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September 1970

EVALUATION OF THE WANL
INTEGRATED NUCLEAR,
RADIATION, AND SHIELDING
STANDARD DESIGN METHOD
(WISDM)

VOLUME 1

MASTER

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VOLUME 1

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ABSTRACT

The WANL Integrated Nuclear, Radiation and Shielding Standard Design Method (WISDM) used at the Westinghouse Astronuclear Laboratory in nuclear and radiation design and analysis of the NERVA Nuclear Subsystem is described in this report. An evaluation of the accuracy of the WISDM methods and techniques is made by comparison with experimental results. WISDM is an integral part of an overall Design Engineering Method which is being developed to automate the mechanical, thermal, structural, nuclear, and radiation analyses and design of the NERVA Nuclear Subsystem.

Methods related to predictions in the following nuclear, radiation, and shielding analysis areas are discussed herein: 1) absolute reactivity (for component sizing or fuel loading), 2) control drum worth, 3) fission distribution (for fuel zoning or coolant allocation) 4) material reactivity worth (for evaluation of the effect of non-nominal compositions or conditions), 5) shield performance, 6) radiation heating rates (for component cooling requirements), and 7) radiation environment (for radiation damage).

This report is separated into two volumes. Volume 1 describes the WISDM techniques, evaluates the accuracy of the techniques by comparison with applicable experimental data, and discusses areas requiring improvement. Detailed statistical evaluation of the comparison of experimental and calculated results is presented for the methods used in calculating control drum worth, fission distribution, shield performance, radiation heating rate, and radiation environment. Volume 2, which describes the reactor systems evaluated by WISDM, is published as a separate volume since it contains Confidential Restricted Data.

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1.0 INTRODUCTION

This report presents a detailed description and evaluation of the WANL nuclear and radiation methods and techniques which have been identified by the acronym "WISDM," WANL Integrated Nuclear, Radiation, and Shielding Standard Design Method. The evaluation is performed by comparison of the analytical results with nuclear and radiation measurements obtained from the NERVA experimental program.

The WISDM nuclear and radiation analyses performed at the Westinghouse Astronuclear Laboratory in support of the NERVA Project are an integral part of an overall Design Engineering Method for the design and evaluation of the NERVA Nuclear Subsystem and its individual components. NERVA is an epithermal uranium-graphite fueled, beryllium reflected system having a median energy of fission of about 500 eV. The goal of this Design Engineering effort is to provide an automated and comprehensive design and evaluation of the Nuclear Subsystem and its components in accordance with the requirements that have been placed on it.

The current method development effort in the nuclear and radiation area is directed toward accomplishing several goals: 1) improvement of the ability of the analytical techniques to accurately predict the NERVA design characteristics, 2) elimination of duplicated effort in overlapping nuclear and radiation analyses, and 3) automation of nuclear and radiation methods to reduce the elapsed time required to perform an analysis and the transfer of information between the Nuclear and Radiation Design and other design groups. Significant progress towards these goals have been accomplished by defining a common neutron energy group structure, consistent neutron cross-section data, and consistent geometrical and material models. Substantial progress has also been made in automating the various methods employed in Nuclear and Radiation Design. The evaluation of the accuracy of the current WISDM techniques, that is described herein, is the basis for improving the methods. The evaluation effort performed to date in the following areas is presented:

- (a) Absolute Reactivity
(for use in component sizing and fuel loading computations)
- (b) Control Drum Worth
- (c) Fission Distribution (for use in fuel zoning or coolant allocation)

- (d) Material Reactivity Worth
(for evaluation of the effect of non-nominal conditions)
- (e) Shield Performance
(for engine component radiation protection)
- (f) Radiation Heating Rates
(for component cooling requirements)
- (g) Radiation Environment
(for radiation damage)

This document will be periodically updated as required to reflect improvements in the techniques and further evaluations and comparisons with experiments. Sufficient data is available in a number of the above categories to allow a statistical analysis of the difference between the analytical and experimental results to derive the average analytical bias and the variability in the analytical technique. Figure 1-1 is a flowchart of this evaluation which results in a technique which can be applied with a rated confidence to the NERVA design analysis tasks.

Figure 1-1. Evaluation of Accuracy of WISDM

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2.0 SUMMARY

The evaluation of the WISDM method defined in this report is based on recent comparisons with experimental data obtained over a period of years from nuclear experiments* in the NERVA Program. These experimental systems are PAX (Permanent Assembly Experiment), and FCX (Flexible Critical Experiment) at the Waltz Mill Experiment Facility, and NRX (NERVA Reactor Experimental) at the Nuclear Rocket Development Station.

The following sections summarize the present conclusions regarding the accuracy of WISDM defined in Chapter 3.

2.1 ABSOLUTE REACTIVITY CALCULATIONS

At the present time, basically two reactor assemblies have been analyzed in toto by WISDM techniques: (1) the PAX-E1, a critical mock-up of the NRX-A6 and (2) the FCX, a graphite reflected U-C epithermal system.

The error in the calculated absolute reactivity was found to be +1.5\$ for the complex PAX-E1 reactor geometry. A smaller +21¢ error in the calculated absolute reactivity for FCX, a reactor of simpler geometry, indicates that a possible reduction in the PAX-E1 error would result from an improvement in modeling in such areas as the core periphery/beryllium reflector region, the use of the transport corrected P_0S_4 scattering approximation, coolant channel streaming, and self-shielding corrections for fuel beads and unfueled elements in NRX/R reactor calculations.

2.2 CONTROL DRUM WORTH CALCULATIONS

The accuracy of WISDM control drum worth calculations has been established by an analysis based on four NERVA reactors: NRX-A6, PAX-G0, PAX-G1, and PAX-G2. The analysis indicates that WISDM in general overpredicts drum span and that the average ratio of measurement to calculation is 0.873 with a one sigma variability of 0.022. The results of a statistical analysis based on this data are presented in Figure 5-7. Use of this information

*A description of the reactors is given in Appendix A (Volume 2).

allows the selection of a multiplicative correction factor for the WISDM calculated drum span that will assure that a chosen design can meet its drum span requirement to a specified probability and confidence level. Given, for example, a criterion of 90 percent probability of meeting a drum span requirement to a confidence level of 95 percent, the best estimate of the drum span of a particular design is obtained by reducing the WISDM nominal calculational result by 22 percent (multiplying by a correction factor of 0.781).

2.3 FISSION DENSITY CALCULATIONS

Fission density calculations with WISDM in the NRX/R type reactors are in error in the vicinity of the core periphery. Detailed analysis of the PAX-E1 reactor revealed maximum errors of 17 percent and average errors of 11 to 12 percent in this area. The situation was confirmed in the PAX-G calculations^{25, 26, 55} where maximum errors of 25 percent were observed on the core periphery. A statistical analysis carried out for results obtained in the PAX-G0, G1, and G2 reactors indicates that the error in the calculated fission density at a given core location can be expected to be relatively constant. For example, analysis at PAX-G/R-1 core location P5F10 shows that the average ratio of measured to calculated fission density is 1.1840 with a one-sigma variability of 0.0107 (error of $18.40 \pm 1.07\%$).

In contrast to this, the FCX reactor fission density calculations were accurate to approximately 5 percent at all mid-axial core locations. In this reactor, errors increased at the top and bottom of the assembly near the reactor edge where the analytical model is known to be less efficient with regard to lack of neutron reflection and control rod simulation.

From the differences in analytical error in PAX-E1/PAX-G and FCX, it is apparent that an important weakness exists in some aspect of the PAX core periphery/reflector and control vane modeling. Two avenues are open in this regard other than the use of effective cross sections tailored to give the correct answer at a specific drum position for a specific reactor design. First, it may be possible to avoid "fit" cross sections by use of an analytical model especially designed to account for flux depression. The second approach is the use of the DOT code for solution of the neutron transport problem by transport theory instead of the diffusion theory of the EXTERMINATOR-2 solution.

2.4 MATERIAL REACTIVITY COEFFICIENT CALCULATIONS

The reactivity coefficient analysis reported here confirmed, in general, the suitability of the use of WISDM perturbation techniques and, specifically, qualified the PERT-1D code for use in NERVA calculations. The accuracy of these calculations for various materials is discussed below.

Reactivity coefficient calculations with WISDM for PAX-E1 and FCX have shown that for the two major constituent materials in NERVA type reactors, namely carbon and enriched uranium, the calculated and measured coefficients as a function of radial position agree generally to within ± 10 percent. One exception to this was the uranium coefficients in the reflector region of FCX where the calculation and measurement differed by a maximum of ± 15 percent.

For the FCX reactor, the calculated niobium coefficients are up to 30 percent higher than the experimental data. As analysis of other materials has shown, the use of effective cross sections, which included resonance self-shielding effects, would considerably improve the agreement between the calculation and the measurement.

In addition to the above, reactivity coefficients for tantalum and molybdenum were calculated in the PAX-E1 assembly and for hydrogen in the FCX assembly. The calculated tantalum reactivity coefficients differed by up to 20 percent from experimental data when effective cross sections were used. For molybdenum, the calculated and experimental coefficients differed by 50 percent even when effective cross sections were used. For hydrogen, the calculated and experimentally determined reactivity coefficients as a function of radial position in the FCX assembly are in excellent agreement.

2.5 RADIATION ENVIRONMENT CALCULATIONS

The accuracy of the radiation environment calculations was evaluated based on results obtained from an analysis of the PAX-E5, PAX-G1A, and FCX reactors.

Calculation of the PAX-E5 radiation environment measurements predicted the shape of the experimental results for all neutron and gamma detectors that were utilized. However, the absolute magnitude of the calculated results were, in general, somewhat lower than the measurements.

Examination of the PAX-E5 comparisons also resulted in the following conclusions:

- 1) ANISN one-dimensional calculations differ significantly from DOT two-dimensional calculations for low energy neutron detectors only. The DOT calculations displayed better agreement with the experimental results. This situation may result from a discrepancy in the buckling (transverse leakage) required in the ANISN code.
- 2) Point kernel neutron data agreed well with experimental results at the top of the 4B Configuration and were also comparable to the data obtained from DOT.
- 3) Photon absorbed dose rate data from point kernel and DOT were comparable.

Detailed comparisons between radiation measurements and calculations for the PAX-G1A reactor resulted in adoption of the form $\log(\text{Experiment}) - \log(\text{Calculation})$ to represent the error in the analytical results. Graphical analysis revealed that errors represented in this manner had a frequency distribution that was essentially "normal." This allowed the use of simple multiplicative factors to describe possible variations in radiation levels.⁵⁶ Table 8-2 summarizes the PAX-G1A midplane average differences between the calculation and measurement and the one sigma limits on this difference. As an example of the use of these data, Table 8-2 indicates that, on the average over the reactor, the calculated neutron absorbed dose rate was 2 percent less than the measured value. Further, the experimental result lay between the product of 0.89 and 1.16 times the calculated data for 68 percent of the observations.

Another example of the accuracy of radiation environment calculations with WISDM is summarized in Table 8-3 which has been reproduced from Reference 57. This table gives a comparison for various neutron energy regions of the computed and measured results at locations internal and external to the Cluster Hot End Support Hardware (CHESH). This reactor component is located outside the core fuel envelope at the aft end of reactor. As in the case of the data appearing in Table 8-2, the "Average Analytical Bias" (Figure 1-1) and the upper and lower one sigma limits for this bias were developed.

The analysis of the neutron activation data for the FCX reactor showed that the shape of the measurements was readily predicted by the analytical methods. However, calculations were consistently lower, by as much as 30 percent, compared to the experimental sulfur and dysprosium data. Near the upper surface of the FCX reactor, the presence of the control rods, which were partially inserted into the core for this series of measurements, caused significant differences between calculations and measurements since the control rods were not simulated in the analytical model.

2.6 SHIELD EFFECTIVENESS CALCULATIONS

Multiplicative correction factors for shield effectiveness calculations are developed for neutron and photon environment internal to (a) a 15-inch BATH shield mock-up and (b) a borated steel-LH₂ shield mock-up. Table 9-1 summarizes the average value of the correction factors and the one sigma upper and lower limits. In the BATH shield mock-up, for example, this correction factor has an average value of 8 percent for calculated neutron flux of energy greater than 2.9 MeV (sulfur response). The upper and lower one sigma limits in this case are 16 and 1 percent, respectively. Comparison of the results for the two shield configurations revealed that the ability to compute neutron environment in the two shields was similar; however, this was not true in the case of photon environment where the nominal correction factors for the CaF₂ TLD response calculations, 0.92 and 0.62, were found to differ by more than three standard deviations.

Correction factors for calculated radiation environment external to a typical internal shield configuration are presented for two locations on a plane tangent to the PVARA dome at a distance of 63 inches from the R-1 core center. The factors, summarized in Table 9-2, are shown for use with both discrete ordinate and point kernel calculations. Correction factors applicable to fast neutron calculations vary from 1.3 to 1.8 for the discrete ordinates method and from 1.8 to 3.1 for the point kernel technique. For photon analyses a slight reversal is observed. That is, for the discrete ordinates approach the uncertainty factor ranges from 1.7 to 2.0, while for the point kernel method the variation is from 1.2 to 1.4.

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3.0 DEFINITION OF THE WANL INTEGRATED NUCLEAR, RADIATION, AND SHIELDING STANDARD DESIGN METHOD (WISDM)

The nuclear, radiation, and shielding calculations performed at WANL are an integral part of an overall Design Engineering plan for the automated analysis and evaluation of a reactor system design or the design of an individual component. The goal of the analysis reported here is an evaluation of the analytical techniques, using appropriate experimental results in order to develop an understanding of errors and uncertainties in the analytical methods. This information is then used to comprehensively evaluate the ability of the component or system to meet the requirements that have been placed on it. This report describes in detail only the nuclear, radiation, and shielding analytical methods which are designated WISDM. For orientation, a brief description of the overall plan for reactor system design is also given. At this stage of the WISDM method development, many of the designated procedures are not as yet automated.

3.1 OVERALL FRAMEWORK OF WANL THERMAL, NUCLEAR, AND STRUCTURAL DESIGN PROCEDURES

The flow chart shown in Figure 3-1 illustrates symbolically the path for information flow during a design evaluation. Explanations of the rectangular and trapezoidal blocks appear below. In this figure a rectangular block represents the analysis activity of an engineering group. Each analysis activity block is identified by a number located in its left-hand corner. The trapezoidal blocks indicate data input or output associated with an analysis activity. The "inverted" trapezoid (base line shorter than the top line) represents an input data requirement, and the regular trapezoid (base line longer than the top line) represents output from an analysis activity. The input and output trapezoids are identified by a number at the upper left-hand corner.

The detailed analysis procedure, computer codes used, and the data flow for Nuclear and Radiation Design Block No. 11 are expanded in Figure 3-2. Those for the Thermal Design, Blocks No. 13 and 19, and those for Structural Analysis, Blocks No. 14 and 21 are documented in References 1 and 59, respectively. Figure 3-2 represents the next level in detail and complexity over Figure 3-1. The interrelationships between the various types of analyses

(thermal - nuclear - structural) can be seen in these figures. For example, in Figure 3-2, input information from Thermal analysis appears in trapezoid designated as 12. The flow path for the development of this information appears as output trapezoids on the appropriate flow charts in Reference 1.

In Section 3.2 the next level in detail and complexity for the nuclear, radiation, and shielding calculations is described.

3.2 NUCLEAR AND RADIATION DESIGN PROCEDURES

The sections that follow describe in detail the analytical procedures for the various types of calculations performed in the Nuclear and Radiation Design areas. Common to all of the procedures are the initial steps of (1) the calculation of regionwise converged neutron flux and current spectra from basic neutron cross-section data and atom densities and (2) the use of these spectra to obtain multigroup cross-sections. Techniques used in these two common areas are discussed in the following subsections.

a. Basic Neutron Cross-Section Data

The generation of basic neutron cross-section data has been carried out in recent years by the WANL Reactor Physics and Mathematics Department. Considerable effort has been expended in preparing and evaluating basic neutron cross-section data libraries (i. e. , smooth cross sections, resonance parameters, scattering kernels, etc.) for use in the GAMB1T¹ computer code (see following section for description). The basic library for GAMB1T comes from the GAM1,⁴ QUERY,²⁷ and BIT²⁹ and other data libraries (see Appendix C) which have been extensively updated and expanded by evaluation at WANL. References 17 - 24 describe the changes and additions to these libraries for the elements or isotopes listed in Table 3-1. These data libraries are the basic neutron cross-section data input to the nuclear analyses as shown in Figure 3-2.

b. Multigroup Neutron Cross-Section Generation

The calculational model employed in generating multigroup neutron cross sections uses spatially dependent fluxes and currents to spectrally reweight fine group cross sections from basic libraries to the chosen multigroup approximation. The technique used was

originally coded in the NSDM¹⁰ code which was a linkage of revised GAM1,⁴ QUERY,²⁷ and TNS²⁸ computer codes to solve for the nonthermal (fast), resonance, and thermal energy range cross sections. This linkage of codes underwent considerable improvement preceding the analysis reported here and these techniques were replaced in the WISDM model by the GAMB1T code. GAMB1T is an improved linkage of the revised GAM1, QUERY, and BIT codes which has been adapted to provide a design analysis tool in an automated (by magnetic tape) environment. The individual codes use the basic data libraries described above and are linked by calculation models to provide a consistent set of neutron cross sections over the entire neutron energy range. Only a description of the GAMB1T capability and linkage is included in the following paragraphs since the preceding capability in NSDM is functionally similar.

GAM1⁴

The GAM1 code used in the GAMB1T system has been modified to include the capability to use either the internally calculated neutron slowing down spectrum (fluxes and currents), using a P_1 or B_1 approximation to the Fourier transformed Boltzmann transport equation or an input neutron flux and current spectrum from magnetic tape or punched cards. A total of 68 energy groups are used to treat the energy range from 10 MeV to 0.414 eV, with each group having a lethargy width of 0.25. Scattering kernels in this energy range are usually treated as full scattering matrices for each isotope. Neutron capture, fission, inelastic scattering and (n, 2n) processes are also included. The resonance absorption is treated by using a modified QUERY method as an integral part of the GAM1 code. The GAM1 and QUERY libraries are combined consistently by the use of "smooth cross sections" in a multigroup format to provide neutron cross-section data in a form compatible with neutron transport and/or diffusion theory codes.

QUERY²⁷

A revised QUERY code has been adapted to the GAM1 code to provide a treatment of resolved and unresolved resonance parameters. The QUERY code computes the fine group cross sections for resonance capture and fission using the method of Adler-Hinman-Nordheim.⁵⁰ These data are combined with the GAM1 fine group data (smooth data) in the GAM1 code to provide total reaction (capture and fission) cross sections in the fine group energy range from 10 MeV to 0.414 eV.

BIT²⁹

The BIT code, which is an integral part of the GAMBIT code solves for the multi-group cross sections in the thermal energy range. In this energy range a neutron may either gain or lose energy as a result of a collision. Treatment of the combined effects of upscatter or downscatter have been handled in a variety of codes, of which the BIT code is one of the latest versions. The BIT code uses either a direct numerical solution of the thermal energy range neutron spectrum, using a P_1 or B_1 approximation to the Fourier transformed Boltzmann equation or an input neutron flux and current spectrum from magnetic tape or punched cards. BIT treats the energy range from 2.38 eV to 0.0 eV in 99 energy groups. Cross-section data are handled as smooth capture or fission with no resonance treatment in this energy range. Scattering cross-section data for moderators in the BIT energy range are treated as either complete 99×99 matrices for each element or isotope or as matrices which satisfy detailed balance. BIT computes multigroup cross sections in the thermal energy range from these 99 fine group data, using calculated or input spectra to group-collapse. In the BIT energy range, only within-group scattering and capture are included for isotopes designated as "absorbers."

GAM1-B1T Linkage

In order to provide multigroup cross-section data which span the entire neutron energy range from 10.0 MeV to 0.0 eV, the GAM1 and B1T codes have been linked by neutron slowing down calculation models. This linkage in energy is achieved by the specification of slowing down models for moderators and absorbers. The moderator slowing down is treated by multigroup scatter-transfer cross sections based on gas kernel scattering. Absorber neutron slowing down from the GAM1 energy range to the B1T range is treated by a stationary target model. This treatment of the neutron slowing down from the GAM1 energy range to the B1T energy range provides for complete cross-section matrices as output of GAMB1T.

c. Selection of Neutron Energy Group Structure

This section describes briefly the basis for the selected WISDM energy group structure. The structures that were ultimately chosen for the 16 group calculations and the 52 group spectrum calculations appear in Table 3-2. These group structures, a compromise of the nuclear and radiation/shielding requirements, were based on the following criteria:

- 1) Adequate high-energy group structure to provide reaction rates for heating calculations, comparisons with activation measurements, and representation of anisotropic scattering.
- 2) Adequate epithermal-energy structure to define the U-235 cross section in the resonance range (lethargy between 12 and 15.5).
- 3) Program storage limits (GAMB1T limit of 32 fast groups and 48 thermal groups).
- 4) Multigroup cross sections collapsible directly to few-group cross sections. This means that every few-group break point must fall at a multigroup break point. The motivation for this criterion is that it allows the use of ANISN to perform the collapse without another pass through GAMB1T to generate cross sections in the desired few-group energy structure.
- 5) Sufficient multigroups in the thermal range to adequately define the energy dependence of absorption and scattering cross sections.

- 6) Thermal group structure to equalize the relative influence of each thermal group in the numerical calculation.
- 7) Reasonable computer running time.

d. Converged Spectra

The procedure for the calculation of converged WISDM regionwise spectra is shown in Figure 3-3. Converged regionwise spectra are developed through iterative use of the LINK¹ code system (see Appendix C), an automated package of three computer programs. The first program, GAMB1T, provides cross section averaging as described above. In the first iteration* the WISDM procedure calls for the use of an infinite medium (P₁) flux and current spectrum for weighting of the fine group (68 fast and 93 thermal) data in the Nuclear Properties data file (see 3.2.a). The infinite medium flux and current spectrum is established (internal to GAMB1T) from the atom densities of the constituents of the region in question. For each region in the "Nuclear Model (8)," a 52 group set of macroscopic cross sections is calculated and placed directly on magnetic tape or disc.** Table 3-2 gives the energy structure of this 52 group scheme. (See Section 3.2.c for a brief discussion of the basis for the selection of this group structure.)

The second step in the LINK calculation is a multigroup solution of the neutron flux distributions in a reactor model, using these macroscopic regionwise neutron cross-section data in the ANISN¹ code (see Appendix C). The purpose of this solution is the recalculation of the neutron flux and leakage (for current) spectra for each region in the system as it interacts with the rest of the system. The radial one-dimensional ANISN calculations are performed with a poison ring to mock-up the control vanes in the reflector of the NERVA reactor. Cross sections for the ring vane are obtained during the spectrum calculation by averaging over an infinite medium beryllium reflector spectrum.⁺ The scalar and angular

*An iteration is defined to be one "pass" through Figure 3-3. The first iteration is not an iteration in the usual sense. For convenience it is so designated.

**Options for punched card output are also available.

⁺A trial was made to determine the effect of using a reflector spectrum updated after each iteration. This added refinement had little effect on the calculation and, therefore, has been dropped.

neutron fluxes from the ANISN calculation are processed in ANISIG-1 (see Appendix C), the third code in the LINK package, through use of a magnetic tape or disc.

ANISIG-1 serves three functions in the WISDM scheme. First, it computes regionwise neutron current in 52 groups from the ANISN angular flux data. Next, it interpolates and extrapolates the 52 group data to obtain flux and current data for 68 fast groups in the fast energy range (10 MeV to 0.414 eV) and 93 groups in the thermal energy range (1.86 eV to 0.0 eV). Table 3-2 which lists the groupwise energy data describes only 155 groups due to 6 "fast" groups (GAM1 groups) which overlap into the thermal range and are not used. Only the first 62 groups of the GAM1 energy range are used in GAMB1T with an energy breakpoint of 1.86 eV. In addition, the ANISIG-1 code computes the conventional standard deviation of the groupwise percent differences between the neutron flux and current in the present and the previous iteration of the spectrum convergence. (This last computation is bypassed in the first "iteration.") The purpose of this last step in ANISIG-1 is to determine if the regionwise flux and current spectrum are converged and if the iterative process shown in Figure 3-3 may be terminated. Samples of this convergence data are shown in Appendix B.

Experience to date in spectrum convergence (also shown in Appendix B) indicates that two iterations are required for satisfactory convergence. Regionwise neutron cross sections computed by GAMB1T in the third LINK iteration, using neutron flux and current data from the second iteration, do not significantly differ from those cross sections developed in the previous iteration.

Thus, two complete passes through the scheme shown in Figure 3-3 have been selected as the minimum required by WISDM. For the NERVA program NRX, FCX, and R reactor series, this minimum is suitable. Reactor systems which differ significantly from these, especially smaller, higher energy systems, may require more than two iterations.

e. Region-Dependent Neutron and Photon Cross-Section Generation

Application of these converged neutron flux and current data to develop neutron cross sections and ultimately neutron/photon cross-sections is illustrated in Figure 3-4. In the case of macroscopic neutron cross sections, a single additional step is required. This is

the use of the GAMB1T code to compute the WISDM prescribed 16 broad-group data (see Table 3-2). This is shown by the output trapezoid which represents data for both one- and two-dimensional calculations. Cross sections for a region/component that is either small in size or does not significantly affect the flux and current spectrum within its physical envelope are obtained by use of spectra from a nearby appropriate region/component. For example, cross sections for the NERVA reflector components (e. g. , springs, etc.) in two-dimensional calculations are obtained by weighting with the converged beryllium reflector spectrum.

Photon cross sections are developed in a manner analogous to the neutron cross sections. Basic photon cross-section data^{1,9} is weighted in the GAMLEG¹ code over a photon spectrum to obtain 13 energy group average (and energy point) cross-section data. The spectrum used in this procedure is not, however, the result of an iterative process as in the neutron case. It is, instead, the energy distribution of photons due to fission of U-235. The result of this computation appears in output trapezoids in Figure 3-4.

For certain other types of calculations, additional cross-section processing is required. This is shown in Figure 3-4 by activity block 11-2 where upscatter removal takes place. The first step in this removal is the use of the APPROPOS code which combines the neutron data from the GAMB1T code (16 groups) and the photon data from the GAMLEG code (13 groups). The resulting 29 group macroscopic cross sections are then processed through axial and radial ANISN one-dimensional transport calculations for removal of the upscatter entries in the neutron cross-section transfer matrix by a reaction rate balance. Figure 3-4 illustrates the calculation of axial and radial flux data output also.

3.2.1 Absolute Reactivity

The method of calculating absolute reactivity or reactivity differences is divided into two areas: calculations of an investigative nature or those at the trade study level and the method for the evaluation of a final design. The application of either of these methods is governed by the detail required of the analysis. The subsections that follow describe in detail these two areas of activity.

3.2.1.1 Reactivity for Trade Studies and Scoping of Conceptual Design

The method for obtaining reactivity at the trade study or conceptual design level is shown in Figure 3-5. This flow chart has been divided into one- and two-dimensional analyses to cover situations when the (preferable) one-dimensional analysis is not suitable. The calculations are performed with the same 16 group cross-section set with the exception of "two-dimensional" regions such as those in the NERVA reflector. One-dimensional calculations contain a ring (cylindrical annulus) mock-up of the control vanes in the reflector of the NERVA reactor design. The atom density and/or cross sections of the materials in the ring, its position, and its thickness are designed to approximate the effect of the actual control vane position on reactivity, fission distribution, and total absorptions in the vane. The WISDM technique for the development of this vane mock-up has been derived from an extension of a previous analysis similar to that reported in Reference 10. Cross sections for control vane material in two-dimensional calculations are obtained by weighting over a beryllium reflector spectrum (see Appendix B).

To evaluate a particular design alternate in a trade study, the one- or two-dimensional path in Figure 3-5 must be followed twice, one for the "base" case and a second time with a case containing the change to be evaluated. The difference in effective multiplication factor (eigenvalue) of the two calculations is then the reactivity worth of the design change.

3.2.1.2 Absolute Reactivity of a Final Design

The analysis of a final design differs from the trade study analyses described above in one important respect. This is the effort that is made to identify and assign a nuclear worth to all of the factors that cannot be or have not been included in the analytical model of the reactor. There is also an additional effort to provide a model that is as accurate as possible, for example, with regard to component masses.

The flow chart for this calculation appears in Figure 3-6, where one vertical path is concerned with the neutron transport solution for the reactor and the other with corrections

to that solution for effects which have not been included in the direct calculation. The neutron transport solution is performed with EXTERMINATOR-2, using macroscopic cross-section data developed in Figure 3-4. The cross sections for each reactor region are obtained by averaging over the converged neutron flux and current spectrum for that region or an appropriate region as described in Section 3.2.

Corrections for effects not included in the neutron transport solution are obtained from three sources. These are:

- (1) reactor test data from NRDS and WANEF (block 17 in Figure 3-2) and
- (2) analytical determinations such as the trade study type of calculations described in Figure 3-5 and use of Reactivity Analysis (Block 11-3 Figure 3-2) to estimate reactivity differences arising from inventory differences.

Thus, if it is desired to analyze Configuration A and neutron transport solution and reactivity is obtained for an approximation to A, A^* , the reactivity difference between A^* and A must be estimated and differences added to the eigenvalue or excess reactivity from the direct calculation. This approach is necessary since in very few instances is it possible for the analytical model to closely resemble the actual reactor in its design or test environment.

3.2.2 Control Drum Worth

The calculation of control drum worth is a direct application of the reactivity analysis described in the previous section. The flow chart for this calculation is shown in Figure 3-7. The determination is based on two EXTERMINATOR-2 calculations that use the same 16 group neutron cross-section sets. In one calculation, the control vanes are represented at their closest position relative to the reactor core ("full in"). In the second, the vanes are represented at their farthest position from the reactor core ("full out"). The control drum worth (span) is the difference between the calculated reactivity values from these two calculations.

3.2.3 Fission Distribution

Fission distribution calculations, like those for reactivity, are divided into several types: (1) those performed for scoping of a conceptual design, (2) those that are performed in the early design stages and are related to adjustments of the gross fission distribution to meet certain design criteria, and (3) detailed calculations to obtain the radial distribution of fission power. The subsections that follow describe these three types of calculations in detail.

3.2.3.1 Fission Distribution for Conceptual Design Scoping

Since there typically does not exist a need for highly accurate analysis during the conceptual stage of an evolving design, the method to be followed in these calculations has been considerably simplified over that of a final design evaluation. The flow chart for these calculations appears in Figure 3-8. The calculations are based on a one-dimensional neutron transport solution with the ANISN code. Neutron cross sections are obtained from GAMB1T using either an infinite medium flux spectrum or a flux and current spectrum from a previous calculation for a similar reactor. Microscopic data are used so that the "feedback" loop of the flow chart can be readily implemented. That is, if the initial region-dependent fuel distribution (uranium atom densities) used in the GAMB1T calculations and later in the ANISN calculation are not such that the design criteria are achieved, changes to the atom densities may readily be made manually and the ANISN calculation repeated any number of times until the criteria are met. This simplified procedure avoids the time and expense of spectral convergence for each of the designs considered.

3.2.3.2 Fission Distribution Calculations for Initial or Preliminary Fuel Zoning

The next step following the scoping of a conceptual design and the development of initial fuel zoning is the initial specification of uranium content in each fuel element. The flow path for this calculation is shown in Figure 3-9. Starting from the Initial Fuel Zoning and other (possibly non-nuclear) criteria, an initial uranium loading map is specified. From this, the region-dependent constituent atom densities are developed and the LINK calculation for converged regionwise neutron spectra (see Section 3.2) is completed.

These spectra are used in the GAMBIT code for fine group cross-section weighting and the production of regionwise 16 broad group cross-section input data for the neutron transport solution. This solution, with EXTERMINATOR-2, is done in an r,θ geometry and produces neutron fluxes for use in the last step in the process, the PFENCER-1 code. The output of PFENCER-1 is a detailed core power map which is examined to determine if the fuel loading distribution first specified has produced a power map that meets the design criteria. If it has, the calculation of "Initial Fuel Loading" is complete. If not, the loading of those elements not meeting the criteria is changed and, depending on the extent of the change, certain portions of the flow chart are repeated until the desired power map is attained and design criteria met.

3. 2. 3. 3 Power Distribution Calculations for Analysis of Final Design Configuration

The WISDM procedure for the calculation of the core power distribution of the final design configuration is shown schematically in Figure 3-10. The calculation is based on a two-dimensional (r,θ) model of the reactor that is augmented by a number of auxiliary calculations that effectively expand the "base" model from an (r,θ) nominal representation (ambient, midplane, as specified model) to that of an (r,θ,z) system (as-built) at the desired hot operating conditions.

The first step, depicted on the left side of Figure 3-10, basically consists of an (r,θ) midplane calculation of the radial variation of the fission density in the core of a reactor for which detailed ambient experimental data have been obtained. The experimental reactor used is as near--neutronically--to the final design configuration as possible. This requirement is based on the use of the calculated results as shown in the midportion of Figure 3-10. The calculated fission/gm-U values from PFENCER-1 are compared to the measured values in each of the locations within the reactor for which experimental data are available. The fission distribution is mapped experimentally in detail in the peripheral regions of the reactor core and in a coarser fashion in the central regions. The fission distribution data from these experiments are supplemented in PFENCER-2 in two ways. These techniques are described further in Section 3. 2. 3. 3. 1. The end effect of these operations is to provide a measured fission/gm U value at each coolant channel location for direct comparison to calculated values from PFENCER-1 for the ambient experimental configuration. The ratio of the

experimental to the calculated fission rate, NC_i , is computed at the location of each coolant channel in the core. These ratios (NC_i 's) are then used in PFENCER-3 as a means of correcting the "Calculated Fission Distribution" for the reactor at design conditions for the errors that exist in the calculated values. This approach assumes that the normalization factors (NC_i) calculated from the data for the ambient experimental reactor configuration will apply to a similar reactor at hot design conditions. The validity of this assumption is supported partially by the accuracy of the NRX-A6 power distribution analysis.¹¹

PFENCER-3 also serves as a vehicle for the inclusion of fission energy that is not deposited at or very near the fission site (neutron kinetic energy and photon energy deposition). This energy is considered separately because its spatial distribution differs significantly from that of the fission energy that is deposited at the fission site (fission fragment and delayed beta kinetic energy). This is discussed further in Section 3.2.3.3.1.

The right side of Figure 3-10 illustrates the development of the fission/gm-U values in the reactor at design conditions for use in PFENCER-3. As in the case of the experimental configuration, the analysis is based primarily on an ambient midplane (r, θ) model of the nominal reactor augmented by auxiliary calculations to account for factors not appearing on the base calculation. These include, for example, the "cold-to-hot swing," the effect of structures external to the core/reflector envelope such as the test cell shields and a number of local effects that cannot be readily or inexpensively included in the (r, θ) model. The techniques used for inclusion of special effects such as these are discussed in Section 3.2.3.3.1.

3.2.3.3.1 Techniques for Inclusion of Special Effects

Several references were made in the previous section to design features that influence the power distribution but are not included in its direct calculation (EXTERMINATOR-2 base case). The techniques used to include these special effects are discussed in this section and examples of these special effects to be factored into the calculation are discussed. The next section describes the application of selected special effects to achieve the desired end results of the fission distribution analysis.

Special effects can be divided into two categories: gross and local. Gross effects refer to design features that would affect the overall distribution of power generation if they had been included in the direct EXTERMINATOR-2 calculation. Gross effects are handled in two ways. First, if they are of an additive nature to the fission power generation, they are entered through the use of PFENCER-3 which permits the fission power values to be modified by the entry of one or more tables of values with associated radii and appropriate weighting factors. The effect is then introduced as follows:

$$\text{Total Power at channel } i = \left[W_1 \text{ FP}(i) + \sum_2^N W_k \text{ SE}_k(i) \right] / \sum_1^N W_k$$

where

FP(i) = the fission power density at channel i

SE_k(i) = the contribution of the special effect k to the power generation in channel i.

W_k = the weighting factor for effect k

N = the total number of effects

FP(i) and each SE(i) have been normalized to a core average value of unity.

The second type of gross effect is simply a multiplicative factor and is handled in the PFENCER-1 code. The multiplicative factors are a table of values with corresponding radii. The code interpolates these data to obtain the value of the special effect to be assigned to each particular channel at its radial position. Azimuthal variations are not included so that gross effects handled by this technique are only radially dependent. Examples of gross effects and their use in PFENCER are given in Table 3-3. Following is a brief description of these examples.

(A. 1) The cold-to-hot swing accounts for the effects on fission density of thermal and fluid condition changes in the reactor from an ambient to a partial or full power condition. It can be derived, for example, by comparing two one-dimensional calculations each performed in a manner similar to that shown in Figure 3-8.

(A. 2) The factor "Shim Concentrations" is included to account for the gross effect of concentrations of nuclear poison shim elements. While placement of shim elements is not part of the design, expedient core assembly techniques may require their placement to achieve the desired critical assembly. A calculation along the flow path of Figure 3-8 is used to estimate the gross radial effect of a spatial nonuniformity of shims.

(A. 3) Concentrations of materials that deviate from their specified value are handled in much the same manner as (A. 2). The effect may be, for example, a group of fuel elements with high uranium content. Because of their spatial concentration, they will cause more than a local effect (i. e., an effect within themselves) and may, indeed, change the local neutron spectrum on a large enough scale to affect a significant number of elements. Under these circumstances, the use of a gross radial correction factor is justified and desirable from a convenience standpoint in lieu of a recalculation of the EXTERMINATOR-2 base case (right side of Figure 3-10).

(A. 4) The inclusion of the gamma heating effect is necessary because the spatial variation of the energy deposition resulting from photon energy absorption and neutron kinetic energy absorption is significantly different from the spatial variation of fission fragment and delayed beta energy deposition. The magnitude and distribution of this effect is calculated by the NAGS computer code (Block 11-8 in Figure 3-2) and is factored into the power calculation in PFENCER-3 as an additive effect. The "fine structure" (distribution within the fuel element) of the gamma heating is entered as a local effect in PFENCER-1. Only the radial variation of gamma heating is entered as a gross effect.

(A. 5) The heat conduction effect results from the fuel cluster geometry and the energy that is generated in the unfueled element by gamma and neutron heating in excess of that removed by the cooling fluids. This energy is conducted to the fuel elements and adds to the energy load on the particular channels affected. It is handled in PFENCER-3 in a manner similar to gamma heating.

