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(Contract NAS-8-20414)

WANL-TME-1597

7 March 1967

Westinghouse Astronuclear Laboratory

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

**Midterm Presentation At Marshall Space Flight Center
February 28, 1967**

MASTER

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INFORMATION CATEGORY

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J. M. Kutzler
AUTHORIZED CLASSIFIER

3/13/67
DATE

Mr. H. Stern of Marshall Space Flight Center
is the technical coordinator of
this work, performed by Westinghouse Astronuclear
Laboratory under contract NAS-8-20414

ABSTRACT

The data in this report was presented at the Marshall Space Flight Center by Westinghouse Astronuclear Laboratory on 28 February 1967. This presentation is pertinent to Contract NAS-8-20414, "Synthesis of Computational Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems." A description of the analytical methods selected for application under this contract is given. These analytical methods include the TRANSPORT, POINT KERNEL, and MONTE CARLO methods. Comparisons of data calculated by each of the three methods with available experimental data from the NERVA Reactor program are presented.

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Section I
INTRODUCTION
presented by
Mr. H. C. Woodsum

TOPICS COVERED IN INTRODUCTION

WORK STATEMENT

BREAKDOWN OF WORK SCOPE

CRITERIA FOR METHOD SELECTION

LIST OF METHODS SELECTED

GENERAL DESCRIPTION OF METHODS

STATUS OF COMPLETION OF CONTRACT

TYPES OF PROBLEMS SOLVED

CONTRACT NAS-8-20414 WORK STATEMENT
FOR
"SYNTHESIS OF CALCULATIONAL METHODS FOR THE
DESIGN AND ANALYSIS OF RADIATION SHIELDS FOR
NUCLEAR ROCKET SYSTEMS"

"THE CONTRACTOR WILL PERFORM COMPARISONS AND CRITICAL EVALUATIONS OF CURRENT
AUTOMATIC MACHINE CALCULATION METHODS USED TO PREDICT RADIATION DOSE LEVELS
AROUND NUCLEAR REACTOR SHIELD CONFIGURATIONS. . . "

"ON THE BASIS OF THE RESULTS OF THIS COMPARISON AND EVALUATION STUDY, AND IN
CONSULTATION WITH MSFC, THE CONTRACTOR SHALL SELECT ONE OR TWO CALCULATION
METHODS BEST SUITED FOR REACTOR PROPULSION APPLICATION AND SHALL
MAKE THESE PROGRAMS OPERABLE ON THE MSFC DATA PROCESSING EQUIPMENT. "

**CONTRACT (NAS-8-20414) WORK STATEMENT
FOR
"SYNTHESIS OF CALCULATIONAL METHODS
FOR THE DESIGN AND ANALYSIS OF RADIATION
SHIELDS FOR NUCLEAR ROCKET SYSTEMS"**

UNDER THIS CONTRACT, WESTINGHOUSE AGREED TO DEVELOP TWO METHODS WHICH MSFC CAN USE IN DESIGNING AND EVALUATING NUCLEAR ROCKETS.

THE FIRST METHOD WILL PROVIDE AN EARLY DESIGN METHOD FOR PERFORMING INITIAL DESIGN CALCULATIONS.

THE SECOND METHOD WILL PROVIDE A FINAL DESIGN METHOD FOR PERFORMING MORE DETAILED AND MORE ACCURATE RADIATION ANALYSIS AND SHIELD DESIGN.

THIS CONTRACT WILL SUPPLY MSFC WITH PROGRAM DESCRIPTIONS AND DECKS WHICH ARE COMPLETELY OPERABLE ON THE MSFC COMPUTER SYSTEM.

BREAKDOWN OF WORK SCOPE

TASK I

**EVALUATE PREVIOUS WORK
SELECT GEOMETRICAL MODELS
REVIEW AVAILABLE METHODS
SELECT PROMISING METHODS
EVALUATE METHODS VS EXPERIMENT**

TASK II

**EVALUATE MOST PROMISING METHODS
SELECT EARLY DESIGN METHOD
SELECT FINAL DESIGN METHOD**

TASK III

**PROGRAM EARLY DESIGN METHOD
FORMULATE AND PROGRAM FINAL DESIGN METHOD
WRITE FINAL REPORT**

CRITERIA FOR EVALUATION OF METHODS

- 1. DETAIL OF INFORMATION OBTAINED.
CAN ANGULAR AND ENERGY DISTRIBUTIONS BE OBTAINED?**
- 2. FLEXIBILITY. WHAT GEOMETRIES CAN BE ACCOMMODATED?**
- 3. CONVENIENCE AND DIRECT USEFULNESS OF OUTPUT.**
- 4. ACCURACY.**
- 5. RUNNING TIME.**
- 6. ANY UNUSUAL FEATURES WHICH ARE ADVANTAGEOUS OR
DISADVANTAGEOUS IN REGARD TO NUCLEAR ROCKET APPLICATIONS.**

LIST OF CODES FOR EARLY
AND FINAL DESIGN METHODS

EARLY DESIGN METHOD

1. KAP V - POINT KERNEL PROGRAM
2. TAPAT SYSTEM - ONE DIMENSIONAL GAMMA AND NEUTRON TRANSPORT PROGRAMS
3. TIC-TOC-TOE- TANK HEATING CODE FOR ON-AXIS PROPELLANT TANK GEOMETRY

FINAL DESIGN METHOD

1. ODD-K-TWO-DIMENSIONAL GAMMA AND NEUTRON TRANSPORT PROGRAM (r, z GEOMETRY)
2. NAGS-ODD-K DATA PROCESSING CODE TO OBTAIN NEUTRON AND GAMMA SOURCES AND HEATING RATES
3. DAFT-ODD-K DATA PROCESSING CODE TO OBTAIN DIRECTIONAL ANGULAR LEAKAGE FLUXES
4. FASTER-MONTE CARLO PROGRAM

