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
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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
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 Calculational 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 1

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."
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

KAPV
POINT KERNEL

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)

TAPAT
ONE DIMENSIONAL
TRANSPORT

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

TIC-TOC-TOE

TANK HEATING

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
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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

CENTER OF ACTIVE CORE

R = 16.5'

53'

16.5'

11'
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 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 GAMMA RAY DOSE RATE
ON THE REACTOR MIDPLANE (RADIAL TRAVERSE)

O PAX TLD DATA

RELATIVE RADIAL DISTANCE
EXPERIMENTAL GAMMA RAY DOSE RATE ON THE SURFACE OF THE PRESSURE VESSEL (AXIAL TRAVERSE)
EXPERIMENTAL GAMMA RAY DOSE RATE ON THE MERIDIAN RING

GAMMA RAY DOSE RATE (R/HR - WATT)

30 50 70 90 110 130 150 170

REACTOR POLAR ANGLE, $\theta$, DEGREES

PAX TLD DATA
NRX-A2 LOW POWER GLASS DATA
NRX-A2 LOW POWER GLASS DATA
NRX-A2 LOW POWER TLD DATA
NRX-A3 LOW POWER TLD DATA
NRX-A3 LOW POWER GLASS DATA
NRX-A3 LOW POWER COBALT PLATE DATA
EXPERIMENTAL GAMMA RAY DOSE RATE ON THE CORE MIDPLANE VERSUS RADIAL DISTANCE

- NRX-A2 LOW POWER GLASS DATA
- NRX-A3 LOW POWER TLD DATA
- NRX-A3 LOW POWER GLASS DATA
- NRX-A3 HIGH POWER COBALT PLATE DATA
- NRX-A3 LOW POWER COBALT PLATE DATA
EXPERIMENTAL FAST NEUTRON FLUX \( (E > 2.9 \text{ MEV}) \) ON THE REACTOR MIDPLANE (RADIAL TRAVERSE)

RELATIVE RADIAL DISTANCE

○ PAX SULFUR DATA
• NRX-A3 HIGH POWER NICKEL DATA
EXPERIMENTAL FAST NEUTRON FLUX (E>2.9 MEV) ON THE SURFACE OF THE PRESSURE VESSEL (AXIAL TRAVERSE)
EXPERIMENTAL FAST NEUTRON FLUX ON THE MERIDIAN RING (E>2.9 MEV)

- NRX-A2 LOW POWER SULFUR DATA
- NRX-A2 HIGH POWER SULFUR DATA
- NRX-A2 HIGH POWER NICKEL DATA
- NRX-A3 LOW POWER SULFUR DATA
- NRX-A3 LOW POWER NICKEL DATA
- NRX-A3 HIGH POWER NICKEL DATA

NEUTRON FLUX

10^2 - 10^3

REACTOR POLAR ANGLE, \( \theta \), DEGREES
EXPERIMENTAL FAST NEUTRON FLUX (E > 2.9 MEV) ON THE CORE MIDPLANE VERSUS RADIAL DISTANCE

DISTANCE FROM CENTER OF REACTOR (FEET)

- ○ PAX SULFUR DATA
- △ NRX-A2 HIGH POWER NICKEL DATA
- ▽ NRX-A2 HIGH POWER SULFUR DATA
- ◊ NRX-A3 HIGH POWER SULFUR DATA
- ▲ NRX-A3 HIGH POWER NICKEL DATA
- ■ NRX-A3 LOW POWER SULFUR DATA
- ○ NRX-A3 LOW POWER NICKEL DATA

NEUTRON FLUX (N/CM²-SEC-WATT)

- 10^0
- 10^1
- 10^2
- 10^3
- 10^4
- 10^5
EXPERIMENTAL DIFFERENTIAL NEUTRON FLUX AT A RADIUS OF 5 FEET FROM THE CENTER OF THE REACTOR ON THE MIDPLANE

NRX-A2 FISSION FOIL MEASUREMENTS
NRX-A3 FISSION FOIL MEASUREMENTS

NEUTRON ENERGY (Mev) vs. NEUTRONS/CM²-SEC-MEV-WATT

10⁵

10⁴

10³

10²
EXPERIMENTAL THERMAL NEUTRON FLUX (E<0.4EV) ON THE MERIDIAN RING

NEUTRON FLUX (N/CM²-SEC-WATT)

