XN-NF-80-19(NP)(A) VOLUME 1 and XN-NF-80-19(NP)(A) VOLUME 1 SUPPLEMENTS 1 & 2

EXXON NUCLEAR METHODOLOGY FOR Boiling water reactors

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NEUTRONIC METHODS FOR Design and Analysis

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EXON NUCLEAR COMPANY, Inc.

XN-NF-80-19(NP)(A) Volume 1 and

XN-NF-80-19(NP)(A) Volume 1 Supplements 1 & 2

Issue Date: 1/28/83

EXXON NUCLEAR METHODOLOGY FOR BOILING WATER REACTORS -NEUTRONIC METHODS FOR DESIGN AND ANALYSIS

This is the NRC approved version of Documents XN-NF-80-19(NP), Volume 1, XN-NF-80-19(NP), Volume 1, Supplement 1, and XN-NF-80-19(NP), Volume 1, Supplement 2, and has been prepared in accordance with NRC guidance.





UNITED STATES NUCLEAR REGULATORY COMMISSION WASHINGTON, D. C. 20555

APR 0 7 1982

Mr. G. F. Owsley, Manager Reload Fuel Licensing Exxon Nuclear Company, Inc. 2101 Horn Rapids Road P. O. Box 130 Richland, Washington 99352

Dear Mr. Owsley:

Subject: Acceptance for Referencing of Topical Report XN-NF-80-19(P) Volume 1, Supplement 1 and Supplement 2

The Nuclear Regulatory Commission (NRC) has completed its review of the Exxon Nuclear Company, Inc. (ENC) Licensing Topical Report XN-NF-80-90(P) Volume 1, dated May 1980; Volume 1, Supplement 1, dated April 1981; and Volume 1, Supplement 2, dated May 1981, all entitled "Exxon Nuclear Methodology for Boiling Water Reactors, Volume 1 Neutronic(s) Methods for Design and Analysis."

Volume 1 includes local neutronic analysis models applicable to individual fuel assemblies and neutronics core analysis methodology applicable to the entire core. Uncertainty analysis methodology and verification of the calculational results are also covered. The neutronics core analysis methodology includes control rod drop, control rod withdrawal, fuel misloading, reactor core and channel hydrodynamic stability, and neutronic input to the total nuclear power plant transient analysis. The neutronic methods are verified by comparing the calculational results with measured reactor data and with higher order calculations. The power distribution uncertainty methodology considers the neutronic models and the measured reactor data.

Volume 1, Supplement 1, provides the responses to USNRC questions pertaining to the Exxon Nuclear Company Neutronics Methodology for Boiling Water Reactors and request number 1 for additional information contained in the letter from R. L. Tedesco to G. F. Owsley dated January 19, 1981. Volume 1, Supplement 2, presents supplemental information on the XTGBWR model, COTRAN model, methods verification, and Uncertainty Analysis. In addition, it includes a complete new section 7 of Volume 1 on the Application of Neutronics Methodology. A copy of our Safety Evaluation is attached.

Mr. G. F. Owsley, Manager

Based on our review of XN-NF-80-19(P) Volume 1; Volume 1, Supplement 1; and Volume 1, Supplement 2, we conclude they describe acceptable analytical methods and computer codes for calculating the neutronic behavior of BWRs with fuel loadings and geometric properties similar to those analyzed in the report. It has been shown that these methods can predict, to an acceptable accuracy, the physics characteristics of operating reactors. We do, however, recommend that the analytical models be continuously verified to insure their applicability.

As the result of our review, we conclude that the Exxon Nuclear Company licensing topical report XN-NF-80-19(P) Volume 1, dated May 1980, as augmented by Volume 1, Supplement 1, dated April 1981 and further augmented by Volume 1, Supplement 2, dated May 1981, is acceptable for referencing in license applications relating to BWR physics analyses to the extent specified and under the limitations in the topical report and the topical report safety evaluation. When this report is referenced, the reference must include both the proprietary and non-proprietary versions.

We do not intend to repeat our review of this topical report when it appears as a reference in a particular license application except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the features described in the topical report, and the Supplements thereto.

In accordance with established procedures, it is requested that Exxon Nuclear Company Inc. publish the approved versions within three months of receipt of this letter. The approved versions should include this letter and the enclosed evaluation following the title page and should appropriately incorporate into Volume 1 the information contained in Supplement 1 and Supplement 2.

Should Nuclear Regulatory Commission criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, Exxon Nuclear Company Inc. and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely, in Styles

James R. Miller, Chief Standardization & Special Projects Branch Division of Licensing

Enclosure: Topical Report Evaluation

SAFETY EVALUATION

FOR THE

EXXON NUCLEAR COMPANY TOPICAL REPORT:

"EXXON NUCLEAR METHODOLOGY FOR BOILING WATER REACTORS

NEUTRONIC METHODS FOR DESIGN AND ANALYSIS" XN-NF-80-19(P)

(VOLUME 1 MAY 1980, SUPPLEMENTS 1 AND 2 APRIL 1981)

Prepared By

Core Performance Branch

Reviewed by:

L. Lois

With Contributions by:

- G. Schwenk
- S. L. Wu

and

- D. Cokinos, BNL
- P. Neogy, BNL
- H. Cheng, BML
- L. Eisenhart, BNL

EVALUATION OF REPORT XN-NF-80-19(P) VOLUME 1, SUPPLEMENT 1 AND SUUPLEMENT 2

Report Number:

1

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Report Title:

Report Date: Originating Organization: Reviewed By: XN-NF-80-19(P) Volume 1, Supplement 1 and Supplement 2 Exxon Nuclear Methodology for Boiling Water Reactors-Neutronic Methods for Design and Analysis, Volume 1, Supplements 1 and 2 May 1980, (Supplements April 1981) Exxon Nuclear Company, Inc. Reactor Physics Section, Core Performance Branch, DSI

Table of Contents

1.0	INTRODUCTION								
2.0	SUMMARY OF EVALUATION P								
	2.1 Neutronics Models for BWR Reactor Core Calculations								
		(a) (b) (c) (d) (e)	XFYRE, Fuel Assembly Depletion Model XTGBWR, Core Simulator Model COTRAN, Reactor Kinetics Code XDT, Multigroup Diffusion Theory XMC, Monte Carlo Neutron Transport						
	2.2	Neut	ronic Core Analysis Methodology	Page	12				
		(a) (b) (c) (d) (e)	Steady State Parameter Calculations Control Rod Drop Accident Fuel Misloading Analysis Control Rod Withdrawal Neutronic Reactivity Parameters						
	2.3	Neutronics Methods Verification Pag							
		(a) (b) (c)	XFYRE Verification XTGBWR Verification COTRAN Verification						
	2.4	Measu	ured Power Distribution Uncertainties	Page	19				
		(a) (b)	Determination of Measured Power Distribution Derivation and Estimation of the Uncertainty in the Measured Power Distribution						
3.0	APPLICATIONS OF THE NEUTRONIC METHODOLOGY								
	 (a) Control Rod Drop Accident (b) Fuel Misloading Error (c) Reactor Core Stability (d) Control Rod Withdrawal 								
4.0	EVALU	ATION	PROCEDURE	Page	23				
5.0	REGUL	ATORY	POSITION	Page	24				
6.0	REFER	ENCES		Page	25				

1.0 INTRODUCTION

The Exxon Nuclear Company has submitted a technical report describing the techniques and the supporting data for the design methodology to be applied in the next four to five years to the reloads of Dresden Units 2 and 3, Grand Gulf Units 1 and 2, Susquehanna Units 1 and 2, LaSalle Units 1 and 2, and Zimmer. The first reload will be Dresden 3 scheduled for a May 1982 start up. The report includes discussions of codes for:

- fuel assembly depletion (XFYRE)
- core simulation (XTGBWR)
- reactor kinetics (COTRAN)
- neutron diffusion (XDT)
- Monte Carlo (XMC)

In addition, neutron flux verification methods and analysis of power distrihution uncertainties are included. As a result of this review, two additional volumes (Supplements 1 and 2) were issued which include additional information in response to staff questions and an extensive documentation of the data base for method verification and the uncertainty analysis. Application of the neutronic methodology to the control rod drop accident, fuel misloading error, and the rod withdrawal accident are discussed. The verification of XFYRE, XTGBWR, and COTRAN is discussed in a separate section. Finally, the measured power distribution uncertainty and its estimation is dealt with in the last section.

In the course of the review, additional information was supplied with an extended data base and treatment for the neutronics models, neutronic data verification, measured power distribution uncertainties, and the application of the neutronics methodology. The information supplied in each section in the original report and the reply to the staff's questions (Supplements 1 and 2) will be reviewed as a whole for each topic.

2.0 SUMMARY OF EVALUATION

7.1 Neutronics Models for BWR Reactor Core Calculations

In this section we shall review the computer codes and the associated cross sections. The neutronics methods include the five codes: XFYRE, for the calculation of fuel neutronic parameters and assembly burnup; XTGBWR, for reactor core simulation; COTRAN, for transient analysis and calculations; XDT, for neutron diffusion calculations; and XMC, for Monte Carlo neutron transport calculations.

The review objectives for analytical methods are stated in SRP Section 4.3 For the review of computer codes, the areas of concern and review are:

- (a) description of the analytical methods used in the nuclear design, including those for predicting criticality, reactivity coefficients, burnup and stability;
- (b) the data base used for neutron cross-sections and other nuclear parameters; and
- (c) verification of the analytical methods by comparison with measured data.

With the above criteria in mind we shall discuss each of the codes listed above.

(a) XFYRE, Fuel Assembly Depletion Model

The XFYRE code will generate fuel neutronic parameters as a function of void and exposure for both controlled and uncontrolled assemblies. XFYRE employes diffusion theory and combines $HRG^{(1)}$ and $THERMOS^{(2)}$ cross-section generation routines. The code uses two dimensional four energy group diffusion theory for the microscopic depletion of BWR assemblies. The code alternates between a spatial calculation of the average flux in each pin cell and a burnup calculation for each pin over an exposure interval maintaining a constant pin power over the interval. The four group cross sections are collapsed from fine group thermal and epithermal spectrum calculations for each pin type within the assembly. The spectrum calculations are repeated at intervals to adjust the multigroup cross-sections for the spectral change with burnup.

Cross-Sections

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XFYRE uses the Battelle Northwest thermal cross-section library⁽³⁾ for all nuclei except for hydrogen scattering for which the ENDF/B⁽⁴⁾ scattering kernel is employed and incorporated into the THERMOS library. THERMOS is used for the thermal neutron spectrum as a function of location in the lattice and solves a set of integral transport equations. The rods are distinguished as standard fuel, fuel with burnable poison and with gadolinia.

The XFYRE and the HRG programs are used for the epithermal cross-sections including calculation of the Dancoff correction. The energy range of 10 Mev to .414 eV is divided into 68 equally spaced lethargy groups using the P_1 approximation and U-235 or Pu-239 fission spectra. A special calculation is performed for the resonance range of U-235, Pu-239, Pu-240, and Pu-241.

For the control rod cross sections, THERMOS and HRG are used in a similar manner for each fuel type as a function of exposure and void. A special one dimensional integral transport calculation is performed for the thermal energy for the control rod blade and fuel assembly. Self shielding factors are obtained from Monte Carlo calculations. In the epithermal region, the control rod cross-sections are obtained from an HRG calculation.

Neutron Flux and Power Distribution

A two dimensional, four energy group diffusion calculation with XFYRE is used to calculate controlled and uncontrolled pin power, neutron flux and fuel assembly reactivity. For the diffusion theory calculations each cylindrical pin cell is homogenized over a 2x2 (x-y) spatial mesh. Additional regions represent the film water-channel mixture, water gap, incore detectors, inert rods, control blade, and control support. These calculations are performed as a function of depletion and void.

Depletion Calculation

Depletion is performed for each pin cell. If the pin contains gadolinia isotopes the cell is divided into four regions. Depletion is carried out in a single energy group (by collapsing the four groups) and with standard transmutation chains. The calculations are performed at predetermined intervals with linear interpolation to ten subintervals for the effect of spectral change. At each interval, the complete four group calculation is repeated. The above subinterval correction is used for all isotopes except the gadolinia for which the concentration and cross-sections are adjusted at each subinterval.

Finally the XFYRE code includes time and power dependence for xenon and samarium and a scheme to calculate the power at the location of the incore detectors.

(b) XTGBWR, Core Simulator Model

The core simulator program XTGBWR is a modification of a similar Exxon program $XTG^{(5)}$. This code is based on a two group, three dimensional, coarse mesh diffusion theory with rapid convergence characteristics. For fuel management calculations XTGBWR will, among other things, compute control rod parameters, thermal hydraulic feedback, Doppler reactivity feedback and fuel shuffling schemes.

The three dimensional geometry is (x, y, z). For core analysis in two dimensions, (x, y) geometry is used and the neutron leakage in the axial direction is calculated using either an input axial buckling or an internally calculated axial buckling. The reactivity effects of the axial buckling are treated through adjustment of the absorption cross-section.

As pointed out earlier, this is a standard diffusion calculation with adjustments for the control rods, instantaneous void and void history, power dependent Doppler, and time and power dependent xenon and samarium. A noteworthy approximation used is that of the effective diffusion coefficient between points i and j as: $D = \sqrt{D_j \cdot D_j}$ An empirical correction factor is applied to the model for controlled nodes to improve the prediction of the nodal power. Boundary conditions have been developed for the fast and thermal flux in the diffusion solution which are approximate for each group and correctly account for neutron leakage at the outer boundaries. Reflected and periodic boundary conditions have also been developed. Cross-sections are computed at near zero, average and maximum void (typically at 0, 40 and 70 percent) for each of 14 exposures from 0 to 40 GWD/MTU. Cross sections at values of exposure or void, different that those provided, are obtained by interpolation. Calculation of controlled element cross-sections for core follow are performed by XFYRE along with the assembly depletion.

Xenon is calculated in XTGBWR from the differential equations for iodine and xenon which allows explicit time and power dependence. Likewise, the promethiumsamarium chain is similarly treated in XFYRE. For startup conditions after shutdown and during operation, the samarium concentration is calculated. XFYRE also calculates nodal fuel temperature as a function of power, burnup, void fraction, and fuel rod design. This information is used by XTGBWR to adjust the fast absorption cross-section to account for the Noppler broadening in uranium and plutonium.

The pressure drop across the fuel assemblies (and the flow bypass) are calculated by a flow model which includes frictional, acceleration and gravitational terms. The orificing and other frictional loss terms such as for tie plates, spacers, etc., are modeled by flow dependent loss coefficients.

An option is provided in XTGBWR to calculate the flow distribution either by direct application of the pressure drop model or by the use of input polynomials approximating the assembly flow as a function of the orificing, hydraulic resistance and power. The coolant enthalpy, the steam quality and the void fraction are calculated from the integral of the heat transferred to the coolant. The local peaking factors are calculated in XFYRE as a function of exposure, voids, fuel type, and control using in part input from XTGBWR.

In the reactor core, incore detectors are placed in approximately one out of every four possible locations such that if the core is operated with quarter core mirror symmetry, all fuel assemblies (excluding those in the periphery) are monitored by an incore detector. Finally the XTGBWR code will perform zero power flux and eigenvalue solutions with no flow, void, or Doppler feedback. The nuclear parameters for the zero power solution are calculated with the XFYRE code.

(c) COTRAN, Reactor Kinetics Code

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The COTRAN code has been assembled from two separate codes i.e., XTRAN⁽⁶⁾ which supplies the neutronics solution and COBRA-IV(7) which determines the thermal hydraulics feedback. The neutronics solutions are based on a two dimensional (r-z) one group neutron diffusion model with one prompt and six delayed neutron groups. Fuel temperature and void reactivity feedback are provided. The thermal-hydraulic feedback is based on a one dimensional model. (By elimination of cross flow.) The COTRAN neutronics model solves the space-time diffusion equation with the assumption that in the volume of each node the flux is constant and separable in space and time. The same diffusion coefficient approximation as in XTGBWR is also used here. The above assumptions lead to an iterative scheme for each time step. COTRAN employs a very simplified technique to account for the effects of the reflector core interface. In this method the flux at a reflector node is assumed to be zero and the diffusion coefficient is adjusted until realistic flux distributions are obtained compared to a more sophisticated static calculation. Diffusion coefficient values from 0 to - will yield boundary conditions from reflecting to vacuum respectively.

COTRAN, Thermal-Hydraulic Methodology

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COTRAN solves the integral conservation equations for mass, energy and momentum in an eulerian control volume, V, bounded by a fixed surface A. This surface may include solid interfaces, such as a fuel rod or structural wall and fluid boundaries, but all solid material is outside V and is addressed in the fuel thermal model. The fluid in V is a single component, two phase mixture of liquid and vapor in thermodynamic equilibrium.

The general balance equations are simplified by making the following assumptions:

- Kinetic energy changes are small compared to internal thermal energy changes.
- Work done by body forces and shear stress is considered to be insignificant.
- * Gravity is the only significant body force.
- * Internal heat generation in the fluid is ignored.
- * Fluid flow is one dimensional.

These assumptions are made with the intention that COTRAN would be applied to the analysis of BWR channels with low speed flow and significant surface heat transfer.

The channel balance equations for mass, energy and momentum may be solved in COTRAN by two independent numerical solution procedures. The first, denoted the "Implict Solution Scheme", solves the partial differential equations by finite difference with the channel divided into a finite number of axial sequents. The difference equations derived for the implicit solution scheme are limited to positive flow rates but allow for two phase slip flow with the assumption that the phases are in thermal equilibrium and that the phase velocities and volume factors are uniformly distributed. The second numerical solution technique, denoted the "Explicit Solution Scheme", solves the cell balance equations directly. To solve the cell balance equations, reverse flow is considered but two-phase flow is assumed to be homogeneous which implies that both phase velocities are equal and that the phase distribution is uniform throughout the control volume. The stepwise solution technique employed in the explicit solution scheme limits its use to transient analysis using small time steps.

Stability Analysis

Section 4.3 of XN-NF-80-19(P) describes a calculational method for determining channel hydrodynamic stability using COTRAN. The method is as follows:

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- An option in the COTRAN computer code is selected by which the steady-state thermal hydraulic feedbacks are held constant throughout the transient. This option essentially decouples the neutronics and allows the calculation of the thermal hydraulic characteristics only.
- Time domain transients are initiated by a rapid change in core pressure or inlet flow.
- 3. The resultant time response is analyzed as follows:
 - (a) the inflection points of the time response are determined and connected by straight lines,
 - (b) the amplitude of each maxima and minima are determined from the base line created in step (a) and,
 - (c) consecutive peak amplitudes of each polarity are compared and the resultant decay ratios averaged to yield an overall decay ratio for the operating state.

The hydrodynamic and core reactivity decay ratios determined by the above procedures are then compared to the operational design criteria for the stability analysis which is specified in terms of the decay ratio.

Evaluation of COTRAN Thermal-Hydraulic Methodology

Sections 3.3.2.1, 3.3.2.2, and 3.3.2.3 were reviewed with emphasis on the following areas:

- Mathematical accuracy of equations presented
- Appropriateness of assumptions made in the derivation of the equations and
- Comparison of the proposed method with standard industry techniques.

The equations were found to be accurate with the exception of the last term in equation (3.3.-14) which should be:

$$\int_{W} H(T_{W} - T_{F}) dA$$

which appears to be a typographical error.

The assumptions made are appropriate for BWR channels with low speed flow and significant surface heat transfer and are acceptable for the analysis of the transients described in XN-NF-80-19(P). However, if COTRAN is to be used for the analysis of different transients additional documentation of the proposed method must be provided before such analysis will be considered acceptable.

The basic equations and solution techniques are consistent with industry practice and are acceptable to the staff.

COTRAN Fuel Models

The purpose of the fuel modeling in COTRAN is to calculate the temperature distribution of a fuel rod during steady-state and transient operation. COTRAN has two available fuel models, TEMP and TEMPFD. The first model uses the Methods of Weighted Residuals (MWR) to derive time-dependent temperature

distributions and cladding surface heat flux to the fluid channel. The TEMP model can be applied to transient conditions of power instability (10% power oscillation) and the turbine-trip accident. The second model calculates the temperature and enthalpy distributions of the fuel rod and the cladding surface heat flux using the finite difference numerical scheme. The TEMPFD model is applied to an adiabatic rod-drop accident, which imposes an adiabatic boundary condition at the fuel cladding interface. The adiabatic rod-drop accident can reach a fuel-melting condition due to the reactivity surge and the corresponding fuel rapid heatup. In both models, temperaturedependent fuel thermal conductivity is assumed.

Relationship with the RODEX2 Code

The RODEX2⁽⁸⁾ code calculates fuel performance for the fuel rod thermalmechanical response during normal operation and provides initialization values principally for LOCA analysis. Also it provides initial values of some physical parameters, such as gap conductance and rod geometry, for input into the COTRAN calculations. The same material properties are used consistently in RODEX2 and COTRAN. Although our review of RODEX2 is not complete, we have not previously required explicit approval of the fuels code that initializes the nuclear analysis, so we find this improved (but as yet unapproved) code to be acceptable for this application.

The Method of Weighted Residuals (MWR)

The MWR technique was systematically introduced by Finlayson⁽⁹⁾ and has been adopted in the NRC audit code, $FRAPCON-2^{(10)}$, which performs steady-state thermal-mechanical calculations. We, therefore, believe its use in COTRAN is appropriate.

The TEMP Model

The form of the time-dependent heat conduction equation formulated by the MWR technique in the TEMP model is in terms of an independent variable θ . The definition of θ is as follows:

$$\theta = \int_{T_0}^{T} k(T') dT'/k_0$$

where T is temperature, k(T') is the thermal conductivity at temperature T', K₀ is the thermal conductivity at the reference temperature T₀. The form of this expression is correct, and the fuel thermal conductivity expression is a fourth-order Taylor series expansion of the RODEX2 conductivity correlation.

We have reviewed the MWR technique used to solve this equation and its convergence, and find this model acceptable.

The TEMPFD Model

As mentioned earlier, a finite difference scheme in space and time was used for this model. The admittance of the cladding is set to zero, thereby achieving the adiabatic condition at the fuel-clad interface. Thus, the cladding is decoupled from the fuel regardless of external conditions. An energy balance equation involving enthalpy for the fuel is the starting point for the finite difference formulation. The solution scheme is rather simple and straight-forward, and we find this model acceptable.

(d) XDT Multigroup Difusion Theory

The XDT⁽¹¹⁾ code is used for special diffusion calculations with crosssections generated by XFYRE. The eigenvalue, relative power, multigroup fluxes and flux and volume weighted cross-sections can be computed with XDT. Zero gradient, zero flux and periodic boundary conditions are available. Standard numerical schemes are employed for the solution of the difference equations, with relaxation to speed up convergence.

(e) XMC, Monte Carlo Neutron Transport

XMC is a code designed to represent complex geometrical details and to provide high energy resolution using a maximum of 2,000 microscopic energy

groups. Resolved resonances are computed using the single level Doppler broadened Breit-Wigner formula. The Battelle Master Library or ENDF/B cross-sections can be used. The energy range is 0-10 MeV, the thermal scattering is based on the Haywood Scattering⁽¹²⁾ and the reaction types includes fission, capture, inelastic and elastic scattering, n-2n scattering and isotropic and anisotropic angular distributions. The geometric capabilities and the energy range allow XMC to evaluate nuclear parameters and the effects of water gaps, control blades, burnable poison rods etc. However, XMC has not been benchmarked and as such can only be used as an ancillary code to increase the confidence of other calculational methods.

2.2 <u>Neutronic Core Analysis Methodology</u>

This section covers the methods used for the analysis of steady state operation and specific transient and accidents which use the codes and numerical methods discussed in the preceding paragraphs. The transients and accidents analysed are: control rod drop, fuel misloading, and control rod withdrawal. Also analyzed are reactor core neutronic parameters used as input to LOCA analyses.

(a) Steady State Parameter Calculation

The simulator model XTGBWR is the major tool for the computation of steady state reactor operation. Some important parameters are: power distribution and K_{eff} , control rod patterns, cold critical, cold shutdown margin, exposure distribution, void distribution and void history and time and power dependent xenon and samarium.

XTGBWR uses a modified two group coarse mesh diffusion theory to solve for the fast flux. The thermal flux is generated from the fast flux. Corrections are made to account for strong flux gradient areas near controlled nodes and at interfaces. Two (x, y) and three dimensional (x, y, z)geometries are available. The cross-sections are specified for each node and the code utilizes information for control rods, instantaneous void, void history, power dependent Doppler and time and power dependent xenon and samarium to adjust the cross sections to fit the actual conditions at each node. The outer boundary conditions are specified via an extrapolation distance for each node for the fast flux so as to account for neutron leakage. The thermal flux boundary condition is treated by approximating the thermal flux profile in a one dimensional two group representation of the fuel-reflector interface.

Assuming a uniform fast source in the fuel region the core and reflector equations are solved to determine fast and thermal flux distributions and the K_{eff} eigenvalue. After the first iteration a new source distribution is established and the iteration is repeated until convergence is obtained

The solution for the flux leads to the nodal power which in turn allows the calculation of the average planar linear heat generation rate (APLHGR) the linear heat generation rate (LHGR) and the critical power ratio (CPR) for each fuel node in the core.

Control rod effects for three dimensional core follow calculations are performed with XTGBWR with cross-section for the controlled nodes supplied by XFYRE. This information is supplied in terms of controlled to uncontrolled cross section ratios as a function of voids, exposure and fuel type.

XTGBWR will compute the eigenvalue with zero power, no flow, void or Doppler feedback. Cross-section input for this calculation is obtained from XFYRE as a function of fuel type, void history and moderator temperature.

An obvious extention of the above calculation is the computation of the cold shutdown margin.

XTGBWR requires cross-sections as a function of exposure, voids and material type. The base cross-section are obtained by burning the fuel to whatever exposure is desired. The isotopic composition of each node is a function of the void history. A correction for this is applied through a set of multipliers to the base cross-section.

The void fraction and void history for each node is calculated from the coolant enthalpy. The void fraction correlation in XTGBWR is based on a mechanistic description of two phase separated flow.

The XTGBWR calculates time and power dependent xenon and samarium crosssections from the solution of the differential equations for iodine-xenon and promethium-samarium respectively.

(b) Control Rod Drop Accident

The limiting criteria for the control rod drop accident analysis are: (a) maximum deposited enthalpy no greater than 280 cal/gm of fuel and (b) maximum reactor pressure vessel stresses not to exceed the "Service Limit C" as defined in the ASME Code (Section III). The neutronic parameters which affect the rod drop analysis are: (a) the Doppler coefficient, (b) the maximum rod worth, (c) the power peaking, and (d) the delayed neutron fraction. This accident is analyzed with a one group, time-space neutron diffusion theory calculation in (r-z) geometry using the COTRAN program to account for fuel temperature (Doppler) feedback. The cross-sections are calculated using XTGBWR; the accident may be terminated assuming a scram. At first the cross-sections are input and the initial control rod pattern (exposure and void history effects included) is calculated. Temperature distribution for the Doppler feedback is derived from XFYRE and used to model the feedback for the rod drop accident. The core is divided into three radial zones, one for the central dropped rod (four element module) and two outer zones for the partially controlled regions of the scram rods. The control fractions of the outer regions are varied to obtain different control rod worths and $k_{eff} = 1.000$ with a fully controlled central region. The reactivity input curve for the central control rod is calculated with COTRAN from a series of static calculations at different insertion lengths. The peaking factors (axial x radial) are then determined for a fully withdrawn central rod and variable rod worth. Adiabatic boundary conditions are assumed at the fuel pellet-gap interface, i.e., no heat

transfer to the coolant is allowed which is a conservative assumption. The rod drop analysis is performed for a spectrum of start-up conditions with respect to coolant temperature, core size, composition, and exposure. Scram is assumed when the power reaches the scram level. The total transient time is about six seconds. The maximum fuel enthalpy is calculated from the peaking factors for each transient. The maximum reactor pressure is calculated from the total energy generated during the transient.

(c) Fuel Misloading Analysis

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The analysis considers both fuel element misorientation and mislocation. The criterion is the resulting difference in the Minimum Critical Power Ratio AMCPR which is added to the safety-limit MCPR to determine the operating limit. Misorientation error results from the fact that in BWR/2, 3, and 4 lower enrichment is provided for the wider gap side and will result in higher power fuel rods. The calculation begins with a rotated assembly (90 and 180 degrees) depletion using XFYRE. A pin-by-pin calculation follows, using XDT for a four assembly module where one is assumed to be rotated 90 and 180 degrees. Correction factors are obtained for a single bundle local peaking. For fuel assemblies with MCPR control rod step through calculations are performed without fuel misloading. The AMCPR is determined for misorientation, i.e., with no error minus the MCPR with a rotated assembly. Finally, the limiting MCPR is determined.

The fuel misloading error calculations are handled in a similar manner. A high reactivity fuel assembly is assumed to be loaded in place of a low reactivity fuel assembly, which will result in higher local power. The MCPR is computed by direct CPR comparison of misloaded and normal assemblies. First it is assumed that the control rod and loading patterns have been developed using the XTGBWR. The next step is to identify candidate mislocation assemblies, i.e., those which involve a high reactivity assembly in a low reactivity location. Each such location is then burned through the cycle to determine the MCPR for the misloaded and under normal loading conditions. The Δ MCPR is then obtained by direct CPR subtraction and comparison.

(d) Control Rod Withdrawal

This is the analysis of the transient resulting from the withdrawal of a fully inserted control rod until the motion is stopped by a rod block. It is assumed that the reactor is operating at power, that the maximum reactivity rod is being withdrawn and the operator ignores the local power range monitor (LPRM) alarm. The calculation determines the Δ MCPR as a function of the rod block setting. The analysis is performed with XTGBWR as a series of steady-state calculations.

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(e) Neutronic Reactivity Parameters

Plant transients and LOCA analysis require neutronic parameters which characterize the particular core state. Such parameters are: void reactivity coefficient, Doppler reactivity coefficient, scram reactivity distribution, delayed neutron fraction, and prompt neutron lifetime. Each of these parameters is discussed in the following:

- (e.1) Void reactivity coefficient is the fractional change of core reactivity due to a change in the average fractional void. It is the most important of the reactivity feedbacks and is computed with XTGBWR. These calculations consist of a k_{eff} with average voids and a series of the same calculation with different void fractions.
- (e.2) The Doppler reactivity coefficient is computed with XFYRE as a function of fuel temperature T_f and fitted to the $\sqrt{T_f}$ for each fuel type exposure and void.
- (e.3) Scram reactivity is the k_{eff} as a function of the scram bank insertion. The total scram reactivity is calculated using COTRAN. At first a static solution is generated with XTGBWR which provides average cross-section as a function of axial height. Before a COTRAN transient is run it is checked for consistency with XTGBWR.

The time dependent scram reactivity is calculated by subtracting the void reactivity from the total reactivity calculated by COTRAN. Adjustment factors are applied to account for particular plant conditions like inlet pressure, etc.

- (e.4) The deived neutron fraction B_{eff} depends upon the fuel type, hence, it is determined by exposure and volume weighting of the fuel. This is done with XFYRE.
- (e.5) XFYRE, likewise, calculates prompt neutron lifetime, i.e., by exposure and volume weighting of the fuel.

2.3 Neutronics Methods Verification

In the preceding sections the essential features of the calculational techniques and their applications have been described. Some of the computer codes described are known to the staff but there are parts of the calculations or input to these codes which is prepared with auxiliary codes not as well known for example HRG, DASQHE, etc., In this case, the overall performance of the proposed computational scheme cannot be predicted on the basis of the performance of the known parts. Hence, their review is based on verification and error measurement with experimental data. The following, deal with the predictive capability of each of the codes and an assessment of their statistical error. The information pertains not only to the basic document under review (i.e., XN-NF-80-19(P), Volume 1) but also to Supplements 1 and 2 of the same which were submitted by the applicant as a result of questions raised by the staff and its consultants (BNL) during the review. Comparison of calculations to accepted codes and codes of higher order calculation such as Monte Carlo is also made to increase the confidence of the proposed method.

The major set of data which were used are from the Quad Cities 1 gamma scan measurements and TIP data and the Dresden Units 2 and 3 TIP data. Additional more specific data from other reactors will be mentioned in the appropriate section.

(a) XFYRE Verification

The data used are from: (a) Quad Cities 1 end of Cycles 2, 3, and 4, (b) isotopic measurements of a Garigliano fuel assembly, and (c) XMC calculations. The Quad Cities data constitute extensive comparisons of local power as a function of void, burnup, fuel assembly location, and fuel type. There is reasonable agreement between calculated and measured values. Because this comparison covers such a wide range of parameters, it constitutes a significant part of the qualification of the methodology. The Garigliano data are local power distributions from gamma scans of two fuel assemblies. They have variable void fractions, burnup, and include gadolinia fuel rods. Again the comparisons are very reasonable. Finally comparisons are made with XMC calculations and show good agreement. The Quad Cities data show calculated versus measured data from assembly pin gamma scan results (a) as a function of assembly height (with concurrent variations of local burnup and void) and (b) fuel type. The average difference and the standard deviation are calculated for every 8x8 rod assembly at each 6 inch height interval. The maximum difference is -3.6 percent and the standard deviation is around 2.0 percent. The agreement is reasonable and acceptable.

mg.

One of the Garigliano fuel assemblies has been analyzed for fuel isotopic composition. The calculated values of the isotopes show small relative error for those with high concentration and larger error with derivative isotopes of small concentration which is typical in similar calculations. These comparisons seem reasonable and are acceptable. A more detailed comparison of the local r wer distribution and of the effect of control, local voids, and temperature

performed to the results of the Monte Carlo code XMC. XMC is not an approved code but it represents state-of-the-art calculations. The comparison shows good agreement and reinforces the conclusion that XFYRE can compute detailed power distributions. The K= comparison computed with XFYRE are within the uncertainty limits of the corresponding XMC calculation.

(b) XTGBWR Verification

The XTGBWR core simulator is verified by comparison of calculated and measured reactor parameters as a function of power level and exposure. Extensive K_{eff} measured values from Dresden-3 Cycles 5, 6 and 7, Dresden-2 cycles 6 and 7, and Quad Cities-1 Cycles 1, 2 and 3 have been presented. Core exposure and corresponding values of power, voids and flow were varied. The maximum deviation of the calculated values was .007 but the most frequent computed deviation was .001. The hot K_{eff} for the same reactors plotted vs exposure show good uniformity. Calculated values of K_{eff} for Dresden-3, Dresden-2, and Quad Cities-1 as a function of control rod sequence, moderator temperature and exposure show the same consistency of prediction as previously. Extensive TIP measurements (i.e., axial power profiles) have been measured at Dresden-3, Dresden-2, and Quad Cities-1 at the beginning, middle, and end of cycle. XTGBWR full core, 24 axial node calculations corresponding to each TIP traverse indicate good agreement with the measured values. Better agreement was obtained with the end of cycle, fuel assembly gamma scan measurements from Quad Cities Cycles 2 and 4. The performance of XTGBWR in extensive comparisons with measured data is excellent and acceptable.

(c) COTRAN Acceptability and Verification

The COTRAN Thermal-Hydraulic model described in XN-NF-80-19(P) is adequate for the analysis of BWRs with low speed flow and significant surface heat transfer and is acceptable to the staff for the analysis of the transients and accidents described in the report. Likewise, we find the COTRAN fuel models acceptable as used in ENC's nuclear methodology.

COTRAN verification is based on Peach Bottom-2 turbine trip transients, in which the measured and calculated power response is compared. The methods involved in the analysis provided by COTRAN are acceptable and the results of the Peach Bottom-2 comparison are adequate.

2.4 Measured Power Distribution Uncertainties

Uncertainties in the measured values of the reactor parameters and in particular local power are inevitable. The same is true with the input parameters to computer codes. The purpose of this section is to describe the methodology for the quantification of the uncertainty in the measurement of reactor parameters. In power measurements the parameters are the local pin power, the axial and the radial bundle power distribution. In the set of measurements used for the verification of the BWR neutronic methodologies, the computed power distribution uncertainties were: (a) local pin power 2.5 percent, (b) axial bundle power 3.0 percent, and (c) radial bundle power 5.3 percent. It is assumed that the data have normal distribution. The uncertainties are relative in terms of standard deviations. A very large number of data points were utilized for the determination of the uncertainties, providing a good statistical base. TIP measurements from 15 reactor cycles were used along with seven fuel bundle pin-by-pin gamma scans.

(a) Determination of Measured Power Distribution

The measured reactor power distribution is a combination of measured and calculated quantities. The measured data are the readings (and calibrations) of the fixed LPRMs and movable TIPs. The calculated data include the relative core nodal power distribution, the incore detector response distribution and local peaking factors for the fuel rods. The measured nodal power is computed as the average of four calculations of measured nodal power of the nearest radial locations. In this procedure the TIP and LPRM responses are combined.

(b) <u>Derivation and Estimation of the Uncertainty in the Measured</u> Power Distribution

Assuming normal distribution around 1.0 of the ratio of each measured variable to its calculated value, the difference (V-1) represents the distribution of the relative uncertainties associated with the variable V.

The methodology is based on a Taylor series expansion for the calculation of the partial derivatives with respect to the independent variables. All variables are assumed to be independent which is a conservative assumption.

For each set of measured data the components of its uncertainty are computed separately. For the TIP distributions (which form the largest part of the data) the uncertainty has three components: (a) due to the calibration of the TIP system, (b) due to the detector response, and (c) due to the interpolation procedure which utilizes calculated data. The relative standard deviation is the statistical combination (square root of the sum of the squares) of the above components. Symmetric TIP data are used for the TIP system uncertainty. Data were untilized from Quad Cities-1 Cycles 1-5. Duad Cities-2 Cycle 4-5, Dresden-2 Cycles 4-7 and Dresden-3 Cycle 4-7. The standard deviation is 6.0 percent. The uncertainty of the LPRM detector response of 3.4% has been adopted from the previous GE study reported in NEDO-20340. The uncertainty due to the interpolation procedure is computed by the comparison of the synthesized TIP distribution and the measured distribution. (In this comparison, the top and bottom two axial nodes are omitted.) The calculated TIP uncertainty is determined from the difference of the calculated and measured detector distributions. The uncertainty of the measured distribution is removed from the uncertainty of the difference by subtracting. The power distributions are calculated using the XTGBWR code and the corresponding measured values are determined from the TIP responses multiplied by appropriate response-to-power factors. The local power uncertainty including uncertainty in local peaking factors are determined by comparing the calculated pin powers to the pin-by-pin gamma scans of fuel bundles irradiated in the reactor. (This comparison is done in terms of La-140 activity rather than power.) Results of the measured values were obtained from three EPRI reports. EPRI-NP-214, EPRI-NP-512, and one unpublished. The measured relative uncertainty (from the EPRI reports) is about 1.5 percent.

The overall measured nodal power distribution uncertainty is 6.07% in terms of relative standard deviation and 5.82% for the radial pin power distribution.

In the preceding paragraphs it has been shown that the uncertainties in the measured power levels and the corresponding calculated values have been statistically analyzed in an acceptable manner based on a very wide and appropriate data base. The methodology and the data base for the statistical analysis of the uncertainty are acceptable.

3.0 APPLICATIONS OF THE NEUTRONIC METHODOLOGY

As pointed out in the preceeding paragraphs the core related events and transients for which the methodology would be applicable are: (a) control rod drop, (b) fuel assembly misloading, and (c) control rod withdrawal transient. The methodology is applied to these events with conservatively realistic assumptions. The results of these calculations are not compared to experimental nor other code results and as such do not constitute part of the code verification, nevertheless, they add to the credibility of the method and as such are reviewed briefly.

(a) Control Rod Drop Accident

The important parameters in the analysis of a rod drop accident are: rod worth, the Doppler coefficient, the delayed neutron fraction and the local peaking factor. The condition to be satisfied is that the maximum fuel rod enthalpy remain below 280 cal/gm. The range of the variables was selected to envelope anticipated reactor operating conditions for a typical jet pump BWR cycle loaded with a mixture of exposed and fresh reload fuel assemblies. Initial core conditions assumed: hot zero power at saturated moderator temperature of 546°F. The values of the parameter ranges are as follows: Rod Worth 8 mk and 12 mk. Doppler coefficient -11.5×10^{-6} , -10.5×10^{-6} , -9.5×10^{-6} and $-8.5 \times 10^{-6} \ \Delta k/k/°F$ at 773°F, delayed neutron fraction 4.5×10^{-3} , 5.5×10^{-3} and 6.5×10^{-3} and local peaking factor 1.30. Combinations of individual values from the above ranges were selected. Other parameter values were: scram signal at 120% of of rated power, scram delay time .30 seconds, scram velocity 2.54 ft/sec, dropped rod velocity of 3.11 ft/sec and zero voids. The results of these studies indicate that the

sorst combination of parameter is for the highest control rod worth and the smallest Doppler coefficient which yields about 230 cal/gm i.e., well below the limit of 280 cal/gm.

(b) Fuel Misloading Error

There are two parts in the analysis of the fue! misloading error i.e. misorientation and mislocation of a fuel assembly. Further, the assume ion is made that the error is not discovered during core verification, hence. the purpose of the analysis is to determine the operating MCPR limit. (A combination of misoriention and mislocation is not considered because of the very low probability of occurence.) The misorientation calculation determines the largest MCPR for a misoriented assembly during a given cycle. Depletion calculations are performed assuming a fuel assembly rotated by 90° and 180°, computing the local peaking factors at the bottom, midplane and top of the channel assuming 0, 40, and 70 percent voids respectively. For the control rod step through calculation the MCPR is noted with and without the misorientation from which the MCPR is concluded. This analysis is cycle dependent. The fuel misloading analysis has been modified from that described in XN-NF-80-19(P). This modification refers to the location of the pair of fuel assemblies which would result in the maximum Linear Heat Generation Rate (LHGR) which in turn could result in the highest AMCPR. The proposed modification consists of a simplifed diffusion calculation procedure which allows a comparison of the relative power of a misloaded assembly. Once candidate locations have been chosen a burn through the cycle calculation is performed, with and without the misloaded assembly, and the MCPR is calculated. The procedure also is cycle dependent.

Using the typical reactor parameters described above the largest mislocation Δ MCPR was found to be .157.

(c) Control Rod Withdrawl

A control rod withdrawal analysis has been performed to estimate the MCPR and maximum LHGR in such a transient. A starting control rod pattern is established for the typical BWR reactor of these applications and a central

control rod is withdrawn from the fully inserted position. Rod withdrawal results in an increase of the LHGR and decrease of the CPR. The computed MLHGR and MCPR are compared to values of other transients to establish operating limits for the reactor.

4.0 EVALUATION PROCUEDURE

The topical report XN-NF-80-19(P) "Exxon Nuclear Methodology for Boiling Water Reactors-Neutronic Methods for Design and Analyses", Vol. 1 and Supplements 1 and 2 has been reviewed by the Core Performance Branch. The Core Performance Branch has been assisted by BNL under Technical Assistance contract A3370: by D. Cokinos in the review of XTGBWR and overall review. P. Neogy in the review of XFYRE and XDT. L. Eisenhart in the review of XMC the Exxon Monte Carlo code, and H. S. Cheng in the review of COTRAN, all under the direction of J. Carew. The questions formulated by the staff and BNL resulted in the submittal of two additional volumes i.e., Supplements 1 and 2 to the main report. The neutronics include (1) a fuel assembly depletion model (b) a core simulator (c) a reactor kinetics (d) multigroup diffusion theory code and (e) a Monte Carlo code. The use of the codes and models is outlined in a neutronic core analysis methodology section which includes transient analysis and more specifically the control rod drop accident, fuel misloading, stability analysis, control rod withdrawal and the calculation of the reactivity parameters. The neutronic methods verification has been accomplished with the use of an extensive data base from measurements of power and power distribution at Quad Cities 1 and 2, Dresden 2 and 3, Oyster Creek and Carigliano. The methodology and the application for the uncertainty of the data has been developed in detail. (The channel hydrodynamic stability criterion is subject of a separate staff review.)

The review of topical report XN-NF-80-19(P) has been conducted within the guidelines provided by the Standard Review Plan, Section 4.3. Sufficient information is included to permit a knowledgeable person to conclude that the methods and techniques employed are state-of-the-art and are acceptable.

5.0 REGULATORY POSITION

3 . 1

The subject report and its supplements describe acceptable analytical methods and computer codes for calculating the neutronic behavior of BWRs with fuel loadings and geometric properties similar to those analyzed in the report. It has been shown that these methods can predict to an acceptable accuracy the physics characteristics of operating reactors. Therefore, this report (and its supplements) may be referenced in future licensing applications relating to BWR physics analysis. We do, however, recommend that the analytical models be continuously verified to insure their applicability.

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EXXON NUCLEAR METHODOLOGY FOR BOILING WATER REACTORS VOLUME 1 NEUTRONIC METHODS FOR DESIGN AND ANALYSIS

Approved:

To Knipinski. 8/1/80

T. L. Krysinski, Manager BWR Neutronics

Approved:

. . . .