SUPPORTING DATA

1. LIBRARIES OF GAMMA AND NEUTRON CROSS SECTION DATA APPLICABLE TO TRANSPORT OR MONTE CARLO PROGRAM
2. POINT CODE

EARLY DESIGN METHOD DESCRIPTION

INPUT

GEOMETRICAL MODEL DATA
GAMMA AND NEUTRON (FISSION) SOURCE DISTRIBUTIONS
GAMMA SOURCE STRENGTHS

OUTPUT

GAMMA FLUXES, DOSES, OR HEATING RATES
FAST NEUTRON FLUX (MOMENTS DATA)*
FAST NEUTRON DOSE RATES (ALBERT WELTON OR MOMENTS DATA)*
FAST NEUTRON OR GAMMA RAY EXTERNAL HEATING RATES

*OPTIONS

EARLY DESIGN METHOD DESCRIPTION

INPUT

ATOM NUMBER DENSITIES
DIMENSIONS
CROSS SECTIONS
SOURCE (OR SOURCE GUESS)

OUTPUT

NEUTRON FLUXES
FISSION DISTRIBUTION
SECONDARY PHOTON SOURCES*
PHOTON FLUXES*
NEUTRON AND/OR PHOTON INTERNAL HEATING RATES*

* OPTIONS

EARLY DESIGN METHOD DESCRIPTION

INPUT

GEOMETRICAL MODEL
FLOW RATE
GAMMA AND NEUTRON SOURCES

OUTPUT

INTEGRAL TANK HEATING
HEATING RATE DISTRIBUTIONS*
TEMPERATURE INCREASE VS OPERATING TIME*

* OPTIONS

FINAL DESIGN METHOD DESCRIPTION

TWO DIMENSIONAL TRANSPORT

INPUT

NEUTRON CROSS SECTIONS
GEOMETRICAL DESCRIPTION
ATOM DENSITIES

ODD-K (NEUTRON)

OUTPUT

MULTIGROUP REACTOR AND SHIELD
SCALAR FLUXES
MULTIGROUP ANGULAR LEAKAGE FLUXES*
FISSION DISTRIBUTION

* OPTION

FINAL DESIGN METHOD DESCRIPTION

INPUT

MULTIGROUP CONVERGED REACTOR NEUTRON FLUXES FROM ODD-K EIGENVALUE PROBLEM
GEOMETRICAL DESCRIPTION
ATOM DENSITIES

NAGS
DATA PROCESS - 1

OUTPUT

MULTIGROUP DISTRIBUTED PHOTON SOURCES
NEUTRON KINETIC AND (N, α) HEATING
NEUTRON DOSE RATES *

* OPTION

FINAL DESIGN METHOD DESCRIPTION

TWO DIMENSIONAL TRANSPORT

INPUT

FIXED GAMMA SOURCES
PHOTON CROSS SECTIONS
GEOMETRICAL DESCRIPTION
ATOM DENSITIES

ODD-K
(PHOTON)

OUTPUT

MULTIGROUP PHOTON SCALAR FLUXES

MULTIGROUP ANGULAR LEAKAGE FLUXES*

* OPTION

FINAL DESIGN METHOD DESCRIPTION

INPUT

OUTPUT FROM NEUTRON AND PHOTON ODD-K PROBLEMS
GEOMETRICAL DESCRIPTION
ATOM DENSITIES

NAGS DATA PROCESS - 2

OUTPUT

TOTAL GAMMA AND NEUTRON HEATING RATES BY REGION
GAMMA AND NEUTRON HEATING RATE DISTRIBUTIONS
GAMMA AND NEUTRON HEATING RATES BY MATERIAL *

* OPTION

FINAL DESIGN METHOD DESCRIPTION

INPUT

GEOMETRICAL DESCRIPTION
SOURCE DISTRIBUTIONS
GAMMA OR NEUTRON MICROSCOPIC CROSS SECTIONS
POINT, SURFACE, AND/OR VOLUME DETECTOR SPECIFICATIONS
BIASING PARAMETERS

FASTER MONTE CARLO

OUTPUT

MULTIGROUP NUMBER AND ENERGY FLUXES AND VARIANCES
MULTIGROUP DIFFERENTIAL NUMBER AND ENERGY FLUXES
CUMULATIVE NUMBER AND ENERGY FLUXES
MULTIGROUP AND TOTAL RESPONSE WITH VARIANCE LIMITS*
MULTIGROUP FLUX AND TOTAL RESPONSES BY SOURCE REGION
MULTIGROUP FLUX AND TOTAL RESPONSES BY NUMBER OF COLLISIONS*
MULTIGROUP ANGULAR LEAKAGE FLUXES
MULTIGROUP AVERAGE PATH LENGTH MOMENTS*

*OPTION

STATUS OF COMPLETION
OF CONTRACT

TASK I

EVALUATE PREVIOUS WORK - COMPLETE
SELECT GEOMETRICAL MODELS - COMPLETE
REVIEW AVAILABLE METHODS - COMPLETE
SELECT PROMISING METHODS - COMPLETE
EVALUATE METHODS VS EXPERIMENT - COMPLETE

TASK II

EARLY AND FINAL DESIGN METHOD SELECTION - COMPLETE
EVALUATION OF THESE METHODS: COMPLETE FOR NRX CONFIGURATION
AND CONTINUING FOR FLIGHT CONFIGURATION

