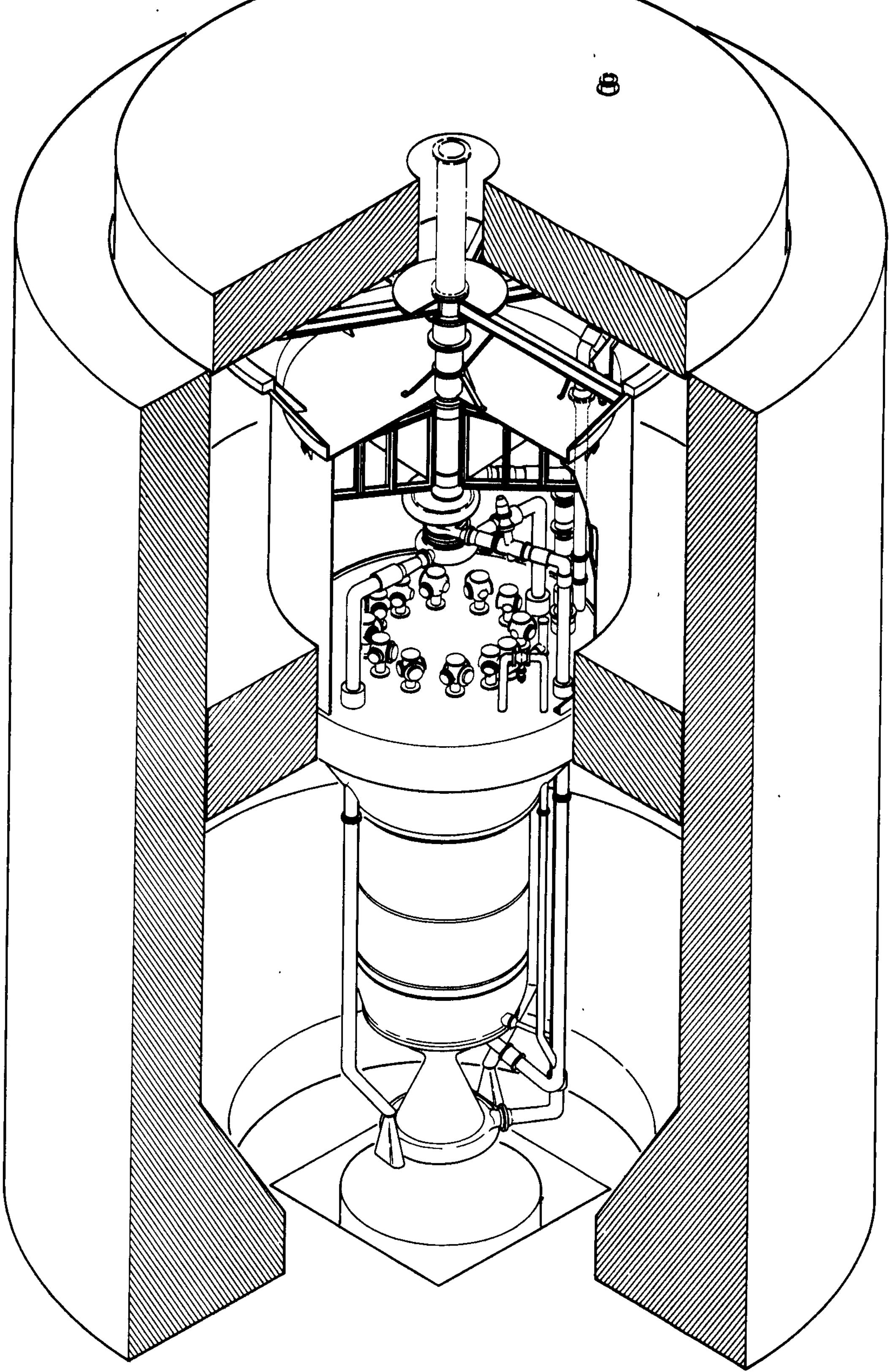


Figure 9

Isometric of XE-Engine
In ETS-1 Engine Test Compartment

It seems quite probable that reliable data exist for the SNAP-8 configuration with an external shadow-shield. However, measurements are unlikely for radiation scattering off of the radiator because these measurements would be complicated by the effects of air scattering and/or facility scattering. For this situation, comparisons may be limited to the data generated by the different analysis methods. For improved validity a relatively large number of particle histories will be examined with the preferred Monte Carlo approach.

D. CANDIDATE ANALYSIS METHODS

Candidate analysis methods are those among existing methods that can treat the radiation analysis problems of nuclear rocket and nuclear electric systems. These problems include treatment of primary gamma rays, secondary gamma rays, scattered gamma rays, primary neutrons, and scattered neutrons.

Each of the above problems can be treated, with varying degrees of adequacy, by use of any one of several methods. Primary gamma ray distributions, for example, can be predicted by a number of codes in each of the point kernel, discrete ordinates, and Monte Carlo classes of analysis methods. Likewise, a relatively wide selection of codes is available for predicting each of the four remaining classes of radiation analysis problems.

Thus a selection must be made of the codes that will be used for calculations on the test problems. This selection should be made only after completing a relatively broad literature and field survey. Table IV provides a tentative list of analysis methods that should be considered for performing calculations on the selected test problems.

TABLE IV

Tentative List of Candidate Analysis Methods

Primary Gammas	Scattered Gammas	Source Terms	Primary Neutrons	Scattered Neutrons
QAD-P5	GGG	DSN/DTF	QAD-P5	TDC/2DF
14-0/14-1	Albedo Methods	TDC/2DF	DSN/DTF	MCS
C-17	MCG	MCS	TC/2DF	FMC-N
DTF-4		FPIP/FPIC	MCS	Albedo Methods
MCG		ACT-II	FMC-N	
FMC-G			05R	
18-1			RENUPAK	

On the basis of comparisons and critical evaluations of the candidate analysis methods, selections can be made for one or two combinations that are best suited for use in nuclear rocket and nuclear electric applications. One possible combination of methods is shown in Figure 10. Note that only one block of input data is shown for the entire program system. "Housekeeping" or connector programs will be required to effect such an integrated one-pass system.

The GECOP code is shown at a location where it can provide, if desired, a plot of the geometric configuration prior to performing any calculations. This will provide assurance that the calculational model matches the desired analytical model. Moreover, it will serve to identify all the differences in a single pass rather than require multiple checkout runs on the computer.

The discrete ordinates method codes DTF-4 and 2DF are shown for generation of secondary gamma source terms. These source terms are then input to the QAD-P5 code for determining the gamma ray environment due to secondary sources.

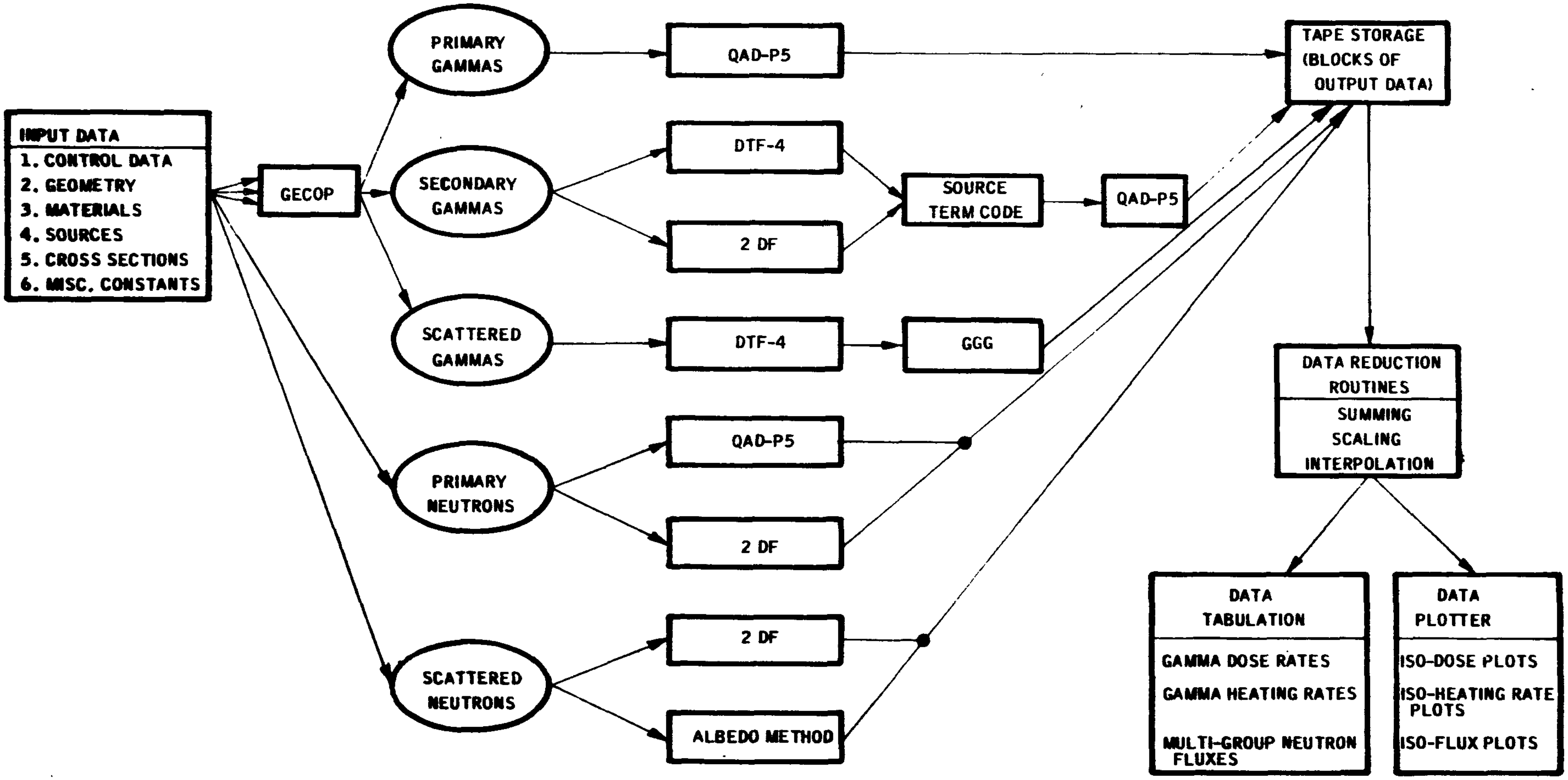


FIGURE 10

CANDIDATE ANALYSIS METHOD

The discrete ordinates method code, DTF-4, is also used in calculations of scattered gamma rays. In these calculations, it provides data on gamma ray spectra for input into the GGG, single-scatter code. This spectra generator is an important element of the analysis method because the gamma scattering process is very dependent on the energy distribution of the incident beam.

Another prominant feature of this combination method is its use of a data reduction routine. This routine can operate on the blocks of data stored on data tape for tabulation as well as plotting. The operating functions can include scaling for the desired power level and units, summing to include the desired structure for energy groups, and interpolation for identifying location of the isodose or iso-flux contours. These data can then be listed and/or plotted as desired.

E. TEST CALCULATIONS

Test calculations are needed to generate data from which to critically evaluate the candidate analysis methods. These data include accuracy, cost, and overall efficiency.

It should be recognized that the accuracy and cost potential of the various computing methods depend strongly on the adequacy of problem definition as well as on the basic nuclear data. Thus it is important to have the test problems prepared by personnel experienced in the use of the candidate methods.

A summary of user decisions that can have a significant effect on both accuracy and cost of the computing method is given in Table VI.

TABLE VI

Summary of User Decisions Affecting Accuracy

User Decisions	Point Kernel Method	Discrete Ordinates Method	Monte Carlo Method
Number and structure of energy groups	X	X	X
Approximations to geometric configuration	X	X	X
Approximations to material composition	X	X	X
Criterion for termination iterative calculations		X	
Number and distribution of mesh points		X	
Number of source points	X		
Distribution of source points	X	X	X
Number of receiver points/zones	X		X
Distribution of receiver points/zones	X		X
Source spectrum	X	X	X

Depending on the background and experience of the particular user, a generally poor technique can be used to provide accurate and inexpensive results.

Likewise, excellent techniques can often provide very poor results if appropriate input data were not provided. Hence to minimize the biasing effect of user experience in this evaluation of computing methods, input data must be prepared by personnel thoroughly familiar with the particular analysis method.

Overall efficiency is associated with the relative ease with which input data can be prepared and with which output data can be used. The engineering time required to prepare input data as well as to interpret output data should be evaluated and recorded for each of the candidate analysis methods. Moreover, a problem log-book should be used containing pertinent data such as problem running time, a summary of results, and the information required for retrieval of the output material. Special arrangements will be required for adequate control over these test calculations because most of them will be performed on the MSFC computer facility.

In regard to use of the MSFC computer, the candidate analysis programs as well as the input data must be transferred from Aerojet to MSFC with minimum time delays and minimum cost. Existing auto-din equipment can be used for most of these transfer requirements. The transmission time will be on the order of 6-hours with negligible costs. However, the existing auto-din equipment is limited in that binary information cannot be transmitted. This is not a serious limitation because most of the data transfers are non-binary. Program decks and other binary data can be shipped air-express to the MSFC computer facility. The requirement for selected binary control information such as those contained in the DTK or DDK codes can be replaced with BCD control information. The actual series of analysis methods that should be used in each of the selected test problems will depend on the survey of existing methods. The series should include candidate methods from each of the point kernel, discrete ordinates, and Monte Carlo classes of codes as well as from the auxiliary codes.

F. PROGRAMMING CONSIDERATIONS

The task for devising a "one pass one system" program for solution of radiation analysis problems has been divided into two sections for this discussion. The first section discusses the integration of existing programs without significant changes from their present state and constructing connector programs to effect the transition and thereby achieve a workable system. This discussion includes the first three stages of work. A truly "one-system" approach with considerations of attendant programming problems is discussed in Stage IV. This second section involves the task of re-programming into FORTRAN IV.

Discussion of the programming effort is divided into four stages. Stage I will be identified as the construction of a data reduction program in FORTRAN IV. Stage II will be the modification of various existing radiation analysis programs from their present state to provide appropriate output for the data reduction program. Stage III will be the construction of a housekeeping or connector program to permit the various component programs written in different systems to run consecutively. Finally, Stage IV would involve rewriting some of the selected programs in FORTRAN IV as appropriate.

The accomplishment of Stage I (Data Reduction Program) requires first that the pertinent output parameters be specified. This output is to take the form of both printed and plotted data. Existing subroutines are available for plotting on both the CAL COMP 570 or the SC-4020. Should the Marshall Space Flight Center have a plotter other than the two mentioned, research would have to be performed to determine whether software for the Marshall plotter exists and, if so, what the specifications for use would be.

In order to make this portion of the overall program compatible with both editions of the composite program, primary input should be from tape and it should be written in FORTRAN IV. The method of input gives rise to several questions that will require resolution. The information that the various radiation analysis programs are currently providing must be determined. If the units of a particular parameter contributed by the component programs differ, a standard unit of output will have to be chosen. Typical radiation analysis output data will have to be examined to determine the space requirements needed by the input routine. Since some of the radiation analysis programs provide output on the basis of the discrete ordinates method, a standard format must be developed for input of the data to the reduction program. Considering that it will be convenient for some of the component programs to output in row order and others in column order of multi-subscripted arrays, the data reduction's input scheme must be developed to handle either situation.

Development of data reduction procedures should be relatively straightforward once the above input questions are resolved. Setting forth the display units and the conversion thereto should pose the only remaining problem in the development of Stage I.

Stage II involves the modification of each radiation analysis component program so that it will output specified parameters in standard units to a common output tape (the data reduction input). Accomplishment of this stage requires first that a correspondence be drawn between the actual program symbol and the desired output symbol. The flow of the program must be sufficiently researched to ensure that the instructions for outputting to tape are placed efficiently. An additional problem area at that point is consistency of units. If the planned output is not in the approved units, appropriate conversions must be made. It must also be remembered that the large size of some of the component programs may require that the proposed tape manipulations be accomplished by a chaining method or through the use of a post-processor program. The introduction of either method will increase the program running time.

A workable prototype system should exist at the conclusion of Stage II since all component programs written under the same system could be run consecutively and their contribution to the consolidated data tape could be added to the existing data tape as they are run. Since essentially three systems are under consideration, FORTRAN IV, FORTRAN II-FAP, FLOCO-SCAT-SAP, three separate runs would have to be made to generate a complete data tape. The complete data tape would then be input to the data reduction portion and subsequently yield the desired print and plot of output data.

For consolidation into a one pass system, Stage III will require a system constructed to receive the required input of each radiation analysis component program, pair it with its corresponding program found on a program storage tape, and generate the system required control cards. Additionally, in order to conserve computer time, programs written in the same system should be grouped together on a secondary input tape. Problem areas here are the initiation and termination of the individual systems involved. Conflicts in physical tape assignments of the affected systems must be circumvented also. A third major factor to be considered is that of the method for handling tape labels (or the absence of) in the affected systems.

Completion of the above three phases would produce a "one pass system" with a minimum amount of disturbance to existing programs that have been checked out.

Stage IV involves recoding appropriate parts of the system in FORTRAN IV. A major item of consideration in evaluating the many shielding programs currently in use is that of inconsistency in the programming system. The spectrum of languages used range from SAP through SCAT, FLOCO, and FORTRAN II to FORTRAN IV. Additionally, several different computers were used in construction of the various programs including the 704, 7094, and the CDC 3600.

Differences in machine storage availability may require the use of methods such as the CHAIN option. While improving problem handling capability, use of this option increases problem running time. Reluctance to use CHAIN would preserve problem running time but at the same time substantially reduce the size of problems that can be handled.

Differences in the machine word sizes (7094-3600) could lead to a difference in prediction accuracy, amount of core storage necessary, and/or problem running time.

The selection of programs written in SAP, SCAT, or FLOCO would necessitate a decoding of same and a subsequent recoding in FORTRAN IV where existing documentation is inadequate. Preliminary studies indicate a strong possibility that some of the component radiation analysis programs selected may be in FLOCO or in FAP. As a yardstick, the GE 15-2 programs contains 5800 instructions. Application of a locally arrived at program construction factor of 11 instructions/hr (1.5 FORTRAN Statements) yields a generation time of 528 hrs for the GE 15-2 program. This construction factor considers the time from problem definition through checked out and documented program. The plus factor of working from a checked out program in the FORTRAN IV program generation is more than offset by the necessity of first having to decode the selected FLOCO or FAP program. Moreover, work performed by other contractors in generating FORTRAN versions of the FMC-N/FMC-G and DTK/DDK programs substantiates this conclusion.

FORTRAN II to FORTRAN IV conversion can be achieved with minimal effort. Core system residence in FORTRAN IV is substantially greater than in FORTRAN II. Consequently marginal FORTRAN II programs would have to be chained when converted to FORTRAN IV. However, CDC FORTRAN to FORTRAN IV conversion requires some additional amount of editing effort. Multi-statement lines and I/O statements are examples of differences that would have to be resolved. As an example, 49 hours of programming effort were expended by Aerojet in converting 2DF, a CDC FORTRAN program written for the CDC 3600, to run locally on the IBM 7094 under a FORTRAN IV system. Memory size incompatibilities also limit the problem size capable of being handled.

In summary, the size and difficulty of the programming effort depends upon the selected program mix (FLOCO, FORT II, etc.) from which the resulting FORTRAN IV program will be constructed. However most of the more promising of existing methods have already been converted into FORTRAN. These include the FMC-N/FMC-G package, the TDC code, and the DTK/DDK package.

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III. PROPOSED PROGRAM

The objectives of this proposed program are to compare and critically evaluate existing analysis methods in relation to their ability to meet the analysis requirements of nuclear rocket and nuclear electric systems to select one or two "best" combinations of methods, and to re-program these into FORTRAN-IV as appropriate and to conduct a seminar for MSFC personnel. Past work reported in ER-8236, Evaluation of Methods for Computing Nuclear Rocket Radiation Fields, will be critically reviewed and then extended to include consideration of additional promising analysis methods in order to achieve the program objectives.

Detailed specifications of the work to be performed, the plan of approach, and schedules are described in this section. The proposed effort is divided into tasks, distributed over three phases. The percentage of total technical effort for each task is shown in Figure 11.