D. H. Timmons, Manager Neutronics Development

Approved:

HAUIDO im

R. B. Stout, Manager Neutronics & Fuel Management

Approved:

G. A. Sofer, Manager Nuclear Fuels Engineering

Vol. 1

TABLE OF CONTENTS

Sect	ion					Page				
1.0	INTR	RODUCTION								
2.0	SUMMARY									
	2.1	FUEL A	ASSEMBLY DEPLETION MODEL (XFYRE)			2				
	2.2	CORE S	SIMULATOR (XTGBWR)			3				
	2.3	REACTO	DR KINETICS MODEL (COTRAN)			4				
	2.4	DIFFUS	SION THEORY MODEL (XDT)			5				
	2.5	MONTE	CARLO MODEL (XMC)			5				
	2.6	CORE A	NALYSIS METHODOLOGY			6				
	2.7	NEUTRONICS METHODS VERIFICATION								
	2.8	MEASURED POWER DISTRIBUTION UNCERTAINTY								
3.0	NEUTRONICS MODELS FOR BWR REACTOR CORE CALCULATIONS									
	3.1	FUEL A	ASSEMBLY DEPLETION MODEL (XFYRE)			12				
		3.1.1	Basic Cross Section Library			13				
		3.1.2	Thermal Cross Sections			13				
		3.1.3	Epithermal Cross Sections			15				
		3.1.4	Control Rod Cross Sections			16				
		3.1.5	Neutron Flux and Power Calculations			17				
		3.1.6	Depletion Calculation			20				
		3.1.7	Xenon and Samarium			22				
		3.1.8	Restart Calculations			23				
		3.1.9	Incore Detector Parameters			23				
TABLE OF CONTENTS CONTINUED

Section									Page
3.2	CORE SI	IMULATOR (XTGBWR)	.,						35
	3.2.1	Core Geometry							36
	3.2.2	Diffusion Theory Model	.,						37
	3.2.3	Boundary Conditions	È						43
	3.2.4	Exposure and Void Dependent Cross Section	s						43
	3.2.5	Control Rod Effects				ķ			44
	3.2.6	Equilibrium and Time Dependent Xenon							44
	3.2.7	Samarium Buildup			4				48
	3.2.8	Doppler Broadened Cross Section	. '						49
	3.2.9	Coolant Flow Distribution	•			÷			50
	3.2.10	Steam Quality and Void Fraction	÷		.*		•		51
	3.2.11	Thermal Limits Calculation						•	51
	3.2.12	Incore Detector Response Calculation							52
	3.2.13	Zero Power Critical Option							54
3.3	REACTO	R KINETICS MODEL (COTRAN)							59
	3.3.1	COTRAN Neutronics Model							60
	3.3.2	COTRAN Thermal-Hydraulic Model							65
	3.3.3	Description of Code Mechanics and Output	Fei	atu	ire	s			90
	3.3.4	Input and Use of Cross Sections							91

TABLE OF CONTENTS CONTINUED

Sect	ion	Pa	ige
	3.4	MULTIGROUP DIFFUSION THEORY MODEL (XDT)	93
	3.5	MONTE CARLO MODEL (XMC)	80
		3.5.1 The XMC Code Package	10
		3.5.2 The XMC Monte Carlo Routines	11
		3.5.3 Treatment of Neutron Absorption	12
		3.5.4 The Neutron Flux and the Neutron Beam	15
		3.5.5 Energy Group Structures	17
		3.5.6 Code Check Tallies	18
		3.5.7 The Source Routine	18
		3.5.8 The XMC Geometry Routines	19
		3.5.9 Path Length Calculations	19
		3.5.10 The XMC Collision Routines	20
		3.5.11 The XMC Output Routines	24
		3.5.12 Cross Section Library	26
		3.5.13 The XMC Loader	29
	3.6	REFERENCES	39
4.0	NEUTR	ONICS CORE ANALYSIS METHODOLOGY	41
	4.1	CONTROL ROD DROP ACCIDENT	41
	4.2	FUEL MISLOADING ANALYSIS	43
		4.2.1 Fuel Misorientation Error	44
		4.2.2 Fuel Mislocation Error	45

TABLE OF CONTENTS CONTINUED

Sect	ion	Page
	4.3	STABILITY ANALYSIS
	4.4	NEUTRONIC REACTIVITY PARAMETERS
		4.4.1 Void Reactivity Coefficient
		4.4.2 Doppler Reactivity Coefficient
		4.4.3 Scram Reactivity
		4.4.4 Delayed Neutron Fraction
		4.4.5 Prompt Neutron Lifetime
	4.5	CONTROL ROD WITHDRAWAL
	4.6	REFERENCES
5.0	NEUT	RONICS METHODS VERIFICATION
	5.1	XFYRE VERIFICATION
	5.2	XTGBWR VERIFICATION
	5.3	COTRAN VERIFICATION
	5.4	REFERENCES
6.0	MEAS	SURED POWER DISTRIBUTION UNCERTAINTY
	6.1	MEASURED POWER DISTRIBUTION DETERMINATION
	6.2	UNCERTAINTY DERIVATION
	6.3	ESTIMATION OF UNCERTAINTY
		6.3.1 Detector Measurements: δ _F
		6.3.2 Calculated Detector Response Distribution: δ_{T} 202
		6.3.3 Calculated Nodal Power Distribution: δ _p
		6.3.4 Local Pin Distribution: δ ₁

LIST OF TABLES

Table				Page
3.1-1	ISOTOPES NORMALLY USED IN XFYRE	•		24
32	TYPICAL EXPOSURES (MWD/MTU) FOR XFYRE DEPLETION CALCULATIONS			25
3.1-3	CORNABLE ISOTOPES IN XFYRE			26
3.3-1	DEFINITIONS OF VARIABLES FOR THERMAL-HYDRAULIC SOLUTIONS			94
3.5-1	ISOTOPES IN THE XMC CROSS SECTION LIBRARY			131
3.5-2	XMC CODE BOUNDARY FUNCTIONS		÷ .	132
5.1-1	GARIGLIANO ISOTOPIC COMPARISON MEASURED/CALCULATED DATA			162
5.1-2	XMC (MONTE CARLO)/XFYRE K∞ COMPARISONS FOR BWR RELOAD FUEL ASSEMBLIES			163
5.2-1	XTGBWR CALCULATED KEFF AND AVERAGE VOIDS FOR DRESDEN-3 CYCLE 5			164
5.2-2	XTGBWR CALCULATED KEFF AND AVERAGE VOIDS FOR DRESDEN-3 CYCLE 6			165
5.2-3	XTGBWR CALCULATED K AND AVERAGE VOIDS FOR OYSTER CREEK CYCLE FF			166
5.2-4	XTGBWR CALCULATED K AND AVERAGE VOIDS FOR OYSTER CREEK CYCLE 8			167
5.2-5	XTGBWR CALCULATED K AND AVERAGE VOIDS FOR QUAD CITIES CYCLE 1			168
5.2-6	XTGBWR CALCULATED KEFF AND AVERAGE VOIDS FOR QUAD CITIES CYCLE 2			169

LIST OF FIGURES

			Page
TYPICAL 8x8 BWR FUEL ASSEMBLY WITH A WIDE AND NARROW WATER GAP			. 27
CYLINDRICAL GEOMETRY FOR THERMOS CALCULATION OF NON-GADOLINIA FUEL ROD			. 2
CYLINDRICAL GEOMETRY FOR THERMOS CALCULATION OF FUEL ROD CONTAINING GADOLINIA			. 29
ACTUAL GEOMETRY AND SLAB GEOMETRY FOR THERMOS CONTROL ROD CROSS SECTION CALCULATION			. 30
XFYRE MESH BOUNDARIES FOR AN 8x8 BWR FUEL ASSEMBLY WITH A WIDE AND NARROW WATER GAP			. 31
XFYRE ISOTOPIC CHAIN			. 32
RELATIVE THERMAL ABSORPTION CROSS SECTION OF GADOLINIUM-157 AS A FUNCTION OF FUEL ASSEMBLY EXPOSURE			. 33
RELATIVE GADOLINIUM-157 NUMBER DENSITY BY SUBREGION AS A FUNCTION OF FUEL ASSEMBLY EXPOSURE			. 34
TYPICAL BWR REACTOR CORE GEOMETRY			. 55
XTGBWR QUARTER CORE SYMMETRY BOUNDARY CONDITIONS			. 56
THREE DIMENSIONAL MESH DESCRIPTION IN XYZ GEOMETRY			. 57
TIP/LPRM IN-CORE ASSEMBLY CROSS SECTION			. 58
CHANNEL CONTROL VOLUME FOR THERMAL-HYDRAULIC BALANCE EQUATIONS			. 95
PLACEMENT OF VARIABLES FOR IMPLICIT SOLUTION			. 96
PLACEMENT OF VARIABLES FOR EXPLICIT SOLUTION			. 97
MATRIX ELEMENTS FOR THIRD ORDER COLLOCATION	÷		. 98
	TYPICAL 8x8 BWR FUEL ASSEMBLY WITH A WIDE AND NARROW WATER GAP CYLINDRICAL GEOMETRY FOR THERMOS CALCULATION OF CYLINDRICAL GEOMETRY FOR THERMOS CALCULATION OF FUEL ROD CONTAINING GADOLINIA ACTUAL GEOMETRY AND SLAB GEOMETRY FOR THERMOS CONTROL ROD CROSS SECTION CALCULATION XFYRE MESH BOUNDARIES FOR AN 8x8 BWR FUEL ASSEMBLY WITH A WIDE AND NARROW WATER GAP XFYRE ISOTOPIC CHAIN RELATIVE THERMAL ABSORPTION CROSS SECTION OF GADOLINIUM-157 AS A FUNCTION OF FUEL ASSEMBLY EXPOSURE RELATIVE GADOLINIUM-157 NUMBER DENSITY BY SUBREGION AS A FUNCTION OF FUEL ASSEMBLY EXPOSURE TYPICAL BWR REACTOR CORE GEOMETRY XTGBWR QUARTER CORE SYMMETRY BOUNDARY CONDITIONS TIP/LPRM IN-CORE ASSEMBLY CROSS SECTION TIP/LPRM IN-CORE ASSEMBLY CROSS SECTION CHANNEL CONTROL VOLUME FOR THERMAL-HYDRAULIC BALANCE EQUATIONS PLACEMENT OF VARIABLES FOR EXPLICIT SOLUTION MATRIX ELEMENTS FOR THIRD ORDER COLLOCATION	TYPICAL 8x8 BWR FUEL ASSEMBLY WITH A WIDE AND NARROW WATER GAP CYLINDRICAL GEOMETRY FOR THERMOS CALCULATION OF NON-GADOLINIA FUEL ROD CYLINDRICAL GEOMETRY FOR THERMOS CALCULATION OF FUEL ROD CONTAINING GADOLINIA ACTUAL GEOMETRY AND SLAB GEOMETRY FOR THERMOS CONTROL ROD CROSS SECTION CALCULATION XFYRE MESH BOUNDARIES FOR AN 8x8 BWR FUEL ASSEMBLY WITH A WIDE AND NARROW WATER GAP XFYRE ISOTOPIC CHAIN RELATIVE THERMAL ABSORPTION CROSS SECTION OF GADOLINIUM-157 AS A FUNCTION OF FUEL ASSEMBLY EXPOSURE RELATIVE GADOLINIUM-157 NUMBER DENSITY BY SUBREGION AS A FUNCTION OF FUEL ASSEMBLY EXPOSURE TYPICAL BWR REACTOR CORE GEOMETRY XTGBWR QUARTER CORE SYMMETRY BOUNDARY CONDITIONS THREE DIMENSIONAL MESH DESCRIPTION IN XYZ GEOMETRY TIP/LPRM IN-CORE ASSEMBLY CROSS SECTION CHANNEL CONTROL VOLUME FOR THERMAL-HYDRAULIC BALANCE EQUATIONS PLACEMENT OF VARIABLES FOR EXPLICIT SOLUTION MARRIX ELEMENTS FOR THIRD ORDER COLLOCATION	TYPICAL 8x8 BWR FUEL ASSEMBLY WITH A WIDE AND NARROW WATER GAP CYLINDRICAL GEOMETRY FOR THERMOS CALCULATION OF NON-GADOLINIA FUEL ROD CYLINDRICAL GEOMETRY FOR THERMOS CALCULATION OF FUEL ROD CONTAINING GADOLINIA ACTUAL GEOMETRY AND SLAB GEOMETRY FOR THERMOS CONTROL ROD CROSS SECTION CALCULATION XFYRE MESH BOUNDARIES FOR AN 8x8 BWR FUEL ASSEMBLY WITH A WIDE AND NARROW WATER GAP XFYRE ISOTOPIC CHAIN RELATIVE THERMAL ABSORPTION CROSS SECTION OF GAOULINIUM-157 AS A FUNCTION OF FUEL ASSEMBLY EXPOSURE RELATIVE GADOLINIUM-157 NUMBER DENSITY BY SUBREGION AS A FUNCTION OF FUEL ASSEMBLY EXPOSURE TYPICAL BWR REACTOR CORE GEOMETRY YIGBWR QUARTER CORE SYMMETRY BOUNDARY CONDITIONS THREE DIMENSIONAL MESH DESCRIPTION IN XYZ GEOMETRY TIP/LPRM IN-CORE ASSEMBLY CROSS SECTION CHANNEL CONTROL VOLUME FOR THERMAL-HYDRAULIC BALANCE EQUATIONS PLACEMENT OF VARIABLES FOR EXPLICIT SOLUTION MATRIX ELEMENTS FOR THIRD ORDER COLLOCATION

LIST OF FIGURES CONTINUED

Figure	이 같은 것이 같은 것이 같은 것을 많은 것이 같이 많은 것이 같이 같이 많이 많이 많이 했다.	Page
3.4-1	MESH DESCRIPTION	. 105
3.4-2	SCHEMATIC DIAGRAM OF 1-D REACTOR	. 106
3.5-1	THE XMC CODE PACKAGE	. 133
3.5-2	GENERAL FLOW DIAGRAM OF THE MONTE CARLO CODE FOR NEUTRONS IMPORTANCE WEIGHTING NOT BEING USED	. 134
3.5-3	BWR FUEL ASSEMBLY GEOMETRY	. 135
3.5-4	PWR QUARTER BUNDLE GEOMETRY	. 136
3.5-5	BOX TYPES WITH DIFFERENT NUMBERS OF INTERNAL REGIONS	. 137
3.5-6	EXAMPLES OF BOXES OF TYPE 1 AT ROD POSITION 1, 11, AND 28, AND AND A BOX OF TYPE 2 AT ROD POSITION 15. NOTE THAT INSERTION OF A BOX AT AN OFF DIAGONAL POSITION ACTUALLY INSERTS TWO BOXES SUCH AS THE BOX AT ROD POSITION 11	. 138
4.1-1	CYLINDRICAL GEOMETRY FOR CONTROL ROD DROP ANALYSIS	. 154
4.1-2	TYPICAL CONTROL FRACTION α_1 VS α_2 FOR CENTRAL ROD FULL IN OR FULL OUT	. 155
4.1-3	TYPICAL SCRAM BANK REACTIVITY WORTH CURVE	. 156
4.2-1	FOUR BUNDLE MODULE WITH MISORIENTATED FUEL ASSEMBLY, 180° ROTATION	. 157
4.2-2	ELEVATION VIEW OF MISORIENTATED FUEL ASSEMBLY	• 158
5.1-1	COMPARISON OF XFYRE CALCULATED/GAMMA SCAN MEASURED LOCAL POWER DISTRIBUTION FOR ENC 8x8 RELOAD FUEL	. 170
5.1-2	COMAPRISON OF XFYRE CALCULATED/GAMMA SCAN MEASURED LOCAL POWER DISTRIBUTION FOR ENC 8x8 RELOAD FUEL	. 171
5.1-3	COMPARISON OF XFYRE CALCULATED/GAMMA SCAN MEASURED LOCAL POWER POWER DISTRIBUTION FOR ENC 8x8 RELOAD FUEL	. 172

LIST OF FIGURES CONTINUED

Figure	Page	
5.1-4	COMPARISON OF XFYRE CALCULATED/GAMMA SCAN MEASURED LOCAL POWER DISTRIBUTION FOR ENC 8x8 RELOAD FUEL	
5.1-5	COMPARISON OF XFYRE CALCUALTED/GAMMA SCAN MEASURED LOCAL POWER DISTRIBUTION FOR ENC 8x8 RELOAD FUEL	
5.1-6	FUEL ROD POSITION IDENTIFICATION FOR GARIGLIANO ISOTOPIC COMPARISON	
5.1-7	XFYRE/XMC (MONTE CARLO) CALCULATED LOCAL POWER DISTRIBUTION FOR OYSTER CREEK BWR RELOAD FUEL, 0%, V - NO GADOLINIA - NO CONTROL 176	
5.1-8	XFYRE/XMC (MONTE CARLO) CALCULATED LOCAL POWER DISTRIBUTION FOR OYSTER CREEK BWR RELOAD FUEL, 32% V - WITH GADOLINIA - NO CONTROL . 177	
5.1-9	XFYRE/XMC (MONTE CARLO) CALCULATED LOCAL POWER DISTRIBUTION FOR OYSTER CREEK BWR RELOAD FUEL, 32% V - NO GADOLINIA - NO CONTROL 178	
5.1-10	XFYRE/XMC (MONTE CARLO) CALCULATED LOCAL POWER DISTRIBUTION FOR OYSTER CREEK BWR RELOAD FUEL, 32% V - NO GADOLINIA - CONTROLLED 179	
5.1-11	XFYRE/XMC (MONTE CARLO) CALCULATED LOCAL POWER DISTRIBUTION FOR OYSTER CREEK BWR RELOAD FUEL, 64% V - NO GADOLINIA - NO CONTROL 180	
5.2-1	CALCULATED KEFF AS A FUNCTION OF CYCLE EXPOSURE	
5.2-2	DRESDEN-3 MEASURED AND XTGBWR CALCULATED TIP COMPARISON	
5.2-3	DRESDEN-3 MEASURED AND XTGBWR CALCULATED TIP COMPARISON 183	
5.2-4	DRESDEN-3 MEASURED AND XTGBWR CALCULATED TIP COMPARISON 184	
5.2-5	DRESDEN-5 MEASURED AND XTGBWR CALCULATED TIP COMPARISON 185	
5.2-6	DRESDEN-3 MEASURED AND XTGBWR CALCULATED TIP COMPARISON 186	
5.2-7	OYSTER CREEK MEASURED AND CALCULATED TIP COMPARISON	
5.2-8	OYSTER CREEK MEASURED AND CALCULATED TIP COMPARISON	
5.2-9	OYSTER CREEK MEASURED AND XTGBWR CALCULATED TIP COMPARISON 189	

LIST OF FIGURES CONTINUED

Figures	<u>s</u>	Page
5.2-10	OYSTER CREEK MEASURED AND CALCULATED TIP COMPARISON	190
5.2-11	OYSTER CREEK MEASURED AND CALCULATED TIP COMPARISON	191
5-2-12	OYSTER CREEK MEASURED AND CALCULATED TIP COMPARISON	. 192
5.2-13	QUAD CITIES 1 EOC2 FUEL ASSEMBLY GAMMA SCAN COMPARISON, XTGBWR CALCULATED/MEASURED LA-140	. 193
5.3-1	COTRAN - PEACH BOTTOM-2 LOW FLOW STABILITY TEST COMPARISON PERIODIC PRESSURE REGULATOR SETPOINT STEP CHANGES	. 194
5.3-2	COTRAN - PEACH BOTTOM-2 LOW FLOW STABILITY TEST COMPARISON RANDOM PRESSURE REGULATOR SETPOINT CHANGES	. 195
6.1	LPRM IN-CORE ASSEMBLY CROSS SECTION	. 205

1.0 INTRODUCTION

The purpose of this report is to document the Exxon Nuclear Company (ENC) methods for the neutronic analysis of boiling water reactors (BWR's). The report is being issued at this time for NRC and utility customer review. Included in this report are local neutronic analysis models applicable to individual fuel assemblies and neutronics core analysis methodology applicable to the entire core. Uncertainty analysis methodology and verification of the calculational results are also covered. The neutronics core analysis methodology includes control rod drop, control rod withdrawal, fuel misloading, reactor core and channel hydrodynamic stability, and neutronic input to the total nuclear plant transient analysis. The neutronic methods are verified by comparing the calculational results with measured reactor data and with higher order calculations. The power distribution uncertainty methodology considers the neutronic models and the measured reactor data. The neutronic methods presented in this report will be used by ENC and utility customers for the design of reload fuel, for reactor in-core physics calculations and for safety and licensing calculations which include accident and transient analyses.

2.0 SUMMARY

Included in this section is a brief description of the computer codes used in the neutronic calculations for boiling water reactors. A summary of the core analysis methodology, verification of the neutronic and fuel management methods, and the method of determining the power distribution uncertainty is provided in the following.

The ENC neutronic methods include the five modules: (1) XFYRE for calculation of fuel neutronic parameters and assembly burnup, (2) XTGBWR for reactor core simulation, (3) COTRAN for transient calculations, (4) XDT for diffusion theory calculations and (5) XMC for Monte Carlo benchmark calculations.

2.1 FUEL ASSEMBLY DEPLETION MODEL (XFYRE)

The nuclear parameters for the BWR assemblies are calculated with the XFYRE computer code. The XFYRE code combines the HRG and THERMOS cross section generating codes, diffusion theory, and an isotopics depletion model to generate fuel neutronic parameters as a function of voids and exposure for both controlled and uncontrolled assemblies.

The calculations performed by the XFYRE code include generation of cross sections for each fuel assembly region, neutron flux and powe. shapes across the fuel assembly, isotopic depletion, flux and volume weighted bundle parameters, and incore detector parameters.

The code uses two dimensional four energy group diffusion theory methods for the microscopic depletion of BWR assemblies. The code alternates between a spatial calculation of the average flux in each pin cell and

a burnup calculation for each pin over an exposure interval, maintaining a constant pin power over the interval. The four energy group cross sections are collapsed from fine group thermal and epithermal spectrum calculations for each pin type within an assembly. The spectrum calculations are repeated at intervals to adjust the multigroup cross sections for the spectral change with burnup.

2.2 CORE SIMULATOR MODEL (XTGBWC)

The ENC core simulator program for the analysis of BWR cores is the XTGBWR code. The XTGBWR code requires two-group cross sections as input and utilizes simulated two-group diffusion theory models to solve for flux and power. The XTGBWR program uses coarse mesh diffusion theory to solve for the fast group flux in each node. The thermal flux is calculated from the fast flux assuming the only source of thermal neutrons is slowing from the fast group and that no thermal leakage occurs within each node. Corrections to the above assumption are made to account for thermal flux gradients at controlled nodes and on the core edge. Inner iterations are performed on the fast group flux, but the thermal flux is updated only after each outer iteration. After a specified number of outer iterations, the cross sections are updated to reflect power dependence on xenon, Doppler and thermal hydraulic feedback.

For fuel management calculations, the XTGBWR code has the following primary capabilities:

- 1. Core calculations in 1/4, 1/2 and full core geometry.
- 2. Control rod dependent parameters.
- 3. Thermal hydraulic void feedback including subcooled boiling.

4

- 4. Equilibrium and time dependent xenon and samarium.
- 5. Power dependent Doppler broadening.
- 6. Void history correction to cross sections.
- 7. Calculation of core K-effective and nodal power distribution.
- Calculation of critical power ratio (CPR), linear heat generation rate (LHGR), and average planar linear heat generation rate (APLHGR), at each node.
- 9. Prediction of the traveling incore probe (TIP) measurements.
- 10. Haling calculations in two or three dimensions.
- 11. Fuel shuffling option.
- 12. Zero power critical option.

2.3 REACTOR KINETICS MODEL (COTRAN)

The ENC program for kinetics analysis of BWR cores is the COTRAN code. COTRAN is a two dimensional (r-z) computer program which solves the space and time dependent neutron diffusion equation with fuel temperature and reactivity feedback. These reactivity feedbacks are determined from a solution of equations of mass, energy and momentum for the coolant coupled with a fuel conduction model. The COTRAN code requires input from the XFYRE and XTGBWR codes including cross sections, rod worths, initial flux and power shapes, peaking factors and other initial condition parameters. The COTRAN code has the capability to accept forcing functions as a function of time for several system parameters. These forcing functions allow COTRAN to model the reactor while including as input the total system feedback.

2.4 DIFFUSION THEORY MODEL (XDT)

The ENC computer program which is used for special single and multi-bundle diffusion theory calculations is the XDT code. These special diffusion theory calculations include four and sixteen bundle calculations, fuel misloading calculations and incore detector calculations. The XDT code calculates the eigenvalue, relative powers, multigroup neutron fluxes, and region cross sections.

2.5 MONTE CARLO MODEL (XMC)

The Exxon Monte Carlo Code (XMC) is a general purpose Monte Carlo code designed primarily to calculate benchmark problems for thermal reactors. These benchmarks are then used as one of the methods to calibrate the other ENC methods. With XMC, the geometrical configuration can be described exactly. This geometrical capability and a coupled space-energy solution of the transport problem makes the Monte Carlo method as contained in XMC superior to other calculational methods for evaluting key bundle nuclear parameters and for calculating the effects of water gaps, control blades and burnable poison rods.

The neutron flux, isotopic reaction rates, group-averaged cross sections, and neutron leakage are calculated in three-dimensional space over the energy range from 0 to 10 MeV. The reaction types considered are fission, capture, inelastic and n-2n scattering, elastic scattering with isotropic or anisotropic angular distributions, and thermal elastic scattering using the ideal gas scattering model. The energy distribution of the neutrons is continuous. However, the cross sections are averaged over up to 2000 microscopic energy groups. Resolved resonance cross sections are caluclated by XMC for each neutron energy using the Doppler-broadened Breit-W-gner single-level formula.

2.6 CORE ANALYSIS METHODOLOGY

Special neutronic calculations are performed to evaluate the control rod drop accident, fuel misloading incident, reactor core and channel hydrodynamic stability, control rod withdrawal incident, and to determine the neutronic input parameters for the plant transients and loss of coolant accidents.

Control Rod Drop

The control rod drop accident assumes that a control rod becomes uncoupled from the drive and remains stuck fully inserted in the reactor core as the control rod drive is withdrawn. The uncoupled control rod is then assumed to drop out of core. The control rod drop calculations are performed with COTRAN in two-dimensional (r-z) geometry with fuel temperature

and moderator density reactivity feedbacks. The reactor neutronic parameters which significantly affect the rod drop analysis include the Doppler reactivity coefficient, the maximum control rod worth, the power peaking (peaking with control rod removed from core) and the delayed neutron fraction.

Fuel Misloading

Two separate incidents are analyzed as part of the fuel misloading analysis. The first incident, which is termed the fuel misorientation error, assumes that a fuel assembly is misoriented, by rotation through 90^o or 180^o from the correct orientation, when loaded into the reactor core. The second incident, the fuel mislocation error, assumes a fuel assembly is placed in the wrong core location during refueling. For both the fuel misorientation error and the fuel mislocation error, the assumption is made that the error is not discovered during the core verification and the reactor is operated during the cycle with a fuel assembly misloaded. The fuel misorientation calculations are performed using the XFYRE, XDT, and XTGBWR codes. The fuel mislocation calculation is performed with the XTGBWR code. The limiting parameter of interest for the fuel misloading error is the MCPR in the misloaded fuel assembly. The fuel misloading analysis determines the difference between the MCPR for the correctly loaded core and the MCPR for the core with a fuel assembly misloaded.

Stability

Stability analysis is concerned with two basic phenomenon, reactor core (reactivity) stablity and channel hydrodynamic stability. Reactor core instability is when the reactivity feedback of the entire core drives the

reactor into power oscillations. Channel hydrodynamic instability is flow oscillations which may impede heat transfer to the moderator resulting in localized power oscillations. Stability is analytically demonstrated if no divergent oscillations develop as a result of perturbations of any critical variable such as core pressure, control rod position, and recirculation flow. The stability analysis is performed with the COTRAN computer code.

Neutronic Input to Plant Transient Analysis

The ENC plant transient and loss of coolant accident calculations require the following neutronic parameters as input:

- 1. Void reactivity coefficient,
- 2. Doppler reactivity coefficient,
- 3. Scram reactivity,
- 4. Delayed neutron fraction, and
- 5. Prompt neutron lifetime.

The above parameters are calculated with the XFYRE, XTG, and COTRAN codes.

Control Rod Withdrawal

The control rod withdrawal error is the widthdrawal of a control rod by the reactor operator from a fully inserted position until the control rod motion is stopped by the rod block. While the control rod is being withdrawn, the reactor power and the local power in the area of the rod which is being withdrawn will increase. The reactor thermal limit of concern as the power increases is the transient minimum critical power ratio (MCPR) limit which protects against critical heat flux. The control rod withdrawn calculation is performed with the XTGBWR code.

2.7 NEUTRONICS METHODS VERIFICATION

The ENC neutronics methods are principally verified by comparing calculated results to measured reactor data. The XFYRE calculated local power distribution and isotopics are compared to gamma scan measurements and destructive isotopic data. In addition the XFYRE code has been benchmarked against the higher order Monte Carlo code. The XTGBWR reactor simulator code is verified by calculating reactor K_{eff} data, measured TIP traces, and bundle gamma scan data. The kinetics calculations performed by the COTRAN code are compared to the measured Peach Bottom-2 data.

2.8 MEASURED POWER DISTRIBUTION UNCERTAINTY

The determination of the uncertainty associated with a measured power distribution is necessary from a reactor safety viewpoint. The safety analyses are performed to assure safe reactor operation with a certain quantified degree of confidence. The uncertainty associated with the reactor power distribution is defined in terms of the relative standard deviations of the independent variables involved in determining power distribution.

The reactor power distributions are combinations of measured reactor data and computer calculated data. The measured reactor data include the fixed local power range monitor (LPRM) in-core detector data and the traveling in-core probe (TIP) detector data. The computer calculated data include the relative core nodal power distribution, the in-core detector response distribution, and the local peaking factors for the fuel rods within each bundle. The predicted relative nodal power and detector response distributions are calculated with the XTGBWR reactor simulator code.

The relative standard deviations of the detector measurements, the calculated detector response distribution, the calculated nodal power distribution, and the local pin power distribution are determined by comparison to measured data. The measured data consist of distributions of TIP and fixed in-core detector responses plus gamma scans of bundles and pins.

3.0 NEUTRONICS MODELS FOR BWR REACTOR CORE CALCULATIONS

The ENC code package for performing reactor core neutronics calculations includes a fuel assembly depletion model (XFYRE), a core simulator model (XTGBWR), a reactor kinetics model (COTRAN), a multigroup diffusion theory model (XDT), and a Monte Carlo model (XMC). The XFYRE code calculates the basic fuel assembly neutronic parameters including the local rod power distribution, the local rod exposure distribution, the two and four group cross section sets and the fuel assembly reactivity. The parameters are calculated as a function of temperature, voids, exposure, power, and control.

The XTGBWR reactor simulator code models the reactor core in two dimensional (X-Y) or three dimensional (X-Y-Z) geometry. The reactor calculations can be performed in one-quarter, one-half, or full core geometry. The code calculates the reactor core reactivity, core flow distribution, nodal power distribution, reactor thermal limit values, and incore detector response.

The reactor kinetics calculations are performed with the COTRAN code. The COTRAN code models the time dependent core neutronics and thermal hydraulics in two dimensional (r-z) geometry with void and Doppler feedback. The code calculates the axial and radial temperature distribution for the fuel rods.

The special single fuel assembly and multi-fuel assembly diffusion theory calculations are performed with the XDT code. The XDT code is a two dimensional-multigroup model for reactor analysis. The code calculates the eigenvalue, relative powers, neutron fluxes, and flux and volume weighted neutronic parameters.

The XMC code is a general purpose Monte Carlo code used primarily to benchmark the bundle depletion code XFYRE. XMC utilizes an exact geometrical description and a coupled space-energy solution of the transport problem which makes XMC higher order than the other methods for evaluating key nuclear parameters.

These computer codes are described in Sections 3.1, 3.2, 3.3, 3.4, and 3.5, respectively.

3.1 FUEL ASSEMBLY DEPLETION MODEL (XFYRE)

The nuclear parameters for the BWR fuel assemblies are calculated with the XFYRE computer code. The XFYRE code combines the basic cross section generating codes, diffusion theory, and depletion models to generate fuel neutronic parameters as a function of voids and exposure for both controlled and uncontrolled assemblies.

The XFYRE code is automated to perform all calculations for the BWR fuel designs with a minimum of required input. A typical 8x8 BWR fuel design with two inert water rods and a control rod is shown in Figure 3.1-1. The XFYRE code can be used to analyze fuel rod arrays up to 11x11, with nonsymmetrical or symmetrical water gaps and with water or zirconium filled inert rods. The input to the XFYRE code consists of basic fuel rod and fuel assembly dimensions, fuel rod enrichments and material densities.

The calculations performed by the XFYRE code include generation of cross sections for each fuel assembly region, calculation of neutron flux and power shapes across the fuel assembly, depletion calculations, calculation of flux and volume weighted bundle parameters, and calculation of incore detector parameters. The methods used by the code to calculate each of the above are described in the following sections.

3.1.1 Basic Cross Section Library

The cross sections in the XFYRE code with the exception of the hydrogen scattering kernel are derived from the Battelle Northwest Master Library.⁽³⁻¹⁾ The ENDF/B⁽³⁻²⁾ scattering kernel (Haywood Kernel)⁽³⁻³⁾ for hydrogen in water was incorporated into the cross section library for the THERMOS⁽³⁻⁴⁾ program. The hydrogen kernel is generated using the FLANGE II code.⁽³⁻⁵⁾ The scattering kernels for the other nuclides are calculated by a Brown-St. John model, using free atom scattering cross sections.

The isotopes normally used in the XFYRE calculations are listed in Table 3.1-1.

3.1.2 Thermal Cross Sections

The thermal cross sections for the fuel rods are calculated with the Exxon revised THERHOS program. The THERMOS code calculates the scalar thermal neutron spectrum as a function of position in a lattice by solving numerically, the integral transport equation. The calculations are performed for 30 energy groups over the energy range $0 \le 0.683$ ev.

For a standard fuel rod with no burnable poison, the fuel rod and surrounding water are transformed into a cylindrical pin cell for calculation of the thermal cross section. The cylindrical cell has a fuel region, a clad region, and an outer water region as shown in Figure 3.1-2. The gap between the fuel pellet and the clad is homogenized with the fuel. The outer diameter of the water region in the cylindrical geometry is selected to give the true volume of the unit cell.

For the standard feul rod, the cell is divided into 15 cylindrical rings, with eight equal thickness rings of fuel, two rings of clad and five rings of water moderator. The THERMOS calculation uses the white boundary condition at the edge of the cell with an albedo of 1.0.

For fuel rods containing gadolinia, the cylindrical geometry is expanded to include an extra region of homogenized cells. The extra region is necessary to obtain the correct thermalization spectrum in the fuel rods containing gadolinia. The cylindrical geometry for the gadolinia pin cells is shown in Figure 3.1-3. [

The cylindrical pin cell and extra region are divided into [] rings for the THERMOS calculation. Since the thermal flux is strongly depressed in the gadolinia-fuel region, the gadolinium cross sections have a strong spatial dependence resulting in a non-uniform depletion of the gadolinium. []

The reflecting boundary condition is used in the THERMOS calculations for the fuel cells with extra regions.

For both the standard fuel rod and the gadolinia fuel rod, the cross sections are flux and volume weighted over the 30 fine energy groups and the fuel pin cell to obtain a one thermal energy group macroscopic cross section.

3.1.3 Epithermal Cross Sections

The epithermal cross sections for each region in the fuel assembly are calculated in XFYRE with the JRG program. The JRG program combines the HRG⁽³⁻⁶⁾ program and the DASQHE⁽³⁻⁷⁾ Dancoff calculation. The epithermal slowing down spectrum calculation is performed with 68 equal lethargy width five energy groups using the P₁ approximation. The calculation is performed over the energy range 10 MeV to 0.414 ev. The 68 fine group fluxes and currents are calculated by one sweep through the group structure, starting from the U-235 or Pu-239 source distribution. The multigroup model uses a full down-scattering matrix, with inelastic, n-2n, and P₀ and P₁ components of elastic scattering explicitly included.

For each fuel rod type in the BWR fuel assembly, the isotopic concentrations are homogenized over the unit cell consisting of the fuel, clad and water associated with each fuel rod. The macroscopic fine group parameters are constructed from the homogenized isotopic concentrations and the microscopic parameters on the HRG data tape.

A special calculation is made in the resonance range for U-235, U-238, Pu-239, Pu-240, and Pu-241 nuclides, using an adaptation of the

Alder, Hinman, and Nordheim(3-8) method to an intermediate resonance approximatation for both the absorber nuclide and an admixed moderator. The Dancoff correction factors that account for the effect of the adjacent fuel rods on the resonance absorption are calculated with the DASQHE program for a square lattice. The resonance contribution for each isotope is allocated to the fine groups in a consistent manner providing self-shielding in both space and energy.

3.1.4 Control Rod Cross Sections

Control rod cross sections are calculated for each fuel type as a function of exposure and void. The control rod cross sections are calculated after the THERMOS and JRG calculations are carried out for each fuel rod type in the fuel assembly. The calculation model includes the fuel assembly and the detailed control rod blade configuration including dimensions and number of poison pins per wing.

The blade is constituted of stainless steel for the support, stiffener (if present) and sheath, B_4C powder in stainless steel clad absorber pins, and the space between the absorber pins and the sheath can be either voided or unvoided water.

In the thermal energy range, a special one-dimensional slab geometry integral transport theory calculation is performed. The control rod blade and fuel assembly are converted into a one-dimensional slab preserving the relative areas of each component. The actual control rod geometry and the geometry for the THERMOS calculation of the control rod blade is shown in Figure 3.1-4.

XN-NF-80-19(NP)(A) Vol. 1 The thermal macroscopic cross sections of the control rod blade are obtained from the THERMOS calculations by editing over the blade region of the fuel assembly. The stainless steel microscopic cross sections for the control rod support are obtained from the blade sheath region.

Γ

The concentrations of the fuel isotopes, the clad, and the water in the pin cells are flux and volume weighted to obtain the homogenized concentrations for the fuel regions. The thermal flux for the flux weighting is obtained from the THERMOS pin cell calculations. []

In the epithermal energy range, the control rod cross sections are obtained from a special HRG calculation. The calculation is performed by homogenizing all regions of the fuel assembly including the control rod to obtain bundle average number densities. The HRG calculation is then performed for the fuel bundle to obtain the slowing down spectrum and the multigroup microscopic cross sections for each nuclide in the fuel assembly.

The epithermal macroscopic cross sections of the control blade are calculated from the boron, carbon, stainless steel, and water isotopic concentrations in the blade and the respective microscopic cross sections.

3.1.5 Neutron Flux and Power Calculation

Controlled and uncontrolled local pin powers, neutron flux, and bundle reactivity are calculated within XFYRE utilizing a four group diffusion theory calculation in two dimensional geometry. This portion of

the program is a modified version of the XDT code described in Section 3.4. These calculations are performed after the detailed energy and spatial calculations have been carried out for each fuel pin type in the assembly.

For the diffusion theory calculations the fuel assembly is transformed into an X-Y geometry as shown in Figure 3.1-5. [] The arrangement of pin cells is symmetric about the assembly diagonal which bisects the control blade slot. Additional regions representing the film water-channel mixture, gap water, incore detectors, inert rods, control blade, and control support complete the geometrical description of the fuel assembly. [

The broad group cross section parameters for the diffusion theory calculations are averaged over the following four broad energy groups:

Broad Energy Groups	Energy Range
1	11.7 kev - 10 Mev
2	2.38 ev - 11.7 kev
3	0.683 ev - 2.38 ev
4	0 ev683 ev

The four energy group diffusion equations can be written

as:

$$D_g \nabla^2 \phi_g - \Sigma_g^r \phi_g + S_g = 0, g = 1, \dots 4$$
 (3.1-1)

where

$$S_{g} = \frac{x_{g}}{K_{eff}} \sum_{g'=1}^{4} (v \Sigma_{f})_{g'} \phi_{g'} + \sum_{g'=1}^{g-1} \Sigma(g' \rightarrow g) \phi_{g'}, \qquad (3.1-2)$$

18

XN-NF-80-19(NP)(A) Vol. 1

and

g = the energy group

 χ_g = the fraction of neutrons generated in energy group g For mesh points which are situated in the center of the mesh interval as shown in the mesh description below

integration over the volume associated with each mesh point yields the difference equations in the following form:

$$\sum_{k=1}^{4} \frac{\overline{D}_{k} A_{k}}{R_{k}} (\phi_{k} - \phi_{0}) - \Sigma_{0}^{r} \phi_{0} V_{0} + S_{0} V_{0} = 0, \qquad (3.1-3)$$

where, for simplicity, the group indices have been omitted, and:

 Σ_{0}^{r} = removal cross section associated with mesh point o,

 S_0 = source rate associated with mesh point o,

 V_{o} = volume associated with mesh point o,

 ϕ_k = flux associated with mesh point k,

 2_k = distance between mesh point k and mesh point o,

 A_{ν} = area of boundary between mesh point k and mesh point o,

 \overline{D}_k = effective diffusion constant between mesh point k and mesh point o.

$$\overline{D}_{k} = \frac{D_{o} D_{k} (\delta R_{o} + \delta R_{k})}{D_{o} \delta R_{k} + D_{k} \delta R_{o}}$$
(3.1-4)

An iterative process is used to solve the difference equation (3.1-3). In XFYRE a successive line over-relaxation algorithm is used to accelerate convergence in the iteration that produces the group fluxes. After the spatial four energy group neutron fluxes are calculated, the power in each fuel pin is calculated from the fission rate and normalized to the average pin power.

The XTGBWR core simulator code requires as input the nodal average two energy group macroscopic cross sections for each fuel type in the reactor. These node average macroscopic cross section parameters are obtained by collapsing the four group cross sections in XFYRE to two energy groups.

3.1.6 Depletion Calculation

The burnup calculation is performed over exposure intervals which are specified by input. A typical set of exposure intervals for a BWR assembly containing gadolinia as a burnable poison is given in Table 3.1-2. The depletion is performed separately for each pin cell in the fuel assembly,

assuming diagonal symmetry of the cells in the assembly. The isotopes which are burned are listed in Table 3.1-3. The four isotopes FPA, FPB, FPC, and FPD are lumped pseudo-isotopes for U-235 in the Nephew⁽³⁻⁹⁾ fission product model. In all fuel pins the non-gadolinia isotopes are burned using an average isotopic concentration for the fuel area. []

The isotopic transmutation calculations performed in the XFYRE program follow the process depicted in Figure 3.1-6. The set of differential equations that govern the transmutation of the subchain, i.e. U-238, Pu-239, Pu-240, Pu-241, Pu-242, can be written as follows using standard notation:

$$\frac{dN^8}{dt} = -\left\langle \sigma_a^8 \right\rangle \cdot \left\langle \phi \right\rangle \cdot N^8$$
(3.1-5)

$$\frac{dN^{9}}{dt} = -\left\langle\sigma_{a}^{9}\right\rangle \cdot \left\langle\phi\right\rangle \cdot N^{9} - \lambda^{9} \cdot N^{9} + \left\langle\sigma_{c}^{8}\right\rangle \cdot \left\langle\phi\right\rangle \cdot N^{8} \qquad (3.1-6)$$

$$\frac{dN^{0}}{dt} = -\left\langle \sigma_{a}^{0} \right\rangle \cdot \left\langle \phi \right\rangle \cdot N^{0} - \lambda^{0} \cdot N^{0} + \left\langle \sigma_{c}^{9} \right\rangle \cdot \left\langle \phi \right\rangle \cdot N^{9} \qquad (3.1-7)$$

$$\frac{dN^{1}}{dt} = -\left\langle \sigma_{a}^{1} \right\rangle \cdot \left\langle \phi \right\rangle \cdot N^{1} - \lambda^{1} \cdot N^{1} + \left\langle \sigma_{c}^{0} \right\rangle \cdot \left\langle \phi \right\rangle \cdot N^{0} \qquad (3.1-8)$$

$$\frac{dN^{2}}{dt} = -\left\langle \sigma_{a}^{2} \right\rangle \cdot \left\langle \phi \right\rangle \cdot N^{2} + \left\langle \sigma_{c}^{1} \right\rangle \cdot \left\langle \phi \right\rangle \cdot N^{1}.$$
(3.1-9)

The depletion calculations are performed with a one group flux which is obtained by collapsing the four group fluxes from the diffusion theory calculation. The XFYRE code takes advantage of the fact that for constant fluxes and cross sections, the solution to the depletion equations can be expressed analytically.

The assumption of constant fluxes and cross sections is a reasonable assumption if the exposure steps are small. []

When the cross sections are regenerated, the average concentration of each isotope for a fuel type is calculated, and then the THERMOS-JRG fine group calculations are performed to obtain new microscopic cross sections for each isotope. The number of fuel types in the fuel assembly is specified as input. Usually all fuel rods with the same uranium enrichment, gadolinia concentration, dimensions, and fuel density are considered as one rod type in the calculation.

[] A typical plot of the gadolinium-157 thermal absorption cross section for each region as a function of exposure is shown in Figure 3.1-7. A typical plot of the gadolium-157 concentration as a function of exposure is shown in Figure 3.1-8.

Since the gadolinium concentrations and cross sections are changing more rapidly than the other isotopes, over each burnup subinterval the gadolinium cross sections []

3.1.7 Xenon and Samarium

For depletion calculations the XFYRE code includes time and power dependent xenon and samarium. The time is calculated by the code from the power, exposure, and fuel weight. At zero exposure there is no xenon or samarium in the calculations.

For restart calculations the code can calculate time dependent xenon and samarium based on the isotopic concentrations from the restart

tape and an input shutdown time. The code has the options of calculating (1) no xenon and no samarium, or (2) no xenon and the samrium from the restart tape in addition to the time dependent option.

3.1.8 Restart Calculations

During an XFYRE depletion calculation, a restart tape can be written that saves sufficient information to perform additional calculations without repeating the burnup calculation. Such additional calculations can include solutions at different temperatures, void conditions, or control conditions.

When a restart and burn calculation is performed where a parameter is changed or when the gadolinia is not depleted, a small burn step of 250 MWD/MT with cross section regeneration is desirable prior to resuming calculations with larger burn steps. The small step is necessary for the accurate extrapolation of cross section during a larger burn interval.

3.1.9 Incore Detector Parameters

The XFYRE code uses a dilute macroscopic thermal fission cross section at the location of the incore detector to calculate the T factor. The T factor is defined as follows:

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If desired, the homogenized four energy group macroscopic cross sections of the incore detector can be input. The code then uses the input cross sections in the diffusion theory calculations and in calculating

Table 3.1-1 Isotopes Normally Used in XFYRE

Isotopes	Isotopes
C12	Am241
016	Am243
H20 W/UPSCAT	Cm242
ZIRCONIUM	Cm243
30455	Cm244
BORON	PFP4-235
U238	PFP1-235
Pu240	PFP2-235
Pu242	PFP3-235
U235	Sm151
Pu239	Gd154
Pu241	Gd155
Xe135	Gd156
Sm149	Gd157
U236	Gd158
Np237	Pu238

Table 3.1-2 Typical Exposures (MWD/MTU) for XFYRE Depletion Calculations



Table 3.1-3 Burnable Isotopes in XFYRE

U-235	Pu-238	Am-241	Xe-135	Gd-154*
U-236	Pu-239	Am-243	Sm-149	Gd-155
U-238	Fu-240	Cm-242	Sm-151	Gd-156
Np-237	Pu-241	Cm-243	PFP4-235	Gd-157
	Pu-242	Cm-244	PFP1-235	Gd-158
			PFP2-235	
			PFP3-235	

* The gadolinia isotopes are burned only in Gd-poisoned pins.