STATUS OF COMPLETION OF CONTRACT

TASK III

- PROGRAM EARLY DESIGN METHOD - KAP V PROGRAM CODING IS COMPLETE
CODE IS OPERATIONAL ON MSFC COMPUTER

STILL REQUIRES: 1. CONVERSION OF TAPAT TO FORTRAN IV
2. FINAL CHECKOUT OF REVISED TIC-TOC-TOE PROGRAM

- FORMULATE AND PROGRAM FINAL DESIGN METHOD

FASTER - MONTE CARLO CODE FORMULATION, CODING, AND CHECKOUT IS COMPLETE
OPERATIONAL ON MSFC COMPUTER

NAGS - CODE FORMULATION, CODING, AND CHECKOUT IS COMPLETE

ODD-K - CODE REQUIRES SOME EFFORT TO MAKE OPERATIONAL ON MSFC COMPUTER

TRANSPORT CODE AND MONTE CARLO CODE LIBRARIES MUST BE COMPLETED

DAFT - CODE CHECKOUT MUST BE COMPLETED

- WRITE REPORT ON EARLY DESIGN METHOD
MUST BE COMPLETED

- WRITE REPORT ON FINAL DESIGN METHOD
MUST BE COMPLETED

**TYPES OF PROBLEMS
SOLVED BY MSFC CODE PACKAGE**

EARLY DESIGN METHOD

**REACTOR INTERNAL RADIATION ENVIRONMENT
(1 d, RADIAL AND AXIAL)**

**REACTOR INTERNAL HEATING RATES
(1 d, RADIAL AND AXIAL)**

SHIELD PARAMETRIC STUDIES

**DIRECT COMPONENT EXTERNAL
ENVIRONMENT**

PROPELLANT TANK HEATING

FINAL DESIGN METHOD

**REACTOR INTERNAL RADIATION ENVIRONMENT
(2 d, R - Z CYLINDRICAL GEOMETRY)**

**REACTOR INTERNAL HEATING RATES
(2 d, R - Z CYLINDRICAL GEOMETRY)**

SHIELD DESIGN ANALYSIS

**DIRECT AND SCATTERED COMPONENT
EXTERNAL ENVIRONMENT**

PROPELLANT TANK HEATING

TABLE OF METHODS CAPABILITIES

EARLY DESIGN	SHIELD DESIGN	1 d ENVIRON- MENT INTERNAL	2 d ENVIRON- MENT		1 d HEAT- ING RATES INTERNAL	2 d HEAT- ING RATES		TANK HEAT	REACTOR SHIELD PARAMETRIC STUDIES
			INTERNAL	EXTERNAL		INTERNAL	EXTERNAL		
1. KAP 2. TAPAT 3. TIC-TOC-TOE	X	X		X	X		X	X X	X X
FINAL DESIGN									
ODD-K NAGS DAFT FASTER	X X		X X	X X X X		X X	X X X X	X X X X,X	

Section II

Geometrical Models and Experimental Data

presented by

Miss M. A. Capo

1. PHYSICAL DESCRIPTION OF THE REACTORS
 - A. NRX TYPE
 - B. FLIGHT TYPE
2. PROPELLANT TANK DESCRIPTION
3. EXPERIMENTAL FACILITIES
 - A. PAX
 - B. PTX-NCX
 - C. NRX
4. EXPERIMENTAL DATA

GEOMETRICAL MODELS

1. TYPICAL NRX REACTOR

PURPOSE: CHECK ANALYTICAL DATA WITH EXPERIMENTAL DATA

2. FLIGHT-TYPE REACTOR

PURPOSE:

- 1) PROVIDE FLIGHT-TYPE REACTOR APPLICABLE TO PROPELLANT HEATING ANALYSIS
- 2) PERMIT APPLICATION OF THE ANALYTICAL METHOD TO THE FLIGHT-TYPE SYSTEM

NRX TYPE REACTOR CONFIGURATION FOR RADIATION ANALYSIS

PROPELLANT TANK CONFIGURATION FOR RADIATION ANALYSIS

TYPICAL NRX REACTOR CROSS SECTIONAL VIEW SHOWING DOSIMETERS ON THE MERIDIAN RING AND PRESSURE VESSEL SURFACE

EXPERIMENTAL DATA

1. GAMMA RAY
 - A. INTERNAL PAX DATA
 - B. EXTERNAL PAX, NRX DATA
2. FAST NEUTRON ($E > 2.9 \text{ Mev}$)
 - A. INTERNAL PAX DATA
 - B. EXTERNAL PAX, NRX DATA
3. FISSION FOIL NEUTRON SPECTRA DATA
 - A. EXTERNAL NRX DATA
4. THERMAL FLUX DATA
 - A. EXTERNAL NRX DATA

EXPERIMENTAL DIFFERENTIAL NEUTRON FLUX AT A RADIUS
OF 5 FEET FROM THE CENTER OF THE REACTOR ON THE MIDPLANE

Section III

TRANSPORT METHOD OF ANALYSIS

presented by

Mr. R. K. Disney

TRANSPORT METHOD

IS AN INTEGRAL PART OF THE ENTIRE RADIATION
ANALYSIS METHODOLOGY IN THAT IT PROVIDES
DISTRIBUTED NEUTRON AND PHOTON SOURCES

- A. SPATIAL
- B. ENERGY
- C. SURFACE ANGULAR

FOR MONTE CARLO AND POINT KERNEL METHODS.

DESCRIPTION OF TRANSPORT METHOD

1. TIME INDEPENDENT SOLUTION OF THE BOLTZMANN TRANSPORT EQUATION
2. MULTIGROUP REPRESENTATION OF THE PARTICLE VELOCITIES
3. DISCRETE ORDINATE REPRESENTATION OF THE PARTICLE DIRECTIONS
4. FINITE DIFFERENCE EQUATION SOLUTION WHERE THE BASIC ASSUMPTION IS THAT FLUX IS LINEAR WITH RESPECT TO r , μ , ϕ IN EACH INTERVAL

PROGRAMS USED IN THE TRANSPORT METHOD

1. TAPAT - ONE-DIMENSION (SLAB, SPHERE, CYLINDER)
2. ØDDK - TWO-DIMENSION (CYLINDER: R-Z)

FLOW OF DATA AND INFORMATION THROUGH THE ONE- AND TWO-DIMENSION CALCULATION PROCEDURES

COMPARISON OF TRANSPORT RESULTS TO EXPERIMENTAL DATA

ONE DIMENSIONAL TRANSPORT METHODS

BALANCE EQUATION:

LEAKAGE

+ COLLISIONS

= SOURCE

SLAB GEOMETRY

$$\mu \frac{\partial N(r, \mu)}{\partial r} + \Sigma_T(r) N(r, \mu)$$

SPHERICAL GEOMETRY

$$\mu \frac{\partial N(r, \mu)}{\partial r} + \frac{1-\mu^2}{r} \frac{\partial N(r, \mu)}{\partial \mu} + \Sigma_T(r) N(r, \mu) \left\{ = q(r, \mu) + \int_0^{2\pi} \int_{-1}^1 \Sigma_S(r, \mu_0) N(r, \mu, \varphi) d\mu' d\varphi \right.$$

CYLINDRICAL GEOMETRY

$$\frac{1-\mu^2}{r} \left\{ \cos \varphi \frac{\partial [r N(r, \mu, \varphi)]}{\partial r} - \frac{\partial [\sin \varphi N(r, \mu, \varphi)]}{\partial \varphi} \right\} + \Sigma_T(r) N(r, \mu, \varphi)$$

FINITE DIFFERENCE METHODS IN TRANSPORT METHODS

ANGULAR FLUX:
$$N(r, \mu)_i = N(r_i, \mu_j) + \frac{\partial N(r_i, \mu_j)}{\partial r} (r_{i+1} - r_i) + \frac{\partial N(r_i, \mu_j)}{\partial \mu} (\mu - \mu_j) + \frac{\partial^2 N(r_i, \mu_j)}{\partial r \partial \mu} (r - r_i)(\mu - \mu_j)$$

where:

$$\frac{\partial N(r_i, \mu_j)}{\partial r} = \frac{N(r_{i+1}, \mu_j) - N(r_i, \mu_j)}{r_{i+1} - r_i}$$

$$\frac{\partial N(r_i, \mu_j)}{\partial \mu} = \frac{N(r_i, \mu_{j+1}) - N(r_i, \mu_j)}{\mu_{j+1} - \mu_j}$$

$$\frac{\partial^2 N(r_i, \mu_j)}{\partial r \partial \mu} = \frac{N(r_{i+1}, \mu_{j+1}) - N(r_{i+1}, \mu_j) - N(r_i, \mu_{j+1}) + N(r_i, \mu_j)}{(r_{i+1} - r_i)(\mu_{j+1} - \mu_j)}$$

DISCRETE ORDINATE REPRESENTATION IN TAPAT
(S_4 ANGULAR QUADRATURE)

SLAB AND SPHERE

CYLINDER

DISCRETE ORDINATE REPRESENTATION IN ODDK
(S_6 ANGULAR QUADRATURE)

TRANSPORT ANALYSIS

INPUT

1. REGION DATA
 - a. ELEMENT IDENTIFICATION NUMBERS
 - b. ELEMENT ATOM DENSITIES
2. MESH INTERVAL DATA
 - a. REGION MESH INTERVAL AND MATERIAL ASSIGNMENTS
 - b. MESH INTERVAL DIMENSIONS

TRANSPORT ANALYSIS

OUTPUT

1. NEUTRON DATA

- a. MULTIGROUP SCALAR FLUXES
- b. REACTION RATES (HEATING AND DOSE RATES)
- c. MULTIGROUP ANGULAR FLUXES AT SURFACE

2. PHOTON DATA

- a. SOURCES (REGION, POINT, AND SEPARABLE r AND z DISTRIBUTION)
- b. MULTIGROUP SCALAR FLUXES
- c. MULTIGROUP ANGULAR FLUXES AT SURFACE
- d. REACTION RATES (HEATING AND DOSE RATES)

TWO DIMENSIONAL TRANSPORT ANALYSIS

CAPABILITIES

TAPAT

1. 100 MESH COORDINATES
2. S_2 OR S_4 ANGULAR QUADRATURE
3. 20 GROUPS
4. P_0 WITH P_1 (IN-GROUP) LEGENDRE
POLYNOMIAL EXPANSION OF
SCATTERING INTEGRAL
5. 30 MATERIALS
6. 40 REGIONS

ONE DIMENSIONAL TRANSPORT SYSTEM

CAPABILITIES

ØDDK (LIMITS ARE TYPICAL SIZE PROBLEMS SINCE
ØDDK IS A VARIABLE DIMENSION PROGRAM)

NEUTRON: P_0 OR TRANSPORT CORRECTED CROSS SECTION INPUT

1. 1200 MESH CELLS
2. S_2, S_4, S_6 ANGULAR QUADRATURE
3. 16 GROUPS
4. 40 MATERIALS
5. 40 REGIONS

PHOTON: P_0 AND P_1 CROSS SECTION INPUT

1. 800 MESH CELLS
2. S_6 ANGULAR QUADRATURE
3. 13 GROUPS
4. 18 MATERIALS (36 CROSS SECTION SETS)
5. 18 REGIONS

EXPERIMENTAL DIFFERENTIAL NEUTRON FLUX AT A RADIUS
OF 5 FEET FROM THE CENTER OF THE REACTOR ON THE MIDPLANE

63

1. LIMITED TO REACTOR-SHIELD GEOMETRIES (I.E., R-Z MODELS)
2. TRANSPORT METHOD EFFICIENTLY PERFORMS A DETAILED DATA PROCESSING AND DATA REDUCTION FUNCTION FOR THE ENTIRE RADIATION ANALYSIS PROCEDURE
3. ACCURACY IS GOOD AND WITH PROPER CROSS SECTION DATA AND PROPER APPLICATION MODERATE SHIELD DEPTH PENETRATION CAN BE ACCOMMODATED
4. COMPUTER RUNNING TIME:
 - COMPLETE TWO-DIMENSION
TRANSPORT METHOD ANALYSIS - 2 HOURS
(IBM 7094 MOD II)
 - COMPLETE ONE-DIMENSION TRANSPORT
METHOD ANALYSIS IN RADIAL AND AXIAL
DIRECTIONS - 40 MINUTES
(IBM 7094 MOD II)
5. UNUSUAL FEATURES:
 - a. SUPPLIES DISTRIBUTED (ANGULAR AND ENERGY) DATA FOR POINT KERNEL AND MONTE CARLO
 - b. TREATS PHOTON AS WELL AS NEUTRON TRANSPORT IN A CONSISTENT MANNER