A. WORK TO BE PERFORMED

1. Literature search and field surveys shall be conducted to evaluate past work and to collect data on existing analysis methods.

2. Promising radiation analysis methods, additional to and including the methods considered in Report No. ER 8236, shall be reviewed for possible application in whole or in part for the selected analysis method.

3. Evaluation criteria for existing analysis methods shall be defined. Consideration shall be given to radiation analysis problems characteristic of nuclear rocket and nuclear electric power systems, to the use of suitable measured data for accuracy evaluation, and to pertinent operational characteristics of the analysis methods.

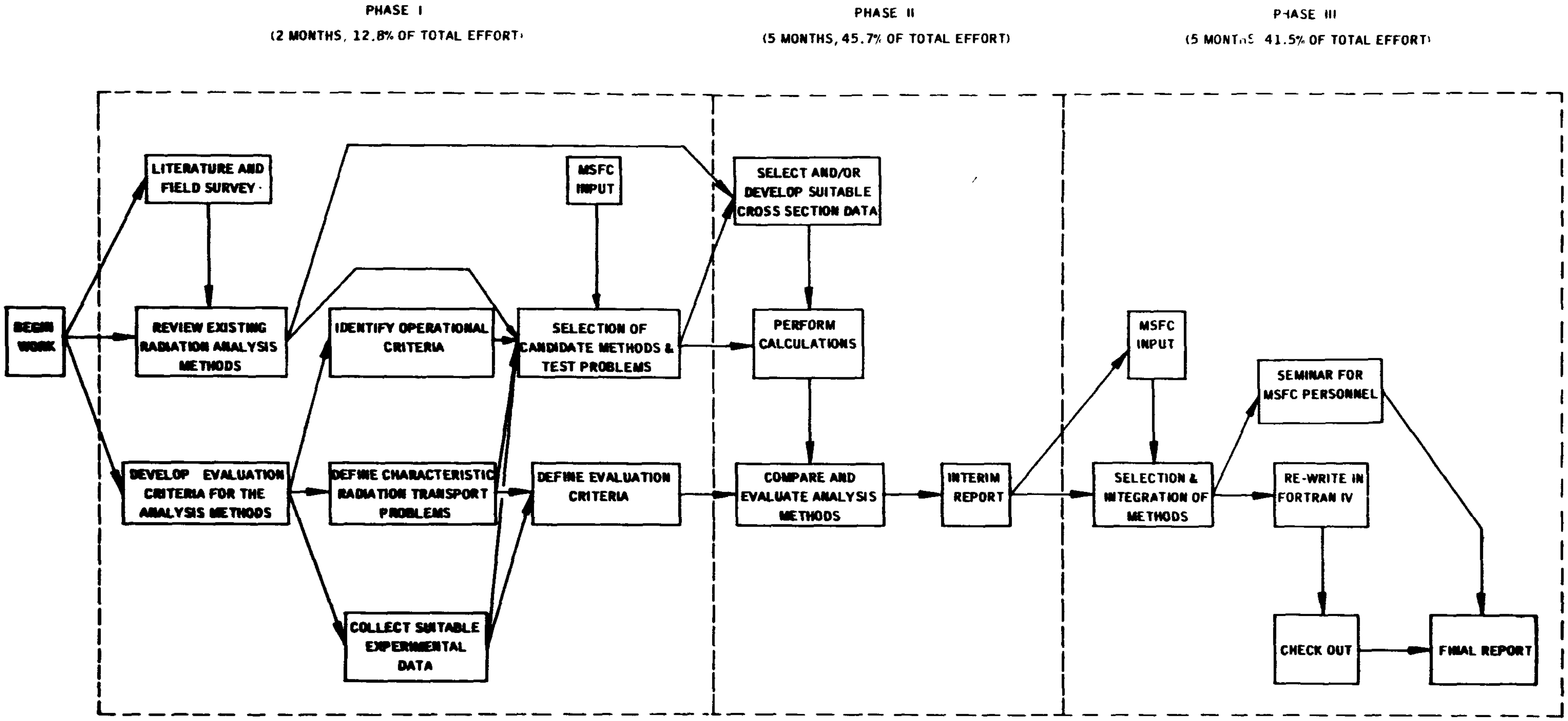


FIGURE 11

FLOW CHART FOR THE WORK PROGRAM

4. Test problems shall be selected and defined for evaluating the promising analysis methods identified by Item No. 2 above.

5. Appropriate nuclear data including multigroup neutron and gamma ray cross sections, interaction probability data, attenuation coefficients, and build-up factors shall be selected and/or developed for use in evaluating the analysis methods.

6. Perform calculations on the test problems selected in Item No. 4 with the candidate analysis methods. These shall include primary neutron and gamma ray calculations, secondary gamma ray calculations, scattered neutron and gamma ray calculations, and gamma ray and neutron streaming calculations.

7. A comparison and critical evaluation of the promising analysis methods shall be made on the basis of the criteria developed in Item No. 3.

8. A selection shall be made for one or at most two analysis methods best suited for nuclear rocket and nuclear electric applications on the basis of Item No. 7 with appropriate input from MSFC personnel. Appropriate modifications and integration of the selected methods shall also be made.

9. The selected combinations of analysis methods shall be re-written as required so as to obtain versions of the programs in FORTRAN-IV language.

10. A seminar shall be conducted to provide MSFC personnel and other personnel, as may be specified by the government, with the instructions necessary to allow complete utilization of the selected methods at MSFC.

11. The reports to be written shall include the agreed upon progress reports as well as an interim report, a user's manual, and a final report.

B. PROGRAM APPROACH

The approach proposed for achieving the program objectives is best shown pictorially (Figure 11). The program, 12-months in duration, is divided into three phases, each of which is described below.

1. Phase I (12.8% of total effort)

The tasks during Phase I include a literature and field survey, a review of existing analysis methods, definition of evaluation criteria, and selection of candidate analysis methods and of test problems.

a. Literature and Field Survey

The proposed program will be initiated with a literature and field survey to locate and identify the existing computer codes which might be of interest to the proposed program. The object of this survey is to determine whether some of the less popular computer codes are more efficient or contain more efficient features than those used in current nuclear rocket and nuclear electric programs.

Moreover, the codes evaluation study recently completed for MSFC (Ref. 1) will be critically reviewed to avoid any unecessary duplication of completed work. The relatively large differences in the candidate methods considered in this study are believed to be caused by a combination of differences in basic nuclear data as well as differences in computing techniques. The proposed approach of performing calculations on basic configurations as well as realistic configurations will provide a basis for identifying weaknesses in basic nuclear data.

A field survey will also be undertaken. It will include consultations with personnel from the Radiation and Shielding Information Center (RSIC) of Oak Ridge National Laboratory. Computer programs of interest, that are not currently available in the Aerojet codes library (see Section V-B), will be acquired along with suitable documentation. Consultations will also be arranged, as appropriate, with radiation analysis personnel from other contractors engaged in radiation analysis activities. The object of these consultations will be to identify those techniques which have been developed for different applications and by personnel with different backgrounds which might contain unique features that warrant evaluation in the proposed program.

b. Review of Existing Methods

The promising methods identified in the literature and field survey will be critically reviewed for possible inclusion in the evaluation task. Consideration will include the basic theory of the various methods as well as their operational suitability. The various methods will be characterized as point kernel, discrete ordinates, Monte Carlo, or some combination method. In addition, computation schemes, convergence criteria, and availability of appropriate input data will be considered. Operational suitability considerations will include geometric flexibility, complexity of input data, potential accuracy of results as well as operational costs.

c. Definition of Evaluation Criteria

The criteria to be used in evaluating the different analysis methods will be defined prior to performing any calculations. These evaluation criteria will be based on the analysis requirements of nuclear rocket and nuclear electric systems, on availability of suitable experimental data, and on program operational characteristics.

Current analysis requirements include treatment of neutron transport through void media followed by thermalizing and absorbing media, neutron and gamma ray scattering in air and off of metallic surfaces, and angular distributions of neutrons emerging from the reactor surfaces. Detailed characteristics of these and of other analysis requirements will be defined in an effort to establish criteria on which suitability of the various analysis methods may be evaluated.

Suitable experimental data will also be collected and evaluated. These will include measured data on basic configurations. The configurations on which experimental data are available include the Kiwi-B, NRX-A, NRX/EST, Phoebus-2, and SNAP-8. Accuracy criteria will be defined on the basis of experimental data that are considered valid and reliable.

Operational characteristics of computing methods will be an important consideration in defining the evaluation criteria. These will include the ability to treat realistic geometries, relative ease with which input data are prepared, and direct useability of the output data.

All of the above considerations will contribute to a detailed definition of the evaluation criteria. These criteria will then be used when comparing the relative suitability of the various condidate analysis methods.

d. Selection of Candidate Analysis Methods and Test Problems

Candidate analysis methods are those which will be selected, with MSFC participation, for use in calculating the test problems. They will include those methods that cannot be rejected on the basis of their ability to treat realistic nuclear rocket and nuclear electric configurations. In cases of redundancy such as in the DSN, DTK, and DTF series of codes, a selection will be made of the most promising method.

Test problems will also be selected, with MSFC participation, to determine the relative suitability of the various candidate analysis methods. These problems will include two basic configurations for evaluation of nuclear data. These will probably be a point source problem and a plate source problem in an infinite water medium.

Two problems representing typical nuclear rocket configurations will also be selected. One of these will be a flight configuration and the other a ground test configuration. Detailed specifications for these problems will consider the configurations for which suitable experimental data are available.

The fifth test problem will represent a typical nuclear electric configuration. This problem may be confined to evaluation of the radiator scattering problem of a system such as SNAP-8, since the radiation analysis problems of nuclear rocket and nuclear electric systems are quite comparable otherwise.

Thus Phase I of the proposed program concludes with selection of the candidate analysis methods and test problems and with the definition of the evaluation criteria.

2. Phase II (45.7% of Total Effort)

The tasks during Phase II include selection and/or development of suitable cross section data, calculations on the selected test problems, comparison and evaluation of the analysis methods and preparation of an interim report summarizing these results.

a. Generation of Cross Section Data

Cross section data will be required for analysis of the selected test problems with the candidate analysis methods. Judicious choice of constants can lead to significant cost savings as well as improvements in accuracy of results.

In general, the required neutron cross sections can be selected from existing libraries of 4, 6, 16, 18, or 24 group constants. However, specific cases such as regions containing LH_2 will require some additional data for the thermal energy groups. The gamma ray group structure will contain at least 5 groups below 2.2 Mev to properly account for the rapid variation of attenuation coefficients with energy at the low end of the energy bound.

b. Calculations on Test Problems

Calculations with the IBM-7094 computer will be performed principally at the MSFC Computer facility. However, some calculations of a checkout nature will be performed at the Aerojet facility in order to minimize delays associated with computer turn-around time.

The candidate analysis methods selected during Phase I will be shipped to the MSFC computer facility for checkout. Concurrently, arrangements will be made for the handling of input/output data that must be transferred between MSFC and Aerojet. The Auto-din network that currently exists at both MSFC and Aerojet will be used to minimize the time required for transfer of the input/output data.

Calculations will be initiated on the two basic configurations for validating the nuclear data and for a preliminary screening of the candidate Monte Carlo methods. Preliminary screening of the candidate Monte Carlo methods will be on the basis of calculations on the basic configurations. These calculations will be adequate for assessment of program flexibility and efficiency as well as program adequacy for treating the radiation transport problems of nuclear rocket and nuclear electric systems. Calculations on the realistic configurations will be performed only with the most promising of the Monte Carlo methods. This approach will allow running a sufficient number of particle histories on the selected test problems for accuracy evaluation.

Distributions for primary gamma and neutron, scattered gamma and neutron, as well as secondary gammas will then be calculated as appropriate for the two nuclear rocket and one nuclear electric configurations. All the candidate analysis methods selected during Phase I will be used in these calculations, with the exception that Monte Carlo Analysis will be performed with only the most promising of the candidate methods. Table VII shows the minimum number of analysis methods that will be used in calculations of the test problems.

TABLE VII

SUMMARY OF CALCULATIONS TO BE PERFORMED

Configuration	Primary Gammas	Scattered Gammas	Source Terms	Primary Neutrons	Scattered Neutrons
Basic Configuration	7		3	6	
Nuclear Rocket - Flight	4	3	2	3	3
Nuclear Rocket - Ground Test		2	2		2
Nuclear Electric - Flight		2			2

Most of these calculations will be performed on the IBM 7094 computer at MSFC. Input data will be delivered to the MSFC computer facility by means of an Auto-din network that currently exists at both MSFC and Aerojet. Compatible equipment for this method data transfer currently exists at both Aerojet and MSFC.

Although most of the computations will be performed on the MSFC computer facility, checkout functions will be performed on the Aerojet computer facility. Use of the Aerojet facility will minimize the turn-around time for the short, checkout-type problems.

c. Interim Report

The results of these calculations, comparisons, and critical evaluations will be documented in an interim report. This report will contain critical evaluations of the various candidate analyses on the basis of criteria defined during Phase I. These criteria will include such parameters as accuracy and adequacy of results, machine and engineering costs associated with the calculations, and flexibility of the analysis method. The object of this interim report is to collect, organize, and interpret the raw data in order that an objective selection can be made, with MSFC participation, of the one or two analysis methods best suited to nuclear rocket and nuclear electric systems.

3. Phase III (41.5% of Total Effort)

The third phase of the proposed program will extend over a 5-month period and will include selection and integration of analysis methods, re-write into FORTRAN-IV, a seminar program, and a final report.

a. Selection and Integration of Methods

On the basis of the data contained in the interim report and in consultation with MSFC, one or two of the candidate methods will be selected. Where appropriate, desirable features from those methods not selected will be incorporated into the selected methods.

The selected methods will, no doubt, become multiple combinations of existing programs. The different programs will require modification for their integration into a program system of a number of independent programs.

Intermediary data processing routines will be written to make the output data of each program compatible with the program system. Moreover, data checking and data reduction routines will also be required. The REON program GECOP, for example, will be incorporated into the program system to allow checking of the problem geometry specifications. Data reduction routines will include scaling, summing, interpolation listing, and plotting functions.

b. FORTRAN-IV Reprogramming

With exception to some of the Monte Carlo programs that might be selected, most of the existing programs are either written in FORTRAN language or have FORTRAN versions. However, they are written either in an older version of FORTRAN identified as FORTRAN II or are written in the FORTRAN language for the CDC computer rather than the IBM computer. The modifications necessary to convert these FORTRAN systems into FORTRAN IV are relatively minor and will be undertaken as soon as analysis method selections are made.

The Monte Carlo methods that may be selected will probably be in either FLOCO or FAP machine language. However, if the FMC-N/FMC-G package is selected, existing FORTRAN versions can be used. Reprogramming such systems as the MCS/MCG or the General Electric Program 18-1 will require about three man-months of programming effort. Such re-programming will be performed as required to provide versions of the selected methods in FORTRAN IV language.

c. Seminar Program

Effective utilization of the selected analysis methods will be achieved by undertaking a seminar program. Detailed planning for this seminar will include consideration of the character of the selected methods as well as the familiarity of MSFC personnel with the different elements of the selected methods.

This seminar will be most beneficial if conducted in two or three 2-week sessions, each separated by about 2 weeks. This type of scheduling will minimize disruption of the normal duties of MSFC personnel as well as provide opportunity for use of the programs to solve problems different from the test cases.

This seminar will be conducted in parallel with the effort to generate FORTRAN IV versions of the selected methods. This approach will allow opportunity for MSFC technical personnel to influence the task of re-writing in FORTRAN IV.

In general, the seminar will consist of lectures as well as problem solving work sessions. The lectures will include descriptions of the various theoretical approaches as well as the limitations of the selected approach.

The proposed interim report and User's Manual will be extremely helpful in providing the necessary introductory information. The interim report will contain detailed descriptions of the test problems as well as identification of the recommended combination of analysis methods. The User's Manual will contain instructions on preparation of input data.

A "User's-Manual" will be drafted prior to the initial meetings of the seminar. This manual will contain detailed instructions on preparation of input data.

This third phase will be concluded with a final report. Information not conveyed in the interim report and in the User's Manual will be included in this final report. It will probably be issued in three parts with re-prints of the interim report and the User's Manual consisting of two of the three parts. Description of the integrated program system as well as of the programs rewritten in FORTRAN IV will be conveyed in the third part of this final report.

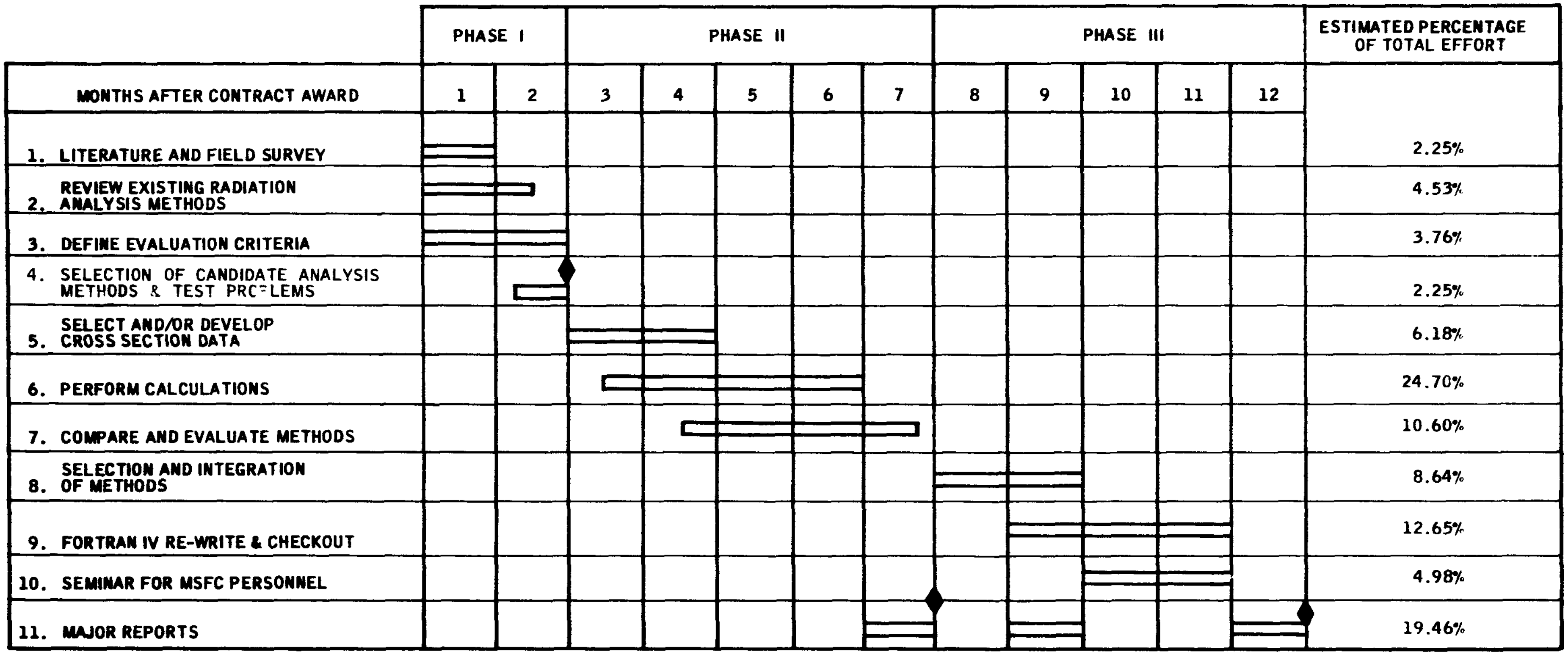
C. SCHEDULES

The proposed schedule is shown in Figure 12. Three phases are indicated along with the estimated percentages of the total effort for each phase and for each of the eleven tasks.

The first phase extends over a two-month period and consists of the preliminary work associated with the review of existing analysis methods. It concludes with the selection of candidate analysis methods and of test problems, which is the first milestone.

The second phase extends over a five-month period and is that phase during which the candidate analysis methods are compared and critically evaluated. This phase terminates with, the second milestone, the publication of an interim report.

The third and final phase also extends over a five-month period and consists of the selection and integration of the best analysis methods, reprogramming into FORTRAN IV and a seminar for MSFC personnel. Two major reports are planned for this phase including a User's Manual and a final report which is the third and final milestone.