XN-NF-&0-19(NP) (A) Vol. 1



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Inert Water Rods

Figure 3.1-1 Typical 8x8 BWR Fuel Assembly with a Wide and Narrow Water Gap



Figure 3.1-2 Cylindrical Geometry for THERMOS Calculation of Non-Gadolinia Fuel Rod


Figure 3.1-3 Cylindrical Geometry for THERMOS Calculation of Fuel Rod Containing Gadolinia



Actual Geometry and Slab Geometry for THERMOS Control Rod Cross Section Calculation Figure 3.1-4

XN-NF-80-19(NP)(A) Vol. 1



Figure 3.1-5 XFYRE Mesh Boundaries for an 8x8 BWR Fuel Assembly with a Wide and Narrow Water Gap



Figure 3.1-6 XFYRE Isotopic Chain



Figure 3.1-7 Relative Thermal Absorption Cross Section of Gadolinium-157 as a Function of Fuel Assembly Exposure



the T factor. The incore detector region for the diffusion theory calculation is defined by the width of the water gaps on the side of the bundle of the control rods. The incore detector location is shown in Figure 3.1-1. The incore detector region in X-Y geometry for the diffusion calculation is shown in the upper right hand corner of Figure 3.1-5.

3.2 CORE SIMULATOR MODEL (XTGBWR)

The Exxon Nuclear core simulator program for the analysis of BWR reactor cores is the XTGBWR code. The XTGBWR code uses the same modified two group diffusion theory as the Exxon Nuclear reactor simulator code for pressurized water reactors (XTGPWR).⁽³⁻¹⁰⁾ The XTGBWR program uses coarse mesh diffusion theory to solve for the fast flux assuming the only source of thermal neutrons is slowing down from the fast group and no thermal leakage occurs within each node. Corrections to this assumption are made to account for thermal flux gradients at controlled nodes and on the core edge. Inner iterations are performed on the fast group flux, but the thermal flux is updated only after each outer itera ion. After a specified number of outer iterations, the cross sections are updated to reflect power dependence on xenon, Doppler and thermal-hydraulic feedback. This method of solution results in rapid covergence.

For fuel management calculations, the XTGBWR code has the following capabilities:

- Core calculations in 1/4, 1/2 and full core geometry with several boundary conditions.
- 2. Control rod dependent parameters.

- Thermal hydraulic model including void feedback, subcooled boiling, and pressure drop flow calculation.
- 4. Equilibrium and time dependent xenon and samarium.
- 5. Power dependent Doppler broadening.
- 6. Void history correction to cross sections.
- 7. Calculation of core K-effective and nodal power distribution.
- Calculation of critical power ratio (CPR), linear heat generation rate (LHGR), and average planar linear heat generation rate (APLHGR) at each node.
- Core exposure calculated from inputs of burnup (MWD/MT), energy (GWD), or time (hours).
- Full edit capability in either two or three dimensions for all arrays.
- 11. Prediction of the traveling incore probe (TIP) measurements.
- 12. Tape or file outputs for restart capability.
- 13. Haling calculations in two or three dimensions.
- 14. Fuel shuffling option.
- 15. Zero power critical option.
- 3.2.1 Core Geometry

X-Y-Z geometry is used in the three dimensional model of XTGBWR. When two dimensional geometry is used the axial nodes are averaged into one plane in the Z direction. A typical X-Y geometry full core configuration is shown in Figure 3.2-1. The code requires the node in the radial direction (X-Y) to be the same size for all fuel assemblies. All nodes in the axial direction must also be the same size but may be a different size than in the radial direction. [] In addition to the full core geometry, the code can be used for quarter core and half core geometry with reflective, repetitive, or 90⁰ rotational boundaries. The boundary conditions for quarter core symmetry are shown in Figure 3.2-2.

For the analysis of cores with axially distributed gadolinia or enrichment, each fuel type may be made up of two or more material types. XTGBWR is capable of handling a different fuel type for each axial plane of the reactor for each bundle location.

For core analysis in two dimensions, the core is modeled in X-Y geometry and the neutron leakage in the axial direction is calculated using either an input axial buckling or the internally calculated geometric axial buckling. The reactivity effects of the axial buckling are treated through adjustment of the absorption cross sections.

3.2.2 Diffusion Theory Model

The XTGBWR program uses a modified coarse mesh two energy group diffusion theory model for steady state analysis of the reactor core. The model is designed to accept void and exposure dependent two group cross sections. The cross sections can be specified on a nodal basis allowing axial and radial effects to be modeled. The conditions under which the cross sections were generated are input, and the XTGBWR code utilizes this information to adjust the cross sections to fit the actual reactor conditions

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in each node. This includes adjustment for control rods, instantaneous void, void history, power dependent Doppler, and time and power dependent xenon and samarium.

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Using standard notation, the basic diffusion theory equa-

$$-D_{g}\nabla^{2}\phi_{g} + \Sigma_{g}^{R}\phi_{g} = \frac{\chi_{g}}{k_{eff}}\sum_{g'=1}^{G} (\nabla\Sigma_{f})_{g}\phi_{g'} + \sum_{g'=1}^{g-1} \Sigma_{s1}(g' \rightarrow g)\phi_{g'} \qquad (3.2-1)$$

Assuming all neutrons are born in the fast group, the two group diffusion equations are

$$-D_{1}\nabla^{2}\phi_{1} + \Sigma_{R_{1}}\phi_{1} = \frac{1}{k_{eff}} (v_{1}\Sigma_{f_{1}}\phi_{1} + v_{2}\Sigma_{f_{2}}\phi_{2})$$
(3.2-2)

$$-D_{2}\nabla^{2}\phi_{2} + \Sigma_{a_{2}}\phi_{2} = \Sigma_{s1} (1 \rightarrow 2) \phi_{1}$$
(3.2-3)

These equations are integrated over the volume of a threedimensional node. To evaluate the leakage term, the volume integral over the Laplacian is changed to a surface integral using Green's theorem

$$\int D\nabla^2 \phi \, dV = \int D \vec{\nabla} \phi \cdot d\vec{A} \tag{3.2-4}$$

Using mesh points at the node centers, the volume integration of equation (3.2-2) yields

$$\sum_{k=1}^{6} \frac{\overline{D}_{k}A_{k}}{dk} (\phi_{k} - \phi_{0}) - \Sigma_{R_{0}}\overline{\phi}_{0}V_{0} = -S_{0}V_{0}\overline{\phi}_{0}$$
(3.2.5)

where 0 refers to the node being calculated and k to the six nearest neighbors shown in Figure 3.2-3. For convevience, the subscript 1 referring to the gast group has been omitted. The notation is as follows:

 Σ_{R_0} = removal cross section = $\Sigma_{s1}(1 \rightarrow 2) + \Sigma_{a_1}$

 $S_{0} = \frac{1}{k_{eff}} \left(v\Sigma_{f_{1}} + v\Sigma_{f_{2}}\overline{\phi}_{2}/\overline{\phi}_{1} \right)$

 $V_0 = volume of node$

 d_{ν} = distance between mesh point k and mesh point 0

 A_{ν} = area of b ary mesh point k and mesh point 0

Dk = effective diffusion coefficient between mesh point k and mesh point 0

$$\overline{D}_{k} = \frac{D_{0}D_{k}(\delta R_{0} + \delta R_{k})}{D_{0}\delta R_{k} + D_{k}\delta R_{0}}$$
(3.2-6)

 $\delta R_0, \delta Rk$ = node size in direction of calculation

If h_x , is the mesh spacing in both the X and Y directions and h_z is the mesh spacing in the z direction, the

$$\frac{A_k}{d_k} = h_z \text{ in } x, y \text{ direction}$$
(3.2-7)

 $\frac{A_k}{d_k} = \frac{h_x^2}{h_z} \text{ in z direction}$ (3.2-8)

$$\overline{D}_{k} = \frac{2D_{0}D_{k}}{D_{0} + D_{k}}$$
(3.2-9)

$$V_0 = h_x^2 h_z$$
 (3.2-10)

Equation (3.2-5) becomes

$$\sum_{k} \frac{2D_{k}D_{0}R_{k}}{D_{k} + D_{0}} (\phi_{k} - \phi_{0}) = -S_{0}h_{x}^{2} \overline{\phi}_{0} + \Sigma_{R_{0}}h_{x}^{2} \overline{\phi}_{0}$$
(3.2-11)

where

$$R_{k} = 1 \text{ if } k \text{ in } x, y \text{ direction}$$

$$= \frac{h_{x}^{2}}{h_{z}^{2}} \text{ if } k \text{ in } z \text{ direction}$$
(3.2-12)

with minimal error, $\overline{\mathtt{D}}_k$ can be approximated by

$$\frac{2D_k D_0}{D_k + D_0} = \sqrt{D_k} \sqrt{D_0}$$
(3.2-13)

with the additional definitions

$$\psi_j = \phi_j \sqrt{D_j} \quad j=0,k \quad (3.2-14)$$

$$\overline{\psi}_0 = \overline{\phi}_0 \sqrt{D_0} \tag{3.2-15}$$

$$P_0 = \sum_{k} R_k \sqrt{D_k} / \sqrt{D_0} + \varepsilon_{R_0} h_x^2 / D_0 \qquad (3.2-16)$$

and with some algebraic manipulation, Equation (3.2-11) can be written as

$$P_{0\psi_{0}} + \sum_{k} R_{k}\psi_{k} = \frac{S_{0}}{D_{0}}h_{x}^{2}\overline{\psi_{0}} - \frac{\Sigma_{R_{0}}}{D_{0}}h_{x}^{2}(\overline{\psi_{0}} - \psi_{0})$$
(3.2-17)

 $\overline{\phi}_0$ is a weighted average flux for node 0 calculated from the mid-point fluxes of node 0 plus the six surrounding k nodes. Specifically $\overline{\phi}_0$ is calculated from

$$\overline{\phi}_0 = b\phi_0 + 2c \sum_{k}^{\sum} R_k \phi_k^0$$
 (3.2-18)

where

$$b = \frac{3 * AFA}{3 * AFA + (1 - AFA)(R + 2)}$$
(3.2-19)

$$c = \frac{1 - AFA}{4 * (3 * AFA + (1 - AFA)(R + 2))}$$
(3.2-20)

 ϕ_k^0 = the flux on the interface between node 0 and node k and is derived using continuity of current.

$$\phi_{k}^{0} = \frac{\psi_{0}}{2\sqrt{D_{k}}} + \frac{\psi_{k}}{2\sqrt{D_{0}}}$$
(3.2-21)

AFA = the weighting factor for the mid-point fast flux.

Using equations (3.2-18) and (3.2-21)

$$\overline{\psi}_{0} = (b + c * r_{0})\psi_{0} + c \sum_{k} R_{k}\psi_{k}$$
(3.2-22)

where

$$r_0 = \sqrt{D_0} \sum_{k} R_k / \sqrt{D_k}$$
 (3.2-23)

The numerical solution is obtained by rewriting equation (3.2-17)

$$\psi_0 = \left(\sum_{k} R_k \psi_k + \frac{\overline{S}_0}{D_0} h_{x^2} \overline{\psi}_0\right) / P_0$$
(3.2-24)

where

$$\psi_0$$
 = average nodal ψ from previous iteration (3.2-25)

$$\overline{S}_{0} = \frac{1}{k_{eff}} \left(v \Sigma_{f_{1}} + v \Sigma_{f_{2}} \overline{\phi_{2}} / \overline{\phi_{1}} \right) - \Sigma_{a_{1}} - \Sigma_{s1} (1 + 2)$$
(3.2-26)

Equation (3.2-24) is used to iterate on the fast flux solution.

The thermal flux is calculated by assuming no thermal leakage among nodes. Equation (3.2-3) reduces to

$$\overline{\phi}_2 = \overline{\phi}_1 \cdot \frac{\Sigma_{s1}(1+2)}{\Sigma_{a2}}$$
(3.2-27)

where $\overline{\phi}_1$ is calculated from Equation (3.2-18). $\overline{\phi}_2$ may then be calculated using the form of Equation (3.2-18) except that all parameters refer to the thermal group. An empirical correction factor is applied to the model for controlled nodes and is used to improve the prediction of the nodal powers.

A new eigenvalue (k_{eff}) is calculated after each outer iteration. This eigenvalue and updated values of $\overline{\phi_1}$ and $\overline{\phi_2}$ are used to compute a source term and the inner iterations are repeated. After each ten or fewer outer iterations, the cross sections are updated to account for the power distribution effects of thermal hydraulic feedback, Doppler broadening and xenon. These changes to the cross sections are described under the respective headings in the following sections of this report. This procedure of inner and outer iterations and cross section updating continues until

convergence or the specified maximum iterations are reached, whichever is sooner. The power distribution in each node is calculated by:

 $P = (\kappa \Sigma_{f_1} \overline{\phi}_1 + \kappa \Sigma_{f_2} \overline{\phi}_2)$ (3.2-29)

3.2.3 Boundary Conditions

3.2.3.1 Outer Boundary

The outer boundary conditions determine the leakage from the core. XTGBWR utilizes an extrapolation distance at which the fast flux goes to zero to determine the flux profile and the leakage of fast neutrons from the nodes on the core-reflector interface. The extrapolation distance is calculated separately for each boundary node and is based or input "reflector" cross section data which represents the neutron diffusion (material) properties of reflector nodes found at the top, bottom, and periphery of the core boundary. []

3.2.3.2 Reflected Boundary

The zero current boundary condition is achieved by simply setting $\phi_k = \phi_0$ in Equation (3.2-5) for a reflected node.

3.2.3.3 Periodic and Other Boundary Conditions

Periodic and other boundary conditions are achieved by setting the flux node value for node k in Equation (3.2-5) to the correct value when a node is a boundary node.

3.2.4 Exposure and Void Dependent Cross Sections

The XTGBWR code requires two group cross section sets as a function of exposure and voids to describe each material type. Cross sections

are input for zero void, near core average void, and near maximum void for each fuel type. A typical hot operating cross section set is input at voids of [] for each of the following exposures (GWD/MT): [] The cross sections are obtained for the actual void and exposure conditions of a node by linearly interpolating between or beyond the input values.

The base cross sections are calculated by burning the fuel at a given void fraction from zero exposure to the maximum exposure that any node of this fuel type is expected to achieve throughout the life of the fuel. At each calculational exposure increment the instantaneous void and the average void history is the same. The void history is defined by the following [

3.2.5 Control Rod Effects

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If a core model is set up such that a fuel node is partially controlled, the fully controlled cross sections are homogenized with the uncontrolled cross sections using the fraction of node height controlled as a mixing factor.

3.2.6 Equilibrium and Time Dependent Xenon

Xenon is calculated within XTGBWR by equations which are solutions to the differential equations for iodine and xenon. This allows the calculation of both time and equilibrium power dependent xenon. The

44

XN-NF-80-19(NP)(A)

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Vol. 1

exposure and void dependent cross section data from the XFYRE code at exposures greater than zero include constant power equilibrium xenon absorption cross sections. The XTGBWR code calculates a base macroscopic xenon free absorption cross section from the input data by subtracting the constant power xenon absorption cross section. The xenon concentration used to calculate the magnitude of the xenon absorption cross section that is subtracted is obtained from the input constant power cross section parameters and the power that was assumed for the XFYRE calculations. The time and power dependent xenon concentration for the actual reactor operating conditions is calculated for each node. The xenon adjustment to the base macroscopic absorption cross section is calculated by multiplying the xenon concentration by a xenon miscroscopic absorption cross section which has been adjusted for actual void and void history effects via Equation 3.2-38. This delta cross section is then added to the macroscopic base thermal absorption cross section.

The differential equations which are used for the formation and decay of the iodine and xenon isotope are:

$\frac{dI}{dt}$	$= -\lambda_{I}I + \gamma_{I} \phi \Sigma_{f}$	(3.2-39)
$\frac{dX}{dt}$	= $-\lambda_{\chi}X - AX + \gamma_{\chi} \phi \Sigma_{f} + \lambda_{I}I$	(3.2-40)

The solution of the differential equations gives the following equations for the time dependent iodine I(t) and xenon X(t)

$$I(t) = I_0 \exp(-\lambda_I \Delta T) + \frac{\gamma_I}{\lambda_I} \left[1 - \exp(-\lambda T) \right] \phi \Sigma_f \qquad (3.2-41)$$

$$X(t) = \frac{(\gamma_{I} + \gamma_{\chi}) \phi \Sigma_{f}}{\lambda_{\chi} + A} + \left[X_{0} - \frac{(\gamma_{I} + \gamma_{\chi}) \phi \Sigma_{f}}{\lambda_{\chi} + A} \right]$$
(3.2-42)

$$-\frac{\gamma_{I}I_{0} - \gamma_{I} \phi \Sigma_{f}}{\lambda_{\chi} + A - \lambda_{I}} \right] * \exp(-(\lambda_{\chi} + A) \Delta T$$

+
$$\frac{\lambda_{I}I_{0} - \gamma_{I} \phi \Sigma_{f}}{\lambda_{\chi} + A - \lambda_{I}} * \exp(-\lambda_{I} \Delta T)$$

where

 λ_{I} = decay constant for iodine

 λ_{χ} = decay constant for xenon

Y_I = fission yield of iodine

 γ_{χ} = fission yield of xenon

△T = time since last time step

In = iodine number density at the last time step

 X_0 = xenon number density at the last time step For equilibrium power conditions, the exponential terms in the above equation are zero and the xenon concentration (X) is given by the following formula:

$$\zeta = \frac{(\gamma_{I} + \gamma_{\chi}) \phi \Sigma_{f}}{\lambda_{\chi} + A}$$
(3.2-43)

where the terms in the equation are given above.

The thermal absorption cross section for each node of fuel is then adjusted to account for xenon thermal absorption by

$$\Sigma_{a_2} = \Sigma_{a_2} (base) + X(t) * \sigma_{a_2}^X$$
 (3.2-44)

where

 Σ_{a_2} (base) = nodal cross section with constant power xenon subtracted.

(3.2-45)

3.2.7 Samarium Buildup

At exposures greater than zero, the base cross sections generated by the XFYRE code contain equilibrium samarium. For hot operating conditions where the reactor has operated for greater than 10 days, the samarium is at equilibrium and no correction to the absorption cross section for non-equilibrium samarium is necessary.

For startup conditions after shutdown the samarium buildup option in the XTGBWR program can be used to calculate time dependent samarium concentration. After the reactor has been shutdown for about 15 days, the promethium has decayed to samarium and the samarium concentration will be

 $Sm (shutdown) = Sm_0 + Pm_0$

where Sm_0 and Pm_0 are the equilibrium concentration of samarium and promethium. The equilibrium concentration of an isotope is

	$\gamma_i(\phi_1\Sigma_{f_2} + \phi_2\Sigma_{f_2})$		
. 1	=		(3.2-46)
1	^i		

where

 N_i = concentration of isotope i γ_i = yield of isotope i λ_i = decay constant of isotope i ϕ_1 = fast flux ϕ_2 = thermal flux

 Σ_{f_1} = fast fission cross section Σ_{f_2} = thermal fission cross section

After startup the excess Sm is burned out exponentially until equilibrium Sm is once again established. The equation for the time dependent added absorption cross section due to samarium depletion is

$$\Delta \Sigma_{a_2} = Pm_0 * \sigma_{a_2}^{Sm} * \exp(-\sigma_{a_2}^{Sm} \phi_2 \Delta T)$$
(3.2-47)

where

 $\sigma_{a_2}^{Sm}$ = thermal samarium absorption equation

 ϕ_2 = thermal flux

 ΔT = time since startup.

3.2.8 Doppler Broadened Cross Sections

The base cross sections are calculated with the XFYRE code at a constant power and fuel temperature. For a given node of fuel in the core the fuel temperature depends on the power, exposure, void fraction, and fuel rod design. Since the Doppler broadening of the uranium and plutonium resonance absorption peaks is dependent on the fuel temperature, the XTGBWR code accounts for the Doppler effects by adjustment of the fast absorption cross section of each node when the cross sections are calculated. The adjustment is made by the following equation:

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3.2.9 Coolant Flow Distribution

The coolant flow distribution is calculated from a hydraulic model of the reactor core described in Reference 3-11. In this model, the core consists of a number of parallel flow paths between upper and lower plenums, with an equal pressure drops across all paths. Each assembly constitutes a separate path and the bypass flow region shared by all the assemblies is a flow path. For each flow path, the pressure drop is calculated by a channel flow model which includes frictional acceleration, and gravitational terms. The effects of orificing, lower and upper tie plates, grip spacers, and other frictional losses are modeled by flow dependent loss coefficients. The effects of power on the flow distribution are included by a void fraction model, described below, and by a two-phase friction multiplier. The coolant flow through each parallel path is adjusted iteratively until the pressure drops for all parallel flow paths are equal within a specified limit.

All assemblies with the same number of rods and the same set of loss coefficients comprise a hydraulic type. The results of the hydraulic model calculations can be used to obtain an empirical relationship between assembly flow and assembly power for each type which describes the flow versus power for the assembly to good accuracy. [] The flow distribution is redetermined and renormalized at each cross section update.

3.2.10 Steam Quality and Void Fraction

The coolant enthalpy (h_{ijk}) at each axial node k of a fuel assembly is calculated from the flow and power by integrating the heat deposited in the water up to the node midpoint using the following formula:

The void fraction at each node is calculated from the coolant enthalpy. The void fraction correlation used in the XTGBWR program is based upon a mechanistic description of two-phase separated flow and incorporates the effects of integral and relative phase slip and is a function of the pressure, mass velocity, flow quality and rod surface heat flux within an assembly. A subcooled void model is included in the void fraction correlation to include the effects of thermal nonequilibrium. The void fraction model is described fully in Reference 3-11.

3.2.11 Thermal Limits Calculation

As an edit option, the XTGBWR code calculates the average planar linear heat generation rate (APLHGR), the linear heat generation rate (LHGR), and the critical power ratio (CPR) for each node of fuel in the core. The APLHGR is calculated from the relative nodal power P_{ijk} and the total reactor thermal power PTH converted to kw/ft as follows:

$$APLHGR_{ijk} = \frac{PTH * P_{ijk} * FPGIF * 12,000}{ND2D * NRODAS_{ij} * HEIGHT}$$
(3.2-52)

where

PTH = core thermal power in MWth
P_{ijk} = relative nodal power
FPGIF == fraction of power deposited within fuel rods
ND2D = fuel assemblies in core
NRODAS_{ij} = fuel rods per assembly at core locations
HEIGHT = core height in inches

The LHGR is the maximum rod power in a node of fuel and is calculated from the APLHGR and the relative local peaking factor (P_1) .

LHGR = APLHGR*P,

(3.2-53)

The local peaking factor is calculated by the XFYRE code and input into the XTGBWR program as a function of exposure, voids, fuel type, and control.

The CPR is calculated in the XTGBWR code for each fuel assembly using the Exxon Nuclear $XN-3^{(3-12)}$ critical power correlation. [].

3.2.12 Incore Detector Response Calculation

The XTGBWR code has the edit capability to calculate the relative incore detector response. The incore detector assembly consists of an outer stainless steel sheath, a stainless steel tube for the traveling incore probe (TIP), four fixed position local power rate monitors (LPRM's) and the signal cables to the LPRM's. A cross section of the BWR incore detector is shown in Figure 3.2-4.

The inc re detector is physically located outside the channel in the water gap corner opposite the control rod. The location of an incore detector relative to the assembly is shown in Figure 3.1-1.

In the reactor core the incore detectors are placed in approximately one out of every four possible locations such that if the core is operated with quarter core mirror symmetry, all fuel assemblies excluding those on the core periphery are monitored by a traveling incore detector. A typical placement of the incore detectors in the core is shown on Figure 3.2-1.

The TIP and LPRM incore detectors are both miniature fission detectors usually containing uranium-235. The signal ouput from the detectors is proportional to the thermal neutron flux.

In the XTGBWR program, the detector response is calculated from the nodal power of each of the four fuel assemblies surrounding the detector. The relative detector response (TIP) at a given axial location is given by the following equation.

Tables of basic T factors are input as a function of exposure and void history as part of each cross section set. These basic factors are corrected for each node to account for the effects of 1) difference between instantaneous voids and average void history, 2) axial variation of instantaneous void, and 3) presence of control rod. The correction factors are reactor

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dependent, but not fuel type dependent. Specifically, for BWR's with asymmetric water gaps, the T factor for an uncontrolled node is [

For BWR's with symmetric water gaps, the T factor for an uncontrolled node is [

For either water gap, the T factor for a controlled node is given by [

The axial TIP correction factor (DC_N) is the ratio of the relative thermal fission density in the actual incore detector to the fission density in a dilute U-235 and water mixture at a specific axial core height. This factor is function of the axial position and the in-channel void fraction,

3.2.13 Zero Power Critical Option

The XTGBWR code will perform zero power flux and eigenvalue solutions with no flow, void, or Doppler feedback. The nuclear parameters for the zero power solutions are calculated with the XFYRE code. Starting with void history and exposure dependent isotopics, the cross sections are calculated for each fuel type in the core at the desired fuel and moderator temperature. The calculations are performed both controlled and uncontrolled.

The zero power nuclear data are input into the XTGBWR code as the ratio of the zero power cross sections divided by the hot operating cross sections. The cross section ratio data are normally input into the



- + Control Rods
- TIP Instrument Assemblies
- X Common Position for all TIP Machines
 - Fuel Assemblies with Large Diameter Orifice
 - Fuel Assemblies with Small Diameter Orifice

Figure 3.2-1 Typical BWR Reactor Core Geometry

Reflective

+ C/L

+-C/L

	1	2	3	4
1	1	2	3	4
5	5	6	7	8
9	9	10	11	12
13	13	14	15	16
С	† /L			

Repetitive

	5	6	7	8
2	1	2	3	4
6	5	6	7	8
10	9	10	11	12
14	13	14	15	16

5

2

10

9

3

7

11

13

4

12

16

C/L

1

1

9

- 1

2

90° Rotational

+C/L

4	13	14	15
	t		
C,	/L		

Figure 3.2-2 XTGBWR Quarter Core Symmetry Boundary Conditions







code at exposures of []. The zero power cross sections for controlled fuel nodes are input as the ratio of controlled cross sections to uncontrolled cross sections. The controlled cross section data are input []. Prior to performing a zero power calculation, the XTGBWR code determines the cross sections for each node of fuel in the core from the input data by interpolating on exposure and void history and considering control, xenon, and samarium. Since there is no thermal hydraulic or power feedback, the cross sections for a given node of fuel do not vary during the flux and eigenvalue calculation.

3.3 REACTOR KINETICS MOLTL (COTRAN)

The ENC reactor kinetics model for the analysis of BWR reactor cores is the COTRAN code. COTRAN is a two dimensional (r-z) computer program which solves the space and time dependent one energy group neutron diffusion equation with one prompt and six delayed neutron groups. Fuel temperature and void reactivity feedback are determined from a solution of the equations of mass, energy and momentum for the coolant coupled with a fuel heat conduction model. The coolant model is a one-dimensional solution of the hydrodynamic equations assuming thermodynamic equilibrium between phases in the two-phase region. The fuel rod model is a two-dimensional solution of the heat conduction equation by the method of weighted residuals in the radial direction with finite differences used for time and axial space derivatives. Axial conduction and temperature dependent fuel thermal conductivity are included.

The COTRAN code integrates the two separate codes XTRAN⁽³⁻¹³⁾ and COBRA IV.^(3-14, 3-15) XTRAN supplies the neutronic solution and COBRA IV, reduced to one dimension by the elimination of cross flow, determines the thermal-hydraulic feedback.

The COTRAN code requires input from the XFYRE and XTGBWR codes including cross sections, rod worths, initial flux and power shapes, peaking factors and other initial condition parameters. The COTRAN code has the capability to accept forcing functions as a function of time for several system parameters. These forcing functions allow COTRAN to model the reactor while including input total system feedback.

3.3.1 COTRAN Neutronics Model

3.3.1.1 Space & Time Dependent Iterative Equation

The one neutron group, space and time dependent neutron diffusion equation with no external sources is:

$$\frac{1}{v(\vec{r},t)} \frac{d\phi(\vec{r},t)}{dt} = \left[(1-\beta)v\Sigma_{f}(\vec{r},t) - \Sigma_{A}(\vec{r},t) \right] \phi(\vec{r},t) + \nabla \cdot D(\vec{r},t)\nabla \phi(\vec{r},t) + \sum_{\varrho} \lambda_{\varrho} C_{\varrho}(\vec{r},t)$$

$$(3.3-1)$$

Applying Ficks Law,

$$J(\vec{r},t) = -D(\vec{r},t)\nabla\phi(\vec{r},t)$$

Equation (3.3-1) becomes

$$\frac{1}{v(\vec{r},t)} \frac{d\phi(\vec{r},t)}{dt} = \left[(1-\beta)v\Sigma_{f}(\vec{r},t) - \Sigma_{A}(\vec{r},t) \right] \phi(\vec{r},t) - \nabla \cdot J(\vec{r},t) + \sum_{\varrho} \lambda_{\varrho} C_{\varrho}(\vec{r},t)$$

$$(3.3-2)$$

Integrating Equation (3.3-2) over the volume (Vi)

of some of node i, assuming the quantities being integrated are separable in space and time and constant over the node, and making use of Gauss' theorem yields.

$$\frac{V_{i}}{V_{i}(t)} \quad \frac{d\phi_{i}(t)}{dt} = \left[(1-\beta)v\Sigma_{fi}(t) - \Sigma_{Ai}(t) \right] \phi_{i}(t)V_{i} - \int_{Surface} J(\vec{r}_{s}) \cdot \overline{n} \cdot dr_{s} + V_{i} \sum_{\varrho} \lambda_{\varrho} C_{i\varrho}(t)$$
(3.3-3)

where

$$\int_{\text{surface}} J(\vec{r}_s) \cdot \vec{n} \cdot d\vec{r}_s = \sum_i J_{ij} A_{ij}$$

and

 J_{ij} = net current per unit area at the interface of nodes i and j. A_{ij} = area of the interface of node i and j. \sum_{i} = summation over all nodes immediately adjacent to node i.

If the flux is assumed to have the following spatial dependence between nodes i and j.

$$\phi(r) = \phi_{i} \frac{1-r}{d_{ij}} + \phi_{j} \frac{r}{d_{ij}}$$

where

 $d_{i,j}$ is the distance between the centroids of nodes i and j, then

$$J_{ij} = -D_{ij} \frac{d\phi}{dr} |_{r=r_{ij}} = \frac{-D_{ij}}{d_{ij}} (\phi_j - \phi_i)^*$$

If it is now assumed that for a small time step

$$(\Delta t = t_2 - t_1)$$

$$\frac{\partial \phi(t)}{\partial t} = \frac{\phi(t_2) - \phi(t_1)}{\Delta t}$$

Then Equation (3.3-3) can be approximated by

$$\frac{1}{v \Delta t} \left[\phi_{i}(t_{2}) - \phi_{i}(t_{i}) \right] = \left(1 - \sum_{\ell} \beta_{\ell} \right) v \Sigma_{f_{i}} \phi_{i}(t_{2}) - \Sigma_{A_{i}} \phi_{i}(t_{2}) + \frac{1}{V_{i}} \sum_{j} \frac{A_{ij} D_{ij}}{d_{ij}} \left[\phi_{j}(t_{2}) - \phi_{i}(t_{2}) \right] - \sum_{\ell} \lambda_{i\ell} C_{i\ell}(t_{2})$$

* D_{ij}, the diffusion coefficient at the interface adjacent nodes, is approximated by: D_{ij} $\simeq \sqrt{D_i} \sqrt{D_i}$

This was shown by Borresen (3-16) to be a very good approixmation when the dimensions of node i are similar to node j.

Solving for $\phi_i(t_2)$, yields

$$\phi_{i}(t_{2}) = \frac{V_{i}\left[\frac{1}{V\Delta t}\phi_{i}(t_{1}) + \sum_{\varrho}\lambda_{i_{\varrho}}C_{i\varrho}(t_{2}) + \right]\sum_{j}\frac{A_{ij}D_{ij}}{d_{ij}}\phi_{j}(t_{2})}{V_{i}\left[\frac{1}{V\Delta t} - (1 - \sum_{\varrho}\beta_{\varrho})_{\nu}\Sigma_{f_{i}} + \sum_{A_{i}}\right] + \sum_{j}\frac{D_{ij}A_{ij}}{d_{ij}}}{j}$$
(3.3-4)

At this time it is necessary to solve for the precursor densities as a function of flux. Recall that the precursor density equation is of the form:

$$\frac{\partial C_{g}}{\partial t} = \beta_{g} \vee \Sigma_{f} \phi - \lambda_{g} C_{g}$$

Making use of the finite difference approximation and using the average value of the flux and precursor density during the time step leads to the following expression:

$$\frac{C_{i\ell}(t_2) - C_{i\ell}(t_1)}{\Delta t} = \beta_{\ell} v \Sigma_{fi} \frac{\phi_i(t_1) + \phi_1(t_2)}{2} - \frac{\lambda \ell}{2} \left[C_{i\ell}(t_2) + C_{i\ell}(t_1) \right]$$

which can be solved for $C_{i\ell}(t_2)$ as:

$$C_{i\ell}(t_2) = \frac{C_{i\ell}(t_1) (2 - \lambda_{\ell} \Delta t) + \beta_{\ell} \Delta t \vee \Sigma_{fi} \left[\phi_i(t_1) + \phi_i(t_2)\right]}{2 + \lambda_{\ell} \Delta t}$$

substituting this expression for $C_{i\ell}(t_2)$ in Equation (3.3-4) yields:

$$\phi_{i}(t_{2}) = V_{i}\phi_{i}(t_{1})\left[\frac{1}{v\Delta t} + v \Sigma_{Fi}\sum_{\ell} \frac{\beta_{\ell}\Delta t\lambda_{\ell}}{2+\lambda_{\ell}\Delta t}\right] + V_{i}\sum_{\ell} \left(\frac{2-\lambda_{\ell}\Delta t}{2+\lambda_{\ell}\Delta t}\right)\lambda C_{i\ell}(t_{1}) + \frac{\sum_{\ell} \frac{A_{ij}D_{ij}\phi_{j}(t_{2})}{d_{ij}}}{V_{i}\left[\frac{1}{v\Delta t} - v\Sigma_{Fi}\left(1 - \sum_{\ell} \frac{2\beta_{\ell}}{2+\lambda_{\ell}\Delta t}\right) + \Sigma_{a_{i}}\right] + \sum_{j} \frac{A_{ij}D_{ij}}{D_{ij}}\right]$$
(3.3-5)

This equation is the iterative equation solved at the end of each time step of length Δt .

3.3.1.2 Calculations at Core-Reflector Interface

COTRAN employs a very simplified technique to account for the effects of the reflector core interface. Infinite difference form, the net leakage (L) into a node i can be expressed as:

$$L_{i} = \sum_{j} \frac{D_{ij}A_{ij}}{d_{ij}} (\phi_{j} - \phi_{i})$$
(3.3-6)

where

 D_{ij} = Effective diffusion coefficient at nodal surface = $\sqrt{D_i D_j}$ A_{ij} = Area of the interface between nodes i and j d_{ij} = Distance between centroids of nodes i and j ϕ_i = Flux of node i ϕ_i = Flux of node j
The technique used in COTRAN if node j is the reflector is to assume that:

 $\phi_j = 0$

and to adjust the value of the reflector diffusion coefficient (D_j) until realistic flux distrubitions are obtained when compared to a more sophisticated static calculation. It can be seen that reflector diffusion coefficients of 0.0 and ∞ yield reflecting and vacuum boundary conditions respectively.

3.3.2 COTRAN Thermal-Hydraulic Model

3.3.2.1 Transient Mixture Balance Laws

The integral balance laws which form the basis of COTRAN are formed on an Eulerion control volume, V, which is bounded by a fixed surface A. This surface may include solid interfaces, such as a fuel rod or structural wall, and fluid boundaries, but all solid material is outside V and composes the fuel thermal model in Section 3.3.2.4. The fluid in V is a single component, two phase mixture of liquid and vapor in thermodynamic equilibrium.

The integral balance on the control volume for the

mixture properties* mas, energy and momentum are:

$$\frac{\partial}{\partial t} \int_{V} \rho dV + \int_{A} \rho (\vec{u} \cdot \vec{u}) dA = 0$$
(3.3-7)

$$\frac{\partial}{\partial t} \int_{V} \rho e dV + \int_{A} \rho e(\vec{u} \cdot \vec{n}) da = \int_{V} (\rho(\vec{t} \cdot \vec{u}) + \rho r) dV + \int_{A} ((\vec{t} \cdot \vec{u}) - \vec{q}) \cdot \vec{n} dA \qquad (3.3-8)$$

$$\frac{\partial}{\partial t} \int_{V} \rho \vec{u} dV + \int_{A} \rho \vec{u} (\vec{u} \cdot \vec{n}) dA = \int_{V} \rho \vec{f} dV + \int_{A} (\vec{T} \cdot \vec{n}) dA \qquad (3.3-9)$$

respectively,** where

 \vec{u} = fluid velocity

 \vec{n} = unit outward normal

e = energy, internal thermal energy, i, and kinetic energy (e = $i+u^2/2$).

f = sum of all body forces acting on the fluid.

r = rate of internal heat generation/unit mass from all sources.

- * It is assumed that the local composition of the mixture can be described by the space-time average vapor volume fraction, ∝. Any mixture variable, Q, can be expressed as the volume weighted sum of the individual phase variables Q = ∞Qv + (1-∞) QL.
- ** Note the integral balance laws are of the form
 change of the total rate at which sum of all sources and
 amount of Q in V ⁺ Q is transported ⁼ sinks of Q inside V

across boundaries

 \dot{T} = surface stress tensor.

 \vec{q} = heat flux vector.

Only these three mixture conservation equations with one mixture equation of state and one relation specifying the relative velocity of one of the phases with respect to the other (or the mixture) are required to treat separated two-phase flow assuming thermal phase equilibrium.

The integral balance laws, Equations (3.3-7)

through (3.3-9), have been written for a single component two-phase mixture with the phases in thermodynamic equilibrium. Since the intended applications of COTRAN are for BWR channels with low speed flow and significant surface heat transfer the following assumptions apply:

- Kinetic energy changes are small compared to internal thermal energy changes
- Work done by body forces and shear stress is considered to be insignificant
- Gravity is the only significant body force
- Internal heat generation in the fluid is ignored
- Fluid flow is one dimensional

Under these assumptions, the only surface integrals of interest, associated with the solid interfaces, are the heat transfer and surface forces.

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The surface heat transfer integral will be modeled by the fourier law, for advection across the fluid boundary, and the product of an emperical surface heat transfer coefficient, H, and temperature difference for the solid interface. That is

$$\int_{A} (\vec{q} \cdot \vec{n}) dA = - \int_{F} K(\vec{\nabla} T \cdot \vec{n}) dA + \int_{W} H(T - T) dA \qquad (3.3-10)$$

where

k = Fluid thermal conductivity

 T_{f} = Local fluid temperature

 T_{W} = temperature of solid boundary

The stress tensor, T, can be written as the sum of a hydrostatic component, p, and a viscous stress tensor, π , as follows

$$\int_{A} (\vec{T} \cdot \vec{n}) da = - \int_{F} p\vec{n} dA + \int_{F} (\vec{\vec{\pi}} \cdot \vec{n}) dA + \left\{ - \int_{W} p\vec{n} dA + \int_{W} (\vec{\pi} \cdot \vec{n}) dA \right\}$$
(3.3-11)

The wall component, in brackets, will be modeled in the momentum equation by empirical friction factor and drag coefficient correlations. For the energy equation, in which work done by shear stresses has been assumed negligible, Equation (3.3-11) reduces to,

$$\int_{A}^{\vec{x}} (\vec{u} \cdot \vec{n}) dA = - \int_{F} \rho(\vec{u} \cdot \vec{n}) dA \qquad (3.3-12)$$

Applying these definitions and assumptions to the

original intergal balances, Equation (3.3-7) through (3.3-9), form the following equations.

$$\frac{\partial}{\partial t} \int_{V} \rho dV + \int_{F} \rho(\vec{u} \cdot \vec{n}) dA = 0$$
(3.3-13)

Energy

Macc

$$\frac{\partial}{\partial t} \int_{V} \rho h dV + \int_{F} \rho h(\vec{u} \cdot \vec{n}) dA = - \int_{F} K(\vec{\nabla} T \cdot \vec{n}) dA + \int_{W} H(T - T) dA \qquad (3.3-14)$$

where

 $\rho h = \rho i + p$ (The temporal derivative of pressure can be ignored

for low-speed flow).

Momentum

$$\frac{\partial}{\partial T} \int_{V} \rho \vec{u} dV + \int_{F} \rho \vec{u} (\vec{u} \cdot \vec{n}) dA = \int_{V} \rho \vec{g} dV - \int_{F} \rho \vec{n} dA + \int_{F} (\vec{\pi} \cdot \vec{n}) dA - \int_{W} \rho \vec{n} dA$$

$$+ \int_{W} (\vec{\pi} \cdot \vec{n}) dA$$
(3.3-15)

3.3.2.2 Channel Equations

In order to consider the essential nature of twophase flow, it is necessary to smooth out its chaotic nature. In deriving the integral balance laws of Section 3.3.2.1 it was assumed that the mixture variables are sufficiently space-time averaged to provide continuous derivatives inside the fixed volume and over its surface.

To solve the integral balance equations the following volume and surface averages are defined for an arbitrary mixture variable, Q,

$$\langle \langle Q \rangle \rangle_{V} = \int_{V} Q dV$$

 $\int_{V} Q dV = \frac{\int_{A} Q(\vec{u} \cdot \vec{n}) dA}{\int_{A} dA}$

Consider the channel section in Figure 3.3-1. The centroid is located at x and its length is Δx . Therefore, the upper and lower surfaces are at $x + \frac{(\Delta x)}{2}$ and $x - \frac{(\Delta x)}{2}$, respectively. The axial flow area is A and the axial velocity is u.

Mass Balance

The mass balance Equation (3.3-13) may be applied directly to the channel control volume.

 $V_{\overline{\partial T}}^{\partial} \ll V_{\Delta}^{\partial} = V_{\Delta}^{\partial} + V_{\Delta}^{\partial} +$

The channel equation is formed by dividing through

by Δx and taking the limit as Δx becomes small

$$A_{\partial t}^{\partial} << p>_{V} + \frac{\partial}{\partial X} <_{p} u>_{A} = 0$$
(3.3-16)

Energy Balance

Formal averaging of the surface heat fluxes will not be defined since commonly used surface heat transfer correlations already imply considerable surface averaging. Instead the average nuclear power in a radial region will be utilized such that the heat input to V from rods is

$$Q_{r} = \Delta x \left[P_{r} \phi H \right] \left[D_{r} \right] T + \Delta x \gamma$$

where

- H = surface heat transfer coefficient
- P = rod perimeter
- = number of rods contained in V
- $\begin{bmatrix} D_r \end{bmatrix}$ T = difference between the rod surface temperature and bulk fluid temperature
- γ = volumetric heat input from direct moderator heating Applying this definition and the energy balance law in Equation (3.3-14) to the channel control volume yields:

$$V\frac{d}{dt} << \rho h >>_{v} + < \rho u h >_{A}^{A} |_{x + \Delta x} - < \rho u h >_{A}^{A} |_{x - \Delta x} = \Delta x \left\{ \begin{bmatrix} P_{r} \phi H \end{bmatrix} \begin{bmatrix} D_{r} \end{bmatrix} T + \gamma \right\}$$

+ A $\left\langle K\frac{dT}{dx} \right\rangle |_{x + \Delta x} - A \left\langle K\frac{dT}{dx} \right\rangle |_{A} |_{x - \Delta x}$

again dividing by Δx and taking the limit as Δx becomes small leads to:

$$A_{\partial t}^{\partial} < \langle \rho h \rangle >_{V} + \frac{\partial}{\partial X} < \rho u h \rangle_{A}^{A} = \left[P_{\phi} H \right] \left[D_{r} \right] \left\{ T \right\} + \frac{\partial}{\partial X} A \left\langle K_{\partial X}^{\partial T} \right\rangle_{A} + \gamma \qquad (3.3-17)$$

Axial Momentum Balance

Before the integral momentum balance Equation (3.3-15) can be applied to the channel control volume, descriptions for the pressure and shear forces on the volume must be derived.

Using an area average, the net axial pressure force acting on the ends of the channel segment can be written as:

$$F_{p} = - \langle p \rangle_{A} A \Big|_{x + \Delta x} + \langle p \rangle_{A} A \Big|_{x - \Delta x}$$

If, however, the area varies axially an additional force, Fw, is exerted by the side walls. If both the area and pressure variation are assumed linear within Δx , this additional force is:

$$Fw = \langle p \rangle_{X} (A \left| x + \Delta X - A \right| x - \Delta x)$$

the total pressure force is simply the sum that is,

$$F_p + Fw = -Ax \left[\left_{x + \Delta x} - \left_{x - \Delta x} \right]$$

As mentioned in section 3.3.2.1 the rest of the solid interface stress integral is approximated by emperical wall friction correlations and form loss coefficients. The axial drag force is computed as:

 $F_D = 1/2 \left[f \Delta x P w + KA \right] <_{\rho} u^2 >_A$

where

f = dimensionless friction factor determined by correlation

Pw = wetted perimeter of the channel

K = total form loss coefficient in Δx

applying the difinition of the hydraulic diameter, D_{μ} ,

$$Fd = 1/2 \left(\frac{f \Delta x}{D_h} + K \right) <_{\rho} u^2 >_A A$$

where

f = 4f

Using these definitions the channel equation

becomes

$$V\frac{d}{dt} <<\rho u >>_{V} + <\rho u^{2} >_{A}A \Big|_{x + \Delta x} - <\rho u^{2} >_{A}A \Big|_{x - \Delta x} = -A_{x} \left(_{x + \Delta x} - <\rho >_{x - \Delta x} \right)$$
$$- 1/2 \left(\frac{f \Delta x}{D_{h}} + K \right) <\rho u^{2} >_{A}A - V <<\rho >>_{V}g COS\Theta$$

dividing by Δx and taking the limit produces:

where

 $\boldsymbol{\theta}$ is the channel orientation angle measured from the vertical.

3.3.2.3 Numerical Solution Procedure

At present, there are two independent solution schemes for the thermal-hydraulic balance equations in COTRAN. One is the implicit technique which provides a direct steady state solution but is limited to positive axial flow rates. The other solution scheme removes the positive flow restriction but is limited to small time steps. The explicit solution is further limited to transient problems although it may be initiallized by an implicit steady state calculation.