REACTOR POLAR ANGLE, , DEGREES

○ NRX-A2 HIGH POWER DATA (BARE MINUS Cd COVERED COBALT)
○ NRX-A3 HIGH POWER DATA (BARE MINUS Cd COVERED COBALT)
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 $r$, $\mu$, $\phi$ 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**
  
  \[ \mu \frac{\partial N(r, \mu)}{\partial r} \]

- **Slab Geometry**

  \[ \mu \frac{\partial N(r, \mu)}{\partial r} + \frac{1 - \mu^2}{r} \frac{\partial N(r, \mu)}{\partial \mu} \]

- **Spherical Geometry**

  \[ \mu \frac{\partial N(r, \mu)}{\partial r} + \frac{1 - \mu^2}{r} \frac{\partial N(r, \mu)}{\partial \mu} + \Sigma_T(r) N(r, \mu) \]

- **Cylindrical Geometry**

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

- \[ = \text{Source} \]

  \[ q(r, \mu) + \int_{0}^{2\pi} \int_{-1}^{1} \Sigma_S(r, \mu, \varphi) N(r, \mu, \varphi) \, d\mu' \, d\varphi \]
ANGULAR FLUX: \( N(r, \mu)_i = N(r, \mu) + \frac{\partial N(r, \mu)}{\partial r} (r_{i+1} - r_i) + \frac{\partial N(r, \mu)}{\partial \mu} (\mu - \mu^*_i) \nn + \frac{\partial^2 N(r, \mu)}{\partial r \partial \mu} (r - r_i) (\mu - \mu^*_i) \)

where:

\[
\frac{\partial N(r, \mu)}{\partial r} = \frac{N(r_{i+1}, \mu^*_i) - N(r_i, \mu^*_i)}{r_{i+1} - r_i}
\]

\[
\frac{\partial N(r, \mu)}{\partial \mu} = \frac{N(r_i, \mu_{i+1}) - N(r_i, \mu_i)}{\mu_{i+1} - \mu_i}
\]

\[
\frac{\partial^2 N(r, \mu)}{\partial r \partial \mu} = \frac{N(r_{i+1}, \mu_{i+1}) - N(r_i, \mu_{i+1}) + N(r_i, \mu_i)}{(r_{i+1} - r_i) (\mu_{i+1} - \mu_i)}
\]
DISCRETE ORDINATE REPRESENTATION IN TAPAT
($S_4$ ANGULAR QUADRATURE)

SLAB AND SPHERE

CYLINDER
DISCRETE ORDINATE REPRESENTATION IN $\phi$DDK
($S_6$ 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

POINT
reaction and transport
cross section library
and physical constants

neutron transport
cross sections
(macroscopic)

neutron reaction
cross sections
and photon source
(microscopic)

photon transport
cross sections
(macroscopic)

ODD-K
Neutron Transport

NAGS IV
Photon Source

ODD-K
Photon Transport

NAGS IV
neutron and photon
heating rates and
dose rates

DAFT
angular multigroup
fluxes for the
FASTER code

\[ \Phi^N(\mu, r \text{ or } z) \]

\[ \Phi^N(r, z) \]

\[ \Phi^N(r^1, z^1) \]

\[ \Phi^K(\mu, r^1 \text{ or } z^1) \]

\[ \Phi^K(r^1, z^1) \]
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

POINT
reaction and transport cross section library and physical constants

- neutron reaction cross sections and photon source (microscopic)
- neutron transport cross sections (macroscopic)
- photon transport cross sections (macroscopic)

Neutron Transport cylinder, slab, or sphere

- FLUX EDIT photon source
  neutron heating rates, dose rate, or fluxes

Photon Transport cylinder, slab, or sphere

- FLUX EDIT photon heating rates, or dose rates
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 GAMMA RAY DOSE RATE ON THE REACTOR MIDPLANE (RADIAL TRAVERSE)