The indicated over-all schedule and percentage allocation of total effort provides the necessary time and depth-of-study for achievement of the program objectives. Although the program could be completed on an earlier schedule, the 12-month period suggested in the Purchase Request was selected for efficient utilization of resources.

D. REPORTS

The reports that will be published include the required monthly progress reports and final report as well as an interim report and a User's Manual. All of these reports will conform to the requirements of the contract.

The two additional reports are included to improve the degree to which program objectives are achieved. The interim report will contain the comparison and critical evaluation data. In addition to formalizing the basis for the selection task, this report will improve the degree to which MSFC personnel can participate in the selection of analysis methods.

The User's Manual will be written as part of the seminar task. It will contain detailed instructions on preparation of input data as well as operation of the program system.

E. NEW TECHNOLOGY

NASA's New Technology clause is actively supported at Aerojet Under Contract SNP-1 which provides for the Development of the NERVA Nuclear Rocket, REON has had a formalized program since October 1964. During this period of time, REON has forwarded to NASA more patent disclosures than had been submitted in the previous 3-1/2 year period of the Development Program. Currently, there are approximately 50 reportable items submitted by REON in various stages of being published as Technical Briefs or to be included in Survey Handbooks. To date, at least two reportable items are known to have been transferred to commercial use. Since a new technology reporting system is already in being, it is not anticipated that any additional direct charge manpower will be required in the conduct of this proposed program. REON is also participating in New Technology reported under NASA Contract SNP-35 for development of the Phoebus-2 Nozzle and other Aerojet divisions are also actively participating in the Technical Utilization Program in relation to the M-1 (NAS 23555) and the SNAP-8 (NAS-5417) Programs which are prime contracts with NASA Lewis Research Center.

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IV. MANAGEMENT ORGANIZATION AND PERSONNEL

Aerojet is actively engaged in the development of nuclear rocket systems (NERVA) as well as nuclear electric systems (SNAP-8) and recognizes the problems associated with the integration of these systems with space vehicles. These development programs require extensive analyses to define component radiation environment using a large complement of available analysis methods. (See Section V-B). Successful completion of the proposed program will help resolve many of the difficulties currently encountered in the accurate predictions of radiation environment.

A. CORPORATE ORGANIZATION

The Aerojet corporate organization includes eight plant operations, virtually autonomous, that report to the President (W. E. Zisch) through Group Vice-Presidents A. L. Antonio and L. W. Mullane. The largest of Aerojet's facilities is the Sacramento, California plant which is directed by Mr. R. B. Young, Vice-President and General Manager.

Recognizing the importance of the proposed program to the continued successful progress of the Nuclear Rocket Program, Aerojet management has chosen to locate the responsibility for this proposed program within the Aerojet structure where it can obtain the maximum benefit from the best managerial and technical talent available in these unique disciplines.

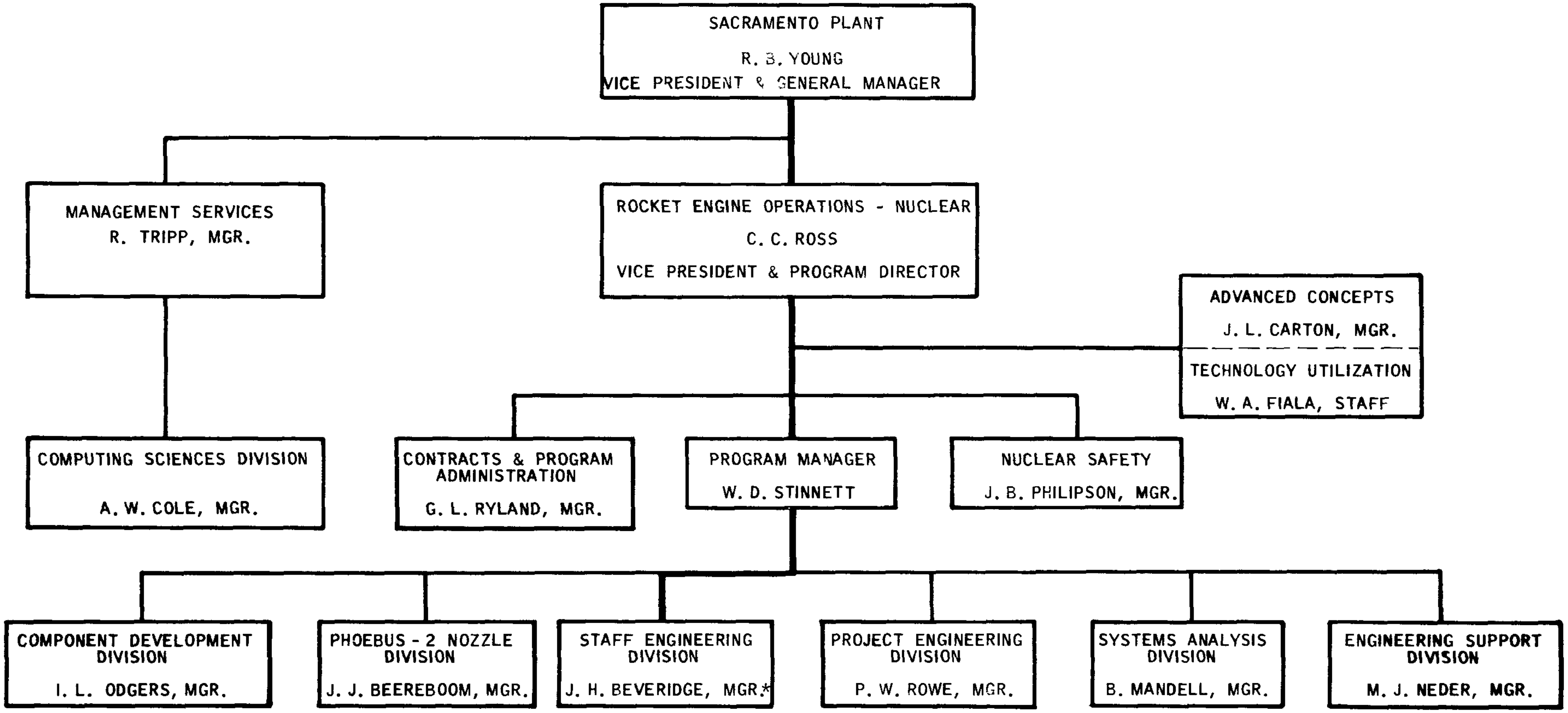
The proposed program for "Synthesis of Methods for Design Analysis of Radiation Shields for Nuclear Rocket Systems" will be carried out at the Sacramento plant by personnel of Rocket Engine Operations-Nuclear (REON) supported as necessary by personnel from the Computing Sciences Division and the SNAP-8 Division.

The Computing Sciences Division is located at the Sacramento Plant and provides the computing support for all of Aerojet's Sacramento operations. The SNAP-8 Division is located at the von Karman Center of Aerojet.

REON maintains prime responsibility for the conduct of the NERVA program and all other phases of nuclear rocket technology for Aerojet under the direction of Dr. C. C. Ross, Vice-President and Program Director. (See Figure 13) Dr. Ross is also Vice-President of Engineering for Aerojet-General Corporation reporting directly to Mr. Zisch, and in this capacity, is able to influence corporate resources to aid REON activities. Dr. Ross continuously reviews the operations and progress of all programs and divisions under his jurisdiction using a highly effective program management technique modeled after the program management system developed by the Navy for the Polaris Program.

The proposed program will be controlled through this system of program review with the use of wall charts and slides (which are kept current on a weekly basis). Information from all pertinent management aids such as PERT, cost data, and milestone charts are used to highlight deviations from the planned program. The variables of time, resources, and technical performance form the basis for managerial action to assure that current and potential problems are isolated in time to take corrective action. Management can readily see the impact created by the chosen course of action.

Dr. Ross also has at his disposal a program evaluation team that maintains a continuing evaluation of the status of each major part of all REON programs. This proposed program, as an element of REON efforts, will come under the surveillance of the evaluation team. These evaluations are presented during the weekly meeting and serve as an aid, not only to Dr. Ross, but also to his subordinates. The meeting participants form a highly qualified technical staff, representing a wealth of experience that can be applied to problem areas that may be presented on the proposed program. This program management and monitoring system will benefit the proposed effort and NASA by continuously exposing the program efforts to experienced, high caliber review.



* PRESENTLY ASSIGNED AS PROJECT MANAGER FOR THE CONCEPTUAL DESIGN OF THE NERVA FLIGHT ENGINE

FIGURE 13
MANAGEMENT ORGANIZATION

B. PROJECT ORGANIZATION

In support of on-going programs, the REON organizational structure provides for divisional-level and administrative organizations (Staff, Project, Component Development, Systems Analysis, and Engineering Support). (Ref. Figure No. 13). The managers of these divisions report to the Program Manager, Mr. W. D. Stinnett.

In addition to current programs, new technical projects of special interest to REON report to the Program Manager as they become operational. Therefore, this program will be established as a project function within the Nuclear Analysis Department, managed by Dr. J. A. Vreeland, reporting through the Staff Engineering Division Manager, Mr. J. H. Beveridge.

The REON Nuclear Analysis Department constitutes a significant concentration of highly skilled radiation analysis experts. Although a relatively small group, they offer 145 man-years of nuclear experience, 27 of which are directly related to nuclear rockets. Performance of this program in such a group will assure the achievement of program objectives. Moreover, the varied technical background represented within this group assures the availability of personnel who are intimately familiar with all the analysis methods that are to be considered in the proposed program.

The proposed program will be managed by Mr. W. R. Butler, the Project Engineer, whose attention will be devoted exclusively to this program. Moreover, he will be the Aerojet representative on all technical matters relating to the proposed program.

Mr. Butler, currently Manager of Reactor Analysis, has had more than ten years of reactor and radiation analysis experience. His recent efforts include radiation environment, nuclear heating, and nuclear induced activity predictions for all components of the NERVA engine system.

These analyses required application of existing Monte Carlo, Discrete Ordinates, and point kernel methods, as well as development of auxiliary support systems.

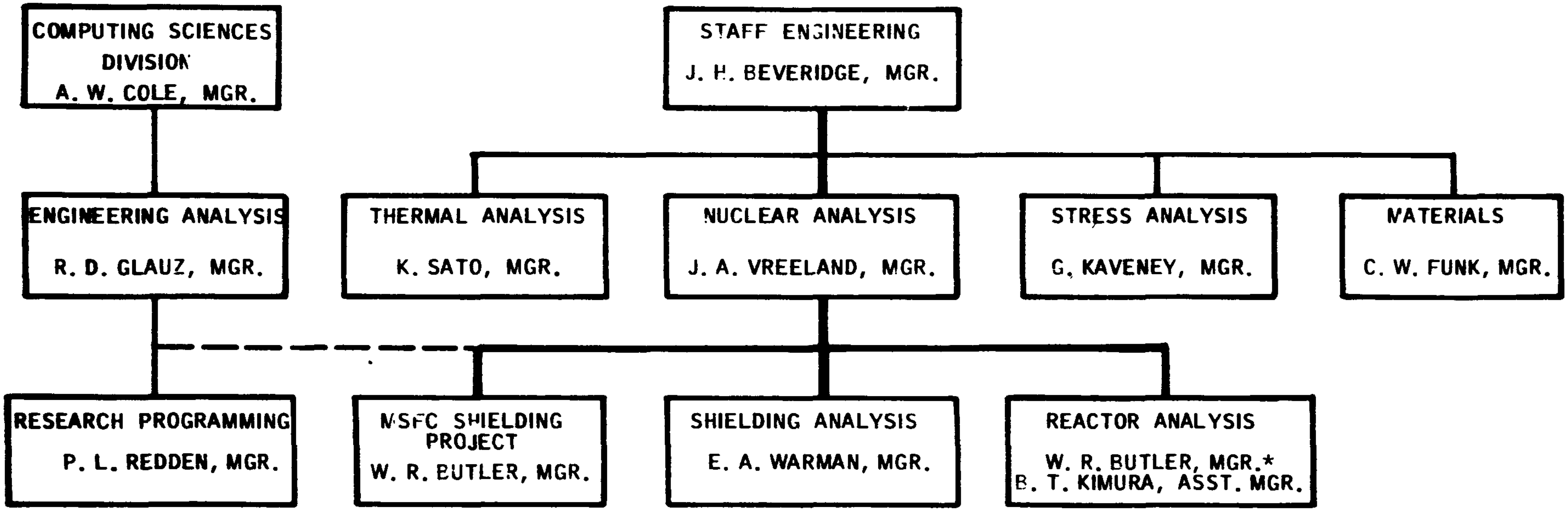
Key personnel who will assist Mr. Butler include Messrs. B. T. Kimura and P. L. Redden. Mr. Kimura has also had more than ten years of extensive theoretical and experimental work on such programs as the nuclear ramjet and aircraft nuclear propulsion programs. Mr. Redden has had extensive programming experience applicable to the requirements of this proposed program. He has programmed in SAP, SCAT, FLOCO, FORTRAN II and FORTRAN IV for the IBM 701, 704, and 7094 series of computers.

The relationship between the proposed project and the programming support is shown in Figure 14. This relationship has been found to be very effective in the course of the NERVA work. Programmers who support the NERVA nuclear analysis requirements and who are thoroughly familiar with the characteristics of nuclear codes, will be available through Mr. Redden to support the proposed program as required.

A partial listing of the personnel who will be assigned and available to this program, and the approximate percentage of time they will devote to it, are as follows:

Walter R. Butler	Manager, Reactor Analysis	100%
Karl O. Koebberling	Nuclear Engineer	50%
Benjamin T. Kimura	Senior Nuclear Specialist	30%
Paul L. Redden	Senior Programmer	30%
Edward A. Warman	Manager, Shielding Analysis	10%
Duke R. Rogers	Senior Nuclear Engineer	10%
Paul Duchon	Senior Systems Engineer	10%
John A. Vreeland	Manager, Nuclear Analysis	5%

Resumes for the above personnel follow.



* FOR THE DURATION OF THE PROPOSED CONTRACT, MR. W. R. BUTLER WILL BE RELIEVED OF HIS PRESENT ASSIGNMENT AS MANAGER OF REACTOR ANALYSIS.

FIGURE 14

PROJECT ORGANIZATION

W. R. Butler

Education

B.S., Electrical Engineering, University of California, Berkeley, 1956

M.S., Nuclear Engineering, University of California, Berkeley, 1957

Additional post graduate work at University of California, Berkeley as well as Sacramento State College

Professional Experience

Aerojet, 1963 to present, Manager, Reactor Analysis Dept. Responsibility for radiation analysis required for design of Phoebus-2 nozzle and pressure vessel and all components of NERVA engine. In this capacity he has directed the analysis of nuclear heat generation in the Phoebus-2 nozzle and pressure vessel as well as a program to maintain current data as design configurations are modified. He has published comprehensive reports on nuclear heating, nuclear activation and nuclear environment of the NERVA Technology engine. In addition he has provided radiation source terms for the design of the shielding required in ground testing of the NERVA Technology engine.

Aerojet, 1957 to 1963, Task Engineer. In this capacity Mr. Butler performed shielding analysis for the Gas-cooled Reactor Experiment (GCRE) as well as the SNAP systems. He also contributed to Aerojet-sponsored studies of propulsion reactors prior to the existence of the NERVA and SNAP programs.

University of California, 1956 to 1957, Graduate Research Engineer. Mr. Butler participated in the design of a test facility for research studies of boiling with forced convection.

W. R. Butler

Publications:

A Transport Theory Analysis of a Gaseous Core Concept, W. R. Butler and H. C. Chang, Trans. ANS 7 379 (1964)

The NERVA Neutron Environment as Affected by the Angular Distribution of Leakage Neutrons, W. R. Butler and J. A. Vreeland, Trans. ANS 6 (1963)

Effects of Space Radiation on Start-up of Orbit's Reactors, W. R. Butler and D. I. Gilbert, Trans. ANS 5 (1962)

Paul Duchon

Education

B.S., Mathematics, Western Reserve University, 1951
M.S., Engineering Mechanics, Case Institute of Technology, 1961
Certificate in EE, West Virginia University, 1944

Professional Experience

Aerojet, 1962 to present, Senior Systems Engineer. Mr. Duchon is responsible for preliminary and conceptual design of power conversion systems and their integration with spacecraft for orbital and planetary missions. He has contributed to SNAP-8 turbo-alternator mechanical design and studied use of SNAP-8 systems for lunar installation as well as an earth orbiting manned space station.

Thompson Ramo-Wooldridge, 1957-1962, Supervisory Engineer. The design engineering and integration of the total power conversion system of SNAP-2 was directed by Mr. Duchon. He also supervised the design of the SNAP-1. He developed the mathematics required in the bearing and pump tests and studied numerous advanced concepts for space power systems.

Designers for Industry, Inc., 1955-1957, Design Engineer. Mr. Duchon developed mechanical and electrical designs for space electrical machines, systems for oil well location, the Vanguard Satellite sequence timer, and switch gear for facsimile transmission and reception.

Teaching Experience. In addition to the experience described above, Mr. Duchon has taught courses at Upland College in California, Fenn College in Ohio and for the Cleveland Board of Education from 1954 to 1955.

B. T. Kimura

Education

B.S.E., Physics, University of Michigan, 1952
A.M., Physics, University of Michigan, 1954
Additional post graduate work at University of Oregon, 1964 and 1965

Professional Experience

Aerojet, 1965 to present, Nuclear Engineering Specialist. Mr. Kimura is responsible for the radiation analysis provided by the NERVA program to design future test stands for ground testing of nuclear rocket engines and stages. In addition he has made substantial contributions to Aerojet-sponsored analyses of radiation exposure to personnel operating nuclear rocket vehicles.

Marquart, 1960 to 1964, Principal Member/Advanced Technical Staff. Mr. Kimura directed extensive theoretical and experimental programs in neutron and gamma ray physics including radiation transport and detection. Design of experiments including dosimetry and the prediction of nuclear interactions using a linear accelerator as a source were part of this activity. Mr. Kimura was also responsible for the radiation analysis of all nuclear systems including nuclear ramjet engines, isotope and reactor space power sources, and NULACE (Nuclear Liquid Air Cycle Engines). This responsibility included prediction of the environment, and the solution of any radiation-induced problems. The design of shielding required for controlling radiation was performed by Mr. Kimura.

Rocketdyne, 1957 to 1960, Senior Research Engineer. Mr. Kimura conducted radiation analysis of nuclear rocket engines comparable to NERVA. These analyses included radiation heating of propellant and structure, shield design, and reactor hazards evaluation.

B. T. Kimura

Convair, 1954 to 1957, Senior Nuclear Engineer. Experiments to evaluate radiation scattering from air, ground, and structure were analyzed by Mr. Kimura. He also evaluated additional experiments to access the effects of shield penetrations.

Engineering Research Institute, 1953 to 1954, Assistant in Research. Mr. Kimura assisted in the development of ferromagnetic cores for tuned circuit applications including the manufacture, evaluation, and interpretation of these cores and their properties. He also participated in studies of inside trajectories, radar cross sections and error analyses related to use of ballistic missiles in defense.

Engineering Research and Development Laboratory, 1952 to 1953, Physicist. Mr. Kimura did investigations in geometrical optics, interferometry and infra-red spectroscopy for the development of a passive infra-red detectors.

Publications

Radiation Exposure to Crew During Operation of a Manned Nuclear Space Vehicle, G. Garrington, B. T. Kimura, and E. A. Warman, Aerojet Report RN-TM-0294, January 1966

Radiation Analysis for the PLUTO Nuclear Ramjet Engine, Paper presented at ARS/ANS/AIS Nuclear Propulsion Conference August 15 - 17, 1962, Monterey, Calif., B. T. Kimura and J. C. Saunders.

Shielding Handbook for Aircraft Designers, B. T. Kimura, ed., Report NARF-57-20T (FZK-9-093), Convair, June 1957 (Classified)

Air and Ground Scattering from the GTR; A Post Analysis, B. T. Kimura, E. Hessee, and G. S. Weller, Report NARF 56-20T, Convair-Ft. Worth, July 1965 (Classified)

Penetration of Radiation through Crew Shields, B. T. Kimura, Report NARF-56-29T, Convair-Ft. Worth, Nov. 1956 (Classified)

K. O. Koebberling

Education

B.S., Nuclear Engineering, North Carolina State University, 1958

Experience

Aerojet, 1965 to present, Nuclear Engineer. Mr. Koebberling has assisted in calculating radiation distribution and nuclear heating for the design of Phoebus-2 and NERVA components. Use of various computer programs including Monte Carlo and S_n codes were required for these analyses.

