A THREE-DIMENSIONAL SYNTHESIS APPROACH TO
REACTOR INTERNAL GAMMA-RAY HEATING*

J. T. West III
Union Carbide Corporation, Nuclear Division**

C. L. Whitmarsh
Babcock & Wilcox, USA

(Preprint of Paper P-II.8 is to be included in Proceedings of Fifth
International Conference on Reactor Shielding, April 18-22, 1977,
Knoxville, Tennessee)

*This work performed at Babcock & Wilcox, Lynchburg, Virginia
**Prime Contractor for the Energy Research and Development Administration.
A THREE-DIMENSIONAL SYNTHESIS APPROACH TO
REACTOR INTERNAL GAMMA-RAY HEATING

J. T. West III
Union Carbide Corporation, Nuclear Division

C. L. Whitmarsh
Babcock & Wilcox, USA

ABSTRACT

The construction of a unique procedure for generating a three-dimensional gamma heating data base has been completed for a PWR reactor design. The data base construction is achieved by the development of synthesizing equations combining several ANISN\(^1\) gamma heating results with several DOT\(^2\) gamma heating results. These three-dimensional equations contain appropriate corrections for particle streaming and scattering effects. This presents a new approach to the problem of three-dimensional data generation and efficient data transfer between radiation transport codes and thermal stress codes.

COMMERCIAL PWR REACTOR INTERNAL DESIGN PROCEDURE

Figure 1 shows a typical procedure for designing the reactor internal structure. The reactor internals must be designed to maintain its integrity under several severe power distributions representing a "maximum power" distribution and a "transient power" distribution. The maximum power distribution occurs during an abnormal operation condition and exists for only short periods of time. Transient power conditions are those power distributions that occur during a normal power transient. The power distributions are generated by the Core Physics Group responsible for fuel cycle analysis. The distributions are then given to the Shielding Analysis Group responsible for neutron and gamma flux analysis. This group uses one and two dimensional discrete ordinates transport codes to generate gamma heating* distributions corresponding to the "maximum" and the "transient" power distributions. These distributions are then given to the thermal and stress analysis groups responsible for structural and mechanical design verification. If there are any areas of concern that require re-evaluation, another design iteration may occur, represented by the dashed line on Figure 1. If the design is verified, then manufacture of the reactor internal components begins and plant licensing procedures are initiated. All of the analysis cumulates with construction and operation of the final reactor internal design.
REACTOR INTERNAL DESIGN FLOW

- GENERATE POWER DISTRIBUTIONS
- GENERATE GAMMA HEATING
- COMPUTERIZED DATA BASE FOR LARGE FLOW OF DATA
- GENERATE THERMAL DISTRIBUTIONS
- PERFORM STRESS ANALYSIS
- MANUFACTURE
- LICENSING
- GENERATE FUNCTIONAL SPECS

FLOW:
- FUEL CYCLE ANALYSIS
- NEUTRON & GAMMA FLUX ANALYSIS
- STRUCTURAL & MECHANICAL DESIGN

CONSTRUCTION & OPERATION
The procedure for designing a nuclear reactor internal structure involves an enormous flow of data between different engineering disciplines applying different methods for performing each specialized analysis. Since the data is generated with computer programs, it is most efficient to transfer this data via the computer. This report summarizes the flow of gamma heating data from a Shielding Analysis Group to a Thermal and Stress Analysis Group. The problems involved in this transfer are:

1) The Shielding Group uses 1 and 2 dimensional geometry models while the Thermal and Stress Group uses 3 dimensional geometry models.

2) Both shielding geometry models and thermal models apply arbitrary geometry grids which seldom correspond.

3) The shielding analysis is performed using metric units and the thermal and stress analysis is performed using British units.

**Gamma Heating Synthesis Equations**

The following equations utilize two dimensional results from several DOT calculations and one dimensional results from several ANISN calculations to synthesize simplistic three dimensional gamma heating distributions. Two-dimensional models of the reactor internals were represented in R-Z and R-O geometry. A coupled neutron gamma cross section library was used with appropriate gamma energy absorption factors to provide gamma heating output directly from the ANISN and DOT calculations.

There are two synthesizing equations used. The first equation is valid above and below the reactor core, but not to its side. The second equation is valid to the side of the reactor core, and has two expressions. Equation 2a is appropriate to the side of the reactor core with the exception being at elevations corresponding to the reactor core formers. Equation 2b is valid inside and to the side of the reactor core formers.

Equation 1 is valid for areas above the upper active fuel lines and areas below the lower active fuel line.

\[ \dot{q}(r_1, z_1, \Theta_1) = \dot{q}_H(r_1, z_1) H(z_1) K(r_1, z_1) C(r_1, \Theta_1) f_{DA} \]  

Equation 2a is valid for areas to the side of the active fuel, below the upper active fuel line and above the lower active fuel line, including the surface of core formers.

\[ \dot{q}(r_1 z_1, \Theta_1) = \dot{q}(r_1 \Theta_1) P(z_1) K(r_1) f_{DA} \]  

Equation 2b is valid for gamma heating in the center of the core-formers and the core-barrel in the shadow of the core-former.

\[ q(r_1, z_1, \Theta_1) = q(r_1, \Theta_1) R(\Delta r) P(Z) K(r) f_{DA} \]
The parameters in the above equations have the following definitions:

- \( r_j \) = radial distance from core center along the core major axis (cm)
- \( z_j \) = axial distance from core midplane (cm)
- \( \theta_i \) = the angle between a radial traverse from core center to a point of interest and a radial traverse from core center on a line perpendicular to the core flat (degrees)

\( \dot{q}(r_j, z_j) \) = gamma heating rates from a \( P_0 \) DOT R-Z cylindrical calculation (watts/cm\(^3\))

\( \dot{q}(r_j, \theta_i) \) = gamma heating rates from a \( P_0 \) DOT R-O at core midplane (watts/cm\(^3\))

\( C(r_j, \theta_i) \) = angular correction factor as applied to the DOT R-Z results (unitless)

\( C(r_j, \theta_i) = \frac{\dot{q}(r_j, \theta_i)}{\dot{q}(r_j, \theta = 0)} \)  

\( H(Z) \) = axial streaming correction accounting for increased axial gamma heating due to water channels offering fairly unobstructed gamma and neutron transport (unitless)

\( f_{DA} \) = normalization factor to account for unnormalized power distribution curves and to ratio DOT results back to ANISN results for added conservatism by not accounting for transverse leakage.