FROM ODD-K 2D TRANSPORT

O PAX TLD DATA

RELATIVE RADIAL DISTANCE
EXPERIMENTAL GAMMA RAY DOSE RATE ON THE SURFACE OF THE PRESSURE VESSEL (AXIAL TRAVERSE)

- ODD-K 2D TRANSPORT

- NRX-A2 LOW POWER GLASS DATA
- NRX-A3 LOW POWER GLASS DATA
- NRX-A3 LOW POWER COBALT PLATE DATA
- PAX TLD DATA

RELATIVE AXIAL DISTANCE
EXPERIMENTAL GAMMA RAY DOSE RATE ON THE MERIDIAN RING

FROM ODD-K 2D TRANSPORT

- PAX TLD DATA
- NRX-A2 LOW POWER GLASS DATA
- NRX-A2 LOW POWER TLD DATA
- NRX-A3 LOW POWER TLD DATA
- NRX-A3 LOW POWER GLASS DATA
- NRX-A3 LOW POWER COBALT PLATE DATA

GAMMA RAY DOSE RATE (R/HR - WATT)

REACTOR POLAR ANGLE, $\alpha$, DEGREES
EXPERIMENTAL GAMMA RAY DOSE RATE ON THE CORE MIDPLANE VERSUS RADIAL DISTANCE

FROM ODD-K 2D TRANSPORT

GAMMA RAY DOSE RATE (R/HR - WATTS)

RADIAL DISTANCE FROM REACTOR AXIS (FEET)

NRX-A2 LOW POWER GLASS DATA
NRX-A3 LOW POWER TLD DATA
NRX-A3 LOW POWER GLASS DATA
NRX-A3 LOW POWER COBALT PLATE DATA
NRX-A3 HIGH POWER COBALT PLATE DATA
EXPERIMENTAL FAST NEUTRON FLUX ($E > 2.9$ MEV) ON THE REACTOR MIDPLANE (RADIAL TRAVERSE)

FROM ODD-K 2D TRANSPORT

○ PAX SULFUR DATA
● NRX-A3 HIGH POWER NICKEL DATA

RELATIVE RADIAL DISTANCE
EXPERIMENTAL FAST NEUTRON FLUX (E>2.9 MEV) ON THE SURFACE OF THE PRESSURE VESSEL (AXIAL TRAVERSE)

FROM ODD-K 2D TRANSPORT

- NEUTRON FLUX (N/cm² - SEC - WATT)
- RELATIVE AXIAL DISTANCE

NRX-A2 LOW POWER SULFUR DATA
NRX-A3 LOW POWER SULFUR DATA
NRX-A3 LOW POWER NICKEL DATA
NRX-A3 HIGH POWER NICKEL DATA
EXPERIMENTAL FAST NEUTRON FLUX ON THE MERIDIAN RING (E > 2.9 MEV)

FROM ODD-K 2D TRANSPORT

NEUTRON FLUX

CM$^2$ - SEC - WATT

10$^2$  10$^3$  10$^4$

20  40  60  80  100  120  140  160

REACTOR POLAR ANGLE, $\alpha$, DEGREES

- NRX-A2 LOW POWER SULFUR DATA
- NRX-A2 HIGH POWER SULFUR DATA
- NRX-A2 HIGH POWER NICKEL DATA
- NRX-A3 LOW POWER SULFUR DATA
- NRX-A3 LOW POWER NICKEL DATA
- NRX-A3 HIGH POWER NICKEL DATA
EXPERIMENTAL FAST NEUTRON FLUX ($E > 2.9$ MEV) ON THE CORE MIDPLANE VERSUS RADIAL DISTANCE

FROM ODD-K 2D TRANSPORT

- PAX SULFUR DATA
- NRX-A2 HIGH POWER NICKEL DATA
- NRX-A2 HIGH POWER SULFUR DATA
- NRX-A3 HIGH POWER SULFUR DATA
- NRX-A3 HIGH POWER NICKEL DATA
- NRX-A3 LOW POWER SULFUR DATA
- NRX-A3 LOW POWER NICKEL DATA