Lockheed, 1959 to 1965, Nuclear Engineer. For six years, with the Lockheed Nuclear Products Division, Mr. Koebberling participated in numerous programs including tank shielding of Aberdeen - OTAC, the Ohio State University, reactor and the South American Tour reactor as well as the Reactor in Flight Test (RIFT) program. In these programs Mr. Koebberling performed Monte Carlo shielding studies, criticality calculations, control rod worth predictions, power distribution analyses, reactor kinetics calculations, radiation effects evaluations and hazards analyses. In addition to the use of many nuclear codes, Mr. Koebberling maintained a library of codes and therefore made substantial contributions to Lockheed Report ER-8236, Evaluation of Methods for Computing Nuclear Rocket Radiation Fields.

Publications

Fission Product Inventory Code, K. D. Koebberling, et al., ER-6906, May 1964

IV, B, Project Organization (cont.)

P. W. Redden, Jr.

Education

A.B., Mathematics, West Virginia State College, 1951

Professional Experience

Aerojet, 1962 to present, Senior Programmer. Mr. Redden has the responsibility for providing the programming support required by Aerojet in the development of nuclear rockets. In this capacity he directs the activity of programmers in generating new codes, in modifying and adapting existing codes to specific requirements, and in incorporating new codes into the Aerojet system. In addition to shielding and nuclear analysis, his activities include programming related to trajectory analysis, component design, interior ballistics and structures analysis. This support requires familiarity with the FLOCO system of programming as well as FORTRAN II and FORTRAN IV.

Aerojet, 1957 to 1962, Group Leader, Rocket Engine Component Development. Direction of programming support for development of rocket engine components, especially turbopump and nozzle designs was administered by Mr. Redden. These codes were written for the IBM 704 and 7094 computers in SAP, SCAT, and FORTRAN II. Engine simulation, data reduction, statistics, reliability, and structures evaluation codes were written by this group under his cognizance.

General Electric, 1954 to 1957, Programmer. Mr. Redden developed programs used in turbine and pump design and the reduction of test data for the IBM 701 computer.

U.S. Army, 1951 to 1954, First Lieutenant. Mr. Redden served in the Artillery in Korea and as an instructor in the Artillery school.

Publications

Polaris and the Digital Computer, P. L. Redden, Polaris Training Bulletin

D. Rogers

Education

B.S. (Summa cum Laude), Engineering Physics, Univ. of Calif., Berkeley, 1958
M.S., Nuclear Engineering, Univ. of Calif., Berkeley, 1959
(Oak Ridge Nuclear Science and Engineering Institute Fellow, 1958 to 1959)

Professional Experience

Aerojet, 1959 to present, Senior Nuclear Engineer. Mr. Rogers performs radiation and analyses related to ground testing of nuclear rocket engines including calculation of radiation scattering, nuclear heat generation, neutron activation and predicted radiation environment. He has also conducted shielding and heat transfer studies on the ML-1 gas cooled reactor and the MARK II AEDC (Arnold Engineering Development Center) Environment Chamber. His ML-1 analyses included design of fuel casks and auxiliary systems as well as shield design.

Institute of Engineering Research, 1958, Junior Engineer. Mr. Rogers conducted semi-theoretical research studies of nucleate boiling heat transfer.

Publications

Preliminary Evaluation of the ML-1 Shutdown Shield Performance, D. Rogers and H. O. Wittum, Jr., Trans ANS 6 (1963)

John A. Vreeland

Education

B.S., Cum Laude, Physics, Presbyterian College, 1949
M.A., Physics, University of Wisconsin, 1951
Ph.D., Physics, University of Wisconsin, 1955

Experience

Aerojet, 1962 to present, Manager, Nuclear Analysis Department. Dr. Vreeland, as Manager of Nuclear Analysis is responsible for the nuclear analysis support required throughout REON. In addition to determining the radiation environment of the NERVA Nuclear Rocket, he directs the nuclear analysis required in the review of facility shielding criteria, hazards documents, shielding designs, as well as the analyses performed in support of Advanced Concepts work.

Dr. Vreeland has participated in such non-corporate activities as Nuclear Rocket Propulsion Symposium conducted at the University of Florida in 1962.

Rocketdyne, 1960-1962, Senior Nuclear Specialist. Dr. Vreeland was responsible for directing the Rocketdyne efforts in the design of very large (NOVA Second Stage) nuclear rocket engines. He also directed the necessary reactor analysis for design of advanced nuclear rockets. Dr. Vreeland conducted a company-sponsored 60-hour course in Nuclear Rocket Theory and course in Modern Physics and Nuclear Physics at the University of California at Los Angeles.

Westinghouse, 1955 to 1960, Senior Nuclear Engineer. Dr. Vreeland was responsible for the technical direction of a reactor physics group. This involved evaluating recent cross-section measurements and predicting cross-sections using various nuclear models of theoretical physics. Dr. Vreeland also had the technical responsibility of reactor analysis for particularly challenging situations

John A. Vreeland

as exemplified by the Westinghouse Variable Loading Test Reactor as well as the planning and execution of related experimental programs at the critical facility. He directed the reactor kinetics program of the Pennsylvania Advanced Reactor.

University of Wisconsin, 1949 to 1955, Research Assistant. Dr. Vreeland conducted a research program measuring and interpreting the nuclear cross sections of various nuclear species. Included in this program were the isotopes of Tungsten, Tin, and Tellarium.

Publications

The NERVA Neutron Environment as Affected by the Angular Distribution of Leakage Neutrons, W. R. Butler and J. A. Vreeland, Trans. ANS 6 (1963)

Comparisons of Some Predicted Reactivity Perturbations with Experiment, Trans. of ANS 3 1,31 (1960)

A Reactor Fuels Measuring Facility, Trans. of ANS 2, 2,88 (1959)

Isotope Shift in the Spectrum of WI, Journal of Physical Society of Japan 13 663 (1958)

On the Production of Experimental Quantities of Isomers, Bull. of APS 2 224 (1957)

The Hyperfine Structure of Tungsten, Physical Review 83 229 (1951)

E. A. Warman

Education

B.S. Physics, Scranton University, 1956

Professional Experience

Aerojet, 1963 to present, Manager Shielding Analysis Department. Mr. Warman is responsible for analysis and shield design for the NERVA Program. These responsibilities include analysis and evaluation of shielding for ground tests of nuclear rocket engines at the Nuclear Rocket Development Station (NRDS), as well as formulation of shielding requirements for flight systems and advanced concepts employing nuclear rocket propulsion systems.

General Dynamics, 1958 to 1963, Shielding Engineering Supervisor. Mr. Warman was responsible for radiation analysis, shield design, and testing in development and construction of nuclear submarine propulsion plants for a number of POLARIS and attack-class vessels. He was also responsible for radiation analysis and shielding studies on General Dynamic's commercial and military sub-marine tankers, nuclear-powered destroyers, and proposal efforts such as the MH-1A program (which employs a $10Mw_e$ reactor mounted on board a converted liberty ship for the U.S. Army Corps of Engineers as an amphibious electrical generating system). He served as an instructor in engineering training programa and lectured in nuclear engineering.

He was guest lecturer in radiation shielding for senior cadets at the U.S. Coast Guard Academy, 1961 and 1962.

Pratt and Whitney Aircraft, 1956 to 1958, Nuclear Engineer. Mr. Warman worked in reactor physics and shielding analysis at the Connecticut Aircraft Nuclear Engine Laboratory (CANEL) and represented Pratt and Whitney on a 14-month assignment at the Oak Ridge National Laboratory. He was assigned to the staff of the Lid Tank Shielding Facility of the Applied Nuclear Physics Division.

E. A. Warman

Publications

Neutron Radiography in Field Use, E. A. Warman, Materials Evaluation, Vol 23, No. 11, November 1965

Radiation Exposure to Crew During Operation of a Manned Nuclear Space Vehicle, E. A. Warman, et al, RN-TM-0294, January 1966

V. SUPPORTING INFORMATION

Aerojet's capability of performing a synthesis of shielding methods is conveyed in this section in terms of experience accumulated in related projects and the corporate facilities which, as a consequence, have been developed.

A. RELATED EXPERIENCE

Aerojet has an extensive background in nuclear space projects as well as projects specifically oriented toward shielding. This experience is concentrated in the NERVA and SNAP programs and various related efforts.

1. Nerva Program (Contract SNP-1)

As prime contractor for the NERVA program Aerojet must perform a variety of radiation analyses intimately related to the requirements of the MSFC shielding study. These analyses are related to the design of the engine system as well as the individual components including the nuclear subassembly and the specification and interpretation of the various nuclear tests conducted in the program.

a. System Design

In addition to the NERVA engine the program includes a sequence of developmental configurations each of which requires an analytical prediction of its environment. In addition to the NRX configurations which are specifically for testing the reactor, the program has considered an NRX/EST, an XE, and an E-Engine configuration all employing an 1100 Mw Technology Engine. The NRX/EST is the first engine test of the program and employs a close coupling between the turbopump and the reactor. Extensive shielding was required to effect an early test of this configuration. This shielding was designed by Aerojet to protect the engine components and extensively analyzed to ascertain that the requirements were met. This analysis is documented in Reference 44. The XE configuration is similar to the NRX/EST in that the engine uses some components that are not radiation hardened. Since this configuration is downward firing with the turbopump located along the axis of the engine, a different shield was required which mates with the ETS-1 facility and which has many penetrations. This shield was also designed by Aerojet and the required analysis including studies of the penetrations is almost complete. A preliminary analysis is given in Reference 42. The E-Engine is another version of the Technology Engine except that it has a Lithium Hydride-Iron internal shield instead of an aluminum flow simulator. It is postulated that the engine components are radiation hardened. Analyses of this configuration are given in References 38, 39 and 40.

As the NERVA program evolves it is certain that additional configurations will be developed culminating with the ultimate 5000 Mw NERVA engine. As the concept of this engine develops, Aerojet is constructing the necessary models to predict the environment of this configuration as well so that the problems of designing such a system can be resolved.

It is evident that the radiation and shielding analyses described have the identical character of those for which a synthesis of methods is sought. Whereas the treatment employed is not optimum in all cases, difficulties in analysis for which acceptable solutions have been found constitute an excellent base from which to initiate an effort to select an optimum approach.

b. Component Design

The design of components to operate in the intense nuclear environment of a NERVA engine produces numerous problems. These problems are eliminated or reduced for some components by the design of appropriate shielding. For other components, however, the necessary relation to the reactor is such that shielding is not possible. Even these problems are readily solved if the environment can be specified within reasonable limits. Examples of such components include the nozzle and the pressure vessel. Prediction of nuclear heat generation rates in these components and specification of the nuclear environment with whatever shielding is necessary is performed by Aerojet for all components as part of the NERVA program.

Though probably more detailed than the requirements of the MSFC program the analyses described here have very stringent limitations on accuracy. This experience is therefore of great value in establishing comparative precision capabilities of various analytical techniques. The design of supplementary shielding is directly applicable to the objectives of the MSFC program.

c. Nuclear Subassembly

The nuclear subassembly is a significant portion of the NERVA engine, especially in the context of radiation analysis. Although Aerojet has a strong subcontractor to manufacture this part of the engine the responsibility for performance necessarily resides with the prime contractor. In addition to maintaining cognizance over the activity, Aerojet retains the responsibility of defining the criteria to be met. In relation to radiation analysis the significant contribution of Aerojet is identification of the function of the shielding which is part of the nuclear subassembly. To write criteria for this shield design Aerojet familiarizes itself with the radiation problems associated with nuclear rockets and modifies the objectives of the nuclear subassembly shielding to eliminate problems as they are identified. In tests planned for the Technology Engine to date, shielding problems are resolved using shields external to the nuclear subassembly. The criteria for the NERVA engine shield, based on the development of a turbopump with a zero net positive suction pressure capability, is presently to control nuclear heating of uncooled structures.

The analyses required to establish criteria for the nuclear subassembly shield and to evaluate the extent to which the criteria are met are analogous to those required for system design. The existence of this effort at Aerojet merely amplifies the depth of experience of great value in selecting and utilizing radiation analysis methods in this program.

d. Nuclear Systems Tests

All of the configurations developed in the NERVA program to date have been for ground testing. The radiation analyses for these systems therefore considered all the scattered and secondary radiation resulting from such tests. The execution of these tests nevertheless results in two additional considerations: (1) the ability to make measurements with which to evaluate the radiation predictions and (2) the incidence of test stand contamination from fission products and neutron activation. For every test Aerojet specifies the dosimetry required to verify the exposure to the components and whatever additional dosimetry is needed to improve its analytical prowess. For example, on the NRX/EST test now being conducted, Aerojet has an experiment which exposes gamma ray and neutron dosimetry to: (1) the total radiation which exists on the test pad; (2) only that radiation incident from the test itself; (3) only that radiation which is reflected from the test cell wall; and (4) only that radiation reflected from the test pad. These components are segregated with an ingeneous arrangement of shields. The results of this experiment will provide an extremely demanding test of the methods used to predict the test results. Additional measurements distributed throughout the complicated geometry of the test article will provide additional bases for evaluating calculations.

The need to predict neutron activation levels at the test pad following a test provides the development of additional calculation methods including the distribution and spectra of incident neutrons, the thermalization and capture of the neutrons, and the decay of the radioactive products of neutron capture. An additional experiment being performed by Aerojet in the current NRX/EST test is the irradiation of samples taken from the shadow walls of Test Cell "A," Test Cell "C" and ETS-1 as well as a soil sample from the Nevada Test Site. These samples are being exposed with measurements of the neutron flux and spectra. They will be subjected to a thorough radio chemical analysis following the test to improve the Aerojet technique in predicting Test Cell activation.

The critique of calculation methods that results from comparison of predictions with measurements in the very complicated geometry is invaluable to the program to be performed for MSFC. Such comparisons have already improved the Aerojet approach to analysis and such improvements will certainly continue. The value of beginning a selection of methods with many of the comparisons already completed should not be overlooked. Some comparisons of experiment with calculations that have been made are included in the appendix.

e. Radiation Effects

The need to develop engine components compatible with the NERVA environment produces a requirement to test materials in a comparable radiation environment so that the design selections can be made with confidence. A similar requirement to proof-test components in appropriate nuclear fluxes exists. To support such a radiation effects testing program, Aerojet has studied all the reactor facilities available which can approximate such fluxes. The environment of each test is then calculated in the course of specifying the test to ascertain whether the experiment will accomplish its objective.

These calculations not only augment the Aerojet experience with radiation having different spectra and distributions than exist in NERVA but they also prepare Aerojet to design experiments in these facilities specifically for evaluation of analytical methods. Although Aerojet is not proposing that such experiments are necessary, it is possible that they might begin to appear attractive as the program proceeds. The application of experience with a variety of radiation sources to the MSFC program will clearly expedite the progress.

2. SNAP Program

Aerojet is the prime contractor for the SNAP-8 system for electrical conversion. Moreover the related experience from this program is expanded by additional studies of the system some of which are specifically related to radiation environment.

a. SNAP-8 Conversion System (Contract NAS 5-417)

In designing the power conversion system for SNAP-8, Aerojet must assure adequate resistance to the radiation environment. Thus calculations are performed to predict the system environment and to establish the shielding requirements of the system. Moreover tests are to be performed to demonstrate system reliability in the anticipated environment.

b. SNAP-8 Shielding Analysis (Contract NAS 5-417)

For the SNAP-8 Program an analysis has been performed of the radiation distributions to be expected near the SNAP-8 reactor in a number of realistic vehicle configurations. Particular areas of interest were neutron activation, secondary gamma production, and radiation scattering from the protuberant SNAP-8 radiators that will also be present in almost all other nuclear space power systems. An effort was also made to select components which would be least affected by high radiation levels. A number of different system configurations were examined and the resulting payload dose rates calculated.

c. Aerospace Environmental Chamber (Contract AF 40(600)-953)

Under contract to the Air Force, Aerojet studied nuclear design criteria for a large aerospace environmental chamber. Reference designs were established for space power reactor systems (fast and epithermal) using various conversion devices (turbo -electric, thermionic, thermoelectric). Problems associated with environmental testing of these reactor systems in the induced space environment of the chamber were investigated as a basis for determining the chamber design criteria. Both neutron and gamma radiation scattering analyses were performed assuming single and multiple scattering in order to determine the payload exposure and optimum positioning of the test systems in the test chamber.

These studies of SNAP systems have given Aerojet a familiarity with the problems associated with radiation analysis of the SNAP systems and a solution to these problems that works. As with the NERVA analyses, the specific solutions may not be optimum but the experience provides a solid basis from which to begin the MSFC study.

3. Other Related Experience

In addition to the experience described above which relates specifically to the requirements of the snythesis of analytical methods, Aerojet has had extensive experience in intimately related programs. Such experience includes design of hardware for the Los Alamos reactor tests, the design of nuclear rocket testing facilities, related activities sponsored by Aerojet, and selected shielding programs.

a. Phoebus-2 Nozzle Program (Contract SNP-35)

Prediction of nuclear heat generation in the large diameter Phoebus-2 nozzle is a significant part of the design since the ability to design without external cooling is marginal and excessive conservatism in the prediction of nuclear heating cannot be tolerated. These calculations are complicated above those performed for NERVA components because of the massive facility shield being designed in the immediate proximity of the nozzle flange. Using Monte Carlo analysis of the heat deposition from gamma radiation in the nozzle, Aerojet was able to reduce the uncertainty in the calculated heating rates to the point where a design without external cooling was possible. The analyses in support of this program are documented in Reference 41. After reviewing the Aerojet calculations for this program, Los Alamos requested that Aerojet extend its analysis to include the Phoebus-2 pressure vessel as well and provide the results to the manufacturer for his design.

The calculation of radiation environment in support of this program does not differ significantly from that employed in support of the design of NERVA components. The significance of this endeavor to the MSFC program is that the Aerojet analysis was reviewed by an independent and competent agency of the government and not only considered adequate but worthy of use more extensive than intended by Aerojet.

b. Facility Design

In addition to the responsibility of designing the NERVA engine Aerojet was chosen to design the facility (ETS-1) in which the Technology Engine is to be tested and the preliminary planning for a second similar facility (ETS-2) (Contract SNPC-4). This design activity included a great deal more extensive radiation analysis than that associated with the use of the facility for a specific test. The calculations performed include operational and post-run isodose plots, nuclear heat generation, material activation, radiation damage, fission-product plateout, facility shielding, and nuclear hazards as related to test stand design, site separation, and site safety. These analyses are documented in Reference 45.

It is probably significant that when design activity on the test stand (E/STS 2-3) intended for NERVA engines and stages was initiated, Aerojet was requested to provide an advisor to the government to cover the nuclear aspects of the design as well as the engine aspects. These two Aerojet representatives are members of a team established to monitor the design effort and are the only industrial members of the team.

The choice of Aerojet to represent the government's interest in the nuclear aspects of the facility design is further indication that an independent appraisal of Aerojet capability in radiation analyses related to nuclear rockets is outstanding.

c. Company Sponsored Work

In addition to the extensive involvement of Aerojet in various contracts requiring extensive radiation and shielding analyses, Aerojet sponsors a considerable level of related activity with objectives defined by Aerojet. Although studies of fluid and metalic core reactors are related, only two programs which are directly related will be described: (1) the storing of enthalpy from radiation absorption and (2) radiation exposure to a crew during operation of a manned nuclear space vehicle. The first of these considers the mechanisms of radiation absorption and attempts to predict the quantity, distribution, and residence times of energy which is introduced without immediate thermal consequences. The results of this activity are documented in Reference 46. Although the radiation analysis for this effort is not sophisticated, it is significant in predicting the attenuation of radiation through the propellant especially when related to temperature measurements.

The second study uses a standard model of the module for the final stage of a nuclear system and a reasonable choice of crew compartment configuration with its space shielding. Radiation to the crew is predicted as a function of remaining propellant and then related to a specified operating cycle. After a detailed consideration of radiation transmitted through the propellant scattered from the walls of the tank and produced from neutron capture in the structure and the propellant; the exposure to the crew is evaluated. The study concludes that, if the allowable dose to the crew from on board radiation is 10% of that expected from space radiation, there is no weight penalty for biological shielding associated with the use of a NERVA nuclear rocket. This study is documented in Reference 43.