(B. 1) The shield backscatter effect tends to increase the neutron density in the core and is computed in a manner similar to that used for (A. 1), (A. 2), and (A. 3). All of these (four) effects are factored into the analysis in PFENCER-1. Because only one table of

values may be entered in the code, the (four) effects are superimposed and transformed into a single correction factor prior to their use.

Local effects arise, in general, from an inability to represent each component separately in the (r,θ) model used in all design analyses. Where appropriate, the materials contained in each component are homogenized over the entire region in which they are predominantly located. In some cases they are omitted entirely from the direct calculation. Local effects are included in PFENCER to account for these factors. The effects can be introduced in either PFENCER-1, 2, or 3. The correct entry point for various kinds of local effects is shown by the examples in Table 3-3. A brief description of these examples follows:

(C.1) Isolated "In-Place Shims" cause a local depression in the fission density that may extend beyond the cluster containing the shim. When this occurs, the effect is handled in two ways. First, the effect on the cluster that physically contains the shim is entered into the code through the use of a table of values with associated radii as measured from the cluster center. This provides a rapid means for correcting the fission density associated with each of the many coolant channels in a single cluster. The correction for the effects of a shim to the fission density assigned to the channels outside the cluster in question is done on a channel-by-channel basis and is more time consuming relative to the first technique.

(C.2) The presence of hydrogen coolant in the support stems for each cluster causes a local increase in fission density and must be handled in a manner similar to effect (C.1). The effect on the gross power shape of this hydrogen is included in the cold-to-hot swing (effect A.1) discussed earlier.

(C.3) The local effect to account for the variation in channel heat load across a fuel element arising from conduction of energy from the unfueled element is the "fine structure" of gross effect A.5. As noted in footnote 3 of Table 3-3, it is not rigorously correct to include it in PFENCER-1.

(C. 4) The effect on local fission density of the unfueled element in a regular cluster is entered into the analysis through PFENCER-2. Since PFENCER-2 only operates on calculations and data related to the ambient experimental configuration, this effect ultimately appears as part of the NC_i correction factor output from PFENCER-2. In effect, this factor in the analysis translates the gross center-of-element (Channel 10) measured data for the central regular portion of the core into a "measured" data value for each channel. (In the peripheral portions of the core, actual measurements are made in many more channels. A "measured" value for every peripheral channel is developed by interpolation from the actual data.)

(D. 1) The perturbation in fission density observed in special test fuel elements (such as the NRX-A6 elements with a matrix additive¹¹) is factored into the analysis in PFENCER-1. Channel-by-channel correction factors must be entered for the element in question as must correction factors for any affected channel outside the special element.

(D. 2) The calculation on the right side of Figure 3-10 is based on the nominal uranium fuel densities as specified in the fuel manufacturing specification document. This is done for convenience and timely completion of the power analysis. The density of fuel in the manufactured element, however, generally differs from the nominal value. The correction for this difference may be made elsewhere (see Section 3.2.3.3.2), but it can be done in PFENCER. If so, it will usually be made on an element basis in PFENCER-3 where the actual uranium mass in grams contained in each element is multiplied by the corrected fissions/gram value. The results for all channels in the core are then renormalized to a core average value of unity before they are combined with the special effects (SE_k) as described by the equation earlier in this section. The correction for this effect can also be made in PFENCER-1 if the uranium loading data are available prior to the analysis. The final use of the analysis will determine at what point the correction factors are added to account for this effect. See Section 3.2.3.3.2 for details.

3.2.3.3.2 Type of Final Design Analysis and Its Relation to Special Effects

This section discusses briefly the types of analyses and their bearing on the inclusion of special effects. Power distribution analyses performed as part of a nuclear design effort

fall into two categories: (1) analysis performed prior to reactor assembly and (2) analysis performed after a test series has been completed. The purpose of the first type of analysis is to provide a predicted power distribution for the actual as-built reactor at design conditions. It should theoretically include all of the effects appearing in Table 3-3 if they apply. However, four of the effects (A. 2, A. 3, C. 1, D. 2) are related to the characteristics of the actual as-built array of fuel elements. Since this first type of analysis is predictive and is, therefore, completed before the reactor assembly takes place, it cannot include the effects of non-nominal distributions of material in the reactor as it is finally assembled. Some of the effects can be included fortunately during the fuel element placement and flow orificing (EPOD)* activity. In fact, effect D. 2 which accounts for non-nominal uranium loadings, is routinely included in the EPOD activity. Effect A. 3, a gross radial correction for concentrations of material that deviate from their specified value, can be included in the EPOD analysis through a combination with a thermal effect. The remaining two items, A. 2 and C. 1, are related to In-Place Shims. These cannot be included in the EPOD work because, at present, orificing and element placement will be completed before shimming. This may, however, not be a problem because the design philosophy behind the in-place shims concept will insure that the local effects are small and gross effects are minimized.

The second type of power analysis is performed after a test series has been completed. The goal of this work is an evaluation of the accuracy of the predicted power distribution. The accuracy evaluation is performed by comparing data from a reactor tested at design conditions with a calculation for the reactor elements as they exist at the time of data acquisition. The special effects not to be included under these conditions are related to the Gamma Heating Correction (A. 4), and the correction for heat conduction from the unfueled element (A. 5 and C. 3). The primary contribution of these two items to power density is at design conditions and they are not directly observable during the post-test radiochemistry. The post-test analysis must, in addition, account for all deviations from the

*This acronym actually stands for Element Placement and Orificing Document.

planned testing program. Typically, the r, θ ambient EXTERMINATOR-2 calculation on the right side of Figure 3-10 must be redone at the effective drum position for the test series and the cold-to-hot swing must be re-evaluated using measured propellant conditions and component temperatures. This recalculation may also include actual material masses and the true element-by-element uranium loading. Since these will be homogenized into annular regions for the EXTERMINATOR-2 calculation, actual loadings may be input to PFENCER-1 to obtain final results. The meaning of nominal loading in PFENCER-1 changes from the spec value to that used in developing macroscopic cross sections for EXTERMINATOR.

See references 3 and 11 for a discussion of the NRX-A6 power analysis in these two areas.

3.2.4 Material Reactivity Coefficients

In general, the differential reactivity worth of a material (termed "reactivity coefficient") is important in two reactor development phases. First, in reactor design, knowledge of reactivity coefficients allows the selection of appropriate materials, that is, materials which may be necessary in order to satisfy a specific design requirement or goal. This area includes not only the effects of variations in the inventory of major constituents but also those of minor constituents such as reactivity poison shims. * Second, in the reactor assembly phase, the use of reactivity coefficients with actual material inventory data permits a final adjustment of the total system reactivity immediately prior to use (or testing). This effort is to assure that the design reactivity (i. e., the design ambient critical control drum position) will be met. This second use of reactivity coefficients is discussed in Section 3.2.1.2 and appears in Figure 3-6.

The calculation of material reactivity coefficients is, basically, a direct application of the reactivity analysis described in Section 3.2.1. However, for convenience and economy, a method has been developed using perturbation theory.³⁹ The method, applicable in both areas discussed above, is described below.

*Such uses of reactivity coefficients as in the calculation of feedback reactivity worth of propellant and material loss during high powered operation also fall into this category.

The method that has been developed to calculate reactivity coefficients is defined in Figure 3-11. The neutron scalar flux and current shown as input is obtained from the calculational scheme outlined in Figure 3-3. These converged spectra are used in the GAMB1T code to obtain 16 group, region dependent microscopic and macroscopic neutron cross section data. The macroscopic cross section data and the nuclear model for the system under analysis is input to the ANISN code in order to calculate (in 16 energy groups) the forward and the adjoint angular flux distribution in the reactor. The final step in the analysis, the one-dimensional perturbation theory calculation, uses the nuclear model, the GAMB1T calculated region dependent microscopic neutron cross sections, and the ANISN calculated forward and adjoint angular fluxes, to calculate the reactivity coefficients as a function of radial position for materials of interest.

The perturbation method used is an improved version of the PERT-1D code.³⁵ PERT-1D is a linear, one-dimensional, first order, perturbation theory code which calculates the reactivity coefficient of a material as a function of position in the reactor. In using the PERT-1D code, it is assumed that perturbations are small in that they do not significantly alter the flux of the reactor system being analyzed. * When this assumption is not justified, the eigenvalue techniques of Section 3.2.1 must be used instead of those described here. Since the reactivity coefficient is computed basically as the product of the forward flux, the adjoint flux and a cross section, the choice of an eigenvalue instead of a perturbation technique is required in situations where the estimated magnitude of the flux perturbation is greater than the uncertainty in the basic cross-section data. See Reference 51, Section 5.2.2 for additional discussion on this subject. ** The transport perturbation equations are presented in detail in Reference 39 and will not be repeated here. However, a change to the perturbation equations was required to account for the effect of transverse leakage.

*Several comparisons of PERT-1D results and direct ANISN reactivity difference results are given in Chapter 7.

**Note also that for certain tasks, perturbation theory is more appropriate. For cross-section evaluation, perturbation theory increases the sensitivity of energy dependence of integrals involving neutron cross sections in multiplying assemblies.⁵²

In ANISN calculations, the leakage in the direction perpendicular (transverse) to the direction of solution, is taken into account by increasing the total neutron absorption term by the factor DB_T^2 (taken from diffusion theory) where:

$$D = 1/3 \sum_{tr} \quad \text{---}$$

$$B_T^2 = \left(\frac{\pi}{H^*} \right)^2$$

and, $H^* = \text{extrapolated transverse dimension.}$

The transverse leakage corrected transport cross section then becomes:

$$\sum_{tr}^{LC} = \sum_{tr} + DB_T^2 = \sum_{tr} + \frac{B_T^2}{3 \sum_{tr}}$$

When subjected to a perturbation in the code, this term becomes:

$$\Delta \left(\sum_{tr}^{LC} \right) = \Delta \sum_{tr} - \frac{B_T^2}{3 \sum_{tr}^2} \Delta \sum_{tr} = \Delta \sum_{tr} \left(1 - \frac{B_T^2}{3 \sum_{tr}^2} \right)$$

The PERT-1D code includes this term at user option.

The PERT-1D code is a highly efficient program with a short running time. Each PERT-1D calculation requires approximately a ten second setup time plus one second for each material for which reactivity coefficients are calculated. Reactivity coefficients are determined for each mesh interval in the calculation. The use of PERT-1D results in a substantial saving of computer time, in that under the previous nuclear, radiation, and shielding design method, reactivity coefficients are calculated by running (for each mesh position of interest) a "base case" and "perturbed" ANISN calculation. The reactivity coefficient was then computed by comparison of these two problems. Under the WISDM procedure for calculating reactivity coefficients (Figure 3-11), the running of expensive "perturbed" ANISN calculations for each mesh position of interest is no longer required.

3.2.5 Radiation Environment, Heating, and Shield Effectiveness

3.2.5.1 Preliminary and/or Parametric Radiation Analysis

This section describes the WISDM analytical procedures employed in radiation and shielding analysis which is of a preliminary or parametric nature. An example of a preliminary analysis is the evaluation of a material change in an Engineering Critical Component. A parametric study may be the evaluation of the effect of varying the thickness of the internal shield.

Figure 3-2 showed the general outline of the procedure employed in Nuclear and Radiation Design. Section 3.1 describes the overall framework of the procedures. Section 3.2d describes the procedure by which a converged neutron spectrum, employed in cross-section computations is obtained, and Figure 3-3 illustrates the procedure. Section 3.2 describes the procedure employed to obtain neutron and photon transport cross sections and photon production data. This procedure was illustrated schematically in Figure 3-4.

Output from Figure 3-4 is therefore, the starting point for the radiation analysis procedure shown schematically in Figure 3-12. Twenty-nine group (16 neutron groups and 13 photon groups) macroscopic cross sections (with upscatter) for each region in the geometry, available as data block 11-1, are input to the one-dimensional discrete ordinate transport code, ANISN. Data block 11-1 in Figure 3-12 also supplies library and cross section data from the APPROPOS code so that a gamma ray transport analysis can be performed in ANISN.

Coupled 29 group neutron-photon ANISN calculations are performed to obtain the neutron and gamma ray fluxes and/or response (e.g., dose rate) internal to the reactor configuration. Either a radial or an axial problem geometry is used depending upon the design problem of interest. This calculation yields data for block 11-7 and/or 11-10.

The 29 group fluxes from ANISN, along with appropriate material library data, are input to the NAGS code which generates material and region-dependent heating rates throughout the geometry for data block 11-8.

To obtain radiation sources for data Block 11-9, a NAGS source calculation is made, employing as input either 29 group (NAGS uses only the first 16 groups) or 16 group neutron fluxes from ANISN and neutron and photon library data. These sources are utilized: 1) to prepare a Common Radiation Analysis Model (CRAM), 2) to evaluate the effect of a design change on the source intensity, and 3) to define region-dependent sources for use in point kernel analysis.

For parametric analyses, the point kernel code, KAP-V, is generally employed due to its flexibility and relatively short computer running time. The KAP-V code yields gamma ray dose rates and fast neutron fluxes and/or dose rates (based on infinite media moment data input) at locations external to the reactor for data Blocks 11-7, 11-8, or 11-10, as required.

3. 2. 5. 2 Detailed Radiation Analysis of the Nuclear Subsystem and/or Component

This section describes the WISDM analytical procedures employed in a detailed radiation analysis of the nuclear subsystem or a component thereof.

Figure 3-2 showed the general outline of the procedure employed in Nuclear and Radiation Design. Section 3.1 describes the overall framework of the procedures. Section 3.2.d describes the procedure by which a converged neutron spectrum, employed in cross-section computations, is obtained, and Figure 3-3 illustrates the procedure. Section 3.2.e describes the procedure employed to obtain neutron and photon transport cross sections and photon production data. This procedure was illustrated schematically in Figure 3-4.

Output from Figure 3-4 is, therefore, the starting point for the detailed radiation analysis procedure shown schematically in Figure 3-13. The 29 group macroscopic cross sections with upscatter removed (Block 11-1) are input to the SATURN code which prepares a cross-section tape for input to the two-dimensional (R,Z) discrete ordinates transport code, DOT (see Block 11-2). *

*The use of SATURN is optional and is required only when 29-group cross sections cannot be processed internal to the DOT problem due to computer storage limitations.

The 29 group radial and axial fluxes serve as a flux guess for the DOT code. An S_2 angular quadrature DOT problem is run to obtain a two-dimensional flux guess for a DOT problem with an S_6 (or higher) angular quadrature. This procedure is followed to economize computer time.

The 29 group flux data from the DOT problem are input to the NAGS code to generate neutron and photon heating rates throughout the reactor model (Block 11-7 and 11-8). The internal radiation environment data, i.e., fluxes and dose rates, are obtained directly from the (R,Z) DOT problem to yield Block 11-10 data.

The 16 group neutron fluxes (first 16 groups of 29 groups) from the DOT problem, along with Block 11-1 data, are input to the NAGS code to generate radiation sources throughout the reactor. These sources may be input to the KAP-V program to calculate the fast neutron and photon environment external to the reactor. The sources are also published as part of the Common Radiation Analysis Model (CRAM).

An alternate procedure for detailed analysis using the DOT code can be employed in certain design problems. The DOT code can be used with 16 group neutron cross sections, as described above, with the converged neutron fluxes processed in the NAGS code to provide photon source distributions. These data are used in a DOT photon transport problem. This sequential analysis provides the same data as calculated by the 29 group procedure, but requires less computer time than the simultaneous calculation of 29 group data.

Since the KAP-V code does not have the capability to calculate thermal and intermediate energy neutron fluxes, the MAP code is employed to transport the neutron fluxes from the reactor surface, through a void, to the external position of interest. From the MAP spectrum, the ratio of intermediate and thermal energy neutron fluxes to fast neutron flux is obtained. This is applied to the fast neutron flux calculated by KAP-V to obtain thermal and intermediate energy neutron fluxes. Therefore, the KAP-V data and the information ratioed by the MAP results provide the radiation environment external to the NSS (NN-4).

Output from either the MAP or KAP-V codes can be employed in the SCAP code to account for the effect on the nuclear subsystem of scattering from external structures, such as the test cell wall, etc.

It should be noted that the techniques described here and in Section 3.2.5.1 are those primarily used to perform radiation analysis. However, specialized calculations are often performed which do not conveniently fall into the schematic flow charts shown in Figures 3-12 and 3-13. These specialized calculations are briefly discussed in the next section.

3.2.5.3 Special Radiation and Shielding Analyses

This section briefly describes radiation analyses which do not conveniently fit into the calculational schemes shown in Figures 3-12 and 3-13.

1. Control Vane Heating - The analytical procedure described in Section 3.2.5.2 is applicable only to an r,z model which does not allow any azimuthal asymmetries. The control vane, which contains a boron poison, is physically located in the beryllium reflector as an off-axis cylindrical annulus. To calculate the heating deposition in the vanes and the heating distribution around a vane, the neutron flux data from an $r-\theta$ geometry EXTERMINATOR-2 calculation (activity 11-2) is employed.

2. Heating Rates from Fission Fragment Decay - For use in thermal analyses the heating rate by material and region from fission product gamma rays must be calculated, as well as that due to prompt fission energy release. To calculate this heating component, the DOT and NAGS photon problems (Figure 3-13) must be rerun for a specific reactor operating time, e.g., a 60-minute run, to obtain these output data separately. This information

is, therefore, applicable to a specific operating time. A special computer code, the S-4 program,* is used to calculate the gamma ray and beta release rate during or after a specific reactor operating profile. Output from the S-4 code, coupled with the DOT-NAGS output, yields the required transient heating rate data. (Delayed neutron fission heating is included in the transient thermal analysis procedure.)

3. Component Activation Analyses - Activation analyses of nuclear subsystem components, as well as various components located in the test cell area, are often required. This analysis is accomplished by reducing the 16 group neutron fluxes from the DOT and/or ANISN programs into 4 group fluxes which are input to the ACT II (Revised) code.* This code computes the activation source in the elements of interest as a function of time. The source terms are input to the KAP-V point kernel code to determine the dose rate from the component of interest.

4. Fission Product Sources After Shutdown - The source due to fission products as a function of time after a specific reactor operating profile is generated by means of the FPIP program.* These sources are then employed in the KAP-V point kernel code which calculates the gamma ray dose rate as a function of time after shutdown external to the nuclear subsystem.

5. Test Cell Analysis - Accurate prediction of the radiation environment in and around the Test Cell area, where reactors are tested at full power conditions, is a formidable task for any one particular computer code. The Test Cell environment is particularly difficult to calculate due to the complexity of the Test Cell geometry, which includes, in the case of NRDS's Test Cell C, a 360° facility shield, the privy roof shield, the test car and associated hardware, and the test cell walls and floor. An intricate, judicious combination of one- and two-dimensional discrete ordinate transport calculations, point kernel, and single scatter and albedo (SCAP) calculations, hand calculations, and engineering judgment are required for this analysis.

*See Appendix C.

6. Uncertainty Analysis and Application of Reactor Test Data – Figure 3-2, which shows the overall Nuclear and Radiation Design Detailed Analysis Procedure, indicates the application of reactor test data and uncertainty analysis (Block 17) in radiation environment and similar calculations. However, in general, it is very difficult to apply this type of data in an automated radiation analysis procedure.

Little or no experimental data has been obtained at Test Cell C internal to the reactor due to the high operating temperatures and radiation levels. The measured internal environment at Test Cell C is complicated by the effect of the test cell hardware, as discussed above.

Therefore, "uncertainty factors" applicable to radiation environment must generally be obtained by comparisons of calculations with measurements made in the PAX reactor operating at ambient conditions. To perform an exact analysis of the environment in the PAX reactor, including mock-ups of internal shielding, it is necessary to employ response functions in the discrete ordinates transport analysis to obtain calculated activation rates which can be compared directly with experimental data.

TABLE 3-1

NEUTRON CROSS-SECTION CHANGES AND ADDITIONS TO
GAM1, BIT and QUERY LIBRARIES

Element or Isotope	Reference	Element or Isotope	Reference
U ²³⁵	17, 19, 24	Si	24
U ²³⁸	17, 19	Ca ⁴⁰	24
Niobium	18, 22	Co ⁵⁹	24
B (B ¹⁰ , Natural)	20, 24	Cu (Cu ⁶³ , Natural)	24
Be	20, 23	Zn	24
W (W ¹⁸² , W ¹⁸³ , W ¹⁸⁴ , W ¹⁸⁶)	21	Y ⁸⁹	24
Mn ⁵⁵	22	Cd	24
In ¹¹⁵	22	Xe ¹³⁵	24
S ³²	22	Eu ¹⁵¹	24
Pb	22	Eu ¹⁵³	24
Zr	22	Dy ¹⁶⁴	24
		Au ¹⁹⁷	24
C	23		
D (In Heavy Water)	23		
H (Para, Ortho, Normal)	23		
Li (Li ⁶ , Li ⁷)	24		
Ni ¹⁴	24		
Mg	24		

TABLE 3-2
WISDM NEUTRON ENERGY GROUP STRUCTURE

Fine-Group Energy Structure				Broad Group Energy Structure		
Energy Group	Lower Energy ^a	Energy Group	Lower Energy ^a	Lower Energy ^a	52 Energy Group	16 Energy Group
1	7.79×10^6	109	0.25	7.79×10^6	1	
2-62	b	110	0.24	6.07×10^6	2	
63	1.725	111	0.23	4.72×10^6	3	
64	1.6	112	0.22	3.68×10^6	4	
65	1.5	113	0.21	2.87×10^6	5	1
66	1.44	114	0.20	2.23×10^6	6	
67	1.4	115	0.19	1.35×10^6	7	2
68	1.35	116	0.18	8.21×10^5	8	3
69	1.30	117	0.17	3.88×10^5	9	4
70	1.25	118	0.16	1.11×10^5	10	5
71	1.20	119	0.15	4.09×10^4	11	
72	1.125	120	0.14	1.50×10^4	12	6
73	1.10	121	0.13	5.53×10^3	13	7
74	1.05	122	0.12	2.04×10^3	14	
75	1.0	123	0.11	583	15	8
76	0.975	124	0.10	167	16	
77	0.950	125	0.095	78.9	17	9
78	0.925	126	0.090	61.4	18	
79	0.90	127	0.085	47.9	19	
80	0.876	128	0.080	37.3	20	
81	0.85	129	0.075	29.0	21	
82	0.835	130	0.070	22.6	22	
83	0.80	131	0.065	17.6	23	
84	0.775	132	0.060	13.7	24	
85	0.75	133	0.055	10.68	25	10
86	0.725	134	0.050	8.32	26	
87	0.700	135	0.045	6.48	27	
88	0.683	136	0.040	5.04	28	
89	0.650	137	0.035	3.93	29	
90	0.625	138	0.030	3.06	30	
91	0.60	139	0.025	2.38	31	
92	0.575	140	0.020	1.86	32	11
93	0.55	141	0.018	1.6	33	
94	0.532	142	0.016	1.35	34	
95	0.50	143	0.014	1.25	35	
96	0.475	144	0.012	0.975	36	
97	0.43	145	0.01	0.700	37	
98	0.414	146	0.009	0.400	38	
99	0.40	147	0.008	0.300	39	12
100	0.385	148	0.007	0.26	40	
101	0.375	149	0.006	0.20	41	
102	0.35	150	0.005	0.16	42	
103	0.325	151	0.004	0.12	43	13
104	0.30	152	0.003	0.10	44	
105	0.29	153	0.002	0.080	45	
106	0.28	154	0.001	0.060	46	14
107	0.27	155	0.0	0.040	47	
108	0.26			0.030	48	
				0.020	49	15
				0.009	50	
				0.005	51	
				0.0	52	16

^aElectron volts

^bEach group from 2 through 62 has a 0.25 lethargy width. Lethargy units are based on 10 MeV.

TABLE 3-3

EXAMPLES OF SPECIAL EFFECTS AND THEIR ENTRY POINT IN PFENCER

<u>Effect</u>	<u>PFENCER Entry Point</u>
GROSS:	
A. Cause Internal to Reactor Envelope	
1. Cold-to-Hot Swing ¹	PFENCER-1 (RT Subroutine)
2. Shim Concentrations	PFENCER-1 (RT Subroutine)
3. Concentrations of Materials that Deviate from Spec	PFENCER-1 (RT Subroutine)
4. Gamma Heating ²	PFENCER-3
5. Radial Variation of Heat Conduction from Unfueled Elements	PFENCER-3
B. Cause External to Reactor Envelope	
1. Shield Backscatter	PFENCER-1 (RT Subroutine)
LOCAL:	
C. Cause External to Fuel Element	
1. In-Place Shim	PFENCER-1 (CLI Subroutine)
2. Support Stem Hydrogen	PFENCER-1 (CLI Subroutine)
3. Heat Conduction from Unfueled Element ³ --the "fine structure" of Gross Effect A.5	PFENCER-1 (CLI Subroutine)
4. Regular Unfueled Element ⁴	PFENCER-2
D. Cause Internal to Fuel Element	
1. Special Test Fuel Elements	PFENCER-1 (CLI Subroutine)
2. Non-Nominal Uranium Loading	PFENCER-3 (or PFENCER-1) ⁵

1. This effect accounts for changes in the fission distribution that occur when reactor contains propellant/coolant and system temperatures are no longer ambient.

2. Energy deposition resulting from photon absorption and neutron kinetic energy loss.

3. Inclusion of this effect in PFENCER-1 is not rigorously correct because the contribution of conduction is additive but is handled in PFENCER-1 as multiplicative. Fortunately the effect is small (approximately 1 percent) and the additive effect can accurately be represented as multiplicative. For example, $1.1 + 0.01 \approx 1.1(1.01)$.

4. This effect is included here for completeness but is of a somewhat different nature. See text for discussion.

5. See Section 3.2.3.3.2.

Figure 3-1. Flow Chart for Nuclear, Thermal, and Structural Analysis

Figure 3-2. Nuclear and Radiation Design NSS Analysis (Block 11)

Figure 3-3. Flowchart for the Calculation of Converged Spectra

Figure 3-4. Flow Chart for the Calculation of Neutron and Photon Cross Sections

Figure 3-5. Flowchart for Calculation of Reactivity in Trade Studies and Scoping of Conceptual Design

Figure 3-6. Flowchart for Calculation of Absolute Reactivity of a Final Design.

Figure 3-7. Flowchart for Calculation of Control Drum Worth

Figure 3-8. Flowchart for Calculation of the Fission Distribution for Conceptual Design Scoping

Figure 3-9. Flowchart for Calculation of Initial Fuel Loading

Figure 3-10. Flowchart for Calculation of Power Map in Final Design at Test Conditions

Figure 3-11. Flowchart for Calculations of Reactivity Coefficients

Figure 3-12. Flowchart for Calculation of Radiation Environment, Heating and Shield Effectiveness for Preliminary and/or Parametric Radiation Analysis

Figure 3-13. Flowchart for Calculation of Radiation Environment, Heating, and Shield Effectiveness for NSS and Component Detailed Radiation Analysis

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4.0 ABSOLUTE REACTIVITY CALCULATIONS

This chapter discusses the calculation of reactivity for the PAX-E1* and the FCX* reactors and the corrections to the calculation for factors not directly included. The corrected reactivity values are compared with experimental data for the accuracy evaluation of the method.

4.1 PAX-E1 REACTOR

4.1.1 Eigenvalue Calculations

The PAX-E1 reactor was built to simulate certain nuclear characteristics of the NRX-A6 reactor. Because of limitations on the availability of certain materials and/or components, only a limited fraction (about 90° azimuthally) of the reactor simulated the NRX-A6 in detail. The remainder of the reactor resembled NRX-A6 in a gross manner. There were three objectives in the present analysis of PAX-E1: evaluation of the WISDM calculational accuracy for (1) absolute reactivity, (2) relative fission distribution, and (3) material reactivity coefficients. The selection of the NRX-A6 Quadrant in PAX-E1 with control drums positioned at 105°** as the model to be simulated in the calculation represents a compromise of a single calculation to serve these needs. (Unfortunately, this selection created the need for a number of additional corrections in the absolute reactivity evaluation. These corrections are discussed in the next section.) The calculation was performed according to Figure 3-6 with cross sections supplied by the scheme shown in Figure 3-4. The model was an EXTERMINATOR-2 (r, θ) representation at the reactor midplane. The finite height of the system was simulated by use of a buckling equal to 0.000450 cm^{-2} .†

The eigenvalue calculated for this system was 1.03261. This translates into an excess reactivity of +4.6\$.

4.1.2 Corrections for Effects Omitted from the Calculation

As noted above, a number of "corrections" were required in the development of the accuracy evaluation. All corrections have been handled from the standpoint of their "Effect on Total Reactivity if Item Had Been Included in the EXTERMINATOR-2 Calculation." Some of the corrections, shown in Table 4-1, are straightforward and no additional comment

*See Appendix A for physical reactor characteristics

**This average drum bank position is that expected to be the average over full power design operation; it is designated "Orifice Angle" (for the NRX-A6 Reactor).

† Reference 5, Pg. E-9.

need be made. However, this is not true of the last four items in the table. Reference to Figures 4-1 and 4-2 should help to clarify these items. Figure 4-1 summarizes schematically all of the reactor systems bearing on this analysis. The portion of Figure 4-1 appearing in Figure 4-2 shows the path for the corrections itemized in Table 4-1. The figures have been divided horizontally into "Experimental Systems" and "Analytical Systems." This division is actually lined off in Figure 4-1 but not in Figure 4-2. The Experimental System column has been subdivided into the two different nomenclatures appearing in the reference material on this subject. Note also that the core material composition of the analytical systems does not include shims.

The evaluation of the absolute accuracy of any calculation requires comparison with a comparable measurement. For the 105° EXTERMINATOR-2 calculation performed here (called EXT2-Quadrant Extrapolated, (QE), the ideal path for such an evaluation is the dashed path shown at the bottom of Figure 4-2. This requires an AMDS calculation that was not available for this evaluation. It is required because of the fuel differences between the real 360° system in the lower left-hand corner of Figure 4-2 and the quadrant-only composition of the EXTERMINATOR-2 calculation (EXT2-QE). The shaded path used here represents an alternate to the ideal path. In the shaded path, the calculation EXT2-QE is related to the reactor as it existed after the first step in the fuel interchange procedure had been completed (refer to Figure 4-1). In this state the reactor was made up largely of PAXD Rezoned Fuel but contained a keyhole of NRX-A6 Build-2 Fuel. The reactor was critical at 88°* with one TaC shim element present in location 3B2A which is included in Table 4-1. That is, if one shim could have been added appropriately to the EXTERMINATOR-2 calculation, the reactivity would have been reduced by 6%. The next item in the table is a result of the difference in drum position between the calculation (105°) and the A6-QE model (88°). The last two items are taken directly from Reference 2 and relate with measured reactivity coefficients the A6-QE model to the measured system in the upper left-hand corner of Figure 4-2. What this means in effect is that the measured system--the reactor as it existed after the first interchange step--was simulated by a reactor with quite a different drum position and core material composition.

*Reference 12, pg. 3 - 11.

The total correction is found to be -3.1% by summing the items in Table 4-1.

4.1.3 Conclusion

Applying the -3.1% correction to the calculated excess reactivity of $+4.6\%$ gives a net error of $+1.5\%$ in the excess reactivity calculated by the WISDM scheme of Figure 3-8 when supplied by macroscopic cross-section data developed in the manner shown in Figure 3-6.

4.2 FCX REACTOR

The Flexible Critical Experiment (FCX) Reactor test program was designed to provide data with which systematic nuclear investigations could be carried out to verify and improve the design methods used for NERVA nuclear calculations. The purpose was to reduce the uncertainty in prediction of such factors as critical mass, power distribution, reactivity coefficients, radiation heating, and kinetic parameters in NERVA systems. The primary design criterion of FCX was minimization of in-core structural material and impurities so that the cleanest possible critical experiments could be performed.⁴⁰

This section examines the absolute reactivity of the FCX reactor using certain overall WISDM techniques. In addition, the analysis also provides an evaluation of certain detailed aspects of the general method. These include:

- Basic Cross Sections (Section 3.2a)
- Multigroup Energy Structure and Broad Group Cross Sections (Sections 3.2c) and 3.2e)
- Method of Spectral Convergence (Section 3.2d)
- Use of 1-D (r) to simulate a 2-D (r,z) situation

4.2.1 Eigenvalue Calculations

The direct eigenvalue calculations for FCX were performed according to the left side of Figure 3-7 with cross sections supplied by the scheme in Figure 3-6. The model was an ANISN radial representation (see Appendix A) at the reactor midplane. The finite height of the system was simulated by use of an axial buckling equal to 0.000487 cm^{-2} . The eigenvalue calculated for this system with a simplified geometrical model was 1.0258 in 16 neutron energy groups. Since the code used here is ANISN, and not EXTERMINATOR-2 as in the PAX-E1 Analysis of Section 4.1, a direct comparison of this eigenvalue with a many-group model (52 groups) is readily available from the last step of the spectrum convergence calculations (See Figure 3-5). The eigenvalue obtained for the last iteration in the calculation of a converged FCX flux and current spectrum was 1.0237 (52 group calculation). This data is summarized in Table 4-2. Also shown are the results of a 16 group DOT calculation in (R, Z) geometry. Note that none of the eigenvalues and the "Direct Reactivity" values into which they have been translated differ significantly, especially the comparable pairs. That is, the first and second entries with different group structures differ only by 29 cents in reactivity. The second and third entries, which are basically different in geometry (radial in ANISN versus (r, z) in DOT), are separated by only 24 cents in reactivity.

Certain modifications to the "regular packed array" of fueled and unfueled elements were necessary to accommodate the seven control rod/fuel follower aluminum guide tubes present in the fueled portion of the FCX. Because the outside diameter of the aluminum tubes was larger than the available maximum inscribed circle of a seven element cluster, it was necessary to modify each fuel element immediately adjacent to each of the seven tubes. This required a modification to a total of 84 fuel elements. Also, it was necessary to remove material from the elements which made up the fuel follower so that they would fit inside the aluminum guide tubes. The total material removed was 17.12 kg of graphite, 3.88 kg of uranium, and 0.75 kg of niobium.

To evaluate the reactivity for the FCX depleted by this amount of material, a detailed geometrical model was developed (see Appendix A) and a 16 group ANISN calculation was performed with (1) an appropriately reduced mass distribution and (2) the aluminum

and stainless steel present in the control rod and guide tube assemblies. The calculation gave an eigenvalue of 1.0109 and indicated a reduction in reactivity of 2.1\$ when these modifications were included.

To verify this rather large effect, the measured reactivity coefficients (See Section 7.2) were applied to the mass reductions quoted above for carbon, uranium, and niobium. The results of this calculations is tabulated below:

Material	Amount Removed, kg	Approximate Reactivity Coefficient, β /kg	Reactivity Effect, \$
C	17.12	+ 8	-1.37
U	3.88	+18	-0.70
Nb	0.75	- 8	+0.06
Net			-2.01\$

Thus, the calculated -2.1\$, which includes other effects as well as those related to the three above materials, is an appropriate estimate of the reactivity change that resulted from use of the more detailed geometrical model.

The last column in Table 4-2 shows the reactivity values corrected for factors which could not be included in the direct calculation. These are discussed in the next section.

4.2.2 Corrections for Effects Omitted from the Calculations

Along with the direct eigenvalue calculations it is necessary to examine in detail the effect of factors which were not included in the direct calculation in order to accurately assess the WISDM capability in this area. The technique used in the direct eigenvalue calculation here was that of Figure 3-5. The corrections discussed in this section are analogous to the overall philosophy shown in Figure 3-6 where the results of a direct reactivity calculation are modified to account for factors which could not or were not included in the calculation. Alternatively, the analysis here can be considered as an evaluation of Figure 3-6 with ANISN replacing EXTERMINATOR-2 in the "Neutron Transport Solution." As in the case of PAX-E1, these factors or corrections are handled from the standpoint of the "Reactivity Change if Effect Had Been Included in the ANISN Calculation." The corrections are summarized in Table 4-3 where they have been divided into two broad categories: Geometric Model Effects and Neutronic Model Effects. This separation has been made in order to identify and apply only those corrections which are consistent with the philosophy of the WISDM evaluation. That is, to base the accuracy evaluation on an analysis where all reactors considered are examined with a given Neutronic Model and the best possible Geometric Model. Thus, the only purpose in examining the neutronic effects listed in the table (which relate to a more detailed model than the present WISDM) was to evaluate the appropriateness of the WISDM model in these areas. The correction then to be applied to the direct eigenvalue calculations in Table 4-2 is just the sum of the Geometric Model Effects, 1.15\$. The five individual effects listed in Table 4-3 are discussed in the subsections that follow.

4.2.2.1 Effect of Test Cell Walls

The reactivity contribution due to the reflection of neutrons from the test cell walls was estimated for PAX-E1, using configuration factors and infinite cylinder transport calculations to be approximately 10 cents.² The FCX assembly is approximately the same distance from the test cell wall as PAX-E1; therefore, the cell reflection reactivity contributions of both reactors were taken to be the same.

4.2.2.2 Correction for Non-circular Shape

Since the initial FCX reactor configuration was nearly hexagonal in cross-sectional shape (see Appendix A) rather than circular, the reactivity contribution of this geometrical effect was investigated to determine if the one-dimensional radial simulation required significant correction.

The method used to determine the magnitude of the effect is outlined briefly below:

- 1) Compute the eigenvalue for an axial ANISN model with radial buckling equal to $(2.405/R_{\text{eff}})^2$ where R_{eff} is the volume-conserving effective radius of the reactor.
- 2) Repeat the first step for a reactor with hexagonal fuel region cross section based on R. L. Murray's³⁰ recent work on this subject. This is accomplished by a rerun of the axial calculation with radial buckling determined from $(2.316572/R_i)^2$ where R_i is the maximum radius of a circle that could be inscribed in the hexagon.

Thus, to account for the hexagonal reactor cross section, the first zero of the Bessel Function, 2.405 (actually 2.404826), is modified to 2.316572 and the effective radius of the system is taken to be the radius of the inscribed circle. Note that in the limit as the number of polygon sides goes to infinity, the radius of the inscribed circle approaches the effective radius, R_{eff} .

The difference between the eigenvalues determined by steps 1 and 2, then, is a measure of the error in terms of reactivity that is to be expected when a hexagonal right-cylinder is approximated in radial calculations by a circular right-cylinder. For the FCX, this error was found to be -96 cents. That is, the radial ANISN calculations for FCX result in an eigenvalue that is estimated to be too high by 96 cents.