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

![Graph showing experimental differential neutron flux vs neutron energy](image-url)
EXPERIMENTAL THERMAL NEUTRON FLUX ($E < 0.4\text{eV}$) ON THE MERIDIAN RING

FROM ODD-K 2D TRANSPORT

NRX-A2 HIGH POWER DATA (BARE MINUS Cd COVERED COBALT)
NRX-A3 HIGH POWER DATA (BARE MINUS Cd COVERED COBALT)

REACTOR POLAR ANGLE, $\alpha$, DEGREES
TRANSPORT METHOD SUMMARY

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

\[ AX^2 + XX_0 + BY^2 + YY_0 + CZ^2 + ZZ_0 = D \]

\[ A(X - X_0)^2 + B(Y - Y_0)^2 + C(Z - Z_0)^2 = D \]

\[ (X - X_0)^2 + (Y - Y_0)^2 = D \]

\[ X = D \quad Y = D \quad Z = D \]
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. MONOVARIANT 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

\[ \psi(\vec{P}_m, E_0) = \exp[-b_T(E_0)] \]

\[ b_T(E_0) = \sum_{m=1}^{M} M_m(E_0) \vec{P}_m \]

COLLIDED FLUX

\[ \psi'(\vec{P}_m, E_0) = B(\vec{P}_m, E_0) \exp[-b_T(E_0)] \]

\[ B(\vec{P}_m, E_0) = \sum_{i=0}^{3} \beta_i(E_0)(b_T(E_0))^i \]

IF, \( b_T(E_0) > 20.0 \), \( B(\vec{P}_m, E_0) = \sum_{i=0}^{3} \beta_i(E_0)(20.0)^i \)
$\psi(p_m, E_n) = \exp \left[ f(W_R, E_n) \right]$ 

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

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

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

$N_R, E_n) = \left[ \sum_{i=0}^{I} \sum_{j=0}^{J} \Delta_{i,j} (E_n)_{j(120.0)}^i \right] + \left[ -\lambda(E_n) (W_R - 120.0) \right]$
EXPERIMENTAL GAMMA RAY DOSE RATE ON THE SURFACE OF THE PRESSURE VESSEL (AXIAL TRAVERSE)

- POINT KERNEL

Gamma Dose Rate (R/hr - watt)

RELATIVE AXIAL DISTANCE

- △ NRX-A2 LOW POWER GLASS DATA
- ○ NRX-A3 LOW POWER GLASS DATA
- □ NRX-A3 LOW POWER COBALT PLATE DATA
- X PAX TLD DATA

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EXPERIMENTAL GAMMA RAY DOSE RATE ON THE MERIDIAN RING

Point kernel (core $\gamma$ + "extra-core" $\gamma$)

Variables:
- Reactor polar angle, $\alpha$, degrees
- Gamma ray dose rate ($\text{yr}^{-1} \text{W}^{-1}$)

Data points:
- PAX TLD data
- NRX-A2 low power glass data
- NRX-A2 low power TLD data
- NRX-A3 low power TLD data
- NRX-A3 low power glass data
- NRX-A3 low power cobalt plate data

Graphical representation showing the distribution of gamma ray dose rate across different reactor polar angles.
EXPERIMENTAL GAMMA RAY DOSE RATE ON THE CORE MIDPLANE VERSUS RADIAL DISTANCE

- NRX-A2 LOW POWER GLASS DATA
- NRX-A3 LOW POWER TLD DATA
- NRX-A3 LOW POWER GLASS DATA
- NRX-A3 HIGH POWER COBALT PLATE DATA
- NRX-A3 LOW POWER COBALT PLATE DATA

GAMMA RAY DOSE RATE (R/HR - WATTS)

RADIAL DISTANCE FROM REACTOR AXIS (FEET)
EXPERIMENTAL FAST NEUTRON FLUX (E>2.9 MEV) ON THE SURFACE OF THE PRESSURE VESSEL (AXIAL TRAVERSE)

POINT KERNEL (BIVARIANT CARBON MOMENTS DATA)

RELATIVE AXIAL DISTANCE
EXPERIMENTAL FAST NEUTRON FLUX ON THE MERIDIAN RING (E>2.9 MEV)