Section IV

POINT KERNEL ANALYSIS

presented by

Miss M. A. Capo

POINT KERNEL METHOD

1. DESCRIPTION OF KAP V CODE

- A. DESIRABLE FEATURES
- B. LIMITATIONS
- C. GEOMETRY
- D. SOURCE
- E. ATTENUATION KERNELS

2. COMPARISON WITH PAX AND NRX EXPERIMENTAL DATA

- A. GAMMA RAY
- B. FAST NEUTRON

3. FLIGHT-TYPE REACTOR AND PROPELLANT TANK GEOMETRY

4. PROPELLANT HEATING ANALYSIS

KAP V - POINT KERNEL CODE

DESIRABLE FEATURES

1. INPUT DATA FLEXIBILITY
2. MULTIPLE DETECTOR POINTS FOR A SINGLE SOURCE REGION
3. MULTIPLE SOURCE REGIONS
4. MULTIPLE GAMMA RAY AND/OR NEUTRON RESPONSE FUNCTIONS
5. BUILT-IN GAMMA RAY CROSS SECTION LIBRARY
6. BUILT-IN BUILDUP FACTOR LIBRARY
7. OPTION FOR FLUX WITHIN A SOURCE REGION
8. ELIMINATION OF + OR - AMBIGUITY INDEX ON BOUNDARY
9. OUTPUT DATA FLEXIBILITY
10. SOURCE DESCRIPTION FLEXIBILITY

KAP V - POINT KERNEL CODE

INPUT DATA LIMITATIONS

30 GAMMA RAY ENERGY GROUPS

30 FAST NEUTRON ENERGY VALUES

100 GEOMETRY ZONES

6 BOUNDARIES PER ZONE

20 MATERIALS AND/OR ELEMENTS

50 COMPOSITIONS (MIXTURES OF MATERIALS OR ELEMENTS)

25 DETECTOR POINTS

SOURCE REGIONS (NO LIMIT)

20 RADIAL SOURCE INTERVALS

20 AXIAL SOURCE INTERVALS

20 AZIMUTHAL SOURCE INTERVALS

20 POLAR SOURCE INTERVALS

10 GAMMA RAY AND/OR NEUTRON RESPONSE FUNCTIONS

KAP V - POINT KERNEL CODE

GEOMETRY SURFACE EQUATIONS

$$AX^2 + XX_{\circ} + BY^2 + YY_{\circ} + CZ^2 + ZZ_{\circ} = D$$

$$A(X - X_{\circ})^2 + B(Y - Y_{\circ})^2 + C(Z - Z_{\circ})^2 = D$$

$$(X - X_{\circ})^2 + (Y - Y_{\circ})^2 = D$$

$$X = D \qquad Y = D \qquad Z = D$$

KAP V - POINT KERNEL CODE

SOURCE GEOMETRY OPTIONS

1. CYLINDRICAL
2. SPHERICAL
3. POINT, LINE, OR DISC
4. VARIABLE AZIMUTHAL SPACING FOR EACH RADIUS

KAP V - POINT KERNEL CODE

SOURCE DISTRIBUTION FUNCTION OPTIONS

CYLINDRICAL - RADIAL

FLAT, COSINE, OR EXPONENTIAL

LINEAR OR EXPONENTIAL INTERPOLATION OF $f(R)$ VS R

PARABOLIC INTERPOLATION OF $F(R')$ VS R'

CYLINDRICAL - AXIAL

COSINE OR EXPONENTIAL

LINEAR OR EXPONENTIAL INTERPOLATION OF $f(Z)$ VS Z

PARABOLIC INTERPOLATION OF $F(Z')$ VS Z'

SPHERICAL - RADIAL

FLAT

LINEAR INTERPOLATION OF $f(R)$ VS R

SPHERICAL - POLAR

FLAT

CYLINDRICAL - AZIMUTHAL

FLAT (UNLESS INPUT AS DISCRETE
POINT SOURCES)

KAP V - POINT KERNEL CODE

ATTENUATION KERNEL OPTIONS

GAMMA RAY

1. UNCOLLIDED FLUX (NO BUILDUP FACTOR)
2. COLLIDED FLUX (WITH BUILDUP FACTOR)

NEUTRON

1. MONOARIANT FIT TO INFINITE MEDIA MOMENTS DATA
2. BIVARIANT FIT TO INFINITE MEDIA MOMENTS DATA
3. MODIFIED ALBERT-WELTON

KAP V POINT KERNEL CODE

GAMMA RAY KERNELS

UNCOLLIDED FLUX

$$\psi(\vec{p}_m, E_g) = \exp[-b_T(E_g)]$$

$$b_T(E_g) = \sum_{m=1}^M \mu_m(E_g) \vec{p}_m$$

COLLIDED FLUX

$$\psi'(\vec{p}_m, E_g) = B(\vec{p}_m, E_g) \exp[-b_T(E_g)]$$

$$B(\vec{p}_m, E_g) = \sum_{i=0}^3 \beta_i(E_g) (b_T(E_g))^i$$

$$\text{IF, } b_T(E_g) > 20.0, \quad B(\vec{p}_m, E_g) = \sum_{i=0}^3 \beta_i(E_g) (20.0)^i$$