The methods used in these analyses are exactly those required to satisfy the prerequisites of the MSFC program. The additional experience gained in performing these studies is of value to the program because of the added limitations to various techniques which were identified in the course of the work. While it is again true that an optimum solution to the analytical problems may not have been reached, an approach which does suffice in these circumstances was defined.

d. Nuclear Rocket Propulsion Systems (Contract AF 33(616) 552)

This study was executed before the NERVA contract. The results are documented in Reference 47. The radiation analysis in this study is the same as that required by the NERVA program and by the MSFC program. However Aerojet performed this work prior to developing the powerful techniques that have evolved in the NERVA work. This effort does, nevertheless, mark the beginning of the Aerojet growth in analytical tools for studying radiation environment of nuclear rockets and is included here to suggest how the MSFC program might have begun were Aerojet's NERVA experience not available.

e. ML-1 Shield Design (Contract AT (10-1)-880)

Shielding against shutdown radiation was the primary objective of the ML-1 shield design. A secondary objective was to reduce operational radiation levels insofar as the reactor package weight limitations would permit.

To achieve these objectives, a series of experiments were performed at the Lid Tank Facility of the Oak Ridge National Laboratory in which the proposed shield was mocked up in slab geometry. The detailed arrangement of the selected shield configuration was established as a result of operational and shutdown measurements made at this facility.

In addition to the Lid Tank Experiments, other tests, measurements, and calculations were performed to investigate other design characteristics of the ML-1 shield. The dose rate from capture gamma sources during operation and activation sources for shutdown conditions were then calculated using these neutron flux distributions. Removal theory was used to calculate the fast and thermal neutron flux distributions in the ML-1 shield for comparison with the results of the SNG code and Lid Tank Measurements.

ML-1 operational levels and heat generation rates were obtained using a multi-group, multi-region, gamma ray attenuation code designed for computing gamma-ray heating and dose rates in infinite or semi-infinite slab shields. This code was used to calculate the operational levels measured in the Lid Tank Experiment, using source strengths calculated for each region. After it was determined that the calculated results were in good agreement with experimental measurements, detailed calculations were performed on the ML-1 shield.

Most of the experimental data used were obtained from ML-1 shielding experiments performed at Idaho. Measurements were made with the reactor at various stages of disassembly (i.e., end cone, the plenum, then baffle, then gas duct removed and shield water drained) as the reactor operated between 0 and 1.8 Mw of thermal power. In this series of experiments, measurements were made of photoneutron levels, radiation steaming through ducts, slots, etc., activation, air scattering, operational neutron and gamma levels, and the effect of changes in boron concentration.

The GCRE Photoneutron Production Experiment measured the relative multiplication photoneutron dose rates were verified by experimental measurements.

These analytical and experimental studies of the ML-1 shield design (Reference 48) add significantly to the Aerojet background. The value of this background should not be minimized, however, since continued reference to experimental data is the essence of radiation analysis. These ML-1 calculations increased Aerojet's capability to perform a synthesis of shield design methods very substantially by identifying which analytical techniques perform acceptably in given circumstances.

B. LIST OF OPERATIONAL DIGITAL COMPUTER CODES

The digital computer codes for radiation analysis that are on a production status at Aerojet are listed in Table VIII. These codes are grouped according to analytical methods including the point kernel, the discrete ordinates and the Monte Carlo methods. the remaining codes, which treat gamma ray and neutron scattering, neutron activation, fission product inventory, cross sections data reduction, input data editing, radiation hazards and reactor kinetics, are grouped together under auxiliary programs. The table describes the basic theory, geometric representation, and functional characteristics for the codes.

TABLE VIII

COMPUTER CODES FOR RADIATION ANALYSIS AT AEROJET

I. POINT KERNEL METHOD

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
QAD-5	Multigroup gamma rays and fast neutrons Multi-region	Three-dimensional. Regions are formed by 7 types of boundary equations including both linear and quadratic equations. Source strength may be described either as a function or as distributed at various points.	Gamma ray attenuation: exponential attenuation with build-up. Neutron attenuation: (a) moments method solution of the Boltzmann transport equation, (b) modified Albert-Welton version of fast neutron removal theory.	Output may include any of the following combinations of data: Gamma ray attenuation and/or dose rate and/or spectra and/or heating rate; neutron attenuation and/or dose rate and/or spectra and/or heating rate. There is no limit on the location and number of detector points.
14-0	Multigroup gamma rays. A single fast neutron group, although provision is made for out-putting the neutron spectra. Multi- region	Regions are formed by translating and rotating rectangles and trapezoids. A number of "basic" regions may be described within each "master" region. Over-lapping of region boundaries is permitted. The source strength is described in cylindrical coordinates; it is assumed independent of Q. Separate exponential and/or cosine functions describe the axial and radial distributions.	Gamma Ray Attenuation: Exponential attenuation with buildup. Fast Neutron Attenuation: A modification of the Albert and Welton version of fast neutron removal theory. Neutron and Gamma Ray Spectra: spectral data computed by moments method solution of the Boltzmann transport equation is input in the form of bivariant polynominal coefficients.	Output may include any combination of the following: shield weight; fast neturon attenuation and/or dose rate and/or neutron spectra; gamma ray attenuation and/or dose rate and/or spectra. Any number of dose points within or outside the shield may be specified.

I, Point Kernel Method (cont.)

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
14-1	Same as 14-0	Same as 14-0 except as follows: The source strength distribution is not assumed independent of Q nor separable in r and z; the source strength distribution is described by inputing point values in cylindrical coordinates.	Same as 14-0	Same as 14-0
14-2	Same as 14-0	Same as 14-0 except as follows: A continuous or discontinuous source strength may be described by inputting point values in three-dimensional rectangular coordinates.	Same as 14-0	Same as 14-0
C-17	Multi-group Multi-region	Three-dimensional	Moments method and point-kernel method.	It calculates the neutron and/or gamma spectra, heat generation rates and/or dose rate at each of a group of point detectors due to each of a group of point sources.

II. DISCRETE ORDINATES METHOD

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
DSN	Multi-group Multi-region	One-dimensional: spherical, cylindrical, slab	Discrete S_n transport.	Iterates on K_{eff}, α, concentration or geometry. Computes neutron fluxes and volume integrated reaction rates. Will also solve either the inhomogeneous surface or volume source problem. Scattering can be either isotropic or linear anisotropic.

II, Discrete Ordinates Method (cont.)

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
DTK	Multi-group Multi-region	One-dimensional, spherical, slab or cylindrical.	Same as DSN.	This code incorporates new difference equations with greater flexibility and faster calculational time than the DSN code.
DTF-2	Multi-group neutron. Multi-region	One-dimensional, spherical, slab or cylindrical.	Same as DTK.	FORTRAN version of DTK.
DTF-4	Multi-group gamma rays and neutrons Multi-region	One-dimensional, spherical, slab or cylindrical	Gamma ray and neutron distributions based on the Boltzmann transport equation using the S_n discrete ordinates method.	FORTRAN version of DTK with additional capability to study gamma ray distribution.
TDC	Multi-group Multi-region	Two-dimensional, r, z	Discrete S_n transport	Similar to DSN but only includes isotropic scattering.
2DXY	Multi-group Multi-region	Two-dimensional, x, y	Discrete S_n transport	Similar to DSN but only includes isotropic scattering.
DDK	Multi-group Multi-region	Two-dimensional: R-O, R-Z and X-Y	Similar to TDC	This code incorporated all two-dimensional coordinate systems and includes both isotropic and anisotropic scattering.
2DF	Multi-group Multi-region	Two-dimensional.	Same as DDK	FORTRAN version of DDK.
III. MONTE CARLO METHOD				
ADONIS	Multi-group Multi-region	Three-dimensional rectangular	Monte Carlo	It calculates the solution to the transport equation for primary neutrons or gamma rays, and their standard deviations.

III, Monte Carlo Method (cont.)

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
FMC-N FMC-G	Multi-group Multi-region	Cylindrical, spherical, slab, others	Monte Carlo	FMC-N and G are programs which apply Monte Carlo methods to simulate neutron and gamma ray life histories, respectively, in a source shield configuration. The programs are designed for flexibility in the geometrical, material, nuclear, and source description of source-shield configurations and variance reduction techniques. Output includes absorption or energy deposition tallies, leakage and entrance tallies, flux and history tallies.
MCS/MCG	Multi-group Multi-region	Three-dimensional configuration of first and second-degree surfaces	Monte Carlo	Variance-reducing techniques written in FLOCO coding system is used to study the neutron and gamma penetration through shields including the scattering and reaction of neutrons on various nuclei.
05R	Multi-group Multi-region	Three-dimensional	Monte Carlo	It computes neutron transport with Monte Carlo method and stores the output on "collision tapes" for analysis. Separate routines are required to extract information from these tapes for use.
SANE-SAGE	Multi-group Multi-region	Spherical	Monte Carlo	It solves the neutron or gamma transport problem in spherically symmetric multilayer geometry.

III, Monte Carlo Method (cont.)

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
18-0	Multi-group Multi-region	Cylindrical, spherical, slab, others	Monte Carlo	Program 18-0 is a digital computer code that applies Monte Carlo methods to simulate neutrons and gamma ray histories in reactor shield assemblies. This program is designed to investigate and determine nuclear heating rates in reactor shield systems, and neutrons and gamma ray leakage distributions in energy and angle for an equivalent point source.
20-2	Multi-group			Approximates cross section dependence on energy. Input for FMC-N, G and 18-0.
20-3	Multi-group			Computer total macroscopic cross section and collision probabilities for specified material and composition. Input for FMC-N and G, and 18-0.
20-4	Multi-group			Program for averaging differential scattering cross sections. Input for FMC-N and G, and 18-0.
20-5	Multi-group			A program for preparation of spectrum tables from evaporation model. Input for FMC-N and G, and 18-0.
20-6	Multi-group			Computes nuclear excitation and transition probabilities from measured gamma ray intensities. Input for FMC-N and G codes.

[redacted] [redacted]

IV. NEUTRON DIFFUSION THEORY METHOD

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
WANDA	Few group Multi-region	One-dimensional: spherical, cylindrical, slab	Diffusion	Calculates k_{eff} and fluxes
AIM-6	Multi-group Multi-region	One-dimensional: spherical, cylindrical, slab	Diffusion	Computes k_{eff} and fluxes.
9-ZOOM	Multi-group Multi-region	One-dimensional: spherical, cylindrical, slab	Diffusion	Computes k_{eff} and fluxes
PDQ-2	Few group Multi-region	Two-dimensional x, y, or r, z	Diffusion	Computes k_{eff} and fluxes
9-ANGIE	Multi-group Multi-region	Two-diemsnional, x, y, or r, z	Diffusion	Computes k_{eff} and fluxes. Permits one group up scattering and two groups down scattering.

V. AUXILIARY SHIELDING CODES

GGG	Multi-group gamma Multi-region ray.	Three-dimensional Same as QAD	3 major processes by which the electromagnetic field of gamma ray may interact with matter are considered which include The Compton effect, photoelectric effect and pair production. For the Compton scattering the Klein-Nishina differential angular cross sections are used.	The code computes single scattering from an isotropic point source. Output includes gamma scattering and direct radiation gamma scattering and without build-up.

V, Auxiliary Shielding Codes (cont.)

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
GRACE-I	Multi-group Multi-region	Finite or semi-infinite slab, sphere, or truncated cone.		Computes gamma ray attenuation and heating in reactor shield. Choice of uniform or exponential source in multiregion slab, truncated cone or spherical shell. Choice of uniform or cosine source in spheres. Buildup factor represented by a double exponential.
GRADE	Multi-group Multi-region	Right circular cylinder and cylindrical shells		Calculates gamma radiation environment quantities around a nuclear rocket in a two-dimensional right circular cylinder
NGASD	One group gammas. Thermal neutron group. Fast Neutron group. Multi-region	Two-dimensional plane geometry. Planar source - any configuration	Emprical equations from FZK-122	NGASD calculates neutron and gamma ray air scattered dose rates in the plane of any planar source at between 20 and 900 feet from the source.
NGD-1	Multi-group (16-group)	Spherical	Diffusion equation with a point source.	Calculates neutron dose rates some distance from the reactor, and the gamma field caused by (n, γ) reaction in air.
NURSE-II		Three-dimensional	Anderson gaseous cloud growth model and Gifford's diffuson equation.	Evaluates radiation hazards resulting from the rapid release of fission products from a nuclear rocket engine. Computes several different doses at positions down- and cross-wind from an excursion point on or near the ground. The code is written in FORTRAN IV for an IBM 7094 with 32K memory.

V, Auxiliary Shielding Codes (cont.)

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
NEUSCAT	One-group fast neutron	Three-dimensional	Albedo Method	It calculates the fast neutron scattering from a plane source in a three-dimensional geometry using the Albedo method.
RISC		Three-dimensional	O. G. Sutton's diffusion equation	Computes the radiation dose to seven body organs as a result of breathing airborne radioactive material at a position down-wind from a reactor. Computations include cloud height and size as well as fallout and washout.
ACT II	Four-group			It provides gamma radiation sources in four groups from neutron activation for use in radiation level and shielding calculations.
FPIP	Four-group gamma ray		Perkins and King Data	It computes as a function of reactor operation and shutdown time, the total decay rates, total average beta energy release rate, total gamma release rate from fission products generated in a U-235 thermal reactor.
FPIC	Seven-group gamma ray		Perkins and King Data	Similar to FPIP with less isotope inventory but requires shorter machine time.
9-NIOBE	Multi-group Multi-region	Spherical	Transport theory	It performs numerical integration of the Boltzmann equation for neutron or gamma transport, time independent in a finite, multi-layered spherically symmetric configuration.

V, Auxiliary Shielding Codes (cont.)

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
DATA REDUCTION CODES				
AGMLR			Least square fit.	Does a least squares fit to a power series, or a sum of exponents, or a sum of sines, or a sum of cosines, or a sum of sines and cosines.
AS-71			Double precision determinant	Solves up to 400 simultaneous algebraic equations in double precision.
AUFOIL				The total neutron flux above a threshold energy and/or the thermal neutron flux is calculated from discriminated gamma counting data from single radionuclid threshold foils.
FADE-I				Program designed to reduce the amount of hand calculation required to analyze foil activation data. Code analyzes foil counting data and computes foil activity and the corresponding neutron flux. Analyzes up to 20 different power levels at irradiation.
FRENIC			Least square fit	Does a least squares fit to a sum of exponents.
REQADO	Multi-group			Reduces the output data of QAD program. It interpolates the data and lists the results for isodose, or isocalifacio plots. The program output tape can be used directly on a plotter.

V, Auxiliary Shielding Codes (cont.)

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
RTDCO	Multi-group			Reduces output data for both TDC and DDK. Performs scaling, summing, and interpolation functions and identifies locations for isoflux contours.

CROSS SECTION CODES

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
GAM	Multi-group Fast Neutrons		B-1 or P-1 approximation	Calculates a spectrum utilizing either the B-1 or P-1 approximation or will take a read-in spectrum. Obtains group averaged constants and transfer matrices for diffusion theory, isotropic Sn theory oranisotripic Sn theory. Calculates Doppler Broadened resonance cross sections for any material, including the fuels, utilizing Adler's method.

V, Auxiliary Shielding Codes (cont.)

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
MUFT-4	Few (1-4) group fast neutrons		B-1 or P-1 approximation and Age Theory	Calculates a spectrum utilizing either the B-1 or P-1 approximation for hydrogen, with other materials by Age theory. Has a resonance treatment which does not include Doppler broadening. Uses statistical theory of inelastic scattering. Yields group averaged constants.
MYS-PRINT	75 sub-group Fast neutron			Aerojet cross section library tape.
MYSTIC	Multi-group fast neutron		Infinite dilution and statistical theory	Using a read-in spectrum, will generate group averaged constants and transfer matrices for diffusion or isotropic Sn theory. Uses infinite dilution resonance treatment and statistical theory inelastic scattering.
SOPHIST V	Multi-group fast neutrons		P-6 approximation	Calculates elastic scattering cross sections and transfer matrices for diffusion, isotropic Sn theory and anisotropic Sn theory. Input is a read-in spectrum and Lengendre coefficients up through a P-6 in either the L or CM system.
SOPHIST I	Multi-group fast and thermal neutrons		Isotropic scattering Maxwellian gas	Calculates the transfer coefficients for diffusion theory over a read-in spectrum assuming isotropic scattering in the CM system. The moderator is treated as a Maxwellian gas.

V, Auxiliary Shielding Codes (cont.)

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
SOPHIST II	Multi-group fast and thermal neutron		Isotropic scattering Maxwellian gas	Calculates cross sections, including Doppler effects, assuming a Maxwellian gas model; spectrum is read-in.
ZOT	Multi-group Fast and thermal neutrons			Reduces multi-group cross sections to few groups. It collapses groups according to a given flux spectrum.
TNS	Multi-group Epithermal and Thermal neutrons		Free gas model	The code combines the features of CRISS-CROSS, GRAYMALKIN, SIGMA GAS and SPECTRUM.
BLOT	One-Group Thermal neutrons Multi-region	Cylinder	P-3 approximation	Solves the analytic soultion to the P-3 equation using Well's method. Yields call fluxes and averaged cross sections.
CRISS-CROSS	Multi-group thermal neutrons			Averages cross sections and related qualitites over a read-in spectrum.
GRAYMALKIN	Multi-group thermal neutron			Computes scattering transfer matrix utilizing a given kernel and spectrum.
SIGMA GAS	Thermal		Free gas model	Calculates scattering kernels utilizing the free gas model.
SPECTRUM	Thermal			Computes infinite medium spectrum for a known moderator kernel and up to 20 absorbers.
SUMMIT	Thermal			Calculates the scattering kernels for a polycrystalline material.
OTHER CODES				
FLAC	Multi-group neutron and gamma rays	Cylinder		Calculates neutron and gamma distribution above and around a cylinder for a specified surface leakage.

V, Auxiliary Shielding Codes (cont.)

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
GECOP				This program computes data points of geometrical configuration according to given equations. Its output tape is used then directly in a plotter to plot 2-dimensional configurations of any cross section in either X-Y, Y-Z, or Z-X plane in any scale. Both the plot and listing are used to check and evaluate the configuration or boundary equations used in computer programs such as QAD, GGG, FMC-N, FMC-G, MCS/MCG, etc. Complete or partial configuration can be drawn with boundary equation numbers and/or zone numbers printed as required.
AIREX			Reactor kinetic equations	Solves the coupled kinetic system using a 4th order Runge-Kutta method.
OSCAG	1 to 6 Regions		Reactor kinetic equations	This reactor kinetics program is for use in nuclear-rocket transient simulation. It solves a wide variety of problems involving the inter-relationships of many neutronics and thermal parameters. It may be used independently in analyzing a reactor that is uncooled or has constant power removal to coolant. In conjunction with an engine-analysis program, it can provide transient simulation for the cooled nuclear rocket engine.

V, Auxiliary Shielding Codes (cont.)

Name	Groups and Region	Geometry	Theory Used	General Description and Remarks
RTS			Reactor kinetic equations	Solves kinetic equations for six delay groups and arbitrary imposed reactivity variation with or without feedback proportional to integrated power.
HATCHET	Few group Multi-region	Spherical	S_n transport, hydrodynamic	Calculates the transient, burst characteristics of a super-prompt, concentric shell, pulsed reactor.

C. COMPUTER FACILITIES

The Computer Sciences Division, Sacramento Plant, Aerojet-General Corporation was established in 1956 to provide complete computing science services to all phases of plant operations. These include, but are not limited to, research, engineering trajectory calculations, engine simulations, quality control, production control, materiel, accounting management, and information retrieval. The current inventory of data processing equipment to support the above activities and which are available to the proposed program is given below.