POINT KERNEL (BIVARIANT CARBON MOMENTS DATA)

NEUTRON FLUX

NEUTRONS

CM^2 - SEC - WATT

REACTOR POLAR ANGLE, \( \alpha \), DEGREES

NRX-A2 LOW POWER SULFUR DATA
NRX-A2 HIGH POWER SULFUR DATA
NRX-A2 HIGH POWER NICKEL DATA
NRX-A3 LOW POWER SULFUR DATA
NRX-A3 LOW POWER NICKEL DATA
NRX-A3 HIGH POWER NICKEL DATA
EXPERIMENTAL FAST NEUTRON FLUX ($E > 2.9$ MEV) ON THE CORE MIDPLANE VERSUS RADIAL DISTANCE.

- PAX SULFUR DATA
- NRX-A2 HIGH POWER NICKEL DATA
- NRX-A2 HIGH POWER SULFUR DATA
- NRX-A3 HIGH POWER SULFUR DATA
- NRX-A3 HIGH POWER NICKEL DATA
- NRX-A3 LOW POWER SULFUR DATA
- NRX-A3 LOW POWER NICKEL DATA

POINT KERNEL (BIVARIANT CARBON MOMENTS DATA)
EXPERIMENTAL DIFFERENTIAL NEUTRON FLUX AT A RADIUS OF 5 FEET FROM THE CENTER OF THE REACTOR ON THE MIDPLANE.

POINT KERNEL (BIVARIANT CARBON MOMENTS DATA)

NRX-A2 FISSION FOIL MEASUREMENTS
NRX-A3 FISSION FOIL MEASUREMENTS
O Refers to Kap Zone Numbers

Point Kernel Code Geometry Model with Two Propellant Tanks
GAMMA RAY HEATING RATE VERSUS DISTANCE FROM THE BOTTOM OF THE ON-AXIS PROPELLANT TANK (TRAVERSE ON THE CENTERLINE)

MONTE CARLO VERSUS POINT KERNEL

FASTER MONTE CARLO

KAP V POINT KERNEL
NEUTRON KINETIC HEATING RATE VERSUS DISTANCE FROM THE BOTTOM OF THE ON-AXIS PROPELLANT TANK (TRAVERSE ON THE CENTERLINE)

MONTE CARLO VERSUS POINT KERNEL

FASTER MONTE CARLO

KAP V POINT KERNEL

USING HYDROGEN MONOVARIANT MOMENTS DATA

DISTANCE FROM BOTTOM OF TANK (CM)
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 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

\[ \phi_k(p_j, \hat{r}_j, E) = \int_0^\infty S_k(p - s\hat{r}_j, \hat{r}_j, E) \exp\left[ -\int_0^s \Sigma^\tau(p - s\hat{r}_j, E) ds' \right] ds \]

\[ S_k(p_j, \hat{r}_j, E) = \int \left( \int_0^\infty \phi_{k-1}(p_j, \hat{r}_j, E') \frac{d^2 \Sigma^S(p_j, \hat{r}_j, E' \rightarrow \hat{r}_j, E)}{d\Omega' dE'} dE' d\Omega' \right) \]

NO \quad CONVERGED

YES

STOP
THE MONTE CARLO METHOD

START

SPECIFY $S_0(\vec{r}, \Omega_j, E)$

$\vec{r}_0$ FROM $p_0^*(\vec{r})$

$W_0^S(\Omega_j, E) = S_0(\vec{r}_0, \Omega_j, E) / p_0^*(\vec{r}_0)$

$k = 0$

$\vec{r}_{k+1}$ FROM $p_{k+1}^*(\vec{r})$

$s_k = |\vec{r}_{k+1} - \vec{r}_k|$

$\Sigma_k = (\vec{r}_{k+1} - \vec{r}_k) / s_k$

$W_k^\phi(E) = W_k^S(\Omega_{k+1}) \exp\left[-\int_0^{s_k} \Sigma_k^2 (\vec{r}_k + s \Omega_{k+1}, E) ds\right]$