4.2.2.3 Effects of Coolant Channel Streaming

Coolant channels in the FCX core affect the reactivity of the system by increasing the average path length traversed by neutrons from the point of birth to capture or escape from the reactor. A correction for this effect is made by including the change in diffusion area due to voids. For isotropic scattering, this correction is made by use of a homogenizing technique where the diffusion area is effectively modified by the factor:

$$H = \frac{V_v + V_s}{V_s}$$

where

V_v = volume of void

V_s = volume of solid

When voids in a reactor have a definite orientation, simple homogenizing techniques are not adequate due to additional anisotropy caused by coolant channel geometry. A simple method of estimating the streaming that results from this anisotropic effect has been derived by Behrens³¹ and was used here to evaluate the magnitude of this effect. According to Behrens' derivation, the change in diffusion area due to circular holes with axis parallel to the centerline of the reactor is expressed as:

$$(\Delta Z)^2 = 1 + 2 V_f + \left[V_f (2r/\lambda) / (\exp (2r / V_f \lambda) - 1) \right] + 1.5 Q r V_f / \lambda$$

where

V_f = volume ratio of holes to material,

r = hydraulic radius of each hole equal to the ratio of twice the volume of each hole to its surface area,

λ = mean free path, and

Q = ratio of mean square passage length through hole to square of mean passage length. These factors have been tabulated by Behrens for different geometrically shaped holes.

Since the mean free path of neutrons is energy dependent, 16 group flux weighted values of $(\Delta Z)^2$ were obtained using a group-dependent mean free path. The ratio of the $(\Delta Z/H)^2$ was found to be 1.019.

The discussion that follows is related to how this factor can be used to compute the effect of streaming on reactivity. Using a one-group approximation, k_{eff} can be expressed as:

$$k_{eff} = \frac{k_{\infty}}{1 + L^2 B^2}$$

where

k_{∞} = the multiplication factor of an infinite array of fuel

L^2 = diffusion area equal to $\frac{D}{\sum_a}$,

D = diffusion coefficient and $\sum_a$ is the macroscopic absorption cross section,

B^2 = total geometrical buckling equal to $B_r^2 + B_z^2$,

B_r^2 = radial buckling, and

B_z^2 = axial buckling.

the change in k_{eff} due to a change in L^2 can be expressed as:

$$\frac{\Delta k}{k} = \frac{1 + L^2 B^2}{1 + L^2 B_r^2 \left(\frac{\Delta r}{H}\right)^2 + L^2 B_z^2 \left(\frac{\Delta z}{H}\right)^2} - 1$$

where

$\left(\frac{\Delta r}{H}\right)^2$ = a change in radial leakage due to holes oriented in the radial direction (this factor is unity here), and

$\left(\frac{\Delta z}{H}\right)^2$ = a change in axial leakage due to holes oriented in the axial direction.

Rearranging and simplifying the above equation may be changed to:

$$\frac{\Delta k}{k} = \frac{1}{F_a + F_r \left(\frac{\Delta r}{H}\right)^2 + F_z \left(\frac{\Delta z}{H}\right)^2} - 1$$

where

$$F_a = \frac{\sum_a}{\sum_a + DB^2}$$

$$F_r = \frac{DB_r^2}{\sum_a + DB^2}$$

$$F_z = \frac{DB_z^2}{\sum_a + DB^2}$$

The factors F_a , F_r , F_z are calculated and printed out by the DOT code in the balance tables and can be used directly in this analysis. Using the DOT calculation noted in Table 4-2, the void streaming correction was calculated for FCX to be -29 cents in reactivity.

4.2.2.4 Effect of Scattering Approximation

The reactivity effect that results from the use of a higher order scattering approximation than WISDM's transport corrected P_0 (and higher order S_n definition than WISDM's S_4) was evaluated as follows.

- 1) A WISDM transport corrected P_0S_4 ANISN calculation was performed with a finite buckling.
- 2) Infinite cylinder calculations were made with ANISN to determine the effects of P_1 and S_n for several cases.
- 3) The results of step 1 were combined with those of step 2 in the equation.

$$k_{\text{eff}}^* (P_1S_n) = k_{\infty} (P_1S_n) \frac{k_{\text{eff}} (P_0S_4)}{k_{\infty} (P_0S_4)}$$

to give the effective multiplication factor that might be expected for higher order P_1S_n . (The asterisk indicates that the k_{eff} value so obtained is approximate and not a direct ANISN result.) Table 4-4 summarizes the calculations performed and gives the result of use of the above equation for two cases.

The highest approximation available with these results, P_3S_8 , indicates that if WISDM calculations were done to this approximation, a reduction in eigenvalue of 0.0018 would be expected. This is equivalent to -25 cents in reactivity.

4.2.2.5 Heterogeneity Effects

In the WISDM mode, FCX materials are homogenized over the analytical regions in which they are placed without correction for localized flux depression or peaking. In this section, two areas of FCX heterogeneity are examined to determine the magnitude of the error that can be expected when making the above approximation in FCX calculations. The analysis is also applicable to NRX reactors because of their similarity to FCX in these two areas.

4.2.2.5.1 Fuel Bead Self-Shielding

The analysis of fuel bead self-shielding is divided into resonance energy effects and thermal energy effects. Fuel beads in the FCX fuel elements are composed of pyrographite coated UC_2 (Ref. 5). It is not possible to calculate directly the correct resonance self-shielding for these beads using the QUERY code due to the numerical interpolation technique which breaks down for very small diameters. The following describes an alternate approach to the evaluation of this effect. Resonance self-shielding factors for spheres are exponentially dependent³² on the radius of the sphere. Since it is expected that QUERY will show this exponential behavior at larger sphere sizes, the resonance self-shielding factors were calculated using the QUERY code for ten, fifty, and one hundred times the original size of the fuel bead and their reactivity effects calculated with ANISN in 16 groups. The results were then plotted, and the true reactivity effect due to the actual fuel bead diameter was determined by extrapolating to the original size. The change in reactivity due to microsphere resonance self-shielding was found to be 3.3 cents.

Thermal self-shielding factors were calculated using the first flight transport approximation.³³ The ANISN calculation was then repeated using self-shielded cross sections to calculate a new eigenvalue for FCX. The result of the calculation indicated a -2.9 cent change in reactivity due to the thermal self-shielding effect.

Adding the contribution from resonance and thermal self-shielding, a negligible net change in reactivity of 0.4 cents is obtained.

4.2.2.5.2 Effects of Unfueled Elements

In order to determine the reactivity effect of homogenization of unfueled elements in the FCX core, cell-weighted macroscopic cross sections were generated as a function of core position. The hexagonal cell geometry in the FCX was approximated by a circular shape and a shell source was used in the ANISN code to simulate the in-coming neutron spectra from outside of the cell (this came from a separate ANISN problem). The resulting cell-weighted cross sections were then used to recalculate a new eigenvalue for FCX. The change in reactivity was found to be -9 cents.

4.2.3 Comparison of Calculations with Measurement

The total of the corrections developed in the previous section for Geometric Model Effects is -1.15\$. This has been applied to the "Direct Reactivity" in Table 4-2 to give the "Corrected Reactivity" appearing in the last column of this table. The result for the base case (last entry) shows that the predicted reactivity difference of FCX is +39 cents. The measured value with the seven control rods fully withdrawn from the core was +18.1 cents.³⁴ Thus, the corrected calculation overestimates the true reactivity by only 21 cents.

4.3 EVALUATION OF ACCURACY OF ABSOLUTE REACTIVITY CALCULATIONS

The analysis of the FCX reactor indicates that in clean geometry, WISDM methods can provide an accurate calculation of absolute reactivity. When viewed in this context, the +1.5 dollar error in the more complex PAX-E1 geometry appears to indicate that an improvement in modeling in such areas as the beryllium reflector region is necessary. Note that a 29 cent reduction in this error is possible if the coolant channel streaming effect as calculated for the FCX reactor in Table 4-3 is included for PAX-E1 and that an additional 34% reduction is possible if the Neutronic Model effects are included.

The analysis of FCX also yields some specific conclusions with regard to certain parts of the WISDM technique. FCX has provided some confirmation in the following areas:

- 1) Cross-section libraries (for the isotopes used in the FCX analysis),
- 2) The multigroup cross-section energy structures,
- 3) The use of two iterations for spectral convergence,
- 4) The use of 1-D (radial) analysis to simulate a 2-D (r, z) situation for scoping studies,

The analysis of FCX also indicates additional effort is appropriate for NERVA reactivity analysis in the following areas:

- 1) The use of the transport corrected P_0S_4 scattering approximation
- 2) Omission of coolant channel streaming
- 3) Self-shielding corrections for fuel beads and unfueled elements in NRX/R reactor calculations.

TABLE 4-1
SUMMARY OF CORRECTIONS TO TWO-DIMENSIONAL CALCULATION FOR PAX-E1

<u>Item Not in Extremator-2 Calculation</u>	<u>Effect on Total Reactivity if Item Had Been Included in the Extremator-2 Calculation, ϵ</u>	
<u>External to Reactor:</u>		
Pressure Vessel ¹	+ 30	
Detectors ²	+ 05	
Walls ³	+ 10	
Test Stand ⁴	+ 07	
<u>Internal to Reactor:</u>		
X Wires ⁵	- 92	
Bundling Mechanism ⁶	- 35	
TaC Shims ⁷	- 06	
Change in Critical Position, ⁸		
Calculated 105 ⁰⁹ to Measured 88 ⁰¹⁰	-114	Discussed in Text
Reactivity Change between A6-QE and PAX-QE ¹¹	-122	
Reactivity Change between PAX-QE and Rezoned PAXD with Keyhole ¹²	+ 7	
	<u>+ 7</u>	
TOTAL	-310ϵ	

1. Ref. 2, Pg. 90.
2. Ref. 2, Pg. 89.
Ref. 13, Pg. 3-21.
3. Ref. 2, Pg. 89.
4. Ref. 2, Pg. 89
5. Ref. 2, Pg. 64.
6. Ref. 2, Pg. 82.
Ref. 13, Pg. 3-21.

7. Ref. 2, Pg. 2.
Ref. 12, Pg. 3-52.
8. This reactivity change is based on drum span
of 7.55\$ (Ref. 12, Pg. 3-10) and a cosine shaped
integral worth curve (see for example Ref. 12,
Pg. 3-8).
9. Ref. 12, Pg. 3-29; See also Figure 4-2.
10. This was calculated by subtraction of 20.8 ϵ (Ref.
12, Pg. 3-11) critical position shown in Table 4-1.
See also Figure 4-2.
11. Ref. 2, Pg. 64; see also Figure 4-2.
12. Ref. 2, Pg. 62; see also Figure 4-2.

TABLE 4-2
SUMMARY OF FCX ABSOLUTE REACTIVITY CALCULATIONS
-DIRECT AND CORRECTED-

Code Used	Number of Groups	Eigenvalue	¹ Direct Reactivity, \$	² Corrected Reactivity, ¢
<u>Simplified Geometrical Model</u>				
ANISN	52	1.0237	3.34	219
ANISN	16	1.0258	3.63	248
DOT	16	1.0275	3.87	272
<u>Detailed Geometrical Model⁴</u>				
ANISN ³	16	1.0109	1.54	39

-
1. Calculated from eigenvalue by $(k - 1)/0.0071$.
 2. Direct Reactivity + Correction from Table 4-3.
 3. WISDM base case.
 4. The detailed geometric model accounts for (1) fuel material removed to accommodate the control rod/fuel follower guide tubes and (2) aluminum and stainless steel present in the control rod and guide tube assemblies.

TABLE 4-3
SUMMARY OF CORRECTIONS TO DIRECT EIGENVALUE
CALCULATION FOR FCX

Effect Not in ANISN Calculation	Reactivity Change if Effect Had Been Included in ANISN Calculation, β
<hr/>	
Geometric Model Effects	
(Factors which could not be included in ANISN Calculation)	
(1) Test Cell Walls	+ 10
(2) Noncircular Shape	- 96
(3) Coolant Channel Streaming	<u>- 29</u>
Total	-115 (To be applied in Table 4-2)
*Neutronic Model Effects	
(Factors which could have been included in ANISN Calculation)	
(1) Scattering Approximation (P ₃ vs P ₀)	- 25
(2) Heterogeneity	<u>- 9</u>
Total	- 34

* These effects have been examined in order to evaluate the appropriateness of the WISDM models in these areas. Since the WISDM model does not include these effects, they have been excluded from the correction applied in Table 4-2.

TABLE 4-4

SUMMARY OF SCATTERING APPROXIMATION CALCULATIONS

<u>Approximation ($P_l S_n$)</u>	<u>Buckling</u>	<u>Eigenvalue</u>	<u>Change in Eigenvalue from Base Case</u>
<u>Calculated with ANISN</u>			
$P_0 S_4$ (Base)	Finite	1.0237	---
$P_0 S_4$	Zero	1.1700	0.1463
$P_1 S_4$	Zero	1.1684	0.1447
$P_3 S_8$	Zero	1.1679	0.1442
<u>Calculated from Equation in Text</u>			
$P_1 S_4$	Finite	1.0223	-0.0014
$P_3 S_8$	Finite	1.0219	-0.0018

EXPERIMENTAL SYSTEMS		ANALYTICAL SYSTEMS	CRITICAL POSITION	SHIM CONTENT OF WANEF SYSTEMS
NOMENCLATURE: WANEF Quarterly PR		WANL-TME-1713
REZONED PAX-D FUEL			90.74°	6 (1-6)B2A
STEP 1 OF FUEL INTERCHANGE (KEYHOLE)		PORTION SHOWN IN FIGURE 4-2

REZONED PAX D FUEL NRX-A6 BUILD 2 FUEL	PAX-O A3-C PAX-ET FUEL	A3-C PAX-QE FUEL	88°	1 382A
STEPS 2, 3, 4, 5 OF FUEL INTERCHANGE
NRX-A6 BUILD 2 FUEL REZONED PAX D FUEL			NA	6 (1-6)B2A
NINE STEPS OF INTERCHANGE COMPLETE
NRX-A6 BUILD 2 FUEL REZONED PAX D FUEL	A6-Q A3-C PAX-E1 FUEL	A3-C A6-QE FUEL	88°	6 (1-6)B2A
NRX-A6 BUILD 2 FUEL REZONED PAX D FUEL		EXT2-QE (SYSTEM CALCULATED IN PRESENT ANALYSIS)	104.8°	29 SEE FIGURE 4-2
NOTE: RINGS WERE USED IN ALL REACTOR CALCULATIONS				

Figure 4-1. Schematics for Reactor Systems Discussed

Figure 4-2. Path for Corrections to Calculated System
(Selected Schematics from Figure 4-1)

5.0 CONTROL DRUM WORTH CALCULATIONS

Control drum worth calculations are basically a direct application of the reactivity type of calculations described in Section 3.1. Following the scheme shown in Figure 3-7, two EXTERMINATOR-2 calculations are performed, one with the control drums simulated at their position closest to the core and the other at their position farthest from the core. The difference in the excess reactivity of the two calculations is, then, the control drum worth. Evaluation of WISDM techniques for control drum worth calculations is based on comparison of NRX-A6 and PAX-G series reactor experimental results with calculated results. Details of the PAX-G calculations and measurements are discussed in References 15, 16, and 53. Details of the NRX-A6 reactor calculation are presented in Section 5.1 and experimental results for NRX-A6 appear in Section 5.2. A comparison of experimental and calculated results is given in Section 5.3. Conclusions are reached regarding the average analytical bias and its variability for control drum worth calculations.

5.1 ANALYTICAL MODEL FOR THE NRX-A6 TWO-DIMENSIONAL CONTROL DRUM WORTH CALCULATIONS AND RESULTING DRUM WORTH

Decision areas in nuclear modeling fall generally into four broad categories. These are (1) mesh line spacing, (2) assignment of mesh cell materials and masses (atom densities), (3) choice of the neutron flux and current spectrum to be used for weighting of cross sections for each mesh cell, and finally (4) the neutron cross sections themselves.¹⁷⁻²⁴ The complete reactor model for the NRX-A6 drum worth calculations is presented in Appendix A. Selected details of the reflector and control vane model will be given here. The NRX-A6 spectra, as described in Section 3.2 d and given in detail in Appendix B were developed from the one-dimensional radial model shown in Figure 5-1 by use of the WISDM LINK system. The reflector is broken up into three zones from an atom density standpoint. In the ANISN spectrum calculations, the two Beryllium-3 zones shown were separately designated in order to conveniently obtain a neutron flux and current spectrum for beryllium material in front of the control vane, Be-3(I) and a spectrum behind the control vane Be-3(O). This made it possible to use more appropriate cross sections than material-related ones in the two-dimensional model.

The material assignment portion of the 2-D model is shown schematically in Figure 5-2 with the control vanes positioned at 0° (full-in) and Figure 5-3 with the control vanes positioned at 180° (full-out). Table 5-1 relates the spectra developed from the one-dimensional regions of Figure 5-1 to the two-dimensional regions of Figure 5-2 and 5-3. Note that "region" is synonymous with "spectrum" on the right-hand side of Table 5-1. (The purpose in dividing the reflector into five regions in the two-dimensional calculation is to learn if two-dimensional spectral effects are important or if the present WISDM use of spectra from a one-dimensional model is adequate.) The cross sections for the boron-10/aluminum control drum vanes were obtained by weighting the cross sections of the individual materials over a Be-3(l) spectrum, i. e., the spectrum with beryllium material in front of the simulated one-dimensional control vane ("ring vane" in Figure 5-1). The actual atom densities for the constituents of the vane were used with no reduction to "effective" atom density values to compensate for self-shielding in the vane. That is, an atom of boron-10 in the center of the vane is assumed to be as effective neutronically as an atom on the vane surface. This assumption was made for convenience and may not be of serious consequence considering that the inclusion of such a neutronically "black" region in the EXTERMINATOR-2 diffusion theory calculation is itself contradictory to one of the basic assumptions made in the derivation of the diffusion theory equations. * The ultimate test of validity of this assumption (and others made in the calculational model) will be the accuracy with which drum span and other parameters (such as absolute reactivity and fission density values) are calculated.

The next level in detail above spectrum assignment is the designation of mesh line spacing. Mesh line spacing in general is related to the degree of detail required in the final results. In the reflector, it has been found that drum span calculations (and core fission density values) are relatively sensitive to significant departures from the actual control drum vane envelope. **Within the restrictions of problem size and running time, then, the simulation of the shape and position of the control drum vanes is made as accurate as possible. The

*See Reference 14, Section 5.10.

**Reference 10 reports that an appropriate simplified control vane model can allow accurate calculations. The control drum positions that can be simulated reasonably well by this simple model are limited ($0, 90, 180^\circ$), however, and it has been replaced by the more general detailed vane model described here.

simulation used for the NRX-A6 control vanes is shown in Figures 5-4 and 5-5 for the 0- and 180-degree positions, respectively. In terms of cross-sectional area, the mock-up is excellent and differs by only 0.35 percent from the actual control vane. Experience has shown that when the cross-sectional area approaches the actual area, the "rubber band perimeter," a critical item in drum worth calculations¹⁰ is also conserved.

Using this model, then, the eigenvalues calculated by EXTERMINATOR-2 for the situation of drums full-in and full-out are:

Drums Full-in	0.990759
Drums Full-out	1.051068

The difference between these two divided by the effective delayed neutron fraction β of 0.0071 results in an NRX-A6 calculated drum span of 8.5 dollars.

5.2 NRX-A6 DRUM WORTH MEASUREMENTS

An extensive analysis of the NRX-A6 drum span measurements at NRDS and simulated NRX-A6 measurements at WANEF (PAX-E1 reactor)¹² is presented in Reference 11 and will not be repeated here. The best estimate of drum worth inferred from these measurements was 7.6 dollars. This value was an average of that obtained by three different techniques at NRDS as follows:

<u>Type of Measurement</u>	<u>Drum Bank Worth, \$</u>
Individual Drum Integral	7.60
Drum Bank Differential	7.68
Drum Bank Integral	7.54

5.3 EVALUATION OF ACCURACY OF CONTROL DRUM WORTH CALCULATIONS

The error in control drum worth calculations by the WISDM method as found from comparison of the measured and calculated NRX-A6 drum span value is 11 percent. Alternatively, the factor by which calculated drum spans must be multiplied to correct for analytical errors is the ratio of measurement to calculation, $7.6/8.5$, or 0.894. Recent WISDM calculations for reactors similar to those of the R series (that are substantially different from

NRX-A6) confirm this situation. Table 5-2 summarizes the results obtained for the PAX-G0,¹⁵ G1¹⁶ and G2A⁵³ reactors along with NRX-A6. The ratio of the measured to calculated drum span--those values appearing in the right-hand column of the table--is the information that will be used in the following discussion to develop (1) an average value for the experimental/analytical ratio--the average analytical bias--and (2) the variability in this bias. These factors are to be considered in the selection of the safety margin that must be allowed to assure that the drum span of the reactor design being evaluated will satisfy some external requirement without unnecessary penalty to the reactor system design.

To allow use of a common statistical method for the calculation of these two factors, a first step is the determination of the underlying population frequency distribution. Toward this end, Figure 5-6 has been prepared. This figure shows the four data points plotted on a probability scale of Cumulative Frequency. When data satisfying a "normal" distribution are plotted in such a fashion, it will fall on a straight line. The extent to which this is not the case will be an indicator of the non-normality of the sample. For the four data points here, it appears that the non-normality is not great enough to be detected by the available sample. This allows, then, the statistical treatment of the data to proceed using the equations from "normal" distribution theory to calculate the parameters that characterize the population from which the four data points have been taken. These calculations are summarized in Table 5-3. The estimates of the population mean (M) and standard deviation (S) are found to be 0.873 and 0.022, respectively. The lower portion of the table shows the manner in which these two parameters are used to develop a measured to calculated ratio consistent with the desired probability and confidence levels. The ratio to be applied to the calculated drum spans in order to assure that the drum span of the reactor design being evaluated will satisfy the requirement is shown in the last column in this lower portion of the table. It has been calculated as $(M - KS)$ where K is a factor to account for (1) the size of the sample used to estimate the population parameters, (2) the desired confidence level, and (3) the fraction of the population that is to be included. These values are plotted in Figure 5-7.

Figure 5-7 provides the upper limit for the drum correction factor for various proportions P and various confidence levels. The value of the correction factor that is selected in any given case is dictated by the nuclear design criteria.

TABLE 5-1

SPECTRUM LIST FOR REGIONS IN TWO-DIMENSIONAL MODEL

<u>Region in Figures 5-2 and 5-3</u>		<u>Spectrum/Region in Figure 5-1</u>
1	(Fueled Core)	Core-1
2	(Fueled Core)	Core-2
3	(Fueled Core)	Core-3
4	(Fueled Core)	Core-4
5	(Fueled Core)	Core-5
6	(Fueled Core)	Core-6
7	(Fueled Core)	Core-7
8	(Fueled Core)	Core-8
9	(Fueled Core)	Core-9
10	(Filler Strip)	Filler Strip
11	(Lateral Support-1)	Lateral Support-1
12	(Lateral Support-2)	Lateral Support-2
13	(Be Reflector)	Beryllium Reflector-1
14	(Be Reflector)	Beryllium Reflector-2
15	(Be Reflector)	Beryllium Reflector-3(I)
16	(Be Reflector)	Beryllium Reflector-3(O)
22	(Be Reflector)	Beryllium Reflector-3(I)
18	(Plunger)	Beryllium Reflector-1
19	(Spring)	Beryllium Reflector-2
20	(Titanium Tie Bolt)	Beryllium Reflector-3(I)
21	(Control Drum Vane)	Beryllium Reflector-3(I)

TABLE 5-2

COMPARISON OF DRUM SPAN CALCULATED WITH WISDM
TO THAT INFERRED FROM MEASUREMENTS

Measurement Made in	Reactor Simulated in Analytical Model	EXTERMINATOR-2 Calculated Drum Span	Drum Span Inferred from Measurement	Ratio Measured/ Calculated
NRX-A6	NRX-A6	a 8.50	e 7.60	0.894
PAX-G0	Test Fuel Sector in PAX-G0	d 7.04* b (6.94)	b 6.26	0.889
PAX-G1	R-1 Sector in PAX-G1**	d 8.00* c (7.88)	c 6.89	0.861
PAX-G2A	R-1 Sector in PAX-G2A	d 7.84	d 6.64	0.847

*Adjusted for $\beta = 0.0070$; original calculation is shown in parentheses.

**Ta included in calculation in an amount to give a correct reactivity worth.

a. See Section 5.1

b. WANL-TME-1910, "PAX-G0 Control Drum Spans and Reactivity Coefficient Measurement," 1/70.

c. WANL-TME-1916, "PAX-G1 Reflector Interchange, Control Drum Span, and Aluminum Reflector Barrel Experiments," 1/70.

d. WANL-TME-2703, "PAX-G2 Control Drum Spans," 8/70.

e. WANL-TNR-223, "NRX-A6 Reactor Test Analysis Report," 8/68.

TABLE 5-3

DRUM SPAN RATIO STATISTICS

Estimate of Population Parameters

Case, i	R_i	Absolute Value of $(R_i - M)$	Cumulative Frequency
1	0.894	0.021	0.875
2	0.889	0.016	0.625
3	0.861	0.012	0.375
4	0.847	0.026	0.125

M = Estimate of the population Mean = $1/4$ (Sum of the R_i values) = 0.873

S = Estimate of the population standard deviation calculated from the above sample =

$$\sqrt{\frac{\sum (R_i - M)^2}{(4-1)}} = \sqrt{\frac{0.001517}{3}} = 0.022$$

Population Fraction and Confidence Level

Population Fraction	Confidence Level, %	K^*	KS	(M-KS)
0.99	95	7.042	0.155	0.718
0.95	95	5.145	0.113	0.760
0.90	95	4.163	0.092	0.781
0.99	90	5.437	0.120	0.753
0.95	90	3.957	0.087	0.786
0.90	90	3.187	0.070	0.803

* From Table V (Tolerance Factors for Normal Distributions - One Sided), SNPO-C-1, "NERVA Program Structural Design Requirements," 12/19/68.

Figure 5-1. One-Dimensional (Radial) Model of NRX-A6 Reactor Configuration

613446-3B

Figure 5-2. Schematic of Region Assignment for NRX-A6/
Drums Full-In Calculation by Exterminator-2

Figure 5-3. Schematic of Region Assignment for NRX-A6/
Drums Full-Out Calculation by Exterminator-2

Figure 5-4. Detailed Control Vane Material Assignment for NRX-A6/Drums Full-In Calculation

Figure 5-5. Detailed Control Vane Material Assignment
for NRX-A6/Drums Full-Out Calculation

Figure 5-6. Examination of Drum Span Ratio Data For Normality

WITH THE SPECIFIED CONFIDENCE LEVEL, AT LEAST P
PERCENT OF THE CORRECTION FACTORS WILL BE GREATER THAN THAT SHOWN

Figure 5-7. Upper Limit of the Drum Span Correction Factor for Various Proportions (P) and Confidence Levels

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6.0 FISSION DISTRIBUTION CALCULATIONS

This chapter discusses the calculation of fission distribution in the PAX-E1* and FCX* reactors, using the techniques discussed in Section 3.2.3.3. Since both of these reactors were ambient experiments, only the portion of Figure 3-10 associated with the ambient experimental configuration is evaluated here. Thus, the overall evaluation of power distribution calculational accuracy cannot be presented at this time.

6.1 PAX-E1 REACTOR

6.1.1 Calculated Radial Fission Distribution

The PAX-E1 reactor was built to simulate certain nuclear characteristics of the NRX-A6 reactor. Because of limitations on the availability of certain materials and/or components, only a limited fraction (about 90° azimuthally) of the reactor simulated the NRX-A6 in detail. However, the remainder of the reactor did resemble NRX-A6 in a gross manner. There were three objectives in the analysis of PAX-E1: evaluation of the WISDM calculational accuracy for (1) absolute reactivity, (2) relative fission power distribution, and (3) material reactivity coefficients. The selection of the NRX-A6 Quadrant in PAX-E1 with control drums positioned at 105° ** as the model to be simulated in the calculation represents a compromise of a single calculation to serve these needs. (Unfortunately, this selection created the need for a number of additional corrections in the absolute reactivity evaluation. These corrections are discussed in Section 4.1.1.) The calculation was performed according to Figure 3-12 with cross sections supplied by the scheme shown in Figure 3-4. The model was an EXTERMINATOR-2 (r, θ) representation at the reactor midplane. The finite height of the system was simulated by inclusion of a buckling of 0.000450 cm^{-1} .⁺ Much of the discussion in Section 5.1 concerning nuclear modeling of the NRX-A6 reactor for drum span calculations is applicable here because of the similarity between PAX-E1 and NRX-A6.

*See Appendix A for physical reactor characteristics.

**This average drum bank position is that expected to be the average over full power design operation; it is designated "Orifice Angle" (for the NRX-A6 reactor).

⁺ Reference 5, Page E-9.

There are only two significant differences between the WISDM calculations for the two reactors: (1) the detailed composition and (2) the drum position simulated by the EXTERMINATOR-2 model. The atom density composition of the NRX-A6 Quadrant in PAX-E1 is presented in Appendix A. The material assignment for the PAX-E1 EXTERMINATOR-2 calculations appears in Figure 6-1 (analogous to Figures 5-2 and 5-3) and the mesh line arrangement for simulation of the control drum vanes at the 105° position is shown in Figure 6-2. For orientation, an overall reactor cross-sectional view is shown in Figure 6-3.

Following is a brief discussion of the sequence of operations followed in activity blocks 11-2 in Figure 3-10.

The output of the EXTERMINATOR-2 code includes groupwise neutron fluxes as well as an (r, θ) neutron production rate map. The EXTERMINATOR-2 calculation uses as input the groupwise macroscopic neutron production cross section, $\nu \Sigma_f$. To calculate the fission density, the groupwise neutron fluxes are used in PFENCER-1 along with uranium atom densities and microscopic fission cross sections (σ_f) to compute the fission density on a per cc-of-fueled graphite basis. This information is then converted in PFENCER-1 to fissions per gram-of-fuel data for ready comparison with experimental data in PFENCER-2.

Selected results of the PFENCER-1 calculation appear in Tables 6-1, 6-2, and 6-3. Table 6-1 presents the calculated fission per gram values along a radial traverse approximately midway between drums (see position on Figure 6-4). Table 6-2 shows similar results along an azimuthal traverse at a radial position of 43.8 cm, the edge of the core in the homogenized calculations. The traverse shown in Table 6-3 is also azimuthal but is taken at a radial position of about 39 cm. The radial position for these last two traverses (also shown in Figure 6-4) was selected from the analysis which will be reported below as radial positions showing unusually high values for NC_f (the ratio of the experimental to the calculated fission per gram values).

Before going on to an evaluation of the calculations, a comment will be given on the philosophy of the comparisons presented here and certain effects that have been omitted.

The direct calculation described above omits four effects which would be included if it was desired to perform the "best possible" detailed calculation of fission* rate in the ambient reactor. These are:

- (a) radial variation of the axial reflectivity ("ZBAR" effect),
- (b) the perturbing local effect of unfueled and TaC central elements,
- (c) the perturbing local effect of azimuthally asymmetric groups of graphite elements at the edge of the core (see Figure 6-3), and
- (d) the deviation of the actual uranium content of an element from its nominal value.

Only the first one of these four effects can typically result in an overall shift in the gross fission pattern (but see the discussion with regard to TaC element and asymmetric periphery). To establish the magnitude of this effect analytically, an (r,z) calculation would be required. However, past experience has shown that the effect is small,** of the order of less than ± 1 percent for NRX-A reactors. Since the purpose of the analysis here is an evaluation of the overall WISDM scheme, the last three items above were omitted since they typically cause small local effects. There is some question concerning the overall effects of "great" numbers of TaC elements. The ultimate inclusion of this effect in the spectrum and cross-section generation operations of WISDM will depend in part on an evaluation of the PAX-G0²⁵ and PAX-G1²⁶ reactor fission distribution. The reactor analyzed here contained 29 TaC shim elements arranged approximately like 6 radial "fingers," one finger in each sector (see Figure 6-4). While it is not clear that this is a sufficiently small number of shims to be omitted from the spectrum and cross-section calculations, the arrangement of the elements does not appear to warrant homogenization over some central core region. The choice between these equally undesirable alternatives was based, however, on another consideration, that past experience in the calculation of Ta reactivity coefficients had indicated large errors were likely if Ta were included directly. For this reason it was decided to omit the Ta from the

*If heating rate was desired, then gamma heating (Effect A. 4 in Table 3-3) would also have to be added.

**See for example, Figure E. 7. 1 of WANL-TME-1565, "XE-1 Nuclear Subsystem . . . Data Book," 4/67.

fission distribution calculation. The perturbing effect of azimuthally asymmetric groups of graphite elements at the edge of the core has been called "localized." Comparisons shown below indicate that this may, indeed, not be the case. The graphite element arrangement (shaded elements) shown in Figure 6-3 illustrates this point. In the range -10 to +5 degrees, graphite elements (both filler strip and support) are not azimuthally regular. The errors in the peripheral elements in this angular interval are, indeed, relatively higher (as will be shown). While there is probably a perturbing effect due to adjacent unfueled elements, the great numbers of these elements in the 5 to 40-degree angular interval may also have an effect on the gross radial fission pattern. Finally, item (d), a correction for small differences between the actual and nominal uranium content of a given fuel element is self-explanatory.

6.1.2 Measured Radial Fission Distribution

Figures 6-5 and 6-6 show the radial variation of the fission per gram distribution, measured by activation of 1-inch long U-Al wires at the PAX-E1 core midplane in the coolant channels of each fuel element. Figure 6-5 gives the variation from the core centerline to a radius of 35 cm. Figure 6-6 shows the peripheral variations in the 35 cm - 44 cm interval. All the experimental data used are listed in Table 6-4. Only channel 10 data (element center channel) is measured out to a radius of 35.0 cm. For PFENCER-2, these data are then smoothed, thereby eliminating any experimental scatter. In the peripheral elements, fissions per gram measurements are made in seven channels (six "corners" and channel 10). These data are used directly by PFENCER-2.

6.1.3 Comparison of Measurement and Calculation

A comparison of the measured and WISDM calculated fission per gram uranium values in PAX-E1 is summarized in Tables 6-1, 6-2, and 6-3 for the three typical radial traverses illustrated in Figure 6-3. For each of the coolant channels falling on or very near the traverse line, the experimental value (from Table 6-4) is listed in the column headed by the letter A. In the column designated by B, the calculated fission per gram value from PFENCER-1 is listed. The next column shows the quotient (A/B) of these two. These values (designated NCF) represent the factors by which the "Calculated Hot Power Factors"

(Figure 3-10) for each coolant channel are to be multiplied to correct for the errors between measurement and calculation. The NC_i values listed in Tables 6-1 through 6-3 are plotted in Figures 6-7 through 6-9. The figures are fairly self-explanatory. The approximate position of the control vane "shadow" on the core periphery is shown in Figures 6-8 and 6-9 in order to point out that the azimuthal variation in the NC_i values is not simply related to the proximity of a particular core location to a control drum (but is not necessarily unrelated to this). That is, the error in calculated fission rate (NC_i) at the 25° position between 2 of the 3 drums associated with a 60° sector is much different than at -5° , midway between the "other" two control drums. The absence of data in certain azimuthal positions in Figure 6-8 and 6-9 (such as between 3 and 13° in Figure 6-8) is a result of either the presence of unfueled support elements or filler strips in these angular intervals. Figure 6-3 illustrates the situation.

Summarizing, the WISDM calculated fission per gram distribution in PAX-E1 has been found to agree relatively well in central portions of the core but exhibits errors as great as approximately 17 percent at the edge of the core. The average error at the edge of the homogenized circular core (such as that appearing in EXTERMINATOR-2 calculations) is 11 to 12 percent. Recent WISDM calculations for reactors similar to those of the R series (that are substantially different from reactors in the NRX series) confirm this situation. See Reference 25 for detailed results obtained in the PAX-G0 reactor, Reference 26 for PAX-G1 results and Reference 55 for PAX-G2 results. Experience in the PAX-G reactor series is also discussed in Section 6.3.

6.2 FCX REACTOR

6.2.1 Calculated Fission Distribution

The fission distribution in the FCX reactor was calculated by the techniques generally outlined in Figure 3-11 and the ambient configuration portion of Figure 3-10 with some exceptions. The principal exception was the substitution of ANISN and DOT in the neutron transport solution (Block 11-2) for EXTERMINATOR-2 in order to qualify these two codes for the "Conceptual Design Scoping" type of activity outlined in Figure 3-6. Also, the

calculations explored the possible advantage of higher order scattering approximations than WISDM's P_0S_4 and the effect of the use of 52 groups in ANISN instead of WISDM's 16 group scheme (Table 3-2).

After development of a converged neutron flux and current spectrum as detailed in Appendix B, cross sections for FCX were generated in 52 and 16 neutron energy groups for ANISN and 16 groups for DOT. Although fuel elements of generally the same loading were used in the fueled portion of the reactor, the core was divided radially into three regions to allow for the effects of spectral changes in the core near the graphite reflector. Similarly, the reflector was subdivided into three regions to account for spectral changes, especially near its interface with the core. The scattering approximation used in the cross sections for the radial ANISN calculations was transport corrected (P_0) and (P_3) data. In the two-dimensional DOT calculation, transport corrected (P_0) cross sections were used with the upscatter components removed to shorten computer running times. The upscatter component removal process is shown by the flow chart in Figure 3-4.

The cross sections developed in this manner were used in ANISN and DOT to compute the radial variation of the fission density across the reactor at the midplane--axial--station and in DOT to compute the changes in axial variation that occur with radial position. The outcome of the radial calculations at midplane is shown in Figure 6-10. The results have been presented on an absolute basis in order that they may be compared directly to the experimental data. To do this, the macroscopic fission cross sections for the comparisons here were based on the atom density of uranium in the measurement foil itself. (The eigenvalue ANISN and DOT calculations used the reactor fuel fission cross sections.) To account for the ANISN and DOT normalization and to place the results on an absolute basis, the following conversion factor was applied to the product of macroscopic foil fission cross section and calculated neutron flux*:

*Power during irradiation has been omitted from the numerator of this equation because results are desired on a "per watt" basis.

$$\frac{K \nu}{kDH} \left(\frac{A_{\text{mid}}}{A_{\text{ave}}} \right)$$

where K is the number of fissions per watt hour

$$= (3.24 \times 10^{10})(3.6 \times 10^3) = 11.664 \times 10^{13} \text{ (based on } 192.9 \text{ MeV/fission),}$$

ν is the average number of neutrons per fission,

k is the eigenvalue of the ANISN or DOT calculation,

D is the density of uranium-235 in the foils (gm/cc),

H is the height of the reactor assembly (cm) and,

$A_{\text{mid}}/A_{\text{ave}}$ is the average value of the ratio of the measured foil activation at the midplane to the axial average.