DATA PROCESSING EQUIPMENT INVENTORY

1 - IBM 7094 Computer
    32K core storage
    Memory access of 2.00 microseconds with build-in overlap
    Two data channels
    15 - 729 Model VI Tape Drives

1 - IBM 7044 Computer
    32K core storage
    Two data channels
    1 - 2302 Disk Storage Units
    6 - 729 Model VI Tape Drive
    4 - 729 Model V Tape Drives
    2 - 1014 Remote Inquiry

1 - IBM 360/30
    16K core storage
    1000 Card per minute reader, 300 Card per minute punch
    3 - 1100 Line per minute printer
    5 - 2401 Magnetic tape units, Model 2

1 - IBM 360/30
    8K core storage
    1000 Card per minute reader, 3 Card per minute punch
    1 - 1100 Line per minute printer
    5 - 2401 Magnetic tape units, Model 2
    1 - Paper tape reader, 500 characters per second
    1 - Paper tape punch, 150 characters per second

V, C, Computer Facilities (cont.)

Peripheral equipment consists of:
1 - IBM 7711 Data Communication Unit
1 - Cal-Comp Plotter
40 - 029, 056, Keypunch and verifiers
2 - 519 Reproducers
2 - 557 Interpreters
1 each 083, 084 Sorter
1 - 188 Collator
1 - 407 Tabulator

D. DOCUMENT FACILITIES

Aerojet-General Corporation is authorized to receive, originate, and transmit classified and unclassified information for the Department of Defense and for the U.S. Atomic Energy Commission. The Aerojet Technical Information Department has all of the standard tools of research for both DOD and AEC materials. The following is a list of materials to which this department has access:

1. Technical abstracts - classified and unclassified,
2. A library of approximately 15,000 research and development reports - classified and unclassified,
3. The Armed Service Technical Information Agency (ASTIA) library at Oakland Army Terminal (approximately 100 miles). This library is one of the most complete Department of Defense libraries on the west coast.
4. Catalogue cards and an index system at AGN which aid literature search,
5. Access, with approval, to the University of California library, both classified and unclassified, and
6. A micro-card library of all unclassified documents issued through Technical Information Service Extension (TISE), received on an automatic distribution basis.

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VI. LIST OF REFERENCES

1. *Evaluation of Methods for Computing Nuclear Rocket Radiation Fields*, Lockheed Georgia Nuclear Laboratory, ER-8236, February 2, 1966.

2. The QAD Code - A Neutron and Gamma Ray Shielding Code, R. Malenfant, Private Communication (LASL).

3. *Shielding Computer Programs 14-0 and 14-1: Reactor Shield Analysis*, J. T. Martin, J. P. Yalch, and W. E. Edwards, XDC 59-2-16, January 23, 1959.

4. *Shield Penetration Programs C-17 and L-63*, D. M. Peterson, NARF-61-39T (FZK-9-170), General Dynamics/Fort Worth, December 1956.

5. *Calculations of the Penetration of Gamma Rays*, H. Goldstein and J. E. Wilkins, Jr., NYO-3075, June 30, 1954.

6. *A Monte Carlo Calculation of the Transport of Gamma Rays*, M. H. Kalos, NDA 56-7, July 1965.

7. *Modified Fast Neutron Attenuation Functions*, USAEC Report XDC 60-2-76, General Electric Company, February 1960.

8. *Penetration of Neutrons from a Point Fission Source through Carbon, Hydrocarbon*, J. Certaine et al., USAEC Report NDA-12-18, Nuclear Development Corporation of America, June 1956.

9. *Penetration of Neutrons from a Point Fission Source through Beryllium and Beryllium Oxide*, H. Goldstein and H. Mechanic, NDA 2092-9, June 1958.

10. *Penetration of Neutrons from a Point Isotropic Fission Sources in Water*, R. Aronson et al. NYO-6269, December 1954.

11. C-17 and R-29 Kernel Integration Code and Data Generator Frusta of Rectangular Pyramids and Cylinders, D. M. Peterson Contributed to RSIC, Oak Ridge National Laboratory, by General Dynamics/Fort Worth.

12. *Applications of Monte Carlo*, H. Kahn, RM-1237-AEC, The Rand Corporation, April 1956.

13. *Flexible Monte Carlo Program FMC-N and FMC-G*, J. J. Loechler and J. E. MacDonald, APEX-706, April 1961.

14. Monte Carlo Codes, FMC-N and FMC-G, E. E. Kinney, TIM-941, July 22, 1965.

15. *Addenda to GEMP-102 Describing Program 18-1*, J. P. Yalch and J. E. MacDonald, Report GEMP-272, General Electric Company, January 1964.

VI, List of References (cont.)

16. *Specialized Reactor - Shield Monte Carlo Program 18-0*, J. E. MacDonald, J. T. Martin, and J. P. Yalch, Report GEMP-102, General Electric Company, October 1961.

17. *Program 20-9, A Program for Computing Flux and Energy Deposition Rates at Specified Points in Program 18-1 Reactor-Shield Configurations*, J. P. Yalch and J. E. MacDonald, GEMP-273, January 1964.

18. *Shielding Computer Program 20-0*, J. E. MacDonald and J. M. Martin, APEX 610, September 1961.

19. *A General Monte Carlo Neutronic Code*, R. R. Johnson, LAMS-2856, May 1963.

20. *05R, General-Purpose Monte Carlo Neutron Transport Code*, D. C. Irving, R. M. Freestone, Jr., F.B.K. Kam, February 1965.

21. *Numerical Solution of Transient and Steady State Neutron Transport Problems*, Bengt Carlson, LA-2260, October 29, 1959.

22. *The DSN and TDC Neutron Transport Codes*, B. Carlson, C. Lee, J. Worlton, LAMS-2255, February 12, 1955.

23. *The DTK and DDK Neutron Transport Theory Codes*, Private Communication, W. J. Worlton (LASL).

24. *2DXY, Two Dimensional, Cartesian Coordinate S_n Transport Calculation*, J. Bengston et al, AGN TM-392, June 1961.

25. The TDC Code, Pratt and Whitney Aircraft-CANEL, TIM-847, September 1964.

26. *DTF Users Manual*, B. G. Carlson, et. al, UNC Phys/Math - 332, 1963.

27. *DTF-IV, A FORTRAN IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering*, K. D. Lathrop, LA-3373, 1965.

27A. *The NERVA Neutron Environment as Affected by the Angular Distribution of Neutron Leakage*, W. R. Butler and J. A. Vreeland, Trans. ANS 6 (1963).

28. *Calculating Transport and Heat Deposition of Gamma Rays by S_n Transport Programs*, R. K. Disney and H. C. Romesburg, Trans. ANS 8 (1965).

29. *GGG - A Gamma Ray Scattering Program*, R. E. Malenfant (LASL), Private Communication.

30. *Tabulated Differential Neutron Cross Sections*, R. Howerton, UCRL-5573.

VI, List of References (cont.)

31. A Nuclear Code to Obtain Multigroup Diffusion and Transport Theory Constants, AGN-TM-369 (Unpublished).

32. AGN-GAM, An IBM 7090 Code to Calculate Spectra and Multigroup Constants, T. P. Wilcox and S. T. Perkins, AGN-TM-407, April 1965.

33. A Series of General Electric Codes to Compute Monte Carlo Constants, GEMP-113 to GEMP-117 (Informal Reports).

34. Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies, G. E. Hansen and W. H. Roach, LAMS-2543, December 6, 1961.

35. Klein-Nishina Photon Cross Sections, APEX-487.

36. X-Ray Attenuation Coefficients from 10 Kev to 100 Mev, NBS-1003.

37. Gamma Ray Cross Section Data, R. A. Mann, DC 60-9-75.

38. Preliminary Report on the NERVA Radiation Environment, RN-S-0133, March 1965.

39. Preliminary Report on Nuclear Radiation Heating of Components in the NERVA Engine, RN-S-0134, April 1965.

40. Preliminary Report on Neutron Induced Activation of NERVA Engine Components, June 1965.

41. Nuclear Analysis in Support of the Phoebus-2 Nozzle Development Program, RP-SR-0001, August 1965.

42. NERVA XE-Engine Shield Design Description, RN-TM-0179, April 1965.

43. Radiation Exposure to Crew During Operation of a Manned Nuclear Space Vehicle, RN-TM-0294, 3 January 1966.

44. Evaluation of NRX/EST Engine System Radiation Environment, RN-TM-0188, 7 June 1965.

45. Engine Test Stand No. 2: Nuclear Analysis for Conceptual Design, (Classified) Aerojet Report No. 2785, 1964.

46. Evaluation of Hydrogen Conversion Phenomenon as Related to the NERVA Nozzles, J. H. Valentine et al, RN-TM-0326, January 1965.

47. Analytical and Preliminary Design Studies of Nuclear Rocket Propulsion Systems, Aerojet Report No. 1999, June 1961.

48.. ML-1 Shielding Design Report, AEC Report IDO 28609, (1963).

APPENDIX

Summary of Comparisons: Prediction vs. Experiment

As part of Aerojet's continuing effort to improve analysis methods for radiation environment, numerous comparisons have been made between analysis results and measured results. These comparisons include analysis vs. analysis, analysis vs. experiment, and experiment vs. experiment. Parallel analyses are performed with different methods on those problems requiring accurate predictions and for which suitable experimental data are not available.

Much like analysis, methods development is also required for the acquisition of suitable experimental data. The large amount of measured data taken from the series of NERVA and Kiwi tests require comparison and evaluation to determine the validity of techniques used for calibration and normalization. Some of the recently completed work is described in this appendix.

I. NERVA DATA

A considerable amount of data are available for the radiation environment about the NERVA and Kiwi reactors. These include gamma ray dose rates, neutron fluxes and neutron spectra.

The data were measured with dosimeters located on the meridian ring (MR) and on the equatorial ring (ER). The meridian ring is a semi-circle in the plane of the reactor axis with 5-feet radius originating from the center of the reactor core. The equatorial ring is a 5-feet radius circle concentric with the reactor and located at the core midplane.

A. GAMMA RAY DOSE RATES

Comparisons of gamma ray dose rates predicted by QAD-PJ and measured from NRX-A are shown in Figure A-1. The comparison data shown are obtained from the expression: $(D_m - D_c)/D_m$. In this expression D_m is the measured dose rate and D_c is the calculated dose rate.

Note that the agreement is within 10% at the nozzle end and within 34% at the reactor midplane.

B. FAST NEUTRON FLUX

Comparisons of fast neutron flux as predicted with TDC and QAD-P5 against measured values are shown in Figures A-2 and A-3, respectively. The differences between TDC predictions and measured data range from a high of -59.5% and a low of +61.5%. The differences between QAD-P5 (water moments data) range from a high of -87.2% and a low of +34.8%. Use of QAD-P5 results with beryllium moments data or carbon moments data lead to significantly larger differences.

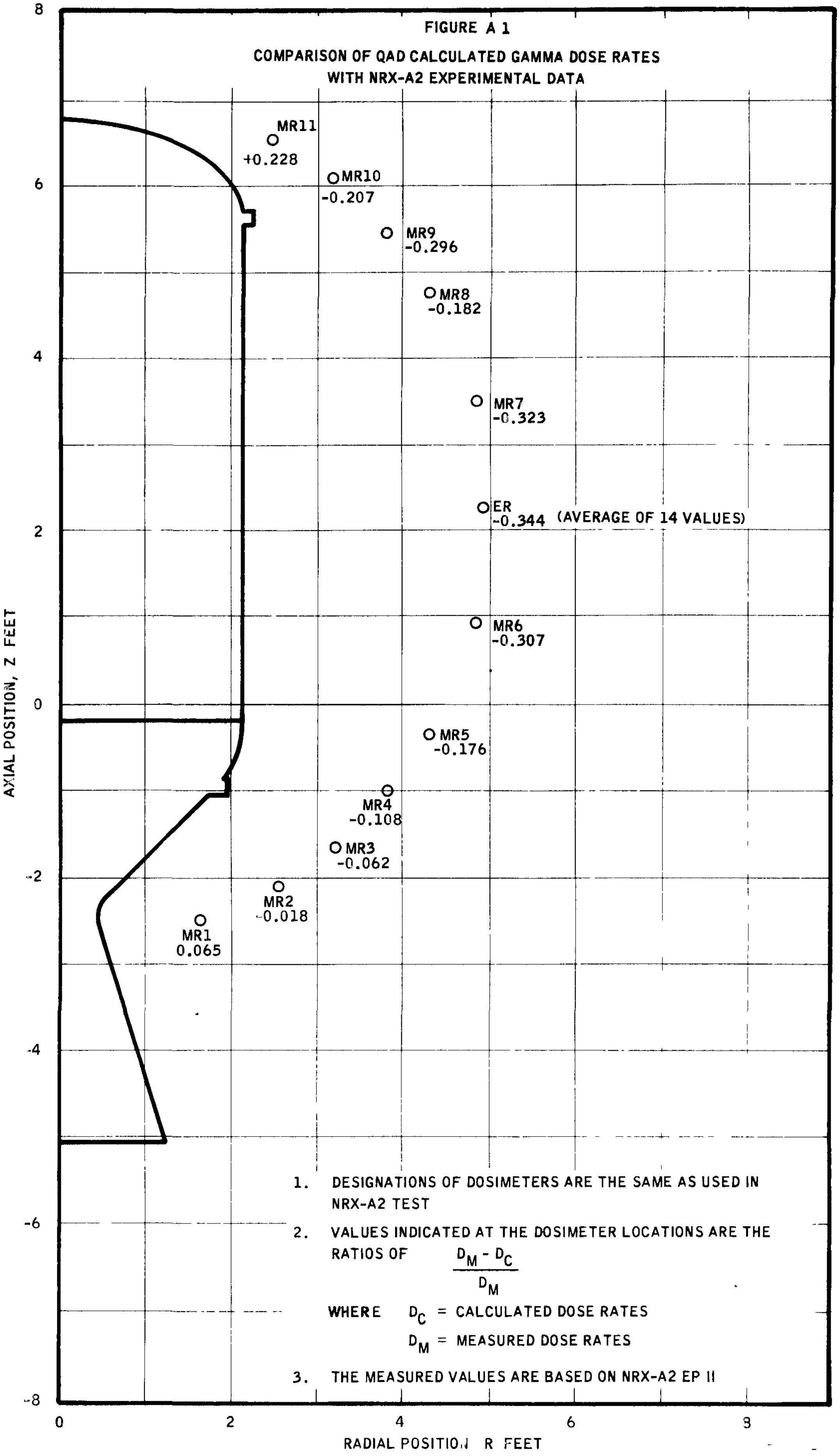


FIGURE A 1

COMPARISON OF QAD CALCULATED GAMMA DOSE RATES
WITH NRX-A2 EXPERIMENTAL DATA

1. DESIGNATIONS OF DOSIMETERS ARE THE SAME AS USED IN NRX-A2 TEST
2. VALUES INDICATED AT THE DOSIMETER LOCATIONS ARE THE RATIOS OF $\frac{D_M - D_C}{D_M}$

   WHERE D_C = CALCULATED DOSE RATES

   D_M = MEASURED DOSE RATES
3. THE MEASURED VALUES ARE BASED ON NRX-A2 EP II

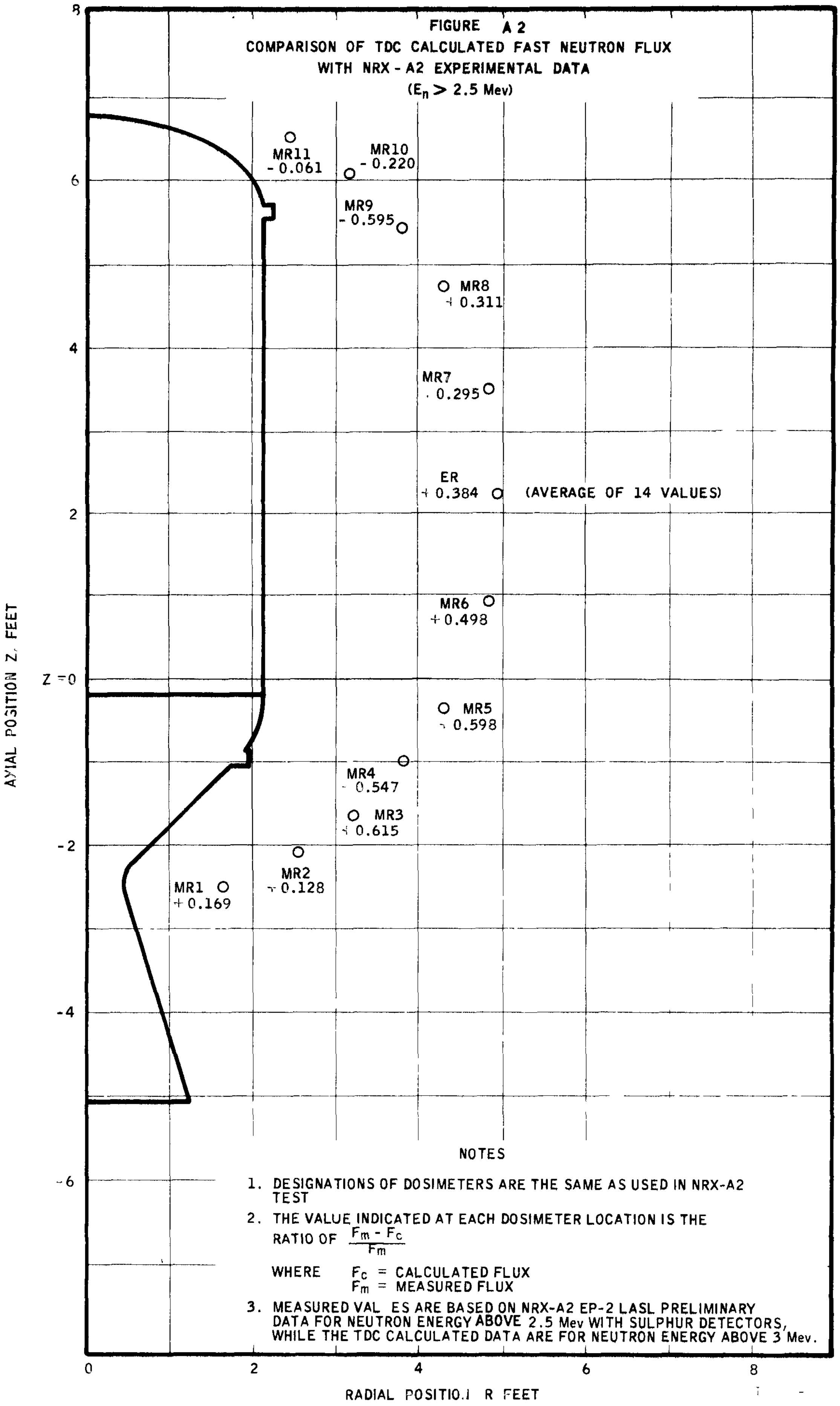


FIGURE A 2
COMPARISON OF TDC CALCULATED FAST NEUTRON FLUX
WITH NRX - A2 EXPERIMENTAL DATA
($E_n > 2.5$ Mev)

NOTES

1. DESIGNATIONS OF DOSIMETERS ARE THE SAME AS USED IN NRX-A2 TEST
2. THE VALUE INDICATED AT EACH DOSIMETER LOCATION IS THE RATIO OF $\frac{F_m - F_c}{F_m}$

   WHERE F_c = CALCULATED FLUX
   F_m = MEASURED FLUX
3. MEASURED VALUES ARE BASED ON NRX-A2 EP-2 LASL PRELIMINARY DATA FOR NEUTRON ENERGY ABOVE 2.5 Mev WITH SULPHUR DETECTORS, WHILE THE TDC CALCULATED DATA ARE FOR NEUTRON ENERGY ABOVE 3 Mev.