The implicit solution scheme includes options for two-phase slip models, void-quality relations and two-phase friction multipliers. The explicit procedure uses only the homogeneous equilibrium model for two-phase flow.

Both solution procedures employ the same fuel temperature model. This heat conduction model uses the method of weighted residuals by the orthogonal collocation technique. The model incorporates the Kirchoff transformation so that temperature-dependent thermal conductivity may be considered. The fuel is interfaced with the fluid thermalhydraulics by means of a surface heat transfer correlation specified by code input. Further details of the fuel model are presented in Section 3.3.2.4.

The two COTRAN solution schemes employ the reference pressure approach. This is, the local fluid density is assumed to be a function of the local enthalpy and a spatially uniform reference pressure. The assumption is valid as long as spatial pressure variations are small compared to the system pressure. In order to solve the three channel equations derived in Section 3.3.2.2, as well as the state equation, the area and volume averaged terms must be related so that the equations can be rewritten in terms of four primary variables. These are axial mass flow rate, M, mixture static enthalpy, h, or flowing enthalpy \hat{h} , mixture density, ρ , and pressure, P. The definitions required to form these variables are different between the two methods since the implicit solution is formed in terms of flowing quality and includes slip whereas the explicit method requires static quality and no slip between phases. Primary variables used in the solution are defined in Table 3.3-1.

Implicit Solution Scheme

If the implicit solution scheme is chosen in COTRAN, the problem is limited to positive flow rates. However, two phase slip flow can be considered with the assumption that the phases are in thermal equilibrium and that the phase velocities and volume fractions are uniformly distributed within the control volume.

Defining the flowing enthalpy and quality as:

 $\hat{h} = \frac{\langle \rho uh \rangle}{\langle \rho u \rangle}$

and

$$\hat{x} = \frac{\langle \alpha \rho v^{U} v^{\rangle}}{\langle \rho u \rangle}$$

respectively, and realizing that the assumption of uniform phase distribution implies that:

 $m = A < \rho u > = A < \langle \rho u \rangle >$

leads to the derivation of the continutiy equation from Equation (3.3-16) as:

$$A_{\partial t}^{\partial \cdot} \rho + \frac{\partial}{\partial x} m = 0$$
 (3.3-19)

Using the definition of flowing enthalpy, the

energy equation, Equation (3.3-17) becomes:

$$A\frac{\partial}{\partial t} \rho h + \frac{\partial}{\partial x} m \hat{h} = Q$$

where Q = the terms on the right hand side of Equation (3.3-17) factoring yields,

$$A\frac{\partial}{\partial t}\rho h + m\frac{\partial}{\partial x}\hat{h} + \hat{h}\frac{\partial}{\partial x}m = 0.$$
 (3.3-20)

at this point it is beneficial to define a new property, $\psi,$ introduced by Tong in 1965 $^{(3-17)},$ which is defined as

$$\psi = \rho(h-h)/h_{fa}$$

which can be rearranged to

$$ph = ph - h_{fa}\psi$$

substituting Equation (3.3-20) reduces to

$$A\left(\frac{\partial}{\partial t}\circ\hat{h} - h_{fg}\frac{\partial\psi}{\partial t}\right) + m_{\partial x}^{\partial}\hat{h} + \hat{h}\frac{\partial}{\partial x}m = Q$$

introducing the continuity equation, Equation (3.3-19) for $\hat{h} \; \frac{\partial}{\partial x} \; m$ gives

$$A\left(\frac{\partial}{\partial t}\rho\hat{h} - \hat{h}\frac{\partial}{\partial t}\rho - h_{fg}\frac{\partial\psi}{\partial t}\right) + m\frac{\partial}{\partial x}h = Q$$

which is equivalent to

$$A\left(\rho - h_{fg} \frac{\partial \psi}{\partial \hat{h}}\right) \frac{\partial \hat{h}}{\partial t} + m \frac{\partial \hat{h}}{\partial x} = 0$$
(3.3-21)

The axial momentum equation, Equation (3.3-18)

is rewritten by use of the definitions of the mementum velocity, \hat{u} , and the assumption of uniform phase distribution:

$$\frac{\partial}{\partial t}$$
 m + $\frac{\partial}{\partial x}$ mu = F

where F contains all the terms on the right hand side of Equation (3.3-18), applying the definition of the specific volume for momentum, \hat{v} , and differentiating,

$$\frac{\partial m}{\partial t} + \frac{V}{A} 2m \frac{\partial m}{\partial X} + m^2 \frac{\partial}{\partial X} \frac{V}{A} = F$$
(3.3-22)

applying the continuity equation yields

$$\frac{\partial m}{\partial t} - 2m \hat{v} \frac{\partial p}{\partial t} + m^2 \frac{\partial v/A}{\partial X} = F$$

Equations (3.3-19), (3.3-21) and (3.3-22) are the basic equations from which the implicit numerical scheme is derived. These partial differential equations are approximated by finite differences when

the channel is divided into a finite number of axial segments and the variables assigned positions on the computational mesh.

The computational mesh illustrated in Figure 3.3-2 leads to the following finite difference equations:

Fluid Continuity

$$\frac{1}{\Delta t}\overline{A}_{i} \left(\rho_{i} \rho_{i}^{n}\right) + \frac{m_{i}-m_{i-1}}{\Delta X} = 0$$
(3.3-23)

Energy

$$\frac{1}{\Delta t}\overline{A}_{i} (p_{i}^{n} - h_{fg} \frac{\partial \psi}{\partial \hat{h}}) (\hat{h}_{i} - \hat{h}_{i}^{n}) + m_{i-1} (\hat{h}_{i} - \hat{h}_{i-1}) = p\phi\overline{H} \left[D_{r} \right] \left\{ T_{r} \right\}$$

$$(3.3-24)$$

+
$$\gamma_i + \frac{1}{\Delta x^2} (kAi (T_{i+1} - T_i) - KAi - 1 (T_i - T_{i-1}))$$

Axial Momentum

$$\frac{\mathbf{m}_{i}-\mathbf{m}_{i}}{\Delta t} - 2\overline{m} \frac{\hat{\mathbf{v}}_{i}}{\Delta t} \left(\frac{\mathbf{p}_{i}-\mathbf{p}_{i}}{\Delta t}\right) + \mathbf{m}_{i-1}^{2} \left(\frac{(\mathbf{v}/A)_{i}-(\hat{\mathbf{v}}/A)_{i-1}}{\Delta X}\right) = -\overline{A} \left(\frac{\mathbf{p}_{i}-\mathbf{p}_{i-1}}{\Delta X}\right)$$
(3.3-25)

 $\overline{A}_{i}K_{L}m_{i}^{2}$ - $\overline{A}p_{i}gcos\theta$

where

$$\overline{A}_i$$
 = average flow area = 0.5($A_i + A_{i-1}$)

$$K_{L} = \frac{v \varepsilon f_{\phi}}{2D_{h}A_{i}^{2}} + \frac{Kv}{2\Delta XA_{i}^{2}}$$

T = channel temperature

Tr = rod surface temperature

 $\overline{m} = 0.5 (m_i + m_{i-1})$

and the superscripts are defined as

n = previous time step (no superscript implies present time)

v = present time but previous iteration.

Equations (3.3-23), (3.3-24) and (3.3-25) are the iterative equations used in the implicit solution scheme of COTRAN.

Explicit Solution Scheme

Unlike the finite difference technique employed by the implicit solution the explicit solution scheme solves the cell balance equations directly. To solve the cell balance equations, two-phase flow is assumed to be completely homogeneous with the phases in thermal equilibrium. This restriction on fluid modeling is compensated, however, by the capability of addressing reverse flow conditions.

The homogeneous assumption implies that both phase velocities are equal (no slip) and that the phase distribution is uniform throughout the control volume. These assumptions lead to the following definitions:

• <pu> = <<p>u> = <<p>u

where h = average enthalpy = <<ph><<ph><<ph>

• $x = \frac{\alpha \rho v}{\langle \langle \rho \rangle \rangle} = \text{static quality}$

• $h = xh_v + (1-x)h_g$

To form the difference equations requires the formulation of a computational cell and the assignment of primary variables to the computational mesh. The cell used in the explicit solution is shown in Figure 3.3-3. A cell balance leads to the following equations for mass, energy and momentum conservation⁺.

 $\frac{Mass}{\hat{A}_{i}} \Delta x \frac{d}{dt} \rho_{i}^{+} m_{i}^{-} m_{i-1}^{-} = 0 \qquad (3.3-26)$

Energy

$$\hat{A}_{i} \Delta X \frac{d}{dt} \rho_{i}h_{i} + m_{i}h_{i}^{*} - m_{i-1}h_{i-1}^{*} = \Delta X \left\{ P_{r}\phi H \right\} \left[D_{r} \right] \left\{ T_{i} \right\} + \gamma_{i}$$

$$+ \frac{1}{\Delta X} \left\{ Ak_{i} \left(T_{i+1} - T_{i} \right) - A\hat{k}_{i-1} \left(T_{i} - T_{i-1} \right) \right\}$$

$$(3.3-27)$$

Momentum

$$\frac{d}{dt} = m_{i} + \hat{A}_{i} \frac{1}{\Delta x} (p_{i+1} - p_{i}) + \frac{1}{2} \left(\frac{f}{D_{H}} + \frac{k}{\Delta x} \right) \left(\frac{m^{2}}{\rho^{*}A} \right)_{i}$$

$$- \frac{1}{\Delta x} \left(\hat{u}_{i+1} m^{*}_{i+1} - \hat{u}_{i} m^{*}_{i} \right) = -\hat{A}_{i} \hat{\rho}_{i} g \cos \theta \qquad (3.3-28)$$

+ All terms on the right hand side of the equals sign are computed from the previous time step information.

80

XN-NF-80-19(NP)(A) Vol. 1 Where the superscript * denotes convected quanti-

ties. That is, the enthalpy convected by m_i in the energy equation is denoted by h^* and is defined as:

$$h_{i}^{*} = h_{i}^{i} \text{ if } m_{i}^{i} > 0$$

 $h_{i}^{*} = h_{i+1}^{i} \text{ if } m_{i}^{i} < 0$

The basis of the explicit solution is an explicit energy equation using flows and energies from the previous time step to form the convective terms. Consider the abgreviated forms of the cell balance equations:

Mass

$$\hat{A}_{i} \frac{\Delta x}{\Delta t} (\rho_{i} - \rho_{i}^{n}) + m_{i} - m_{i-1} = 0$$
(3.3-29)

Energy

$$\hat{A}_{i\Delta t} (\rho_{i}h_{i} - \rho_{i}h_{i}^{n}) + m_{i}h_{i}^{*} - m_{i-1}h^{*}_{i-1} = Q_{i}^{n}$$
(3.3-30)

Momentum

$$m_i - m_i^n - \Delta t \frac{Ai}{\Delta x} (p_{i+1} - p_i) - t F_i^n$$
 (3.3-31)

where the superscript n denotes previous time step

By inverting the state equation, enthalpies can be expressed in terms of specific volumes, v, i.e., ++

 $h_i^* = h_o + \frac{\partial h}{\partial v} \Big|_p (v_i^* - v_o)$

apply this equation to the energy balance yields:

$$\begin{array}{l} m_{i}v_{i}^{*} - m_{i-1}v_{i-1}^{*} - \frac{\partial v}{\partial h}\Big|_{p} Q_{i}^{n} = \left(v_{0} - \frac{\partial v}{\partial h}\Big|_{p}h_{0}\right) \left(\hat{A}_{i} \frac{\partial x}{\partial t} \left(\rho_{i} - \rho_{i}^{n}\right) + m_{i} - m_{i-1}\right)$$

$$(3.3-32)$$

by continuity the right hand side equals zero.

The left side of Equation (3.3-32) is the basis of the explicit solution scheme. At the beginning of a time step the fuel model is evaluated and the explicit terms Q and F are determined. An initial estimate of \tilde{m} is obtained from the momentum equation based on pressures and flows from the previous time step. When this value is used in Equation (3.3-32) there will be a residual error, E_i , that is:

$$\widetilde{\mathbf{m}}_{i}\mathbf{v}_{i}^{*} - \widetilde{\mathbf{m}}_{i-1}\mathbf{v}_{i-1}^{*} - \frac{\partial \mathbf{v}}{\partial \mathbf{h}}\Big|_{p} \mathbf{Q}_{i}^{n} = \mathbf{E}_{i}$$
(3.3-33)

++ h_0 and v_0 define a reference state close enough to h^* and v^* so that $h^{2}v$ can be assumed constant between the two states.

This residual error is reduced to near zero in all computational cells by adjusting the pressure and flows in each cell in an iterative loop. The pressure change, Δp_i , needed to reduce E_i to zero in any cell is computed from $\partial E/\partial p$.

$$\Delta p_i = \frac{-E_i}{\partial E/\partial p}$$
(3.3-34)

The total derivative is formed by holding the specific volume constant:

$$\frac{\partial E}{\partial p_{i}} = \frac{\partial E}{\partial \tilde{m}_{i}} \frac{\partial \tilde{m}_{i}}{\partial p_{i}} + \frac{\partial E}{\partial \tilde{m}_{i-1}} \frac{\partial \tilde{m}_{i-1}}{\partial p_{i}}$$

The flow differentials are formed from Equation (3.3-31).

$$\frac{\partial \tilde{m}_{i}}{\partial p_{i}} = \frac{\Delta t \hat{A}_{i}}{\Delta x}$$

$$\frac{\partial \tilde{m}_{i-1}}{\partial p_{i}} = \frac{\Delta t \hat{A}_{i-1}}{\Delta x}$$
(3.3-35)

from Equation (3.3-33).

$$\frac{\partial E}{\partial \tilde{m}_i} = v_i^*$$

and

$$\frac{\partial E}{\partial \tilde{m}_{i-1}} = v^*_{i-1}$$

therefore

$$\frac{\partial E}{\partial p_{i}} = \frac{\Delta t}{\Delta x} \left[v_{i}^{*} \hat{A}_{i} + v_{i-1}^{*} \hat{A}_{i-1}^{*} \right]$$
(3.3-37)

The pressure change is computed from Equation (3.3-34) and then used to update the cell flow and density. Flows are updated by the momentum derivatives Equations (3.3-35) and (3.3-36). These updated flows are then used in the continuity equation to determine the new density and specific volume. This procedure is repeated over all cells until the maximum error, E, is less than a specified value. The solution is then considered converged.

3.3.2.4 COTRAN Fuel Model

The conductive heat transfer model used in COTRAN calculates the internal temperature distribution of the fuel rod and the surface heat flux to the adjacent fluid channel. The model⁽³⁻¹⁸⁾, which is a combination of the Method of Weighted Residuals (MWR) in the radial coordinate and finite differences in time and the axial coordinate, can include options for axial conduction and temperature dependent fuel thermal conductivity.

Fuel Interior

The fundamental heat conduction equation is:

$$\rho c_{\partial T} = \nabla \cdot (K \nabla T) + q^{2}$$
(3.3-38)

where

K = local thermal conductivity and

q' = volumetric rate of heat generation in the fuel rod.

We can write this equation in cylindrical coordinates as:

$$\frac{\partial c_{\partial T}}{\partial t} = \frac{1}{p^2 r} \frac{\partial}{\partial r} \frac{rK(T)\partial T}{\partial r} + \frac{\partial}{\partial x} \frac{K(T)}{\partial x} + q^{2}$$
(3.3-39)

where

0

r = r¹/R
r¹ = radial coordinate
R = fuel radius

Making use of Kircoffs Transformation,

$$= \frac{1}{K_0} \int_{T_0}^{T} K(T) dT = G(T)$$
(3.3-40)

where ko is the conductivity at the reference temperature To, allows Equation (3.3-39) to be written as:

$$\rho c \frac{Ko}{K(T)} \frac{\partial \theta}{\partial t} = \frac{Ko}{R^2 r} \frac{\partial}{\partial r} r \frac{\partial \theta}{\partial r} + \frac{\partial}{\partial x} K(T) \frac{\partial T}{\partial x} + q^{---}$$
(3.3-41)

If the radial coordinate is approximated by the symmetric polynomial of the form:

$$\theta(r) = d_1 + d_2 r^2 + d_3 r^4 + \dots + d_n r^{2n-2} = \sum_{i=1}^{N} (r^{2i-2}) di$$

and evaluated at the N radial positions, yields,

$$\Theta(r_j) = \sum_{i=1}^{N} (r_j^{2i-2}) di$$
(3.3-42)

or, rewritten in matrix notation

$$\left\{ \theta \right\} = \left[Q \right] \left\{ d \right\}$$

where

$$Q_{ji} = r_j^{2i-2}$$

In COTRAN the radial positions (r_j) are taken to

be the roots of orthogonal polynomials as defined by Finlayson $(1974)^{(3-19)}$.

From Equation (3.3-42) the first and second radial derivatives can be derived:

$$\frac{\partial \theta}{\partial r_j} \Big|_{r_j} = \sum_{i=1}^{N} (2i-2) r_j {(2i-3) \atop d_i} d_i$$
 (3.3-43)

and

$$\frac{1}{r} \frac{\partial}{\partial r} r \frac{\partial \theta}{\partial r} = \sum_{i=1}^{N} (2i-2) (2i-2) r_{j}^{(2i-4)} d_{i} \qquad (3.3-44)$$

(3.3-46)

which may be written in matrix notation as:

$$\frac{\partial}{\partial r} \left\{ \theta \right\} = \left[c \right] \left[0 \right]^{-1} \left\{ \theta \right\}$$
(3.3-45)

and

$$\frac{1}{r} \frac{\partial}{\partial r} r \frac{\partial}{\partial r} \left\{ \theta \right\} = \left[D \right] \left[Q \right]^{-1} \left\{ \theta \right\}$$

where

$$\begin{cases} d \\ has been replaced by [Q] ^{-1} \\ \theta \\ c_{ji} = (2i-2) r_j^{2i-3} \\ D_{ji} = (2i-2)^2 r_j^{2i-4} \end{cases}$$

substituting Equation (3.3-46) into Equation (3.3-41 yields)

$$\rho c \frac{K_0}{K(T)} \frac{\partial \theta}{\partial t} = \frac{K_0}{R^2} \begin{bmatrix} B \end{bmatrix} \left\{ \theta \right\} + \frac{\partial}{\partial x} K(T) \frac{\partial T}{\partial x} + q^{---}$$
(3.3-47)

where

 $\begin{bmatrix} B \end{bmatrix} = \begin{bmatrix} D \end{bmatrix} \begin{bmatrix} Q \end{bmatrix}^{-1}$

Approximating the axial conduction term by a central finite difference and the time derivative by a forward finite derivative yields the heat conduction model at N-1 interior nodel positions.

$$\frac{\rho c K_{0} \theta}{\Delta t K_{1}} - \frac{K_{0}}{R^{2}} \sum_{l=1}^{N} B_{ll} \theta_{l} = \frac{\rho c K_{0}}{\Delta t K_{1}} \theta_{l}^{n} + q^{m}$$

$$+ \frac{2}{\Delta x^{2}} \left[\frac{\tilde{T}_{j-1} - \tilde{T}_{i}}{\frac{1}{K_{j-1}} + \frac{1}{K_{1}}} + \frac{\tilde{T}_{j+1} - \tilde{T}_{i}}{\frac{1}{K_{j+1}} + \frac{1}{K_{1}}} \right]$$
(3.3-48)

Fuel Clad Interface

The boundary condition at the fuel surface is

handled by a lumped resistance technique. The equation is:

$$\frac{\partial \Theta}{\partial r} = H_{a} \left(T_{N} - T_{N+1} \right)$$
(3.3-49)

where

 T_N = fuel exterior surface temperature

 T_{N+1} = clad exterior surface temperature

$$\frac{1}{H_g} = \frac{1}{H_c} + \frac{Y_c}{Kc}$$

Hc = Fuel-clad gap conductance

Yc = Clad thickness

Kc = Clad conductivity

applying Equation (3.3-45) produces:

$$-\frac{K_{0}}{R} \sum_{l=1}^{N} A_{N_{1}} \theta_{l} = H_{g} (T_{N} - T_{N+1})$$
(3.3-50)

where

$$[A] = [C] [Q]^{-1}$$

Cladding

A transient energy balance for the lumped clad is:

$$pc \frac{\partial^{T} N+1}{\partial t} = \frac{Hg}{Yc} \left(\frac{r_{N}}{r_{N+1}} \right) \left({}^{T} N^{-T} N+1 \right) - \frac{Hs}{Yc} \left({}^{T} N+1 {}^{-T} F \right) + Kc \frac{\partial^{2} T_{N+1}}{\partial x^{2}}$$
(3.3-51)

where Hs is the clad surface heat transfer coefficient and T_F is the fluid temperature. Using Kircoffs transformation, an implicit time derviative, and an explicit axial conduction term gives:

$$\frac{(\rho c)_{c}}{\Delta t} \frac{K_{0}}{K_{N+1}} \stackrel{\Theta N+1}{=} \frac{(\rho c)_{c}}{\Delta t} \frac{K_{0}}{K_{1}} \stackrel{\Theta N+1}{=} + \frac{K_{c}}{\Delta x^{2}} \left(\stackrel{\sim}{T}_{1+1}^{-2T} \stackrel{\sim}{N+1} + \stackrel{\sim}{T}_{1-1} \right)$$

$$+ \frac{Hg}{Y_{c}} \left(\frac{r_{N}}{r_{N+1}} \right) \left(\stackrel{T}{N} \stackrel{-T}{N+1} \right) - \frac{Hs}{Y_{c}} \left(\stackrel{T}{T}_{N+1} \stackrel{-\stackrel{\sim}{T}_{F}}{F} \right)$$

$$(3.3-52)$$

The implicit temperature T_N and T_{N+1} appearing in Equations (3.3-50) and 3.3-52) are evaluated by a truncated Taylor series as:

$$T = T^{n} - \frac{G(T^{n}) - \theta(T)}{G(T^{n})}$$
(3.3-53)

where G is defined in Equation (3.3-40) and G⁻ is the derivative of G with respect to T.

Solution Scheme

The boundary conditions at the fuel-clad interface Equation (3.3-50) and the clad surface Equation (3.3-52) are combined with the differential heat conduction Equation (3.3-48) to yield a matrix equation of the form

$$\begin{bmatrix} a \end{bmatrix} \left\{ \theta \right\} = \begin{bmatrix} Q \end{bmatrix}$$

for the transformed temperatures at one axial level. These matricies (shown) in Figure 3.3-4 for a third order orthogonal collocation are solved by an iterative Gauss-Siedel procedure. Once θ is determined, the temperature solution is evaluated by Equation (3.3-53). This temperature solution is then utilized to determine a rod average fuel temperature for Doppler Feedback.

3.3.3 Description of Code Mechanics and Output Features

COTRAN will initially determine the static flux, power and hydraulic distribution corresponding to the input it has received. This includes a user specified option for an input axial power profile. When this option is selected the code will iterate on control density in each node until the desired power shape is achieved. If no transient calculations are to be performed the code will edit the results, punch out control densities and fluxes for a restart if so desired, and exit. If transient calculations are to be made, equilibrium precursor concentrations will be determined and all production corss sections will be divided by the calculated K_{eff} to insure $K_{eff} = 1.000$ at the beginning of the transient.

The initial time step size is 0.0001 seconds for the explicit solution (0.005 seconds for the implicit solution). The forcing functions are updated and the new thermal-hydraulic solution determined. This solution is used to update the cross sections and a new flux calculation is made. At the end of each time step the precursor densities are

updated for the next time period. The input variable LATCH is used to control the time step size and thus relieve the user of choosing time step sizes. The time step size is doubled after LATCH time steps have been taken consecutively during which the number of flux iterations to achieve convergence has not exceeded 25. If at any time more than 60 iterations are required, the time step is halved. This scheme permits small time steps during times of large changes in power level and inversely, large time steps during periods of slow change.

3.3.4 Input and Use of Cross Sections

COTRAN requires two sets of two group macroscopic cross sections for each fuel type in the problem. These cross section sets describe the material in its entirely uncontrolled and completely controlled states [______]. A control density array can then be input by the user or calculated by the code to describe the initial conditions of the core. Linear interpolation is utilized to determine the cross sections for each fuel node at a given control density and void fraction. [

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bases for requiring two group cross sections as input is that normally only

the fast neutron group cross sections vary with fuel temperature.^(3-20,3-21) Primarily the fast absorption and slowing down cross sections are affected by fuel temperature. COTRAN, therefore, allows the two group cross section values to change with fuel temperature (based on the average nodal fuel temperature at each time step) and then collapses to new one group values for the next solution. This feature allows a COTRAN calculation to exhibit many of the characteristics of a two group solution at the much reduced computer time of a one group neutron diffusion theory code.

It has been found that the fast neutron cross sections affected by fuel temperature vary linearly as the square root of the fuel temperature $({}^{O}K)$. (3-21) These rates of change cross sections with fuel temperature are input for each material type.

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The input cross section tables for COTRAN are calculated using XTGBWR. [

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3.4 MULTIGROUP DIFFUSION THEORY MODEL (XDT)

The Exxon Nuclear model used to perform multigroup diffusion theory calculations for BWR fuel assemblies is the XDT code. The XDT code was developed from the $2DB^{(3-22)}$ code that was written for fast reactor multigroup calculations.

Table 3.3.1 Definitions of Variables for Thermal-Hydraulic Solutions

Symbol	Description and Units Definition		
m	Axial mass flow rate, (lb/sec)	m = <pu> A</pu>	
ρ	Mixture density, (lb/ft ³)	$\rho = <<\rho>>_{V} + (1-\alpha)\rho_{\ell}$	
p	Two-phase momentum density, (lb/ft ³)	$\hat{\rho} = \frac{\langle \rho u \rangle^2}{\langle \rho u^2 \rangle}$	
	In terms of void-quality	$\frac{1}{\rho} = \frac{(1-x)^2}{\rho_{\ell}(1-\alpha)} + \frac{x^2}{\rho_{V\alpha}}$	
ŷ	Momentum specific volum , (ft ³ /1b)	$\hat{v} = \frac{1}{\rho}$	
v	Mixture specific volume, (ft ³ /lb)	v = 1/p	
	In terms of static quality	$v = xv_v + (1-x)v_{\ell}$	
h	Mixture enthalpy, (Btu/lb)	$h = \frac{\langle \langle \rho h \rangle \rangle}{\langle \langle \rho \rangle \rangle}$	
	In terms of static quality	$h = xh_v + (1-x)h_g$	
ĥ	Flowing enthalpy, (Btu/1b)	$\hat{h} = \langle \rho u h \rangle / \langle \rho u \rangle$	
	In terms of flowing quality	$\hat{h} = \hat{x}h_v + (1-\hat{x})h_{\ell}$	
â	Flowing quality	$\hat{x} = \langle \alpha \rho_V u_V \rangle / m$	
х	Static qualtiy (vapor mass fraction)	x = <<αρ _y >>/ρ	
ψ	Function defining relationship between h and \hat{h} , (lb/ft ³)	$\psi = \rho_{\ell} x (1-\alpha) \\ -\rho_{\nu} \alpha (1-x)$	
α	Vapor volume fraction	$\alpha = (\rho_{\ell} - \rho) / (\rho_{\ell} - \rho_{V})$	
u	Axial velocity, (ft/sec)	$u = \langle \rho u \rangle / \rho = m / \rho A$	
Р	Pressure, (1b _f /ft ²)	$P = P_v = P_{\ell}$	
А	Subchannel flow area, (ft ²)		



Figure 3.3-1 Channel Control Volume for Thermal-Hydraulic Balance Equations



• i-1

Figure 3.3-2 Placement of Variables for Implicit Solution

• i+1



• i-1



i+1



Figure 3.3-4 Matrix Elements for Third Order Collocation

The XDT code is used to perform special diffusion theory calculations such as four and sixteen bundle calculations, fuel misloading calculations, and incore detector calculations. The cross sections for the XDT code are generated by the XFYRE depletion model.

The XDT code calculates the eigenvalue, relative powers, multigroup neutron fluxes, and flux and volume weighted cross sections.

Eigenvalues are computed in XDT by standard source-iteration techniques. Group rebalancing and successive over-relaxation with line inverson are used to accelerate convergence. Adjoint solutions are obtained by inverting the input data and redefining the source terms.

Variable dimensioning is used to make maximum use of the available fast memory. Since only one energy group is in the fast memory at any given time, the storage requirements are insensitive to the number of energy groups.

Neutron Balance Equations

The multigroup diffusion equations can be written in the form

$$D_g \nabla^2 \phi g - \Sigma_g^r \phi_g + S_g = 0, g = 1, ... N$$
 (3.4-1)

where

$$S_{g} = \frac{\chi_{g}}{k_{eff}} \sum_{g'=1}^{N} (v \Sigma_{f})_{g} \phi_{g'} + \sum_{g'=1}^{g-1} \Sigma(g' \rightarrow g) \phi_{g'}$$
(3.4-2)

and:

N	=	number of energy groups,
g	=	energy group index,
φg	=	flux in group g,
Sg	=	source in group g.
Dg	=	diffusion constant for gourp g (= $1/3 \Sigma_g^{tr}$),
(v∑f)g		fission source cross section for group g,
Σ(g*→g)	=	group transfer cross section from g' to g,
Σg	=	removal cross section from group g
		N

$$= \left[\sum_{g=g+1}^{a} + \sum_{g'=g+1}^{n} \sum_{\Sigma(g \neq g')} \right],$$

x g

= fission source fraction in group g,

K_{eff} = effective multiplication constant.

The mesh points in the XTD code are located in the center of the homogeneous mesh interval (see Figure 3.4-1). This choise leads to a more clean-out calculation and interpretation of all reaction rates.

The spatical difference equations are obtained by integrating Equations (3.4-1) and (3.4-2) over the volume associated with each mesh point. For the (i,j) mesh point shown in Figure 3.4-1, the radial integration would be from $(R_i - \frac{\delta R_i}{2})$ to $(R_i + \frac{\delta R_i}{2})$, and the axial integration would be from $(Z_j - \frac{\delta Z_j}{2})$ to $(Z_j + \frac{\delta Z_j}{2})$.
The leakage terms are obtained by first transforming the volume integral over the Laplacian to a surface integral using Green's theorem,

$$\int DA^2_{\phi} dV = \int D\vec{\nabla}\phi \cdot d\vec{A} \cdot (3.4-3)$$

The flux gradients at the mesh boundary are obtained by interpolating the two contiguous flux values, Thus, volume integration of Equation (3.4-1) for mesh point o (see Figure 3.4-1) leads to the expression

$$\sum_{k=1}^{4} \frac{\overline{D}_{k}A_{k}}{\mathcal{L}_{k}} (\phi_{k} - \phi_{0}) - \Sigma_{0}^{r}\phi_{0}V_{0} + S_{0}V_{0} = 0, \qquad (3.4-4)$$

where, for simplicity, the gourp indices have been omitted, and

 Σ_{0}^{r} = removal cross section associated with mesh point o,

 S_{o} = source rate associated with mesh point o,

V = volume associated with mesh point o,

 ϕ_k = flux associated with mesh point k,

 \mathfrak{l}_k = distance between mesh point k and mesh point o,

 A_{L} = area of boundary between mesh point k and mesh point o,

 \overline{D}_k = effective diffusion constant between mesh point k and mesh point o

$$\overline{D}_{k} = \frac{D_{o} D_{k} (\delta R_{o} + \delta R_{k})}{D_{o} \delta R_{k} + D_{k} \delta R_{o}} .$$
(3.4-5)

Finally, Equation (3.4-4) can be recase into a form more convenient for performing flux iterations. That is

$$\phi_{0} = \frac{S_{0}V_{0} + \sum_{k=1}^{4} C_{k}\phi_{k}}{C_{5}}, \qquad (3.4-6)$$

where

$$C_k = \frac{\overline{D}_k A_k}{2k} \quad k=1, \dots 4$$
, (3.4-7)

and

$$C_5 = \Sigma_0^r V_0 + \sum_{k=1}^4 C_k . \qquad (3.4-8)$$

Discussion of Boundary Conditions

Three standard conditions are available in XDT: $\vec{\nabla}\phi = 0$, $\phi = 0$, and periodic. These are described below using a slight modification to the nomenclature developed in the foregoing sections.

Zero Flux Gradient

Consider the left hand boundary of the one-dimensional reactor shown in Figure 3.4-2. Let us now imagine that a pseudo mesh interval, interval o, has been added on the left hand side of the boundary with the same composition and thickness of interval 1. Clearly, then if $\vec{\nabla}\phi = 0$ at the boundary, $\phi_0 = \phi_1$. Therefore, since $(\phi_0 - \phi)$ vanishes, the coefficient of $\phi_0 - \phi_1$, C_1 (see Equation (3.4-4)), is immaterial--hence C_1 can be set equal to zero. The calculation is performed assuming that ϕ_0 does not exist and $C_1 = 0$.

Zero Flux

Again, imagine that a pseudo mesh interval with the same composition as interval IM has been added to the right hand side of the right boundary. Now, since $\phi_{IM} \neq 0$ and $\phi_{IM+1} = 0$, the coefficient of $(\phi_{IM}-\phi_{IM+1})$ in Equation 3.4-4 cannot be disregarded. In fact, from Equation (3.4-7), it is clear that

$$C_{K} = \frac{D_{K} A_{K}}{.5 \ \delta R_{IM} + .71 \ \lambda_{tr}}$$
(3.4-9)

where λ_{tr} is assumed to equal $1/\Sigma_{tr}$.

Note, as in the $\vec{\diamond}_{\phi} = 0$ case, that there is no contribution of the pseudo flux in Equation (3.46). For a zero flux gradient, $C_{K} = 0$; whereas for a zero flux, $\phi_{K} = 0$.

Periodic Flux

Period boundary conditions are available for the top, bottom, left and right boundaries. In this option,

$$\phi_{0} = \phi_{IM}$$
 (3.4-10)

$$\phi_1 = \phi_{\text{IM}+1}$$
 (3.4-11)

and

$$C_{K}(1 \rightarrow IM) = \frac{\overline{D}_{K} A_{K}}{.5(\delta R_{1} + \delta R_{IM})}$$
(3.4-12)

It should be stressed that the pseudo mesh intervals discussed above are not in any way a part of the code. They are mentioned here only for heuristic purposes.

104

Solution of Different Equations

The eignvalue and flux profiles are computed by standard sourceiteration techniques, i.e., by using an initial fission source distribution, the flux profiles in each gorup are sequentially computed beginning in the top (highest energy) group. After the new fluxes for all groups have been calculated, a new fission source distribution is computed from the new flux profiles. The mulitplication ratio, λ , is then obtained by simply taking the ratio of the new fission source rate to the old (previous iteration) fission source rate. The above sequence of events is called an outer iteration.

Before each new outer iteration, the fission spectrum is multiplied by $1/\lambda$, so that λ approaches unity as the iteration proceeds. The effective multiplication constant is simply the product of the successive λ 's. Convergence is assumed when $|1-\lambda| < \varepsilon$, where ε is an input parameter.

Fission source over-relaxation is employed in XDT to accelerate convergence. The procedure is as follows: After the new fission source rate profile, $F_1^{\nu+1}$, is calculated, a second "new" value, $F_2^{\nu+1}$, is computed by magnifying the difference between the new fission source rate and the old fission source rate. Thus,

 $F_2^{\nu+1} = F^{\nu} + B^{-}(F_1^{\nu+1} - F^{\nu})$,

(3.4 - 13)



Figure 3.4-1 Mesh Description



Figure 3.4-2 Schematic Diagram of 1-D Reactor

(3.4 - 15)

where β^{-} is the fission source over-relaxation factor. $F_{2}^{\nu+1}$ is then normalized to give the same total source as $F_{1}^{\nu+1}$.

The group-fluxes are computed using successive line overrelaxaton (SLOR). That is, the fluxes on each vertical (or horizontal) line are simultaneously computer (by the familiar Crout reduction technique) and then over-relazed using the algorithm

$$\phi^{\nu+1} = \phi^{\nu} + \beta(\phi^{\nu+1} - \phi^{\nu}) , \qquad (3.4-14)$$

where ß is the over-relaxation factor. In R-0 problems or problems involving periodic boundary conditions, direct inversion is performed on veritical lines beginning at the left boundary and proceeding by column to the right boundary. In triangular problems, direct inversion is performed along horizontal lines beginning at the bottom boundary and proceeding by row to the top boundary. In all other situations, direct inversion is used along the dimension with the most mesh points. One mesh sweep is defined as one inner iteration.

The flux over-relaxation factor, β , is an input parameter. The fission source over-relaxation factor, β , is computed internally from the ad hoc expression

 $\beta^{-} = 1.0 + .6(\beta - 1)$.

The flux in each group is normalized (by balancing the total source and loss rate) immediately before each group-flux calculation. Thus, one-region problem with zero-gradient boundary conditions would be solved exactly in one outer iteration.

It should be mentioned that an altering direction SLOR scheme (single line inversion for rows and then columns in alternation) is included as an option to enhance convergence for problems involving tight mesh spacing in both dimensions.

3.5 MONTE CARLO MODEL (XMC)

The Exxon Monte Carlo Code (XMC) is a general purpose Monte Carlo code developed from the Battelle Monte Carlo Code $(BMC)^{(3-23)}$. XMC was designed to calculate thermal reactor benchmark problems. XMC is capable of describing the exact geometrical description of a light water fuel assembly. This geometrical capability and a coupled space-energy solution of the transport equation mak s the Monte Carlo method in XMC a highly accurate method for evaluating key nuclear parameters and the effects of water gaps, control blades and burnable poison rods in light water reactor fuel bundle.

XMC uses basic cross section and neutron scattering data to calculate the various neutronic events. Thus, its accuracy is limited only by the accuracy of the basic cross sections and the number of neutron histories which are run for each problem.

The neutron flux, reaction rates by isotope and region, groupaveraged cross sections, neutron leakage rates, and the standard deviation for each of these parameters are calculated in three dimensional space over the energy range from 0 to 10 MeV. The reaction types included are fission, capture, inelastic scattering, n-2n scattering, elastic scattering with

isotropic or anisotropic angular distributions, and thermal scattering based $(1 \text{ a scattering law generated using the Haywood}^{(3-3)}$ representation of the phonon spectrum for water. The energy distribution of the neutrons is continuous. However, the cross sections are averaged over 190 microscopic energy groups. Resolved resonance cross sections are calculated by the code for each neutron energy using the Doppler-broadened Breit-Wigner single-level formula.

The isotopic material cross sections are processed from the ENDF/B format⁽³⁻²⁾. The data can be provided either from the Battelle Master Library⁽³⁻¹⁾, from the ENDF/B Library, or from any source which can be put into the ENDF/B format.

The XMC code geometry routines can handle any region that can be enclosed by a set of boundaries of the general form

$$A(x-x_0)^2 + B(y-y_0)^2 + C(z-z_0)^2 - K = 0.$$

Provisions are included for several special forms of the above equations including planes, cylinders and spheres. Also, there is a special region geometry routine for a rectangular lattice of clad fuel rods.

Statistics for the values calculated are obtained by making a series of calculations on equal sized sets of neutron histories called "batches" and averaging the results from each batch. The initial spacevelocity-angle coordinates for each neutron of a batch are either picked from a random source distribution or from the fission particles produced by the previous batch.

Neutron absorption is accounted for by reducing the weight of the neutron at each collision. When the weight has been reduced sufficiently the neutron are terminated by a Russian Roulette process.

3.5.1 The XMC Code Package

The XMC code package consists of two parts: the Monte Carlo code and the cross-section data library with the processing support codes, XMCLIB and LIBR. The general flow diagram for the XMC code package is shown in Figure 3.5-1. The XMCLIB code uses isotopic or material cross sections and reaction rates data to prepare a cumulative cross section library (CCT). The data is obtained from a library tape which is written in the ENDF/B format. The CCT library tape is saved between problems. New isotopes can be added to an old CCT tape. Changes to the CCT tape are made using the LIBR code. A list of the isotopes available in XMC is given in Table 3.5-1.

The XMC code consists of three segments or overlays comprised of the input routines, the Monte Carlo routines and the output routines. The input routines use card input and the CCT library tape to prepare the data needed by the Monte Carlo and output routines. The Monte Carlo routines are then loaded and the Monte Carlo calculation performed. The output routines are used to process, print, and/or plot the Monte Carlo output. Statistics are calculated for the various output values.

The next section describes the theory and techniques which are used in the Monte Carlo calculation. Following this, some of the major Monte Carlo routines are described along with the input-output, and the XMC loader routine.

3.5.2 The XMC Monte Carlo Routines

The flow diagram shown in Figure 3.5-2 describes the sequence of events to follow the histories of a "batch" of neutrons. This figure describes the path followed if importance weighting is not being used. The XMC code uses a "beam" type Monte Carlo technique instead of a "particle" method. It starts each track with a beam of neutrons. The beam is used for tallying the flux, leakage, and reaction rates. The beam strength is reduced by the negative exponential of the mean free path traveled between each collision. A beam is followed until it is terminated either by leakage or by Russian Roulette.

The neutron history is initiated by generating a starting location for the neutron beam. The region, the distance to the next boundary, the material type, the mean free path for the material and the region tally number are determined for this location. If a resonance calculation is required the resonance routines are called and the cross section value is added to the total cross section. The distance-to-collision is compared to the number of mean free paths to the boundary. Neutrons are either moved to the collision point or the region boundary which ever is nearer. If a

collision occurs the collision routine checks to see if a fission occurred and stores the fission neutrons in the fission bank. The collision routine uses the probability of nonabsorption to reduce the beam weight and determine if the beam is to be tracked further. It then determines the scattering nuclide and scattering event. A new direction and velocity are calculated and the scattered neutrons are stored. As the Monte Carlo calculation progresses, the flux, leakage, and reaction rates are tallied.

The flow of the calculation is altered slightly if region importance weighting is being used. The main difference is that the calculation-of-distance-to-collision is made after the mean free path is obtained and must be made again in each region.

3.5.3 Treatment of Neutron Absorption

Neutron absorption is accounted for by reducing the importance weighting of a neutron by the absorption probability at each collision. The reduction is done by multiplying the weight before the collision by the non-absorption probability (P_{na}) for the collision. This process can be written as Wt' = Wt x P_{na} . The non-absorption probability can be defined as the total scattering neutron production divided by the total cross section or

$$P_{na} = \frac{\sum_{s} + \sum_{in} + 2\sum_{n-2n}}{\sum_{T}}$$

For a n-2n scattering event the extra neutron is treated as negative absorption. Since the weight will never become zero, the code terminates neutrons using the method previously stated. These are Russion Roulette and "weight ratioing".

Russian Roulette.

Russion Roulette, as the name implies, uses chance to determine if a neutron survives. The Russion Roulette routine is only used if the importance weighting of a neutron has been reduced below the minimum weight. Given a minimum weight (Wt min) and a survival weight (Wt s), Russian Roulette is performed by picking a random number (ξ) between 0 and 1. If the ratio of the neutron weight to the survival weight is greater than the random number, then the neutron is given the survival weight, otherwise the weight is set to zero and the tracking is terminated. This can be written as

if Wt < Wt_{min} and if $\xi \leq \frac{Wt}{Wt_s}$ then set Wt = Wt_s but if $\xi > \frac{Wt}{Wt_s}$ then set Wt = 0 and terminates the neutron.

On the average, with a large number of samples, the weight will be preserved by this method.

Weight Ratioing

For systems which are very thermal it may take a very large number of collisions to terminate neutrons by using Russian Roulette. This

is because the weight reduction at each collision is very small; i.e., P_{na} -1. Often this results in obtaining answers concerning thermal parameters that which are statistically very accurate while the accuracy of the epithermal parameters is very poor. A method called "weight ratioing" was devised to allow control of the time spent in the thermal range compared to the epithermal range.

Weight ratioing uses two flux tally sets: one tally set for neutrons slowing down and one for neutrons that have been slowed down past an entry energy (E_{th}) . The neutrons reaching the thermal tally range (second tally set) can have their energy increased above E_{th} but the results are still tallied into the thermal tally set.

The weight ratio method tallies the results of a neutron history into tally set 1 until it is slowed down past the energy E_{th} . Russian Roulette is then played with each neutron so that R_{th} neutrons entering the thermal tally set are rejected for each one entering the thermal tally range. The ones that survive are followed and the results tallied into a second tally set. The final tallies are the sums of the values from the first tally set plus the values from the second tally set times a weight ratio. This ratio is the weight entering the thermal tally range or approximately $1/(1-R_{th})$. This method statistically conserves the neutron absorption and allows a way of controlling the statistics obtained for epithermal parameters.

3.5.4 The Neutron Flux and the Neutron Beam

The neutron flux at energy E integrated over volume and energy is the total neutron track length in the volume, or

$$\phi_{if} = \int_{V_f} \int_{E_i} \phi(E,V) dEdV = \sum_{n=1}^{N} \ell_n Wt_n$$

where V_f is the volume of region f, ℓ_n is the track length and Wt_n is the weight of the n'th neutron in region f and energy group i. Note that the reaction rate for the k'th event is $R_{i,f,k} = \Sigma_{i,f,k} \phi_{if}$ where $\Sigma_{i,f,k}$ is the macroscopic cross section.

Instead of tallying the track length of individual neutrons the XMC code tallies the estimated track length for a beam of neutrons going in the same direction as the individual neutron. The flux contribution to a region and energy group can be written as

$$\phi_{if} = \sum_{n=1}^{N} \lambda_{if} I_n (1 - \exp(-S_{bn}/\lambda_{if}))$$

where λ_{if} is the mean free path in the i'th energy group of region f, S_{bn} is the distance to the outside boundary of the region for the n'th beam, and 'n is the beam strength at the beginning of the n'th flight of the beam. The beam strength is set equal to the weight of the neutron at the start of the neutron path. It is then diminished by the factor $\exp(-S_{bn}/\lambda_{if})$ at each boundary crossing. Note that the neutron is followed until it has a collision.

or leaks from the cell while the beam is followed even after the collision. If the neutron leaks then the beam is also terminated; otherwise the beam is tracked until terminated by Russian Roulette.

The leakage tallied by the beam crossing a boundary is just the beam strength, I, at the boundary. The leakage is tallied as a function of the broad-energy groups.

The XMC code has provisions for zone importance weighting. Zone importance weighting provides a method for spending more time in regions (zones) of greater importance while reducing the time spent in regions of low importance. When importance weighting is used the number of neutrons in the beam and the weight of each neutron is modified when the beam passes between regions of different importance.

The XMC code can be used in two problem modes. One mode, the fission descendant problem, allows fission neutrons to be born as a result of collisions. These fission progeny are stored and used as starting neutrons for the next batch. The other mode, called the direct source problem, obtains all starting neutrons from the source routine. In this case, there is no need to save the fission neutrons, so the fission reaction is not sampled.