The last two factors, H and the A ratio, are only used for normalization of the radial ANISN calculation and are not needed for the (r,z) DOT calculation. The result of the above computations is the number of fissions per gram of uranium in the foil per hour of irradiation per watt of power (fission/gm U/watt hour).

Referring to Figure 6-10, differences between the 16 group DOT and 52 group ANISN* fission density profiles computed with the transport corrected (P_0) scattering approximation are not significant. This is not the case, however, for the 52 group P_3 and transport corrected P_0 calculations. The P_3 calculations do result in a higher fission density in the reflector by about 7 percent and a lower fission density in general in the fueled core.

Results of the DOT (r,z) calculations, at several radial positions (normalized as noted above), are shown in Figures 6-11 and 6-12. These two graphs show the variation in axial fissions per gram of foil per watt hour of irradiation along traverses in the core (Figure 6-11) and the graphite reflector (Figure 6-12).

6.2.2 Measured Fission Distribution

The measured fission density distribution was obtained from activation of fully enriched 1 mil uranium foils of 0.25 inch diameter in a series of experiments (FCX-B)³⁶

*The 16 group transport corrected P_0 calculations with ANISN have not been shown, but results are very similar to the 52 group analysis.

conducted about a year after the first group of measurements (FCX-A)³⁴ following initial criticality of FCX. The foils were placed in the reactor in a radial pattern at midplane and an axial pattern at a number of radial positions. After irradiation for 60 minutes of a nominal power level of 80 watts, the foils were beta counted and appropriate corrections were applied.³⁶ The results were placed on an absolute basis through the use of gold wires that were simultaneously irradiated with the foils specifically for intercalibration with a separate experiment for which the absolute power had been determined radio-chemically. The experimental results expressed in terms of fissions per gram uranium foil per watt hour of irradiation are plotted in Figures 6-10, 11, and 12 for comparison with the calculation.

Note that in these and other foil irradiations (discussed in Chapter 8) where the reactor was operated for an extended time period at equilibrium, it was necessary to partially insert the control rods into the core a distance of 13 cm. This was not included in the calculations, but must be considered in the accuracy evaluation.

6.2.3 Comparison of the Calculated and the Measured Fission Density Distribution in FCX

A comparison of the measured and calculated fission densities is presented in Figures 6-10, 6-11, and 6-12. As is evident from the curves, the agreement between the calculation and the measured values is generally good except in the reflector and particularly at the outer edge of the reactor. This is shown by the radial distribution in Figure 6-10 and by the more detailed axial distributions in Figure 6-12. The error in the calculated distribution at the radial position of the measurement nearest the edge of the reactor, 62.95 cm (the edge of homogenized reactor is at a radial position 64.3751 cm), is about 5 percent at midplane for effectively all of the calculated distributions shown. This error increases to over 50 percent at the bottom of the reactor and to about 30 percent at the top (Figure 6-12). It is believed that this higher error at axial positions near to the upper and lower surfaces of the reactor is a result of neutron reflection that was not simulated in the (r,z) DOT analytical model (see Appendix A). Another factor not included in the base model, which has been evaluated from a fission density standpoint, is the effect of material removal to accommodate the seven control rod/fuel follower aluminum guide tubes present in the reactor (see Section 4.2.2.3). A repeat of the base ANISN calculation with the material removed showed a

5 percent increase in calculated fission density on the midplane at the edge of the reactor. This results in about a 30 percent decrease in the error of 15 percent at this reactor location.

Another factor related to control rod effects is the fact that the control rods were inserted into the reactor about 13 cm (about 10 percent of the reactor height) in order to maintain an equilibrium power level during the foil irradiation. This effect has not been analyzed in detail but would be expected qualitatively to cause an increase in the local measured fission density near the upper surface of the core because of the increase in fast neutron slowing down caused by the control rod polyethylene. Note that this is the opposite of what might "normally" be expected to occur due to a control rod insertion accompanied by a depressed total neutron flux. Obviously, at some point of insertion, the situation does revert to "normal." The comparison of calculation and experimental data shown in Figure 6-11 tends to confirm this qualitative assessment. That is, the error in the calculation is greater at the axially uppermost measured point on the 5.4 cm radius curve than on the 32.24 cm radius curve. The data at 5.4 cm was one fuel element removed from the outside surface of the central control rod assembly, while the 32.24 cm position is well outside the outer ring of six control rods (see Appendix A).

Figure 6-10 shows that an improvement in calculated fission density in mid-reflector was possible using a P_3 scattering approximation. The improvement was about 7 percent and brought the calculated distribution into excellent agreement with the measurement in this position.

6.3 EVALUATION OF ACCURACY OF FISSION DENSITY CALCULATIONS

Fission density calculations in the NRX/R type reactors are particularly in error in the vicinity of the core periphery. Detailed analysis of the PAX-E1 reactor revealed maximum errors of 17 percent and average errors of 11 to 12 percent in this area. The situation was confirmed in the PAX-G0, G1, and G2 calculations^{25,26,55} where maximum errors of over 25 percent were observed on the core periphery. In contrast to this, the FCX reactor fission density calculations were accurate to approximately 5 percent at all mid-axial core locations. (In this reactor errors increased at the top and bottom of the assembly near the reactor edge

where the analytical model is known to be more deficient with regard to lack of neutron reflection and control rod simulation.) From the differences in analytical error in PAX-E1 and FCX, it is apparent that an important weakness exists in some aspect of the PAX-E1 core periphery/reflector simulation and, very likely, the control vane mock-up. Although the technique for spatial modeling can be improved and some progress has been made in this area,³⁷ it appears that the control vane cross-section development scheme for WISDM (see Section 5.1) is an area where improvements would be of significant value. Two approaches are open in this regard other than the use of effective cross-sections tailored to give the correct answer at a specific drum position for a specific reactor design. First, it may be possible to avoid the use of tailored cross sections by developing control vane cross sections from an analytical model especially designed to account for flux depression. This possibility has been explored briefly.³⁸ The second approach is the use of the DOT-IIW code for solution of the neutron transport problem by transport theory instead of the diffusion theory of the EXTERMINATOR-2 solution.

In the discussion that follows, the evaluation of the accuracy of fission density calculations is extended by a statistical examination of PAX-G0, G1, and G2 experience with WISDM in order that a design or safety margin analysis typical of that illustrated in Figure 1-1 can be carried out. A sample of the overall radial variation in NC_i values in these three reactors appears in Figure 6-13 for PAX-G2A.⁵⁵ The NC_i values obtained at various core locations in the three reactors are tabulated in Tables 6-5 through 6-8. The first of these, Table 6-5, shows the NC_i values in Channel 10 of each fuel element on the core periphery. The last two columns in the table give the mean NC_i and the estimate of the population standard deviation as computed from the three values at each of the core locations tabulated. Note that the standard deviations vary substantially in the table from high value of 6.35 percent to a low of 0.45 percent. This appears to be readily explainable in terms of the position of the various fuel elements in the test sector as illustrated in Figure 6-14. Elements like P3D (with 6.35% standard deviation) which are near the edge of the sector azimuthally are affected by the non-sector type reflector experimentally but not analytically. Thus, higher NC_i values are to be expected in these locations. Since the reflector in PAX-G0

was uniform azimuthally, the NC_i at this location in PAX-G0 is lower (1.16791) relative to PAX-G1/G2 (1.28716 and 1.26541) and the standard deviation is therefore relatively higher than near the center of the test sector. For example, in element P5F the standard deviation is 1.07 percent. Since this core location gives NC_i variations which are believed to be typical of what might be expected in a situation where an experimental configuration can be simulated in a relatively accurate manner analytically, P5F has been used for a detailed statistical analysis. (This is also shown particularly by Table 6-6 where the elements selected fall along a line through the Drum B centerline. The standard deviations are in every case less than 1 percent.) The statistical analysis is summarized in Table 6-9 for core location P5F10. The upper portion of this table shows the development of two appropriate population parameters, mean M of 1.184 and standard deviation S of 0.0107. The lower part of the table shows the manner in which these two parameters are used to develop an NC_i value for location P5F10 that will be consistent with the confidence level and the desired probability of it not being exceeded on a subsequent reactor ("Nuclear Design Criteria" in Figure 1-1). These NC_i values appear in the last column. The chosen value is to be applied as described in Figure 3-12 to the calculated fission density at operating conditions in order to assure that the power generated in this fuel element will not exceed that for which coolant has been allocated. The values in this last column have been calculated as $M + KS$ where K is a factor⁵⁴ to account for (1) the size of the sample used to estimate the population parameters, (2) the desired confidence level and (3) the fraction of the population that is to be included. The values are plotted in Figure 6-15. Note that the use of $M + KS$ will give a correction that is higher than the mean NC_i and result in selectively lower element fuel loadings. With Nuclear Design criteria at "90, 90" for example, where KS is 0.0456, the fuel loading for P5F will be reduced by 4 percent.

In summary, Figure 6-15 represents a typical design or safety margin analysis and provides the lower limit for the NC_i value (at P5F10) for various proportions P and various confidence levels. The value of the NC_i correction factor that is selected for any given location is dictated by the nuclear design criteria.

TABLE 6-1

COMPARISON OF CALCULATED AND MEASURED FISSION PER GRAM
FOR A TRAVERSE BETWEEN DRUMS
(BETWEEN CLUSTERS J3 AND J4)

LOCATION	RADIAL POSITION, CM	A EXPERIMENTAL FISSION/GM	B CALCULATED FISSION/GM	A/B (NC_i)
O 0 F 10	1.9	1.0000	0.9993	1.0007
A 1 G 10	3.6	0.9980	0.9967	1.0013
A 1 E 10	6.4	0.9870	0.9848	1.0022
B 2 C 10	9.2	0.9760	0.9754	1.0006
C 2 B 10	11.6	0.9480	0.9510	0.9968
C 2 D 10	15.2	0.9145	0.9177	0.9965
D 2 G 10	16.4	0.8960	0.8938	1.0025
D 2 E 10	20.0	0.8600	0.8529	1.0083
F 2 F 10	22.4	0.8210	0.8093	1.0145
F 3 B 10	25.4	0.8050	0.7916	1.0169
F 3 D 10	28.1	0.7895	0.7696	1.0259
G 3 B 10	29.5	0.7940	0.7713	1.0294
G 3 E 10	33.7	0.8725	0.8421	1.0361
H 3 G 10	35.2	0.9434	0.9337	1.0104
H 3 F 10	37.6	1.1254	1.0884	1.0340
H 3 E 10	37.9	1.3791	1.3418	1.0278
J 4 C 3	39.1	1.6001	1.5185	1.0537
J 4 C 10	40.0	1.8321	1.7842	1.0268
J 4 C 17	40.8	2.0999	2.0699	1.0145
J 4 D 1	41.2	2.2279	2.2005	1.0125
J 4 D 8	42.1	2.6021	2.5964	1.0022
J 4 M 3	42.4	2.8139	2.7471	1.0243
J 4 M 19	43.8	3.6765	3.4817	1.0559

TABLE 6-2

COMPARISON OF CALCULATED AND MEASURED FISSIONS PER GRAM
FOR AN AZIMUTHAL TRAVERSE AT THE EDGE OF THE HOMOGENIZED CORE

LOCATION	PFENCER ANGULAR POSN, DEG.	A EXPERIMENTAL FISS/GM	B CALCULATED FISS/GM	A/B (NC _i)
J 3 H 8	-10.1	3.9201	3.5128	1.1159
J 3 H 19	-8.1	3.7214	3.4657	1.0738
J 3 N 8	-7.6	3.7032	3.4564	1.0714
J 3 N 19	-5.6	3.6316	3.4313	1.0584
J 3 T 8	-5.1	3.6034	3.4275	1.0513
J 3 T 19	-3.1	3.5915	3.4229	1.0493
J 3 U 8	-2.6	3.6336	3.4243	1.0611
J 3 U 19	-0.6	3.5856	3.4398	1.0424
J 4 M 19	1.9	3.6765	3.4818	1.0559
J 4 H 8	2.4	3.6788	3.5000	1.0511
J 5 J 19	13.4	3.9177	3.4553	1.1338
J 7 H 17	23.2	4.2332	3.6398	1.1630
J 7 H 19	24.3	4.2048	3.6202	1.1615
J 7 K 17	27.5	4.2324	3.6121	1.1717
J 7 K 19	28.6	4.2177	3.6255	1.1633
J 9 H 17	38.4	3.8954	3.4338	1.1344

TABLE 6-3

COMPARISON OF CALCULATED AND MEASURED FISSIONS PER GRAM
FOR AN AZIMUTHAL TRAVERSE AT A RADIAL POSITION OF 39 CM

LOCATION	PFENCER ANGULAR POSN, DEG.	A EXPERIMENTAL FISS/GM	B CALCULATED FISS/GM	A/B (NC _i)
J 3 C 17	-11.1	1.6809	1.5678	1.0721
J 3 D 3	-9.7	1.6765	1.5222	1.1014
J 3 E 3	-6.9	1.6234	1.4945	1.0862
J 3 R 3	-4.1	1.5332	1.4849	1.0325
J 3 W 3	-1.3	1.5535	1.4926	1.0408
J 4 C 3	1.5	1.6001	1.5185	1.0537
J 4 B 19	4.1	1.6318	1.4813	1.1016
J 4 G 8	4.6	1.6310	1.4924	1.0929
J 4 G 19	6.8	1.6773	1.5468	1.0844
J 5 C 10	8.6	1.6515	1.4992	1.1016
J 5 G 17	13.0	1.6962	1.4978	1.1325
J 5 F 3	14.4	1.7311	1.5459	1.1198
J 6 C 8	14.8	1.6671	1.4876	1.1207
J 6 G 19	21.7	1.7141	1.5390	1.1138
J 7 C 12	24.8	1.8010	1.5666	1.1496
J 7 G 8	27.0	1.8100	1.5657	1.1560
H 7 E 17	30.1	1.7302	1.5361	1.1264
J 8 B 3	33.9	1.6300	1.4584	1.1177
J 9 D 12	37.0	1.6544	1.4849	1.1141
J 9 E 1	37.4	1.7395	1.5430	1.1273
J 9 F 3	41.4	1.6125	1.4645	1.1011
J 9 R 10	43.2	1.6461	1.5011	1.0966
J 9 W 17	46.1(-13.9)	1.8203	1.6647	1.0935
J 9 V 12	48.7(-11.3)	1.7806	1.6962	1.0498

TABLE 6-4

PAX-E1 FISSION/GRAM MEASUREMENTS

2 A 1 F 10	.9940	2 D 2 E 10	.8600	2 F 1 D 10	.8340	2 G 1 B 03	.8565
2 A 1 G 10	.9980	2 D 3 F 10	.8790	2 F 1 E 10	.8235	2 G 1 B 08	.9590
2 A 1 H 10	.9975	2 D 3 G 10	.9000	2 F 2 F 10	.7895	2 G 1 B 10	.9205
2 A 1 C 10	.9915	2 D 3 B 10	.9070	2 F 2 G 10	.7952	2 G 1 B 12	.8830
2 A 1 D 10	.9855	2 D 3 C 10	.8920	2 F 2 B 10	.7950	2 G 1 B 17	.9999
2 A 1 E 10	.9870	2 D 3 D 10	.8725	2 F 2 C 10	.7895	2 G 1 B 19	.9620
2 H 1 F 10	.9630	2 D 3 E 10	.8655	2 F 2 D 10	.7940	2 G 1 C 01	1.0613
2 H 1 G 10	.9760	2 D 4 F 10	.8655	2 F 2 E 10	.7930	2 G 1 C 03	.9929
2 H 1 B 10	.9725	2 D 4 G 10	.8865	2 F 3 F 10	.7920	2 G 1 C 08	1.1606
2 H 1 C 10	.9570	2 D 4 B 10	.9000	2 F 3 G 10	.8025	2 G 1 C 10	1.0755
2 H 1 D 10	.9450	2 D 4 C 10	.8900	2 F 3 B 10	.8050	2 G 1 C 12	1.0428
2 H 1 E 10	.9480	2 D 4 D 10	.8690	2 F 3 C 10	.7950	2 G 1 C 17	1.1818
2 H 2 F 10	.9660	2 D 4 E 10	.8575	2 F 3 D 10	.7895	2 G 1 C 19	1.1193
2 H 2 G 10	.9800	2 E 1 F 10	.8050	2 F 3 E 10	.7893	2 G 1 D 01	1.2237
2 H 2 B 10	.9855	2 E 1 G 10	.8185	2 F 4 F 10	.7930	2 G 1 D 03	1.1707
2 H 2 C 10	.9760	2 E 1 B 10	.8150	2 F 4 G 10	.8045	2 G 1 D 08	1.3650
2 H 2 D 10	.9615	2 E 1 C 10	.8000	2 F 4 B 10	.8100	2 G 1 D 10	1.2639
2 H 2 E 10	.9570	2 E 1 D 10	.7920	2 F 4 C 10	.8000	2 G 1 D 12	1.2197
2 C 1 F 10	.9145	2 E 1 E 10	.7940	2 F 4 D 10	.7905	2 G 1 D 17	1.4203
2 C 1 G 10	.9300	2 E 2 F 10	.8210	2 F 4 E 10	.7895	2 G 1 D 19	1.3136
2 C 1 B 10	.9265	2 E 2 G 10	.8405	2 F 5 F 10	.7900	2 G 1 E 01	1.2093
2 C 1 C 10	.9070	2 E 2 B 10	.8405	2 F 5 G 10	.7785	2 G 1 E 03	1.1265
2 C 1 D 10	.8920	2 E 2 C 10	.8210	2 F 5 B 10	.8065	2 G 1 E 08	1.3029
2 C 1 E 10	.8960	2 E 2 D 10	.8065	2 F 5 C 10	.7985	2 G 1 E 10	1.2175
2 C 2 F 10	.9265	2 E 2 E 10	.8065	2 F 5 D 10	.7900	2 G 1 E 12	1.1356
2 C 2 G 10	.9450	2 E 3 F 10	.8290	2 F 5 E 10	.7891	2 G 1 E 17	1.3645
2 C 2 H 10	.9480	2 E 3 G 10	.8500	2 F 6 F 10	.7900	2 G 1 E 19	1.2685
2 C 2 C 10	.9320	2 E 3 B 10	.8565	2 F 6 G 10	.7905	2 G 2 F 10	.8645
2 C 2 D 10	.9145	2 E 3 C 10	.8380	2 F 6 B 10	.7945	2 G 2 G 10	.8160
2 C 2 E 10	.9110	2 E 3 D 10	.8185	2 F 6 C 10	.7930	2 G 2 B 10	.8190
2 C 3 F 10	.9200	2 E 3 E 10	.8150	2 F 6 D 10	.7890	2 G 2 C 10	.8725
2 C 3 G 10	.9390	2 E 4 F 10	.8245	2 F 6 E 10	.7945	2 G 2 D 10	.9660
2 C 3 B 10	.9500	2 E 4 G 10	.8450	2 G 1 F 01	.9916	2 G 2 E 10	.9575
2 C 3 C 10	.9390	2 E 4 B 10	.8575	2 G 1 F 03	.9290	2 G 3 F 10	.8235
2 C 3 D 10	.9200	2 E 4 C 10	.8430	2 G 1 F 08	1.0771	2 G 3 G 10	.7945
2 C 3 E 10	.9100	2 E 4 D 10	.8220	2 G 1 F 10	.9944	2 G 3 B 10	.7940
2 D 1 F 10	.8600	2 E 4 E 10	.8145	2 G 1 F 12	.9445	2 G 3 C 10	.8160
2 D 1 G 10	.8765	2 E 5 F 10	.8100	2 G 1 F 17	1.0877	2 G 3 D 10	.8675
2 D 1 B 10	.8725	2 E 5 G 10	.8290	2 G 1 F 19	1.0113	2 G 3 E 10	.8725
2 D 1 C 10	.8500	2 E 5 B 10	.8430	2 G 1 G 01	.8767	2 G 4 F 10	.8120
2 D 1 D 10	.8345	2 E 5 C 10	.8345	2 G 1 G 03	.8568	2 G 4 G 10	.7900
2 D 1 E 10	.8380	2 E 5 D 10	.8145	2 G 1 G 08	.9602	2 G 4 B 10	.7891
2 D 2 F 10	.8765	2 E 5 E 10	.8045	2 G 1 G 10	.8902	2 G 4 C 10	.7975
2 D 2 G 10	.8960	2 F 1 F 10	.7945	2 G 1 G 12	.8716	2 G 4 D 10	.8340
2 D 2 B 10	.8975	2 F 1 G 10	.7893	2 G 1 G 17	.9730	2 G 4 E 10	.8465
2 D 2 C 10	.8790	2 F 1 B 10	.7891	2 G 1 G 19	.8950	2 G 5 F 10	.8190
2 D 2 D 10	.8605	2 F 1 C 10	.8030	2 G 1 B 01	.9106	2 G 5 G 10	.7920

2 G 5 B 10	.7890
2 G 5 C 10	.7945
2 G 5 D 10	.8260
2 G 5 E 10	.8490
2 G 6 F 10	.8490
2 G 6 G 10	.8030
2 G 6 H 10	.7920
2 G 6 C 10	.8010
2 G 6 D 10	.8425
2 G 6 E 10	.8810
2 G 7 F 10	.9360
2 G 7 G 10	.8465
2 G 7 B 10	.8120
2 G 7 C 10	.8260
2 G 7 D 10	.8960
2 G 7 E 10	.9800
2 H 3 F 01	1.0970
2 H 3 F 03	1.0220
2 H 3 F 08	1.2104
2 H 3 F 10	1.1254
2 H 3 F 12	1.0997
2 H 3 F 17	1.2673
2 H 3 F 19	1.2074
2 H 3 G 01	.9083
2 H 3 G 03	.8971
2 H 3 G 08	1.0136
2 H 3 G 10	.9434
2 H 3 G 12	.9152
2 H 3 G 17	1.0551
2 H 3 G 19	.9922
2 H 3 B 01	.9142
2 H 3 B 03	.9009
2 H 3 B 08	.9823
2 H 3 B 10	.9546
2 H 3 B 12	.9172
2 H 3 B 17	1.0457
2 H 3 B 19	1.0144
2 H 3 C 01	1.0711
2 H 3 C 03	1.0012
2 H 3 C 08	1.1740
2 H 3 C 10	1.1117
2 H 3 C 12	1.0876
2 H 3 C 17	1.2470
2 H 3 C 19	1.1997
2 H 3 D 01	1.2844
2 H 3 D 03	1.2477
2 H 3 D 08	1.4655

2 H 3 D 10	1.3606
2 H 3 D 12	1.3212
2 H 3 D 17	1.5728
2 H 3 D 19	1.4647
2 H 3 E 01	1.3251
2 H 3 E 03	1.2543
2 H 3 E 08	1.4617
2 H 3 E 10	1.3791
2 H 3 E 12	1.3139
2 H 3 E 17	1.5780
2 H 3 E 19	1.4990
2 H 4 F 01	1.0283
2 H 4 F 03	.9834
2 H 4 F 08	1.1001
2 H 4 F 10	1.0607
2 H 4 F 12	1.0236
2 H 4 F 17	1.1613
2 H 4 F 19	1.1317
2 H 4 G 01	.8616
2 H 4 G 03	.8523
2 H 4 G 08	.9446
2 H 4 G 10	.9032
2 H 4 G 12	.8729
2 H 4 G 17	.9962
2 H 4 G 19	.9419
2 H 4 B 01	.8501
2 H 4 B 03	.8457
2 H 4 B 08	.9006
2 H 4 B 10	.8754
2 H 4 B 12	.8858
2 H 4 B 17	.9559
2 H 4 B 19	.9255
2 H 4 C 01	.9291
2 H 4 C 03	.9076
2 H 4 C 08	.9837
2 H 4 C 10	.9758
2 H 4 C 12	.9744
2 H 4 C 17	1.0366
2 H 4 C 19	1.0411
2 H 4 D 01	1.0939
2 H 4 D 03	1.0847
2 H 4 D 08	1.2169
2 H 4 D 10	1.1645
2 H 4 D 12	1.1547
2 H 4 D 17	1.3158
2 H 4 D 19	1.2712
2 H 4 E 01	1.1762

TABLE 6-4

2 H 4 E 03	1.1586
2 H 4 E 08	1.2918
2 H 4 E 10	1.2565
2 H 4 E 12	1.2220
2 H 4 E 17	1.3988
2 H 4 E 19	1.3435
2 H 5 F 01	1.0092
2 H 5 F 03	.9738
2 H 5 F 08	1.0857
2 H 5 F 10	1.0527
2 H 5 F 12	1.0235
2 H 5 F 17	1.1459
2 H 5 F 19	1.1272
2 H 5 G 01	.8695
2 H 5 G 03	.8614
2 H 5 G 08	.9427
2 H 5 G 10	.8940
2 H 5 G 12	.8869
2 H 5 G 17	.9678
2 H 5 G 19	.9280
2 H 5 H 01	.8403
2 H 5 H 03	.8280
2 H 5 H 08	.8641
2 H 5 H 10	.8820
2 H 5 H 12	.8579
2 H 5 H 17	.9219
2 H 5 H 19	.9360
2 H 5 C 01	.8939
2 H 5 C 03	.8818
2 H 5 C 08	.9254
2 H 5 C 10	.9365
2 H 5 C 12	.9526
2 H 5 C 17	1.0023
2 H 5 C 19	1.0115
2 H 5 D 01	1.0325
2 H 5 D 03	1.0545
2 H 5 D 08	1.1379
2 H 5 D 10	1.1069
2 H 5 D 12	1.1358
2 H 5 D 17	1.2433
2 H 5 D 19	1.2067
2 H 5 E 01	1.1294
2 H 5 E 03	1.1021
2 H 5 E 08	1.2239
2 H 5 E 10	1.1933
2 H 5 E 12	1.1827
2 H 5 E 17	1.3457

2 H 5 E 19	1.3300
2 H 6 F 01	1.0767
2 H 6 F 03	1.0584
2 H 6 F 08	1.1724
2 H 6 F 10	1.1391
2 H 6 F 12	1.1509
2 H 6 F 17	1.2599
2 H 6 F 19	1.2487
2 H 6 G 01	.9182
2 H 6 G 03	.8940
2 H 6 G 08	.9675
2 H 6 G 10	.9493
2 H 6 G 12	.9524
2 H 6 G 17	1.0308
2 H 6 G 19	1.0108
2 H 6 H 01	.8580
2 H 6 H 03	.8529
2 H 6 H 08	.9014
2 H 6 H 10	.8974
2 H 6 H 12	.8855
2 H 6 H 17	.9649
2 H 6 H 19	.9706
2 H 6 C 01	.9076
2 H 6 C 03	.8942
2 H 6 C 08	.9576
2 H 6 C 10	.9480
2 H 6 C 12	.9802
2 H 6 C 17	1.0291
2 H 6 C 19	1.0438
2 H 6 D 01	1.0591
2 H 6 D 03	1.0902
2 H 6 D 08	1.1537
2 H 6 D 10	1.1601
2 H 6 D 12	1.1687
2 H 6 D 17	1.2732
2 H 6 D 19	1.2586
2 H 6 E 01	1.2106
2 H 6 E 3	1.2113
2 H 6 E 8	1.3192
2 H 6 E 10	1.3033
2 H 6 E 12	1.3109
2 H 6 E 17	1.4570
2 H 6 E 19	1.4517
2 H 7 F 01	1.2544
2 H 7 F 03	1.2359
2 H 7 F 08	1.3818
2 H 7 F 10	1.3365

2	H	7	F	12	1.3774
2	H	7	F	17	1.5119
2	H	7	F	19	1.5306
2	H	7	G	01	.9957
2	H	7	G	03	1.0043
2	H	7	G	08	1.0970
2	H	7	G	10	1.0669
2	H	7	G	12	1.0952
2	H	7	G	17	1.1794
2	H	7	G	19	1.1835
2	H	7	B	01	.9048
2	H	7	B	03	.9157
2	H	7	B	08	.9546
2	H	7	B	10	.9696
2	H	7	B	12	.9616
2	H	7	B	17	1.0581
2	H	7	B	19	1.0763
2	H	7	C	01	.9572
2	H	7	C	03	.9772
2	H	7	C	08	1.0527
2	H	7	C	10	1.0418
2	H	7	C	12	1.0858
2	H	7	C	17	1.1339
2	H	7	C	19	1.1638
2	H	7	D	01	1.1866
2	H	7	D	03	1.2283
2	H	7	D	08	1.3011
2	H	7	D	10	1.3087
2	H	7	D	12	1.3498
2	H	7	D	17	1.4539
2	H	7	D	19	1.4812
2	H	7	E	01	1.3999
2	H	7	E	03	1.4014
2	H	7	E	08	1.5205
2	H	7	E	10	1.5364
2	H	7	E	12	1.5464
2	H	7	E	17	1.7302
2	H	7	E	19	1.7662
2	J	3	F	01	1.3437
2	J	3	F	03	1.2272
2	J	3	F	08	1.5138
2	J	3	F	10	1.3840
2	J	3	F	12	1.2812
2	J	3	F	17	1.5760
2	J	3	F	19	1.4401
2	J	3	G	01	1.1232
2	J	3	G	03	1.0368

2	J	3	G	08	1.2429
2	J	3	G	10	1.1460
2	J	3	G	12	1.0755
2	J	3	G	17	1.3146
2	J	3	G	19	1.1888
2	J	3	B	01	1.1326
2	J	3	B	03	1.0552
2	J	3	B	08	1.2510
2	J	3	B	10	1.1966
2	J	3	B	12	1.1043
2	J	3	B	17	1.3430
2	J	3	B	19	1.2695
2	J	3	C	01	1.3931
2	J	3	C	03	1.3224
2	J	3	C	08	1.5906
2	J	3	C	10	1.4874
2	J	3	C	12	1.4224
2	J	3	C	17	1.6809
2	J	3	C	19	1.5828
2	J	3	D	01	1.7886
2	J	3	D	03	1.6765
2	J	3	D	08	2.0628
2	J	3	D	10	1.9164
2	J	3	D	12	1.7698
2	J	3	D	17	2.2335
2	J	3	D	19	2.0679
2	J	3	E	01	1.7811
2	J	3	E	03	1.6234
2	J	3	E	08	2.0467
2	J	3	E	10	1.8673
2	J	3	E	12	1.6771
2	J	3	E	17	2.1755
2	J	3	E	19	1.9570
2	J	3	H	01	3.3024
2	J	3	H	03	3.0261
2	J	3	H	08	3.9201
2	J	3	H	10	3.4826
2	J	3	H	12	3.2184
2	J	3	H	17	4.1561
2	J	3	H	19	3.7214
2	J	3	J	01	2.4630
2	J	3	J	03	2.2257
2	J	3	J	08	2.9608
2	J	3	J	10	2.6322
2	J	3	J	12	2.4166
2	J	3	J	17	3.1346
2	J	3	J	19	2.8523

TABLE 6-4

2	J	3	K	01	1.0
2	J	3	K	03	1.0
2	J	3	K	08	1.0
2	J	3	K	10	1.0
2	J	3	K	12	1.0
2	J	3	N	01	3.1699
2	J	3	N	03	2.8788
2	J	3	N	08	3.7032
2	J	3	N	10	3.3440
2	J	3	N	12	3.0324
2	J	3	N	17	3.9567
2	J	3	N	19	3.6316
2	J	3	V	1	2.2605
2	J	3	V	3	2.0531
2	J	3	V	8	2.6253
2	J	3	V	10	2.4037
2	J	3	V	12	2.2347
2	J	3	V	17	2.7909
2	J	3	V	19	2.5883
2	J	3	W	1	1.6738
2	J	3	W	3	1.5535
2	J	3	W	8	1.9716
2	J	3	W	10	1.8090
2	J	3	W	12	1.6784
2	J	3	W	17	2.1083
2	J	3	W	19	1.9309
2	J	3	R	01	1.6743
2	J	3	R	03	1.5332
2	J	3	R	08	1.9659
2	J	3	R	10	1.8064
2	J	3	R	12	1.6562
2	J	3	R	17	2.1351
2	J	3	R	19	1.9726
2	J	3	S	01	2.2796
2	J	3	S	03	2.0906
2	J	3	S	08	2.6644
2	J	3	S	10	2.4223
2	J	3	S	12	2.2738
2	J	3	S	17	2.8173
2	J	3	S	19	2.6261
2	J	3	T	01	3.0355
2	J	3	T	03	2.8602
2	J	3	T	08	3.6034
2	J	3	T	10	3.2989
2	J	3	T	12	3.0326
2	J	3	T	17	3.9165
2	J	3	T	19	3.5915

2	J	3	U	1	3.0575
2	J	3	U	3	2.7917
2	J	3	U	8	3.6336
2	J	3	U	10	3.2879
2	J	3	U	12	2.9968
2	J	3	U	17	3.9351
2	J	3	U	19	3.5856
2	J	4	F	01	1.9463
2	J	4	F	03	1.7952
2	J	4	F	08	2.3033
2	J	4	F	10	2.1103
2	J	4	F	12	1.9712
2	J	4	F	17	2.4733
2	J	4	F	19	2.2915
2	J	4	G	01	1.3929
2	J	4	G	03	1.3290
2	J	4	G	08	1.6310
2	J	4	G	10	1.5320
2	J	4	G	12	1.4588
2	J	4	G	17	1.8089
2	J	4	G	19	1.6773
2	J	4	B	01	1.3308
2	J	4	B	03	1.2669
2	J	4	B	08	1.5160
2	J	4	B	10	1.4669
2	J	4	B	12	1.3929
2	J	4	B	17	1.6988
2	J	4	B	19	1.6318
2	J	4	C	01	1.6762
2	J	4	C	03	1.6001
2	J	4	C	08	1.9254
2	J	4	C	10	1.8321
2	J	4	C	12	1.7848
2	J	4	C	17	2.0999
2	J	4	C	19	2.0466
2	J	4	D	01	2.2279
2	J	4	D	03	2.1956
2	J	4	D	08	2.6021
2	J	4	D	10	2.5016
2	J	4	D	12	2.4726
2	J	4	D	17	2.9237
2	J	4	D	19	2.7989
2	J	4	H	01	3.1167
2	J	4	H	03	3.0330
2	J	4	H	08	3.6788
2	J	4	H	10	3.4901
2	J	4	H	12	3.4394

2 J 4 H 17	4.1143
2 J 4 H 19	3.9784
2 J 4 J 01	1.0
2 J 4 J 03	1.0
2 J 4 J 08	1.0
2 J 4 J 10	1.0
2 J 4 J 12	1.0
2 J 4 K 01	2.6649
2 J 4 K 03	2.4643
2 J 4 K 08	3.1461
2 J 4 K 10	2.8957
2 J 4 K 12	2.7245
2 J 4 K 17	3.4505
2 J 4 K 19	3.2333
2 J 4 M 1	3.0095
2 J 4 M 3	2.8139
2 J 4 M 8	3.5813
2 J 4 M 10	3.3007
2 J 4 M 12	3.0901
2 J 4 M 17	4.0041
2 J 4 M 19	3.6765
2 J 5 F 01	1.8131
2 J 5 F 03	1.7311
2 J 5 F 08	2.1240
2 J 5 F 10	1.9878
2 J 5 F 12	1.9487
2 J 5 F 17	2.3166
2 J 5 F 19	2.2199
2 J 5 G 01	1.3176
2 J 5 G 03	1.2922
2 J 5 G 08	1.5505
2 J 5 G 10	1.4650
2 J 5 G 12	1.4238
2 J 5 G 17	1.6962
2 J 5 G 19	1.6270
2 J 5 B 01	1.1930
2 J 5 B 03	1.1716
2 J 5 B 08	1.3488
2 J 5 B 10	1.3239
2 J 5 B 12	1.2906
2 J 5 B 17	1.5299
2 J 5 B 19	1.4992
2 J 5 C 01	1.4830
2 J 5 C 03	1.4147
2 J 5 C 08	1.6871
2 J 5 C 10	1.6515
2 J 5 C 12	1.6081

2 J 5 C 17	1.8789
2 J 5 C 19	1.8582
2 J 5 D 01	2.0046
2 J 5 D 03	2.0016
2 J 5 D 08	2.2987
2 J 5 D 10	2.2478
2 J 5 D 12	2.2710
2 J 5 D 17	2.6081
2 J 5 D 19	2.5482
2 J 5 J 01	3.1072
2 J 5 J 03	3.0198
2 J 5 J 08	3.5439
2 J 5 J 10	3.4520
2 J 5 J 12	3.3845
2 J 5 J 17	4.0280
2 J 5 J 19	3.9177
2 J 5 H 01	2.7969
2 J 5 H 03	2.7338
2 J 5 H 08	3.3099
2 J 5 H 10	3.1210
2 J 5 H 12	3.0507
2 J 5 H 17	3.4551
2 J 5 H 19	3.6799
2 J 5 K 01	2.5122
2 J 5 K 03	2.4187
2 J 5 K 08	2.8955
2 J 5 K 10	2.7838
2 J 5 K 12	2.7512
2 J 5 K 17	3.2292
2 J 5 K 19	3.1923
2 J 5 L 01	1.0
2 J 5 L 03	1.0
2 J 5 L 08	1.0
2 J 5 L 10	1.0
2 J 5 L 12	1.0
2 J 6 F 01	1.9740
2 J 6 F 03	1.8775
2 J 6 F 08	2.3283
2 J 6 F 10	2.1876
2 J 6 F 12	2.1169
2 J 6 F 17	2.5952
2 J 6 F 19	2.4627
2 J 6 G 01	1.3756
2 J 6 G 03	1.3748
2 J 6 G 08	1.6188
2 J 6 G 10	1.5492
2 J 6 G 12	1.5113