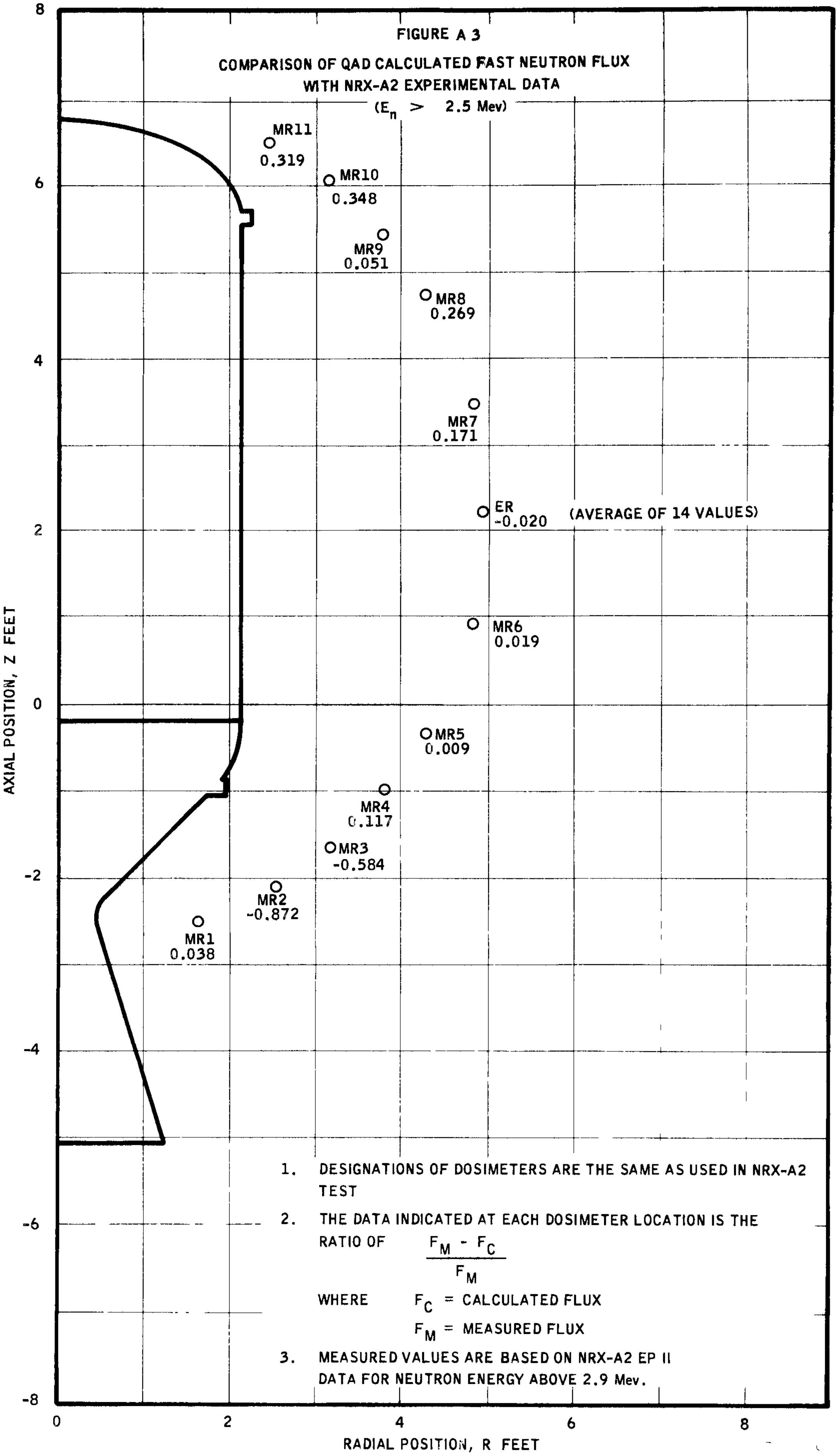


FIGURE A 3

COMPARISON OF QAD CALCULATED FAST NEUTRON FLUX WITH NRX-A2 EXPERIMENTAL DATA

($E_n > 2.5$ Mev)

1. DESIGNATIONS OF DOSIMETERS ARE THE SAME AS USED IN NRX-A2 TEST
2. THE DATA INDICATED AT EACH DOSIMETER LOCATION IS THE RATIO OF $\frac{F_M - F_C}{F_M}$

   WHERE F_C = CALCULATED FLUX

   F_M = MEASURED FLUX
3. MEASURED VALUES ARE BASED ON NRX-A2 EP II DATA FOR NEUTRON ENERGY ABOVE 2.9 Mev.

I, Nerva Data (cont.)

C. NEUTRON SPECTRA

Comparisons of neutron spectra as predicted with TDC and QAD-P5 against measured data are shown in Figures A-4 and A-5, respectively. TDC is shown to be in good agreement with measured spectra with some small deviations for neutrons below 2.5 Mev. QAD-P5 spectra are best when the results with beryllium moments data used. Note that the water moments data provided the best QAD-P5 agreement with measurements for fast neutrons.

II. MONTE CARLO ANALYSIS OF THE GENERAL DYNAMICS (ASTR) EXPERIMENT ON NEUTRON PENETRATION THROUGH HYDROGEN

A. INTRODUCTION

The prediction of fast neutron dose rates at the crew compartment of the nuclear rocket vehicle involves an evaluation of transmission of radiation through various quantities of liquid hydrogen. The hydrogen is distributed with a non-uniform thickness above the engine because of the conically shaped bottom of the propellant tank. This, together with the fact the crew compartment is located at a large distance from the tank bottom, permits neutrons to arrive at the payload with small angle scattering (and hence small energy loss), and with track lengths in the propellant that are shorter than those near the axis of the tank. This situation is illustrated by the track A B C (as opposed to A' B' C') shown in the Figure below. Under this circumstance, the usual point kernel procedure for penetration calculation will tend to under-predict the actual dose rate.

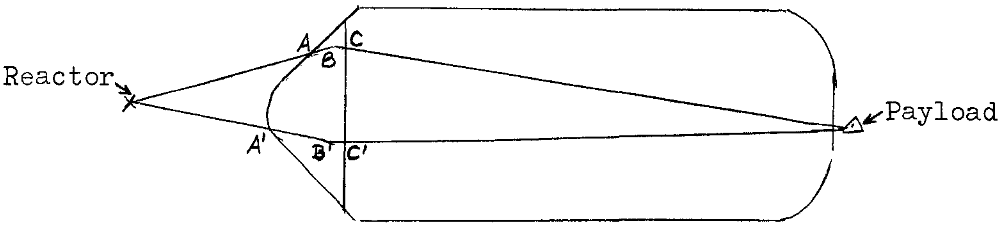
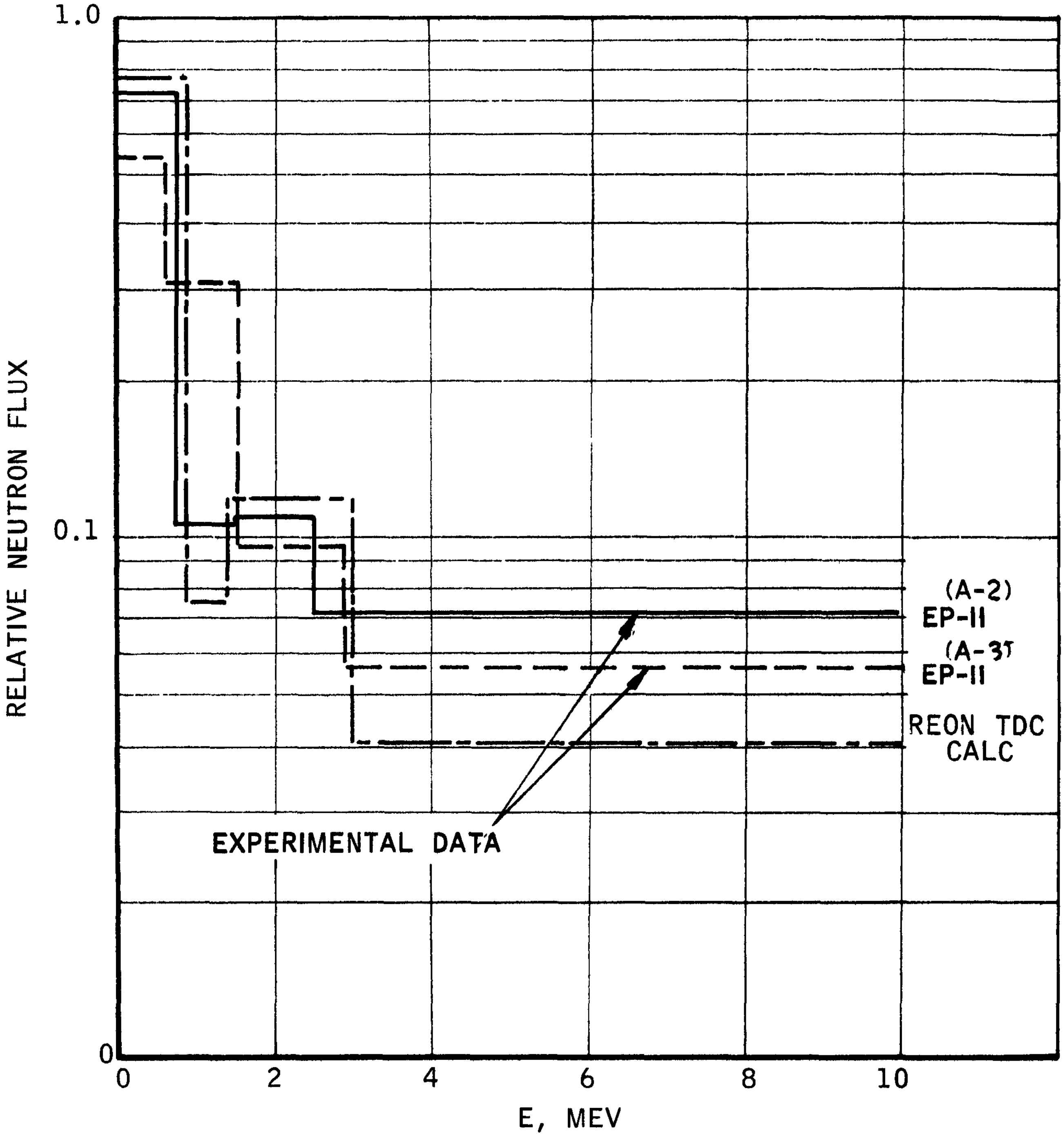


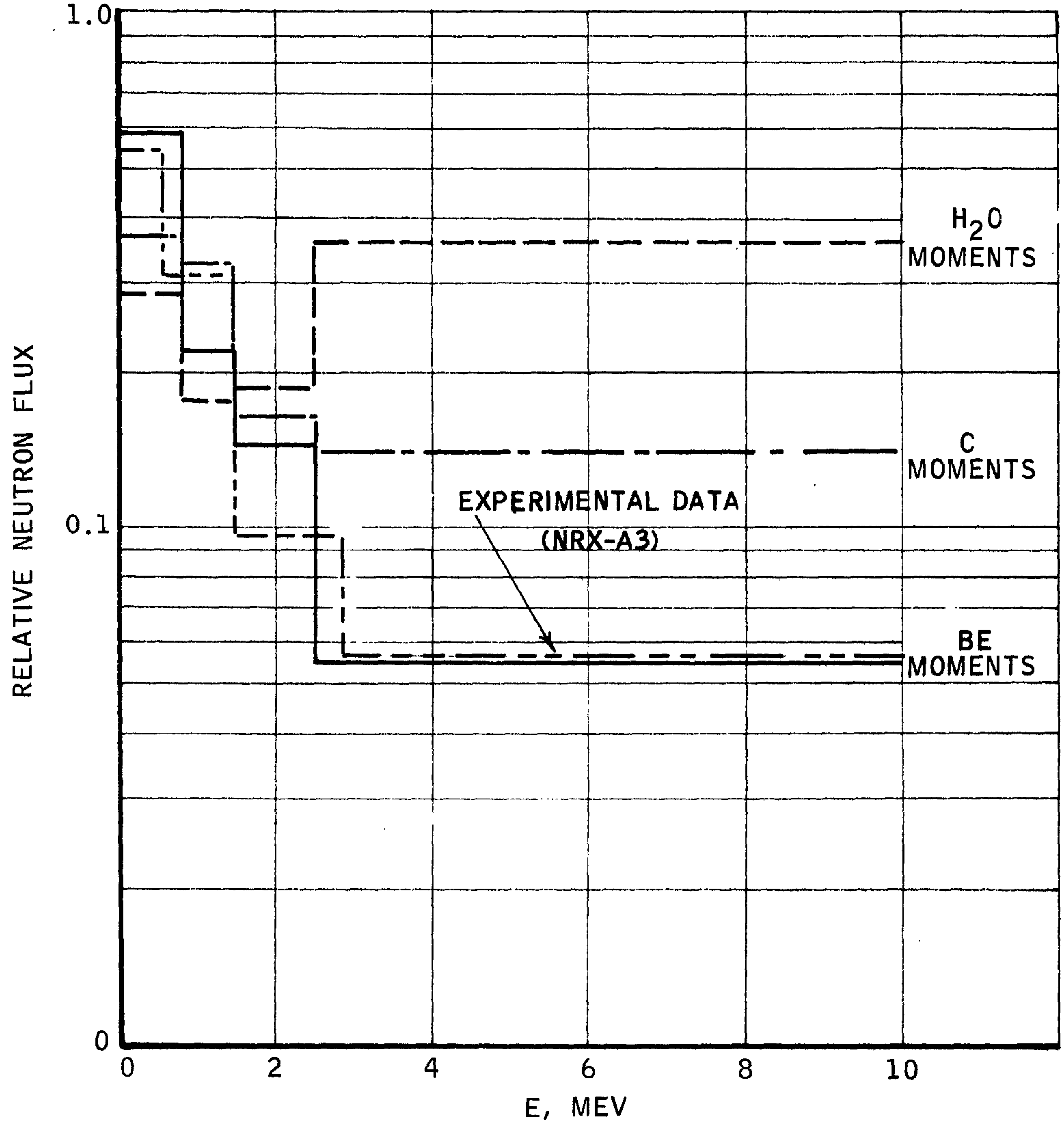
FIGURE A4



COMPARISON OF RELATIVE NEUTRON SPECTRA AT EQUATORIAL RING, EP-II

ANALYTICAL VS. EXPERIMENTAL DATA

FIGURE A 5



COMPARISON OF RELATIVE NEUTRON SPECTRA AT EQUATORIAL RING, EP-II

CALCULATED WITH QAD-P5 USING VARIOUS MOMENTS DATA

An alternative to the point kernel method for the solution of the above neutron transmission problem is the Monte Carlo procedure. To provide guides helpful in an efficient application of the Monte Carlo method, a limited analysis was conducted of a General Dynamics experiment[A1] on neutron penetration through liquid hydrogen. In this analysis the leakage spectrum of the ASTR (Aerospace Shield Test Reactor) was assumed as a source and an attempt was made theoretically to reproduce the experimental flux attenuation in hydrogen of neutrons with energies greater than 2.9 Mev. The basic calculations were performed with the General Electric Flexible Monte Carlo Code - Neutron, FMC-N.[A2]

B. ANALYSIS

1. Preliminary Calculations

Before any analysis of the experimental results were begun, calculations were performed with FMC-N to provide a test of the "exponential transformation" as a variance reduction scheme for deep penetration calculations. In this technique, variance is reduced by a more frequent sampling of the deep penetration event by use of input parameters which adjust cross sections. The bias introduced is removed by a simple deterministic correction, given by an exponential factor. An attempt to solve these transmission problems by using a straight Monte Carlo analogue without resort to any variance reducing scheme would result in a prohibitive expenditure of computer time.

The conditions for these initial calculations were identical with those assumed in a detailed study conducted by Burrell (Reference A3). The geometry consisted of a cylindrical slab of liquid hydrogen (density 0.07 gm/cm^3) whose diameter-to-thickness ratio is large. A monodirectional beam of 8 Mev neutrons are directed into the slab and the current of neutrons with energies greater than 0.01 Mev are counted. The hydrogen cross sections were obtained from BNL-325,[A4] and were chosen at a sufficient number of points, so that straight line segments connecting adjacent values yield a close representation of the actual cross section over the energy range of interest.

Slab thicknesses of 83.96 and 167.92 cm were considered. The results, represented as buildup factors (ratio of uncollided plus scattered current to uncollided current), are compared with those of Burrell below.

	Neutron Buildup Factor	
Thickness, cm	Present Calculation	Burrell's Result
83.96	4.3	4.4
167.92	6.5	6.6

The agreement is good, giving confidence in the exponential transformation option of FMC as a scheme for calculating deep penetrations.

2. Monte Carlo Mockup of the General Dynamics Experiment

The General Dynamics experiment "was designed to simulate the radiation source and liquid-hydrogen propellant-tank of a typical nuclear rocket system design."[A1] The experimental arrangement consisted of the ASTR (Aerospace Shield Test Reactor) and a 125-gal hydrogen dewar, positioned in the ASTR tank as shown to approximate scale in Figure A-6. For greater details, one is referred to References A1 and A5. The dewar, placed within a liner tank, is positioned so that it sits directly above the reactor. Neutron flux measurements were taken with foils at various points within the hydrogen tank.

In the present calculations the configuration was mocked up as shown in Figure A-7. Only one medium of interaction is assumed, namely liquid hydrogen. Several small detector regions are defined in the system which enable the evaluation of flux from the track length tally of the FMC-N program. Actually, 33 regions and 25 surfaces were defined in the calculation, but for the sake of clarity all are not shown in Figure A-7. The reactor is approximated by a point isotropic source located 6.73 in. from the center of the core and in the direction of the hydrogen tank. The location of this "effective" point source is taken to