Associated with each neutron is a set of parameters which describe its position, direction, and velocity. The position is described by the position vector x,y,z where the units are in centimeters and an index

which tells which geometrical region the neutron is in. The direction is defined by the three normalized direction cosines α , β , and γ . The velocity is defined by the microscopic energy-velocity group which the velocity falls into. The velocity units are centimeters per micro second.

3.5.5 Energy Group Structures

The energy range for XMC is from 0 to 10 MeV or any part of this energy range. The neutron energy-velocity distribution is continuous in the XMC calculation; however, the energy dependent input is divided into energy groups. There are two different energy group structures; these are the micro-groups, and broad-groups.

The micro-group is the smallest energy group structure. The cross sections and reaction probabilities are group-averaged over each micro-group. The code uses the micro-averaged values for any velocity which falls within the velocity limits of the micro-group. The fluxes are also tallied for each micro-group.

Most calculations with the XMC code use 190 energy groups with 60 groups below 1 ev. Above 1 ev the groups are equally spaced in lethargy.

The broad-groups are the energy groups used for output. The boundaries of the broad-groups must coincide with those of the microgroups but each broad-group may contain one or more micro-groups. The

broad-group structure is usually picked to coincide with a group structure used by typical multi-group reactor cell codes. The leakage and reaction rate tallies are made by broad-group.

3.5.6 Code Check Tallies

A set of values are tallied as the Monte Carlo calculation is being made which serve as code check tallies. They are a set of values which are not needed as output, but serve as a means of checking the performance and characteristics of the Monte Carlo code. The code check tallies are obtained directly from the neutron histories and are not results obtained using the flux estimator. The answers obtained using the code check tallies are usually not as accurate as those obtained using the flux estimators.

The code check tallies include such things as the number of initial fission or source neutrons, the number and weight of neutrons to and from collision, the number of neutrons leaking from the system, the loss of beam strength and weight by application of Russian Roulette, and the number and weight of neutrons entering, and leaving the thermal tally range, etc.

The code check tallies are written out at the completion of each Monte Carlo batch. Average values with associated statistical errors are also written by the output routines.

3.5.7 The Source Routine

The source routine is used to pick the initial parameters for the neutron histories. If the calculation is a fission descendant problem, the source routine will only be used for the first batch and only the

spatial parameters will be picked. For a direct source problem, all the parameters are generated by the source routine.

The source routine in the XMC code is very versatile. It allows the selection of the neutron spatial distribution using combinations of point, equal volume, cosine $\frac{\pi}{2} \frac{r}{R}$, J_o (2.405 r/R), and (sin π r)/ π r distributions. The velocities can be picked using combinations of point, fission, Maxwellian, and/or slowing down distributions. The angular distribution is either isotopic or mono-directional.

3.5.8 The XMC Geometry Routines

The geometry routine determines the region that the neutronbeam is in and calculates the distance to the nearest boundary. This routine also contains the boundary conditions. The routine follows source, fission or collision neutrons and determines if these neutrons collide in the region, leave the region, or encounter a boundary. If a boundary is encountered the neutron can leak or be reflected isotropically or with a mirror image reflection.

There are geometrically eight types of boundarys which can be used. Table 3.5-2 lists these boundary functions.

3.5.9 Path Length Calculation

The distance from one collision to the next collision is determined by randomly sampling from the distribution $P(\lambda)d\lambda = exp(-\lambda) d\lambda$ where λ is the mean free path for the appropriate region and energy.

The mean free path (λ) is stored for each material and micro group. If the material contains resonances and the energy is in the resonance region, the values of the resonance cross sections are calculated. The resonance cross sections are then combined with the micro group cross sections to obtain the mean free path. The equations explicitly include the Doppler broadening of both absorption and scattering. The values of the cross sections are calculated at the neutron energy on a point basis (ie, not at the micro-group energy).

In the unresolved resonance region, the cross section for a given energy is obtained from the contribution of the two nearest resonances. In this region the value of Γ_n is found from a Porter-Thomas distribution while Γ_f is taken from an exponential distribution. The nearest resonances to a given energy are found from the level spacing.

3.5.10 The XMC Collision Routines

When a collision occurs the code considers the following events: absorption, fission, inelastic scattering, and elastic scattering. The elastic scattering can be treated as isotropic or anisotropic slowing down scattering. Thermal scattering is treated by ENDF/B kernels.

Neutron Absorption and Fission

XMC first reduces the neutron weight to account for absorption. Then the code checks to see if a scattered neutron is to be produced. The fission probability is checked to see if any fission neutrons were produced. N fission neutrons are produced if wt.P_f+ $\xi \ge N$ where ξ is a

random number and P_f is the ratio of the neutron fission production rate (νz_f) to the total reaction rate (z_T) in the material. Note that N will seldom be greater than 1. The fission neutrons are stored in the fission bank.

Selection of Scatterer

If the neutron survived absorption then the scattering isotope is selected. To save computer space and time use is made of a "heavy scatterer." More than one isotope in a material can be specified as heavy isotopes. All of the heavy isotopes elastic scattering cross sections are combined to form one elastic scatterer. A mass for the heavy scatterer is also specified. Scattering from a heavy scatterer is treated as if the scattering mass were infinite (no energy change) unless the neutron is in a region having an isotope with resonance parameters and the neutron has a velocity such that a resonance calculation was made. Then the mass of the special heavy scatterer will be used.

The scatterer is picked by selecting a random number ξ and comparing it to the cumulative scattering probabilities.

Inelastic Scattering

Inelastic scattering is treated in two ways and the n-2n scattering is combined with it. The two models used for inelastic scattering are discrete level energy loss and the evaporation model. Both models assume that scattering is isotropic in the center of mass system.

Inelastic Spectrum Table

The inelastic spectrum table (discrete energy loss) uses a table of the inelastic excitation level energies along with a probability for each one. If the neutron energy is below a certain excitation energy that level can not be excited. An approximation is made in that it is assumed that cross section for excitation of a certain level is a constant in energy above the level energy.

Evaporation Model

The evaporation model is used to select the emergent energy from the Maxwellian distribution

$$P(E^{-}) = \frac{E^{-}}{T_{n}^{2}} e^{-E^{-}/T_{n}}$$

Actually $(V^{-})^2 = 1.91322 \text{ E}^{-1}$ is selected. T_n is the nuclear temperature and is calculated by $T_n = a_0 + a_1 V + a_2 V^2 + a_3 V^3$. The coefficients a_0 , a_1 , a_2 , a_3 are fitted by least squares to the data on the ENDF/B library for T_n . The new velocity is then calculated as

$$V^{-} = \left[T_{n}(\phi_{1} + \phi_{2}) \right]^{1/2}$$

where ϕ_1 and ϕ_2 are random numbers selected from an exponential distribution. It is also required that V⁻ \leq V. The Maxwellian is sampled correctly as shown in the equation

$$P(x) = \int_{0}^{x} e^{-y} e^{-(x-y)} dy = xe^{-x}.$$

122

XN-NF-80-19(NP)(A) Vol. 1

It is assumed that the evaporation model is used only for heavy nuclei and so the conversion from the C.M. to the lab system is not made.

Nonthermal Elastic Scattering

Nonthermal elastic scattering can be treated as isotropic or anisotropic. If the scattering is isotropic, a random set of direction cosines are used as the new direction cosines in the center of mass. For anisotropic scattering the cosine of the scattering angle, μ is selected by making use of the sample rejection technique.

Once the new scattering cosine μ is selected, the new velocity and the direction cosines are selected. If the isotope is a heavy scatterer the velocity is not changed.

Thermal Elastic Scattering

The thermal elastic scattering cross sections are processed from the ENDF/D data files using the FLANGE⁽³⁻⁴⁾ code. In particular, data for neutron scattering by hydrogen in water has been tabulated at temperatures from 273° K to 800° K based on the Haywood model of the phonon spectrum. The Haywood model takes into account the effects of the vibrational and rotational modes of hydrogen atoms bound in the water molecule.

The scattering kernel calculated from $S(\alpha, \beta, T)$ is too large to use directly in the Monte Carlo calculation. Thus, the scattering kernel is divided into downscattering and upscattering components. Then, the two dimensional array based on $S(\alpha, \beta)$ is used to determine the scattering angle. The upscattering, downscattering, and angular components were

calculated at a number of points using a modified version of FLANGE. Data points were then picked so that an accurate linear interpolation could be used. The base data was calculated at 276^OK. Correction factors are used for higher temperatures. The FLANGE calculations were made at the ENDF/B data temperatures.

3.5.11 The XMC Output Routines

The XMC output routines prepare the output and print it and/or plot it. The two main tasks in preparing the output are preparing averaged values of fluxes, cross sections, reaction rates, and leakages along with some other cell parameters and obtaining statistics for the values which are calculated.

Statistics

Statistics are obtained by processing a series of equal sized batches and uveraging the results for the batches. It is possible to leave some of the first batches out of the average. This might be done to damp out the effects of the source distribution. There are two common types of average values obtained from the Monte Carlo calculation. One type is a direct answer such as flux or reaction rate and the other type is a ratio such as flux averaged cross sections. The statistical error for a single averaged value \overline{a} is $S_{\overline{a}}$ where $S_{\overline{a}}$ is the error for one standard deviation. Let N be the number of equal sized batches and a_n be the value of a for the n^{th} batch. The values of \overline{a} and $S_{\overline{a}}$ then become:

$$\overline{a} = \frac{1}{N} \sum_{n=1}^{N} a_n$$

and

$$S_{\overline{a}} = \left[\left\{ \sum_{n=1}^{N} a_n^2 - N(\overline{a})^2 \right\} / N(N-1) \right]^{1/2}$$
$$= \left[\left\{ \sum_{n=1}^{N} a_n^2 - \left(\sum_{n=1}^{N} a_n \right)^2 / N \right\} / N(N-1) \right]^{1/2}$$

The statistical error for a ratio R is

$$S_{R} = \begin{bmatrix} N & N \\ \sum_{n=1}^{N} a_{n}^{2} - 2R \sum_{n=1}^{N} a_{n}b_{n} + R^{2} \sum_{n=1}^{N} b_{n}^{2} \end{bmatrix} \begin{bmatrix} 1/2 & N \\ / \sum_{n=1}^{N} b_{n}, \\ n=1 \end{bmatrix}$$

where

$$R = \sum_{n=1}^{N} a_n / \sum_{n=1}^{N} b_n.$$

Note that for a ratio the values of a and of b may have large statistical errors and the statistical error of the ratio can be very small or even zero. This is accounted for by the correlation term

$$(-2R \sum_{n=1}^{N} a_n b_n).$$

The statistical error is written out in a way which reduces the printing space. The statistical error is forced to have the same exponent as the value. If a value and its error were $1.0685 \times 10^4 \pm 5.2 \times 10^2$ then the numbers would be printed out as $1.0685+04 \ 0.052$.

Broad Energy Averaging Groups

The output routines tally over broad groups. These broad group energy boundaries must coincide with the micro-group boundaries. However, the broad groups do not need to cover the entire energy range or to have joining boundaries. In fact, the boundaries can overlap.

The broad groups are used for flux and reaction rate integrals. These broad group fluxes are also used to obtain statistics on the ratios between different energy broad groups or between regions for a given broad group.

Printed Output

The output code prints the results of the Monte Carlo calculation. First the averaged code check tallies are printed. Next the broad-group fluxes and leakages are printed. Then, the broad-group average cross sections and reaction rates for each isotope in each region are printed along with the reaction rates for the region. The reaction rates for the entire cell along with an infinite and effective multiplication constant follow. Finally the micro-group fluxes for each region are printed and/or plotted.

3.5.12 Cross Section Library

The cross sections in the current XMC library were derived from the Battelle Northwest Master Library (BNML); except for the hydrogen thermal scattering kernal (which is from ENDF/B).

The resonance cross sections can either be group averaged and placed in with the smooth cross sections or they can be placed in a resonance parameter table with a "floor" being placed in with the smooth cross sections. The "floor" accounts for the resonances and resonances contributions not included in the resonance parameter table. The resonance parameter table contains resolved and unresolved resonance parameters.

The resonance parameter table preparation routine determines what energy width around each resonance will be accounted for by the resonance parameter table. The resonance contribution outside of this energy width will be added to the smooth cross sections as a "floor". The potential scattering cross section is also included in the "floor".

The resolved resonance contribution to the average group cross section is determined in one of two ways. If the resonance energy is more than 250 half widths outside the group, the contribution is integrated analytically. Otherwise, the resonance contribution is calculated for a number of points and numerical integration is made.

The analytical integration* assumes that the flux is 1/E and that the resonance equations are not Doppler broadened.

The numerical integration is made by calculating the Doppler broadened resonance cross sections at a number of points using the resonances within 250 half widths of the micro-group boundaries. The point cross sections are then multiplied by the correct weighting function and numerically integrated.

*Derived by Dr. J. L. Carter, Jr., while at Battelle-Northwest Laboratories.

The points are selected so that they are optimumly placed for an integration of $(1 + X^2)^{-1}$ using logarithmic interpolation. Additional points are then added as needed so that the maximum energy spread between points is less than $\Delta Ei/Np$ where Np = A + B.Emin_i. The i is the group number and A and B are input numbers.

The weighting functions which are used are either 1/E or $1/E \cdot \sigma_T(E)$ where $\sigma_T(E) = \sigma_{Tp}(E) + \sigma_{Tin} + \sigma_{T_A}$ and $\sigma_{T_p}(E)$ is the total cross section at each point. $\sigma_{T_{III}}$ is the total cross section of the other isotopes in the material per atom of this isotope (an input number), and σ_{T_A} is the analytical contribution from the other resonances of this isotope. The point values are multiplied by unity or $1/\sigma_T(E)$ and the 1/E weighting is accounted for in the numerical integration. The integration assumes that the logarithm of the cross section varies linearly with the logarithm of the energy.

The unresolved resonance contribution is caluclated at a number of points using the equations as derived in the $ETOE^{(3-24)}$ and $MC2^{(3-25)}$ codes. The point values are then integrated numerically using linear log-log interpolation and assuming a 1/E flux. The point calculations account for Doppler broadening and use the narrow resonance approximation. The scattering from other isotopes in the material is again an input number.

The XMCLIB code first processes the anisotropic scattering data into group averaged Legendre coefficients in the center of mass system. The a's as used by the XMC code are then obtained by matrix multiplication

and normalized. The normalization is the maximum value of P(u) for $-1 \le u \le 1$ and is found by examining P(u) at 101 equally spaced values of u. The a's are also adjusted so that P(u) is never negative and so that the number of random selections of u per acceptance will not average more than 20.

The XMC code uses the same energy distribution data for inelastic and n-2n scattering. However, the information for both inelastic and n-2n scattering are placed on the CCT tape by the XMCLIB code.

It is assumed that the energy distribution from inelastic scattering will be of two forms. The two forms are discrete excitation level and the evaporation model. One isotope can have both forms. The evaporation model is used above the energy range covered by the discrete energy level model.

The temperature coefficients for the evaporation model are calculated by fitting a cubic in velocity to the ENDF/B library values. The probabiltiy of exciting a discrete level is calculated by the XMCLIB code by integrating over energy the probability times the product of the inelastic scattering cross section and the flux weighting function.

3.5.13 The XMC Loader

In order to simplify the data preparation for XMC, a special input routine was written. This routine, called the XMC LOADER, requires less than a dozen cards to prepare all the input to run an XMC case. The use of the XMC LOADER greatly reduces the time to prepare input and the chances of user errors.

The LOADER will prepare input for controlled and uncontrolled BWR fuel bundles, as well as PWR bundles. Additionally, several types of pin-cell geometries are treated by the LOADER.

The output of the LOADER is a file or card deck from which XMC runs. Since XMC plots the geometry of the problem run, a comparison to the actual desired input is easily made.

Some of the available geometry options are shown in Figures 3.5-3, 3.5-4, 3.5-5, and 3.5-6. Figures 3.5-3 and 3.5-4 show BWR and PWR fuel bundles, while Figure 3.5-5 shows some of the pin-cell "box types". These box types may also be placed in a bundle as shown in Figure 3.5-6.

Table 3.5-1 Isotopes in the XMC Cross Section Library

Hydrogen (in water)	Zirconium
Boron (natural)	Silver-107
Boron-10	Silver-109
Boron-11	Cadmium-112
B ₄ C	Indium-115
Carbon	Gadolinium-155
Nitrogen	Gadolinium-157
Oxygen	Uranium-235
Aluminum	Uranium-236
Silicon	Uranium-238
Chromium	Plutonium-238
Manganese	Plutonium-239
Iron	Plutonium-240
Nickel	Plutonium-241
304 Stainless Steel	Plutonium-242

Cobalt

Table 3.5-2 XMC Code Boundary Functions

Boundary Function

 $r = x - x_{0}$ $r = y - y_{0}$ $r = z - z_{0}$ r = y - ax - b $r = x^{2} + y^{2} - R^{2}$ $r = (x - x_{0})^{2} + (y - y_{0})^{2} - R^{2}$ $r = x^{2} + y^{2} + z^{2} - R^{2}$ $r = A(x - x_{0})^{2} + B(y - y_{0})^{2} + C(z - z_{0})^{2} - K$

Description Plane at X = x_0 Plane at y = y_0 Plane at z = z_0 Plane on y = ax + b Cylinder of radius R centered on z axis Cylinder of radius R centered at x_0, y_0

Sphere of radius R centered at the origin

General boundary function



Figure 3.5-1 The XMC Code Package



Figure 3.5-2 General Flow Diagram of the Monte Carlo Code for Neutrons Importance Weighting not Being Used



Figure 3.5-3 BWR Fuel Assembly Geometry

* Rod position numbers



Figure 3.5-4 BNR Quarter Bundle Geometry

* Rod position numbers

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Figure 3.5-6 Examples of Boxes of Type 1 at Rod Position 1, 11, and 28, and a Box of Type 2 at Rod Position 15. Note that Insertion of a Box at an Off Diagonal Position Actually Inserts Two Boxes such as the Box at Rod Position 11.

*Rod position number XN-NF-80-19(NP)(A) Vol. 1

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4.0 NEUTRONICS CORE ANALYSIS METHODOLOGY

This section covers the special methods for analyzing the control rod drop accident, the fuel misloading incident, reactor core and channel hydrodynamic stability, the control rod withdrawal incident, and the methods for calculating the neutronics parameters which are input to the plant transient and loss of coolant accident analyses.

4.1 CONTROL ROD DROP ACCIDENT

The control rod drop accident assumes a control rod becomes uncoupled from the drive and remains stuck fully inserted in the reactor core as the control rod drive is withdrawn. The uncoupled control rod is then assumed to drop out of the core.

The primary reactivity feedback mechanism that limits the power during a rod drop accident is the Doppler reactivity. The control rod scram ensures a final reactor shutdown state.

The limiting criteria for the control rod drop accident analysis are the following:

- The maximum deposited enthalpy during the accident in a fuel rod at any axial location shall not exceed 280 calories per gram.
- The maximum reactor pressure during the accident shall not cause reactor pressure vessel stresses to exceed the "Service Limit C" as defined in the ASME Code. (4-1)

XN-NF-80-19(NP) (A) Vol. 1

The reactor neutronic parameters which significantly affect the rod drop analysis include the Doppler reactivity coefficient, the maximum control rod worth, the power peaking (peaking with control rod removed from core) and the delayed neutron fraction. For a given type of reactor, the maximum deposited enthalpy is parameterized as a function of the above variables.

For maximum deposited enthalpies less than 280 calories per gram, prompt fuel rupture does not occur and the heat transfer from the fuel to the coolant is by convection. The reactor coolant pressure is calculated for limiting values of Doppler, rod worth, power peaking, delayed neutron fraction, and scram bank reactivity worth.

Rod Drop Analysis Method - The rod drop calculations are performed with the COTRAN computer code described in Section 3.3. The COTRAN code solves the space and time dependent neutron diffusion equation in twodimensional (r-z) geometry with fuel temperature and moderator density reactivity feedbacks. COTRAN employs a nodal method based directly on a one-group finite difference technique for the solution of the time dependent neutron diffusion equation. The one-group cross-sections used in the iterative flux solution are determined from input two-group values and modified at each time step by thermal feedback. The input two-group cross sections for COTRAN are calculated using the XTGBWR code following the procedure outlined in Section 3.3.

XN-NF-80-19(NP) (A) Vol. 1

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The space and time dependent COTRAN neutronic model is capable of computing a rapid reactor transient initiated by a reactivity insertion caused by a control rod being removed from the core. Since the model utilizes the two-dimensional (r-z) geometry, the code can calculate the rapidly changing flux distribution as a control rod leaves the core and the scram rod bank simultaneouly enters the core.

COTRAN initially determines the static flux and power distribution corresponding to the problem input. The initial time step for the rod drop analysis is 0.0001 seconds. The code then automatically determines the time step interval based on the number of iterations necessary to achieve convergence. This method permits small time steps during times of large changes in power level, and inversely, large time steps during periods of slow perturbation. Therefore, the code efficiently solves the transient problems without the user choosing time step sizes. Six groups of delayed neutron precursors are employed in the transient analysis.

The following is a step-by-step description of the procedure employed to perform the control rod drop accident analysis.

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4.2 FUEL MISLOADING ANALYSIS

At the present time two separate incidents are analyzed as part of the fuel misloading analysis. The first incident which is termed the fuel misorientation error assumes that a fuel assembly is misoriented by rotation

XN-NF-80-19(NP) (A) Vol. 1

through 90° or 180° from the correct orientation when loaded into the reactor core. The second incident, the fuel mislocation error, assumes a fuel assembly is placed in the wrong core location during refueling. For both the fuel misorientation error and the fuel mislocation error, the assumption is made that the error is not discovered during the core verification and the reactor is operated during the cycle with a fuel assembly misloaded.

The limiting parameter of interest for the fuel misloading error is the MCPR in the misloaded fuel assembly. The fuel misloading analysis determines the difference between the MCPR for the correctly loaded core and the MCPR for the core with a fuel assembly misloaded. The resulting Δ MCPR for the misloading error is then compared with the Δ MCPR determined from the transient analysis for the cycle. The largest Δ MCPR is then added to the transient MCPR safety limity to determine the operating MCPR limit.

4.2.1 Fuel Misorientation Error

For the fuel misorientation error analysis, a limiting fuel assembly in the reactor core is assumed to be rotated 90° or 180° from the normal orientation. The fuel misorientation error is important for the fuel assemblies in the BWR/2, BWR/3 and BWR/4 reactor cores. In these cores the fuel assemblies are offset in the core lattice to provided a wider gap between the fuel channels where the control rods are inserted. To account for the moderating effects of the water in the wider gap, the fuel assemblies are designed with lower enrichment fuel rods next to the wide water gaps.

XN-NF-80-19(NP)(A) Vol. 1

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If a fuel assembly is charged into the reactor core in the rotated orientation, higher enrichment fuel rod powers higher than design values. The misoriented analysis is performed for the most severe case of the MCPR limiting fuel assembly being rotated 180 degrees. Four bundle calculations are performed to demonstrate that the 180 degree rotation is the most limiting case. The procedure used to calculate the power in a misorientated fuel assembly and the resulting MCPR consists of the following five steps:

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4.2.2 Fuel Mislocation Error

For the fuel mislocation error an incorrect fuel assembly is assumed to be loaded in one of the core locations during refueling. The limiting case is the loading of a high reactivity fuel assembly in place of a low reactivity fuel assembly. A misloading error in one of the inner core modules will produce a localized area of higher reactivity resulting in higher power and lower CPR than planned. The following procedure is used to find the lowest CPR that would result from a fuel mislocation. The CPR for the misloaded assembly compared to the CPR for the core with no assembly misloaded gives the AMCPR for the mislocation error.

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When the reactor is operating, the ENC core monitoring method will use all measured data including the local power range monitor (LPRM) data to determine the power of each node of fuel in the core. If a

XN-NF-80-19(NP)(A) Vol. 1

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high reactivity fuel assembly is mislocated in the core, the misloaded assembly will cause that area of the core to be higher in power than planned. The higher power will be detected by the LPRM detectors resulting in the measured operating MCPR being lower than the operating MCPR predicted by the XTGBWR core simulator code in the above procedure. The more accurate fuel mislocation analysis that will be performed when required includes the following additional calculations:

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4.3 STABILITY ANALYSIS

Stability can be defined for an operating system as follows: a system is stable if, following an input perturbation, the transient returns to a steady, non-cyclic state. Stability analysis is concerned with two basic phenomenon, reactor core (reactivity) stability and channel hydrodynamic stability. Reactor core instability is when the reactivity feedback of the entire core drives the reactor into power oscillations. Channel hydrodynamic instability is flow oscillations which may impede heat transfer to the moderator resulting in localized power osciallations. Stability is analytically demonstrated if no divergent oscillations develop as a result of perturbations of any critical variable, such as core pressure, control rod position, and recirculation flow.

The criterion to be evaluated is the decay ratio X_2/X_0 , designated as the ratio of the magnitude of the second over shoot to the first overshoot resulting from a step perturbation. For a time domain analysis, the

decay ratio is determined from the time response containing several oscillations by averaging the decay ratios determined from each successive overshoot. When the decay ratio X_2/X_0 is less than 1.0, the reactor core is stable. Thus, the ultimate performance criteria for the stability analysis is specified in terms of the decay ratio as:

Reactor Core (reactivity) stability $x_2/X_0 < 1.0$

Channel hydrodynamic stability $X_2/X_0 < 1.0$

These criteria are demonstrated for all usual and unusal operating conditions of the reactor that may occur during the course of the fuel in-core lifetime. For stability purposes, the most severe conditions to which these conditions will be applied are:

> Natural circulation flow at a power corresponding to the rod block power limit condition, and

2. End of cycle power distributions at low power operation.

Although the ultimate performance criteria ensure absolute reactor stability, an operational design guide is applied for all expected power and flow conditions encountered in normal operation. The most limiting condition expected corresponds to minimum normal flow.

Stability analysis is performed with the COTRAN computer code described in Section 3.3. The COTRAN code solves the space and time dependent neutron diffusion equation in two-dimensional (r-z) geometry with fuel temperature and moderator density reactivity feedback. These reactivity

feedbacks are determined from a solution of the equations of mass, energy and momentum for the hydrodynamic channels coupled with a fuel conduction model. As such, the COTRAN code provides the time response of important core parameters following a system perturbation. The calculational method for the reactor core (reactivity) stability analysis is as follows:

The hydrodynamic and core reactivity decay ratios determined by the above procedures are then compared to the operational design criteria. If the criteria is met for all usual and unusual operating conditions of the reactor that may occur during the reload cycle then the core is stable.

The neutronics models used in the plant transient and loss of coolant analyses require several neutronic input parameters which characterize the reactor core at a particular operating state. These parameters

1. Void reactivity coefficient,

4.4 NEUTRONIC REACTIVITY PARAMETERS

- 2. Doppler reactivity coefficient,
- Scram reactivity,

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are:

- 4. Delayed neutron fraction, and
- 5. Prompt neutron lifetime.

148

XN-NF-80-19(NP)(A) Vol. 1

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These parameters which vary with cycle exposure and core average void fraction are determined for each reactor condition to be analyzed. The neutronic parameters and their calculational methodology are discussed in the following sub-sections.

4.4.1 Void Reactivity Coefficient

The void coefficient of reactivity is the fractional change ore reactivity produced by a change in the core average void fraction. The void coefficient is dependent on the specific operating state and core average void level.

The void coefficient of reactivity is calculated with the reactor simulator code, XTGEWR. The calculational method for an operating state is as follows:

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4.4.2 Doppler Reactivity Coefficient

The Doppler coefficient of reactivity is the fractional change in core reactivity produced by a change in the core average fuel temperature. The Doppler coefficients for each fuel type in the core are determined with the XFYRE computer code described in Section 3.1. The calculational procedure for determining the core average Doppler coefficient is as follows:

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The scram reactivity is defined as the core reactivity change as a function of the scram bank insertion. The total scram reactivity is calculated with the reactor kinetics model COTRAN as follows:

4.4.4 Delayed Neutron Fraction

The delayed neutron fraction is calculated for each fuel type by the XFYRE code described in Section 3.1 as a function of exposure. For the plant transient analysis, a core average delayed neutron fraction (β_{eff}) is determined by exposure and volume weighting the fuel type dependent delayed neutron fraction.

4.4.5 Prompt Neutron Lifetime

The prompt neutron lifetime is calculated with the XFYRE code for each fuel type in the core. The calculations are performed at core average voids as a function of exposure. The core average prompt neutron lifetime is calculated by exposure and volume weighting the fuel type dependent neutron lifetimes.

4.5 CONTROL ROD WITHDRAWAL

The control rod withdrawal error is the withdrawal of a control rod by the reactor operator from a fully inserted position until the control rod motion is stopped by the rod block. For the analysis, the reactor is assumed to be in a normal mode of operation with the control rods being withdrawn in the proper sequence and all reactor parameters within the

150

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XN-NF-80-19(NP)(A)

Vol. 1

Technical Specification limits and requirements. The most limiting case is when the reactor is operating at power with a high reactivity worth control rod fully inserted. To maximize the worth of the control rod, the reactor is assumed to be xenon free and the control rod with the maximum rod worth is selected as the rod to be withdrawn. When necessary, the partially withdrawn control rods in the core are adjusted slightly to place the fuel near the inserted control rod on thermal limits.

During the control rod withdrawal transient the reactor operator is assumed to ignore the local power range monitor (LPRM) alarms and the rod block monitor (RBM) alarms and continue to withdraw the control rod until the control rod motion is stopped by the control rod block.

While the control rod is being withdrawn, the reactor power and the local power in the area of the rod which is being withdrawn will increase. The reactor thermal limit of concern as the power increases is the transient minimum critical power ratio (MCPR) limit which protects against critical heat flux. The control rod withdrawal analysis will determine the Δ MCPR during the transient as a function of the rod block setpoint. The Δ MCPR values for the control rod withdrawal are compared to the Δ MCPR values for the other transients to determine the operating MCPR limit and rod block set point that will protect the MCPR safety limit of the reactor.

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The control rod withdrawal incident is analyzed as a series of steady state calculations since the rate of power increase is slow compared to the time constants for heat transfer and delayed neutrons. The calculations are performed with Exxon Nuclear Company's reactor core simulator code, XTGBWR. The calculational method for the control rod withdrawal is as follows:

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4.6 REFERENCE

4-1 ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components", American Society of Mechanical Engineers.







Figure 4.1-2 Typical Control Fractions a1 vs a2 for Central Rod Full In or Full Out



Figure 4.1-3 Typical Scram Bank Reactivity Worth Curve





Figure 4.2-1 Four Bundle Module with Misorientated Fuel Assembly, 180° Rotation



Figure 4.2-2 Elevation View of Misorientated Fuel Assembly

158

XN-NF-80-19(NP)(A) Vol. 1

5.0 NEUTRONICS METHODS VERIFICATION

The ENC neutronics methods are verified by comparing calculations to measured reactor data and to calculations made by higher order methods. Methods verification for the XFYRE, XTGBWR, and COTRAN codes is presented in this section.

5.1 XFYRE VERIFICATION

The local power distributions calculated by the XFYRE code are verified by comparison to fuel rod gamma scan measurements. Comparisons of the calculated and measured local power distributions are shown in Figure 5.1-1 through 5.1-5. The measurements were performed by removing the fuel rods from the fuel assembly and measuring the La-140 activity at a given core height. The measured data shown in Figures 5.1-1 through 5.1-5 have not been corrected for core flux tilt effects caused by control rod and fuel exposure.

In addition to comparison to gamma scan results, the accuracy of the XFYRE calculational model for the microscopic depletion of a BWR fuel assembly has been verified by comparison with measured isotopics from a Garigliano fuel assembly (5-1). Figure 5.1-6 is a representation of the Garigliano BWR fuel assembly for which the measurements were made. Table 5.1-1 shows the comparison between the measured and XFYRE calculated isotopics.

The XFYRE code has been benchmarked with the higher order XMC code described in Section 3.5. A series of XMC calculations have been performed to evaluate the effects of in-channel voids, control, gadolinia, and temperature on the BWR fuel assembly reactivity and local power distribution. A comparison of k_{∞} calculated with the XMC and XFYRE codes is shown in Table 5.1-2. The local power distributions calculated with the XMC and XFYRE codes are shown in Figures 5.1-7 though 5.1-11.

The XFYRE calculations were performed with the standard x-y geometry described in Section 3.1. The XMC calculations were performed using an exact geometrical representation for the fuel, clad, channel, and control rod blade. The cross section library was identical for the XFYRE and XMC calculations.

5.2 XTGBWR VERIFICATION

The XTGBWR reactor core simulator code is verified by comparing the calculated and measured reactor parameters. The reactor core follow data for the Oyster Creek, Dresden-3, and Quad Cities reactors are listed in Tables 5.2-1 through 5.2-6. The k_{eff} values calculated by XTGBWR for the critical reactor condition are plotted as a function of cycle exposure on Figure 5.2-1. The k_{eff} data are corrected for known reactivity biases including the effects of "crud", incore instruments, sources, and fuel assembly spacers.

XN-NF-80-19(NP)(A) Vol. 1

A comparison of measured and calculated traveling in-core probe (TIP) data for the Oyster Creek and Dresden-3 reactors is presented in Figures 5.2-2 through 5.2-12. All XTGBWR calculations were performed with a full core model using 24 axial nodes.

The fuel assembly gamma scan measurements made at the Quad Cities 1 reactor at the end of Cycle $2^{(5-2)}$ are compared to the XTGBWR calculation results in Figure 5.2-13. The measured data are La-140 activity which is proportional to the power generation in the last few weeks of the reactor operation. The calculated La-140 activity is determined from the XTGBWR power distribution.

5.3 COTRAN VERIFICATION

The reactor kinetics calculations performed by the COTRAN code are compared to the Peach Bottom-2 transient measurements. (5-3) A comparison of the measured and calculated relative power response for the periodic step change in the pressure regulator setpoint is shown in Figure 5.3-1. The measured and calculated data for the random pressure regulator setpoints changes are shown in Figure 5.3-2. For both of the comparisons, the measured reactor pressure response was input into the COTRAN calculations as a forcing function.

Rod Position	1 1A	1 2B	3C	40	5E	7G	8H	93	9A	
	10355	10060	8939	8653	8736	10309	12424	14180	13785	
EXPOSURE MWD/MTU	10395	9807	8699	8344	8471	10337	12741	14336	13615	
	7.67	12.16	13.31	13.16	13.18	11.84	10.23	5.34	5.48	
U-235 Kg/MTU	7.93	12.50	13.38	13.67	13.58	12.15	10.34	5.55	5.92	
	1.60	1.87	1.67	1.71	1.63	1.81	1.98	1.89	1.82	
U-236 Kg/MTU	1.38	1.53	1.41	1.38	1.39	1.59	1.83	1.70	1.65	
	974.1	969.9	970.0	970.4	970.4	969.6	968.8	971.6	972.2	16
U-238 Ka/MTU	975.0	970.7	970.9	970.8	970.6	. 969.9	969.1	972.4	973.0	N
	3.741	3.874	4.167	4.200	4.240	4.186	3.872 •	3.534	3.504	
Pu-239 Kg/MTU	3.248	3.674	4.040	4.25?	4.341	4.052	3.605	3.082	· 3.090	
	1.127	.887	.814	.771	.777	.949	1.143	1.487	1.432	
Pu-240 Kg/MTU	1.213	.931	.812	.767	.775	.967	1.181	1.567	1.505	Vo.
	.445	. 355	. 340	.335	.338	. 394	.442	.591	.557	1. 1
Pu-241 Kg/MTU	.365	. 325	.289	.231	.291	.371	.462	.540	.507	30-1
	.088	.050	.039	.037	.036	.055	.064	. 185	. 168	9(NP
Pu-242 Kg/MTU	.077	.047	.033	.029	.031	.053	.094	. 181	• 158)(A)

Garigliano Isotopic Comparison Measured/Calculated Data Table 5.1-1

Table 5.1-2 XMC (Monte Carlo)/XFYRE K∞ Comparisons for BWR Reload Fuel Assemblies

Case	K∞ XMC	K∞ XFYRE
OC 0% V, No Gd, No Control	1.3073 ± .0018	1.305
OC 32% V, No Gd, No Control	1.2889 ± .0018	1.289
OC 32% V, No Gd, Control	0.970 + .0024	0.970
OC 32% V, 1.0 w/o Gd ₂ 0 ₃ , No Control	1.2111 <u>+</u> .0010	1.212
OC 64% V, No Gd, No Control	1.2590 ± .0017	1.256
OC Cold, No Gd, No Control	1.289 + .0023	1.296
OC Cold, No Gd, Control	1.100 <u>+</u> .0030	1.108

Cycle Exposure MWD/MTU	Keff	Average Voids	Power MWt	10 ⁶ Flow
195.6	1.003	.33	2339	97.8
663.2	1.003	.37	2400	94.4
782.9	1.002	.37	2475	97.8
1660.0	1.002	.34	2386	98.2
2377.6	1.002	.35	2328	98.0
2599.6	0.999	.38	1772	66.7
3028.7	1.001	.36	2444	96.9
3256.2	1.002	.35	2413	97.7
3632.7	1.003	.34	2306	97.5
3968.7	1.003	.34	2317	98.0
4289.8	1.004	.34	2277	97.7
4716.8	1.004	.31	2100	97.6
5068.4	1.004	.30	1948	97.6
5597.3	1.004	.26	1758	98.0
5930.2	1.004	.23	1602	97.7
6294.8	1.004	.21	1449	97.7
6634.9	1.000	.21	1449	97.7

Table 5.2-1 XTGBWR Calculated K and Average Voids for Dresden-3 Cycle 5

XN-NF-80-19(NP)(A) Vol. 1

Table 5.2-2 XTGBWR Calculated K and Average Voids for Dresden-3 Cycle 6

Cycle Exposure MWD/MTU	K_eff	Average Voids	Power MWt	10 ⁶ Flow
MWD/MTU 250.9 330.3 550.8 719.1 876.3 993.6 1432.5 1674.0 1882.5 2221.2 2480.8 2825.6 3177.6 3275.2 3335.9 3657.1 3843.5 4216.8 4583.7 4874.1 5080.9 5305.1 5600.0 5987.3 6222.1	reff 1.006 1.007 1.004 1.005 1.005 1.005 1.005 1.003 1.003 1.004 1.003 1.003 1.004 1.004 1.004 1.004 1.006 1.006 1.006 1.006 1.006 1.006 1.006 1.004 1.004 1.004 1.004 1.004 1.004	Voids .35 .34 .33 .36 .35 .35 .35 .34 .35 .34 .35 .34 .36 .35 .34 .36 .35 .38 .42 .36 .34 .35 .35 .34 .35 .35 .34 .35 .34 .35 .34 .35 .34 .35 .34 .35 .34 .35 .34 .35 .34 .35 .34 .34 .35 .34 .34 .34 .31 .34 .32 .31	MWt 2218 2382 2445 2103 2364 2418 2409 2408 2471 2458 2491 2338 2292 1894 2423 2376 2412 2450 2304 1903 1892 1877 1744 1808 1706	10° 1b/hr 83.3 93.6 97.3 74.6 91.8 95.3 97.0 94.5 95.3 97.7 97.9 98.0 85.3 63.1 95.5 95.4 98.1 97.8 94.0 67.6 75.4 83.7 65.5 65.3 71.4
6481.6 6732.8 6923.4 7142.8	1.003 1.004 1.005 1.002	.30 .27 .25 .25	1701 1717 1718 1718	72.7 84.0 94.3 94.3

Exposure MWD/MTU	_K _{eff}	Average Voids	Power MWt	10 ⁶ Flow
121.3	1.005	.34	1766	56.4
293.0	1.006	.35	1752	55.7
565.9	1.005	.35	1877	55.8
869.0	1.004	.36	1828	51.5
939.8	1.003	.36	1795	51.8
1232.8	1.002	.36	1817	51.8
1536.0	1.002	.33	1787	52.2
1889.6	1.001	.35	1878	54.9
2401.2	1.001	.35	1833	53.4
2233.2	1.003	.35	1883	58.3
2536.4	1.002	.36	1867	55.9
2890.0	1.005	.32	1856	59.9
3021.4	1.005	.32	1892	60.5
3304.3	1.004	.34	1893	57.2
3668.1	1.005	.36	1893	57.6
3961.2	1.006	.33	1890	60.8
4254.2	1.006	.33	1892	60.6
4628.1	1.005	.37	1820	53.5
4890.8	1.006	.35	1887	61.0
5103.0	1.007	.35	1817	59.5
5244.5	1.005	.36	1781	61.0
5668.9	1.004	.35	1667	61.0
5931.6	1.005	.34	1594	61.0
6224.7	1.005	.34	1481	61.0
6558.1	1.006	.29	1372	61.0
6861.3	1.006	.27	1281	61.0

Table 5.2-3 XTGBWR Calculated K and Average Voids for Oyster Creek Cycle 7

Table 5.2-4 XTGBWR Calculated K and Average Voids for Oyster Creek Cycle 8

Cycle Exposure MWD/MTU	K_eff	Average Voids	Power MWt	10 ⁶ Flow
262.7	1.001	.36	1926	60.2
495.1	0.999	.38	1929	57.9
747.8	0.999	.38	1917	58.7
1253.0	0.997	.37	1914	59.5
1525.9	0.997	.38	1918	57.6
1748.2	0.997	.37	1912	57.6
2051.3	0.996	.37	1912	60.0
2405.0	0.997	.36	1806	53.3
2728.4	0.999	.34	1883	58.9
3132.6	0.999	.36	1779	53.8
3304.3	0.999	.35	1914	58.4
3526.6	0.999	.35	1921	58.0
3799.5	1.000	.34	1906	59.7
4102.6	1.000	.35	1906	59.3
4345.2	1.002	.35	1810	56.5
4729.1	1.001	.36	1910	59.6
4870.6	1.001	.37	1906	59.0
5133.3	1.002	.37	1892	60.1
5355.7	1.002	.37	1867	60.0
5689.1	1.001	.39	1777	59.5
5860.9	1.001	.38	1724	59.8
6103.4	1.000	.37	1660	60.0

Cycle Exposure Power Keff MWD/MTU MWt 272.3 0.997 2184 712.1 1.003 2235 881.9 1.002 2240 1470.6 1.003 2197 2238.9 1.002 2450 3190.2 1.000 2413 3836.2 0.998 2197 4074.2 1.001 2320 4730.1 0.997 2377 5301.6 0.997 2337 6559.2 0.997 2225 6807.3 0.998 2210 7397.0 0.996 2267 7659.4 0.997 2187 7980.2 0.997 2203

Table 5.2-5 XTGBWR Calculated K and Average Voids for Quad Cities Cycle 1

XN-NF-80-19(NP)(A) Vol. 1

Table 5.2-6 XTGBWR Calculated K and Average Voids for Quad Cities Cycle 2

Cycle Exposure MWD/MTU	K _{eff_}	Power MWt
245.8	0.997	2171
677.9	1.002	2156
1136.5	0.995	2096
1502.5	0.998	2411
1855.2	1.000	2500
2886.9	1.000	2463
3951.7	0.997	2474
4648.3	1.000	2153
5609.5	0.999	1829
5911.5	0.998	1713
6324.5	0.998	1547
6954.5	0.998	1487

XN-NF-80-19(NP)(A) Vol. 1



Wide-Wide Gap

Calculation Measured -

OC Fuel Assembly	-	UD 3109
Distance Above Bottom of Fuel	-	27.5 inches
Exposure	-	4,500 MWD/MTU
Void Fraction	-	0
Gamma Scan Data	-	La-140

Figure 5.1-1 Comparison of XFYRE Calculated/Gamma Scan Measured Local Power Distribution for ENC 8x8 Reload Fuel

171

XN-NF-80-19(NP)(A) Vol. 1

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OC Fuel Assembly	-	UD 3109	Calculation Measured
Distance Above Bottom of Fuel	-	47.0 inches	
Exposure	-	3,800 MWD/MTU	
Void Fraction	-	0.34	
Gamma Scan Data	-	La-140	

Comparison of XFYRE Calculated/Gamma Scan Figure 5.1-2 Measured Local Power Distribution for ENC 8x8 Reload Fuel

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Calculation Measured

OC Fuel Assembly	-	UD 3109
Distance Above Bottom of Fuel		105.6 inches
Exposure	-	3,400 MWD/MTU
Void Fraction	-	0.65
Gamma Scan Data	-	La-140

Figure 5.1-3 Comparison of XFYRE Calculated/Gamma Scan Measured Local Power Distribution for ENC 8x8 Reload Fuel


12

OC Fuel Assembly	-	UD 4070	Calculation
Distance Above Bottom of Fuel	-	47 inches	
Exposure	-	3,900 MWD/MTU	
Void Fraction	-	0.31	
Gamma Scan Data	-	La-140	
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Figure 5.1-4 Comparison of XFYRE Calculated/Gamma Scan Measured Local Power Distribution for ENC 8x8 Reload Fuel



Figure 5.1-5 Comparison of XFYRE Calculated/Gamma Scan Measured Local Power Distribution for ENC 8x8 Reload Fuel



Control Rod Location

> Figure 5.1-6 Fuel Rod Position Identification for Garigliano Isotopic Comparison

175

XN-NF-80-19(NP)(A) Vol. 1

0.943 0.934 <u>+</u> .010	1.077 1.072 <u>+</u> .007	1.086 1.088 <u>+</u> .009	1.035 1.060 <u>+</u> .018	1.040 1.049 <u>+</u> .016	1.101 1.104 <u>+</u> .020	1.096 1.106 <u>+</u> .03
1.168 1.146 <u>+</u> .010	1.059 1.050 <u>+</u> .013	0.936 0.947 +.008	0.894 0.902 <u>+</u> .011	0.900 0.911 <u>+</u> .014	0.955 0.947 <u>+</u> .018	
1.100 1.097 <u>+</u> .011	0.991 1.010 <u>+</u> .010	0.877 0.886 <u>+</u> .011	0.840 0.841 <u>+</u> .012	0.846 0.840 <u>+</u> .013		
1.095 1.098 <u>+</u> .010	0.984 0.970 <u>+</u> .011	0.871 0.872 <u>+</u> .009	0.833 0.849 <u>+</u> .015			
1.152 1.167 <u>+</u> .017	1.039 1.019 <u>+</u> .014	0.914 0.889 <u>+</u> .016				
0.924 0.919 <u>+</u> .021	1.042 1.018 <u>+</u> .020					
1.092 1.065 <u>+</u> .018	XFYRE XMC +10					