TABLE 6-4

2 J 6 G 17	1.8050
2 J 6 G 19	1.7141
2 J 6 B 01	1.2138
2 J 6 B 03	1.2072
2 J 6 B 08	1.3809
2 J 6 B 10	1.3558
2 J 6 B 12	1.3470
2 J 6 B 17	1.5906
2 J 6 B 19	1.5865
2 J 6 C 01	1.4586
2 J 6 C 03	1.4362
2 J 6 C 08	1.6671
2 J 6 C 10	1.6311
2 J 6 C 12	1.6612
2 J 6 C 17	1.8660
2 J 6 C 19	1.9030
2 J 6 D 01	2.0152
2 J 6 D 03	2.0543
2 J 6 D 08	2.3230
2 J 6 D 10	2.3066
2 J 6 D 12	2.3801
2 J 6 D 17	2.6681
2 J 6 D 19	2.6674
2 J 6 H 01	2.8754
2 J 6 H 03	2.8840
2 J 6 H 08	3.3269
2 J 6 H 10	3.2691
2 J 6 H 12	3.3489
2 J 6 H 17	3.8384
2 J 6 H 19	3.7793
2 J 6 J 01	1.0
2 J 6 J 03	1.0
2 J 6 J 08	1.0
2 J 6 J 10	1.0
2 J 6 J 12	1.0
2 J 6 K 01	2.7918
2 J 6 K 03	2.6563
2 J 6 K 08	3.2614
2 J 6 K 10	3.0833
2 J 6 K 12	3.0471
2 J 6 K 17	3.7131
2 J 6 K 19	3.5684
2 J 7 F 01	2.2605
2 J 7 F 03	2.1966
2 J 7 F 08	2.6350
2 J 7 F 10	2.5225
2 J 7 F 12	2.5359

2 J 7 F 17	2.9154
2 J 7 F 19	2.8939
2 J 7 G 01	1.5426
2 J 7 G 03	1.5456
2 J 7 G 08	1.81
2 J 7 G 10	1.7479
2 J 7 G 12	1.7643
2 J 7 G 17	2.0576
2 J 7 G 19	2.0076
2 J 7 B 01	1.3270
2 J 7 B 03	1.3308
2 J 7 B 08	1.4987
2 J 7 B 10	1.5125
2 J 7 B 12	1.4935
2 J 7 B 17	1.7387
2 J 7 B 19	1.7334
2 J 7 C 01	1.5578
2 J 7 C 03	1.5643
2 J 7 C 08	1.77
2 J 7 C 10	1.76
2 J 7 C 12	1.8010
2 J 7 C 17	2.02
2 J 7 C 19	2.06
2 J 7 D 01	2.20
2 J 7 D 03	2.22
2 J 7 D 08	2.52
2 J 7 D 10	2.54
2 J 7 D 12	2.62
2 J 7 D 17	2.87
2 J 7 D 19	2.95
2 J 7 H 01	3.1328
2 J 7 H 03	3.1624
2 J 7 H 08	3.6745
2 J 7 H 10	3.6100
2 J 7 H 12	3.6764
2 J 7 H 17	4.2332
2 J 7 H 19	4.2048
2 J 7 J 01	1.0
2 J 7 J 03	1.0
2 J 7 J 08	1.0
2 J 7 J 10	1.0
2 J 7 J 12	1.0
2 J 7 K 1	3.1987
2 J 7 K 3	3.1326
2 J 7 K 8	3.6864
2 J 7 K 10	3.5991
2 J 7 K 12	3.6715

2	J	7	K	17	4.2324
2	J	7	K	19	4.2177
2	J	8	F	01	2.7888
2	J	8	F	03	2.7594
2	J	8	F	08	3.2733
2	J	8	F	10	3.2130
2	J	8	F	12	3.2350
2	J	8	F	17	3.7893
2	J	8	F	19	3.7496
2	J	8	G	01	1.9251
2	J	8	G	03	1.9603
2	J	8	G	08	2.2412
2	J	8	G	10	2.2426
2	J	8	G	12	2.2699
2	J	8	G	17	2.5602
2	J	8	G	19	2.5343
2	J	8	B	01	1.5972
2	J	8	B	03	1.6300
2	J	8	B	08	1.8352
2	J	8	B	10	1.8305
2	J	8	B	12	1.8361
2	J	8	B	17	2.1316
2	J	8	B	19	2.1594
2	J	8	C	01	1.8289
2	J	8	C	03	1.8761
2	J	8	C	08	2.0920
2	J	8	C	10	2.1406
2	J	8	C	12	2.2323
2	J	8	C	17	2.4274
2	J	8	C	19	2.5058
2	J	8	D	01	2.6480
2	J	8	D	03	2.6883
2	J	8	D	08	3.0159
2	J	8	D	10	3.0967
2	J	8	D	12	3.2084
2	J	8	D	17	3.5260
2	J	8	D	19	3.6832
2	J	8	E	01	1.0
2	J	8	E	03	1.0
2	J	8	E	08	1.0
2	J	8	E	10	1.0
2	J	8	E	12	1.0
2	J	9	H	01	3.0409
2	J	9	H	03	3.0651
2	J	9	H	08	3.3356
2	J	9	H	10	3.4539
2	J	9	H	12	3.5261

2	J	9	H	17	3.8954
2	J	9	H	19	4.0058
2	J	9	K	01	2.3545
2	J	9	K	03	2.4618
2	J	9	K	08	2.6832
2	J	9	K	10	2.7493
2	J	9	K	12	2.8584
2	J	9	K	17	3.2013
2	J	9	K	19	3.2001
2	J	9	L	01	1.0
2	J	9	L	03	1.0
2	J	9	L	08	1.0
2	J	9	L	10	1.0
2	J	9	L	12	1.0
2	J	9	P	01	2.7289
2	J	9	P	03	2.8246
2	J	9	P	08	3.0142
2	J	9	P	10	3.1465
2	J	9	P	12	3.3367
2	J	9	P	17	3.4709
2	J	9	P	19	3.7137
2	J	9	F	01	1.6038
2	J	9	F	03	1.6125
2	J	9	F	08	1.7555
2	J	9	F	10	1.7937
2	J	9	F	12	1.8568
2	J	9	F	17	2.0397
2	J	9	F	19	2.0828
2	J	9	G	01	1.1929
2	J	9	G	03	1.2318
2	J	9	G	08	1.3209
2	J	9	G	10	1.3260
2	J	9	G	12	1.3566
2	J	9	G	17	1.4822
2	J	9	G	19	1.4856
2	J	9	B	01	1.0478
2	J	9	B	03	1.0667
2	J	9	B	08	1.1232
2	J	9	B	10	1.1521
2	J	9	B	12	1.1638
2	J	9	B	17	1.2758
2	J	9	B	19	1.3045
2	J	9	C	01	1.1272
2	J	9	C	03	1.1397
2	J	9	C	08	1.2050
2	J	9	C	10	1.2335
2	J	9	C	12	1.3018

TABLE 6-4

2	J	9	C	17	1.3343
2	J	9	C	19	1.3960
2	J	9	D	01	1.4114
2	J	9	D	03	1.4729
2	J	9	D	08	1.5453
2	J	9	D	10	1.5967
2	J	9	D	12	1.6544
2	J	9	D	17	1.7514
2	J	9	D	19	1.8211
2	J	9	E	01	1.7395
2	J	9	E	03	1.7800
2	J	9	E	08	1.9046
2	J	9	E	10	1.9872
2	J	9	E	12	2.0426
2	J	9	E	17	2.1691
2	J	9	E	19	2.2914
2	J	9	V	01	1.9062
2	J	9	V	03	1.7130
2	J	9	V	08	2.2272
2	J	9	V	10	1.9802
2	J	9	V	12	1.7806
2	J	9	V	17	2.2885
2	J	9	V	19	2.0764
2	J	9	W	01	1.4878
2	J	9	W	03	1.3670
2	J	9	W	08	1.7420
2	J	9	W	10	1.5544
2	J	9	W	12	1.4343
2	J	9	W	17	1.8203
2	J	9	W	19	1.6259
2	J	9	R	01	1.5717
2	J	9	R	03	1.4275
2	J	9	R	08	1.8166
2	J	9	R	10	1.6461
2	J	9	R	12	1.4966
2	J	9	R	17	1.9233
2	J	9	R	19	1.7396
2	J	9	S	01	2.2165
2	J	9	S	03	1.9508
2	J	9	S	08	2.5446
2	J	9	S	10	2.2621
2	J	9	S	12	2.0690
2	J	9	S	17	2.6395
2	J	9	S	19	2.3785
2	J	9	T	01	2.7724
2	J	9	T	03	2.5566
2	J	9	T	08	3.2661

2	J	9	T	10	2.9575
2	J	9	T	12	2.7101
2	J	9	T	17	3.5402
2	J	9	T	19	3.1598
2	J	9	U	1	2.6775
2	J	9	U	3	2.4025
2	J	9	U	8	3.1359
2	J	9	U	10	2.7937
2	J	9	U	12	2.4932
2	J	9	U	17	3.3520
2	J	9	U	19	2.9774

TABLE 6-5

PERIPHERAL ELEMENT CORRECTION FACTORS
(Azimuthal Traverse at Radial Position of 45 cm)

Core Location (Channel 10)	NC _i Values			Estimate of Population	Estimate of Population
	PAX-G0	PAX-G1	PAX-G2	Mean	Standard Deviation
P3D	1.16791	1.28716	1.26541	1.24016	0.0635
P3E	1.17380	1.27959	1.28398	1.24579	0.0638
P3K	1.16284	1.25491	1.26796	1.22857	0.0573
P3L	1.17627	1.23340	1.24626	1.21864	0.0373
P4D	1.18288	1.21258	1.21541	1.20362	0.0180
P4E	1.23847	1.21385	1.20942	1.22058	0.0157
P4F	1.23918	1.19920	1.18799	1.20879	0.0161
P4L	1.26600	1.21623	1.21207	1.23143	0.0300
P5D	1.23392	1.20016	1.20999	1.21469	0.0174
P5E	1.22271	1.20763	1.21618	1.21550	0.0076
P5F	1.19572	1.17460	1.18170	1.18400	0.0107
P5K	1.18053	1.20251	1.18593	1.18966	0.0115
P5L	1.15653	1.20851	1.19826	1.18777	0.0275
P6D	1.16390	1.20611	1.18852	1.18617	0.0212
P6E	1.16526	1.23566	1.21119	1.20404	0.0357
P6F	1.19242	1.20363	1.20163	1.19923	0.0060
P7L	1.18649	1.19189	1.18292	1.18710	0.0045
P7D	1.18967	1.15779	1.15167	1.16638	0.0204
P7E	1.18974	1.14188	1.15634	1.16265	0.0245
P7F	1.19909	1.15241	1.14796	1.16649	0.0283
P8D	1.19959	1.14519	1.13039	1.15839	0.0364
P8E	1.22505	1.14645	1.14846	1.17332	0.0448
P8F	1.19061	1.15155	1.15316	1.16510	0.0221
P1C	1.13958	1.13170	1.13153	1.13428	0.0046
P1D	1.14173	1.17084	1.16639	1.15965	0.0157
P1E	1.17849	1.16432	1.16692	1.16991	0.0075
P2D	1.12565	1.18136	1.19256	1.16652	0.0358
P2E	1.13662	1.22076	1.21466	1.19068	0.0469

TABLE 6-6

CORRECTION FACTORS FOR CORE LOCATIONS ALONG A RADIAL TRAVERSE
THROUGH THE DRUM B CENTERLINE

Core Location (Channel 10)	NC _i Values			Estimate of Population	Estimate of Population
	PAX-G0	PAX-G1	PAX-G2	Mean	Standard Deviation
00D	1.00050	1.00049	1.00051	1.00050	0.000008
A1B	1.00128	1.00208	1.00358	1.00231	0.0012
A1C	1.00590	1.00457	1.00654	1.00567	0.0010
B2F	1.02348	1.01888	1.02040	1.02092	0.0023
B2E	1.02394	1.02059	1.02178	1.02177	0.0017
C3G	1.02415	1.02341	1.02405	1.02387	0.0004
C3D	1.02624	1.02776	1.02769	1.02723	0.0009
D4B	1.02777	1.03079	1.02955	1.02937	0.0015
D4C	1.02890	1.03242	1.03066	1.03066	0.0018
E4F	1.05734	1.05913	1.05444	1.05697	0.0024
E4E	1.05778	1.06021	1.05542	1.05780	0.0024
F5G	1.06500	1.06911	1.06434	1.06615	0.0026
F5D	1.06607	1.07058	1.06564	1.06743	0.0027
G6B	1.07098	1.07660	1.07152	1.07303	0.0031
G6C	1.07339	1.07793	1.07291	1.07474	0.0028
H6F	1.09344	1.08501	1.08281	1.08709	0.0056
H6E	1.10135	1.09616	1.09290	1.09680	0.0043
J7G	1.11515	1.11061	1.09755	1.10777	0.0091
J7D	1.12764	1.12065	1.12479	1.12436	0.0035
P6G	1.15354	1.15972	1.15575	1.15634	0.0031
P6F	1.19242	1.20363	1.20163	1.19923	0.0060

TABLE 6-7
CORRECTION FACTORS FOR CORE LOCATIONS
ALONG AN AZIMUTHAL TRAVERSE AT A RADIAL POSITION OF 35 CM

Core Location (Channel 10)	NC _i Values			Estimate of Population	Estimate of Population
	PAX-G0	PAX-G1	PAX-G2	Mean	Standard Deviation
H3G	1.08955	1.09980	1.09400	1.09445	0.0051
H3F	1.08849	1.08260	1.09413	1.08841	0.0058
H4B	1.08787	1.09216	1.08912	1.08972	0.0022
H4G	1.09408	1.09644	1.09295	1.09449	0.0018
H4F	1.09214	1.09810	1.08527	1.09184	0.0064
H5B	1.09496	1.08824	1.08505	1.08942	0.0051
H5G	1.09349	1.08678	1.09251	1.09093	0.0036
H5F	1.10829	1.09737	1.09756	1.10107	0.0063
H6B	1.06587	1.06551	1.07909	1.07016	0.0077
H6G	1.08540	1.08592	1.07009	1.08047	0.0090
H6F	1.09344	1.08501	1.08281	1.08709	0.0056
H7B	1.08062	1.08161	1.08173	1.08099	0.0006
H7G	1.08689	1.08162	1.08302	1.08384	0.0027
G7C	1.07758	1.08044	1.07643	1.07815	0.0021
G7D	1.08545	1.08920	1.08620	1.08695	0.0020
G7E	1.09125	1.09364	1.09075	1.09188	0.0015
G1B	1.08015	1.07737	1.08246	1.07999	0.0025
G1G	1.09596	1.09354	1.08964	1.09305	0.0032
G1F	1.08884	1.09186	1.08034	1.08701	0.0060
G2D	1.08731	1.09308	1.09023	1.09021	0.0029
G2E	1.08553	1.09265	1.08976	1.08931	0.0036

TABLE 6-8
CORRECTION FACTORS FOR CORE LOCATIONS ALONG AN AZIMUTHAL TRAVERSE
AT RADIAL POSITION OF 43 CM

Core Location (Channel 10)	NC _i Values			Estimate of Population	Estimate of Population
	PAX-G0	PAX-G1	PAX-G2	Mean	Standard Deviation
P2J	1.09992	1.12072	1.10484	1.10849	0.0109
P3B	1.13934	1.16191	1.17320	1.15815	0.0172
J3B	1.11911	1.13372	1.13119	1.12801	0.0078
J3E	1.12114	1.14427	1.15662	1.14068	0.0180
J4C	1.10624	1.11199	1.10407	1.10743	0.0041
J4D	1.09009	1.10657	1.09990	1.09885	0.0083
P4H	1.16726	1.16272	1.16036	1.16345	0.0035
J5D	1.15910	1.14519	1.15706	1.15378	0.0075
J5F	1.15618	1.13900	1.14633	1.14717	0.0086
J6D	1.14167	1.13814	1.12588	1.13523	0.0083
J6F	1.14159	1.14027	1.13522	1.13903	0.0034
P6B	1.11767	1.13920	1.13163	1.12950	0.0109
J7D	1.12764	1.12065	1.12479	1.12436	0.0035
P7H	1.13811	1.11261	1.10639	1.11904	0.0168
J8C	1.12908	1.11577	1.10980	1.11822	0.0099
J8E	1.14132	1.11726	1.10985	1.12281	0.0165
P8K	1.11267	1.10486	1.11745	1.11166	0.0064
P8M	1.09512	1.07688	1.07950	1.08383	0.0099
H1D	1.09962	1.08764	1.08714	1.09147	0.0071
H1E	1.08247	1.09015	1.08282	1.08515	0.0043

TABLE 6-9

FISSION DENSITY STATISTICS FOR CORE LOCATION P5F10

Estimate of Population Parameters

<u>Case, i</u>	<u>NC_i</u>	<u>Absolute value of (NC_i - M)</u>
PAX-G0, 1	1.19572	0.01172
PAX-G1, 2	1.17460	0.00940
PAX-G2, 3	1.18170	0.00230

M = Estimate of population mean = 1/3 (Sum of NC_i values) = 1.18400

S = Estimate of population standard deviation
calculated from the above sample =

$$\sqrt{\frac{\sum (NC_i - M)^2}{(3 - 1)}} = 0.0107$$

Population Fraction and Confidence Level

<u>Population Fraction</u>	<u>Confidence Level, %</u>	<u>K*</u>	<u>KS</u>	<u>M + KS</u>
0.99	95	10.552	0.1129	1.2969
0.95	95	7.655	0.0819	1.2659
0.90	95	6.158	0.0659	1.2499
0.75	95	3.804	0.0407	1.2247
0.99	90	7.340	0.0785	1.2625
0.95	90	5.310	0.0568	1.2408
0.90	90	4.258	0.0456	1.2296
0.75	90	2.602	0.0278	1.2118

* Ref 54

Figure 6-1. Schematic of Region Assignment for PAX-E1/Drums at 105° Calculation by Exterminator-2

Figure 6-2. Detailed Control Vane Material Assignment for PAX-E1 with Vanes at 105°

Figure 6-3. Position of Traverses Used

Figure 6-4. TaC Shim Locations in PAX-E1
(Criticality at 104.8°)

Figure 6-5. Measured Radial Variation of the PAX-E1 Fission Distribution

Figure 6-6. Measured Radial Variation for the PAX-E1 Peripheral Fission Distribution

Figure 6-7. Comparison of Calculated and Measured Fission Rate in PAX-E1
for a Traverse Between Drums in the NRX-A6 Quadrant
(Between the J3 and J4 Clusters)

Figure 6-8. Comparison of Calculated and Measured Fission Rate in PAX-E1 for an Azimuthal Traverse at the Edge of the Homogenized Core

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Figure 6-9. Comparison of Calculated and Measured Fission Rate in PAX-E1 for an Azimuthal Traverse at a Radial Position of 39 cm

Figure 6-10. Comparison of Measured and Calculated Radial Fission Density Distribution in FCX at Midplane

Figure 6-11. Comparison of Measured and Calculated Axial Fission Density Distribution at Several Radial Positions in the Core of FCX

Figure 6-12. Comparison of Measured and Calculated Axial Fission Density Distribution at Several Radial Positions in the FCX Reflector

Figure 6-13. PAX-G2A Central Channel Radial NC_i Variation

Figure 6-14. PAX/R-1 Core Configuration Illustrating PAX-G1 and G2 Control Drum Arrangement Relative to Test Sector

With the Specified Confidence Level, at least: P Percent of the NC_i Values will be Less than that shown at Location P5F10

Figure 6-15. Lower Limit of the Fission Density Correction Factor for Various Proportions (P) and Confidence Levels at PAX-G/R Reactor Core Location P5F10

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7.0 MATERIAL REACTIVITY COEFFICIENT CALCULATIONS

This chapter discusses the calculation of reactivity coefficients in the PAX-E1 and FCX reactors using the techniques discussed in Section 3.2.4 and compares the calculated results with measurements. In the last section, the accuracy evaluation of WISDM material reactivity coefficient calculations is presented.

7.1 PAX-E1 REACTOR

The PAX-E1 reactor assembly, built in support of the NRX-A6 reactor design effort, consisted of a 90 degree azimuthal simulation of NRX-A6 using Build 2 fuel and a 270 degree segment made up of re-zoned PAX-D fuel. The actual NRX-A6 beryllium reflector rings were used for the reflector. This assembly has been fully documented in WANL-TME-1739¹² and its major physical characteristics are summarized in Appendix A.

After an extensive review of experiments in the PAX reactor series, the PAX-E1 configuration was selected for the WISDM evaluation for the following reasons:

- 1) There exists for the PAX-E1 reactor extensive experimental results for reactivity coefficients of many major constituent materials as well as absolute reactivity and fission distribution data.¹²
- 2) Detailed mass inventory data were readily available for this configuration, since the NRX-A6 reactor was recently shimmed relative to the PAX-E1.²
- 3) The PAX-E1 reactor assembly presents a relatively simple non-core geometry for analysis since the actual NRX-A6 reflector rings were used.

7.1.1 Calculated Reactivity Coefficients

The reactivity coefficients of carbon, fully enriched uranium (93.15 w/o U-235), tantalum, niobium, molybdenum, beryllium and hydrogen were calculated as a function of radial position in the PAX-E1 reactor using the procedure detailed in Figure 3-11. Sixteen group microscopic and macroscopic neutron cross-section data were produced by GAMBIT using the neutron scalar flux and current as calculated by the LINK system (see Figure 3-3) for the PAX-E1 reactor. The standard fine group cross-section libraries that were used are specified in Appendix C. The macroscopic cross-section data and the PAX-E1 nuclear model were input to the ANISN one-dimensional discrete ordinate transport code, and the forward and adjoint

angular fluxes were calculated and placed on magnetic tape. The ANISN calculation contained a 90 degree ring vane prescription to simulate the control drum vanes. The reactivity coefficients as a function of radial position were calculated with the PERT-1D code, using the GAMBIT microscopic cross-section data and the ANISN forward and adjoint angular fluxes. Results are shown in Figures 7-1 through 7-7.

7.1.2 Comparison of Calculated and Measured Reactivity Coefficients

7.1.2.1 Carbon

The axially averaged reactivity coefficient of carbon was calculated as a function of radial position throughout the PAX-E1 assembly. Results are compared in Figure 7-1 with the measured carbon reactivity coefficients at three radial locations in the core.¹³ As can be seen, the calculated and the measured coefficients for carbon are in very good agreement.

7.1.2.2 Enriched Uranium

The calculated axially averaged reactivity coefficient of fully enriched uranium (93.15 w/o U-235) in the PAX-E1 core is shown in Figure 7-2 together with data taken at four radial locations in the core.¹³ The calculated and measured data are in good agreement.

7.1.2.3 Tantalum

For tantalum, 16 group homogeneous and effective heterogeneous cross sections were developed. The effective heterogeneous cross sections included resonance self-shielding and thermal self-shielding factors.* Both of these cross-section sets were used to calculate tantalum reactivity coefficients in the PAX-E1. The calculated results are compared to the measurements that were made in the PAX-E1 core¹³ in Figure 7-3. The calculated reactivity coefficients for tantalum, using the effective heterogeneous cross section data agree to within 20 percent with the measured data near the reactor centerline. The error at the 32 cm position is only 2 percent. However, when the homogeneous cross-section data are used, calculated results are in error by about a factor of two.

*Thermal self-shielding effects were computed by the first flight transport approximation described in Reference 33.

7.1.2.4 Niobium

Homogeneous as well as effective heterogeneous cross sections (which only include resonance self-shielding) were developed for niobium. The reactivity coefficients calculated in the core from these data are shown in Figure 7-4. Niobium reactivity coefficients were not measured in the PAX-E1 assembly; therefore, in Figure 7-4 the calculated data are compared to the available niobium measurements in the PAX-E0 and PAX-E2 assemblies.¹³ Since the PAX-E0 and PAX-E2 reactors differed from the PAX-E1 assembly, a direct comparison of the calculated and measured niobium reactivity coefficients is not possible. However, it can be seen from Figure 7-4, as was the case for tantalum in Figure 7-3, that the use of effective heterogeneous cross sections provides better agreement between the calculation and the measurement.

7.1.2.5 Molybdenum

For molybdenum, homogeneous and effective heterogeneous cross sections (which only include resonance self-shielding) were computed to test the effect of self-shielding. In calculating the molybdenum reactivity coefficient, the contributions due to the isotopes Mo^{92} and Mo^{94} were neglected. This assumption is valid since these isotopes make a negligible contribution to the total molybdenum cross section. The results of these reactivity coefficient calculations are compared to the PAX-E1 measured data for Mo wires¹² in Figure 7-5. As is shown in this figure, the calculated reactivity coefficients, using homogeneous cross sections, are in error by 70 to 80 percent. In contrast to the cases of niobium and tantalum, however, use of the heterogeneous cross-section data does not result in substantial improvement. The average error in the core is only reduced to the 40 - 60 percent range.

Comparison of the calculated self-shielding effects of the molybdenum wires with comparable measured data also shows a substantial difference. The measured resonance self-shielding effect (extrapolated to a zero wire diameter) is about twice as large as the calculated value obtained using the homogeneous and heterogeneous cross-section data. This discrepancy may be due to an error in the resonance parameters in the molybdenum cross-section library or to the fact that the QUERY code (see Section 3.2.b) does not properly

handle heterogeneous resonance self-shielding factor calculations for isotopes having predominantly scattering resonances. The isotopes of molybdenum have predominantly scattering resonances.

7.1.2.6 Hydrogen and Beryllium

The reactivity coefficients of hydrogen and beryllium were also calculated for the PAX-E1 assembly. Figure 7-6 presents the hydrogen reactivity coefficient as a function of radial position throughout the assembly and Figure 7-7 describes the beryllium reactivity coefficient in the reflector. No experimental data are available for these elements, and the calculations are included for purposes of completeness.

7.2 FCX REACTOR

The Flexible Critical Experiment (FCX) reactor is a clean geometry facility built for the purpose of evaluating and improving analytical methods used in NERVA nuclear, radiation, and shielding design and in the evaluation of basic nuclear data (e. g. , cross sections). The FCX reactor consists of a core of NERVA type fueled clusters and a reflector of graphite clusters. Unlike the PAX reactor series, the FCX contains no control drums, but rather seven control rods which have fuel followers attached. The FCX assembly has been documented in WANL-TME-1474³⁴ and its major physical characteristics are summarized in Appendix A.

7.2.1 Calculated Reactivity Coefficients

The calculated reactivity coefficients of carbon, fully enriched uranium (93.15 w/o U-235), niobium, and hydrogen are presented as a function of radial position in the FCX assembly in Figures 7-8 through 7-12. At the time the FCX reactivity coefficients were calculated, the WISDM procedure for this computation was under development and, therefore, in addition to the standard procedure detailed in Figure 3-11, some reactivity coefficients were calculated by direct ANISN calculations (i. e. , comparison of reactivity from "base case" and "perturbed" ANISNs) in order to qualify the use of PERT-1D. In each case, the method of calculation is specified.

PERT-1D calculations were performed in 16 and 52 energy groups. Neutron cross-section data in the required group structure were produced by GAMB1T, using the converged neutron scalar flux and current as calculated for the FCX reactor. The standard fine group

cross-section libraries used are specified in Appendix C. Macroscopic cross-section data and the FCX nuclear model were input to the ANISN one-dimensional neutron transport code, and the forward and adjoint angular fluxes were calculated. The reactivity coefficients as a function of radial position were then developed with the PERT-1D code (see Section 3.2.4) from GAMBIT microscopic cross-section data and the ANISN calculated forward and adjoint angular fluxes.

7.2.2 Comparison of Calculated and Measured Reactivity Coefficients

7.2.2.1 Carbon

The reactivity coefficient of carbon was calculated as a function of radial position in the FCX assembly, using 16 and 52 group PERT-1D calculations and 16 group direct ANISN calculations. Results are presented in Figure 7-8. Also shown in this figure are the results of a 52 group PERT-1D calculation which does not include a transverse leakage correction (see Section 3.2.4) and the experimentally determined carbon reactivity coefficients at several radial locations in the core and reflector.³⁴ As can be seen from Figures 7-8, the PERT-1D calculations (both 16 and 52 group) which include a transverse leakage correction give very good agreement with the experimental data, while the PERT-1D calculation without a transverse leakage correction differs substantially from the measured data. The results from the 16 and 52 group PERT-1D calculations do not appreciably differ from one another. The 16 group direct ANISN calculation substantially agrees with the PERT-1D results.

7.2.2.2 Enriched Uranium

The reactivity coefficient of fully enriched uranium (93.15 w/o U-235) in the FCX assembly, calculated by PERT-1D using 16 and 52 group cross sections, are shown in Figure 7-9 together with the measured data at several radial locations in the reactor.³⁴ The results of 16 group direct ANISN calculations at two radial positions in the core are also shown in the figure. The reactivity coefficients calculated by the 16 and 52 group PERT-1D calculations are in close agreement with one another throughout the FCX assembly and are in good agreement with the measured data in the core. However, in the reflector the calculated data are as much as 50 percent higher than the experimental results.

Using the GAMB1T code, effective, enriched uranium, 16 group cross sections in the graphite reflector were generated, which included resonance self-shielding and thermal flux depression factors. The results of a 16 group PERT-1D calculation using these effective uranium cross sections are shown in Figure 7-10, together with the measured data.³⁴ (The cross sections for the core were not changed so that there is no effect shown for the core.) As can be seen, the use of effective cross sections yields a considerable improvement in the agreement between calculation and experiment. As was noted above, the "uncorrected" uranium cross sections in the reflector yielded PERT-1D calculated reactivity coefficients that differed from experimental data by as much as 50 percent. Figure 7-10 shows that the error in the reflector has been reduced to less than 15 percent when effective uranium cross-section data are used.

7. 2. 2. 3 Niobium

A 16 and 52 group PERT-1D analysis was performed to calculate the niobium reactivity coefficients as a function of radial position in the FCX assembly. Figure 7-11 presents the results of these calculations, together with the experimental data³⁴ for several radial positions in the FCX core. As has been seen previously, the 16 and 52 group PERT-1D reactivity coefficient calculations are in close agreement with one another. However, in the FCX core the calculation is as much as 30 percent higher than the experimental results. In the reflector, no experimental data are available for comparison.

As the analysis of the PAX-E1 reactor has shown (see Section 7. 1. 2. 4), the use of effective cross sections, which include resonance self-shielding effects, would result in considerably better agreement between the calculated and measured niobium reactivity coefficients (e. g. , see Figure 7-2). Another possibility for the discrepancy between calculation and measurement, as shown in Figure 7-11, is an error in the niobium cross-section library data used in GAMB1T.

7. 2. 2. 4 Hydrogen

For hydrogen, 16 and 52 group PERT-1D and direct 15 group ANISN calculations were performed to obtain the hydrogen reactivity coefficients in the FCX assembly as a function of radial position. The results of these calculations are presented in Figure 7-12,

together with experimental data at eight radial locations in the FCX core and reflector. As is shown in Figure 7-12, the agreement between the calculated and measured hydrogen reactivity coefficients is excellent.

The hydrogen reactivity worths in FCX were measured by inserting polyethylene (CH₂) tubes or rods into single-hole FCX graphite elements at the radial location of interest. To obtain the radial hydrogen reactivity coefficients, the reactivity worth of carbon in the polyethylene (see Figure 7-8) was subtracted from the total reactivity effect.

7.3 EVALUATION OF THE ACCURACY OF REACTIVITY COEFFICIENT CALCULATIONS

In the evaluation of WISDM, reactivity coefficients in the PAX-E1 and FCX reactors have been calculated using the techniques defined in the flow chart of Figure 3-11. The analysis has shown that for the two major constituent materials in NERVA type reactors, namely carbon and enriched uranium, the calculated and experimentally determined reactivity coefficients as a function of radial position agree to within ± 10 percent, with the exception of the uranium coefficients in the reflector region (of FCX) which were calculated to within ± 15 percent.

Reactivity coefficients for niobium were also calculated for both the PAX-E1 and FCX critical assemblies. For the PAX-E1 reactor, no experimental data exist, so direct comparison is not possible. (Figure 7-4 compares the PAX-E1 calculated results with PAX-E0 and PAX-E2 experimental data.) For the FCX reactor the calculated niobium coefficients are up to 30 percent higher than the experimental data. As analysis of other materials has shown (e. g., see Figure 7-3 for tantalum) the use of effective cross sections, which include resonance self-shielding effects, would considerably improve the agreement between the calculation and the measurement.

In addition to the above, reactivity coefficients for tantalum and molybdenum were calculated in the PAX-E1 assembly and for hydrogen in the FCX assembly. The calculated tantalum reactivity coefficients differed by up to 20 percent from experimental data when effective cross sections were used. (When cross sections that did not include resonance and thermal self-shielding effects were used, the difference between the calculated and experimental data was almost 100 percent.) For molybdenum, the calculated and experimental

coefficients differed by 50 percent even when effective cross sections were used. This discrepancy may be due to an error in the molybdenum resonance parameters or to the fact that QUERY does not properly handle heterogeneous resonance self-shielding factor calculations for isotopes having predominantly scattering resonances, as is the case with molybdenum. For hydrogen, the calculated and experimentally determined reactivity coefficients as a function of radial position in the FCX assembly are in excellent agreement.

Figure 7-1. Radial Variation of the Carbon Reactivity Coefficient in the PAX-E1 Core

Figure 7-2. Radial Variation of the Uranium Reactivity Coefficient in the PAX-E1 (Core)

Figure 7-3. Radial Variation of Tantalum Reactivity Coefficient in the PAX E1 Core

Figure 7-4. Radial Variation of the Niobium Reactivity Coefficient in the PAX-E1 Core

Figure 7-5. Radial Variation of the Molybdenum Reactivity Coefficient in the PAX-E1 Core

Figure 7-6. Radial Variation of the Hydrogen Reactivity Coefficient in the PAX-E1

Figure 7-7. Radial Variation of the Beryllium Reactivity Coefficient in the PAX-E1 Reflector

Figure 7-8. Comparison of the Calculated and Measured Radial Reactivity Coefficient of Carbon in the FCX Reactor

Figure 7-9. Comparison of the Calculated and Measured Radial Reactivity Coefficient of Fully Enriched Uranium in the FCX Reactor

Figure 7-10. Comparison of the Calculated and Measured Radial Reactivity Coefficient of Fully Enriched Uranium in the FCX Reactor using Effective Uranium Cross Sections in the Reflector

Figure 7-11. Comparison of the Calculated and Measured Radial Reactivity Coefficient of Niobium in the FCX Reactor

Figure 7-12. Comparison of the Calculated and Measured Radial Reactivity Coefficient of Hydrogen in the FCX Reactor

8.0 RADIATION ENVIRONMENT CALCULATIONS

This section presents a comparison between the measured radiation environment and the calculated environment, using the WISDM techniques described in Section 3.0. The comparison was carried out for the FCX and PAX-E5 experimental critical assemblies. The FCX analysis is based on measured neutron activation data in the core and reflector assembly. PAX-E5 measurements allow evaluation of analytical techniques for calculating neutron activation and photon dose rate for configurations similar to the shield mock-ups placed on top of the aft end of the PAX-E assembly. A summary of PAX-G1A statistical analysis is also presented.

8.1 FCX REACTOR

Section 3.2.5 outlines the general analytical scheme which was employed to obtain the calculated results presented in this section. Appendix A-4 provides a description of the FCX reactor, including schematic diagrams, the component masses in the FCX reactor, the geometrical model used in the analysis, and the region atom densities in the geometrical model. Appendix D provides a description of the manner in which reaction rates are obtained from the DOT and ANISN codes. These reaction rates represent the saturated foil activity which is to be compared directly with the measured data. A detailed discussion of the cross sections used for activation calculations, together with a discussion of the effect of resonance self-shielding, flux depression factors, and cadmium transmission factors for foil activation, are also presented in Appendix D.

8.1.1 Experimental Data

Two series of activation measurements were made in the FCX assembly. These were the FCX A and the FCX B series of experiments. Since the FCX A series had a limited number of measurements (for example, 5 radial and 5 axial), the FCX B series of measurements was planned to provide more spatial resolution which would permit a better comparison of calculated and measured results. In addition, a more accurate power calibration for the FCX B series of measurements was performed.

Activation measurements for the detectors listed in Table 8-1 were made on the reactor midplane to define the radial distribution. The axial distribution was measured at various radial positions. All of the activities are reported in terms of saturated activities

(number of disintegrations per second per watt of reactor power per gram of activated isotope in the foil) except the activity of W^{186} . W^{186} activities are relative data normalized to unity at the center of the core. The absolute power level of the FCX reactor was determined from the radiochemical analysis of an FCX fuel element and a detailed measurement of the radial fission density distribution. There was a 7.5 percent discrepancy in absolute power calibration of the FCX A series of measurements; therefore, the activation data in FCX A were multiplied by a factor of 1.075.

8.1.2 Calculations

Calculations for the FCX assembly to evaluate axial distributions were performed with the two-dimensional, r, z geometry, discrete ordinates transport code, DOT. These calculations employed the standard WISDM 16 neutron energy group structure and P_0 (transport corrected) cross sections.

Calculations for the FCX assembly to evaluate radial distributions were obtained from the DOT calculations noted above or WISDM 52 or 16 neutron energy group ANISN calculations. ANISN calculations employed P_0 (transport corrected), P_1 and P_3 neutron cross sections.

8.1.3 The Effect of Geometry on the Comparison Between Calculations and Measurements

The geometrical effects which may have introduced errors in the FCX activation calculations are: 1) the non-circular cross section of the cylindrical FCX reactor and (2) the removal of fuel and graphite for the purpose of inserting control rod assemblies. The change in activation due to a non-circular core and reactor cross section was not analyzed since this involves an azimuthal variation of neutron flux; however, this effect should be small and should affect only results at the edge of the reactor. The analysis of the effect on foil activation of removal of fuel and graphite for insertion of the control rod assemblies indicated that the maximum expected change in activation rate was less than 5 percent. This occurred near the center of the core and in the reflector due to the radial shift in the fission distribution.

8.1.4 Comparison of Calculations and Measurements

8.1.4.1 Bare and Cadmium-Covered Au Detector

FCX A series measurements used 5-mil gold foils and FCX B series measurements used 0.5-mil gold foils. These foils have different resonance and thermal self-shielding factors as noted in Appendix D. In both series of experiments, axial as well as radial foil activation measurements were made. The Au detectors have a $1/\nu$ absorption cross section in the thermal energy range with a principal resonance at 4.9 eV.

Radial comparisons of calculated and measured saturated activities for bare and cadmium-covered Au foils are shown in Figures 8-1 and 8-2. The agreement between the calculations and the measurements is reasonably good. The FCX B series measurements with 0.5-mil Au foils are approximately ten percent higher than the calculated results. Since the difference shows a consistent bias, the uncertainties may be due to the absolute value of counter efficiencies, power calibration, or consistent bias in the activation cross sections. The FCX A series measurements with 5-mil gold foils (Figure 8-2) are in good agreement, with no significant difference occurring between calculated and experimental data.