$s_k^2 p_{k+1}^*(\vec{r}_{k+1})$

$k = k + 1$

$W_k^S(\Omega_j, E) = \int_0^\infty W_{k-1}^\phi(E') d^2 \Sigma_k^S (\vec{r}_k, \Omega_{k-1}, E' \rightarrow \Omega_j, E) dE'$

NO

CUTOFF

YES

STOP
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 (\( \epsilon, \delta \)) 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
\[ p_o^*(\vec{r}) dV = u^*(\Omega) \nu^*(s) s^2 ds d\Omega \]

\[ \nu^*(s) = C \Sigma_V^*(\vec{r}_d + s\vec{\Omega}) \exp\left[-\int_0^s \Sigma^*(\vec{r}_d + s\vec{\Omega}) ds\right] / s^2 \]
OPTIMUM SCATTERED SOURCE SAMPLING

\[ P_1 = \int_0^{\frac{h}{2}} \Sigma^* (\vec{r}_k + s \vec{S}_d) \, ds \]
\[ P_2 = \int_0^{\frac{h}{2}} \Sigma^* (\vec{r}_d - s \vec{S}_d) \, ds \]
\[ P_T = P_1 + P_2 \]
\[ \eta_1 = \int_0^t \Sigma^* (\vec{r}_k + s' \vec{S}) \, ds' \]
\[ \eta_2 = \int_0^t \Sigma^* (\vec{r}_d + t' \vec{S}) \, dt' \]

\[ \frac{P_1}{P_T} \geq \delta, \quad \frac{P_1(\vec{r})}{P_T} = C_n P_1 \quad q_1^*(\vec{\Omega}) \quad \exp \left[ -\eta_1 - \eta_2 \right] \quad s^2 \, ds \, d\Omega \]
\[ s^2 \, t^2 \]
\[ \mu = \vec{\Omega} \cdot \vec{\Omega}_d, \text{ IF } \mu > 0 \text{, } s \leq h/2 \mu \text{ AND } t \geq h/2 \]

\[ \frac{P_2}{P_T} < \delta, \quad \frac{P_2(\vec{r})}{P_T} = C_n' P_2 \quad q_2^*(\vec{\Omega}') \quad \exp \left[ -\eta_1 - \eta_2 \right] \quad t^2 \, dt \, d\Omega' \]
\[ s^2 \, t^2 \]
\[ \mu' = -\vec{\Omega}' \cdot \vec{\Omega}_d, \text{ IF } \mu' > 0 \text{, } t \leq h/2 \mu' \text{ AND } s \geq h/2 \]

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EXPERIMENTAL GAMMA RAY DOSE RATE ON THE REACTOR MIDPLANE (RADIAL TRAVERSE)

FASTER MONTE CARLO CODE

O PAX TLD DATA

RELATIVE RADIAL DISTANCE
EXPERIMENTAL GAMMA RAY DOSE RATE ON THE SURFACE OF THE PRESSURE VESSEL (AXIAL TRAVERSE)

- FASTER MONTE CARLO CODE

GAMMA DOSE RATE (R/HR - WATT)

RELATIVE AXIAL DISTANCE

△ NRX-A2 LOW POWER GLASS DATA
□ NRX-A3 LOW POWER GLASS DATA
○ NRX-A3 LOW POWER COBALT PLATE DATA
X PAX TLD DATA
EXPERIMENTAL GAMMA RAY DOSE RATE ON THE MERIDIAN RING

GAMMA RAY DOSE RATE (R/HR - WATT)

REACTOR POLAR ANGLE, \( \alpha \), DEGREES

- O PAX TLD DATA
- \( \triangle \) NRX-A2 LOW POWER GLASS DATA
- \( \triangle \) NRX-A2 LOW POWER GLASS DATA
- \( \lozenge \) NRX-A2 LOW POWER TLD DATA
- \( \bullet \) NRX-A3 LOW POWER TLD DATA
- \( \bullet \) NRX-A3 LOW POWER GLASS DATA
- \( \times \) NRX-A3 LOW POWER COBALT PLATE DATA
EXPERIMENTAL GAMMA RAY DOSE RATE ON THE CORE MIDPLANE VERSUS RADIAL DISTANCE