Figure 5.1-7 XFYRE/XMC (Monte Carlo) Calculated Local Power Distribution for Oyster Creek BWR Reload Fuel, 0% V - No Gadolinia -No Control

XN-NF-80-19(NP)(A) Vol. 1

0.987 1.015 <u>+</u> .009	1.109 1.129 <u>+</u> .009	1.073 1.086 <u>+</u> .012	0.975 0.969 <u>+</u> .007	0.994 0.988 <u>+</u> .013	1.079 1.058 <u>+</u> .014	1.082 1.082 <u>+</u> .01
1.229 1.255 <u>+</u> .013	1.100 1.114 <u>+</u> .011	0.905 0.884 <u>+</u> .008	* 0.407 0.380 <u>+</u> .004	0.824 0.813 <u>+</u> .011	0.922 0.898 <u>+</u> .011	
1.167 1.188 <u>+</u> .011	1.040 1.049 <u>+</u> .008	0.872 0.873 <u>+</u> .008	0.773 0.759 <u>+</u> .006	0.775 0.755 <u>+</u> .010		
1.173 1.173 <u>+</u> .011	1.051 1.047 <u>+</u> .009	0.899 0.898 <u>+</u> .009	0.821 0.833 <u>+</u> .011			
1.248 1.255 <u>+</u> .011	1.128 1.117 <u>+</u> .014	0.973 0.962 <u>+</u> .014				
1.011 1.006 <u>+</u> .011	1.143 1.139 <u>+</u> .012					
1.193 1.216 <u>+</u> .016	XFYRE XMC +10					

* 1.0 w/o Gd203

Figure 5.1-8

XFYRE/XMC (Monte Carlo) Calculated Local Power Distribution for Oyster Creek BWR Reload Fuel 32% V - With Gadolinia -No Control

0.940 0.944 <u>+</u> .013	1.074 1.076 <u>+</u> .015	1.076 1.076 <u>+</u> .015	1.016 0.994 <u>+</u> .011	1.016 0.998 <u>+</u> .012	1.075 1.083 <u>+</u> .017	1.065 1.082 +.01
1.170 1.182 <u>+</u> .013	1.068 1.064 <u>+</u> .010	0.935 0.912 <u>+</u> .010	0.883 0.856 <u>+</u> .011	0.885 0.848 <u>+</u> .009	0.940 0.937 <u>+</u> .016	
1.107 1.121 <u>+</u> .013	1.002 1.007 <u>+</u> .007	0.878 0.881 <u>+</u> .005	0.829 0.840 <u>+</u> .008	0.832 0.819 <u>+</u> .012		
1.106 1.130 <u>+</u> .011	1.001 1.001 <u>+</u> .009	0.875 0.872 <u>+</u> .008	0.827 0.825 <u>+</u> .009			
1.169 1.203 +.015	1.064 1.068 <u>+</u> .012	0.928 0.940 <u>+</u> .013				
0.943 0.942 <u>+</u> .010	1.070 1.071 <u>+</u> .012					
1.111 1.141 +.019	XFYRE XMC +10					

Figure 5.1-9 XFYRE/XMC (Monte Carlo) Calculated Local Power Distribution for Oyster Creek BWR Reload Fuel, 32% V - No Gadolinia -No Control

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XN-NF-80-19(NP) (A) Vol. 1

0.918 0.961 <u>+</u> .009	1.176 1.188 <u>+</u> .012	1.305 1.324 <u>+</u> .012	1.328 1.341 +.016	1.395 1.406 <u>+</u> .013	1.518 1.527 <u>+</u> .026	1.526 1.542 +.025
0.948 0.966 <u>+</u> .009	1.083 1.096 <u>+</u> .011	1.093 1.079 <u>+</u> .010	1.128 1.105 <u>+</u> .009	1.193 1.181 <u>+</u> .011	1.307 1.323 <u>+</u> .015	
0.763 0.755 <u>+</u> .009	0.929 0.944 <u>+</u> .012	0.968 0.981 <u>+</u> .011	1.015 1.001 <u>+</u> .014	1.082 1.089 <u>+</u> .016		
0.680 0.672 <u>+</u> .009	0.843 0.843 <u>+</u> .013	0.892 0.883 <u>+</u> .013	0.944 0.928 <u>+</u> .012			
0.645 0.620 <u>+</u> .009	0.796 0.778 <u>+</u> .008	0.840 0.846 <u>+</u> .012				
0.465 0.435 +.006	0.676 0.653 <u>+</u> .014					
0.467 0.446 +.011	XFYRE XMC +10					

Figure 5.1-10

XFYRE/XMC (Monte Carlo) Calculated Local Power Distribution for Oyster Creek BWR Reload Fuel, 32% V - No Gadolinia -Controlled

0.917 0.968 <u>+</u> .009	1.060 1.095 <u>+</u> .012	1.062 1.070 <u>+</u> .012	0.996 0.989 <u>+</u> .009	0.989 0.984 <u>+</u> .012	1.038 1.040 <u>+</u> .009	1.013 1.035 +.016
1.155 1.201 <u>+</u> .012	1.077 1.083 <u>+</u> .010	0.944 0.920 <u>+</u> .013	0.884 0.848 +.010	0.878 0.840 <u>+</u> .011	0.925 0.908 <u>+</u> .012	
1.103 1.138 <u>+</u> .011	1.023 0.999 <u>+</u> .008	0.894 0.853 <u>+</u> .008	0.837 0.793 <u>+</u> .008	0.832 0.791 <u>+</u> .013		
1.109 1.119 <u>+</u> .012	1.028 1.000 <u>+</u> .009	0.899 0.863 <u>+</u> .009	0.841 0.816 <u>+</u> .009			
1.175 1.222 <u>+</u> .009	1.097 1.101 <u>+</u> .011	0.961 0.950 <u>+</u> .012				
0.949 0.974 <u>+</u> .010	1.097 1.116 <u>+</u> .016					
1.102 1.184 +.014	XFYRE XMC +1g					

Figure 5.1-11 XFYRE/XMC (Monte Carlo) Calculated Local Power Distribution for Oyster Creek BWR Reload Fuel, 64% V - No Gadolinia -No Control







Figure 5 2-2 Dresden-3 Measured and XTGBUR Calculated TIP Comparison



Figure 5.2-3 Dresden-3 Measured and XTGBWR Calculated TIP Comparison



Figure 5.2-4 Dresden-3 Measured and XTGBUR Calculated TIP Compartson



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Figure 5.2-6 Dresden-3 Measured and XTGBWR Calculated TIP Comparison



Figure 5.2-7 Oyster Creek Measured and Calculated TIP Comparison



Figure 5.2-8 Oyster Creek Measured and Calculated TIP Comparison



Figure 5.2-9 Oyster Creek Measured and Calculated TIP Compartson



Figure 5.2-10 Oyster Creek Measured and Calculated TIP Comparison





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Figure 5.2-12 Oyster Creek Measured and Calculated TIP Comparison

				.50 .49											
	_			.68 .64	.85 .80										
			.64 .61	.83 .78		1.27				Calc Meas		Calcu LA-14	lated/M 0 Activ	Measured vity	1
re Edge)		.48 .50	.75	.91 .86		1.13	1.15 1.11					Stand of Pe	ard Dev rcent I	viation Differen	= 3.1% ices (1o
		.60 .59			1.09	1.37 1.35		1.40 1.34							
	.49 .49		.90 .87	1.03			1.16 1.15	1.18 1.18	1.15 1.15						
.37 .39	.61 .60		.95 .94	1.26	1.10	1.12 1.11	1.14 1.12	1.24 1.20		1.31 1.30					
		.86		1.09		1.12 1.10	1.11 1.12		1.10 1.11		1.07				
	.73 .72	.90 .88		1.30 1.34	1.12 1.13	1.34 1.35	1.13 1.13	1.32 1.33	1.10	1.07 1.09		1.26 1.28			
.48 .51			.99 .99	1.06		1.09	1.08 1.12	1.09 1.11			1.05	1.06	1.06		
.49	.75	.89	.97	1.02	1.04	1.05	1.07	1.05	1.04	1.03	1.02	1.04	1.08		

Figure 5.2-13 Quad Cities 1 EOC2 Fuel Assembly Gamma Scan Comparison, XTGBWR Calculated/Measured La-140 193

XN-NF-80-19(NP)(A)





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5.4 REFERENCES

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XN-NF-80-19(NP)(A) Vol. 1

6.0 MEASURED POWER DISTRIBUTION UNCERTAINTY

The determination of the uncertainty associated with a measured power distribution is necessary to insure safe reactor operation. The safety analyses are performed to assure safe reactor operation with a certain quantified degree of cor fidence; thus, the uncertainty associated with the measured reactor power distribution must be quantified. The uncertainty analysis presented in this section begins with a concise mathematical expression of the method for determining the measured power distribution. The uncertainty is then defined in terms of the relative standard deviations of the independent variables involved in the measured power distribution determination. Methods to estimate the relative standard deviations of the independent variables from measured data are described. Using the relative standard deviation estimates in conjunction with the equation for the measured power distribution uncertainty, results in an estimate of the measured power distribution uncertainty expressed as a relative standard deviation.

6.1 MEASURED POWER DISTRIBUTION DETERMINATION

Reactor measured power distributions are combinations of measured reactor data and computer calculated data. The measured reactor power distribution data include the fixed local power range monitor (LPRM) in-core detector data and the traveling in-core probe (TIP) detector data.

The LPRM data are electric current readings proportional to the neutron flux level at four axial elevations in a number of radial locations. The radial locations are distributed in a uniform lattice throughout the

core. The LPRM detectors are fission chambers using U-235 as the fissionable isotope. The LPRM detectors are intercalibrated utilizing the TIP data. The TIP system consists of a number of movable fission chamber detectors (about 1" long) which can each enter a number of the radial locations at which the fixed LPRM detectors are located. The movable TIP detectors are all capable of entering one of the radial positions to allow intercalibration of the TIP system. Figure 6.1 is a drawing of an in-core instrument tube which contains both the four LPRM detectors and the TIP tube. Figure 6.2 depicts typical radial locations for both fixed and movable in-core detectors in a BWR core. Each radial location contains the equipment shown in Figure 6.1.

The computer calculated data include the relative core nodal power distribution, the in-core detector response distribution, and the local peaking factors for the fuel rods. The predicted relative nodal power and detector response distributions are calculated with the XTGBWR reactor simulator code described in Section 3.2. The XTGBWR code is a three dimensional modified two group diffusion theory reactor simulator program. The code uses large mesh sizes to perform full core nodal power calculations with time dependent xenon and samarium.

The local peaking factors are calculated by the XFYRE and XDT codes described in Sections 3.1 and 3.4. The XFYRE code is a single bundle depletion model that performs a microscopic depletion of each fuel rod in the fuel assembly. The XDT code is a diffusion theory program used to perform multibundle power distribution calculations.

198

XN-NF-80-19(NP)(A)

Vol. 1

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The synthesis of the measured power distribution can be viewed to occur in two phases. Phase I consists of the fixed LPRM in-core detector calibration. Phase II consists of combining the individual fixed LPRM incore detector distribution measurements with XTGBWR calculated data to produce the measured power distribution. An outline of the procedure is presented here.

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6.2 UNCERTAINTY DERIVATION

The uncertainty in the power distribution, P_{ijk} , can be derived based upon the measurement procedure formulation as expressed. The notation is simplified by rewriting for a single node ijk. In the following development, the index i will denote each fixed LPRM in-core detector used to determine P_{ijk} with ND denoting the number of detectors used.

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6.3 ESTIMATION OF UNCERTAINTY

The uncertainties, in terms of relative standard deviations, [] are determined by comparison to measured data. The measured data consist of distributions of TIP and fixed in-core detector responses plus gamma scans of bundles and pins. The majority of the data consists of TIP and fixed incore detector distributions. This is due both to the limited amount of gamma scan data available, and to the limited core conditions represented by gamma scans.

XN-NF-80-19(NP)(A) Vol. 1

6.3.1 Detector Measurement: 6_F

[] utilizes measured data which consist of a relative distribution of fixed in-core detector responses, F_i . The fixed detectors are located at four axial elevations in each of a number of radial locations, []. The fixed detector responses are calibrated to TIP system measurements at regular intervals and are adjusted for the reduction in sensitivity to the neutron flux as a function of burnup between calibrations to the TIP system.

The uncertainty of the fixed in-core detector data [] is comprised of two sources. First, the uncertainty due to the TIP system which is acquired through the calibration process. Second, the uncertainty associated with the fixed in-core detector response itself.

The uncertainty in the TIP system measurements can be derived from symmetric TIP data. A core which is loaded 1/8 core symmetric and is operating with an 1/8 core symmetric rod pattern will have a number of pairs of instrumented radial locations which will have the same neutron flux distribution. Differences between the TIP responses in these positions can be used to define the TIP system measurement uncertainty.

The uncertainty in the TIP system measurements will be divided into two sources. First, the radial effects due principally to the random offset of the TIP from the center of the water region between channels,

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[] is defined. Second, a term, [] is defined to represent all other sources of uncertainty. Define d_{ij} as the <u>relative</u> difference between symmetric TIP pair i at axial elevation j. The two sources of uncertainty can then be estimated as: [

The uncertainty in the fixed in-core detector response itself can be estimated from repeated measurements of the fixed detector response during a period of time when the power distribution is stable. The uncertainty associated with the fixed in-core detector lies in its ability to reproduce the response to which it was calibrated. The adjustment of the detector response due to a reduction in sensitivity through depletion of the U-235 introduces additional uncertainty, but this effect is negligible relative to that due to detector reproducibility and calibration to the TIP system. The uncertainty due to the fixed in-core detector reproducibility, [] is determined as follows.

Let [] represent the relative differences between two measurements j and k at position i.

[

The uncertainty in the calibration, [] of a fixed in-core detector is the sum of the uncertainty in the TIP system measurement to which the fixed in-core detector is normalized and the uncertainty of the detector response being normalized.

XN-NF-80-19(NP)(A) Vol. 1

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The uncertainty of the fixed in-core detector distribution [] is the uncertainty associated with the calibration plus the uncertainty of the fixed in-core detector reproducibility.

6.3.2 Calculated Detector Response Distribution: δ_T

The uncertainty in the calculated detector response distribution can be determined by comparison to measured detector distributions, either from the TIP system or from the fixed in-core detectors. The relative standard deviation in the calculated detector response distribution can be determined as follows:

[

[

To define the relative standard deviation in [] the calculated detector distribution, the uncertainty in [] must be removed from [] Equation 6.25 below represents the uncertainty in T if the TIP system measurements were used and Equation 6.26 represents the uncertainty if fixed incore detector measurements are used [].

6.3.3 Calculated Nodal Power Distribution: δ_B

There are two sources of measured data which can be used in determining the uncertainty in the calculated nodal power distribution, [] The relative standard deviation [] can be derived from the calculated detector distribution uncertainty or it can be derived by comparing to gamma scan measurements of bundle power distributions. Both methods will be utilized.

XN-NF-80-19(NP) (A) Vol. 1

The derivation of [] the relative standard deviation [] from the uncertainty in [] will be described first. The detector response distribution in XTGBWR is determined from the nodal power distribution using detector response-to-power factors. [

]. The equation

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from which T is determined can be written as follows, ignoring the normalization term.

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Following the approach described in Section 6.2, the relative variance [] can be written [] as in Equation 6.27. The random variables are treated as being independent. Covariance terms may need to be defined altering the equations, if analysis of the data indicates dependency among the random variables.

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6.3.4 Local Pin Distribution: δ_1

The pin power distribution is determined by multiplying the nodal power, [] by a local power distribution factor, []. Local factors for each fuel type are calculated by the XFYRE and XDT codes and input to the XTGBWR code as a function of exposure, void, and control state (controlled or uncontrolled). XTGBWR interpolates among the input data to determine a value for the particular exposure, void and control state at node ijk.

The uncertainty in local peaking factors are determined by comparing the calculated pin powers to the pin by pin gamma scans of bundles which have been irradiated in a reactor. To perform the comparisons, the pin by pin power distributions from XFYRE/XDT must be converted to La-140 distributions, since the gamma scans measure La-140 distributions rather than power distributions.

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XN-NF-80-19(NP)(A) Volume 1 Supplement 1

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EXXON NUCLEAR METHODOLOGY FOR BOILING WATER REACTORS VOLUME 1 NEUTRONIC METHODS FOR DESIGN AND ANALYSIS

APPROVED:

2 TZK 4/24/81 Manage

T. L. Krysinski, M BWR Neutronics

APPROVED:

6/25/81 CWL

JW. Lindenmeier, Manager eutronics Development

APPROVED:

30 JUNPI

R. B. Stout, Manager Neutronics and Fuel Management

APPROVED:

G. F. Owsley, Manager

G. F. Owsley, Manager Reload Fuel Licensing

APPROVED:

G. A. Sofer, Manager Nuclear Fuels Engineering

EXON NUCLEAR COMPANY, Inc.

XN-NF-80-19 (NP)(A) Volume 1 Supplement 1

This supplement provides the responses to USNRC questions pertaining to the proposed Exxon Nuclear Company Neutronics Methodology for Boiling Water Reactors (XN-NF-80-19 (P), Volume 1). In addition, this supplement fulfills the request for additional information contained in a letter from R. L. Tedesco (USNRC) to G. F. Owsley (ENC) dated January 19, 1981.

-1-

Supplement 1

- Q1. In the description of the core depletion code XFYRE the energy range of the thermal cross sections range from 0 to .683 ev., while epithermal slowing down spectrum calculation is performed over the range of 10 MeV to .414 ev. No explanation is given for this slight overlap of the energy ranges. Therefore provide a discussion of the energy boundaries used for the fast and thermal spectral codes. Further, in the description of the broad energy groups, no such overlap is indicated. (paragraph 3.1.5) Either with a .414 or a .683 ev cut off, how is the contribution of the upscattered neutrons to the Pu-240 resonance at 1 ev accounted for?
- A1. The HRG slowing down calculations are performed with 68 fine energy groups over the energy range 0.414 ev to 10 MeV.

The 1.056 eV resonance in Pu-240 is treated in XFYRE in the epithermal range.

Finally, the resonance is isolated in later diffusion theory calculations by choosing one of the broad groups to span only the 0.683-2.38 eV range.
Volume 1

Supplement 1

- Q2. In XFYRE the thermal cross sections are calculated separately for standard fuel rods and for rods with gadolinia and for rods adjacent to the water gap. How are the spectral effects of the water gap on peripheral fuel rods taken into account?
- A2. The effects of the water outside the channel are modeled in the XFYRE code by performing a four (4) energy group two-dimensional (X-Y geometry) fine mesh diffusion theory calculation. The calculation uses reflecting boundary conditions with the boundaries located midway between the fuel assembly channels.

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XN-NF-80-19 (NP)(A) Volume 1

The

Supplement 1

- Q3. In the XFYRE code, in a calculation of the incore detector parameters and specifically in the T factor discuss: (a) the significance of the "conversion constant c" and (b) how are the spectral and spatial effects of the water gap accounted for so that the incore detector response is correctly determined. Is the contribution of the nonthermal fissions to the T factor neglected? If so why?
- A3. The T factor, DC factor and nodal power are combined and normalized to give a relative incore detector response.

constant c is a fixed rate function numerical number selected for convenience in data handling. The constant c cancels out when the relative incore detector response is determined. The T factor is / input into the XTGBWR code as a function of exposure, instantaneous voids, void history, control, and fuel type.

The DC factor is a correction to the T factor to account for actual detector composition effects and the epithermal fission effects neglected in the T factor calculation. Typical numerical values for the DC factor at a constant void fraction at each XTGBWR node are shown in Table 3.1. The spectral and spatial effects of the water gap are modeled in the XFYRE code by performing the fuel bundle diffusion theory calculation with a two-dimensional fine mesh geometry and four (4) energy groups.

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Table 3.1	Numerical	Values	for	Axial	Tip
	Correction	Factor	(DC	Facto	or)

XTG Node	LPRM <u>Wires</u>	DC Value
1	0	
2	0	
3	0	
4	1	
5	1	
6	1	
7	1	r
8	- 1	
9	1	
10	2	
11	2	
12	2	
13	2	
14	2	
15	2	
16	3	
17	3	
18	3	
19	3	
20	3	
21	3	
22	4	
23	4	
24	4	

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*Plunger spring at top of LPRM assembly.

XN-NF-80-19(NP)(A)

Volume 1

Supplement 1

- Q4. There is only one instance of a gadolinia bearing fuel rod which compares the XFYRE And XMC calculated local power distribution. (Oyster Creek, Figure 5.1-8) The difference in this case is 7.1%. Is this adequate to establish the ability of XFYRE to model gadolinia bearing fuel rods?
- A4. An XFYRE and XMC calculated local power distribution for a fuel design containing 3.0 w/o Gd₂O₃ is shown in Figure 4.1. The adequacy of the XFYRE code to model the gadolinia bearing fuel rods is demonstrated by the good agreement in the calculated reactor k data and the comparison of measured and calculated TIP traces. In addition the Oyster Creek fuel rod gamma scan measurements (see response to Question 5) show the XFYRE code accurately predicts the power generation in the fuel rods containing gadolinia.

In order to insure maintaining adequate margin to limits, it is necessary to demonstrate the adequacy of predictory core and assembly reactivity, and the power peaking on the limiting fuel rods. Since good agreement between calculated and measured valves exist, the ENC neutronics methodology is judged adequate to maintain sufficient margin to plant operating limits.

-6-

XN-NF-80-19 (NP)(A) Volume 1 Supplement 1

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Figure 4.1 Comparison of XFYRE and XMC Local Power Distribution for C-Lattice Fuel Design with Six w/o Gd₂O₃ Rods, No Control. 40% Voids, Zero Exposure

XN-NF-80-19 (NP) (A) Volume 1 Supplement 1

- Q5. The ratio of XFYRE calculated to measured local power distributions presented in Figure 5.1-4 shows variations between 0.915 and 1.140 for fuel rods next to a water gap. Discuss the reason for these large differences between measurement and calculation, and the ability of XFYRE to calculate these fuel rods. Are any of these rods during the expected exposures the high powered rods in the bundle?
- A5. The local power comparisons shown in Figures 5.1-1 through 5.1-5 in XN-NF-80-19 (P), Volume 1, do not consider the flux tilts in the reactor core associated with control rods, incore detectors, exposure and void effects.

In order to illustrate the effects of flux tilts upon single assembly, calculations, a multi-bundle diffusion theory calculation was performed for the assembly shown in Figure 5.1-4. This calculation modeled the assembly in question as well as adjacent assemblies and any control rods which were present in the sixteen assembly model. The XFYRE calculated cross sections were used as input into this multi-bundle calculation. The resulting local power peaking for the assembly in question is shown in Figure 5.1.

The results in Figure 5.1 compared to those previously presented (Figure 5.1-4 of XN-NF-80-19 (P), Volume 1) indicate a general improvement in the local power peaking with the exception of the fuel rod immediately adjacent to the fuel rod in the wide-wide gap. This lack of improved agreement is attributed to an apparent deficiency in the measured gamma scan results. Figure 5.2 shows the measured gamma scan results for the assembly in question and indicates a large disparity in the measured values between the two symmetric rods adjacent to the wide-wide gap. This lack of anticipated agreement between these two fuel rods lends some uncertainty to the comparison for these rods which is illustrated is Figure 5.1. In that this fuel rod is not anticipated to be a limiting fuel rod, the lack of agreement is anticipated not to significantly affect the establishment of reactor operating limits. Thus, the use of XFYRE to establish reactor operating limits is considered justified.

XN-NF-80-19 (NP)(A) Volume 1 Supplement 1

EXXON NUCLEAR PROPRIETARY

Figure 5.1 Comparison of 16 Bundle XDT Calculated / Gamma Scan Measured Local Power Distribution for OC 8x8 Reload Fuel

XN-NF-80-19 (NP) Volume 1 Supplement 1

EXXON NUCLEAR PROPRIETARY

Figure 5.2 Measured Gamma Scan Results for ENC OC Reload Fuel

Supplement 1

- Q6. Multigroup bundle calculations are generally required to adequately treat the strong spatial variation in the thermal spectrum that exists across a bundle. Is the Exxon four-group calculation adequate to determine bundle average cross sections?
- A6. Strong spatial variations in the thermal flux spectrum in a BWR bundle exist primarly in the vicinity of gadolinia bearing fuel pins and near control blades. For this reason, XFYRE utilizes special models designed to simulate the essential features which contribute to the thermal flux spectrum transitions in these regions of the bundle.

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Comparisons have been made between results obtained from the production version of XFYRE (the four-group version described in the documentation) and a special five-group version developed for testing purposes. The five-group XFYRE retains the epithermal group structure, however the thermal region is treated using a twogroup model with upscatter from group five to group four. The epithermal and thermal spectrum calculational procedure is identical in both versions. The results of the comparison are presented in Figures 6.1 and 6.2. These results indicate no substantial differences in bundle reactivity or local peaking distribution predictions. The conclusion of this comparison effort is that the significant thermal spectrum effects are retained in the four-group XFYRE model.

XN-NF-80-19 (NP) Volume 1 Supplement 1

EXXON NUCLEAR PROPRIETARY

Figure 6.1 One Thermal Group / Two Thermal Group Calculated Local Power Distribution For Oyster Creek BWR Reload Fuel, 0% V - No Gadolinia - No Control Zero Exposure

XN-NF-80-19 (NP)(A) Volume 1 Supplement 1

EXXON NUCLEAR PROPRIETARY

Figure 6.2 One Thermal Group / Two Thermal Group Calculated Local Power Distribution for Oyster Creek BWR Reload Fuel, 64% V - No Gadolinia - No Control Zero Exposure Q7. An empirical correction factor F is applied (Equation 3.2-28) for the prediction of the thermal flux and the improvement of the nodal power of controlled nodes. With regard to F:(a) describe how is it established (b) it is dependent or position, burnup, voids or other parameters? and (c) give typical values for F. Because controlled and noncontrolled node cross sections are used, why is the F required?

A7. A recently developed model improvement has been adopted for the thermal flux treatment in XTGBWR. The revised model eliminates the necessity for applying the correction factor, F. In the revised model, nodally dependent thermal averaging factors are calculated based on the thermal cross sections appropriate for the node in question.

A detailed description of the new thermal flux model including derivation of the thermal averaging factor and the nodal interface thermal neutron current approximation is presented in Section 3.2.2 of XN-NF-80-19 (P), Volume I, Supplement 2.

-14-

Volume 1

Supplement 1

- Q8. For boundary conditions on the outer boundary it is stated that for boundary nodes with more than one reflector face, a multiplier is derived from fine mesh two-dimensional diffusion theory. With regard to this multiplier, (a) what fine mesh diffusion calculations are performed and how is the multiplier calculated? (b) what parameters does this multiplier depend on? (c) what is the range of its applicability and (d) give some typical values of this multiplier.
- A8. The revised thermal flux model described in the response to Question #7 also eliminates the necessity for the multiplier which corrects the thermal flux estimate for boundary nodes which have more than one reflector interface.

A detailed description of the boundary node thermal flux model including appropriate derivations is presented in Section 3.2.3 of XN-NF-80-19 (P), Volume I, Supplement 2.

-15-

XN-NF-80-19 (NP)(A)

Volume 1

Supplement 1

- Q9. The exposure steps for the input of exposure and void dependent hot operating cross sections appear too large. Gadolinia-loaded fuel requires finer exposure intervals. Please discuss.
- A9. At the present time cross section data for up to exposures can be input into the XTGBWR reactor simulator code and the exposure values can be different for different fuel types. The specific exposures for a given fuel type are selected to accurately model the variation of the nuclear parameters with exposure and gadolinia depletion.

Comparisons

of XFYRE K data and XTGBWR K data for gadolinia concentrations of 1.5 and 3.0 w/o are given in Tables 9.1 and 9.2. The comparison between the XFYRE data and the XTGBWR ' values demonstrate the adequacy of the interpolation scheme used in XTGBWR to determine intermediate values of K.

-16-

Table 9.1 Comparison of XFYRE and XTG k Data (40%V) for Fuel Design with 1.5 w/o Gd_20_3 Concentration

EXXON NUCLEAR PROPRIETARY

Table 9.2 Comparison of XFYRE and $\lambda TG \ k_{\infty} Data (40\%V)$ for Fuel Design with 3.0 w/o $^{\infty}Gd_2O_3$ Concentration

EXXON NUCLEAR PROPRIETARY

y 1

Volume 1

Supplement 1

- Q10. In addition to XFYRE and XDT what other codes are used to prepare input data for the XTGBWR? For example there is no indication how (i.e., by what mechanism and by what program) are the hot operating cross sections introduced in the XTGBWR.
- A10. Most of the data is input into the XTGBWR code in table form. Data fitting Fortran programs are used to prepare the fit coefficients where empirical formulas rather than table lookup is used to describe the input data for the XTGBWR code. Examples of data input as fit coefficients include the void history corrections to the cross section data, the T factors and the Doppler fit coefficients.

Y

Supplement 1

- Q11. A multiplier (VHR) is developed for cross section correction as a function of exposure void history and instantaneous void. However, there is no indication as to what cross sections and energy groups or for what void history or instantaneous voids it is applied.
- All. The void history correction term (VHR) is applied to

The correction is applied for all values of

-20-

Supplement 1

- Q12. Cross sections have been computed for a number of conditions, such as exposure voids, etc. Give a flow chart including the necessary ancillary calculations for a typical fuel assembly cross section calculation.
- A12. The fuel assembly depletion calculations are performed at specific void fractions with the fuel temperature and moderator temperature held constant. The depletion calculations are normally performed uncontrolled; however, the XFYRE code has the capability of depleting the fuel assembly controlled. The effects of control, fuel temperature, and other void histories on the nuclear parameters are evaluated by performing solutions at the conditions of interest.

A chart of typical XFYRE calculations is shown in Table 12.1.

Table 12.1 Chart of Typical XFYRE Calculations

EXXON NUCLEAR PROPRIETARY

- Q13. With respect to the effect of the control rods on the cross sections, comment on the following questions:
 - (a) for what exposures are cross sections for controlled fuel nodes inputed?
 - (b) are control history effects taken into account?
 - (c) what types of calculations are performed which involve partially controlled nodes? How is power peaking handled in such calculations?
 - (d) why are controlled to uncontrolled ratios (and not straight controlled cross sections) used to represent the controlled nodes?

A13. (a)

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- (b) The control history effects are not taken into account for scatter load fuel management schemes where the control rod patterns are changed at appropriate exposure intervals.
- (c)
- (d) The controlled to uncontrolled cross sections ratios

An example of the controlled to uncontrolled cross section ratios as a function of exposure for a D lattice fuel design with 2.0 w/o Gd_2O_3 is shown in Figure 13.1.



Ratios D Lattice Fuel 2 w/o Gd203 40%V

- Q14. What is the functional dependence, range of applicability, limitations and typical values of the parameters EDOPO, EDOP1 and VDOP1 in equation (3.2-49)? Is linear dependence adequate to describe the Doppler cross section variation as a function of exposure?
- A14. The Doppler correction in the XTGBWR code is applied to the The correction is a function of exposure, voids, and power. The Doppler correction is applicable over the range of fuel temperatures in the operating reactor.

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The calculated and fit DOPCOF for epithermal absorption is shown in Figure 14.1.

XN-NF-80-19 (NP)(A) Volume 1 Supplement 1

EXXON NUCLEAR PROPRIETARY

Figure 14.1 Doppler DOPCOF Factor for Epithermal Absorption

XN-NF-80-19 (NP)(A) Volume 1 Supplement 1

- Q15. With respect to coolant flow distribution comment on the following questions?
 - (a) when is the approximate equation (3.2-50) used instead of the pressure drop model?
 - (b) how many hydraulic types are used in a typical calculation?
 - (c) is bypass flow accounted for in XTGBWR?
 - (d) is bypass voiding considered?
 - (e) is equation 3.2-50 adequate to obtain the dependent CPR on assembly flow?
- A15. (a) The XTGBWR methodology used to determine the coolant flow distribution is consistent with the thermal hydraulic methods described in XN-NF-80-19(P), Volume 3. At the present time, it is anticipated that the pressure drop model will be used for all calculations.
 - (b) XTGBWR has the capability of modeling the hydraulic performance of each fuel type within the core. This includes fuel assemblies of different design as well as fuel assemblies of the same design in different orifice zones.
 - (c) Yes.

- (d) Voiding in the bypass regions is not considered in XTGBWR. It is anticipated that at the time of insertion of ENC reload fuel into higher power density JP-BWR's the majority of the remaining fuel assemblies will have drilled lower tie plates. Since the ENC fuel assemblies will also have drilled lower tie plates, the potential for voiding in the bypass region is minimal if not zero. In the case of lower poer density JP-BWR's, voiding in the bypass region has traditionally not been regarded as significant.
- (e) The MCPR for the fuel assemblies will be determined with the Exxon Nuclear Company XN-3 critical power correlation or a demonstrated equivalent. The actual calculated assembly flow is used in the calculation of the assembly MCPR. Equation 3.2-50 is determined for each flow distribution calculation and

An example

of the hydraulic demand curves is shown in Figure 15.1.



- Q16. For the correction factor T, for the incore detector response calculation, comment on the following:
 - (a) describe the physical model for the correction factors given by equations 3.2-55, 3.2-56, and 3.2-57.
 - (b) how are the coefficients TA, TB, TC, and TD established? Comment on the statement that the above coefficients are dependent on the reactor but not on the fuel type.
 - (c) what is the accuracy of the approximation of the above equations?
- A16. The incore detector response T factors are calculated with the XFYRE bundle depletion code. The void history dependence of the T / factor is determined by

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equations 3.2-55, 3.2-56, and 3.2-57 used to model the void history effects are determined empirically to fit the calculated data. The equation 3.2-56 as reported has a typographical error. The correct equation 3.2-56 is a follows:

The coefficients are selected to minimize the error in fitting the data with the model. A comparison of the calculated data, the fit data, and the fitting error for one fuel type is shown in Table 16.1.

The effects of the void history on the T factor is different for the D and C lattices where the water gap between the channels is a different thickness. A typical plot of the T factor data for a D lattice fuel design and a C lattice fuel design is shown in Figures 16.1 and 16.2. The solid curves in Figures 16.1 and 16.2 represent the calculated T factor during the standard depletion calculation, The

XN-NF-80-19 (NP)(A)

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Volume 1 Supplement 1

individual data points in the above figures show

As stated above, this data is used to determine the T factor fit and the validity of the fit is demonstrated in Table 16.1. For a given lattice type, calculations of the T factor have shown insignificant sensitivity to different fuel types. Thus, the same are assumed for all fuel types.

P 1

XN-NF-8C-19 (NP)(A) Volume 1 Supplement 1

Table 16.1 Void History T Factor Fit Data for C Lattice

E GWD/MTU	<u>VH</u>	<u>_vi</u>	T VH,VI	T <u>E,VH**</u>	T <u>Fit*</u>	Error, %
1						
2						
4						
10						
20						
30						
10						
10						
10						
20						

XN-NF-81-19 (NP) (A) Volume 1 Supplement 1

EXXON NUCLEAR PROPRIETARY

Figure 16.1 T Factor for D Lattice

XN-NF-81-19 (NP) (A) Volume 1 Supplement 1

EXXON NUCLEAR PROPRIETARY

Figure 16.2 T Factor for C Lattice

XN-NF-80-19 (NP)(A)

5 2

Volume 1

Supplement 1

Q17. Why are cold cross sections introduced in XTGBWR in the form of cold to hot cross section ratios? Is the full set of cross section calculations performed at both hot and cold conditions?

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A17. The ratio of the cold cross sections to the hot operating cross sections is

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-34-

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XN-NF-80-19(NP)(A) Volume 1 Supplement 1

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- Q18. No data are given for the results of cold calculations and comparisons with measurements (e.g., cold critical control rod patterns). Are such data available for evaluation of XTGBWR calculations of cold shutdown margin and rod worth?
- A18. The comparison of calculated and measured criticals are included in Section 5.0 of XN-NF-80-19 (P), Volume 1, Supplement 2.

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XN-NF-80-19 (NP)(A)

Volume 1

Supplement 1

- Q19. Explain how the local peaking factor calculated by XFYRE as a function of exposure, voids, control and fuel type is used in the XTGBWR program.
- A19. The local peaking factors calculated with the XFYRE code are input into the XTGBWR code in tabular form as a function of fuel type, exposure, void, and control.

XN-NF-80-19 (NP) (A)

Volume 1

Supplement 1

- Q20. Are the exposure points at which the cold cross ratios are input (i.e., 2,000, 6,000, 10,000, 15,000, and 35,000 MWD/MT) adequate to accurately predict the maximum reactivity and the cold shutdown margin?
- A20. The exposures selected for input of the cold to hot cross section ratios depends on the specific fuel type.

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-37-

XN-NF-80-19 (NP)(A)

Volume 1

Supplement 1

Q21. In the report, an attempt has been made to validate or benchmark the calculations of this methodology. However, it seems that his effort has been inadequate. For example: (a) there is no identification of the source of the operating data shown for the Quad Cities eigenvalue results for Cycles 1 and 2, (b) no TIP comparisons are given for Quad Cities 1 and the data shown are not adequate to judge the performance of XTGBWR, (c) the report does not show results of a typical simulation for central and peripheral detector responses at BOC, MOC, and EOC, (d) why has not COTRAN been benchmarked against the Peach Bottom-2 turbine trip tests? etc.

It will be necessary for the validation of the methodology to use generally accepted (and to the extent possible clean) data. The data base should be sufficiently large as to contribute to a definitve methodolgoy validation.

- A21. (a) The Quad Cities reactor operating data is reported in the EPRI reports EPRI NP-240 and EPRI NP-214.
 - (b,c) Addition TIP data including calculated-measurement comparisons for Quad Cities and comparisons for central and peripheral detectors at BOC, MOC, and EOC will be presented in Supplement 2 of XN-NF-80-19 (P), Volume 1.
 - (d) The COTRAN computer code has been compared against the Peach Bottom-2 turbine trip tests and the comparisons documented in XN-NF-79-71 and supplements. In addition, the COTRAN code was used in an analysis of the licensing basis transient. This comparison was provided to the staff in a letter report.

(1) ENC Reference Report RHK:056:80 dated October 29, 1980.

-38-
- Q22. Inasmuch as the accuracy of the XMC is limited by the basic cross sections, comment on the following questions:
 - (a) How does the Battelle Master Library compare with the ENDF/B-IV and V data?
 - (b) What fission spectrum is used and how does it compare with ENDF/B-IV and V?
 - (c) Is the resonance cross section structure, represented explicity in the XMC calculations? If an approximation.
 - (d) What error is incurred in the basic cross section representation by assuming an asymptotic 1/E shpae in integrating Doppler
 broadened resonance shapes considering resonance overlap.
 - (e) What isotopes are considered as "heavy scatters" and what is the effect of this opproximation.
 - (f) The code does not use a resonance miltilevel approximattion; what is the effect on capture and fission rates?
- A22. (a) No direct comparisons have been made between the Battelle Master library (BML) utilized in the XMC calculations and the ENDF/B-IV and V data. However, detailed comparisons have been made by BNWL and ENC between the BML and ENDF/B-II and III data.

As discussed in reference (1), calculated values of k for criticals using ENDF/B-II are significantly lower than with the BML. Analysis of the results showed that the primary reason was the epithermal capture in U-238, ENDF/B-II giving values that were too high by about 10%. Similar results were observed by McCrosson (2).

XN-NF-80-19 (NP)(A) Volume 1 Supplement 1

In particular, the ENDF/B-III and -IV U-238 epithermal capture data are known to be high (6,7,8,9). Thus, reactivity calculations using these data would be expected to produce lower values of k.

Recent comparison made by others of U-238 data from ENDF/B-IV and V to benchmark lattices and LWR bundles (10,11,12) indicate a trend of increasing reactivity estimates due to reduced U-238 capture cross sections. It is clear from the reported results of this benchmark testing the overestimate of $\rho 28$ that has persisted for a decade is slowly being removed by improved U-238 resonance parameters.

Overall,

ENC believes that BML data are adequate for their intended purpose.

References

 U. P. Jenquin, et. al., Effect of ENDF/B-II Data For U-238 on Critical Correlations, <u>TRANS AM NUC SOC</u>, Vol. <u>15</u>, P461, 1972.

XN-NF-80-19 (NP)(A) Volume 1 Supplement 1

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- O. Ozer and W. J. Eich, Analysis of LWR Benchmark Experiments With ENDF/B-V Data, <u>TRANS AM NUC SOC</u>, Vol. <u>34</u>, P772, 1980.
- (b) XMC uses a Maxwellian distribution to represent the fission spectrum.

$$P(E)dE = \frac{2}{T\sqrt{\pi}} \left(\frac{E}{T}\right)^{1/2} e^{-E/T} dE$$

Supplement 1

- (c) Resonances are explicitly represented by a single level Breit-Wigner formula. In BML, the important resonances are treated explicitly while small, unimportant resonances and the unresolved resonances are treated as infinitely dilute and included in the smooth (190 Micro-group) cross sections.
- (d) The 1/E assumption should have no significant effect on infinite dilution resonances. For the case of resolved resonances which are treated explicitly, BML has a table specifying how many resonances are to be considered in determining the cross sections within each of the 190 microgroups. The table entries are chosen to insure that all significant resonance overlap is adequately treated.
- (e) In XMC all isotopes heavier than are considered to be "heavy scatters".

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(f) The resonance formalism and data which were processed to prepare the BML did not include parameters for the multilevel approximation. All data were based on the best available single level parameters for the Breit-Wigner formulation. The data in the BML have been adjusted (see Reference 1 to question #22a) to give excellent agreement with critical experiments. That such adjustments are necessary to ENDF/B data has been noted by many authors over the past decade.

In reaching this conclusion fertile and fissile itotopes are considered separately. For the fissile isotopes, such as U-235 and Pu-239, a multilevel formalism would be more accurate. For these isotopes the level spacings are quite close even at low energies. For example, for U-235 the average level spacing of s-wave resonances is about 1 eV and resonance overlap effects are significant (1). On the other hand for U-238 the average s-wave level spacing is about 20 eV and overlap is not significant for the low energy resonances.

For U-238 almost half of the self-shielded resonance integral in a typical LWR comes from the 6.67 eV resonance. This is a large single isolated s-wave resonance. It can be well represented by a single level formula if the value of the capture width is well chosen. Several recent articles have addressed this point. Olsen, et al (2,3,4), have shown that while a multilevel

-42-

Volume 1

Supplement 1

formulation does give a more accurate shape for the total cross section, single level parameters can be defined for the low energy resonances that remove most of the observed errors in U-238 epithermal capture.

De Saussure, et. al., (5) compared a multilevel formula derived from the Reich and Moore formalism to the singlelevel and multilevel Breit-Wigner formulas. They showed that in the energy range from 100 eV to 680 eV the multilevel Breit-Wigner method is a more accurate representation of the U-238 total cross section. However, below 10 eV they conclude that the ENDF/B-IV file, which is based on the singlelevel Breit-Wigner formula, is adequate. Further, they point out that if only one level is considered for a single isolated resonance, / all three formulations become identical.

Staveloz, et al (6), and Liou and Chrien (7) have also discussed the appropriate capture widths to use in a single level formula. While Olsen, et al state that the multilevel formulation will reduce the observed error in epithermal U-238 capture, Bhat (8) has reported that using the reduced capture widths for the low energy resonances also significantly reduces the discrepencies between measured and calculated resonance integrals. Thus, it would appear that either using the multilevel formulation or adjusting the capture widths for the principle resonances in a single level formulation gives accurate results. Because of the simplicity of the formulation and the excellent results for LWR's, ENC has chosen to use the single level formulas.

In the case of the fissile nuclei, the effective resonance integrals are close to the infinite dilute values for typical LWR systems. In these cases the single level formulas are sufficient.

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XN-NF-80-19 (NP)(A) Volume 1

Supplement 1

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-44-

Volume 1

Supplement 1

- Q23. With respect to the geometrical and neutron tracking aspects of XMC, comment on the following questions:
 - (a) What is the difference of beam and particle tracking, when is each applied and how do the results differ?
 - (b) What values and how were they selected for the Russian Roulette parameters?
 - (c) Does the Roulette scheme conserve particle weight? Is the combination of Russian Roulette and weight rationing an unbiased process?
 - (d) Give the FLANGE derived thermal scattering, what order of scattering is treated in standard XME calculations? How is , thermal upscattering beyond the limit of the kernel treated?
 (of particular improtance for P-420 resonance at about 1 eV).
 - (e) From what version of BMC has XMC derived? What are the major modifications, if any?
- A23. (a) Particle tracking is the traditional method for producing Monte Carlo estimates of collision densities and multiplication constants. However, XMC is employed as a tool to evaluate broad group cross sections and power distribution estimates in fuel lattice geometries in addition to multiplication constants. Therefore an efficient flux estimator is required in XMC. This is the function of the neutron beam tracking concept. XMC is capable of treating more than one particle in a beam,

Between collisions, no distinction is made between a beam and a particle; their weight and parameters are treated separately but are numerically identical.

The incident beam intensity for each tally region is used to accumulate estimates of the enrgy dependent flux in that tally region. Following collisions of the particle, the postcollision parameters are evaluated and temporarily stored while the "uncollided" beam is followed and flux tallies accumulated until the beam is terminated by leakage or Russian Roulette. The XMC returns to the particle collision site establishes a new beam with the same attributes as the collided particle and repeats the process. The final beam is followed to its conclusion after the particle is terminated.

-45-

Volume 1

Supplement 1

It is well known that detailed flux and cross section estimation using Monte Carlo techniques requires many more histories than are required to obtain a statistically significant estimate of the multiplication constant. However, by extending each particle trajectory as an "uncollided" beam to its termination as a useful estimator, it is possible to obtain both flux and multiplication results with a reasonable number of histories.

(b) The Russian Roulette parameters used in XMC are as follows:

Particle	e Tracking	Beam Tracking					
Miniumm	Survival	Minimum	Survival				
Weight	Weight	Weight	Weight				

Epithermal Thermal

The values were chosen by trial and error being careful to preserve particle weight and overall neutron balance.

(c) Experience in running XMC shows that the Russian Roulette scheme does conserve particle weight. Check tallies of the weight lost to Russian Roulette kill events are accumulated and edited in all XMC runs.