Comparisons of axial calculations with measurements for 0.5 and 5.0-mil gold foils are shown in Figures 8-3, 8-4, and 8-5. Experimental results for 0.5-mil foils are at least 10 percent higher than the calculated results, whereas, the experimental results for 5.0-mil foils are slightly lower than the calculated results. Axial measurements near the top of the fuel elements at a radius of 5.4 cm are almost twice as high as the calculations. This effect is most probably due to the presence of control rods which were partially inserted (~ 13.0 cm) into the reactor assembly during the foil activation.

8.1.4.2 Bare and Cadmium-Covered Indium Detectors

The FCX A and FCX B series measurements used 10-mil indium foils containing 5.75 weight percent indium in an aluminum matrix. The indium foils have a $1/\nu$ absorption cross section in the thermal energy range with the principal resonance at 1.45 eV.

A comparison of calculated and measured saturated activities as a function of radial position for indium foils is shown in Figure 8-6. The agreement between calculated and the measured results is good, with the FCX A values approximately 10 percent lower than the FCX B series measured values.

Comparisons of calculated and measured saturated activities as a function of axial position are shown in Figures 8-7, 8-8, and 8-9. Calculated results are in good agreement with the measurement, except near the top of the reactor close to the centerline as shown in Figure 8-7. At this location, the FCX B series of measurements are approximately five times higher than the calculations; whereas the FCX A series of measurements are less than a factor of two higher than calculations (Figure 8-9). The discrepancy in both cases is most probably due to the partial insertion of the control rod bank during the measurements in the FCX B series. The rod bank was inserted ~13 cm or ~10% of the core height.

8.1.4.3 Bare and Cadmium-Covered Dysprosium Detectors

FCX A series measurements used 5-mil foils containing 3.75 weight percent Dy in an aluminum matrix. FCX B series measurements used 10-mil foils containing 5.87 weight percent Dy in an aluminum matrix. These foils do not have an appreciable self-shielding factor, and direct scaling by weight percent to account for the difference in the mass of Dy in the foils was used to obtain saturated activities.

Experimental results for both the FCX A and FCX B series measurements are higher by 10 to 20 percent than the calculated values. The calculated and measured results have approximately the same shape as shown by Figures 8-10 and 8-11.

In the preceding discussion on the Au and In detectors, reasonable agreement with the experimental data was obtained. These detectors respond in the same energy range as Dy, and, therefore, it can be concluded that the calculated neutron flux is reasonably accurate. Therefore, the discrepancy was assumed to be in the Dy activation cross-section data. A comparison of the Dy cross section in the GAMBIT library with the Dy cross sections in the THERMOS⁹ library revealed that the GAMBIT data are based on a thermal cross section ($E = 0.025$ eV) of 2520 barns; whereas, the THERMOS data are based on a thermal cross section ($E = 0.025$ eV) of 2710 barns. This represents an increase in the Dy activation cross section

of 7% which would increase the calculated Dy activity and reduce the discrepancy between the calculated and experimental data.

Figures 8-12, 8-13, and 8-14 compare the saturated Dy activities as a function of axial position. From these figures, it can be seen that the measured results are consistently higher than calculations. In addition, the activities near the top of the core in the FCX B series of measurements tend to peak, whereas, the measurements in the FCX A series of experiments do not display this behavior. As in the case of the Au and In data, this is probably due to the partial insertion of the control rod bank.

8.1.4.4 Bare and Cadmium-Covered Mn Detectors

Bare and cadmium-covered detectors containing 80 weight percent Mn were irradiated in FCX. These measurements were performed in the FCX A series of experiments only. Mn has a $1/v$ absorption cross section with the principal resonance at 337 eV. Figure 8-15 compares the calculations and experiment radially on the midplane, whereas Figure 8-16 illustrates the comparison as a function of axial position at a fixed radius of 6.46 cm. From these figures it can be seen that there is good agreement between calculations and experiment. In addition, there was no peaking in the activation rate at the top of the reactor in the experimental data. This is probably due to the high energy of the principal resonance which would not be affected by the flux trap effect of the control rod bank.

8.1.4.5 Bare and Cadmium-Covered W Detectors

Activities from isotope W^{186} in natural W foils were measured in FCX A series of experiments. Only relative activities normalized to unity at the core center were available. The calculations were normalized to unity at the same location for comparison purposes. The W^{186} isotope has a $1/v$ cross section in thermal energy range with the principal resonance occurring at 18.8 eV.

Figure 8-17 compares calculated and experimental W data as a function of radial position on the core midplane, while Figures 8-18 and 8-19 provide similar data as a function of axial position at a radius of 7.7 and 63.9 cm. From these figures, it can be seen that there is reasonably good agreement between calculated and experimental data.

8.1.4.6 Bare and Cadmium-Covered Cu Detectors

Activities for the Cu^{63} isotope in natural Cu foils were measured in the FCX B series of experiments. Cu^{63} is essentially a $1/v$ absorber in the thermal and epithermal energy range with the largest resonance occurring at 580 eV.

Figure 8-20 provides a comparison of the calculated and experimental Cu data as a function of radial position on the core midplane. Figures 8-21 and 8-22 provide similar data as a function of axial position at a radius of approximately 5.5 and 49.5 cm. From these figures, it can be seen that there is reasonable agreement between the calculated and experimental data. Most of the differences were within the limits of experimental accuracy; however, the bare Cu data on the reactor midplane did exhibit one anomaly at the core reflector interface where the experimental data were noticeably lower than calculations.

8.1.4.7 Bare Sulfur Activities

Sulfur pellets were irradiated in order to measure the high energy neutron flux distribution. Irradiation of sulfur produces P^{32} via an (n, p) reaction which has a threshold at approximately 2.9 MeV. Therefore, the neutron flux with an energy greater than 2.9 MeV is obtained by determining the P^{32} content of the sulfur pellet. These measurements were performed in both the FCX A and FCX B series of experiments.

Figure 8-23 provides a comparison of the calculated and experimental sulfur results as a function of radial position on the core midplane. Figures 8-24, 8-25, and 8-26 provide similar data as a function of axial position at several radial locations. From these figures, it can be seen that the FCX A series of measurements are generally 20 to 30 percent lower than the FCX B series of measurements. This occurred because a less accurate analytical chemistry technique was employed to determine the P^{32} content of the FCX A pellets. The procedure was modified for FCX B so that the FCX B sulfur data represents the reliable experimental information.

It can be seen from these figures that the calculated results are generally 20 to 30 percent lower than the FCX B data. However, the calculated and experimental data do have approximately the same shape. The difference in magnitude could be due to either thermal neutron activation in sulfur, the fission spectrum employed in the calculations,

or the sulfur cross sections. Further analysis is required to determine the cause of the difference.

8.1.5 Summary of FCX Results

The analysis of the neutron activation data taken in the FCX experiments indicated that there was generally good agreement between the calculated and experimental data except at the edges of the assembly. The differences at these locations may be due to the non-circular geometry of the FCX assembly or to the effect of the test cell environment on the neutron fluxes. The analysis also revealed that the calculated and experimental data for both Dy and S had approximately the same shape, but the calculated data were consistently lower for both detectors. Therefore, further investigation is required to understand these differences.

8.2 PAX-E5 REACTOR

Section 3.2.5 outlines the general analysis procedure which was employed to obtain the results presented in this section. Appendix A-3 provides a description of the PAX-E5 reactor and mock-up shield assemblies in which the measurements were taken. The description includes schematic diagrams and photographs of the assemblies, the component masses in the PAX-E5 reactor, the geometrical model employed in the analysis, and the region atom densities in the model. Appendix D provides a description of development of reaction rate response employed with DOT and ANISN neutron or photon calculated fluxes to obtain detector activation or response. The calculated neutron reaction rates for gold, dysprosium, U-238, and sulfur represent saturated foil activity which can be compared directly with the experimental data. Appendix D also contains a detailed discussion of the neutron activation cross sections used in these calculations, including a discussion of resonance self-shielding, flux depression factors, and cadmium transmission factors for the various foil detectors.

The response functions which were applied to the neutron and photon fluxes to obtain the fast neutron and photon (gamma ray) absorbed dose rates do not represent a reaction rate, or activity, in the detector as did the preceding neutron activation cross sections. These response functions represent the factors which convert the neutron number flux (n/cm^2 -sec-w) to the fast neutron absorbed dose rate (rads-tissue / hour-watt) and

factors which convert the photon energy flux ($\text{Mev}/\text{cm}^2\text{-sec-w}$) to the photon absorbed dose rate ($\text{Rads-carbon}/\text{hour-watt}$). The fast neutron absorbed dose rate and photon absorbed dose rate detectors are calibrated in these units, and a direct comparison of calculated and experimental data is made on these terms.

8.2.1 Experimental Data

In the series of radiation and shielding experiments ^{7, 8} conducted on the PAX-E5 reactor, measurements for two experiments were compared with analytical predictions. These two series of measurements were performed in mock-ups which were placed on top of the aft end of the PAX-E5 reactor. The first assembly, which was designated as configuration 4B, consisted of the PAX reactor and a mock-up of a typical core support plate, plenum region, and pressure vessel for a NERVA type reactor. The second assembly, which was designated as configuration 7D, consisted of Configuration 4B with a BATH (Boron Carbide-Aluminum-Titanium-Hydride) mock-up shield (a real density of $79 \text{ gm}/\text{cm}^2$) interposed between the core support plate mock-up and the plenum-pressure vessel mock-up.

Neutron activation measurements were made with gold, dysprosium, U-238, and sulfur detectors. The activation characteristics of these detectors are described in Table 8-1. In addition, the fast neutron absorbed dose rate was obtained with silicon diode (Phylatron) detectors and the photon absorbed dose rate was obtained with CaF_2 thermoluminescent detectors (TLD). In both assemblies, the experimental data were obtained as a function of axial position along the centerline of the mock-up and as a function of radial position at various axial positions internal to the mock-up.

All neutron results are reported in terms of saturated activities (number of disintegrations per second per watt of reactor power per gram of activated isotope in the foil). Fast neutron absorbed dose rate is reported in terms of the neutron absorbed dose rate ($\text{rads-tissue per hour per watt of reactor power}$). Photon absorbed dose rate is reported in units of $\text{rads-carbon per hour per watt of reactor power}$.

8.2.2 Calculations

Calculated results in the mock-ups on the aft end of the PAX-E5 reactor as a function of radial position were obtained with the two-dimensional, r - z geometry, discrete ordinates transport code, DOT. Calculations in the mock-ups as a function of axial position on the centerline were obtained both from DOT and the one-dimensional, discrete ordinates, transport code ANISN. All calculations employed the WISDM 16 neutron energy group structure and 13 photon energy group structure. All calculations were based on P_1 photon cross sections and either P_1 or P_0 (transport corrected) neutron cross sections.

The ANISN and the DOT calculations performed for this analysis were a simultaneous solution for both the neutron and photon fluxes. The ANISN calculations used a symmetric S_8 angular quadrature order, while the DOT calculations employed an asymmetric angular quadrature order S_{124} . In addition, calculations at a limited number of internal locations were also obtained via the KAP V point kernel code, using the DOT-NAGS WISDM technique to obtain the neutron and photon distributed sources for input to KAP-V.

8.2.3 Differences in Analytical Models and Experimental Configuration

The differences between the calculated and experimental results can in part be attributed to differences between the analytical models and the experimental configuration. There were essentially two major differences:

- 1) The analysis did not include the test cell environment which might have affected the measurements taken at the extremities of the assembly.
- 2) The analytical model employed a circular representation of the mock-ups used on the PAX-E5 reactors while the actual simulation was fashioned from layers of square plates. This would also lead to differences between calculations and measurements at the extremities of the assembly.

8.2.4 Comparison of Calculated and Experimental Results

8.2.4.1 Bare and Cadmium Covered Gold Data

The PAX-E5 series of measurements of gold foil activation employed 0.5 mil thick gold foil. Characteristics of this detector are described in Section 8.1.4.1.

Figures 8-27 and 8-28 provide, respectively, the comparison between calculations and experimental bare and cadmium covered gold data as a function of axial position on the centerline of Configuration 4B. From these figures, it can be seen that the DOT calculated results agree quite well with the experimental data. The ANISN activity values agree with DOT values up to an axial position of approximately 138 cm and beyond an axial position of approximately 165 cm. Between these two positions, the ANISN activity is below the experimental data by approximately 23 percent. This difference between the DOT and the ANISN results is most probably due to the analytical treatment of the buckling (transverse leakage term) in ANISN. The single buckling term (energy and region independent) used in ANISN, was apparently not adequate for this analysis. There is no significant difference between the ANISN calculations using P_0 (transport corrected) neutron cross sections and those based on P_1 neutron cross sections as noted in Figure 8-27 and 8-28.

Figures 8-29 and 8-30 compare the calculation and the experimental bare and cadmium covered gold data as a function of axial position on the centerline of Configuration 7D. Analytical results were obtained with the ANISN code only. From these figures, it can be seen that the calculations agree reasonably well with the experimental data in shape; however, they are approximately 23 percent below the experimental results. The difference between calculations and experiment become larger at the top of the assembly (axial position of 200 cm). This effect might be due to test cell environment contributing to the experimental data. From the calculations, it is noted that the shield modules produce large variations in the activation of gold. In addition, at the top of the assembly, there appears to be a difference between the calculations with P_1 neutron cross sections and the calculations with P_0 (transport corrected) cross sections.

8.2.4.2 Bare and Cadmium Covered Dysprosium Data

The PAX-E5 series of dysprosium foil measurements employed a 5 mil thick foil containing 3.75 weight percent of dysprosium. Characteristics of this detector are described in Section 8.1.4.3. Figures 8-31 and 8-32 provide, respectively, a comparison between calculations and experimental bare and cadmium covered dysprosium data as a function of axial position on the centerline of Configuration 4B. From these figures, it can be seen that

the DOT activities agree quite well with the bare dysprosium data except at the top of the assembly. This may be due to the effects of the test cell environment on the experimental data. The shape of the DOT data for cadmium covered Dy agrees quite well with the experimental results; however, the calculated results were approximately 27 percent below the experimental data. As in the case of the gold data, ANISN and DOT agree up to an axial position of approximately 138 cm and beyond an axial position of approximately 165 cm. The different neutron scattering approximations (P_0 transport corrected versus P_1) used in the calculations yield similar results.

Figure 8-33 provides a comparison between calculations and experimental bare dysprosium data as a function of radial position at various axial locations in Configuration 4B. From this figure, it can be seen that there is reasonable agreement between the analysis and the measurements. The difference becomes larger at the edge of the assembly. This effect could be due to either the test cell environment and/or the circular representation of the mock-up in the calculations.

8.2.4.3 Cadmium Covered U-238 Fission Rate Data (Fast Neutron Flux, $E > 1.5$ MeV)

The PAX-E5 series of U-238 measurements employed a 5 mil thick U-238 foil which contained 190 ppm of U-235. Cadmium covers were used to partially suppress neutron induced fission in U-235 so that the primary effect measured was the fast fission of U-238. Calculations indicated that less than 2 percent of the fission rate was due to fission in U-235. Therefore, the U-238 foil data correspond approximately to the integrated fast neutron flux with energies greater than 1.5 MeV (the threshold energy for fast fission in U-238.)

Figures 8-34 and 8-35 provide a comparison between calculations and experimental U-238 fission rate data as a function of axial position on the centerline of Configuration 4B and Configuration 7D. From Figure 8-34, it can be seen that the DOT and ANISN activities agree with one another and agree reasonably well with the experimental data for Configuration 4B. For Configuration 7D (Figure 8-35), it is seen that the shape of the calculated data from ANISN agrees with that of the experimental data; however, the magnitude is approximately 35 percent higher than the experimental data. It should be

noted that the ANISN results based on P_0 (transport corrected) neutron cross sections agree with those based on P_1 neutron cross sections for Configuration 4B but display a difference from one another for Configuration 7D which increases with distance from the core inlet.

Analysis from the KAPV point kernel code is also shown on Figure 8-34 for Configuration 4B. The KAP V data point at approximately 160 cm is considerably below the DOT, ANISN, and experimental data; however, the KAP V data point at the top of the assembly ($Z = 167$ cm) agrees quite well with all the data, both calculated and experimental.

Figure 8-36 compares the calculated and the experimental U-238 fission rate as a function of radial position at four axial locations in Configuration 4B. From this figure, it can be seen that there is reasonable agreement between the DOT and the experimental results as was the case in the axial traverse (Figure 8-34). In addition, the data from KAP-V, presented at the top of the assembly, agree reasonably well with both the experimental and DOT results. It should be noted the the DOT and KAP-V calculated U-238 fission rates display no significant difference from the measurements at the edge of the assembly in the radial direction.

8.2.4.4 Bare Sulfur Data (Fast Neutron Flux $E > 2.9$ MeV)

The PAX-E5 series of sulfur measurements employed sulfur pellets. These pellets produce P^{32} via an (n,P) reaction when irradiated. The reaction has a threshold energy of 2.9 MeV. Thus, sulfur data corresponds to the integrated neutron flux with energies greater than 2.9 MeV. Characteristics of this detector are presented in Section 8.1.4.7.

Figures 8-37 and 8-38 compare the calculated and experimental sulfur activity as a function of axial position on the centerline of Configuration 4B and Configuration 7D, respectively. From Figure 8-37, it can be seen that the calculated activity for DOT and ANISN do not differ substantially. Agreement with the shape of the experimental results for Configuration 4B is good, but all the calculated results are approximately 30 percent below the experimental results. For Configuration 7D (Figure 8-38), it is seen that the activity values from ANISN agree with the experimental results except at the top of the assembly where the measurements may be affected by the test cell environment. The ANISN results employing P_0 (transport corrected) neutron cross sections agree with those using P_1 neutron cross sections

for Configuration 4B. For the deeper penetration shown for Configuration 7D (Figure 8-38) differences between P_0 (transport corrected) and P_1 results become evident. This effect is due to the use of transport correction with isotropic scattering to approximate the anisotropic scattering of neutrons.

Two points calculated by the KAP V point kernel code are also shown on Figure 8-37. These data display the same behavior as did the KAP data for U-238 where agreement at the top of the assembly was good.

Figure 8-39 compares the calculated and experimental sulfur activity as a function of radial position at four axial locations in Configuration 4B. From this figure, it can be seen that the DOT results are approximately 30 percent below the experimental results as was the case for the axial traverse of the data (Figure 8-37). The KAP V data shown in Figure 8-39 at the top of the assembly shows excellent agreement with the experimental results.

8.2.4.5 Fast Neutron Absorbed Dose Rate Data

Fast neutron dose measurements in PAX-E5 employed a p-i-n junction silicon diode (Phylatron) to measure the fast neutron absorbed dose rate. Fast neutron irradiation of the detector displaces nuclei in the diode. The resulting change in the resistivity of the diode that occurs can be measured after irradiation. Phylatrons are calibrated so that the change in resistivity corresponds to the fast neutron absorbed dose rate which is reported in units of rad (tissue) per hour.

Figures 8-40 and 8-41 provide a comparison between calculated and experimental neutron absorbed dose rate as a function of axial position on the centerline of Configuration 4B and Configuration 7D. From Figure 8-40, it can be seen that the DOT and ANISN calculations agree with one another and with the measurements. For Configuration 7D (Figure 8-41), it is seen that the calculated dose rate from ANISN agrees with the experimental results except at the top of the assembly where the measurements may have been affected by the test cell environment. It should be noted that the ANISN calculations employing P_0 (transport corrected) neutron cross sections agree with those performed with P_1 neutron cross sections for Configuration 4B but differences occur for Configuration 7D. These differences increase as a function of depth of penetration into the shield.

Carbon moments (infinite media) and modified Albert-Welton kernel calculated data from KAP V are also shown on Figure 8-40. It is seen that the carbon moments data agree reasonably well with the DOT and ANISN results and with the measurements; however, the Albert-Welton dose rates are considerably lower. These differences are due to the basic principles used in deriving the carbon moments data of Albert-Welton.

Figure 8-42 compares the calculated and experimental fast neutron absorbed dose rate as a function of radial position at four axial locations in Configuration 4B. From this figure, it can be seen that the DOT dose rates agree reasonably well with the measurements as was the case for the axial traverse (Figure 8-40). The KAP V carbon moments data shown in this figure exhibit the same shape as the DOT and the experimental dose rates. However, the KAP data presented is for traverse 4 and is therefore higher than either the DOT or experimental results. A difference of comparable magnitude occurred in the axial traverse.

8.2.4.6 Gamma Ray Absorbed Dose Rate Data

The PAX-E5 series of measurements employed $\text{CaF}_2\text{:Mn}$ thermoluminescent detectors (TLD's) to measure the photon (gamma ray) absorbed dose rate. Photons interact with the carrier electrons in the $\text{CaF}_2\text{:Mn}$ to produce holes in the electron energy band. Upon heating, thermal agitation produces electrons which fall into the holes in the energy band to produce photons with wavelength of about 5000 angstroms. The amount of light produced is proportional to the absorbed photon dose rate and is measured with a light meter after irradiation by placing the TLD's in an oven. The TLD's were calibrated so that the experimental measurement of the absorbed photon dose rate could be reported directly in terms of rads-carbon per hour.

Figures 8-43 and 8-44 provide a comparison between the calculated and experimentally determined photon absorbed dose rate as a function of axial position on the center-line of Configuration 4B and Configuration 7D. From Figure 8-43, it is seen that the DOT and ANISN calculations agree with one another and with the experimental results. For Configuration 7D (Figure 8-44), it is seen that the calculated dose rate from ANISN has approximately the same shape as the experimental results except at the top of the assembly where the measurements may be affected by the test cell environment.

However, the calculated results for Configuration 7D are approximately 15 percent below the experimental results. It should be noted that both sets of ANISN calculations for each configuration employed P_1 cross sections for photons and the reference to the difference in angular representation in the cross sections refers only to the neutron cross sections. The difference between P_0 and P_1 neutron cross sections would only affect the production of secondary photons, a second order effect in this configuration. Therefore, these calculations do not display a difference between the photon data obtained with P_0 neutron cross sections and the data obtained with P_1 neutron cross sections. However, in a calculation where secondary photons are a major contributor to the dose rate, the use of P_0 or P_1 neutron cross sections could lead to a significant difference in the calculated photon dose.

KAPV calculated data are also shown on Figure 8-43. The KAP-V data agree very well with the DOT, ANISN, and experimental results, both at the top of the assembly and internal to the assembly.

Figure 8-45 provides a comparison between calculated and experimental photon absorbed dose rate as a function of radial position at four axial locations in Configuration 4B. From this figure, it can be seen that the DOT and KAP V calculations agree with the experimental results. This was also the case for the axial traverse of the data (Figure 8-43).

8.2.5 Summary of PAX-E5 Results

Section 8.2.4 provided a detailed discussion of the comparison between calculations and experimental data, and a discussion of the differences in the various calculations that were performed. The following is a summary and some generalized conclusions with regard to these results:

- 1) The calculations predicted the shape of the experimental results for all detectors in both the 4B and 7D configurations; however, the absolute magnitude of the calculated results differed from the experimental results for the following detectors:
 - a) Cadmium covered dysprosium data in Configuration 4B,
 - b) $U-238$ data in Configuration 7D,
 - c) Sulfur data in Configuration 4B, and
 - d) Photon absorbed dose rate data in Configuration 7D.

- 2) ANISN calculations differed significantly from DOT calculations for low energy neutron detectors only. The DOT calculations displayed better agreement with the experimental results. This situation may be a result of a discrepancy in the buckling(transverse leakage) required by ANISN but not in DOT.
- 3) Neutron results from ANISN calculations employing P_0 (transport corrected) neutron cross sections agreed with those based on P_1 neutron cross sections for Configuration 4B but differed from one another for Configuration 7D. It is not possible to determine which set of cross sections provided better agreement with experimental data because there is no consistent trend in the agreement of the calculated and measured results.
- 4) Neutron data from KAPV (point kernel) agreed well with experimental results at the top of the 4B Configuration and were comparable to the data obtained from DOT.
- 5) Photon absorbed dose rate data from KAP V and DOT were comparable.
- 6) The test cell environment may have affected all experimental results obtained at the top of Configuration 7D where this factor would be a significant portion of the total measured effect.
- 7) The test cell environment and the use of a circular geometry to represent the square mock-up assembly may have been the cause of the difference between the calculated and experimental data at the edge of the mock-up assembly in the radial direction. This difference occurred only for low energy dysprosium neutron data.

8.3 EVALUATION OF ACCURACY OF RADIATION ENVIRONMENT CALCULATIONS

The evaluation of the accuracy of Radiation Environment Calculations has been divided into two categories:

- 1) A qualitative discussion of the results obtained from the analysis described above for the FCX and PAX-E5 reactors, and
- 2) A summary of recently published ^{56, 57} statistical analysis performed for a calculation/measurement comparison on the PAX-GIA reactor.

8.3.1 FCX and PAX-E5 Reactors

The analysis of the internal neutron activation data for the FCX reactor indicated that there was generally good agreement between the calculations and the measurements except at the edges of the assembly. This was particularly the case near the upper surface of the reactor where the presence of the control rods (which were partially inserted into the core for these measurements) caused greater differences in the comparisons at their location (the central portion of the reactor) than was observed in the reflector. The errors in the calculated results near the assembly edges are thus attributable not only to nuclear modeling, but also to the simulation of the test cell environment in the analytical model and to lack of simulation of the control rods.

The FCX analysis also revealed that the calculated and experimental results for both Dy and S had approximately the same shape radially, but the calculations were consistently lower for both detectors.

In the case of the PAX-E5 radiation environment measurements, the calculations predicted the shape of the experimental results for all detectors in both the 4B and 7D configurations; however, the absolute magnitude of the calculated results differed from the experimental results for the following detectors:

- a) Cadmium covered dysprosium data in Configuration 4B,
- b) U-238 data in Configuration 7D,
- c) Sulfur data in Configuration 4B, and
- d) Photon absorbed dose rate data in Configuration 7D.

Note that the results for sulfur and dysprosium confirm the situation observed in the FCX comparisons.

Examination of the comparisons for PAX-E5 also results in the following conclusions:

- 1) ANISN calculations differ significantly from DOT calculations for low energy neutron detectors only. The DOT calculations displayed better agreement with the experimental results. This situation may be a result of a discrepancy in the buckling (transverse leakage) required in ANISN but not in DOT.

- 2) Neutron results from ANISN calculations employing P_0 (transport corrected) neutron cross sections agreed with those based on P_1 neutron cross sections for Configuration 4B but differed from one another for Configuration 7D. It is not possible to determine which set of cross sections provided better agreement with experimental data because there is no consistent trend in the agreement of the calculated and measured results.
- 3) Neutron data from KAPV agreed well with experimental results at the top of the 4B Configuration and were comparable to the data obtained from DOT.
- 4) Photon absorbed dose rate data from KAPV and DOT were comparable.
- 5) The test cell environment may have affected all experimental results obtained near the top of Configuration 7D where this factor would be a significant portion of the total measured effect.
- 6) The test cell environment and the use of a circular geometry to represent the square mock-up assembly may have been the cause of the difference between the calculated and experimental data at the edge of the mock-up assembly in the radial direction. This difference occurred only for low energy dysprosium neutron data.

8.3.2 Statistical Analysis of PAX-G1A Results

This section presents selected results from a recently published statistical comparison of radiation environment calculations and measurements internal to the PAX-G1A reactor. References 56 and 57 provide detailed comparisons and descriptions of these analyses. Four items are discussed:

- 1) The best functional form representing the difference between calculations and experiments,
- 2) The average difference between calculations and measurements ("Average Analytical Bias" of Figure 1-1),
- 3) The one sigma limits on the average difference ("Variability" of Figure 1-1), and

- 4) A statistical indication of whether the shape of the calculated and experimental data is the same.

Part of the statistical analysis of the difference between the calculated and experimental data was performed with a graphical method. This technique was employed because it is relatively easy to apply and yields information which is very difficult to obtain from a direct numerical analysis.

The graphical analysis consists of plotting the difference between the calculated and experimental data as a function of the cumulative percent on a probability scale. If the differences between calculations and experiment have a normal distribution, the plot will represent the integral of a normal distribution curve. The probability scale is employed because the integral of a normal distribution will appear as a straight line in the plot. Therefore, the best straight line through the data will provide the following information:

- 1) The value of the difference between the calculated and experimental data obtained from the straight line at a cumulative probability of 50% is the average value of the difference ("Average Analytical Bias" of Figure 1-1). This occurs because the average value of a normal distribution occurs at the 50% location in the integral of the distribution.
- 2) The difference between the values on the straight line at the 50% and 84% locations (or the 16% and 50% locations) provides the value of one standard deviations in the differences between calculations and experiments. This occurs because in a normal distribution 68% (plus and minus 34%) of the data points fall within one standard deviation of the average of the distribution. Therefore, in the integral of a normal distribution plotted as a function of cumulative percent, 68% of the data lie between the 16% and 84% locations.
- 3) The ability of a straight line to represent the data provides an indication of whether the variations in the differences between calculated and experimental data are randomly distributed. If the data follow a straight line closely, then the variation in the differences is random and there is no

statistically significant difference between the shape of the calculated data and the shape of the experimental data (note that this does not rule out the possibility of an average difference, or constant bias, between the two sets of data).

Because of the accuracy required in these comparisons, the average difference and standard deviation were determined numerically and employed to plot the best straight line. Therefore, the primary use of the plots was to determine the correlation between the shapes of the two sets of data. This analysis is impossible to perform numerically for a function of two variables (even for one variable) which is represented by a high order transcendental function when there are not considerably more data available.

By examining the plots of the difference between the calculated and experimental data, bad data points will immediately become apparent and can be excluded from the analysis. In addition, the data over the "entire reactor"* were first plotted with the location in the reactor from which the data were obtained being tagged. By examining this plot, it was possible to determine whether the experimental and calculated data agreed in either magnitude, or shape, or both over portions of the reactor but not over the entire reactor. If this appeared to be the case, then the plot was redone over sections of the reactor.

Because of the amount of effort involved, and because the two variables (r and z) would confuse and complicate the analysis, the midplane data as a function of radius were analyzed to obtain average differences, standard deviations, and comparisons of shape. Selected axial shapes were examined, and no differences in the shapes of the data were found which could not be attributed to either random variations or differences such as test cell environment reflection which was not represented in the calculations.

In determining the difference between calculated and experimental data, it was first necessary to determine the best analytical form for the difference because the choice of an incorrect form would yield differences which did not have a normal distribution and, hence, the averages and standard deviations would not be meaningful.

*Entire reactor is defined here as the active core, filler strip, lateral support, and reflector.

In order to determine the form of the difference, the data for sulfur were analyzed using different forms. The sulfur data were chosen because the shape is representative of most of the shapes obtained, and visual comparison of the calculated and experimental data indicated that the two sets of data had the same shape, with a constant difference between the two sets of data. The following forms for the difference between the two sets of data were tested:

E = experimental value

C = calculated value

1) $E - C$

2) $\frac{E - C}{E}$

3) $\log (E - C)$

4) $\log E - \log C.$

Of these four forms for the difference between calculated and experimental data, the form of the difference represented by $\log E - \log C$ was chosen to perform the analysis for all data because the statistical analysis of the sulfur data employing this form indicated that the difference between calculated and experimental data had a normal distribution. This was determined by plotting the differences obtained for the four forms on a probability scale and determining which form yielded the best straight line. The difference between the calculated and experimental sulfur data using analytical form four is plotted as an example in Figure 8-46, and, as can be readily seen, a straight line, which indicated a normal distribution, quite readily fits the data.

It should be emphasized that form four was chosen strictly on a statistical basis; however, the use of this form led to the use of simple multiplicative uncertainty factors to describe possible variations in radiation levels. This is demonstrated in Ref. 56.

Figures 8-47 through 8-49 are additional examples of the probability plots for the differences between the calculated and experimental reactor midplane data in Ref. 56. Table 8-2 provides a summary of PAX-G1A core midplane average differences between the calculation and measurement and the one sigma limits on this difference. As an example

of the use of these data, Table 8-2 indicates that, on the average over the reactor, the calculated neutron absorbed dose rate was 2% less than the experimental data. Further, the experimental data lay between the product of 0.89 and 1.16 times the calculated data for 68% of the observations.

Examination of the data indicates that it was necessary to subdivide the U-238 and bare dysprosium data into core and reflector data in an attempt to obtain an analytical shape which was comparable to the experimental shape. The remainder of the data could be analyzed on an "entire reactor" basis. It should be noted that the thermal flux ($E < 0.4$ eV) dysprosium data were divided into a core and a reflector component. This was done because the core midplane data were compared directly with experimental data, while in the reflector the calculated data were adjusted for azimuthal variations prior to comparison with the experimental data in the spring holes. Examination of the ($E < 0.4$ eV) dysprosium data in Table 8-2 indicates that the core and reflector calculated data have essentially the same average difference and standard deviation from experimental data. Therefore, only a single set of uncertainty factors which apply to the "entire reactor" will be employed with these data.

Figure 8-46 through 8-49 show that the difference between several types of calculated and experimental data is fitted reasonably well by a straight line. This indicates that variations in the difference are random, and therefore, the shape of the calculated data is the same as the shape of the experimental data. This conclusion applies to the following sets of data:

- 1) Sulfur data over the "entire reactor", fast neutron flux ($E > 2.9$ MeV)
- 2) U-238 data in the core, fast neutron flux ($E > 1.5$ MeV)
- 3) Dysprosium data over the "entire reactor," thermal neutron flux ($E < 0.4$ eV)
- 4) Fast neutron absorbed dose rate data over the "entire reactor "
- 5) Gamma ray absorbed dose rate data over the "entire reactor "
- 6) The Bragg-Gray heating rate data over the region in which it was measured

While the U-238 data in the reflector and the bare dysprosium data do not appear to be fitted by a straight line, it can only be concluded that the shapes of the calculated and experimental data may be different or that insufficient data were obtained to evaluate the hypothesis.

Another example of the accuracy of radiation environment calculations with WISDM is summarized in Table 8-3 which has been reproduced from Ref. 57. This table compares the computed and measured results for various neutron energy regions at locations internal and external to the Cluster Hot End Support Hardware (CHESH). These components are located outside the core fuel envelope at the aft end of the reactor. As in the case of the data appearing in Table 8-2, the first column of numerical information gives the "Average Analytical Bias" (Figure 1-1). The next two columns give the upper and lower one sigma limits for this bias.

TABLE 8-1

ACTIVATION DETECTOR CHARACTERISTICS

<u>Detector</u>	<u>Type</u>	<u>Main Resonance Energy</u>	<u>Threshold Energy</u>
Au ¹⁹⁷	1/V, resonance	4.9 eV	---
In ¹¹⁵	1/V, resonance	1.45 eV	---
Dy ¹⁶⁴	Thermal	---	---
Mn ⁵⁵	1/V, resonance	337 eV	---
W ¹⁸⁶	resonance	18.8 eV	---
Cu ⁶³	resonance	580 eV	---
U ²³⁸	threshold	---	1.3 MeV
S ³²	threshold	---	1.0 MeV

TABLE 8-2
STATISTICAL ANALYSIS OF THE DIFFERENCE BETWEEN CALCULATED AND EXPERIMENTAL DATA *

Data Type and Location	Average Difference Between Calculations and Experiments	One Standard Deviation in the Difference Between Calculations and Experiments	
		Upper Limit	Lower Limit
Neutron Dose Rate - Entire Reactor ⁽⁵⁾	1.02	1.16	0.89
Gamma Ray Dose Rate - Entire Reactor ⁽⁵⁾	0.96	1.02	0.90
U-238 Data ⁽¹⁾ - Core	0.96	0.97	0.95
U-238 Data ⁽¹⁾ - Reflector	0.84	0.89	0.79
Sulfur Data ⁽²⁾ - Entire Reactor ⁽⁵⁾	1.32	1.54	1.13
Bare Dysprosium Data - Core	1.22	1.53	0.96
Bare Dysprosium Data - Reflector	1.04	1.09	0.99
Dysprosium ($E < 0.4$ eV) Data ⁽³⁾ - Core	1.00	1.15	0.87
Dysprosium ($E < 0.4$ eV) Data ⁽³⁾ - Reflector	0.99	1.14	0.87
Beryllium Heating Rate ⁽⁴⁾ - Reflector	1.28	1.33	1.24
Carbon Heating Rate ⁽⁴⁾ - Core	1.17	1.19	1.16
Aluminum Heating Rate ⁽⁴⁾ - Reflector	1.24	1.28	1.21
Stainless Steel Heating Rate ⁽⁴⁾ - Core	1.03	1.06	1.00

- (1) Neutron Flux ($E > 1.5$ MeV)
 (2) Neutron Flux ($E > 2.9$ MeV)
 (3) Neutron Flux ($E < 0.4$ eV)
 (4) Photon Heating Rate Only

- (5) Entire reactor refers only to the location in which these measurements were performed, namely, the core, lateral support, and reflector.

*Source : Ref. 56.

TABLE 8-3

STATISTICAL ANALYSIS OF THE DIFFERENCE BETWEEN CALCULATED AND EXPERIMENTAL DATA *

Data Type and Location	Average Ratio of Experimental to Calculated Data	One Standard Deviation in the Ratio Between Calculated and Experimental Data	
		Upper Limit	Lower Limit
U-238 Data ⁽¹⁾ - Internal to CHESH	0.81	0.83	0.80
U-238 Data ⁽¹⁾ - At CHESH Surface	0.83	0.84	0.81
Sulfur Data ⁽²⁾ - Internal to CHESH	1.60	1.66	1.54
Sulfur Data ⁽²⁾ - at CHESH Surface	1.64	1.73	1.55
Dy Thermal Flux Data ⁽⁴⁾ - Internal to CHESH	0.24	0.29	0.19
Dy Thermal Flux Data ⁽⁴⁾ - at CHESH Surface	0.24	0.37	0.16
Phylatron Data ⁽⁵⁾ - Internal to CHESH	0.68	0.77	0.60
Phylatron Data ⁽⁵⁾ - at CHESH Surface	0.58	0.65	0.52
CaF ₂ TLD Data ⁽⁶⁾ - Internal to CHESH	0.85	1.01	0.71
CaF ₂ TLD Data ⁽⁶⁾ - at CHESH Surface	0.82	1.22	0.55

(1) Neutron Flux ($E > 1.5$ MeV)

(6) Photon Dose Rate

(2) Neutron Flux ($E > 2.9$ MeV)(4) Neutron Flux ($E < 0.4$ eV)

(5) Neutron Dose Rate

*Source: Ref 57.