- FASTER MONTE CARLO CODE

\[ 10^1 \]

\[ 10^0 \]

\[ 10^{-1} \]

\[ 10^{-2} \]

\[ 10^{-3} \]

\[ 10^{-4} \]

\[ 10^{-5} \]

\[ 10^{-6} \]

\[ 10^{-7} \]

\[ 10^{-8} \]

NRX-A2 LOW POWER GLASS DATA
NRX-A3 LOW POWER TLD DATA
NRX-A3 LOW POWER GLASS DATA
NRX-A3 LOW POWER COBALT PLATE DATA
NRX-A3 HIGH POWER COBALT PLATE DATA

RADIAL DISTANCE FROM REACTOR AXIS (FEET)

GAMMA RAY DOSE RATE (R/HR - WATTS)
EXPERIMENTAL FAST NEUTRON FLUX \( (E \geq 2.9 \text{ MEV}) \)
on the reactor midplane (radial traverse)

- Faster Monte Carlo code
- NRX-A3 high power nickel data
- Pax sulfur data

Relative radial distance
EXPERIMENTAL FAST NEUTRON FLUX (E>2.9 MEV) ON THE SURFACE OF THE PRESSURE VESSEL (AXIAL TRAVERSE)

RELATIVE AXIAL DISTANCE

NEUTRON FLUX (N/CM² - SEC - WATT)

- NRX-A2 LOW POWER SULFUR DATA
- NRX-A3 LOW POWER SULFUR DATA
- NRX-A3 LOW POWER NICKEL DATA
- NRX-A3 HIGH POWER NICKEL DATA
EXPERIMENTAL FAST NEUTRON FLUX ON THE MERIDIAN RING (E > 2.9 MEV)

FASTER MONTE CARLO CODE

NEUTRON FLUX

CM^2 - SEC - WATT

NRX-A2 LOW POWER SULFUR DATA
NRX-A2 HIGH POWER SULFUR DATA
NRX-A2 HIGH POWER NICKEL DATA
NRX-A3 LOW POWER SULFUR DATA
NRX-A3 LOW POWER NICKEL DATA
NRX-A3 HIGH POWER NICKEL DATA

REACTOR POLAR ANGLE, \( \alpha \), DEGREES

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EXPERIMENTAL FAST NEUTRON FLUX (E>2.9 MEV) ON THE
CORE MIDPLANE VERSUS RADIAL DISTANCE

NEUTRON FLUX (N/cm^2 - SEC.WATT)

0 10 20 30 40 50 60 70
DISTANCE FROM CENTER OF REACTOR (FEET)

○ PAX SULFUR DATA
△ NRX-A2 HIGH POWER NICKEL DATA
▽ NRX-A2 HIGH POWER SULFUR DATA
◇ NRX-A3 HIGH POWER SULFUR DATA
▲ NRX-A3 HIGH POWER NICKEL DATA
■ NRX-A3 LOW POWER SULFUR DATA
○ NRX-A3 LOW POWER NICKEL DATA

FASTER MONTE CARLO CODE
EXPERIMENTAL DIFFERENTIAL NEUTRON FLUX AT A RADIUS OF 5 FEET FROM THE CENTER OF THE REACTOR ON THE MIDPLANE.

The diagram shows the neutron flux as a function of neutron energy, plotted on a logarithmic scale. The flux is given in units of neutrons/cm²·sec·MeV·Watt. The graph compares the experimental measurements with those from a faster Monte Carlo code and fission foil measurements for NRX-A2 and NRX-A3.

Neutron Energy (Mev):
- 10^-2 to 10^0
- 10^0 to 10^1

Neutron Flux (neutrons/cm²·sec·MeV·Watt):
- 10^0 to 10^5

The data points for the faster Monte Carlo code are shown with solid lines, while the fission foil measurements for NRX-A2 and NRX-A3 are indicated with dashed and dotted lines, respectively.