Weight ratioing is essentially a special form of Russian Roulette biasing. Therefore it can be shown that the combination of Russian Roulette biasing for terminating particles (and/or beams) and weight ratioing for terminating a given fraction of the particle histories as they become thermal constitutes an unbiased process.

(d) XMC uses separate upscatter and downscatter S(α, β) tables at specified temperatures

to evaluate energy and momentum transfers. XMC interpolates linearly among the tabulated data to obtain data at a given velocity and temperature. Since the XMC tabulated scattering law data are taken directly from the ENDF/B scattering law data, the question regarding order of scattering is not applicable.

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(e) XMC was derived from the version of BMC described in BNWL-1433 published in June, 1970.

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The basic methods are unchanged from the original.

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- Q24. With respect to the XMC verification and benchmarking comment on the following questions:
 - (a) Has the XMC been benchmarked and against know critical configuration and especially against gadolinia loaded criticals?
 - (b) Have any adjustments been made to XMC and/or XFYRE regarding the results for K shown on page 188? How many histories were required for the uncertainty shown?
 - (c) What was the void content outside the channel for the XMC, XFYRE comparisons when the channel voids were 32% and 64%.
 - (d) Is XMC used to infer TIP measurements?
- A24. (a) XMC has not been benchmarked against critical experiments.
 - (b) No k-biases have been applied to either the XMC or XFYRE results reported on page 188 of XN-NF-80-19, Vol. I.

- (c)
- (d) XMC is not used to infer TIP measurements or TIP conversion factors.

XN-NF-80-19 (NP[A) Volume 1 Supplement 1

- Q25. The one group neutronic model depends on the effective constants chosen. While the diffusion coefficient approximation has been demonstrated for static conditions (x, y, z) geometry, its applicability is not as obvious for (r, z) geometry under transient conditions. Are the one group cross sections adequate for the range of their intended transient applicability?
- A25. In the axial direction, the diffusion coefficient approximation is the same for cylindrical and rectangular geometry.

The diffusion coefficient approximation utilized in COTRAN in the radial coordinate is:

where

COTRAN generates one-group cross sections from two-group input at the beginning of each time step to reflect temperature, void and control density variations with time. These one-group cross sections are calculated in the following manner:

- Two group parameters are determined at the current void and control state by linear interpolation of input values.
- Two group cross sections are updated for temperature utilizing input derivatives with respect to the square root of absolute temperature.
- The current fast/thermal flux ratio is determined assuming no thermal leakage

$$\frac{\phi_1}{\phi_2} = \frac{{\Sigma_a}^2}{{\Sigma_1} + 2}$$

4) One-group parameters are calculated by flux weighting

$$\Sigma = \frac{\Sigma_1 \phi_1 + \Sigma_2 \phi_2}{\phi_1 + \phi_2}$$

These one-group parameters are then used for the diffusion theory solution and the procedure repeated for the next time step.

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XN-NF-80-19 (NP)(A) Volume 1 Supplement 1

Q26. With respect to the feedback model, comment on the following questions:

- (a) How is the actual control rod pattern converted into an equivalent
 (r, z) control density profile?
- (b) How is a reactor scram simulated?

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- (c) What cross sections are treated as a function of the square root of the fuel temperature for the Doopler feedback and are the void and control fractions accounted for?
- A26. (a) Control rod patterns are converted into an equivalent (r, z) control density profile as follows:
 - The radial regions are defined by assuming specific assemblies in (x, y) geometry to a specific radial region for the (r, z) geometry model.
 - For a given control rod pattern in (x, y, z) geometry, the control fraction (density) is calculated for each of the radial regions defined in (1) above.
 - 3. These control fractions are then input as the average control fraction for a given radial region in (r, z) geometry.
 - (b) Reactor scram is simulated by an instantaneous calculation of the scram bank insertion based on the scram bank velocity and the time increment since initiation of rod movement. This insertion depth is then used to update nodal control rod densities. Cross sections for a time step are based upon linear interpolation between the fully controlled and uncontrolled cross sections for each node.
 - (c) The changes in Σ_{1} and $\Sigma_{1} \rightarrow 2$ with temperature effectively model the Doppler feedback mechanism. The changes in Σ_{1} and $\Sigma_{1} \rightarrow 2$ are input as changes per change in \sqrt{T} since it has been found that they vary essentially linearly with \sqrt{T} over the fuel temperature range of interest (<5,000°F). The Σ_{1} and $\Sigma_{1} \rightarrow 2$ cross sections are calculated at each time step and then collapsed to new one-group values for the next solution. The Doppler feedback is a function of both void and control.

It has been found that it is conservative with respect to the rod drop analysis to ignore the small changes in the other two group cross sections with fuel temperature. This is shown in an example provided in Table 26.1 for the changes in cross sections and $k_{\rm m}$ from

Volume 1

Supplement 1

546°F to 5000°F. The reduction in k is smaller, when assuming that only Σ_{1} and $\Sigma_{1} \rightarrow_{2}$ change with the increase in fuel temperature. In the context of the rod drop analysis which depends on the reduction in k due to the fuel temperature increase to arrest the accident, it is conservative to reduce k by a smaller increment than expected.

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Table 26.1 Fuel Temperature Effects on Two-Group Cross Sections and $\rm K_{\infty}$

Parameters (x 10 ²)	Base Case Cross Sections at 546 F	All Cross Sections at 5,000 °F	$\Sigma_{1\rightarrow2}^{\text{and}}$ at 5,000 °F
vΣrl			
$\sum_{a1} \Sigma_{1+2}$ $\nabla \Sigma_{f2}$			
Σ_{a2}			
K _∞ * ∆K _∞			

XN-NF-80-19 (NP)(A)

Volume 1

Supplement 1

Q27. Referring to the Hydraulic model, comment on the following points:

- (a) On page 78, it is stated that internal heat generation in the fluid is ignored. Yet on page 82, the direct moderator heating is included in the energy equation.
- (b) In the description of the explicit solution scheme, the quantities A and A* are not defined.
- (c) It is stated that the implicit solution scheme includes options for two-phase slip models, void-quality relations and two-phase friction multipliers. Give descriptions or references for these correlations.
- (d) It is not clear whether the code accounts for subcooled boiling. If not, discuss the implication of this approximation.
- A27. (a) For the control rod drop analysis, an adiabatic boundary condition is assumed at the pellet-gap interface and no energy is deposited in the coolant. For the non-adiabatic boundary condition, direct moderator heatings are included in COTRAN as a volumetric heat source in the energy equation.
 - (b) The terms A and A* in the explicit solution scheme are the flow area for node i and the average flow area of nodes i and j, respectively. The * superscript denotes that j = i + 1 depending on the flow direction.
 - (c) The void-quality relation and two-phase friction multipliers utilized in COTRAN are reported in the document "Methodology for Calculations of Pressure Drop in BWR Fuel Assemblies," (XN-NF-79-59), October, 1979.
 - (d) COTRAN accounts for subcooled boiling. The subcooled boiling model is described in the above document.

- Q28. In the derivation of equation 3.3-41 from 3.3-49, what assumptions or approximations have been made (if any)?
- A28. The transformed equation (3.3-41) is derived by replacing the time and radial derivatives of temperature in equation (3.3-39) with equivalent functions of Θ , the conductivity integral. Additional information is shown in Section 3.3.2.4 of Supplement 2 to XN-NF-80-19(P), Volume 1.

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XN-NF-80-19 (NP)(A) Volume 1

Y

Supplement 1

- Q29. What is the potential error in treating the gap and the clad as single regions (linear temperature profile, equation 3.3-49 and the heat balance, equation 3.3-51)?
- A29. A revised temperature model with separate regions for the gap and clad is given in Supplement 2 to XN-NF-80-19(P), Volume 1.

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-56-

Supplement 1

- Q30. In the analysis of the rod drop accident, COTRAN is primarily applicable to a center rod due to its (r, z) geometry. If the maximum worth rod is off center, how is COTRAN utilized? There are three (3) radial zones to simulate control rod distribution. Discuss the adequacy of this approximation in conjunction with the definition and determination of the control fractions.
- A30. The rod drop model consists of a four (4) bundle module where the rod is dropped and two (2) surrounding rings of fuel. The fourbundle module represents the module with the maximum rod worth. More information on modeling of the core for the rod drop accident analysis will be presented in Supplement 2 of XN-NF-80-19(P), Volume 1.

XN-NF-80-19 (NP) (A) Volume 1 Supplement 1

- Q31. Provide a more detailed discussion of the maximum pressure attainable during a rod drop accident.
- A31. Following the unlikely occurrence of a rod drop, an increase in reactor system pressure results due to the deposition of the energy produced during the transient into the coolant. This increase in reactor system pressure is not anticipated to result in penetration of stress limits as defined in Section III of the ASME boiler and pressure vessel code. This is because the total energy deposited and the associated increase in reactor system pressure during a rod drop event is not high relative to other events such as turbine trip without bypass or main steamline isolation valve closure, both of which are quantitatively analyzed.

*

Q32. It is stated that the more conservative static scram reactivity is used in the rod drop analysis. Discuss how the scram curve is utilized to simulate reactor scram.

A32. The statement in the document that the static scram reactivity is used in the rod drop analysis is incorrect. The dynamic scram reactivity is calculated by the COTRAN code as the control rods are inserted into the core. In the rod drop analysis, the performance of the scram system is conservatively modeled. This includes the use of the overpower trip setpoint, and conservatively selecting the scram velocity and delay times.

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XN-NF-80-19 (NP)(A) Volume 1

Supplement 1

- Q33. The specific conditions of the rod drop accident considered for the analysis are not discussed. Provide a description of the accident conditions.
- A33. The primary reactor core variables that affect the deposited enthalpy include the Doppler reactivity, the reactivity worth of the dropped rod, the core power peaking, and the delayed neutron fraction. The rod drop analysis is performed over a range of conditions to bracket the expected values of the above primary variables. The calculated deposited enthalpy is then parameterized as a function of the variables such that for a given reactor cycle, each variable can be calculated and the deposited enthalpy confirmed to be within the established limit.

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- Q31. In addition to question 21 regarding benchmarking and validation, consider the following questions related to XTGBWR verification:
 - (a) How is the K_{eff} value corrected for crud?

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- (b) What data are used and how are they obtained?
- A34. The K values calculated by the XTGBWR code are corrected by a bias to account for crud, spacers, and incore hardware including the LPRM detectors, IRM detectors, SRM detectors, and neutron sources. The K bias is determined by comparing the XTGBWR calculated results with the measured data.

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Volume 1

Supplement 1

- Q35. Equation (6.11) is valid only if the random variables are independent. The same approach is followed and the same assumption is made following the derivation of equation (6.27). Covariance terms would be needed if analysis indicated that the variables are not independent. Such analysis to establish the independence of the variables is not indicated nor is it carried out.
- A35. The variables which are assumed to be independent are those in equations 6.4 and 6.27 show below.

The origins and definitions of the terms in the above two equations indicate that they are either independent of one another or that their dependence is such that the assumption of independence results in a conservative uncertainty estimate.

Supplement 2 to XN-NF-80-19 (p), a correlation coefficient between the variables in Equation 6.27 is added since the assumption of independence is overly conservative in this instance. Q36. In the estimation of the uncertainty (paragraph 6.3) differences between TIP responses in symmetric positions are used to define the system measurement uncertainty. The uncertainty term δ_A is defined to represent all other sources than the radial effects. However, should a fuel assembly misloading or misorientation is present, there are no provisions to either recognize it or avoid incorporating its effects in the value of δ_A .

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A36. The effects of a fuel assembly misloading or misorientation are accounted for by a separate ENC "accident" analysis. The presence of a misloading or misorientation would tend to increase the estimation of both $\delta_{\rm R}$ and $\delta_{\rm A}$, a conservative result.

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XN-NF-80-19(NP)(A) Volume 1 Supplement 2 07/28/81

EXXON NUCLEAR METHODOLOGY FOR

BOILING WATER REACTORS

VOLUME 1

NEUTRONICS METHODS FOR

DESIGN AND ANALYSIS

APPROVED: T. L. Krysinski, Manager Date BWR Neutronics APPROVED: D. H. Timmons, Manager Date Neutronics Development **APPROVED:** 8 JUL 81 R. B. Stout, Manager Date Neutronics & Fuel Management **APPROVED:** . 1 G. F. Owsley, Manager Date Reload Fuel Licensing **APPROVED:** Date G. A. Sofer, Manager Fuel Engineering & Technical Services

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XN-NF-80-19(NP)(A) Volume 1 Supplement 2

Supplement 2 to the Exxon Nuclear Methodology for Eoiling Water Reactors, Volume 1, Neutronic Methods for Design and Analysis, presents supplemental information on the XTGBWR model (Section 3.2), the COTRAN model (Section 3.3), methods verification (Section 5.0), and Uncertainty Analysis (Section 6.0). In addition, Supplement 2 contains Section 7.0 on the Application of Neutronics Methodology.

XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

TABLE OF CONTENTS

i

														Page No.
3.0	NEU	TRONICS	MODELS FOR BWR CORE CA	LCULATIO	ONS									3-1
		3.2.2	Diffusion Theory Mode	1		•								3-2
		3.2.3	Boundary Conditions .		• •	•								3-12
			3.2.3.1 Outer Bounda	ry									•	3-12
			3.2.3.2 Reflected Box	undary .										3-16
			3.2.3.3 Periodic and	Other B	lour	ndar	·у	Cor	nd	iti	or	15		3-16
			3.2.3.4 COTRAN Fuel M	Models .					ŀ					3-17
5.0	NEUT	RONICS	METHODS VERIFICATION .				.,							5-1
	5.1	XFYRE	ERIFICATION					÷						5-1
	5.2	XTGBWF	VERIFICATION											51
	5.3	COTRAN	VERIFICATION											5-2
	5.4	REFERE	CES											5-4
6.0	MEAS	URED PO	ER DISTRIBUTION UNCERT	AINTY .										6-1
	6.1	MEASUR	D POWER DISTRIBUTION D	ETERMIN	ATI	ON								6-2
	6.2	DERIVA POWER	ION OF THE UNCERTAINIY ISTRIBUTION	IN THE	ME	ASU	REI			ľ		ľ		6-9
	6.3	ESTIMA	ION OF UNCERTAINTY .		•									6-14
		6.3.1	Synthesized TIP Distri	bution				1	ŀ	2				6-14
		6.3.2	Calculated TIP Distrib	ution .			Ç							6-20
		6.3.3	Calculated Power Distr	ibution			į,		ł					6-23
		6.3.4	.ocal Pin Distribution				0							6-26
		6.3.5	Bundle Gamma Scan Compa	arisons										6-28
		6.3.6	Summary of the Measured	1 Power	Dis	str	ibu	ti	on					6-31
							1				5 2			0.01

Volume 1, Supple

TABLE OF CONTENTS (Continued)

			Page No.
7.0 AP	APPL	ICATION OF NEUTRONICS METHODOLOGY	7-1
	7.1	CONTROL ROD DROP ACCIDENT	7-2
		7.1.1 Description of Example Problem	7-2
		7.1.2 Range of Parametric Studies	7-3
		7.1.3 Discussion of Typical Results	7-6
		7.1.4 Results of Parametric Studies	7-7
		7.1.5 Example Application of Results	7-8
		7.1.6 General Applicability	7-10
	7.2	FUEL MISLOADING ERROR	7-20
		7.2.1 Fuel Misorientation Error	7-21
		7.2.2 Fuel Mislocation Analysis	7-40
	7.3	REACTOR CORE STABILITY	7-48
		7.3.1 Analysis Model and Conditions	7-48
		7.3.2 Analysis Methodology and Stability Margins	7-49
	7.4	CONTROL ROD WITHDRAWAL	7-54
	7.5	REFERENCES	7-60

XN-NF-80-19(NP)(A)

Volume 1, Supplement 2

LIST OF TABLES

Table No.		Page No.
5.2-1	XTGBWR CALCULATED K FOR DRESDEN-3 CYCLEEF AND AVERAGE VOIDS	5-5
5.2-2	XTGBWR CALCULATED KEEF AND AVERAGE VOIDS	5-6
5.2-3	XTGEWR CALCULATED KEFF AND AVERAGE VOIDS FOR DRESDEN-3 CYCLE	5-7
5.2-4	XTGBWR CALCULATED K FOR DRESDEN-2 CYCLEEF. AND AVERAGE VOIDS	5-8
5.2-5	XTGBWR CALCULATED KEFF AND AVERAGE VOIDS FOR DRESDEN-2 CYCLEEFF	5-9
5.2-6	XTGBWR CALCULATED K QUAD CITIES-1 CYCLEEFF. AND AVERAGE VOIDS FOR	5-10
5.2-7	XTGBWR CALCULATED K QUAD CITIES-1 CYCLEE2F. AND AVERAGE VOIDS FOR	5-11
5.2-8	XTGBWR CALCULATED K QUAD CITIES-1 CYCLE ^E 5 ^F	5-12
5.2-9	XTGBWR CALCULATED K QUAD CITIES-1 CYCLEEF AND AVERAGE VOIDS FOR	5-13
5.2-10	XTGBWR CALCULATED STARTUP KEFF RESULTS FOR DRESDEN-3	5-14
5.2-11	XTGBWR CALCULATED STARTUP KEFF RESULTS FOR DRESDEN-2	5-15
5.2-12	XTGBWR CALCULATED STARTUP KEFF RESULTS FOR	5-16
5.3-1	PEACH BOTTOM-2 EOC2 STABILITY TESTS	5-17
6.1	DATA BASE SUMMARY	6-36
6.2	TIP SYMMETRY	6-37

iii

XN-NF-80-19(NP)(

Volume 1, Supplement 2

LIST OF TABLES (Continued)

Table No.		Page No.
6.3	MEASURED POWER DISTRIBUTION UNCERTAINTY COMPONENTS	6-38
6.4	MEASURED POWER DISTRIBUTION UNCERTAINTY	6-40
7.1-1	COTRAN RADIAL REPRESENTATION	7-11
7.1-2	INITIAL CORE CONDITIONS	7-12
7.1-3	CONTROL ROD DROP VARIABLES	7-13
7.1-4	DELAYED NEUTRON CONSTANTS	7-14
7.1-5	MAXIMUM FUEL ROD ENTHALPY (CALORIES PER GRAM)	7-15
7.2-1	MAXIMUM LOCAL PEAKING FACTOR VERSUS EXPOSURE FOR NON-ROTATED AND 180° ROTATED BUNDLE	7-35
7.2-2	MAXIMUM S-FACTOR VERSUS EXPOSURE FOR NON-ROTATED AND 180° ROTATED BUNDLE	7-36
7.2-3	K-INFINITY VERSUS EXPOSURE FOR NON-ROTATED AND 180 ROTATED BUNDLE	7-37
7.2-4	EFFECT OF MISORIENTATION ON LOCAL PEAKING, S FACTORS AND K	7-38
7.2-5	MAXIMUM LHGR AND AMCPR DUE TO ASSEMBLY MISLOCATION	7-39
7.4-1	CONTROL ROD WITHDRAWAL RESULTS	7-55

iv

XN-NF-80-19(NP)(A)

Volume 1, Supplement 2

LIST OF FIGURES

FIGURE		PAGE
3.3-4	FUEL PIN MODEL	3-28
5.1-12 through 5.1-35	QUAD CITIES-1 EOC2 MEASURED LOCAL POWER DISTRIBUTION/ XFYRE CALCULATED LOCAL POWER DISTRIBUTION/PERCENT DIFFERENCE BETWEEN MEASURED AND CALCULATED	5-18 - 5-41
5.1-36 through 5.1-59	QUAD CITIES-1 EOC3 MEASURED LOCAL POWER DISTRIBUTION/ XFYRE CALCULATED LOCAL POWER DISTRIBUTION/PERCENT DIFFERENCE BETWEEN MEASURED AND CALCULATED	5-42 - 5-65
5.1-60 through 5.1-67	QUAD CITIES-1 EOC4 MEASURED LOWER POWER DISTRIBUTION/ XFYRE CALCULATED LOCAL POWER DISTRIBUTION/PERCENT DIFFERENCE BETWEEN MEASURED AND CALCULATED	5-66 - 5-73
5.2.1	XTGBWR CALCULATED HOT OPERATING Keff FOR DRESDEN 3	5-74
5.2.2	XTGBWR CALCULATED HOT OPERATING Keff	5-75
5.2.3	XTGBWR CALCULATED HOT OPERATING Keff FOR QUAD CITIES-1	5-76
5.2-4 through 5.2-22	COMPARISON OF XTGBWR CALCULATED (-O-) AND MEASURED (-) TIP DATA FOR D3, CYCLE 7	5-77 - 5-95
5.2-23 through 5.2-27	COMPARISON OF XTGBWR CALCULATED (-0-) AND MEASURED (-) TIP DATA FOR D2, CYCLE 7	5-96 - 5-101
5.2-28 through 5.2-39	COMPARISON OF XTGBWR CALCULATED (-0-) AND MEASURED (-) TIP DATA FOR QC1, CYCLE 2	5-102 - 5-113
5.2-40 through 5.2-52	QUAD CITIES-1 EOC2 FUEL ASSEMBLY GAMMA SCAN COMPARISONS, XTGBWR CALCULATED/MEASURED La-140 ACTIVITY	5-114 - 5-126
5.2-53 through	QUAD CITIES-1 EOC4 FUEL ASSEMBLY GAMMA SCAN COMPARISONS, XTGBWR CALCULATED/MEASURED La-140	5 107 5 110
0.2.03		5-127 - 5-143

XN-NF-80-19(NP) (A)

Volume 1, Supplement 2

LIST OF FIGURES (contd.)

FIGURE		PAGE
5.3-1	COTRAN - PEACH BOTTOM 2 LOW FLOW STABILITY TEST - POWER COMPARISON, PERIODIC PRESSURE REGULATOR SETPOINT CHANGES	5-144
5.3-2	COTRAN - PEACH BOTTOM 2 LOW FLOW STABILITY TEST - POWER COMPARISON, RANDOM PRESSURE REGULATOR SETPOINT CHANGES	
5.3-3	PTI-COTRAN TRANSIENT POWER RESPONSE TO 7 PSI RAMP DECREASE IN SYSTEM PRESSURE	5-145
5.3-4	PT3-COTRAN TRANSIENT POWER RESPONSE TO 7 PSI RAMP DECREASE IN SYSTEM PRESSURE	5-140
6.1	LPRM IN-CORE ASSEMBLY CROSS SECTION	6-41
6.2	BWR TYPICAL INSTRUMENT LOCATIONS	6-42
6.3	RADIAL TIP SYMMETRY	6-43
6.4	NODAL TIP SYMMETRY	6-44
6.5	PLANAR TIP SYMMETRY	6-45
6.6	RADIAL TIP SYNTHESIS	6-46
6.7	NODAL TIP SYNTHESIS	6-47
6.8	CALCULATED VS. MEASURED RADIAL TIP DISTRIBUTION	6-48
6.9	CALCULATED VS. MEASURED PLANAR TIP DISTRIBUTION	6-49
6.10	CALCULATED VS. MEASURED LOCAL POWER DISTRIBUTION	6-50
6.11	QUAD CITIES EOC2 BUNDLE GAMMA SCAN RESULTS - COMPARISON OF MEASURED AND CALCULATED Ba-140 DISTRIBUTION	- 6-51
6.12	QUAD CITIES EOC4 BUNDLE GAMMA SCAN RESULTS - COMPARISON OF MEASURED AND CALCULATED Ba-140 DISTRIBUTION	6-52
7.1-1	DETERMINATION OF CONTROL FRACTIONS	7-16
7.1-2	CORE AVERAGE POWER VS. TIME DURING CONTROL ROD DROP ACCIDENT.	7-17

XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

LIST OF FIGURES (contd.)

FIGURE						PAGE
7.1-3	MAXIMUM FUEL ROD ENTHALPY					7-18
7.1-4	RELATIVE ENTHALPY CORRECTION FOR VARIATIONS IN $\overline{\mathfrak{F}}$					7-19
7.2-1	LOCAL POWER DISTRIBUTION FOR NORMAL AND 180° ROTATED BUNDLE FOR A VOID FRACTION OF 0.0 AT 0 MWD/MT		1		į	7-23
7.2-2	S FACTORS FOR NORMAL AND 180° ROTATED BUNDLE FOR A VOID FRACTION OF 0.0 AT 0 MWD/MT					7-24
7.2-3	LOCAL POWER DISTRIBUTION FOR NORMAL AND 180° ROTATED BUNDLE FOR A VOID FRACTION OF 0.0 AT 10,000 MWD/MT.		•			7-25
7.2-4	S FACTORS FOR NORMAL AND 180° ROTATED BUNDLE FOR A VOID FRACTION OF 0.0 AT 10,000 MWD/MT		1			7-26
7.2-5	LOCAL POWER DISTRIBUTION FOR NORMAL AND 180° ROTATED BUNDLE FOR A VOID FRACTION OF 0.4 AT 0 MWD/MT	•	•			7-27
7.2-6	S FACTORS FOR NORMAL AND 180° ROTATED BUNDLE FOR A VOID FRACTION OF 0.4 AT 0 MWD/MT					7-28
7.2-7	LOCAL POWER DISTRIBUTION FOR NORMAL AND 180° ROTATED BUNDLE FOR A VOID FRACTION OF 0.4 AT 0 MWD/MT					7-29
7.2-8	S FACTORS FOR NORMAL AND 180° ROTATED BUNDLE FOR A VOID FRACTION OF 0.4 AT 10,000 MWD/MT					7-30
7.2-9	LOCAL POWER DISTRIBUTION FOR NORMAL AND 180° ROTATED BUNDLE FOR A VOID FRACTION OF 0.7 AT 0 MWD/MT					7-31
7.2-10	S FACTORS FOR NORMAL AND 180°F ROTATED BUNDLE FOR A VOID FRACTION OF 0.7 AT 0 MWD/MT					7-32
7.2-11	LOCAL POWER DISTRIBUTION FOR NORMAL AND 180° ROTATED BUNDLE FOR A VOID FRACTION OF 0.7					
	n 10,000 mm0/m1	*		*	*	/-33

vii

XN-NF-80-19(NP)(A)

Volume 1, Supplement 2

LIST OF FIGURES (contd.)

FIGURE	PAGE	
7.2-12	S FACTORS FOR NORMAL AND 180° ROTATED BUNDLE FOR A VOID FRACTION OF 0.7 AT 10,000 MWD/MT	
7.2-13	ASSEMBLY ARRANGEMENT USED IN ESTIMATING MISLOCATED ASSEMBLY POWER	;
7.2-14	COMPARISON OF ASSEMBLY RELATIVE POWER AND MCPR FOR THE CORRECTLY LOADED AND MISLOADED CORE AT THE WORST CORE LOCATION	
7.3-1	POWER FLOW MAP	
7.3-2	DECAY RATIO VS. REACTOR POWER	
7.3-3	RATED POWER/FLOW, NATURAL CIRCULATION INTERSECTION - COTRAN TRANSIENT POWER RESPONSE TO 4 PSI RAMP DECREASE IN SYSTEM PRESSURE	\$
7.4-1	STARTING CONTROL ROD PATTERN FOR CONTROL ROD WITHDRAWAL ANALYSIS	;
7.4-2	AMCPR AS A FUNCTION OF CONTROL ROD POSITION	
7.4-3	RBM RESPONSE AS A FUNCTION OF CONTROL ROD POSITION, CHANNEL A+C	3
7.4-4	RBM RESPONSE AS A FUNCTION OF CONTROL ROD POSITION, CHANNEL B+D	,
Volume 1, Supplement 2

1

3.0 NEUTRONICS MODELS FOR BWR REACTOR CORE CALCULATIONS

The following pages present supplemental information on Section 3.0. Three subsections, 3.2.2, 3.2.3 and 3.3.2.4, are effected. Each subsection is fully reproduced with supplemental description and equations being identified by a vertical line in the right hand margin.

*

3.2.2 Diffusion Theory Model

The XTGBWR program uses a modified coarse mesh two energy group diffusion theory model for steady state analysis of the reactor core. The model is designed to accept void and exposure dependent two group cross sections. The cross sections can be specified on a nodal basis allowing axial and radial effects to be modeled. The conditions under which the cross sections were generated are input, and the XTGBWR code utilizes this information to adjust the cross sections to fit the actual reactor conditions in each node. This includes adjustment for control rods, instantaneous void, void history, power dependent Doppler, and time and power dependent xenon and samarium.

Using standard notation, the basic diffusion theory

equation is

$$-D_{g}\nabla^{2}\phi_{g} + \Sigma_{g}^{R}\phi_{g} = \frac{\chi_{g}}{k_{eff}} \sum_{g'=1}^{G} (\nabla\Sigma_{f})_{g'}\phi_{g'} + \sum_{g'=1}^{g-1} \Sigma_{s1}(g' \neq g)\phi_{g'} \quad (3.2-1)$$

í

Assuming all neutrons are born in the fast group, the two group diffusion equations are

$$-D_{1}\nabla^{2}\phi_{1} + \Sigma_{R_{1}}\phi_{1} = \frac{1}{k_{eff}} (v_{1}\Sigma_{f_{1}}\phi_{1} + v_{2}\Sigma_{f_{2}}\phi_{2}) \qquad (3.2-2)$$

$$-D_{2}\nabla^{2}\phi_{2} + \Sigma_{a_{2}}\phi_{2} = \Sigma_{s1} (1 \rightarrow 2) \phi_{1}$$
(3.2-3)

These equations are integrated over the volume of a three-dimensional node. To evaluate the leakage term, the volume integral over the Laplacian is changed to a surface integral using Green's theorem

$$\int D\nabla^2 \phi \, dV = \int D \overline{\nabla} \phi \cdot d\overline{A} \tag{3.2-4}$$

Using mesh points at the node centers, the volume integration of equation (3.2-2) yields

$$\sum_{k=1}^{6} \frac{\overline{D}_{k}A_{k}}{d_{k}} (\phi_{k} - \phi_{0}) - \Sigma_{R_{0}}\overline{\phi}_{0}V_{0} = -S_{0}V_{0}\overline{\phi}_{0}$$
(3.2-5)

where 0 refers to the node being calculated and k to the six nearest neighbors shown in Figure 3.2-3. For convenience, the subscript 1 referring to the fast group has been omitted. The notation is as follows:

$$\Sigma_{R_0} = \text{removal cross section} = \Sigma_{s1}(1 \rightarrow 2) + \Sigma_{a_1}$$

$$S_0 = \frac{1}{k_{eff}} \left(\nabla \Sigma_{f_1} + \nabla \Sigma_{f_2} \overline{\phi_2} / \overline{\phi_1} \right)$$

$$V_0 = \text{volume of node}$$

- d_{ν} = distance between mesh point k and mesh point 0
- A_k = area of boundary between mesh point k and mesh point 0
- \overline{D}_k = effective diffusion coefficient between mesh point k and and mesh point 0

$$\overline{D}_{k} = \frac{D_{0}D_{k}(\delta R_{0} + \delta R_{k})}{D_{0}\delta R_{k} + D_{k}\delta R_{0}}$$
(3.2-6)

 δR_0 , δRk = node size in direction of calculation

If h_{χ} is the mesh spacing in both the x and y directions and h_{χ} is the mesh spacing in the z direction, then

$$\frac{d_k}{d_k} = h_z \text{ in x,y direction}$$
(3.2-7)

$$\frac{A_k}{d_k} = \frac{h_x^2}{h_z} \text{ in z direction}$$
(3.2-8)

$$\overline{D}_{k} = \frac{2D_{0}D_{k}}{D_{0} + D_{k}}$$
(3.2-9)

$$V_0 = h_{\chi}^2 h_{\chi}$$
 (3.2-10)

Equation (3.2-5) becomes

$$\sum_{k} \frac{2D_{k}D_{0}R_{k}}{D_{k} + D_{0}} (\phi_{k} - \phi_{0}) = -\overline{S}_{0}h_{x}^{2} \overline{\phi}_{0} + \Sigma_{R_{0}}h_{x}^{2} \overline{\phi}_{0}$$
(3.2-11)

where

$$R_k = 1$$
 if k in x,y direction (3.2-12)
= $\frac{h_x^2}{h_z^2}$ if k in z direction

With minimal error, $\overline{\mathrm{D}}_k$ can be approximated by

$$\frac{2D_k D_0}{D_k + D_0} = \sqrt{D_k} \sqrt{D_0}$$
(3.2-13)

with the additional definitions

$$\psi_{j} = \phi_{j} / D_{j} \quad j=0,k$$
 (3.2-14)

$$\overline{\psi}_0 = \overline{\phi}_0 \sqrt{D_0} \tag{3.2-15}$$

$$P_0 = \sum_{k} R_k \sqrt{D_k} / \sqrt{D_0} + \Sigma_{R_0} h_x^2 / D_0$$
(3.2-16)

and with some algebraic manipulation, Equation (3.2-11) can be written as

$$P_{0}\psi_{0} - \sum_{k} R_{k}\psi_{k} = \frac{S_{0}}{D_{0}} h_{x}^{2}\overline{\psi_{0}} - \frac{\Sigma_{R_{0}}}{D_{0}} h_{x}^{2}(\overline{\psi_{0}} - \psi_{0}) \qquad (3.2-17)$$

 $\overline{\phi_0}$ is a weighted average flux for node O calculated from the mid-point fluxes of node O plus the six surrounding k nodes. Specifically $\overline{\phi_0}$ is calculated from

$$\overline{\phi}_0 = b\phi_0 + 2c \sum_k R_k \phi_k^0$$
 (3.2-18)

5 3

where

$$b = \frac{3 * AFA}{3 * AFA + (1 - AFA)(R + 2)}$$
(3.2-19)

$$c = \frac{1 - AFA}{4 * (3 * AFA + (1 - AFA)(R + 2))}$$
(3.2-20)

 ϕ_k^0 = the flux on the interface between node 0 and node k and is derived using continuity of current

$$\phi_{k}^{0} = \frac{\psi_{0}}{2\sqrt{D_{k}}} + \frac{\psi_{k}}{2\sqrt{D_{0}}}$$
(3.2-21)

AFA = the weighting factor for the mid-point fast flux. Using equations (3.2-18) and (3.2-21)

$$\overline{\psi}_0 = (b + c * r_0)\psi_0 + c \sum_k R_k \psi_k$$
 (3.2-22)

where

$$r_{0} = \sqrt{D_{0}} \sum_{k} R_{k} / \sqrt{D_{k}}$$
(3.2-23)

The numerical solution is obtained by rewriting equation (3.2-17)

$$\psi_0 = \left(\sum_{k} R_k \psi_k + T_0\right) / P_0$$
 (3.2-24)

where

$$T_{0} = \frac{S_{0}\overline{\phi}_{0}h_{x}^{2}}{\sqrt{D_{0}}} + \frac{h_{x}^{2}}{D_{0}} \Sigma_{s1} (1+2) + \Sigma_{a_{1}} (\psi_{0} - \overline{\psi}_{0})$$

1

with

 $\overline{\phi}_0$ = average nodal fast flux from previous iteration $\overline{\psi}_0$ = $\overline{\phi}_0 \sqrt{D_0}$

 ψ_0 = nodal center point ψ from previous iteration. Equation (3.2 -24) is used to iterate on the fast flux.

The thermal flux in a node is evaluated by considering the thermal absorption in the node as being composed of two components:

- a Absorption of neutrons which slow down within the node (volumetric term).
- b Absorption of thermal neutrons which leak into the node from neighboring nodes (surface term).

The total absorption of thermal neutrons in the node is estimated by superposition of the two components.

Absorption = $\Sigma_{a_2} \overline{\phi}_2$

$$\Sigma_{a_2} \left[\left(\overline{\phi}_2 \right)_v + \sum_{k} \left(\overline{\phi}_2 \right)_k^k \right]$$
 (3.2-25)

where

$$(\overline{\phi}_2)_{V} = \overline{\phi}_1 \cdot \frac{\Sigma_{s1}(1 \rightarrow 2)}{\Sigma_{a_2}}$$

with

 $\overline{\phi}_1$ = average fast flux in the node $\overline{\phi}_2$ = average thermal flux in the node $(\overline{\phi}_2)_V$ = "volumetric" component of thermal flux $(\overline{\phi}_2)_k^k$ = "leakage" component of thermal flux (kth/_L surface 1 contribution).

Thus, the thermal flux is assumed to be approximately asymptotic to the fast flux with a thermal leakage correction applied to improve the estimate.

The leakage component is treated by considering two adjacent nodes for which the thermal current at the mode interface, j_k^0 , is assumed to be known.

Volume 1, Supplement 2

Volume 1, Supplement

A new eigenvalue (k_{eff}) is calculated after each outer iteration. This eigenvalue and updated values of $\overline{\phi}_1$ and $\overline{\phi}_2$ are used to compute a source term and the inner iterations are repeated. After each ten or fewer outer iterations, the cross sections are updated to account for the power distribution effects of thermal hydraulic feedback, Doppler broadening and xenon. These changes to the cross sections are described under the respective headings in the following sections of this report. This procedure of inner and outer iterations and cross section updating continues until convergence or the specified maximum iterations are reached, whichever is sooner. The power distribution in each node is calculated by:

$$P = (\kappa \Sigma_{f_1} \overline{\phi}_1 + \kappa \Sigma_{f_2} \overline{\phi}_2)$$
(3.2-29)

3.2.3 Boundary Conditions

3.2.3.1 Outer Boundary

The outer boundary conditions determine the leakage from the core. XTGBWR utilizes an extrapolation distance at which the fast flux goes to zero to determine the flux profile and the leakage of fast neutrons from the nodes on the core-reflector interface. The extrapolation distance is calculated separately for each boundary node and is based on input "reflector" cross section data which represents the neutron diffusion (material) properties of reflector nodes found at the top, bottom, and periphery of the core boundary.

3.2.3.2 Reflected Boundary

The zero current boundary condition is achieved by simply setting $\phi_k = \phi_0$ in Equation (3.2-5) for a reflected node.

3.2.3.3 Periodic and Other Boundary Conditions

Periodic and other boundary conditions are achieved by setting the flux node value for node k in Equation (3.2-5) to the correct value when a node is a boundary node.

3.2.3.4 COTRAN Fuel Models

There are two fuel models currently available in the COTRAN code, TEMP and TEMPFD. TEMP calculates the internal temperature distribution of the fuel rod and the surface heat flux to the adjacent fluid channel during the steady state and transient. TEMP utilizes the Orthogonal Collocation Method of Weighted Residuals in the radial coordinate and finite difference for the temporal derivative. TEMPFD calculates the internal temperature and enthalpy distribution of the fuel rod and the surface heat flux to the adjacent fluid channel during steady state. During the transient, TEMPFD imposes an adiabatic boundary condition at the fuel-clad interface and calculates the temperature and enthalpy distribution of the fuel rod. TEMPFD utilizes finite difference in space and time to solve the heat conduction equation.

Both temperature models include temperature dependent fuel thermal conductivity. TEMPFD also includes variable fuel heat capacity to accurately predict enthalpy and temperature changes from initial conditions to the fuel melting point.

Transient Heat Conduction Model, TEMP Fuel Interior

The fundamental heat conduction equation, assuming angular symmetry and neglecting axial conduction, can be written for cylindrical geometry as:

$$pc \quad \frac{\partial T}{\partial t} = \frac{1}{R^2 r} \quad \frac{\partial}{\partial r} r \ k(T) \ \frac{\partial T}{\partial r} + q'' \qquad 3.3.38$$

Volume 1, Supplement :

where

k(T)	=	local thermal conductivity
q'''	=	uniform volumetric heat source
r'	=	radial coordinate
R	=	pellet radius
r	=	r'/R

Making use of Kircoff's

Transformation,

$$\theta = \frac{1}{k_0} \int_{T_0}^{T} k(T') dT' = G(T)$$
 3.3.39

where k_0 is the conductivity at the reference temperature T_0 , the temporal and radial derivatives of θ can be derived as:

$$\frac{\partial \theta}{\partial t} = \frac{k(T)}{k_0} \frac{\partial T}{\partial t}$$
 3.3.40

and

$$\frac{\partial \theta}{\partial r} = \frac{k(T)}{k_0} \quad \frac{\partial T}{\partial r} \qquad 3.3.41$$

Using these equations to substitute for the temporal and radial derivatives of temperature in equation 3.3.38 yields

$$\int c \frac{k_0}{k(T)} \frac{\partial \theta}{\partial t} = \frac{k_0}{R^2 r} \frac{\partial}{\partial r} r \frac{\partial \theta}{\partial r} + q''' \qquad 3.3.42$$

which is solved by the orthogonal collocation method of weighted residuals. In this method, a polynomial expansion for θ is substituted into the differential equation and solved at N interior collocation points and at the

1 2

pellet surface r = 1.0. Thus, θ is expanded as a symmetric polynomial,

$$\theta(r) = d_1 + d_2 r^2 + d_3 r^4 + \dots + d_{N+1} r^{2N} = \sum_{i=1}^{N+1} (r^{2i-2}) di$$
 3.3.43

where di are the unknown coefficients of the expansion.

Evaluating this polynomial at the N+1 collocation points defined by Finlaysen $\binom{(3-19)}{gives}$

$$\theta$$
 (rj) = $\sum_{i=1}^{N+1} r_j^{2i-2} di$ 3.3.44

or, in matrix notation,

 $|\theta| = [Q] \{d\}$; where $Q_{ji} = r_j^{2i-2}$

From equation 3.3.43, the first

and second radial derivatives can be calculated as

$$\frac{d\theta}{\partial r} |_{r_j} = \sum_{i=1}^{N+1} (2i-2) r_j^{2i-3} di \qquad 3.3.45$$

and,

$$\frac{1}{r} \frac{\partial}{\partial r} r \frac{\partial \theta}{\partial r} \Big|_{r_j} = \sum_{i=1}^{N+1} (2i-2)^2 r_j^{(2i-4)} d_i \qquad 3.3.46$$

which, in matrix notation, is

$$\frac{\partial}{\partial r} \left| \theta \right| = [C] [Q]^{-1} \left\{ \theta \right\}$$
 3.3.47

and

$$\frac{1}{r} \quad \frac{\partial}{\partial r} \quad r \quad \frac{\partial}{\partial r} \quad \left| \theta \right| = \left[D \right] \left[Q \right]^{-1} \quad \left| \theta \right| \qquad 3.3.48$$

r 5

where

 $\begin{bmatrix} Q \end{bmatrix}^{-1} | \theta | = \{d\} \\ C_{ji} = (2i-2) r_j^{2i-3} \\ D_{ji} = (2i-2)^2 r_j^{2i-4} \end{bmatrix}$

Substituting equation 3.3.48 in equation 3.3.42 yields

$$\rho c \frac{k_0}{k(T)} \frac{\partial \theta}{\partial t} = \frac{k_0}{R^2} [B] |\theta| + q''' \qquad 3.3.49$$

where $[B] = [D] [Q]^{-1}$

Approximating the time derivative by a forward finite difference produces the heat conduction equation at the N+1 interior nodal positions:

$$\frac{\rho c k_0}{\Delta t k_i} \theta_i - \frac{k_0}{R^2} \sum_{\ell=1}^{N+1} B_{i\ell} \theta_{\ell} = \frac{\rho c k_0}{\Delta t k_i} \theta_i^n + q''' \qquad 3.3.50$$

Fuel-Clad Interface

The imposed boundary condition

at the fuel surface is:

$$\frac{-k(T)}{R} \frac{\partial T}{\partial r} \Big|_{R} = Hg \frac{R_{N+2}}{R} (T_{N+1} - T_{N+2}) \qquad 3.3.51$$

where

 T_{N+1} = Fuel surface temperature T_{N+2} = Clad interior surface temperature Hg = Fuel-clad gap conductance R_{N+2} = R_{CI} = Clad interior radius

5 5

Utilizing the conductivity integral for the radial coordinate (3.3.41) yields

$$\frac{-k_{0}}{R} \frac{\partial \theta}{\partial r} \Big|_{R} = Hg \frac{R_{N+2}}{R} (T_{N+1} - T_{N+2}) \qquad 3.3.52$$

Substituting equation 3.3.47 gives

 $[A] = [C] [Q]^{-1}$

$$\frac{-k_{o}}{R_{N+2}} \sum_{\ell=1}^{N+1} A_{N+1,\ell} = Hg (T_{N+1} - T_{N+2})$$
 3.3.53

where

Fuel Cladding

The transient heat conduction

equation for the clad region can be written as:

 $\rho c \quad \frac{\partial T}{\partial t} = \frac{1}{r'} \frac{\partial}{\partial r'} r' k(T) \frac{\partial T}{\partial r'}$

In COTRAN this equation is solved

by conventional finite difference techniques for the clad interior surface, centerline and exterior surface temperatures. The resultant difference equations are:

Interior Surface,

$$pc \frac{\partial T}{\partial t} = \frac{4 k_{N+5/2}}{Y_c^2} (2+Yc/2R_{N+2})(T_{N+3}-T_{N+2}) + \frac{4 Hg}{Y_c} (T_{N+1}-T_{N+2})$$
 3.3.54

where

 $k_{N+5/2} = (k (T_{N+2}) + k (T_{N+3}))/2$ $R_{N+3} = Clad centerline radius$ $Y_{C} = Clad thickness$

2

3.3.