Figure 8-1. Comparison of FCX B Series of Measured and Calculated Radial Saturated Activities of Bare and Cadmium Covered Au Foils in FCX at Midplane (0.5 Mil Foils)

Figure 8-2. Comparison of FCX A Series of Measured and Calculated Radial Saturated Activities of Bare and Cadmium Covered Au Foils at Midplane (5.0 Mil Foils)

Figure 8-3. Comparison of FCX B Series of Measured and Calculated Axial Saturated Activities of Bare and Cadmium Covered Au Foils (Calculated with 16 Group DOT for 0.5 Mil Foils)

Figure 8-4. Comparison of FCX B Series of Measured and Calculated Axial Saturated Activities of Bare and Cadmium Covered Au Foils (Calculated with 16 Group DOT for 0.5 Mil Foils)

Figure 8-5. Comparison of FCX A Series of Measured and Calculated Axial Saturated Activities of Bare and Cadmium Covered Au Foils (Calculated with 16 Group DOT for 5.0 Mil Foils)

Figure 8-6. Comparison of FCX A and FCX B Series of Measured and Calculated Radial Saturated Activities of Bare and Cadmium Covered In Foils in FCX Midplane

Figure 8-7. Comparison of FCX B Series of Measured and Calculated Axial Saturated Activities of Bare and Cadmium Covered in Foils (Calculated with 16 Groups DOT)

Figure 8-8. Comparison of FCX B Series of Measured and Calculated Axial Saturated Activities of Bare and Cadmium Covered in Foils (Calculated with 16 Group DOT)

Figure 8-9. Comparison of FCX A Series of Measured and Calculated Axial Saturated Activities of Bare and Cadmium Covered in Foils (Calculated with 16 Group DOT)

Figure 8-10. Comparison of FCX B Series of Measured and Calculated Radial Saturated Activities of Bare and Cadmium Covered Dy Foils in FCX at Midplane

Figure 8-11. Comparison of FCX A Series Measured and Calculated Radial Saturated Activities of Bare and Cadmium Covered Dy Foils

Figure 8-12. Comparison of FCX B Series of Measured and Calculated Axial Saturated Activities of Bare and Cadmium Covered Dy Foils
(Calculated with 16 Group DOT)

Figure 8-13. Comparison of FCX B Series of Measured and Calculated Axial Saturated Activities of Bare and Cadmium Covered Dy Foils
(Calculated with 16 Group DOT)

Figure 8-14. Comparison of FCX A Series of Measured and Calculated Axial Saturated Activities of Bare and Cadmium Covered Dy Foils
(Calculated with 16 Group DOT)

Figure 8-15. Comparison of FCX A Series of Measured and Calculated Radial Saturated Activities of Bare and Cadmium Covered Mn Foils in FCX at Midplane

Figure 8-16. Comparison of FCX A Series of Measured and Calculated Axial Saturated Activities of Bare and Cadmium Covered Mn Foils
(Calculated with 16 Group DOT)

Figure 8-17. Comparison of FCX A Series of Measured and Calculated Radial Saturated Activities of Bare and Cadmium Covered W Foils in FCX at Mid-plane (Normalized to 1.0 at the Centerline)

Figure 8-18. Comparison of FCX, A Series of Measured and Calculated Axial Saturated Activities of Bare and Cadmium Covered W Foils (Normalized to 1.0 at the FCX MidPlane)

Figure 8-19. Comparison of FCX A Series of Measured and Calculated Axial Saturated Activities of Bare and Cadmium Covered W Foils (Normalized to 1.0 at the FCX Midplane)

Figure 8-20. Comparison of FCX B Series of Measured and Calculated Radial Saturated Activities of Bare and Cadmium Covered Cu Foils in FCX at Midplane

Figure 8-21. Comparison of FCX B Series of Measured and Calculated Axial Saturated Activities of Bare and Cadmium Covered Cu Foils (Calculated with 16 Group DOT)

Figure 8-22. Comparison of FCX B Series of Measured and Calculated Axial Saturated Activities of Bare and Cadmium Covered Cu Foils (Calculated with 16 Group DOT)

Figure 8-23. Comparison of FCX A and FCX B Series of Measured and Calculated Radial Saturated Activities of Bare S Tablets

Figure 8-24. Comparison of FCX A and FCX B Series of Measured and Calculated Axial Saturated Activities of Bare S Tablets (Calculated with 16 Group DOT)

Figure 8-25. Comparison of FCX B Series of Measured and Calculated Axial Saturated Activities of Bare 5 Tablets (Calculated with 16 Group DOT)

Figure 8-26. Comparison of FCX B Series of Measured and Calculated Axial Saturated Activities of Bare S Tablets (Calculated with 16 Group DOT)

Figure 8-27. Comparison of Experimental and Calculated Saturated Bare Gold-197 Activity (Thermal Neutron Flux) vs Axial Position on the Reactor Centerline, Configuration 4B

Figure 8-28. Comparison of Experimental and Calculated Saturated Cadmium Covered Gold-197 Activity vs Axial Position on the Reactor Centerline, Configuration 4B

Figure 8-29. Comparison of Experimental and Calculated Saturated Bare Gold-197 Activity vs Axial Position on the Reactor Centerline, Configuration 7D

Figure 8-30. Comparison of Experimental and Calculated Saturated Cadmium Covered Gold-197 Activity vs Axial Position on the Reactor Centerline, Configuration 7D

Figure 8-31. Comparison of Experimental and Calculated Saturated Bare Dysprosium-164 Activity vs Axial Position on the Reactor Centerline, Configuration 4B

Figure 8-32. Comparison of Experimental and Calculated Saturated Cadmium Covered Dysprosium-164 Activity vs Axial Position on the Reactor Centerline, Configuration 4B

Figure 8-33. Comparison of Experimental and Calculated Bare Dysprosium-164 Activity vs Radial Distance from the Reactor Centerline, Configuration 4B

Figure 8-34. Comparison of Experimental and Calculated U-238 Fission Rate (Fast Neutron Flux, $E > 1.5$ MeV) vs Axial Position on the Reactor Centerline, Configuration 4B

Figure 8-35. Comparison of Experimental and Calculated U-238 Fission Rate vs Axial Position on the Reactor Centerline, Configuration 7D

Figure 8-36. Comparison of Experimental and Calculated U-238 Fission Rate (Fast Neutron Flux $E > 1.5$ MeV) vs Radial Distance from the Reactor Centerline, Configuration 4B

Figure 8-37. Comparison of Experimental and Calculated Saturated Sulfur Activity (Fast Neutron Flux, $E > 2.9$ MeV) vs Axial Position on the Reactor Centerline, Configuration 4B

Figure 8-38. Comparison of Experimental and Calculated Saturated Sulfur Activity vs Axial Position on the Reactor Centerline, Configuration 7D

Figure 8-39. Comparison of Experimental and Calculated Saturated Sulfur Activity vs Radial Distance from the Reactor Centerline Configuration 4B

Figure 8-40. Comparison of Experimental and Calculated Neutron Absorbed Dose Rate vs Axial Position on the Reactor Centerline, Configuration 4B

Figure 8-41. Comparison of Experimental and Calculated Neutron Absorbed Dose Rate vs Axial Position on the Reactor Centerline, Configuration 7D

Figure 8-42. Comparison of Experimental and Calculated Neutron Absorbed Dose Rate vs Radial Distance from the Reactor Centerline, Configuration 4B

Figure 8-43. Comparison of Experimental and Calculated Gamma Ray Absorbed Dose Rate vs Axial Position on the Reactor Centerline, Configuration 4B

Figure 8-44. Comparison of Experimental and Calculated Gamma Ray Absorbed Dose Rate vs Axial Position on the Reactor Centerline, Configuration 7D

Figure 8-45. Comparison of Experimental and Calculated Gamma Ray Absorbed Dose Rate vs Radial Distance from the Reactor Centerline, Configuration 4B

Figure 8-46. Probability Plot of Sulfur Data

Figure 8-47. Probability Plot of Neutron Dose Rate Data

Figure 8-48. Probability Plot of Gamma Ray Dose Rate Data

Figure 8-49. Probability Plot of Beryllium Heating Rate Data

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9.0 SHIELD EFFECTIVENESS CALCULATIONS

This section presents a summary of shield effectiveness results published in Reference 58, "Nuclear Subsystem Shielding Provision Study Experiments" (WANL-TME-1912). The analysis allows evaluation of the accuracy of WISDM calculations both internal and external to a typical R-1 type shield. In addition, the evaluation is divided into neutron and photon related effects.

Section 9.1 contains brief introductory material. In Section 9.2, an evaluation of WISDM computational accuracy in shield effectiveness calculations is summarized from the statistical analysis of Reference 58.

9.1 INTRODUCTION

A series of experiments was performed at WANEF to provide data that would allow the WISDM shielding effectiveness calculations to be evaluated for two types of R-1 internal shield mock-ups. These experiments provided information on the internal and external radiation environments of 20.32, 30.48, and 38.10 cm thick Boron Carbide-Aluminum-Titanium Hydride (BATH) shield mock-ups and a Borated Stainless Steel-Liquid Hydrogen shield mock-up positioned on the aft end of the PAX-GOC reactor.⁵⁸ Measurements were made to determine the radiation environment internal to the simulated shield, using passive dosimetry to measure fast neutron fluxes, thermal neutron fluxes, and gamma ray-absorbed dose rates. Measurements of the external radiation environment were made with active dosimetry to determine fast neutron fluxes and dose rates, and gamma ray-absorbed dose rates.

An in-depth analytical evaluation of the experimental assemblies was performed using the WISDM system appearing in Figures 3-12 and 3-13. Results of both one- and two-dimensional discrete ordinates calculations of neutron and photon radiation levels internal to the various simulated shields appear in Reference 58. Two-dimensional analysis is summarized here. Likewise, analytical results external to the experimental configurations are presented for both discrete ordinates and point kernel methods.

9.2 EVALUATION OF THE ACCURACY OF SHIELD EFFECTIVENESS CALCULATIONS

The evaluation presented in the following paragraphs is a summary of material appearing in Reference 58.

The method used for evaluation of the accuracy of shield effectiveness calculations is briefly:*

- (1) Perform detailed comparison of measurements and calculations to identify errors in analytical results
- (2) Perform appropriate averaging of these errors to develop uncertainty factors which are the average analytical bias and variability in this bias.

These uncertainty factors can then be applied to calculated radiation environment to estimate possible variations in calculated results. Section 5.5 of Reference 56 (WANL-TME-1914, Radiation Environment in Various NERVA Components) contains a detailed discussion of the manner in which the statistical analysis was performed to obtain the uncertainty factors to be applied to the calculated environment.

9.2.1 Internal Radiation Environment

Table 9-1 summarizes the nominal uncertainty factor and its one-sigma limits for shield effectiveness calculations internal to the shield mock-ups. The uncertainty factors are provided for both the neutron and photon environment determined internal to shield Configuration 16C, the ~15 inch BATH shield mock-up, and shield Configuration 17, the borated steel-LH₂ shield mock-up. As an example of the use of these data, Table 9-1 indicates that the nominal uncertainty between the calculated and experimental sulfur data in shield Configuration 16C is 1.08. Further, the experimental data lay between the product of 1.16 and 1.01 times the calculated data for 68 percent of the observations.

*See Section 5.5, Reference 56 for a detailed description.

An examination of the nominal uncertainty factor indicates that the difference in the nominal uncertainty factor for the neutron data between the two shield configurations is within two standard deviations of one another. The difference in the nominal uncertainty for the photon TLD data between the two shield configurations is not within three standard deviations of one another. This implies that the ability to calculate the neutron environment in both shield configurations is similar; however, the ability to calculate the photon environment in the borated steel-LH₂ shield mock-up appears to be significantly different.

It should be noted that an average ratio of calculated to experimental results not equal to unity may be due to causes other than a discrepancy between the calculated and actual fluxes.* Therefore, the uncertainty in the calculated environment may be more appropriately represented by a modification to the upper and lower limits that have been assigned to the individual items. The limits are modified so that the nominal correction would be unity in Table 9-1 by dividing the one-standard-deviation upper limit in the ratio of experimental to calculated data by the average ratio of experimental to calculated data. The following equation provides the formulation for determining other than one-standard-deviation uncertainty factors, F , from this information:

$$F = S^m$$

where

F = the uncertainty factor

m = the desired level of uncertainty (i. e. , 1.0 sigma, 2.3 sigma, 3.0 sigma, -1.0, -2.3, -3.0, etc.)

$$S = \frac{\text{One Standard Deviation Upper Limit of Ratio}}{\text{Average Ratio of Experimental to Calculated Data}}$$

*See Section 5.3.2.1, Reference 56.

9.2.2 External Radiation Environment

Uncertainty factors for calculated radiation environment external to a shield envelope were also developed. * Nominal uncertainty values appear in Table 9-2 for two general locations on a plane tangent to the PVARA (dome) at a distance of 63.0 inches from the R-1 core center. Factors are shown for use with two different analytical techniques: Discrete Ordinate and Point Kernel.

From Table 9-2, it is seen that on the plane in question, uncertainty factors applicable to fast neutron calculations vary from 1.3 to 1.8 for the discrete ordinates method and from 1.8 to 3.1 for the point kernel technique. For photon analyses a slight reversal is observed. That is, for the discrete ordinates approach the uncertainty factor ranges from 1.7 to 2.0, while for the point kernel method the variation is from 1.2 to 1.4.

*An additional extrapolation was required so that the factors would be applicable to an R-1 size shield. See Reference 58 for details.

TABLE 9-1
UNCERTAINTY FACTORS FOR THE DIFFERENCE BETWEEN CALCULATED
AND EXPERIMENTAL DATA INTERNAL TO THE SHIELD MOCK-UPS*

Shield Configuration and Data Type	Average Ratio of Experimental to Calculated Data	One Standard Deviation in the Ratio of Experimental to Calculated Data	
		Upper Limit	Lower Limit
Shield Configuration 16C			
Sulfur Data ⁽¹⁾	1.08	1.16	1.01
U-238 ⁽²⁾	0.82	1.08	0.61
Dy Thermal Flux ⁽³⁾	0.60	0.83	0.43
CaF ₂ TLD Data ⁽⁴⁾	0.92	0.96	0.88
Shield Configuration 17			
Sulfur Data ⁽¹⁾	0.91	1.04	0.79
U-238 Data ⁽²⁾	1.25	1.74	0.91
Dy Thermal Flux Data ⁽³⁾	1.12	1.27	0.99
CaF ₂ TLD Data ⁽⁴⁾	0.62	0.67	0.58

1) Neutron Flux E > 2.9 MeV

2) Neutron Flux E > 1.5 MeV

3) Thermal Neutron Flux E < 0.4 eV

4) Photon Dose Rate

*Source: Reference 58.

TABLE 9-2
UNCERTAINTY FACTORS TO BE APPLIED TO CALCULATED RADIATION LEVELS
FORWARD OF THE R-1 REACTOR*

	DISCRETE ORDINATES		POINT KERNEL	
	NEAR REACTOR CENTERLINE	NEAR REACTOR PERIPHERY	NEAR REACTOR CENTERLINE	NEAR REACTOR PERIPHERY
FAST NEUTRONS	1.3	1.8	1.8	3.1
PHOTONS	1.7	2.0	1.2	1.4

*Source: Reference 58

Tabulated factors are applicable to computed radiation environment on a plane tangent to the PVARA dome at a distance of 63.0 inches from the core center.

10.0 REFERENCES

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14. Glasstone and Edlund, "The Elements of Nuclear Reactor Theory," D. Van Nostrand Co., Inc., 1952.
15. WANL-TME-1910, "PAX-G0 Control Drum Spans and Reactivity Coefficient Measurements," 1/70.
16. WANL-TME-1916, "PAX-G1 Reflector Interchange, Control Drum Span, and Aluminum Reflector Barrel Experiments," 1/70.

17. WANL-TMI-358, "Calculation of Resonance Integrals, With Emphasis on U^{235} and U^{238} , Including Doppler Broadening," D. W. Drawbaugh and W. C. McCune, 1962.
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20. WANL-TME-1228, "Changes in the GAM-TNS Program and Cross Section Library," G. Gibson, M. Melnick, W. Gray, 1965.
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27. "An IBM-704 Program for Calculating Effective Absorption Cross Sections by the Narrow Resonance or Narrow Resonance Infinite Mass Approximations," D. Drawbaugh and G. Sandowsky, CEND 94.
28. WANL-TME-263, "TNS, An IBM-7090 Program for Computing Thermal Neutron Spectra and Multithermal Group Cross Sections," C. A. Stevens and S. Young, 1962.
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30. R. L. Murray et al., "Geometric Buckling of Polygonal Reactors," Nuclear Science and Engineering 34, No. 1, 1968.
31. D. J. Behrens, "The Effect of Holes in a Reacting Material on the Passage of Neutrons," Proceedings of Physical Society 62A, 607, 1949.
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33. R. B. Meghrebian and D. K. Holmes, "Reactor Analysis," McGraw-Hill Book Company, Inc., New York (1960).
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37. WANL-TME-2724, "LOBO--A Computer Program to Azimuthally Mesh and Approximate Circles, Annuli and Partial Annuli," 8/70.
38. Unpublished work by P. Sipush and E. Iwinski.
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41. J. R. Stehn et al., "Neutron Cross Sections," Brookhaven National Laboratory, BNL-325, Second Edition, Supplement 2, 1966.
42. F. Domanic and V. L. Sailor, Physical Review, 119, 208, 1960.
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44. J. B. Garg et al., Physical Review, 134, B985, 1964.
45. G. Gibson et al., "A Single Level Analysis of U^{235} Based on Recent σ_t , σ_f and σ_c Measurements; A n + U^{235} Multigroup Cross-Section Library," WANL-TME-1586, 1967.
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51. H. H. Hummel and D. Okrent, "Reactivity Coefficients in Large Fast Power Reactors," American Nuclear Society Monograph, 1970.
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57. WANL-TME-1915, "Radiation Environment on CHESH Components," 8/70.
58. WANL-TME-1912, "Nuclear Subsystem Shielding Provision Study Experiments," 9/70.
59. WANL-TME 2754, "Structural Analysis Report Nuclear Subsystem," 2/71.

APPENDIX A

See VOLUME 2

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APPENDIX B

CONVERGED SPECTRA

This appendix contains tabular and graphical information related to the flux spectra used for analysis of the reactor systems in this report. Space limitations do not allow a complete documentation of all of the WISDM converged spectra used for cross-section averaging; however, a comprehensive sampling has been included. The graphical information shows:

- (1) Spectral changes in a reactor assembly (Figures B-1, B-2, B-3, B-12, B-13, B-14, B-15, B-16)
- (2) The effect of spectral reweighting (Figures B-4, B-5, B-6, B-7, B-8, B-9)
- (3) The adequacy of the PAX-E1 mock-up of the NRX-A6 reactor (Figures B-10 and B-11)
- (4) The similarity of the FCX and PAX reactor series (Figure B-17).

Graphs in the second area show both the need for spectral reweighting and the effect of several iterations. Information such as this and that appearing in Tables B-1 through B-4 were the basis of the adoption of two iterations as the WISDM standard. This is especially the case in the Table B-4 information where percent differences between the second and the third iteration (for FCX) fluxes are shown to be generally not significant. Economy and scheduling have, of course, been considered in the selection of two iterations.

Each of the spectra shown in the figures has been normalized to unity to allow meaningful direct comparisons.

TABLE B-1
SPECTRUM CONVERGENCE DATA FOR NRX-A6 REACTOR
CALCULATIONS

<u>Iteration Number</u>	<u>Eigenvalue</u>	<u>*Change in Eigenvalue, cents</u>
1**	1.033799	---
2	1.030971	-39.8
3	1.030955	- 0.2

* This is the change in eigenvalue relative to the previous iteration. The conversion to cents is based on an effective "beta" equal to 0.0071.

** The cross sections used in the ANISN calculation for the first iteration were averaged over an infinite medium spectrum.

TABLE B-2
SPECTRUM CONVERGENCE DATA FOR PAX-E1 REACTOR CALCULATIONS

<u>Iteration Number</u>	<u>Eigenvalue</u>	<u>*Change in Eigenvalue, cents</u>
1**	1.042224	---
2	1.039464	-38.9
3	1.039113	- 4.9

* This is the change in eigenvalue relative to the previous iteration. The conversion to cents is based on an effective "beta" equal to 0.0071

** The cross sections used in the ANISN calculation for the first iteration were averaged over an infinite medium spectrum.

TABLE B-3

SPECTRUM CONVERGENCE DATA FOR FCX REACTOR CALCULATIONS

<u>Iteration Number</u>	<u>Eigenvalue</u>	<u>*Change in Eigenvalue, cents</u>
1**	1.027250	---
2	1.023842	48
3	1.023700	2

* This is the change in eigenvalue relative to the previous iteration. The conversion to cents is based on an effective "beta" equal to 0.0071.

** The cross sections used in the ANISN calculation for the first iteration were averaged over an infinite medium spectrum.

TABLE B-4

DETAILED REGIONWISE DATA ON FLUX CONVERGENCE
IN FCX SPECTRUM CALCULATIONS

Region (See Figure A4-5)	Average ¹ Percent Difference between Iteration 3 and 2 Flux Values		Largest Percent Difference between Flux Values found in Iteration 3 and 2 (Group Observed in)	
	<u>Fast Groups</u>	<u>Thermal Groups</u>	³ <u>Fast Groups</u>	<u>Thermal Groups</u>
Core-1 (1)	0.41	1.08	1.39 (46)	10.18(1) ²
Core-2 (2)	0.36	1.75	1.03 (46)	16.71(2)
Core-3 (2)	0.35	0.99	-0.97 (41)	8.21(1)
Reflector-1 (4)	0.40	0.63	-1.03 (41)	-2.28 (54)
Reflector-2 (5)	0.38	0.22	-1.17 (42)	1.20 (1)
Reflector-3 (6)	0.40	0.16	-1.18 (42)	0.30 (63)

¹ The averages listed here (output of the ANISIG-1 Code) are the standard deviation of the fine group percent differences.

² Fast Groups are numbered from the highest energy to the lowest. Thermal groups are numbered from the lowest energy to the highest.

³ The tabulation here is limited to the range 10 MeV to 1.86 eV (uppermost 62 GAM groups).

Figure B-1. Comparison of Converged Spectra in NRX-A6 - Core Center to Core Edge

Figure B-2. Comparison of Converged Spectra in NRX-A6 - Core Edge to Beryllium-3 Inner

Figure B-3. Comparison of Converged Spectra in NRX-A6 - Beryllium-3 Inner to Beryllium-3 Outer

Figure B-4. Effect of Spectral Reweighting - Infinite Medium (Zeroth Iteration) vs First Iteration on Core Centerline in PAX-E1

Figure B-5. Effect of Spectral Reweighting - First Iteration
vs Third Iteration on Core Centerline in PAX-51

Figure B-6. Effect of Spectral Reweighting - Infinite Medium (Zeroth Iteration) vs First Iteration on Core Periphery in PAX-E1

Figure B-7. Effect of Spectral Reweighting - First Iteration
vs Third Iteration on Core Periphery in PAX-E1

Figure B-8. Effect of Spectral Reweighting - Infinite Medium (Zeroth Iteration) vs Third Iteration in Filler Strip in PAX-E1

Figure B-9. Effect of Spectral Reweighting - Infinite Medium (Zeroth Iteration) vs Third Iteration in Beryllium Reflector (3 Inner) in PAX-E1

Figure B-10. Comparison of Converged Spectra - NRX-A6 and PAX-E1
in Center of Core

Figure B-11. Comparison of Converged Spectra - NRX-A6 and PAX-E1
in Reflector (3 Inner)

Figure B-12. Selected Converged Spectra in PAX-E5 - Core Center and Core Edge

Figure B-13. Selected Converged Spectra in PAX-E5 - Core Edge and Beryllium Reflector (3 Inner)

Figure B-14. Selected Converged Spectra in PAX-E5-Simulated Shield (Region 22, Figure A3-8) and the First BATH Shield Mock-up Module (Region 40, Figure A3-8)

Figure B-15. Comparison of Converged Spectra in FCX - Core Centerline to Core Edge

Figure B-16. Comparison of Converged Spectra in FCX - Core Edge to Reactor Edge

Figure B-17. Comparison of Converged Spectra on Reactor Centerline - FCX and PAX-E1

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APPENDIX C

NUCLEAR AND RADIATION CODE ABSTRACTS

<u>Code Abstract</u>	<u>Page</u>
ACT-II	C- 22
ANISIG-1	C-7
ANISN-W	C-9
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APPROPOS	C-10
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EXTERMINATOR-2	C-8
GAMBIT	C-4
GAMLEG	C-5
KAP-V	C-18
LINK	C-2
MAP	C-20
NAGS	C-13
PERT	C-17
PFENCER	C-15
R-102	C-21
SATURN	C-23
SCAP	C-19
S-4	C-24

LINK

Code Discs and Tapes: (CGBNØ1)(Tape: CGBNØ1D)
(MLJD, DISK) (Tape: SRN, ANISN, BINARY)
(MLASIG2, DISNEY) (Tape: MLASIG, DISNEY)

Computer: CDC-6600

LINK¹ is an automated system of separate codes that are used sequentially to produce spectrally converged neutron transport cross sections for use in one- and two-dimensional neutronic calculations. The system consists of three major programs: GAMB1T,² ANISN-W,³ and ANISIG-1.⁴ GAMB1T performs the cross section spectral weighting and group reduction from fine group data libraries.⁵ ANISN-W uses these cross sections in a one-dimensional transport theory calculation to calculate a revised few-group spectrum. ANISIG-1 takes this spectrum as input, generates a new fine group spectrum as required by GAMB1T, and computes the changes that have occurred in the spectrum relative to that used in the cross-section weighting and group reduction. The procedure may then be repeated any number of times desired. All data transfer is done automatically through magnetic discs or tapes.

References:

1. a. Internal Document (NRD-69-93), "LINK User's Manual," R. S. Kaiser, March 20, 1969.
1. b. Internal Document (NRD-70-287), "LINK User's Manual," R. S. Kaiser, September 21, 1970.
2. a. WANL-TME-1752, "GAMB1T Program," G. Collier, G. Gibson, 4/68.
2. b. WANL-TME-1969, "Second Version of the GAMB1T Code," G. Collier, G. Gibson, L. Moran, R. K. Disney, R. S. Kaiser, November 1969.
3. a. K-1693, "A User's Manual for ANISN, A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," W. W. Engle, Jr., 3/30/67.
3. b. Internal Document (WANL-TMI-1967), "Revised WANL ANISN Program User's Manual," R. G. Soltesz, April 1969.

4. a. Internal Document (NRD-68-614), "Transmittal of ANISIG-1 User's Manual," L. Ortenberg, 8/27/68.
4. b. Internal Document (NRD-69-37), "Revised Input Instructions for ANISIG-1," L. Ortenberg, 8/27/68.
5. The identification labels for the standard cross-section tapes and their corresponding disc files are:

FSGAM28, RD1, RD2	Disc: MLGAM28, RD1, RD2
CGB1T15	Disc: MLB1T15
CGGAS15	Disc: MLGAS15
GWQRY1, RD1, RD2	Disc: MLQRY1, RD1, RD2

GAMBIT

Code Tape: (CGBNØ1)

Code Disc: (CGBNØ 1D)

Computer: CDC-6600

GAMBIT¹ is a FORTRAN IV code that provides broad-group flux and current averaged neutron cross sections. They are based on a thermal cross-section library² consisting of 99 fine groups (0 - 2.38 eV), and a fast-neutron library³ consisting of 68 fine groups (0.414 eV - 10 MeV, 1/4 lethargy intervals). The fluxes and currents required to obtain the broad-group cross sections may be calculated with GAMBIT or may be entered into the code. In addition, a Maxwellian spectrum may be used in the thermal range. If the flux is calculated, either the B1 or P1 methods of solving the infinite medium transport equation may be used. The code provides cross sections for a full P(1) transport calculation or a P(0) transport corrected calculation. Output from GAMBIT is obtained on punched cards usable in EX TERMINATOR-2, ANISN-W, or DOT-IIW as well as magnetic tape for use in ANISN-W and DOT-IIW.

References:

1. a. WANL-TME-1752, "GAMBIT Program," G. Collier, G. Gibson, 4/68.
1. b. WANL-TME-1969, "Second Version of the GAMBIT Code," G. Collier, G. Gibson, L. Moran, R. K. Disney, R. S. Kaiser, November 1969.
2. Identification labels are: (CGB1T15)
(CGGAS15)
3. Identification Labels are (FSGAM28, RD1, RD2)
(GWQRY1, RD1, RD2)

GAMLEG

Code Tape: N/A* (Binary object deck and FORTRAN source deck on file in NRD)

Code Disk: N/A

Computer: CDC-6600

GAMLEG¹ is a WANL modified and corrected version of the original GAMLEG code² written by K. D. Lathrop of LASL. The code solves for multigroup photon transport cross sections in up to a P₅ scattering approximation using a numerical integration of the Klein-Nishina equation for photon inelastic scattering from a free electron to obtain Compton scattering cross sections.

The absorption cross sections for pair production and photoelectric effects are included as pointwise data from library decks or at option from a basic data tape.³ These data were obtained from Reference 4. Compton absorption cross sections are solved for by the code.

Solution of the scattering cross sections in the multigroup approximation is based on an energy flux solution such that the transport code using GAMLEG cross sections is based on energy flux.

Cross-section averaging in each group is based on a flux or source weighting to obtain group-averaged cross sections. In addition, the scattering of photons below a bottom-group cutoff energy is included as a pseudo-absorption in the bottom group to preserve energy.

GAMLEG is used to generate multigroup photon transport cross sections for use in APPROPOS, ANISN-W, DOT - IIW, or KAP - V. A microscopic multigroup set of GAMLEG photon transport cross sections exists on magnetic tape⁵ for 13 group design calculations.

References:

1. WANL-PR(LL)-034, Volume 3, "Cross-Section Generation and Data Processing Techniques," R. G. Soltész, R. K. Disney, S. L. Zeigler, August 1970.
2. LA-3267, "GAMLEG--A FORTRAN code to Produce Multigroup Cross Sections for Photon Transport Cross Sections," K. D. Lathrop, April 23, 1965.
3. Basic Data Tape: (ZS43, GAMLEG, LIBRARY)
*Not available.

4. LA-3753, "Photon Cross Sections from 0.001 to 100 MeV for Elements 1 through 100," E. Stern and H. I. Israel, November 15, 1967.
5. Standard 13 photon group GAMLEG produced data tape: (ZS48, 13GRP, EFXSEC).

ANISIG-1

Code Tape: (MLASIG, DISNEY)

Code Disc: (MLASIG2, DISNEY)

Computer: CDC-6600

The ANISIG-1¹ program serves two functions in the LINK technique² for the development of spectrally converged neutron cross-section data. First, the program provides neutron flux and current input in 161 neutron energy groups for spectral weighting in the GAMBIT³ program. Second, it computes the conventional standard deviation of the groupwise percent differences between the neutron flux and current in the "present" and the previous iteration in the spectrum convergence. Calculations are performed by reactor region in any of four broad-group neutron energy structures with option to use an input group structure. ANISIG-1 is coupled² to GAMBIT and ANISN-W⁴ for automated usage.

References:

1. a. Internal Document (NRD-68-814), "Transmittal of ANISIG-1 User's Manual," L. Ortenberg, 8/27/68.
1. b. Internal Document (NRD-69-37) "Revised Input Instructions for ANISIG-1," L. Ortenberg, 2/4/69.
2. a. Internal Document (NRD-69-93), LINK User's Manual," R. S. Kaiser, March 20, 1969.
2. b. Internal Document (NRD-70-287), LINK User's Manual," R. S. Kaiser, September 21, 1970.
3. a. WANL-TME-1752, "GAMBIT Program," G. Collier, G. Gibson, 4/68.
3. b. WANL-TME-1969, "Second Version of the GAMBIT Code," G. Collier, G. Gibson, L. Moran, R. K. Disney, R. S. Kaiser, November 1969.
4. a. K-1693, "A User's Manual for ANISN, A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," W. W. Engle, Jr., 3/30/67.
4. b. Internal Document (WANL-TMI-1967), "Revised WANL ANISN Program User's Manual," R. G. Soltesz, April 1969.

EXTERMINATOR-2

Code Tape: (ØL8, ØL3WITH, ØVERLAY)

Code Disc: N/A

EXTERMINATOR-2¹ is a FORTRAN-IV code that solves the multigroup two-dimensional neutron diffusion equations in (X, Y), (R, Z), or (R, θ) geometry. The code is variable dimensioned so that the only mesh point/group restriction on the size of a particular problem that the program can solve is the limitation imposed by the computer core size. The code is used in calculations of fission distribution and reactivity effects in R- θ models of design concepts and their alternates.

References:

1. ORNL-4078, "EXTERMINATOR-2: A FORTRAN IV Code for Solving Multigroup Neutron Diffusion Equations in Two Dimensions," R. B. Fowler et al., 4/67.

ANISN-W

Program Tape: (SRN, ANISN, BINARY)

Program Disk: (MLJD, DISK)

Computer: CDC-6600

ANISN-W¹ is a WANL modified version of the original ANISN code distributed by ORNL.² The code solves the one-dimensional discrete ordinate transport equations for slabs, cylinders, and spheres in multigroup form, using a general anisotropic scattering approximation represented by Legendre polynomials. The code is used to calculate neutron and photon flux distributions in multiregion problems based on eigenvalue, fixed distributed source, or fixed boundary source (shell source) solutions. Variable dimensioning of the computer program provides much flexibility in problem size. Various applications of the code to design problems are:

1. Forward and adjoint fluxes for perturbation theory analyses.
2. Neutron and photon fluxes for shield performance evaluation including heat generation rates, transmitted energy flux, and dose rates.
3. Eigenvalues for use in reactivity effects of design changes.

References:

1. Internal Document (WANL-TMI-1967), "Revised WANL ANISN Program User's Manual," R. G. Soltesz, April 1969.
2. K-1693, "A User's Manual for ANISN, A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," W. W. Engle, Jr., 3/30/67.

APPROPOS

Code Tape: (ZS69, NEW, APPRØ)

Code Disk: N/A

Computer: CDC-6600

The APPROPOS¹ code is a data processing link which prepares multigroup, simultaneous, neutron-photon cross-section data and neutron reaction cross-section data for radiation transport codes. The APPROPOS code processes GAMBIT produced multigroup, neutron cross-section data and multigroup, neutron reaction cross-section data (e. g. , radiative capture, fission, inelastic scatter, and elastic scatter) to provide spectral-weighted, neutron cross-section data in a reduced number of energy groups. These data are then processed with input-specified elemental atom densities, gamma ray production data due to neutron reactions, and GAMLEG produced photon cross-section data to provide macroscopic, coupled, neutron-photon cross-section data, neutron cross-section data, and photon cross-section data. Output data from the APPROPOS code are placed on punched cards and/or magnetic tape for use:

- 1) In simultaneous neutron and photon transport calculations in ANISN-W or DOT-IIW, or
- 2) In the NAGS code to do photon source calculations using ANISN-W or DOT-IIW scalar flux data tapes.

The APPROPOS code is the successor of the POINT² code and performs similar or expanded data processing functions in the design activities.

References:

1. WANL-PR(LL)-034, Volume 3, "Cross-Section Generation and Data Processing Techniques," R. G. Soltesz, R. K. Disney, S. L. Zeigler, August 1970.
2. WANL-PR(LL)-010, Volume 2, "Synthesis of Computational Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems," June 1967.

DOT-IIW

Code Tape: (DKDØTG, BINARY, PRØGRAM)

Disk: N/A

Computer: CDC-6600

DOT-IIW,¹ a FORTRAN-IV code, is a WANL modified version of the DOT-II code² which solves the linear, energy-dependent, Boltzmann transport equation for two-dimensional r-z, x-y, and r- θ geometries. The gradient or convection term in the Boltzmann equation is approximated by a finite difference technique known as discrete ordinates or Carlson's S_n method. The scatter integral is approximated by expanding the differential cross section in a Legendre series which allows the integral to be computed by quadrature. DOT-IIW will solve forward or adjoint, homogeneous or inhomogeneous problems. The inhomogeneous problems may have a volume distributed source or a specified angular flux at the right or top boundaries; fissions may be included for a subcritical system. Homogeneous (eigenvalue) problems will determine the following:

(1) Static multiplication factor, " k ," (2) Time absorption, "Rossi α ," (3) Concentration for a specified " k ," and (4) Zone thickness for a specified " k ."

DOT-IIW is a successor to the two-dimensional transport codes DOT, TDC, DDK, ODD-K and DDF. For those familiar with these codes, the primary differences between DDK and DOT-IIW are given in the following:

- (1) DOT-IIW is programmed in variable dimension FORTRAN-IV and is compatible with any system having FORTRAN-IV capability.
- (2) General anisotropic scattering is allowed.
- (3) Boundary sources may be treated by specifying the angular flux at the right or top boundaries.
- (4) Angular fluxes may be printed or written on binary tape.

The WANL version of the DOT-IIW code is an integral part of the two-dimensional transport analysis system capable of calculating the complete radiation environment (neutron, prompt and delayed gamma ray) of a nuclear system. WANL modifications to the original ORNL version of the DOT code include:

- 1) Use of asymmetric quadrature techniques in the Z direction
- 2) Complete automation with associated codes via magnetic tape
- 3) An assembly language version of subroutine GRIND which reduces running time by a factor of 2 over the all-FORTRAN version.

References:

1. WANL-TME-1982, "User's Manual for the DOT-IIW Discrete Ordinates Transport Computer Code," R. G. Soltesz, R. K. Disney, G. Collier, December 1969.
2. CTC-INF-952, "Development of Two-Dimensional Discrete Ordinates Transport Theory for Radiation Shielding," F. R. Mynatt, F. J. Muckenthaler, P. N. Stevens, August 1969.

NAGS

Code Tape: (ZS103, NAGS, SØURCE)

Code Disk: N/A

Computer: CDC-6600

The NAGS¹ code is a special purpose code which serves as a link between the neutron transport solution and the photon transport solution, using the discrete ordinates transport codes ANISN-W² and DOT-IIW.³ In addition, this program processes neutron and photon energy flux data simultaneously to calculate total heating and heating distributions in a reactor. The program also provides neutron and photon source distribution data for subsequent use in point kernel or Monte Carlo analyses.

The six basic operations of the NAGS program are:

- 1) Normalization of multigroup neutron and photon energy fluxes to units of per fission, per watt, or a selected power level.
- 2) Redefinition of the mesh cell description for use in photon problems.
- 3) Calculation of photon and neutron sources and distribution and heating rates for each region in the problem.
- 4) Calculation of the total photon and neutron sources and fissions in the reactor.
- 5) Preparation of fixed distributed neutron or photon source tapes for direct use in DOT-IIW.
- 6) Punched output of radial and axial source distributions by region, region averaged photon source spectra, and other region parameters for direct use in KAP-V.