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## CONVERGENCE HISTORY - PHOTON TRANSPORT  
### (MIDPLANE DETECTORS)

### NUMBER OF ITERATIONS

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<th>Point</th>
<th>100</th>
<th>200</th>
<th>400</th>
<th>% CHANGE 100 TO 400</th>
<th>% CHANGE 200 TO 400</th>
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<td>1</td>
<td>3.30 (+1)</td>
<td>3.07 (+1)</td>
<td>3.57 (+1)</td>
<td>+7.0</td>
<td>+14.0</td>
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<td>2</td>
<td>2.86 (+1)</td>
<td>3.40 (+1)</td>
<td>2.89 (+1)</td>
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<td>-1.8</td>
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<td>3</td>
<td>2.01 (+1)</td>
<td>1.65 (+1)</td>
<td>1.54 (+1)</td>
<td>-30.1</td>
<td>-7.1</td>
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<td>4</td>
<td>1.12 (+1)</td>
<td>1.13 (+1)</td>
<td>1.26 (+1)</td>
<td>+11.1</td>
<td>+9.5</td>
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<td>5</td>
<td>6.88 (+0)</td>
<td>7.80 (+0)</td>
<td>7.51 (+0)</td>
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<tr>
<td>6</td>
<td>3.78 (+0)</td>
<td>4.74 (+0)</td>
<td>4.67 (+0)</td>
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<td>7</td>
<td>7.77 (-1)</td>
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<td>8.12 (-1)</td>
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<td>7.24 (-3)</td>
<td>7.16 (-3)</td>
<td>+5.9</td>
<td>-1.1</td>
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<td>REACTOR PARTICLES</td>
<td>NRX NEUTRON</td>
<td>NRX PHOTON</td>
<td>FLIGHT NEUTRON</td>
<td>FLIGHT PHOTON</td>
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<td>SURFACES</td>
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<td>REGIONS</td>
<td>54</td>
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<td>DATA LOCATIONS</td>
<td>5500</td>
<td>6000</td>
<td>6700</td>
<td>6200</td>
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<tr>
<td>RUNNING TIME FOR POINT DETECTOR (100 HISTORIES)</td>
<td>90 s</td>
<td>48 s</td>
<td>60 s</td>
<td>45 s</td>
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</table>
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 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

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<th>METHOD</th>
<th>NEUTRONS/CM$^2$ - SEC - WATT</th>
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<tbody>
<tr>
<td></td>
<td>$E &gt; 2.9$ Mev</td>
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<tr>
<td>NRX-A2 MEASUREMENT</td>
<td>2.1 (3)</td>
</tr>
<tr>
<td>NRX-A3 MEASUREMENT</td>
<td>2.2 (3)</td>
</tr>
<tr>
<td>POINT KERNEL</td>
<td>2.0 (3)</td>
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<tr>
<td>FASTER MONTE CARLO</td>
<td>1.3 (3)</td>
</tr>
<tr>
<td>ODD-K TRANSPORT</td>
<td>1.1 (3)</td>
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</table>

LOCATION: 5 FEET FROM THE CORE CENTER ON THE MIDPLANE
SUMMARY

THERMAL NEUTRON ENVIRONMENT (E < 0.4 ev)

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

### APPLICATION OF METHODS

<table>
<thead>
<tr>
<th>OUTPUT DATA</th>
<th>METHOD</th>
<th>POINT KERNEL</th>
<th>TRANSPORT</th>
<th>MONTE CARLO</th>
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<td>Neutron Fission Distribution</td>
<td></td>
<td>NO</td>
<td>YES</td>
<td>NO</td>
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<tr>
<td>Internal Photon Sources</td>
<td></td>
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<td>YES</td>
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<td>Internal Environment</td>
<td></td>
<td>YES</td>
<td>YES</td>
<td>YES</td>
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<td>Surface Energy and Angular Distributions</td>
<td></td>
<td>NO</td>
<td>YES</td>
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<td>External Environment</td>
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<td></td>
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</tr>
<tr>
<td>a. Gamma Ray</td>
<td></td>
<td></td>
<td>YES</td>
<td>YES</td>
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<tr>
<td>b. Fast Neutron</td>
<td></td>
<td></td>
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<td>YES</td>
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<td>c. Thermal Neutron</td>
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<td>NO</td>
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<td>Propellant Heating</td>
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<td>Payload Environment</td>
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