\( P(Z) \) = axial neutron power distribution between the lower active fuel line and the upper active fuel line; this function is meaningless anywhere else (unitless)

The "K Function" is an order of scatter correction factor used to correct the \( P_0 \) transport corrected DOT calculations for forward scattering. This is accomplished by defining "K" radially and axially separate as the ratio of a 1-D \( P_3 \) ANISN gamma heating to the 1-D \( P_0 \) ANISN gamma heating at the same location:

\[ K_r(r) = \frac{\dot{q}(r) P_3 \text{ANISN}}{\dot{q}(r) P_0 \text{ANISN}} \]  

\[ K_z(z) = \frac{\dot{q}(z) P_3 \text{ANISN}}{\dot{q}(z) P_0 \text{ANISN}} \]

The definition of \( K(r,z) \) is as follows:

\[ k(r,z) = 1 + [(K(z) - 1)^2 + (K(r) - 1)^2]^{1/2} \]

This definition of \( K(r,z) \) makes \( K(r,z) \) continuous.
The function \( K(r, z) \), as the actual two-dimensional radial and axial scattering correction factor, is not known exactly. This method maintains a conservative estimate as to the appropriate combination of \( K(r) \) and \( K(z) \). Geometric changes near the upper and lower corners of the reactor core increase the extent of the scattering correction approximation.

**Gamma Heating Calculational Flow**

The calculational flow to generate these functions involves a sequence of ANISN and DOT runs as shown in Figure 2. For large reactor models, a number of approximations and/or simplifications are necessary because of computer storage limitations and prohibitive cost of long running times. For this reason the two dimensional DOT runs were made with a \( P_0 \) transport library corrected \( P_1 \) scattering effects and with a reduced number of energy groups as compared with the one dimensional ANISN calculations. The effect of the reduced number of energy groups had a negligible effect on the bulk gamma heating but the reduction from a \( P_3 \) Legendre Expansion of the scattering cross section to a \( P_0 \) had a large effect on the bulk gamma heating. This is the reason for the "\( K \) Function" described in Equations 4-6.

The geometric modeling in two dimensions was limited in accounting for three dimensional streaming effects through axial water channels. Therefore streaming factors were calculated for the lower axial internals and for the upper axial internals. This was accomplished by running several x-y DOT calculations using a homogeneous model and using a heterogeneous model conserving the same water to steel volume ratios at each elevation. The results of the heterogeneous and homogeneous DOTs were radially averaged at each elevation and the ratio of the two was calculated to yield the "\( H(z) \) Function".

All of the DOT and ANISN results were saved on computer magnetic tape and processed with the "MAXTABL Program" which created a data file with all of the necessary information for the synthesis equations. This data base was then accessed with the "HOTBOX Program" by the Structura, and Thermal Analysis Unit. "HOTBOX" read in the discrete geometry model used in the thermal and stress codes and integrated over the volume of each grid using the synthesis equations to calculate the volumetric gamma heat generation for each grid. HOTBOX used linear logarithmic interpolation in one and two dimensions to estimate the gamma heating at arbitrary locations. The result of this approach was a direct link between the discrete ordinate transport codes used in the neutron and gamma flux analysis with the finite element heat transfer codes used in the thermal and stress analysis.

**CONCLUSION**

The limited geometric capabilities of discrete ordinate transport codes is enhanced by synthesizing several one and two dimensional calculations to simulate three dimensional geometry. This approach has proven itself efficient as an organized method of generating, manipulating, and
GAMMA HEATING CALCULATIONAL FLOW

RADIAL
ANISN
40 GROUP - P3

RADIAL
ANISN
26 GROUP - P0

AXIAL
ANISN
40 GROUP - P3

AXIAL
ANISN
26 GROUP - P0

DOT X-Y
ABOVE/BELLOW CORE
HETEROGENEOUS

CALCULATE
RADIAL AVERAGE
EACH ELEVATION

DOT X-Y
ABOVE/BELLOW CORE
HOMOGENEOUS

CALCULATE
RADIAL AVERAGE
EACH ELEVATION

DOT R-Z
LOWER & UPPER
MAX. & TRANSIENT

DOT R-O
MAX. & TRANSIENT

ANISN RADIAL
MAX. & TRANSIENT

ANISN AXIAL
LOWER & UPPER
MAX. & TRANSIENT

RATIO
K(r)

(K_r,z)

RATIO
K(r)

MATLAB
\( \phi(r,z,\theta) \)
3-D SYNTHESIS
DATA FILE

NORMALIZATION
CONSTANT

STRUCTURAL
AND
THERMAL
ANALYSIS
INPUT

HOTBOX
transmitting large quantities of gamma heating results to thermal and stress finite element codes on a production design basis.

The calculational flow to generate these functions involves a series of ANISN and DOT runs. The result of applying this method is the creation of a 3-D gamma heating data base where with an appropriate interpolation technique a gamma heating value can be generated given an arbitrary coordinate.

REFERENCES