References:

1. WANL-PR(LL)-034, Volume 3, "Cross-Section Generation and Data Processing Techniques," R. G. Soltesz, R. K. Disney, S. L. Zeigler, August, 1970.
2. Internal Document (WANL-TMI-1967), "Revised WANL ANISN Program User's Manual," R. G. Soltesz, April 1969.

3. WANL-TME-1982, "User's Manual for the DOT-IIW Discrete Ordinates Transport Computer Code," R. G. Soltesz, R. K. Disney, G. Collier, December 1969.

PFENCER

Code Tape: N/A (Binary object deck and FORTRAN source deck on file in NRD)

Code Disk: N/A

Computer: IBM-360 Model 75

PFENCER,¹ for Power Factor Evaluation and Normalization to Critical Experiments--Revised, is a group of three FORTRAN-IV programs which are used to calculate the steady-state power distribution across an NRX/R core during design operation. Using magnetic tape output from the EXTERMINATOR-2² program, PFENCER-1 calculates the fission reaction rate at each EXTERMINATOR-2 mesh point, interpolates this (R, θ) information to obtain a fission per cc fuel value at each coolant channel location and, then, converts these to a normalized fissions per gram uranium value. These data are used by PFENCER-2 to determine the ratio between the calculated fission/gram value and the corresponding experimental value measured in a reactor mock-up at WANEF. The output of PFENCER-2, correction factors for the calculated fission per gram values, is used by PFENCER-3 to compute the fission reaction rate at design conditions from PFENCER-1 information for the reactor at design conditions. All data transfer between the PFENCER programs is via magnetic tape.

References:

1. Internal Document (WANL-TMI-1999), "User's Manual, PFENCER-1, PFENCER-2, and PFENCER-3," B. Jacob, L. I. Ortenberg, C. P. Saalbach, January 1970.
2. ORNL-4078, "EXTERMINATOR-2: A FORTRAN-IV Code for Solving Multigroup Neutron Diffusion Equations in Two Dimensions," T. B. Fowler et al., April 1967.

AMDS

Code Tape: N/A (Binary object deck and FORTRAN source deck on file in NRD)

Code Disk: N/A

Computer: CDC-6600

The Automated Material Data System (AMDS) program¹ performs a number of calculations concerned with the total reactivity of one reactor system relative to another.

Calculations, which are limited to the reactor core, include the following:

- (a) Mass by material which is contained in regions defined at input, both actually and axially reactivity weighted for either a completely built core or a core in any stage of assembly.
- (b) The reactivity differences between the present core and any previous core for all of the materials involved.
- (c) Atom densities² corresponding to item (a).

Mass data are stored on magnetic tapes and are readily updated as more components become available.

References:

1. WANL-TME-1355, "NRX-A4/EST Reactivity Analysis with the Automated Material Data System," D. J. Hill, January 1966.
2. Internal Document (NRD-68-449), "Status of AMDS Computer Program," D. J. Hill, May 28, 1968.

PERT

Code Tape: N/A (Binary Object Deck and FORTRAN Source Deck on File in NRD)

Code Disk: N/A

Computer: CDC-6600

The PERT¹ code calculates the reactivity effects of small changes in reactor composition. The calculations are based on linear perturbation theory² and can be done in either one or two dimensions. One-dimensional calculations use forward and adjoint angular flux output tapes from ANISN-W³ to calculate the spatial distribution of the reactivity coefficient of any particular material as well as region and reactor average values. A similar capability is available in two dimensions⁴ using DOT-IIW⁵ magnetic tape output.

References:

1. a. ORNL 1968, "Unpublished PERT-1D Code," F. R. Mynatt.
1. b. Internal Document (NRD-68-607), "Instruction Manual for WANL PERT-1D," S. Salah, August 21, 1969.
2. G. E. Hansen and C. Maier, "Perturbation Theory of Reactivity Coefficients for Fast Neutron Critical Systems," NSE, 8, 532-542 (1960).
3. Internal Document (WANL-TMI-1967), "Revised WANL ANISN Program User's Manual," R. G. Soltesz, April 1969.
4. Internal Document (NRD-69-294), "PERT-2D Section of Project 493 Report," J. Jedruch, October 8, 1969.
5. WANL-TME-1982, "User's Manual for the DOT-IIW' Discrete Ordinates Transport Computer Code," R. G. Soltesz, R. K. Disney, G. Collier, December 1969.

KAP-V

Code Tape: (ZS34, KAPV, ØLD)

Code Disk: N/A

Computer: CDC-6600

KAP-V¹ is a successor to the QAD² series of point kernel codes. The code employs the point kernel method to calculate radiation levels at detector points located within or outside a complex radiation source geometry describable by a combination of intersecting quadratic surfaces. The KAP technique is used to calculate gamma ray and/or fast neutron flux, dose, or heating rates. The attenuation function, or kernel, for gamma rays employs exponential attenuation on the line-of-sight from a point source representing a finite volume of a source region to a detector point. Infinite media buildup for a specified material is applied on each line-of-sight to approximate multiple scattering effects. Fast neutron attenuation functions included are: (1) a modified Albert-Welton for fast neutron dose using neutron removal cross sections, (2) a bivariate polynomial expression for computing neutron spectra, using infinite media moments data, and (3) a monovariate polynomial expression for computing neutron spectra using infinite media moments data. The code contains gamma ray cross-section data and buildup data as subroutines. Source description can be cylindrical, spherical, disc, line, or point representations with a variety of distribution functions. The code accepts as input the NAGS punched data and at user option will produce punched data cards for direct use in the SCAP code.

References:

1. WANDL-PR(LL)-034, Volume 6, "Point Kernel Techniques," R. G. Soltesz, R. K. Disney, August 1970.
2. LA-3573, "QAD: A Series of Point Kernel General-Purpose Shielding Programs," Richard E. Malenfant, April 5, 1967.

SCAP

Code Tape: (ZS32, SCAP, PRØGRAM)

Code Disk: N/A

Computer: CDC-6600

The SCAP¹ code employs a single scatter point kernel method to calculate scattered radiation levels at detector points located within or outside a complex radiation source geometry describable by a combination of intersecting quadratic surfaces. The SCAP technique requires as input the energy and polar angle dependent meridian ring data for gamma rays from a KAP-V or MAP problem. These data are used to represent a pseudo point source of radiation. The integration of the scatter volume is handled by a trapezoidal technique in conjunction with an equal polar angle subdivision in spherical coordinates. Klein-Nishina equations are employed for gamma ray scattering and gamma ray absorption using GAMLEG or user input data. Output from the code includes direct, scatter by zone (with and without buildup on the scatter leg), scatter by polar angle, scatter by zone compositions, and scatter volumes. In addition, meridian ring data are integrated in fine detail for use in the scatter calculations.

The code has the complete geometry capability of the KAP-V code and employs variable dimensioning in FORTRAN for flexibility in geometry description.

Reference:

1. WANL-PR(LL)-034, Volume 6, "Point Kernel Techniques," R. G. Soltesz, R. K. Disney, August 1970.

MAP

Code Tape: (ZS97, MAPP, FIDØ)

Code Disk: N/A

Computer: CDC-6600

The MAP¹ code reconstructs the angular dependence of neutron or photon flux as a function of position along some arbitrary plane employing as input the reactor surface angular fluxes from the two-dimensional transport code, DOT. These reconstructed angular fluxes are then automatically used as a fixed boundary source in the DOT two-dimensional transport code, thus extending the geometrical capability of the two-dimensional transport analysis procedure. The MAP code is used to compute external environments of neutron and photons, using the entire energy and angular data at a DOT problem surface. The integration of particle leakage is handled by a numerical technique and energy dependent fluxes as well as response functions are available from the output.

References:

1. WANL-TME-2706, "Code Description and User's Manual for the MAP Radiation Transport Computer Code," R. K. Disney, R. G. Soltesz, J. Jedruch, S. L. Zeigler, August 1970.

R-102

Code Tape: N/A (Binary Program Deck and Source Deck on File in NRD)

Code Disk: N/A

Computer: CDC-6600

R-102¹ is a FORTRAN-IV code that solves the one energy group space-independent reactor kinetics equations in an inverse fashion. A library of delayed neutron precursor parameters is contained in the code. The code takes as input the measured time history of the nuclear power during an entire test series and calculates the reactivity changes that must have occurred to produce the measured power variations. The code is also used for analysis of a single test or portion of a test.

References:

1. a. ANL-7411, "Argonne Code Center: Compilation of Program Abstracts," M. K. Butler et al., January 1968 - Reactor Code Abstract 168, J. J. Kaganove.
1. b. Internal Document (RA-3571), "First Progress Report on R-102 Checkout," E. Iwinski, L. Ortenberg, November 10, 1966.
1. c. Internal Document (NRD-67-67), "Second Progress Report on R-102 Checkout," E. Iwinski, L. Ortenberg, January 1968.
1. d. Internal Document, (NRD-68-45), "Anomalous R-102 Calculation of NRX-A6 Drum Worth," L. Ortenberg, E. Iwinski, January 1968.

ACT-II (Revised)

Code Tape: N/A (Binary object deck and FORTRAN source deck on file in NRD)

Code Disk: N/A

Computer: CDC-6600

The ACT-II (Revised) code¹ calculates source strength (MeV/cc-sec) for both gamma ray emissions (in 4 energy groups) and beta particle emissions (in 5 energy groups) as a function of neutron exposure and decay time. In addition, the code calculates the activity in microcuries/cc of each radionuclide present in the activated material. Calculations may be performed for 48 different target elements for which a total of 223 nuclear reactions are possible with a total of 178 different radionuclides produced.

The ACT-II (Revised) code retains all options originally available in the ACT-I² and ACT-II² computer codes.

References:

1. WANL-TME-598, "Materials Activation Program - ACT-II, Revised," W. S. Brown, L. O. Ricks, P. C. Heiser, November 1963.
2. WANL-TNR-063 (with Addendum 1), "Activation Source Strength Program, ACT-I and ACT-II," P. C. Heiser, L. O. Ricks, September 1962.

SATURN

Code Tape: (ZS47, SATURN, PR~~O~~GRAM)

Code Disk: N/A

Computer: CDC-6600

Data processing functions performed by the SATURN computer code¹ are as follows:

1. Preparation of group dependent or independent cross section data magnetic tapes or disk files.
2. Inclusion of transverse leakage corrections to macroscopic cross-section data.
3. Preparation of macroscopic cross sections using mixing table operations.
4. (At option) adjoint cross section data is provided for use in DOT-IIW.

These functions are performed in SATURN as a sequence of logical steps to minimize the computer core memory storage required to prepare cross-section data.

Reference:

1. WANL-PR(LL)-034, Volume 3, "Cross-Section Generation and Data Processing Techniques," R. G. Soltesz, R. K. Disney, S. L. Zeigler, August 1970.

S-4

Code Tape: N/A (FORTRAN source deck on file in NRD)

Code Disk: N/A

Computer: CDC-6600

The S-4 code¹ calculates the fission product energy release rate as a function of time both during and after reactor operation for decaying beta particles and delayed gamma rays. A convolution integral which describes the above phenomena is exactly solved by use of double precision arithmetic. At least 64 bits are required for use of the code.

Reference:

1. Internal Document (NRD-67-111), "U²³⁵ Fission Product Energy Release Rate (Program S-4)," R. G. Soltesz, R. N. Nassano, April 20, 1967.

APPENDIX D

DESCRIPTION OF SPECIAL ANALYTICAL TECHNIQUES

D-1 ACTIVATION CROSS SECTIONS

Neutron activation measurements were used to evaluate calculated neutron energy spectra and neutron cross sections. Materials used in activation detectors were Au^{197} , In^{115} , Dy^{164} , Mn^{55} , W^{186} , U^{238} , U^{235} , S^{32} , Cu^{63} , and natural Cd. The neutron cross sections were on GAMB1T library tapes, and following is a discussion of the origin of the cross sections for each of these elements.

Activation cross sections for In^{115} , Mn^{55} , and S^{32} were added to GAMB1T libraries by the Reactor Physics and Mathematics Department at WANL. The resonance parameters in In^{115} and Mn^{55} were obtained from BNL-325,⁴¹ and branching ratios for the various (n, γ) decay chains for these isotopes were obtained from Reference 42.

Sulfur cross sections were obtained from the measured results reported in Reference 43. The detailed sources for the cross sections of Au^{197} , Dy^{164} , and Cu^{63} have been published in Reference 22. The source of the cross sections for W^{186} are reported in Reference 21. The cross sections for U^{238} were generated by Gibson, et al.,²⁰ using the U^{238} cross-section measurements by Garg, et al.⁴⁴ Cadmium cross sections were obtained from the original GAM library⁴ and U^{235} cross sections were obtained from Reference 45.

D-2 RESONANCE SELF-SHIELDING AND FLUX DEPRESSION FACTORS

Self-shielding of a single isolated resonance is defined as the ratio of the probability of a certain resonance reaction occurring in a lump to the probability of the same reaction occurring in an infinitely dilute mixture. This ratio is always equal to or smaller than unity. The two contributions to resonance self-shielding are: (1) spatial self-shielding and (2) energy self-shielding.

The spatial self-shielding is a result of the outer region of the lump absorbing or scattering the neutrons and shielding the inner portion of the lump; it is equal to the ratio of the average flux in the absorber to the flux without the absorber.

Energy self-shielding is a result of the depletion of neutron flux density at energies corresponding to that of the resonance. For single isolated resonances, effective capture cross sections can be calculated using the single level Breit-Wigner⁴⁸ formula which includes the effect of Doppler Broadening to account for relative motion of the incident neutron and the target nucleus.

In the present analysis, self-shielding factors were obtained from Reference 46. This calculation included the scattering and Doppler Broadening effect of resonance detectors in a slab geometry. The calculation assumed a $1/E$ dependence of energy flux. For narrow resonances, like the activation resonances, this is a good approximation. Table D2-1 summarizes the resonance self-shielding factors of Reference 46 for the various activation detectors. Figures D2-1 and D2-2 display the resonance self-shielding factors for Au^{197} , In^{115} , Mn^{55} , and W^{186} as a function of foil thickness.

Flux depression factors are important only at thermal energies and this effect is due to neutrons being absorbed in passing through the foil. This gives rise to a depression in the flux for a distance from the foil comparable to the neutron mean free path in the diffusing media. For most of the foils used in this analysis, the thermal flux depression factor was less than 1 or 2 percent. The exception to this was the 5 mil gold foils which had a maximum depression of approximately 20 percent. The thermal flux depression factors were obtained through the use of first-flight transport theory³³ and are listed in Table D2-1.

D-3 CADMIUM TRANSMISSION FACTORS

Energy-dependent transmission factors were employed to calculate cadmium filtered activities. An expression for these transmission coefficients has been derived in Reference 47 for an incident isotropic source. This is given by Equations D-1 and D-2:

$$T(E) = \left[\left\{ E_3(X_f^T) - E_3(X_f^T + X_d) \right\} + \frac{\sum_s}{2\sum_f} \int_0^{X_f^T} \left\{ E_3(X_f^T - X) - E_3(X_f^T + X_d - X) \right\} \right. \\ \left. \left\{ E_2(X) + E_2(2X_f^T + X_d - X) \right\} dX \right] / \left\{ \frac{1}{2} - E_3(X_d) \right\} \quad (D-1)$$

for thick detectors, and

$$T(E) = E_2(X_f^T) + \frac{\Sigma_s}{2\Sigma_f} \int_0^{X_f^T} E_2(X_f^T - X) \left\{ E_2(X) + E_2(2X_f^T - X) \right\} dX \quad (D-2)$$

for infinitely thin detectors

where:

$$E_2(Y) = \int_0^1 e^{-Y/\mu} d\mu = \int_1^\infty u^{-2} e^{-Yu} du$$

$$E_3(Y) = \int_0^1 \mu e^{-Y/\mu} d\mu = \int_1^\infty u^{-3} e^{-Yu} du$$

$$\mu = \cos \theta$$

$$\theta = \text{angle of incidence for neutrons}$$

$$X_f = \Sigma_f(E)t_f \quad (f = \text{filter})$$

$$X_d = \Sigma_d(E)t_d \quad (d = \text{detector})$$

$$t_f = \text{thickness of filter}$$

$$t_d = \text{thickness of detector}$$

$$\Sigma_s = \text{macroscopic scattering cross section of filter for neutrons of energy } E$$

$$\Sigma_f^T(E) = \text{macroscopic total cross section of the filter for neutrons of energy } E$$

$$\Sigma_d(E) = \text{macroscopic reaction cross section of the detector for neutrons of energy } E$$

The numerical solution of the second term on the right in Equations D-1 and D-2 is difficult. However, this can be avoided by using a better approximation. When only the first term on the right is used for transmission coefficient calculations, the transmission is correct for a pure absorber and underestimated for other absorbers. If $\sum_f^T(E)$ is replaced by the absorption cross section of the filter, $\sum_f^a(E)$, in Equations D-1 and D-2, and the second term in these equations is neglected, then the transmission coefficient is again correct for a pure absorber but overestimated for other absorbers with scattering. Therefore, the true transmission factor is in between the two values calculated from Equation D-1 or D-2. In fact, the arithmetic average of these two values was found to be closer to the correct value than the exact solution calculated by Tsuchihashi.⁴⁷ This was determined by comparing the various calculated values with experimental measurements.³⁴ Using these relationships, the transmission coefficient becomes

$$T(E) = \frac{\left[\left\{ E_3(X_f^T) - E_3(X_f^T + X_d) \right\} + \left\{ E_3(X_f^a) - E_3(X_f^a + X_d) \right\} \right]}{\left[1 - 2 E_3(X_d) \right]}$$

for thick detectors, and

$$T(E) = \left[E_2(X_f^T) + E_3(X_f^a) \right] / 2$$

for infinitely thin detectors.

D-4 ACTIVATION RATE

The expected activation rates for each of the detectors was computed for the basic and cadmium covered condition from the formulas below.

$$\begin{aligned} \text{Bare:} \quad A_s &= K' C \sum_{g=1}^{16} \bar{\sigma}_g \bar{\phi}_g \\ \text{With Cd Filter:} \quad A_s &= K' C \sum_{g=1}^{16} \bar{\sigma}_g T_g \bar{\phi}_g \end{aligned}$$

where:

A_s	=	calculated saturated activity
$\bar{\sigma}_g$	=	group average effective reaction cross section (includes self shielding)
K'	=	factor to place results on an absolute basis (See Section 6)
C	=	conversion factor = $0.6025/A$
A	=	atomic weight of detector
$\bar{\phi}_g$	=	group average flux from DOT and ANISN
T_g	=	group average transmission coefficient of cadmium filter
g	=	neutron energy group index

The saturation activity is the activity approached for an irradiation that is long compared to the mean life of the reaction product. When this condition is not satisfied, the experimental results can readily be corrected to account for this factor so that the resultant values can be compared directly to the calculated activity obtained from the above two formulas.

TABLE D2-1 SELF-SHIELDING FACTORS FOR RESONANCE AND THERMAL DETECTORS

Detector	Resonance Energy or Thermal Energy Range	Type of Self-Shielding	Self-Shielding Factor
Au ¹⁹⁷ (5 mil)	61.5 eV	Resonance	0.350
Au ¹⁹⁷ (5 mil)	4.9 eV	Resonance	0.155
Au ¹⁹⁷ (5 mil)	0.060 - 0.040 eV	Thermal	0.980
Au ¹⁹⁷ (5 mil)	0.040 - 0.030 eV	Thermal	0.960
Au ¹⁹⁷ (5 mil)	0.030 - 0.020 eV	Thermal	0.940
Au ¹⁹⁷ (5 mil)	0.020 - 0.009 eV	Thermal	0.860
Au ¹⁹⁷ (5 mil)	0.009 - 0.005 eV	Thermal	0.805
Au ¹⁹⁷ (0.5 mil)	0.005 - 0.0 eV	Thermal	0.739
Au ¹⁹⁷ (0.5 mil)	61.5 eV	Resonance	0.782
Au ¹⁹⁷ (5 mil)	4.9 eV	Resonance	0.468
In ¹¹⁵ (10 mil alloy)	9.10 eV	Resonance	0.964
In ¹¹⁵ (10 mil alloy)	3.86 eV	Resonance	0.980
In ¹¹⁵ (10 mil alloy)	1.46 eV	Resonance	0.730
Mn ⁵⁵	1,080 eV	Resonance	0.840
Mn ⁵⁵	337 eV	Resonance	0.890
W ¹⁸⁶	18.8 eV	Resonance	0.620
Cu ⁶³ (5.5 mil)	580 eV	Resonance	0.732

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Figure D2-1. Resonance Self-Shielding Factors of In and Au Foils

Figure D2-2. Resonance Self-Shielding Factors of Mn and W Foils

APPENDIX E

WANEF EXPERIMENTAL TECHNIQUES

This Appendix documents briefly the techniques used at the Westinghouse Astronuclear Experimental Facility (WANEF) for the measurement of the nuclear characteristics of a reactor mock-up.

The overall flow chart for WANEF activity is shown in Figure E-1. Additional charts shown in Figures E-2 through E-14 are provided to further clarify the flow path of information and data. The numbering scheme for the activity blocks and the input/output blocks is similar to that used in Section 3. The five basic experimental activities (Blocks 401 through 407) are described in the sections that follow. Abstracts are given for the computer codes referenced in the figures.

NOTE: Material in this Appendix has been taken from DRM 52738, "Flow Charts for WANEF Experimental Procedures, Data Reduction, and Reporting," W. D. Rankin. See this DRM for material referenced here.

E-1.1 Reactivity Worths

Operate Reactor to Measure Sample Reactivity (Block 403.1)

a. Small reactivity changes are determined by measuring the reactivity of the reactor at the same drum bank angle with and without the sample present. The resulting reactivity can be measured with either the period pick-off device or the reactivity computer. The period pick-off device measures the asymptotic period which must then be converted to reactivity with the in-hour equation. The reactivity computer displays reactivity directly.

b. Large reactivity changes ($> 30\%$) are measured by determining the new critical drum bank position and using the known value of control drum worths ($\%$ /degree) to calculate the reactivity change.

Period Pick-Off (Block 403.2)

The reactor period is indicated directly by this device by measuring the time required for the reactor power to increase by a factor of e , as determined by a neutron detector located external to the reactor.

Checkout Reactivity Computer (Blocks 403.3, 404.4)

The analog computer amplifiers are balanced according to the method given in the PC 12 Operating Manual.

Reactivity Computer (Blocks 403.4, 404.5)

An EAI PC 12 analog computer is programmed to solve the time-dependent point reactor kinetics equations. A neutron sensitive ion chamber located outside the reactor provides a current which is proportional to power level for the computer.

See Reference 1, p. 3-4. *

* References for this appendix are given in Section E-3, p. E-14.

MIDAS-1 (Block 403.5, 404.6)

MIDAS-1 is the Model Integrated Data Acquisition System Mark 1 in use at WANEF which can take data from the PAX reactor using the Packard Bell PB-250 computer. Up to six analog signals can be processed by the multiplexed analog-to-digital converter which is interfaced to the central processor. Time signals are obtained from the master clock at WANEF. Input/Output are controlled by program DATA, which can accept up to three inputs and provides typed listing and punched tape outputs.

E-1.2 Control Drum Worths

Operate Reactor to Achieve Initial Conditions for the i^{th} Drum Calibration (Block 404.1)

The reactor is placed on a positive period with the i^{th} drum at 180° , its nearest neighbors at some prescribed angle, and the remaining drums at a uniform drum bank angle. The reactor power level is permitted to increase until the signal to the analog reactivity computer is about midscale in its operating range. Measurement of the i^{th} drum worth can then begin.

Measure i^{th} Drum (Block 404.2)

The chart recorder on the analog reactivity computer records a single integral drum worth as the i^{th} drum is run in continuously from 180° to 0° .

Data Processing (Block 404.3)

The integral drum worth curves from the reactivity computer are differentiated by hand to provide differential drum worths.

DATA (Block 403.5, 404.6)

Code Tape: (Assembly Tape and Binary Tape on File at WANEF)

Computer: PB-250

DATA is the program used by MIDAS-1 to control the gathering of data. Up to three different analog inputs can be selected by means of sense switches. The digital output is in the form of a typed list and a punched paper tape.

EXFIT (Block 403.6)

Code Tape: (Fortran Source Tape and Binary Tape on File at WANEF)

Computer: PB-250 or CDC-6600

EXFIT fits a simple exponential by a least square method to suitable power level data acquired from the PAX reactor with the MIDAS-1 system. Input values are obtained from the punched paper tape output of MIDAS-1/DATA. The computer result is used to calculate reactivities.

DRMSPAN (Block 404.7)

Code Tape: (Fortran Source Deck & Binary Deck on File at WANEF)

Computer: CDC-6600

DRMSPAN accepts control drum position and reactivity data from the PAX reactor using MIDAS-1 and DATA. Input punched cards are produced from the punched tape output of DATA using the tape-to-card converter. DRMSPAN output is in the form of listings and graphs of integral and differential control drum reactivity worth.

E-1.3 Fission Distribution

CREATE (Block 405.1)

Code Tape: (Fortran Source Deck on File at WANEF)

Computer: CDC-6600

CREATE is used to prepare a master tape containing the radial and azimuthal positions of flow channel locations in a given core design. Input to the program consists of a card deck containing this information. The master tape is used by EXPADO 2 to insert radial and azimuthal information into a data tape containing the results of experimental in-core dosimetry measurements.

EXPADO 1 (Block 405.2, 407.9.7)

Code Tape: (Binary Object Deck and Fortran Source Deck on File at WANEF)

Computer: CDC-6600

EXPADO 1 performs preliminary data reduction of all passive dosimetry data obtained at WANEF from fission or activation dosimeters.

Input to the code consists of: 1) data representing counts and times recorded from scaling and timing systems used to measure the activity on the dosimeters, 2) tables of dosimeter locations, and 3) miscellaneous constants required for the data reduction which may include tables of dosimeter weights.

Operations performed by the code include: 1) listing of input constants, tables, and data, 2) correlation of data table with dosimeter location table, 3) correction for counting system background, resolving time, and the relative efficiency of individual counters in the system, 4) optional corrections for dosimeter weight and decay time, 5) optional ratio to normalizing dosimeter, 6) listing of computed background rates, 7) statistical analysis of scatter in data, 8) averaging of multiple counts on the same dosimeters with rejection of data deviating from average beyond desired limits, 9) optional re-normalizing on the basis of computed relative efficiencies of the individual counters in the system, 10) uncertainty calculation involving computation of error propagation during each step of the data reduction. Extensive output tables are provided to enable tracing the root of unexpected or questionable results. In addition, error flags are provided to call attention to unusual computational results or procedures.

Two binary tapes are written to provide input to EXPADO 2 and EXPADO 3.

EXPADO 2 (Block 405.3)

Code Tape: (Binary Object Deck and Fortran Source Deck on File at WANEF)

Computer: CDC-6600

EXPADO 2 operates on the output tape from either EXPADO 1, EXPATLD, or EXPAPHY, which contain partially analyzed data and produces a second tape of identical format into which has been inserted radius and angle information representing the location of dosimeters which had been in reactor flow channels. As a source of this position information, it uses the output tape from the program, CREATE, which consists of a table of radius and core angle versus a location code describing a given flow channel. EXPADO 2 matches location codes in the two tapes for each data point requested and transfers the corresponding position information. Error messages are generated for locations on the data tape which cannot be matched.

The output tape from EXPADO 2 is used as input to the program EXPADO 3.

EXPADO 3 (Block 405.4, 407.9.2, 407.9.5, 407.9.8)

Code Tape: (Binary Object Deck and Fortran Source Deck on File at WANEF)

Computer: CDC-6600

EXPADO 3 is used in the scheme of passive dosimetry data reduction at WANEF. Its input consists of the output tapes from either EXPADO 1, EXPATLD, or EXPAPHY, which may or may not have been modified and rewritten by EXPADO 2.

The program operations include the following: 1) perform operations on the dosimeter location coordinate to scale them or shift their zero reference, 2) optionally write an output deck in the form requested for input to the PFENCER program, 3) write output tables suitable for final reports (especially for relative fission flux and for TLD and Phylatron measurements), 4) write an output deck in form suitable for input to EXPADO 4, EXPASIG, or COREAVG, 5) write a binary tape in form suitable for input to PADOSE.

PADOSE (Block 405.5, 407.9.3, 407.9.6, 407.9.11)

Code Tape: (Fortran Source Deck and Binary Deck on File at WANEF)

Computer: CDC-6600

PADOSE is used to produce computer drawn plots of the reduced passive dosimetry data from experiments at WANEF. Data input is via magnetic tape as written by EXPADO 3 or EXPADO 5. Instructions for each plot to be drawn are input on cards.

EXPASIG (Block 405.6)

Code Tape: (Fortran Source Deck and Binary Deck on File at WANEF)

Computer: CDC-6600

EXPASIG is used to compare two sets of data representing two independent measurements of the fission flux distribution at identical or corresponding locations in the reactor. Data input is via the card output from EXPADO3. Results are confidence values of the reproducibility of the measurement technique. Other output consists of data and plots showing details of the statistics of the data (see Reference 2).

E-1.4 Power Calibration

COREAVG (Block 406.1)

Code Tape: (Fortran Source Deck on File at WANEF)

Computer: CDC-6600

COREAVG is used to compute an average value of the fissions per gram in one sector of the core with respect to the fissions per gram in the center of the core. The computation makes use of experimental data of relative fission flux as output on cards from EXPADO 3. The computer result is used in the calculation of core power levels.

E-1.5 Radiation Environment

Foils (Block 407.1)

Foils currently on hand at WANEF include dysprosium, gold, sulfur, U-235, U-238, Pu-239, and Th-237. Other materials can be added when necessary.

Checkout Gamma Counters (Block 407.2)

The checkout of the gamma counters consists of checking the amplifier wave forms with an oscilloscope and of adjusting the high voltage and discriminator settings by means of standard Cs-137 sources and a multichannel analyzer. The count rates obtained from the sources and the background count rates are then compared to normal values to verify proper operation of the system.

Checkout Beta Counters (Block 407.3)

The checkout of the beta counters consists of checking the voltage plateaus of the detectors with standard sources and verifying that the system functions mechanically and electronically. The count rates of the sources and the background count rates are compared to normal values to verify proper operation of the system.

Checkout Phylatron System (Block 407.5)

The checkout of the Phylatron system involves verification that the instrument functions properly and then recording the voltage readings of each Phylatron to be used in the experiment.

Calibrate Bragg-Gray detector (Block 407.6)

Calibration of the Bragg-Gray detectors is a two-step procedure given in Figure E-10. In step 1, the RCF gamma radiation pattern is measured using a Co-60 source and the NBS calibrated CO₂ detector. In step 2, the detector to be calibrated is exposed in the measured radiation field as a function of distance from the source and the necessary calibration constant and 1 σ uncertainty is obtained by least squares fitting the experimental data by the BGOMFIT program.

Calibrate U-238 detector (Block 407.7)

Calibration of the U-238 detector, given in Figure E-11, is accomplished by exposing the detector to the known flux of the NBS calibrated neutron source at several different distances. The calibration constant is obtained by least squares fitting the data with the BGOMFIT program.

Intercompare WX-5362 detectors (Block 407.8)

The procedure for intercomparing the ORNL calibrated WX-5362 detector with two similar detectors is given in Figure E-12 (see Volume 1, Appendix A of Reference 3).

BGOMFIT (Blocks 407.6.1, 407.6.3, 407.7.1)

The BGOMFIT, (Reference 4), computer code is used to expedite U-238 fast fission detector calibration, WX-5362 thermal neutron detector calibration, CO₂ gamma ray absorbed dose rate detector calibration, and the Bragg-Gray gamma ray absorbed dose rate detectors calibration, (see References 5, 6, 7,8). The BGOMFIT code is a modification to one portion of the OMNIFIT2 (Reference 9) computer code. BGOMFIT calculates the log₁₀ of the input data, does the least squares curve fitting calculations, and then calculates the anti-log of the original input data. This process is applied if the first function (NFIT = 1) is selected, otherwise the BGOMFIT code is identical to the OMNIFIT2 computer code.

EXPATLD (Block 407. 9.1)

Code Tape: (Binary Object Deck and FORTRAN Source Deck on File at WANEF)

Computer: CDC-6600

EXPATLD performs data reduction of thermoluminescent dosimeter (TLD) readings taken for radiation dosimetry purposes at WANEF.

Input to the code consists of: 1) a card deck with one card per dosimeter containing information including dosimeter location, serial number and type, and the instrument reading of the dose received, 2) miscellaneous constants required for the data reduction.

Operations performed by the code include: 1) listing of input constants and data tables, 2) corrections of the data for background, for scale conversion factors, and, optionally, for individual dosimeter calibration factors, 3) an uncertainty calculation based on input of uncertainties in the various constants and on uncertainties in the readout process. Output tables are provided to enable tracing the steps of the data reduction.

Two binary tapes are written to provide input to EXPADO3 (or to EXPADO2).

EXPAPHY (Block 407. 9.4)

Code Tape: (Binary Object Deck and FORTRAN Source Deck on File at WANEF)

Computer: CDC-6600

EXPAPHY performs data reduction of Phylatron readings taken for radiation dosimetry purposes at WANEF.

Input to the code consists of: 1) a card deck with one card per dosimeter containing information including dosimeter location, serial number, and type, the time of readout, and the Phylatron voltage readings as originally received, just before irradiation and after irradiation, 2) miscellaneous constants required for the data reduction.

Operations performed by the code include: 1) listing of input constants and data tables, 2) conversion of the data from voltage readings to doses by means of a look-up table and interpolation, 3) data corrections for previous dosage to the Phylatrons and for fading after exposure, 4) an uncertainty calculation based on input of uncertainties in the various constants and on uncertainties in the readout process. Output tables are provided to enable tracing the steps of the data reduction.

Two binary tapes are written to provide input to EXPADO3 (or to EXPADO2).

EXPADO4 (block 407. 9.9)

Code Tape: (Binary Object Deck and FORTRAN Source Deck on File at WANEF)

Computer: CDC-6600

EXPADO4 performs one step of the data reduction of activation foil and fission foil dosimetry data taken at WANEF.

Input to the code consists of: 1) data cards output during the previous phase of the data reduction procedure by the code EXPADO3, 2) a deck of normalization constants required to convert relative numbers to absolute activities or fission rates and/or fluxes, including estimates of the uncertainties involved, 3) miscellaneous other constants.

Operations performed by the code include, 1) listing of the normalization constants and data tables, 2) normalization of the data.

Two binary tapes are written to provide input to EXPADO5.

EXPADO5 (Block 407. 9.10)

Code Tape: (Binary Object Deck and FORTRAN Source Deck on File at WANEF)

Computer: CDC-6600

EXPADO5 completes the data reduction process for activation foil and for fission foil dosimetry data taken at WANEF.

Input to the code consists of two binary tapes written by EXPADO4.

Operations performed by the code include 1) listing of input constants transmitted from EXPADO4, 2) matching of and subtraction of bare and cadmium-covered data points to obtain subcadmium or thermal-type data where required, 3) output of tables in a form suitable for final reports, 4) output of a card deck for a computer-readable record of the results, 5) output of the results on a binary tape in a form suitable for input to PADOSE.

RNPARX2 (Block 407. 10. 1)

The RNPARX2 (Reference 10) computer code subtracts instrument background, applies conversion factors to the detector and position data, calculates a standard deviation for each data point, and sorts the calculated data by detector and configuration. The program calculates differences between background and foreground configurations. The results of the differences calculated and the foreground data are written onto two separate tapes. These tapes are utilized by the PARX22 program.

PARX22 (Block 407.10.2)

The PARX22 (Reference 10) computer code generates two types of graphs from the two tapes (total and direct) created by the RNPARX2 computer code. Type 1 graphs consist of plots generated for each configuration and PAX angle for the tables requested, while Type 2 graphs consist of plots for a fixed elevation versus a calculated polar angle. The "total" data tape contains data for all the configurations for all of the reports available. The "direct" data tape contains data for all the configurations with a background configuration subtracted from each.

HDRC (Block 407.10.3)

The HDRC (Reference 11) computer code is used to reduce the fast neutron dose rate data from the Hurst proportional counter (Reference 12). This code calculates an integrated pulse height distribution above each channel of integral data. The sums of the integrated data are added to give the total area. The integrated pulse height distribution data are then plotted versus channel numbers. A linear least squares fit is made to a selected portion of the curve and the dose rate is then calculated as a function of the area under the fitted curve.

E-2 Flow Charts

The flow charts are shown in Figures E-1 through E-14. Figure E-1 presents the overall flow chart for WANEF. Figures E-2 and E-3 show the planning stages of an experiment (blocks 401 and 402). Figures E-4 through E-8 detail the types of experiments performed (blocks 403 through 407). Figures E-9 through E-12 show calibration procedures (blocks 407.4, 407.6, 407.7, and 407.8). Figures E-13 and E-14 show the data acquisition and data reduction procedures for radiation environment experiments, (blocks 407.9 and 407.10). Further description of the blocks are given in Section E-1.

E-3 References

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Figure E-1. Westinghouse Astronuclear Experimental Facility

Figure E-2. General Test Planning
Block 401
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Figure E-3. Detailed Planning
Block 402

Figure E-4. Reactivity Worths
Block 403
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Figure E-5. Control Drum Worths
Block 404
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Figure E-6. Power Distribution
Block 405
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Figure E-7. Power Calibration
Block 406
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Figure E-9. Calibrate TLD System
Block 407.4

Figure E-10. Calibrate Bragg-Gray Detector
Block 407.6
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Figure E-11. Calibrate U-238 Detector
Block 407.7
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Figure E-12. Intercompare WX-5362 Detectors
Block 407.8
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Figure E-13. Passive Data Acquisition and Reduction
Block 407.9

Figure E-14. Active Data Acquisition and Reduction
Block 407.10
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ABSTRACT The WANL Integrated Nuclear, Radiation and Shielding Standard Design Method (WISDM) used at the Westinghouse Astronuclear Laboratory in nuclear and radiation design and analysis of the NERVA Nuclear Subsystem is described in this report. An evaluation of the accuracy of the WISDM methods and techniques is made by comparison with experimental results. WISDM is an integral part of an overall Design Engineering Method which is being developed to automate the mechanical, thermal, structural, nuclear, and radiation analyses and design of the NERVA Nuclear Subsystem.		
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