KAP V POINT KERNEL CODE

BIVARIANT NEUTRON SPECTRA KERNEL

$$\psi(\vec{\rho}_m, E_n) = \exp[f(W_R, E_n)]$$

$$f(W_R, E_n) = \sum_{i=0}^I \sum_{j=0}^J \Delta_{ij} E_n^j W_R^i$$

$$W_R = \left[\sum_{m=1}^M \Sigma_m \vec{\rho}_m \right] / \Sigma_R$$

IF, $W_R > 120.0 \text{ gm/cm}^2$ then;

$$N_R, E_n) = \left[\sum_{i=0}^I \sum_{j=0}^J \Delta_{ij} (E_n)^j (120.0)^i \right] + \left[-\lambda(E_n)(W_R - 120.0) \right]$$

EXPERIMENTAL DIFFERENTIAL NEUTRON FLUX AT A RADIUS
OF 5 FEET FROM THE CENTER OF THE REACTOR ON THE MIDPLANE

SUMMARY - POINT KERNEL METHOD

1. THE KAP V CODE PROVIDES A QUICK AND REASONABLY ACCURATE MEANS OF DETERMINING EXTERNAL RADIATION ENVIRONMENT
2. ACCURACY OF EXTERNAL ENVIRONMENT
 - A. PREDICTS GAMMA RAY EXTERNAL ENVIRONMENT WITHIN A FACTOR OF TWO
 - B. PREDICTS EXTERNAL FAST NEUTRON FLUX ($E > 2.9 \text{ Mev}$) GENERALLY BETTER THAN A FACTOR OF TWO IN AREAS WHERE CARBON MOMENTS DATA ARE APPLICABLE
3. PROPELLANT HEATING
 - A. EXCELLENT AGREEMENT WITH MONTE CARLO FOR GAMMA RAY HEATING ON CENTERLINE
 - B. ADDITIONAL ANALYSIS IS REQUIRED TO DETERMINE WHICH MOMENTS DATA SHOULD BE EMPLOYED FOR NEUTRON KINETIC HEATING

Section V

MONTE CARLO ANALYSIS

presented by

Mr. T. M. Jordan

THE MONTE CARLO FINAL DESIGN METHOD

- THE MONTE CARLO METHOD
- THE FASTER CODE
- COMPARISONS WITH EXPERIMENTAL DATA

THE BOLTZMANN TRANSPORT EQUATION

THE MONTE CARLO METHOD

THE FASTER PROGRAM

(A FORTRAN ANALYTIC SOLUTION OF THE TRANSPORT EQUATION BY RANDOM SAMPLING)

WRITTEN IN FORTRAN FOR THE IBM 7094 COMPUTER

QUADRIC SURFACE GEOMETRIES

MULTIPLE POINT, LINE, SURFACE, AND/OR VOLUME SOURCES

NEUTRONS OR GAMMA RAYS

POINT, SURFACE, AND/OR VOLUME DETECTORS

BIASED RANDOM SAMPLING

SIMULTANEOUS TREATMENT OF ALL PARTICLE ENERGIES

COMPUTER ORIENTED FEATURES

COMPLETELY WRITTEN IN FORTRAN IV

USES OVERLAY FEATURE

VARIABLE INPUT AND OUTPUT UNIT DESIGNATIONS

USES NO AUXILIARY TAPES (ALL INTERNAL)

NO FIXED LIMITS (VARIABLE DIMENSIONS)

USES STANDARD MATHEMATICS LIBRARY

INDIVIDUAL DATA CARDS LISTED AS INPUT

COMPATIBLE WITH CDC-6600

CALCULATED QUANTITIES

FLUX GROUPS MAY BE COLLAPSED FROM CROSS SECTION GROUPS

MULTIPLE RESPONSE FUNCTIONS

POINT, SURFACE, AND/OR VOLUME DETECTORS

ANALYTIC ESTIMATION OF FLUXES

NUMBER AND ENERGY FLUXES: GROUPWISE, DIFFERENTIAL, AND CUMULATIVE

COEFFICIENT OF VARIATION AND AVERAGE ENERGY BY GROUP

RESPONSE FUNCTIONS BY GROUP AND TOTALS WITH LIMITS ON VARIATION COEFFICIENTS

GROUPWISE FLUXES AND TOTAL RESPONSES BY SOURCE

GROUPWISE FLUXES AND TOTAL RESPONSES BY NUMBER OF COLLISIONS

GROUPWISE ANGULAR MOMENTS AND TOTAL RESPONSES

GROUPWISE LENGTH-OF-FLIGHT MOMENTS AND TOTAL RESPONSES

GEOMETRIC FEATURES

SEPARATE DESCRIPTION OF SURFACES BOUNDING REGIONS

GENERAL SURFACE EQUATION; $a_0 + a_1x + a_2y + a_3z + a_4x^2 + a_5y^2 + a_6z^2 + a_7xy + a_8yz + a_9zx = 0$

INTERNAL EXPANSION OF MORE COMMON FORMS OF PLANES, CONES, ELLIPTICAL CYLINDERS AND ELLIPSOIDS

REGION DESCRIPTION BY LISTING BOUNDING SURFACES

INTERNAL CALCULATION OF "AMBIGUITY INDICES" USING "POINT-IN REGION"

INTERNAL CONSISTENCY CHECK USING "POINT-IN REGION"

RELATION OF ALL CALCULATIONS TO SURFACES RATHER THAN REGIONS

INTERNAL CALCULATION OF "MOST PROBABLE NEXT REGIONS"

INTERNAL CALCULATION OF "EXTERIOR BOUNDARIES"