FIGURE A 6

EXPERIMENTAL ARRANGEMENT

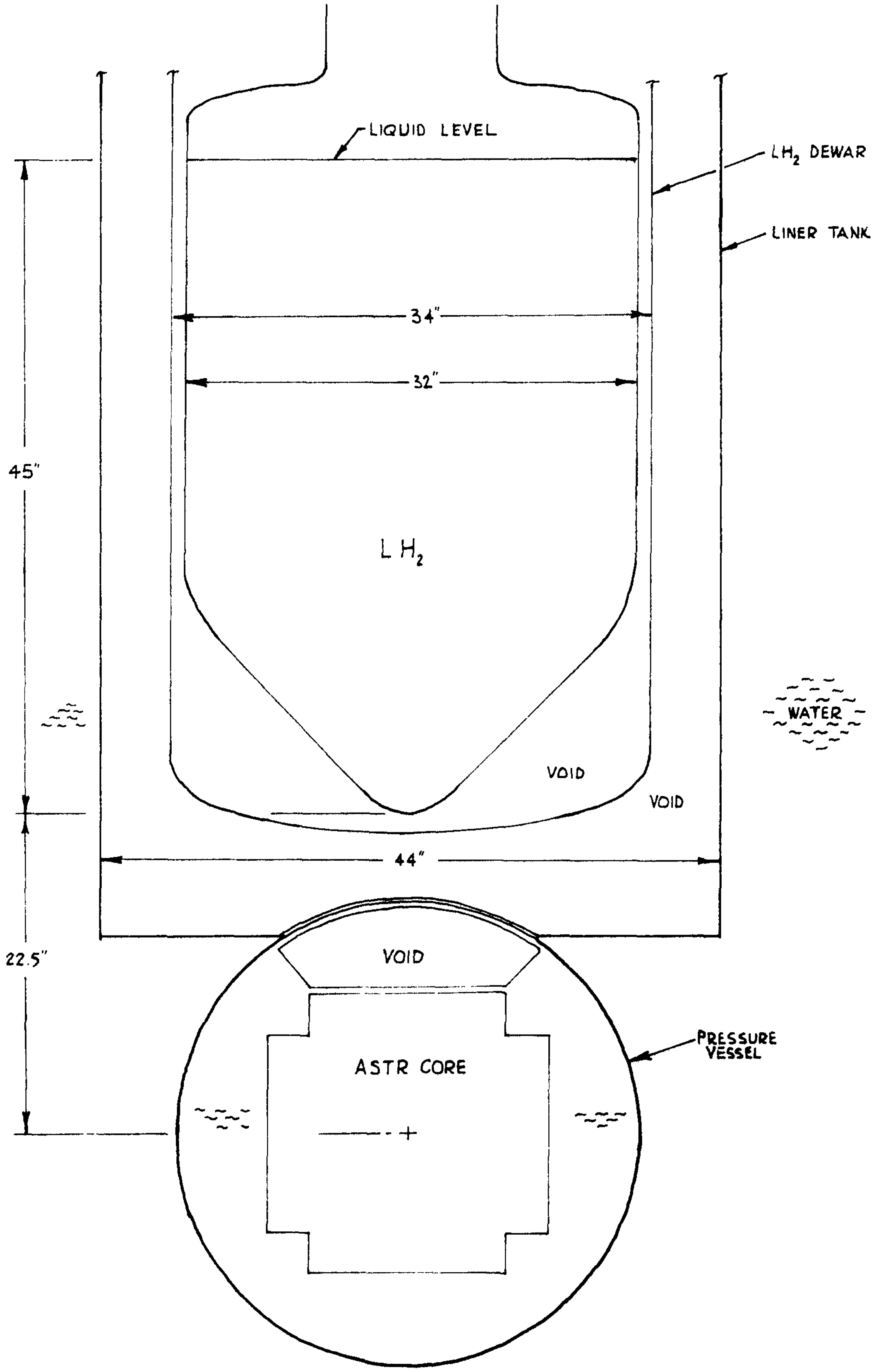
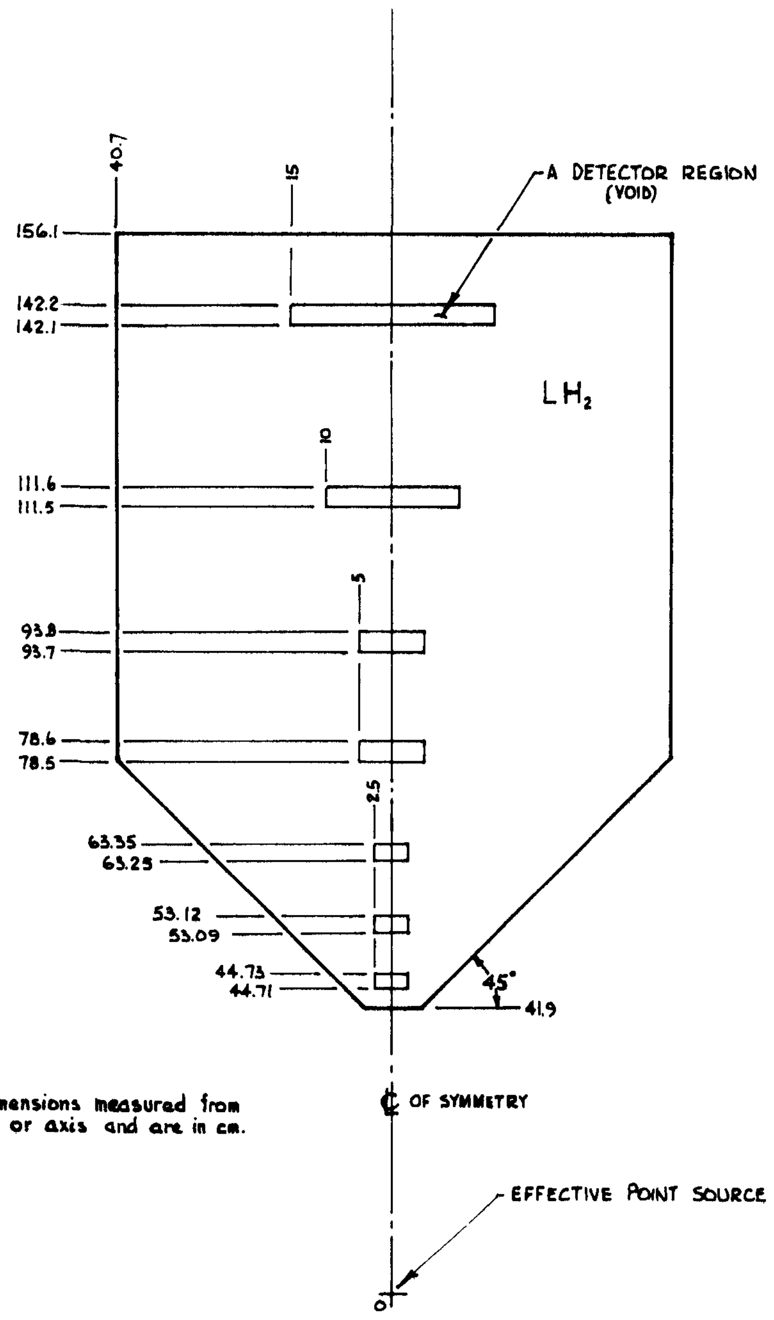


FIGURE A 7

MONTE CARLO CALCULATION MODEL OF HYDROGEN DEWAR



All dimensions measured from source or axis and are in cm.

be that point from which an inverse square relation maintains in describing attenuation of neutron flux in air. Using the data on ASTR radiation map in air,[A5] it was found that the above location of the "effective" point source yields an inverse square attenuation that equaled the experimental attenuation to within ±5% between 25 and 65 inches from the core center. It should be pointed out that an "effective" source as is defined here is not a unique method for representing the source. A selection from various types of surface sources with different spacial and angular distribution can be made but the effort required to determine whether a more satisfactory choice is possible is outside the scope of this study.

A total of 11,000 neutron histories were studied with the source neutrons distributed as follows:

a. Between 0° - 9° (measured from the reactor-tank axis), 1000 neutrons each at 4, 6, 7.25 and 8.5 Mev.

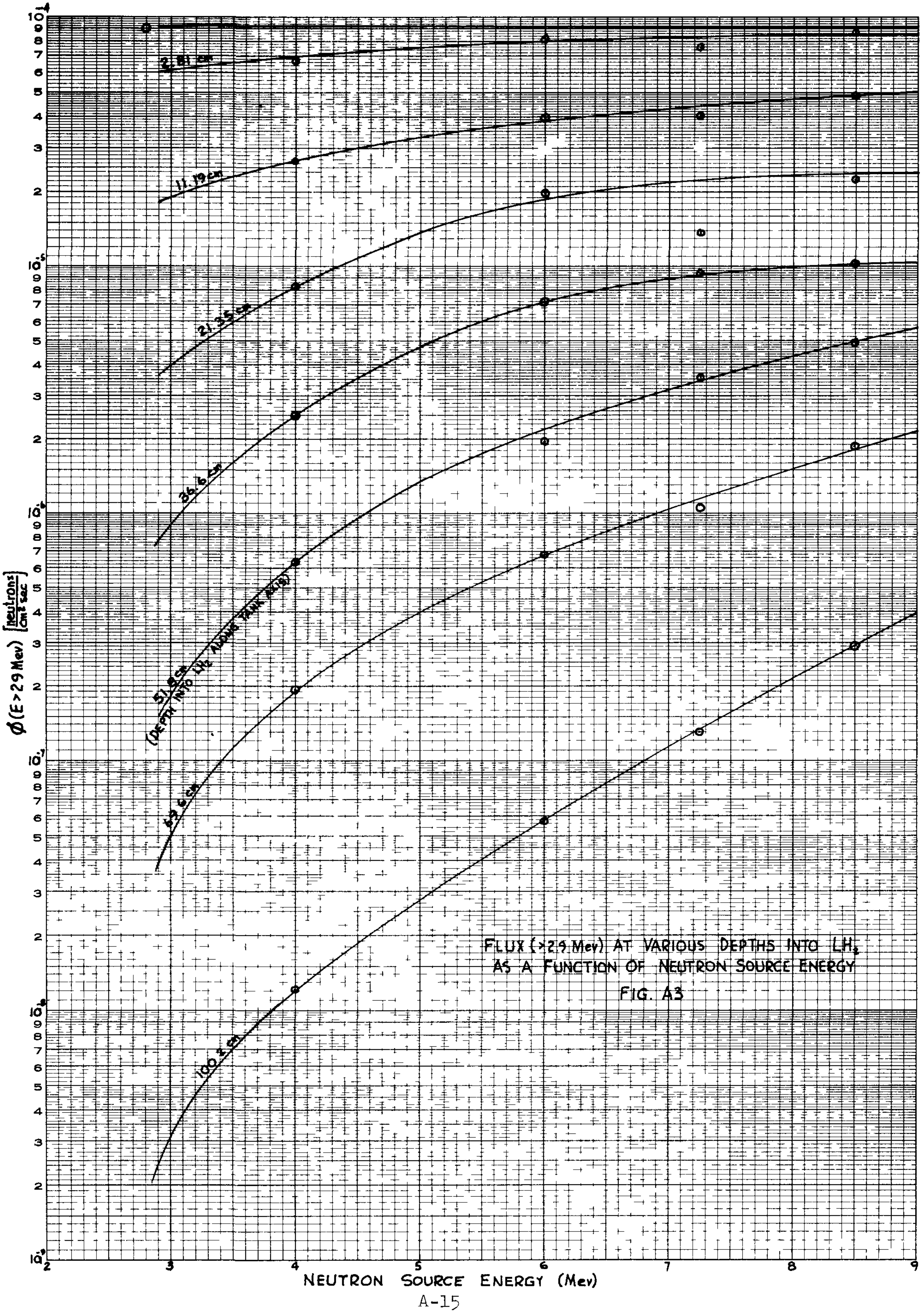
b. Between 9° - 19°, 1000 neutrons at same energies.

c. Between 19° - 26°, 800 neutrons at 4, 7.25 and 8.5 Mev, 600 neutrons at 6 Mev.

The maximum angle of 26° corresponds to the angle subtended by the tank used in the experiment.

The results of the calculations, normalized to a source strength of 2 neutrons/sec, are summarized in Figure A-8. Since the detector has a threshold of 2.9 Mev, the flux values at this energy are readily obtained by the exponential and inverse distance-square law. Except for the three top Monte Carlo values at 7.25 Mev, the results fall essentially on smooth curves.

FIGURE A 8



FLUX (>2.9 Mev) AT VARIOUS DEPTHS INTO LH_2
AS A FUNCTION OF NEUTRON SOURCE ENERGY
FIG. A3

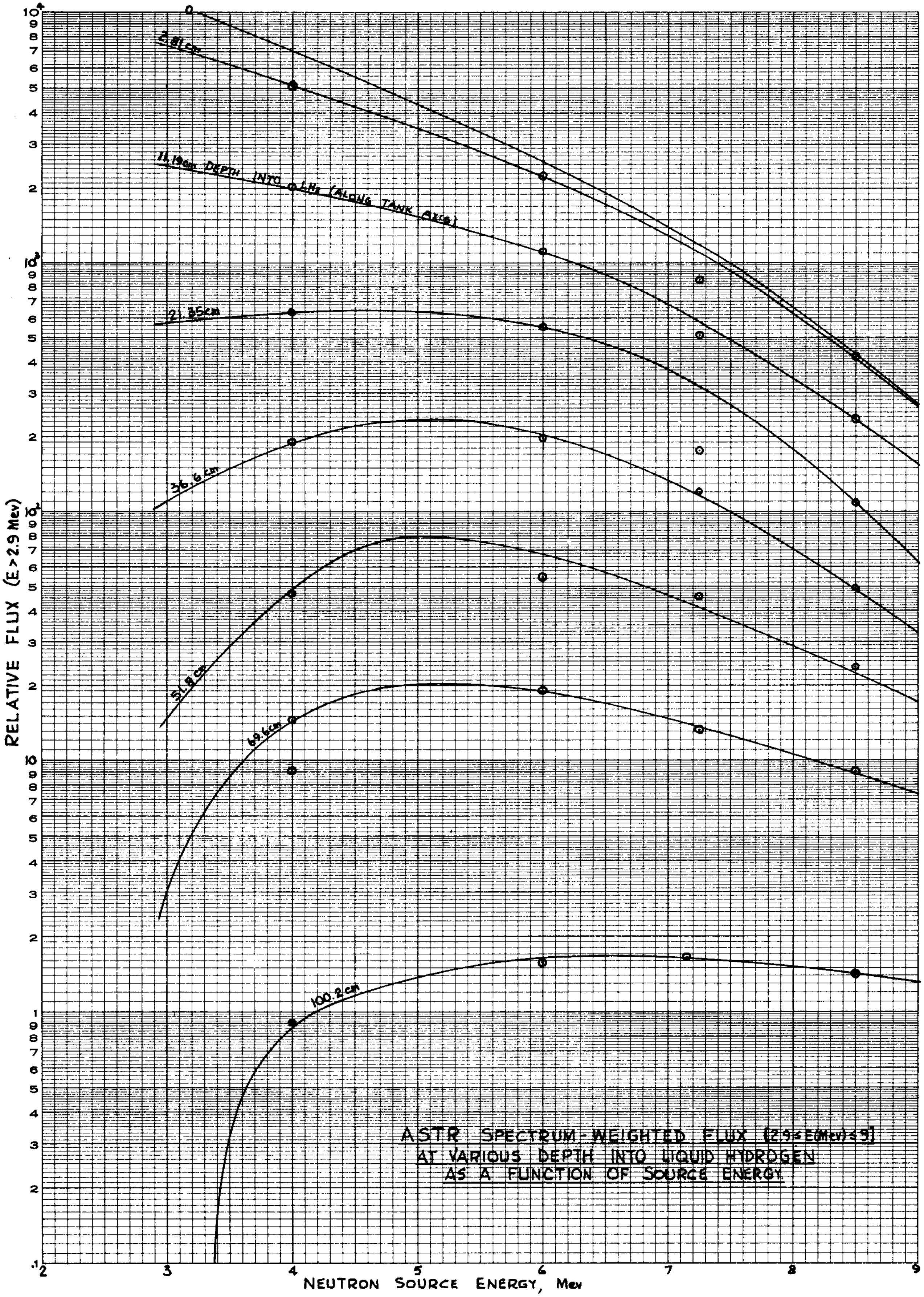
The same results are presented in Figure A-9 but now are weighted according to the ASTR leakage spectrum.[A5] It is noticed that although the high energy contribution is depressed somewhat there is still a significant contribution from source energies greater than 9 Mev for penetrations greater than around 60 cm of hydrogen. At 100.2 cm of hydrogen, it is estimated that the integral flux above 9 Mev is 40-50% of the flux from 2.9 to 9 Mev.

The flux of neutrons between 2.9 and 9 Mev has been obtained by numerically integrating curves of Figure A-9. The variation of this flux with distance in hydrogen is compared with that of experiment in Figure A-10. Again, it is pointed out that the intent here is to compare penetration within hydrogen, hence the normalization indicated in Figure A-10.

C. DISCUSSION OF RESULTS

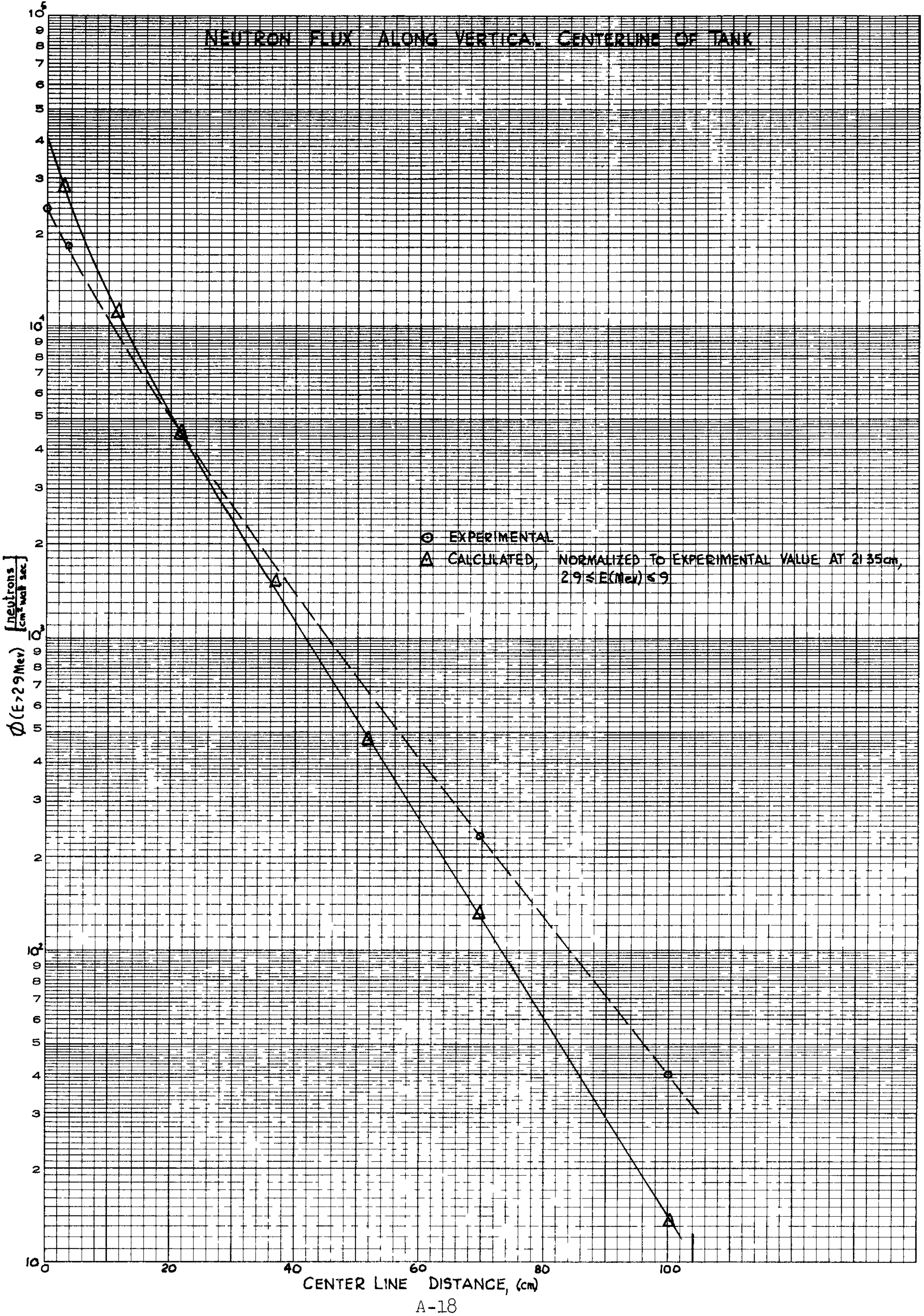
Comparison of the final calculated results with the experimental data raises a number of questions. The major sources of the discrepancies, however, are not obscure. The cause of the difference near the origin is mostly caused by the shadowing effect of piping below the tank bottom, which is not accounted in the calculation. If one assumes the shielding effect of the piping to be approximated by that of a small-diameter, stainless steel disc of 0.32 in. thickness and apply a correction based on this assumption, the difference near the origin would be only about 25% relative to the experimental point. One of the sources of discrepancy at the large thickness is evident from the results of Figure A-9; there is a sizeable contribution to flux due to neutrons above 9 Mev which are not included in these calculations. If one applies a factor of ~ 1.5 (an estimate of the correction for the contribution above 9 Mev) to the calculated value at 100 cm, there is improvement but still a factor of two discrepancies yet to be explained. The causes of the remaining difference are probably due to (a) neutrons streaming along the void spaces surrounding the hydrogen, and then being deflected laterally into the tank, thus enhancing the experimental flux at the detectors located deep within the hydrogen, and (b) the assumption of a point isotropic source in the calculation, when in reality the leakage neutrons are emitted with a preferential forward direction.

FIGURE A 9



ASTR SPECTRUM-WEIGHTED FLUX [2.9≤E(Mev)≤9]
AT VARIOUS DEPTH INTO LIQUID HYDROGEN
AS A FUNCTION OF SOURCE ENERGY

FIGURE A 10


NEUTRON FLUX ALONG VERTICAL CENTERLINE OF TANK
○ EXPERIMENTAL
△ CALCULATED, NORMALIZED TO EXPERIMENTAL VALUE AT 21.35cm,
2.9 ≤ E(Mev) ≤ 9
Ø (E>2.9 Mev) [neutrons / cm² watt sec]
10⁵
10⁴
10³
10²
10
9
8
7
6
5
4
3
2
0
20
40
60
80
100
CENTER LINE DISTANCE, (cm)

REFERENCES - APPENDIX

A-1 Nuclear Radiation Heating in Liquid Hydrogen, W. A. Hehs, et al., General Dynamics/Fort Worth, Report FZK-186-1, June 1964.

A-2 Analytical and Preliminary Design Studies of Nuclear Rocket Propulsion Systems, Aerojet Report No. 1999, June 1961

A-3 Nuclear Radiation Transfer and Heat Deposition, M. O. Burrell, NASA TND-1115, August 1962.

A-4 Neutron Cross Sections, D. J. Hughes and R. B. Schwartz, Brookhaven National Laboratory, Report BNL-325, July 1958.

A-5 Spatial and Energy Distribution of the Nuclear Field in the ASTR Radiation Effects Test Cell, General Dynamics/Fort Worth, FZK-169, August 1963.