CALCULATES REGION OCCUPIED BY SPECIFIED POINT

CALCULATES ALL PATH LENGTH ELEMENTS ALONG RAY AT ONE PASS

DOES NOT USE (ϵ , δ) BOUNDARY CROSSING SEARCH

CALCULATES NORMAL DERIVATIVES AT SPECIFIED BOUNDARY CROSSINGS

FINAL PRINT INCLUDES BOUNDARY SEARCH PARAMETERS

THE FIXED SOURCE

MULTIPLE SOURCES

SEPARABLE SPATIAL, ANGULAR, AND ENERGY DISTRIBUTIONS

RECTANGULAR, CYLINDRICAL, AND/OR SPHERICAL GEOMETRIES

EACH OF THE 3 SPATIAL VARIABLES MAY BE CONTINUOUS OR DISCRETE

THE COORDINATE SYSTEM FOR ANGULAR DISTRIBUTIONS ROTATES WITH THE RADIUS VECTOR

EACH OF THE 2 ANGULAR VARIABLES MAY BE CONTINUOUS OR DISCRETE

CONTINUOUS SPATIAL AND ANGULAR DISTRIBUTIONS SPECIFIED BY TABULATING v , $f(v)$

INTERNAL NORMALIZATION OF ALL DISTRIBUTIONS

SOURCE SPECTRUM NORMALIZED TO TOTAL ENERGY OR PARTICLES

GROUP-WISE SOURCE OBTAINED BY INTEGRATING INPUT SPECTRUM

INPUT SPECTRUM DEFINED BY TABULATING: DIFFERENTIAL NUMBER SPECTRUM

DIFFERENTIAL ENERGY SPECTRUM

GROUP INTEGRATED PARTICLES

GROUP INTEGRATED ENERGY

CROSS SECTIONS

GAMMA RAY SCATTERING USING INTERNAL CODING OF KLEIN-NISHINA FORMULA

INTERNAL CALCULATION OF ENERGY ABSORPTION COEFFICIENTS

NEUTRONS USING MULTIGROUP CONSTANTS

INTERNAL CALCULATION OF KINETIC HEATING RESPONSES

EXACT TREATMENT OF ANGULAR DEPENDENCE OF SCATTERING FROM HYDROGEN

HYDROGEN DENSITIES BY REGION

BIASING

BIASED SELECTION OF INITIAL POSITION VECTOR

SELECT SOURCE VOLUME AND THEN VARIABLES

USE SPHERICAL PSEUDO-SOURCE SUPERIMPOSED OVER REAL SOURCES

BIASED SELECTION OF INITIAL DIRECTION VECTOR

SELECT ANGLES IN COORDINATE SYSTEM ORIENTED ALONG RADIUS VECTOR

SELECT ANGLES IN COORDINATE SYSTEM ORIENTED TOWARD PREFERRED POINT

BIASED SELECTION OF SCATTERING POINT

NON-ESCAPE WEIGHTING IMPLICIT

BIASED SELECTION OF SCATTERED DIRECTION

TOWARDS PREFERRED POINT AND IN ORIGINAL DIRECTION

NON-CAPTURE PROBABILITY WEIGHTING IMPLICIT

TYPICAL BIASING DATA

- GROUP IMPORTANCE
- LINEAR BUILDUP COEFFICIENTS
- RATIOS OF FORWARD TO BACKWARD SCATTERING PROBABILITIES
- SCALING PARAMETERS FOR BUILT-IN BIASING FUNCTIONS

OPTIMUM FIXED SOURCE SAMPLING

$$p_0^*(\vec{r}) dV = u^*(\vec{\Omega}) v^*(s) s^2 ds d\Omega$$

$$v^*(s) = C S_V^*(\vec{r}_d + s\vec{\Omega}) \exp\left[-\int_0^s \Sigma^*(\vec{r}_d + s'\vec{\Omega}) ds'\right] / s^2$$

OPTIMUM SCATTERED SOURCE SAMPLING

$$P_1 = \int_0^{h/2} \Sigma^*(\vec{r}_k + x\vec{\Omega}_d) dx$$

$$P_2 = \int_0^{h/2} \Sigma^*(\vec{r}_d - x\vec{\Omega}_d) dx$$

$$P_T = P_1 + P_2$$

$$\eta_1 = \int_0^s \Sigma^*(\vec{r}_k + s'\vec{\Omega}) ds'$$

$$\eta_2 = \int_0^t \Sigma^*(\vec{r}_d + t'\vec{\Omega}') dt'$$

$$\frac{P_1}{P_T} \geq \frac{1}{3}, \quad P_1^*(\vec{r}) dV = \frac{C_n P_1}{P_T} q_1^*(\vec{\Omega}) \frac{\exp[-\eta_1 - \eta_2]}{s^2 t^2} s^2 ds d\Omega$$

$$\mu = \vec{\Omega} \cdot \vec{\Omega}_d, \text{ IF } \mu > 0, s \leq h/2\mu \text{ AND } t \geq h/2$$

$$\frac{P_1}{P_T} < \frac{1}{3}, \quad P_2^*(\vec{r}) dV = \frac{C'_n P_2}{P_T} q_2^*(\vec{\Omega}') \frac{\exp[-\eta_1 - \eta_2]}{s^2 t^2} t^2 dt d\Omega'$$

$$\mu' = -\vec{\Omega}' \cdot \vec{\Omega}_d, \text{ IF } \mu' > 0, t \leq h/2\mu' \text{ AND } s \geq h/2$$

RELATIVE RADIAL DISTANCE

EXPERIMENTAL DIFFERENTIAL NEUTRON FLUX AT A RADIUS
OF 5 FEET FROM THE CENTER OF THE REACTOR ON THE MIDPLANE

CONVERGENCE HISTORY - PHOTON TRANSPORT
(MIDPLANE DETECTORS)

NUMBER OF ITERATIONS

<u>Point</u>	<u>100</u>	<u>200</u>	<u>400</u>	<u>% CHANGE 100 TO 400</u>	<u>% CHANGE 200 TO 400</u>
1	3.30 (+1)	3.07 (+1)	3.57 (+1)	+ 7.0	+14.0
2	2.86 (+1)	3.40 (+1)	2.89 (+1)	+ 1.0	- 1.8
3	2.01 (+1)	1.65 (+1)	1.54 (+1)	-30.1	- 7.1
4	1.12 (+1)	1.13 (+1)	1.26 (+1)	+11.1	+ 9.5
5	6.88 (+0)	7.80 (+0)	7.51 (+0)	+ 8.4	- 3.9
6	3.78 (+0)	4.74 (+0)	4.67 (+0)	+18.9	- 1.5
7	7.77 (-1)	8.58 (-1)	1.16 (+0)	+33.6	+25.8
8	8.12 (-1)	6.95 (-1)	6.44 (-1)	-26.2	- 7.9
9	3.29 (-1)	3.16 (-1)	3.28 (-1)	- 0.3	+ 3.7
10	1.84 (-1)	1.93 (-1)	2.25 (-1)	+18.2	+14.2
11	5.03 (-2)	4.42 (-2)	5.31 (-2)	+ 5.3	+16.7
12	6.74 (-3)	7.24 (-3)	7.16 (-3)	+ 5.9	- 1.1

TYPICAL FASTER PROBLEMS				
REACTOR	NRX	NRX	FLIGHT	FLIGHT
PARTICLES	NEUTRON	PHOTON	NEUTRON	PHOTON
GROUPS	13	23	13	23
MATERIALS	20	20	24	24
SOURCES	1	24	5	14
SURFACES	49	49	37	37
REGIONS	54	54	44	44
DATA LOCATIONS	5500	6000	6700	6200
RUNNING TIME FOR POINT DETECTOR (100 HISTORIES)	90 s	48 s	60 s	45 s

SUMMARY

- SIMPLIFIED INPUT
- FLEXIBLE GEOMETRY
- DETAILED CALCULATION OF FLUXES
- CONVENIENT AND USEFUL OUTPUT
- SIMULTANEOUS TREATMENT OF PARTICLES
- COMPLETE USE OF BIASING
- COMPATIBLE WITH VARIOUS COMPUTERS

Section VI

SUMMARY

presented by

Mr. H. C. Woodsum

SUMMARY

GAMMA RAY ENVIRONMENT

1. POINT KERNEL IS IN CLOSER AGREEMENT WITH EXPERIMENTAL DATA THAN MONTE CARLO OR 2d TRANSPORT.

USE OF INFINITE MEDIA BUILDUP FACTORS IN THE GAMMA RAY KERNEL IMPLIES THAT THE POINT KERNEL WOULD BE HIGHER THAN THE MONTE CARLO OR TRANSPORT DATA.

2. MONTE CARLO AND 2d TRANSPORT ARE GENERALLY LOWER THAN THE EXPERIMENTAL DATA.
 - A. INTERNAL ENVIRONMENTS ARE WITHIN 10% OF THE EXPERIMENTAL DATA.
 - B. EXTERNAL ENVIRONMENTS FROM BOTH METHODS ARE WITHIN A FACTOR OF 1.5.

SUMMARY

FAST NEUTRON ENVIRONMENT ($E > 2.9\text{MEV}$)

1. POINT KERNEL IS IN CLOSER AGREEMENT WITH THE EXTERNAL ENVIRONMENTS DATA
USE OF CARBON INFINITE MEDIA MOMENTS
DATA IN THE NEUTRON SPECTRA DATA IMPLIES
THAT THE POINT KERNEL WOULD BE HIGHER
THAN THE MONTE CARLO OR TRANSPORT DATA
2. BOTH THE MONTE CARLO AND TRANSPORT DATA ARE GENERALLY LOWER THAN THE
EXPERIMENTAL DATA BY AS MUCH AS A FACTOR OF 2
THIS MAY BE DUE TO THE MULTIGROUP CROSS
SECTION INPUT DATA OR THE USE OF THE
TRANSPORT-CORRECTED P_0 APPROXIMATION
USED IN BOTH CODES.

SUMMARY OF FISSION FOIL
DATA COMPARISON

METHOD	NEUTRONS/CM ² - SEC - WATT		
	E > 2.9 Mev	E > 1.5 Mev	E > 0.6 Mev
NRX-A2 MEASUREMENT	2.1 (3)	6.9 (3)	1.4 (4)
NRX-A3 MEASUREMENT	2.2 (3)	7.2 (3)	9.7 (3)
POINT KERNEL	2.0 (3)	8.3 (3)	1.9 (4)
FASTER MONTE CARLO	1.3 (3)	6.9 (3)	9.7 (3)
ODD-K TRANSPORT	1.1 (3)	6.3 (3)	1.2 (4)

LOCATION: 5 FEET FROM THE CORE CENTER ON THE MIDPLANE

SUMMARY

THERMAL NEUTRON ENVIRONMENT ($E < 0.4 \text{ eV}$)

THE 2 d TRANSPORT DATA ARE IN VERY GOOD AGREEMENT WITH THE MEASURED DATA ON THE MERIDIAN RING.

SUMMARY
APPLICATION OF METHODS

OUTPUT DATA	METHOD		
	POINT KERNEL	TRANSPORT	MONTE CARLO
NEUTRON FISSION DISTRIBUTION	NO	YES	NO
INTERNAL PHOTON SOURCES	NO	YES	NO
INTERNAL ENVIRONMENT	YES	YES	YES
SURFACE ENERGY AND ANGULAR DISTRIBUTIONS	NO	YES	YES
EXTERNAL ENVIRONMENT			
a. GAMMA RAY	YES	NO	YES
b. FAST NEUTRON	YES	NO	YES
c. THERMAL NEUTRON	NO	NO	YES
PROPELLANT HEATING	YES	NO	YES
PAYLOAD ENVIRONMENT	YES	NO	YES