3.3.5

Clad Centerline

$$\rho c \frac{dT}{dt} = \frac{4}{Yc^2} \left\{ K_{N+7/2} \left(1 + \frac{Yc}{4R_{N+3}} \right) \left(T_{N+4} - T_{N+3} \right) - K_{N+5/2} \left(1 - \frac{Yc}{4R_{N+3}} \right) \left(T_{N+3} - T_{N+2} \right) \right\} \quad 3.3.$$

where
$$k_{N+7/2} = (k(T_{N+3}) + k(T_{N+4}))/2$$

Clad Exterior Surface i=N+4

$$\rho c \frac{dT}{dt} = -\frac{8}{Yc^2} \left(1 - \frac{Yc}{4R_{N+4}}\right) k_{N+7/2} \left(T_{N+4} - T_{N+3}\right) - \frac{4}{Vc} \frac{H_{surf}}{Yc} \left(T_{N+4} - T_{w}\right) \qquad 3.3.$$

where

 R_{N+4} = Clad exterior radius H_{surf} = Surface heat transfer coefficient_F. T_w = Bulk fluid temperature

Substituting equation 3.3.40 and approximating the time derivative by a forward finite difference, equations 3.3.54 through 3.3.56 become:

$$\frac{pc \ k_{0}}{k_{N+2}} \frac{\Theta_{N+2}}{\Delta t} = \frac{pc \ k_{0}}{k_{N+2}} \frac{\Theta_{N+2}^{n}}{\Delta t} + \frac{4 \ k_{N+5/2}}{Yc^{2}} (2 + Yc/2R_{N+2})(T_{N+3}-T_{N+2}) + \frac{4 \ Hg}{Yc} (T_{N+1}-T_{N+2})$$

$$3.3.$$

$$\frac{\rho c}{k_{N+3}} \frac{\Theta_{N+3}}{\Delta t} = \frac{\rho c}{k_{N+3}} \frac{\Theta_{N+3}^{n}}{\Delta t} + \frac{4}{Yc^{2}} \left\{ k_{N+7/2} \left(1 + \frac{Yc}{4R_{N+3}} \right) (T_{N+4} - T_{N+3}) - k_{N+5/2} \left(1 - \frac{Yc}{4R_{N+3}} \right) (T_{N+3} - T_{N+2}) \right\}$$

Clad Exterior Surface

Clad Contauling

$$\frac{\rho c}{k_{N+4}} \frac{\Theta_{N+4}}{\Delta t} = \frac{\rho c}{k_{N+4}} \frac{\Theta_{N+4}^{H}}{\Delta t} - \frac{8}{Yc^{2}} \left(1 - \frac{Yc}{4R_{N+4}}\right) k_{N+7/2} \left(T_{N+4} - T_{N+3}\right) - \frac{4}{Vc} \frac{H_{surf}}{Yc} \left(T_{N+4} - T_{W}\right)$$

Equations 3.3.57, 3.3.58 and

3.3.59 provide the finite difference equations for the clad region. The implicite temperatures T_{N+1} , T_{N+2} , T_{N+3} and T_{N+4} are evaluated by a truncated Taylor series as:

$$T_{i} = \hat{T}_{i} - \frac{G(\hat{T}_{i}) - \Theta(T_{i})}{G'(\hat{T}_{i})}$$
 3.3.60

where G is defined in equation 3.3.39, G' is the derivative of G with respect to T, and \hat{T}_i is the temperature solution at node i for the previous iteration.

Solution Scheme

Equations 3.3.57 through 3.3.60

are combined with the fuel-clad interface equation (3.3.53) and the differential heat conduction equation (3.3.50) to yield a matrix equation $[a] \{e\} = [Q]$

for the transformed temperature, θ . These matrices are solved by an iterative Gauss-Siedel procedure. Once θ is determined, the temperature solution is evaluated by equation 3.3.60.

Adiabatic Transient Fuel Model, TEMPFD

The adiabatic transient fuel mode

solves the heat conduction equation for an average fuel rod with 8 equal volume fuel nodes and, for the steady-state, two clad nodes as shown in Figure 3.3-4. Prior to initialization of the transient, the admittance to the clad is set to zero, thereby decoupling the fuel conduction equations

7 5

from the clad and moderator region. A steady-state energy balance on each fuel node yields:

$$Q_N - Y_i (T_i - T_{i+1}) + Y_{i-1} (T_{i-1} - T_i) = 0$$
 3.3.61

Similarly, for the clad nodes:

$$Y_{i-1}(T_{i-1}-T_i) - Y_i(T_i-T_{i+1}) = 0$$
 3.3.62

where

Referring to Figure 3.3-4, the radii are given by:

$$R_{1} = R_{F}/2 \sqrt{2}$$

$$R_{2} = R_{F}/2$$

$$R_{3} = R_{F} \sqrt{3/8}$$

$$R_{4} = R_{F} \sqrt{2}$$

$$R_{5} = R_{F} \sqrt{5/8}$$

$$R_{6} = R_{F} \sqrt{3}/2$$

$$R_{7} = R_{F} \sqrt{3}/2$$

$$R_{8} = R_{F}$$

$$R_{9} = (R_{CI} + R_{C0})/2$$

$$R_{10} = R_{C0}$$

$$R_{F} = Fuel radius$$

$$R_{CI} = Clad inside radius$$

$$R_{C0} = Clad outside radius$$

1 2

The heat conduction path lengths are:

$$\Delta R_{12} = (R_1 + R_2)/2$$

$$\Delta R_{i,i+1} = (R_{i+1} - R_{i-1})/2 \qquad 2 \le i \le 7$$

$$\Delta R_{8q} = R_F - \sqrt{7/8} R_F$$

The internode heat transfer areas are:

$$A_{i,i+1} = 2\pi R_i \Delta x$$

$$A_{8g} = 2\pi R_F \Delta x$$

$$A_{9c} = 2\pi R_{C0} \Delta x = 2\pi R_{10} \Delta x$$

where $\Delta x = node height$

The admittances for the fuel nodes

are derived from the thermal conductivity (k_i) , the internode heat transfer areas, and the heat conduction path lengths:

$$r_i = \frac{\kappa_i A_i, i+1}{\Delta R_{i,i+1}}$$
 $1 \le i \le 7$ 3.3.63

The admittance of the node of more than one material is:

$$r_{8} = \frac{1}{\frac{\Delta R_{8g}}{K_{8} A_{8g}} + \frac{1}{H_{g} A_{8g}} + \frac{R_{8} \ln R_{9}/R_{CI}}{A_{8g} k_{c}}}$$

where Hg = Fuel-clad gap coefficient

k_c = Clad thermal conductivity

$$9 = \frac{\frac{1}{R_{10} \ln R_{10}/R_{9}}}{\frac{R_{10} \ln R_{10}}{A_{9c} \ kc}}$$

and $Y_{10} = H_{surf} A_{9c}$

where H_{surf} is the surface heat transfer coefficient.

With these definitions, equations 3.3.61 and 3.3.62 are solved for the steady-state temperature distribution. During the transient, a time-dependent form of equation 3.3.61 is used for the fuel nodes, i.e.,

$$P_i V_i \frac{\partial H_i}{\partial t} = Q_N - Y_i (T_i - T_{i+1}) + Y_{i-1} (T_{i-1} - T_i) \frac{1 \le i \le 8}{1 \le 3.3.64}$$
 3.3.64

where p_i = Density of node i V_i = Volume of node i H_i = Enthalpy of node i Y_i = Thermal admittance of node i (equation 3.3.63) Y_8 = 0.0 during transient

Thus, the clad and moderator temperatures do not enter into the transient fuel temperature distribution. The time derivative of enthalpy is approximated by

$$\frac{\partial H_i}{\partial t} = \frac{\Delta H}{\Delta t}$$

The fuel temperature for each node at the end of the time step Δt is determined from the following:

$$T = T^{n} + \Delta H/C_{f} \qquad 3.3.65$$

where C_f , the fuel heat capacity, will be evaluated as a function of the fuel temperature of the node by the following:

 $C_f = 0.03868 + TA [.08345 + TA (-.0624 + TA (.01806 - .00159 TA))]$ where TA = $T_f/1000$.

It should be noted that when the contained enthalpy of the fuel reaches a value (269 cal/gm) corresponding to the heat content at which the fuel (UO_2) begins to melt, the fuel temperature remains constant during the entire melting phase of the fuel. When the fuel is completely melted, the value for C_f is taken to be 0.1244.



Figure 3.3-4 Fuel Pin Model

Vol. 1, Supplement 2

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5.0 NEUTRONICS METHODS VERIFICATION

Section 5.0 supplements the methods verification data presented in XN-NF-80-19(P), Volume 1. Comparisons between calculations and measurements include the Quad Cities-1 gamma scan measurements and Dresden-3, Dresden-2, and Quad Cities TIP data. In addition, the XTGBWR and COTRAN results have been updated to reflect the code improvements described in Sections 3.2 and 3.3 of this supplement.

5.1 XFYRE VERIFICATION

Comparisons of XFYRE calculations with the Oyster Creek gamma scan results, the Garigliano isotopic measurements, and the XMC Monte Carlo results are presented in XN-NF-80-19(P), Volume 1. The Quad Cities-1 end of Cycle $2^{(5-2)}$, end of Cycle $3^{(5-4)}$, and end of Cycle $4^{(5-5)}$, fuel rod gamma scan measurements have been compared to the XFYRE calculated fuel rod powers and the results are shown in Figures 5.1-12 through 5.1-67. The Quad Cities gamma scan measurements were performed by removing fuel rods from the fuel assembly and measuring the La-140 activity as a function of core height. In that the tie rods and water rods were not gamma scanned, the measured and calculated powers appear as zero in Figures 5.1-12 through 5.1-67.

5.2 XTGBWR VERIFICATION

The XTGBWR reactor core simulator code is verified by comparing the calculated and measured reactor parameters. The reactor core follow data for the Dresden-3, Dresden-2, and Quad Cities reactors is presented Tables 5.2-1 through 5.2-9. The hot operating K_{eff} values calculated by

Vol. 1, Supplem

XTGBWR are plotted as a function of cycle exposure on Figures 5.2-1 through 5.2-3. The XTGBWR calculated startup critical data is shown on Tables 5.2-10, 5.2-11 and 5.2-12. The K_{eff} data are corrected for reactivity biases including the effects of "crud", incore instruments, sources, and fuel assembly spacers.

A comparison of measured and calculated traveling in-core probe (TIP) data for the Dresden-3, Dresden-2, and Quad Cities-1 reactors is presented in Figures 5.2-4 through 5.2-39. The data shown is typical for beginning of cycle, middle of cycle, and end of cycle for each of the three reactors.

The Quad Cities-1 end of Cycle $2^{(5-2)}$ and end of Cycle $4^{(5-5)}$ fuel assembly gamma scan measurements have been compared to be XTGBWR calculated nodal powers. The calculated/measured results for Cycle 2 are shown on Figures 5.2-40 through 5.2-52. The Cycle 4 results are shown on Figures 5.2-53 through 5.2-69. The measured data is La-140 activity. The XTGBWR calculated nodal powers were converted to La-140 activity for the comparisons.

5.3 COTRAN VERIFICATION

The reactor kinetics calculations performed by the COTRAN code are compared to the Peach Bottom-2 transient measurements (5-3). A comparison of the measured and calculated relative power response for the periodic step change in the pressure regulator setpoint is shown in Figure 5.3-1. The measured and calculated data for the random pressure

Vol. 1, Supplement 2

regulator setpoint changes are shown in Figure 5.3-2. For both of the comparisons, the measured reactor pressure response was input into the COTRAN calculations as a forcing function.

The primary goal of the step and periodic pressure regulator tests at Peach Bottom-2 was to determine the core stability margins at several operating points. The reactor core stability margin was determined from an empirical model fitted to the experimentally derived transfer function measurement between core pressure and the average neutron flux signals (APRM). The stability margins for the pressure tests are reported as decay ratios (X_2/X_0) . The decay ratio is defined as the ratio of the magnitudes of successive maxima, or minima, of the transient response to a step perturbation. In this case the perturbation is core pressure.

The determination of the core reactivity decay ratio can be determined from a COTRAN transient by initiating the transient with a rapid change in core pressure or control rod position. The resultant time response is analyzed to determine the magnitude of successive power oscillations and the decay ratio for the operating state, as described in Section 4.3 Reference 7-1.

The decay ratios reported for the above stability tests exhibit a large degree of stability in the power-to-void closed-loop response with the maximum decay ratio being 0.34 for PT3. The relative power responses, as calculated by the COTRAN code, are presented in Figures 5.3-3 and 5.3-4 for stability transients conducted at PT1 and PT3 initial conditions. A comparison of the calculated and measured decay ratios for these two points are presented in Table 5.3-1.

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5.4 REFERENCES

- 5-1 A. Ariemma, et al., "Experimental and Theoretical Determination of Burnup and Heavy Isotope Content in a Fuel Assembly Irradiated in the Garigliano Boiling Water Reactor", EUR 4638, 1971.
- 5-2 M. B. Cutrone and G. F. Valby, "Gamma Scan Measurements at Quad Cities Nuclear Power Station Unit 1 Following Cycle 2", EPRI NP-214, July 1976.
- 5-3 L. A. Carmichael and R. O. Niemi, "Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2", EPRI NP-564, June 1978.
- 5-4 D. W. Merth and B. A. Zolotar, "Gamma Scan Measurements of Quad Cities Nuclear Power Station Unit 1 Following Cycle 3", EPRI NP-512, July 1977.
- 5-5 Personal Communication from B. A. Zolotar (EPRI) to J. S. Holm (ENC), February 1981.

Tables 5.2-1 through 5.2-5 are proprietary and have been deleted.

XN-NF-80-19(NP)(

Vol. 1, Sup

1

Cycle Exposure MWD/MTU	Power MWt	10 ^{Elow}
292.3	2185	84.39
/12.1	2235	99.61
881.9	2240	94.65
1470.6	2197	97.58
2238.9	2450	97.97
3190.2	2414	95 30
3836.2	* 2197	94 84
4074.3	2320	94 72
4736.8	2377	02 03
5301 6	2227	00.05
6031 3	2014	50.95
6550 2	2014	/3.50
6007.2	2225	97.89
0807.3	2210	94.14
/3//.0	2267	95.62
7659.4	2187	97.73
8060.7	2203	95,94

Table 5.2-6 XTGBWR Calculated K and Average Voids for Quad Cities-1 Cycle 1

Table 5.2-7 XTGBWR Calculated K and Average Voids for Quad Cities-1 Cycle 2

Cycle Exposure MWD/MTU	Keff	Average Voids	Power MWt	10 ^{Elow}
677.9 1502.5 1855.2 2886.9 5609.5 5911.5 6324.5 6454.5			2286 2412 2500 2463 1829 1713 1547 1487	88.01 97.42 96.68 94.23 93.87 94.10 95.70 94.90

XN-NF-80-19(NP)(

Vol. 1, Supplem

1

Table 5.2-8 XTGBWR Calculated K and Average Voids for Quad Cities-1 Eycle 3

Cycle Exposure MWD/MTU	Keff	Average Voids	Power MWt	10 ^{Elow}
445.3			2441	96.30
1989.6			2423	98.20
2783.2			2445	98.40
3753.2			2190	97.60
4221.1			2126	96.80
			F 1	7.7
Vol. 1, Supplement 2

Cycle Exposure MWD/MTU	Keff	Average Voids	Power <u>MWt</u>	10 ^{Elow}
412.4			2416	95.70
849.6			2103	/8.0/
1498.0			2402	90.94
1944.1			2450	99.09
2356.5			2454	97.96
2731.0			2482	97.38
3839 4			2495	98.59
4347.4			1890	61.95
4733.9			2217	89.06
5173.3			2387	97.38
5471.1			2442	97.54
5824.5			2283	92.88
6043.1			2023	81.32
6559.6			2053	97.62
6846.3			1456	55.10
6983.8			1242	41.05
/056.5			1342	75 27
7843.2			1342	75.27

Table 5.2-9 XTGBWR Calculated K and Average Voids for Quad Cities-1 Cycle 4

Vol. 1, Suppleme

Table 5.2-10 XTGBWR Calculated Startup K Results for Dresden-3

Vol. 1, Supplement 2

Table 5.2-11 XTGBWR Calculated Startup K Results for Dresden-2

XN-NF-80-19(NP)(

Vol. 1, Supplem

Table 5.2-12 XTGBWR Calculated Startup K Results for Quad Cities-1

Cycle	Cycle Exposure MWD/MTU	Date	Moderator Temperature F	Reactor Period Sec.
1	0.0	4/5/72	147 ⁰	230
1	2600.0	2/8/73	160 ⁰	300
1	3400.0	5/7/73	120 ⁰	300
1	4480.0	8/7/73	120 ⁰	45
1	6270.0	1/6/74	180 ⁰	300

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TABLE 5.3-1 PEACH BOTTOM-2 EOC 2 STABILITY TESTS

Test _#	Power (% rated)	Flow (% rated)	Pressure (psia)	INLET ENTHALPY (BTU/1BM)	Measured Decay Ratio	CALCULATED DECAY RATIO
PT1	60.6	51.3	1000	508,9	0.1206	.1336
PT3	59.2	38.0	1005	528.4	0.3441	,3630

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XN-NF-80-19(NP)(A)
Vol. 1, Supplement

Figures 5.1-12 through 5.1-67 are proprietary and therefore deleted.



XN-NF-80-19(NP)(A)





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Vol. 1, Supplement ?



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XN-NF-80-19(NP)(A) Vol. 1, Supplement

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XN-NF-80-19(NP)

Vol. 1, Supplem



XN-NF-80-19(NP)(A)



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XN-NF-80-19(NP) (A)



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Vol. 1, Supplen









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Volume 1, Supplement 2

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Vol. 1, Supplement

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Vol.

1, Supplem







Vol.

Supplement

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Vol.

1, Supplem







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Vol.

1, Supplement

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XN-NF-80-19(NP)(A)













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Figure 5.2-43 Quad Cities-1 EOC2 Fuel Assembly Gamma Scan Comparisons, XTGBWR Calculated/Measured La-140 Activity

Vol. 1, Supplement 2



XTGBWR Calculated/Measured La-140 Activity

Vol. 1, Supplem



Vol. 1, Supplement 2



Fuel Assembly Gamma Scan Comparisons, XTGBWR Calculated/Measured La-140 Activity

Vol. 1, Supplemen



Vol. 1, Supplement 2



Vol. 1, Supplement





XTGBWR Calculated/Measured La-140 Activity







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Figure 5.2-62 Quad Cities-1 EOC4 Fuel Assembly Gamma Scan Comparisons, XTGBWR Calculated/Measured La-140 Activity







Fuel Assembly Gamma Scan Comparisons, XTGBWR Calculated/Measured La-140 Activity


XN-NF-80-19(NP)(A)





Fuel Assembly Gamma Scan Comparisons, XTGBWR Calculated/Measured La-140 Activity











reverse Power

Relative Power



Time (seconds)

Figure 5.3-2 COTRAN- Peach Bottom-2 Low Flow Stability Test-Power Comparison, Random Pressure Regulator Setpoint Changes



Time (seconds)

Figure 5.3-3 PT1 - COTRAN Transient Power Response to 7 psi Ramp Decrease in System Pressure



Figure 5.3-4 PT3 - COTRAN Transient Power Response to 7 Psi Ramp Decrease in System Pressure

Volume 1, Supplement 2

6.0 MEASURED POWER DISTRIBUTION UNCERTAINTY

The information presented below is intended to supplement Section 6 of the report XN-NF-80-19(P) "Exxon Nuclear Methodology for Boiling Water Reactors", Volume 1, "Neutronics Methods for Design and Analysis". Section 6 of the referenced report described the procedure by which the uncertainty associated with the measurement of a BWR power distribution would be determined. The ENC methodology for measuring the power distribution in a BWR reactor was also discussed. In this report the procedure by which the uncertainties in the measured power distribution are determined is presented again and the measured power distribution uncertainties which result from application of this methodology to the data base are presented.

The data base from which the values of the individual uncertainties are estimated consists of TIP (Traversing Incore Probe) system measurements and gamma scan measurements. The TIP system measurements are taken from 15 reactor cycles: Quad Cities Unit 1, Cycles 1, 2, 3, 4, and 5; Quad Cities Unit 2, Cycles 4 and 5; Dresden Unit 2, Cycles 4, 5, 6 and 7; and Dresden Unit 3, Cycles 4, 5, 6 and 7. The gamma scan measurements were performed at the Quad Cities Unit 1 reactor at the end of Cycles 2, 3 and 4. Eighth core gamma scan measurements made at the end of Cycles 2 and 4. A total of seven UO₂ bundles were gamma scanned on a pin by pin basis.

The following sections present the formulation and results of the uncertainty analysis in detail. A detailed description of the measured power distribution determination procedure is given in Section 6.1. A derivation of the uncertainties associated with this determination of the measured power distribution 1.° presented in Section 6.2. The quantification of the measurement uncertainties in terms of the primary sources of uncertainty is detailed in Sections 6.3.1 to 6.3.5 and the specification of the power distribution measurement uncertainty is presented in Section 6.3.6.

6.1 MEASURED POWER DISTRIBUTION DETERMINATION

Reactor measured power distributions are combinations of measured reactor data and computer calculated data. The measured reactor power distribution data include the fixed local power range monitor

Volume 1, Supplement 2

(LPRM) in-core detector data and the traversing in-core probe (TIP) detector data. The LPRM data are electric current readings proportional to the neutron flux level at four axial elevations in a number of radial locations. The radial locations are distributed in a uniform lattice throughout the core. The LPRM detectors are fission chambers using U-235 as the fissionable isotope. The LPRM detectors are intercalibrated utilizing the TIP data. The TIP system consists of a number of movable fission chamber detectors (about 1" long) which can each enter a number of the radial locations at which the fixed LPRM detectors are located. The movable TIP detectors are all capable of entering one of the radial positions to allow intercalibration of the TIP system. Figure 6.1 is a drawing of an in-core instrument tube which contains both the LPRM detectors and the TIP tube. Figure 6.2 depicts typical radial locations for both fixed and movable in-core detectors in a BWR core. Each radial location contains the equipment shown in Figure 6.1.

The computer calculated data include the relative core nodal power distribution, the in-core detector response distribution, and the local peaking factors for the fuel rods. The predicted relative nodal power and detector response distributions are calculated with the XTGBWR

Volume 1, Supplement

reactor simulator code described in Section 3.2, XN-NF-80-19.The XTGBWR code is a three dimensional modified two group diffusion theory reactor simulator program. The code uses large mesh sizes to perform full core nodal power calculations with time dependent xenon and samarium.

The local peaking factors are calculated by the ..FYRE code described in Section 3.1 of XN-NF-80-19. The XFYRE code is a single bundle depletion model that performs a microscopic depletion of each fuel rod in the fuel assembly.

The synthesis of the measured power distribution can be viewed to occur in two phases. Phase I consists of the fixed LPRM in-core detector calibration. Phase II consists of combining the individual fixed LPRM in-core detector distribution measurements with XTGBWR calculated data to produce the measured power distribution. An outline of the procedure is presented here.

XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

Volume 1, Supplement 2

6.2 DERIVATION OF THE UNCERTAINTY IN THE MEASURED POWER DISTRIBUTION

The uncertainty in the measured power distribution is derived based upon the definition of the measured power distribution

XN-NF-80-19(NP) (A)
Volume 1, Supplement 2

6-11

XN-NF-80-19 (NPXA) Volume 1, Supplement 2

XN-NF-80-19(NP) (A)
Volume 1, Supplement 2

XN-NF-80-19(NP) (A)
Volume 1, Supplement 2

Volume 1, Supplement

6.3 ESTIMATION OF UNCERTAINTY

The uncertainties,

are determined by comparison to measured data. The measured data consists of distributions of TIP detector responses plus gamma scan measurements of bundles and pins. The majority of the data consists of TIP distributions.

6.3.1 Synthesized TIP Distribution

utilizes measured and calculated data. The measured data consists of a relative distribution of fixed in-core detector response, $F_{ijk'}$. The fixed detectors are located at four axial elevations in each of a number of radial locations.

The fixed detector responses are calibrated to TIP system measurements at regular intervals and are adjusted for the reduction in sensitivity to the neutron flux as a function of burnup between calibrations to the TIP system. The uncertainty in the synthesized TIP distribution is composed of three sources: the uncertainty due to the TIP system which is acquired through the calibration process, the uncertainty associated with the fixed in-core detector response itself, and the uncertainty added by the interpolation procedure which utilizes the calculated data.



The data base used to define the TIP measurement uncertainty is summarized in Table 6.1. Data were utilized from a number of cycles in four reactors: Quad Cities Unit 1, Cycles 1 to 5; Quad Cities Unit 2,

XN-NF-80-19(NP)(A Volume 1, Supplement

Cycles 4 and 5; Dresden Unit 2, Cycles 4 to 7; Dresden Unit 3, Cycles 4 to 7.

The uncertainty in the LPRM detector response has been previously determined by General Electric in their report NEDO-20340, "Process Computer Performance Evaluation Accuracy", J. F. Carew, June 1974. A value for δ_{LPRM} of 3.4% is reported in Section 3.1.2.2. This is the value which will be used in this analysis.

XN-NF-80-19(NP)(A) Volume 1, Supplement 2

The last term in the determination of the uncertainty in the synthesized TIP distribution is the synthesis procedure uncertainty. The synthesis procedure uncertainty is that portion of the uncertainty due to interpolation between LPRM axial locations. The synthesis uncertainty can be determined by measuring the TIP distribution and then creating a synthesized TIP distribution which uses the TIP distribution

6.3.2 Calculated TIP Uncertainty

The uncertainty in the calculated TIP response distribution can be determined by comparison to measured TIP distributions. The relative standard deviation in the calculated TIP distribution can be determined as follows:

XN-NF-80-19(NP) (A)
Volume 1, Supplement 2

The measured data base used to evaluate the calculated TIP distribution uncertainty is summarized in Table 6.1. The data were taken from full core TIP measurements at three reactors: Quad Cities Unit 1, Cycles 1, 2, 3; Dresden Unit 2, Cycles 6 and 7; and Dresden Unit 3, Cycles 6 and 7.

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6.3.3 Calculated Power Distributions

The uncertainty in the calculated power distribution will be determined.

Comparisons of the calculated power distributions to measured power distributions are presented in Section 6.3.5.

XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

6.3.4 Calculated Local Power Uncertainty

The pin power distribution is determined by multiplying the nodal power, B_{ijk} , by a local power distribution factor. Local factors for each fuel type are calculated by the XFYRE code and

Volume 1, Supplement 2

input to the XTGBWR code as a function of exposure, void, and control state (controlled or uncontrolled). XTGBWR interpolates among the input data to determine a value for the particular exposure, void and control state at node ijk.

The uncertainty in local peaking factors are determined by comparing the calculated pin powers to the pin by pin gamma scans of bundles which have been irradiated in a reactor. To perform the comparisons, the pin by pin power distributions from XFYRE must be converted to Ba-140 distributions, since the gamma scans measure Ba-140 distributions rather than power distributions.

6.3.5 Bundle Gamma Scan Comparisons

The correlation coefficients in the equations of Section 6.3.3 are determined from two gamma scan measurements. The gamma scan measurements measure the relative La-140 activity in irradiated
bundles which is proportional to the power distribution of the bundles prior to shutdown. The gamma scan measurements utilized were performed at the Quad Cities Unit 1 reactor following Cycles 2 and 4. The Cycle 2 results are reported in EPRI-NP-214, July 1976, "Gamma Scan Measurements at the Quad Cities Nuclear Power Station Unit 1 Following Cycle 2". The Cycle 4 results have not yet been published. A draft copy of the results was obtained from EPRI.

To compare the XTGBWR calculated power distributions to the gamma scan results, Cycles 1, 2, 3 and 4 were modeled and depleted with the XTGBWR model. The power distributions obtained from these calculations were then converted to Ba-140 distributions for comparison to the gamma scan results. The conversion method from calculated power distributions to calculated Ba-140 distributions is detailed in the report EPRI-NP-214.

The comparisons of the calculated and measured bundle power distribution at the end of Cycles 2 and 4 are shown in Figures 6.11 and 6.12, respectively. The comparison and normalization of the relative distributions excludes all mixed oxide bundles and all bundles on the core periphery. The figures show only those bundles in an 1/8 core for simplicity in presentation. The exterior bundles were

XN-NF-80-19(NP)(A Volume 1, Supplement :

excluded since these bundles are of low power and therefore, not important from a safety standpoint. Inclusion of the edge bundles does not affect the resultant correlation coefficient or the relative standard deviations to a significant degree.

The comparisons shown in Figures 6.11 and 6.12 indicate that the agreement between the measured and calculated bundle power distributions is quite good. The largest deviations occur in the low power bundles near the core edge. The largest deviation in the core interior is -6.5%; the difference occurs in the Cycle 4 comparison and the bundle is adjacent to a mixed oxide bundle. The relative standard deviations for the comparisons are as follows: two dimensional comparison, Cycle 2 - 2.82%; Cycle 4 - 3.63%; nodal distribution, Cycle 2 - 4.85%; Cycle 4 - 7.69%. The calculations appear better radially than axially.

6.3.6 <u>Summary of the Measured Power Distribution Uncertainty</u> The measured power distribution uncertainty is

derived in Section 6.2 based upon the formulation of the measured power distribution.

XN-NF-80-19(NP) (A)
Volume 1, Supplement 2

XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

Table 6.1 Deta Base Summary

Component

Data Source

TIP System Reactor measurements: Dresden Unit 2, Cycle 4,5,6,7 Dresden Unit 3, Cycles 4,5,6,7 Quad Cities Unit 1, Cycles 1,2,3,4,5 Quad Cities Unit 2, Cycles 4,5

LPRM System Report NEDO-20340 - June 1974

TIP and Power Calculation, XTGBWR calculations of the following: TIP Synthesis Dresden Unit 2, Cycles 6,7 Dresden Unit 3, Cycles 6,7 Quad Cities Unit 1, Cycles 1,2,3

Local Power XFYRE calculations and gamma scan within a measurements for Quad Cities Unit 1, Bundle EOC 2,3,4, a total of 7 bundles in the three cycles

Calculated The above gamma scan data for bundles Bundle Power, Correlation Coefficient 6-37 through 6-40

XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

Tables 6.2 through 6.4 are proprietary and therefore deleted.



XN-NF-80-19(NP)(A)





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XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

Figures 6.3 through 6.10 are proprietary and therefore deleted.

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			.61 .65 6.0	.76 .80 5.3			Relativ Bundle Planar	e stand correla correla	lard dev ition co ition co	iation efficionefficion	2.82 ent .5 ent .6	3
		.58 .61 4.6	.74 .79 5.9		1.18 1.18 .2							
		.68 .71 4.9	.82 .87 5.2		1.02 1.07 4.4	1.06 1.10 3.2						
			-	1.01 1.03 2.4	1.30 1.28 -1.2		1.28 1.33 3.6					
		.84 .85 2.1	.95 .98 2.5			1.10 1.11 .9	1.12 1.13 .4	1.10 1.11 .6	Measur Calcul % Diff	ed ated erence	<u>c-m</u> *	100
.58		.90	1.22	1.08	1.06	1.07 1.09 2.1	1.15 1.18 2.8		1.24		1	
1.4	.81 .82		1.05		1.05 1.07 1.5	1.07 1.07 0.0		1.06 1.06 2	ĺ	1.03 1.03 0.0		1
.69	.85		1.28	1.08	1.29 1.26 -2.5	1.09 [°] 1.08 3	1.27 1.25 -1.6	1.07 1.06 3	1.05 1.04 9		1.23 1.20 -2.5	
		.95	1.02 1.01		1.06 1.04 -1.3	1.07 1.05 -2.6	1.06 1.05 7			1.03 1.01 -2.2	1.04 1.02 -2.2	1.05
.71	.84	.91	.98	1.02	1.02 1.01	1.05 1.03 -2.1	1.05 1.02 -3.0	1.03 1.01 -2.3	1.01 .99 -1.9	1.01 .99 -2.1	1.04 1.00 -3.5	1.07 1.02 -4.6

Figure 6.11 Quad Cities EOC2 Bundle Gamma Scan Results . Comparison of Measured and Calculated BA-140 Distributions

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Volume 1, Supplement 2

		1	1		1					
.58 .61 6.5	.53 .56 5.9	.54 .56 4.4	.46 .47 1.6	.36 .36 1.6]				
.73 .77 4.2	.99 .98 4	.75 .79 4.9		.57 .61 7.0	.45 .49 7.5]	
1.01 1.01 3.7	1.06 1.04 -1.3				.72 .76 6.2		.49 .52 6.0	.39 .41 5.9		
.86 .88 2.6		1.01 1.03 2.0		.87 .89 2.0		.74 .79 6.0		.52 .57 8.9	.39 .43 8.4	
1.12 1.09 2.3	.95 1.01 6.5		1.19 1.18 7	1.22 1.17 -4.0	1.13 1.11 -1.6		.87 .89 2.4	.88 .87 -1.1	.61 .67 10.3	
.89 .92 3.0	1.29 1.25 -3.2	.97 1.01 3.8	1.33 1.33 0.0	1.02 1.02 1	1.31 1.27 -2.8	.92 .96 3.4		.78 .82 4.6		
1.17 1.13 3.6	1.04 1.07 2.2		1.26 1.25 4		1.07 1.09 2.5	1.23 1.20 -2.4	1.14 1.15 .4			
.97 .97 2		.99 1.03 3.5		1.08 1.12 3.9		.99 1.02 2.6	Measu Calcu % Dif	red lated ference	<u>c-m</u> *	100%
1.21 1.17 3.6	1.22 1.22 .2		1.10 1.13 2.4	1.32 1.27 -3.6	1.27 1.25 -1.9				m	
1.00 .99 1.2	1.34 1.32 -1.5	1.07 1.11 3.6		1.07 1.12 4.6		•				
1.22 1.19 2.2	1.09 1.10 .9		1.10 1.10 5		Relat	ive Star	ndard D	eviatio	n = 3.6	3
.99 .99 .6		.99 1.01 1.6			Bundle Plana	e Correi r Correi	lation lation	Coeffic Coeffic	ient = ient =	.47 .82
1.25 1.17 6.5	1.26 1.22 -3.1									

Figure 6.12 Quad Cities EOC4 Bundle Gamma Scan Results Comparison of Measured and Calculated BA-140 Distribution

7.0 APPLICATION OF NEUTRONICS METHODOLOGY

This section describes in detail the application of ENC Neutronics Methodology to the licensing analysis of certain core related events. The following sections provide a brief statement of each event, the required special treatment of the basic ENC neutronics methodology, and the results of a sample calculation. The core related events include the control rod drop, fuel assembly misloading, pertubations which could affect reactor stability, and the control rod withdrawal transient.

XN-NF-80-19(NP)(A Volume 1, Supplement

A

7.1 CONTROL ROD DROP ACCIDENT

A parametric evaluation of the control rod drop accident was performed utilizing the COTRAN computer code and methodology described in Reference 7-1 assuming an adiabatic boundary condition and no direct moderator heating. The analysis determined the effects of dropped rod worth, Doppler coefficient, delayed neutron fraction, and fuel rod local peaking factor on the fuel rod enthalpy. The range of the above variables was selected to envelope anticipated reactor operating cycles. Thus, the results from the parametric analysis can be used directly and reported on a plant/cycle specific basis within the assumed range. If the values of the above parameters are outside the assumed range, an extension of this analysis or a specific cycle analysis may be required.

As indicated by the results presented in this document, the maximum fuel rod enthalpy for the anticipated worst set of conditions resulted in a value below the 280 cal/gm limit.

7.1.1 Description of Example Problem

The reactor core configuration for the analysis is a typical jet pump BWR cycle loaded with a mixture of exposed fuel assemblies and a fresh reload batch of ENC fuel. The 2-group input cross sections for COTRAN are obtained from an XTGBWR core simulator model which

description of the core model is given in Table 7.1-1. The initial core conditions are given in Table 7.1-2.

Volume 1, Supplement 2

7.1.2 Range of Parametric Studies

The range of values for the dropped rod worth, Doppler coefficient, delayed neutron fraction, and fuel rod local peaking was selected to envelope anticipated values in jet pump BWR's. The values of other parameters judged to influence the results such as scram reactivity insertion and dropped rod velocity were conservatively selected in the analysis. These values are shown in Table 7.1-3. A brief discussion of each of these variables is given below:

Dropped Control Rod Worth

The total rod worth (mk) is defined as [(k rod out - k rod in)/k rod in] * 1000 and the values used in the analysis are given in Table 7.1-3. In addition, calculations were performed for total rod worths of 13.3 mk and 6 mk. These dropped rod worths are obtained by varying the control fractions, α_1 and α_2 , of the two outer radial zones in the control rod drop model in accordance with the technique described in Reference 7-1. For example, Figure 7.1-1 shows the α_1 and α_2 determination for both 8 and 12 mk.

Doppler Reactivity

The Doppler reactivity feedback for the rod drop calculation is modeled using the change in cross sections as a linear function of the square root of absolute fuel temperature. Specifically the Doppler reactivity is conservatively modeled using the changes in the fast absorption cross section, Σ_{al} , and the slowing down cross section from

Volume 1, Supplement 2

the fast to the thermal energy group, $\Sigma_{1 \rightarrow 2}$. For both Σ_{a1} and $\Sigma_{1 \rightarrow 2}$, $\Delta \Sigma / \Delta$ T is derived using XFYRE⁽⁷⁻¹⁾ restart calculations over a range of temperature and exposure for both the uncontrolled and controlled states. The cross section variation with fuel temperature is then correlated with the Doppler reactivity coefficient ($\Delta k/k/^{0}F$) at a reference fuel temperature of 773⁰F. The parametric Doppler coefficients used herein are -11.5 x 10⁻⁶, -10.5 x 10⁻⁶, -9.5 x 10⁻⁶ and -8.5 x 10⁻⁶ $\Delta k/k/^{0}F$.

Delayed Neutron Fraction

The delayed neutron fraction, $\overline{\beta}$, is varied to cover the range of values from beginning to end of cycle. For this analysis, values of 0.0045, 0.0055, and 0.0065 have been used. Six groups of delayed neutron precursors are employed in the transient analyses. The decay constants and fractional group yields utilized herein are taken from recommended values in Reference 7-2 and presented in Table 7.1-4.

Fuel Rod Local Peaking

The maximum nodal (axial x radial) enthalpy occurs in the dropped rod zone in COTRAN which represents To convert this maximum nodal enthalpy to the maximum enthalpy in a fuel rod at any axial location, the four bundle local power peaking factor (P4B_L) is applied as discussed in Reference 7-1. A typical value of 1.30 has been used as a reference value for $P4B_L$. This factor is applied external to the COTRAN code as a multiplier on the maximum calculated nodal enthalpy.

Dropped Rod Velocity

The dropped rod velocity is set at 95 cm/sec (3.11 ft/sec). Control rod velocity limiter tests discussed in the appendix to Reference 7-3 have shown 3.11 ft/sec to be the maximum drop velocity that could be achieved for control rods incorporating the velocity limiter design.

Scram Reactivity

The overall negative reactivity insertion as a result of the scram is influenced by several items including the scram signal set point, the delay time from the scram signal to start of scram bank motion and the scram bank velocity. The values used in the analysis are shown in Table 7.1-3. In addition, the effect of partially inserted rods was neglected in the analysis. These factors have been combined to provide a conservative scram reactivity insertion for the COTRAN rod drop analyses. The actual scram reactivity insertion in COTRAN occurs dynamically by increasing the control fractions α_1 and α_2 to a maximum of 1.0 (fully controlled) for each axial node according to the input scram bank velocity.

Volume 1, Supplement

7.1.3 Discussion of Typical Results

Typical plots of power versus time obtained in the analysis are shown in Figure 7.1-2. The two sample cases shown are taken from the parameterization for rod worths of 12 mk and 8 mk with the Doppler coefficient set at -9.5 x 10^{-6} and $-10.5 \times 10^{-6} \Delta k/k/^{0}$ F, respectively. In both cases, \overline{B} is 0.0055. As the postulated stuck control rod falls from the core, the power begins to increase rapidly. The scram signal (set at 120% of rated power) occurs at approximately 1.16 \cdot seconds for the 12 mk rod and 1.70 seconds for the 8 mk rod.

Due to the rapidly increasing reactor power, the fuel temperature also rises quickly causing the Doppler feedback to compensate the reactivity insertion produced by the falling rod. The primary power peak as shown in Figure 7.1-2 occurs when the Doppler feedback exactly balances the dropped rod reactivity insertion. Subsequently the Doppler feedback becomes the dominating factor and the core wer is rapidly reduced.

For the higher 12 mk rod worth case, the Doppler feedback arrests the reactivity insertion before the dropped control rod is one-third of the way out. Additional reactivity is added as this dropped rod continues to fall from the core. In the 8 mk case,

Volume 1, Supplemment 2

since the control rod is almost one-half the way out when the Doppler feedback has arrested the reactivity insertion, there is little additional reactivity inserted as this rod continues to fall.

The scram bank begins to enter the core at approximately 1.46 seconds for the 12 mk case and 2.00 seconds for the 8 mk case, as shown in Figure 7.1-2. The Doppler reactivity has clearly arrested the accident and reduced the power below rated prior to the start of scram motion. Furthermore, the negative reactivity effect of the scram bank is not realized until additional time has elapsed to allow the scram bank to reach a significant level in the core. Therefore, considering the assumptions used herein, the scram reactivity is of secondary importance (compared to Doppler reactivity) during the rod drop accident.

7.1.4 Results of Parametric Studies

The results of the reference control rod drop accident analysis in terms of maximum enthalpy are summarized in Table 7.1-5. These results incorporate the four bundle local peaking $(P4B_L)$ factor of 1.30 and therefore represent the maximum enthalpy in a fuel rod at any axial location. Note that within the bounds of this parameterization, the 280 cal/gm limit is not approached.

XN-NF-80-19(NP)(A Volume 1, Supplement 2

To effectively utilize these results generically over the possible range of Doppler reactivity, delayed neutron fraction and rod worth, Figures 7.1-3 and 7.1-4 have been developed. Figure 7.1-3 provides a plot of maximum fuel rod enthalpy (including P4B_L = 1.30) versus dropped rod worth. A family of curves is provided representing the four Doppler coefficients of -11.5, -10.5, -9.5 and -8.5 x 10⁻⁶ $\Delta k/k/^{O}F$ for a base $\overline{\beta}$ of 0.0055. Figure 7.1-3 can then be used to determine the maximum fuel rod enthalpy at a specified rod worth and Doppler coefficient for a constant $\overline{\beta}$ of 0.0055.

Figure 7.1-4 provides the relative enthalpy versus rod worth for $\overline{\beta}$'s of 0.0045, 0.0055 and 0.0065. This figure is used to correct the results obtained from Figure 7.1-3 for the specific $\overline{\beta}$. The correction factor for the base $\overline{\beta}$ of 0.0055 is defined as 1.0 in Figure 7.1-4.

Although P4B_L is incorporated by assuming a typical value of 1.30, different values of P4B_L can be handled by applying the direct ratio of (actual P4B_L)/ 1.30 to the maximum enthalpy obtained using Figures 7.1-3 and 7.1-4.

7.1.5 Example Application of Results

The results of the rod drop analyses are parameterized as a function of reactor and neutronic variables. When the licensing calculations are performed for a specific reactor cycle, the cycle/core

dependent variables are calculated using both core simulator and bundle design codes. Specifically, the maximum rod worth, Doppler coefficient $(773^{\circ}F)$, core delayed neutron fraction, and four bundle local peaking must be determined and used to obtain the maximum fuel rod enthalpy from Figures 7.1-3 and 7.1-4.

As a sample calculation, the maximum fuel rod enthalpy resulting from hypothetical conditions is calculated below using the parametric results in Figures 7.1-3 and 7.1-4. The conditions prescribed are as follows:

Maximum Rod Worth (mk) = 11.0

Doppler Coefficient $(\Delta k/k/^{\circ}F) = -9.5 \times 10^{-6}$

Delayed Neutron Fraction $(\overline{\beta}) = 0.0065$

Four Bundle Local Peaking (P4B,) = 1.25

Using Figure 7.1-3, the maximum fuel rod enthalpy is determined to be

for the 11 mk rod and a Doppler coefficient of -9.5×10^{-6} . Using Figure 7.1-4, the correction factor for a \overline{B} of 0.0065 is found to be To correct for a P4B_L of 1.25 the ratio of 1.25/1.30 is calculated to be 0.962. The final maximum fuel rod enthalpy for this example is the product of

This resultant enthalpy is then compared to the 280 cal/gm upper limit to demonstrate acceptability.

When the licensing calculations are performed for a specific reactor cycle, the cycle dependent parameters important for the rod drop are calculated. The parameters are dependent on the fuel and core design, the reactor design, and the control rod withdrawal method employed. Using the cycle dependent parameters, the maximum deposited enthalpy is determined and compared to the limiting criteria to verify that the limits would not be exceeded if a rod drop accident were to occur.

7.1.6 General Applicability

The reference control rod drop analysis should be applicable if the variables for a given core and cycle are encompassed by the reference parameterization. The specific values of dropped rod worth, \overline{B} and Doppler coefficient should be within the range used for the reference analysis. However, if any of these variables is out of range in the less limiting direction (i.e., lower rod worth, higher \overline{B} , and more negative Doppler coefficient) which produces lower maximum enthalpy, the nearest value in the parameterization can be used as a conservative estimate of the maximum fuel rod enthalpy. Concerning the four bundle local peaking, as stated previously, this parameter can be varied independently by direct ratio with the assumed reference value of 1.30.

7-10

XN-NF-80-19(NP)(A Volume 1, Supplement

XN-NF-80-19(NP)(A)

Volume 1, Supplement 2

Radial Zone Number	Subzone Number	Width of Subzone <u>∆ Radius (cm</u>)	Number of Bundles Represented
1	1		
2	2		
2	3		,
2	4		,
3	5		
3	6		
3	7		
3	8		
3	9		
3	10		

Table 7.1-1 COTRAN Radial Representation

XN-NF-80-19(NP) (A

Volume 1, Supplement 2

Table 7.1-2 Initial Core Conditions

Power Level: 2527 x 10⁻⁶ Mwt Fuel Temperature: 546⁰F Moderator Temperature: 546⁰F (Saturated) Void Fraction: 0.0

XN-NF-80-19(NP) (A) Volume 1, Supplement 2

Table 7.1-3 Control Rod Drop Variables

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Parametric Values

Dropped Rod Worth: 8 mk, 12 mk Four Bundle Local Peaking: 1.30 Doppler Coefficient: $\Delta k/k/^{O}F (773^{O}F)$ -11.5×10^{-6} -10.5×10^{-6} -9.5×10^{-6} -8.5×10^{-6}

Delayed Neutron Fraction, $\overline{\beta}$: 0.0045

0	0	0	5	5
0	0	0	6	5

Fixed Values

Scram Reactivity:

Scram Signal: 120% Rated Power Scram Delay Time: 0.30 sec Scram Velocity: 77.44 cm/sec (2.54 ft/sec) Dropped Rod Velocity: 95 cm/sec (3.11 ft/sec)

Delayed Neutron Group (i)	Fractional Group Yield (B _i /B)	Decay Constant λ_i (sec)
1	0.038	0.0127
2	0.213	0.0317
3	0.188	0.115
4	0.407	0.311
5	0.128	1.40
6	0.026	3.87

Table 7.1-4 Delayed Neutron Constants

XN-NF-80-19(NP)(A)

Volume 1, Supplement 2

Control Rod Worth	Doppler Coefficient ∆k/k/0F (x 10 ⁶)	Delayed Neutron Fraction (\overline{B})				
(mk)		0.0045	0.0055	0.0065		
6	-10.5					
8 8 8 8	-8.5 -9.5 -10.5 -11.5					
12 12 12 12	-8.5 -9.5 -10.5 -11.5					
13.3	-9.5					

Table 7.1-5 Maximum Fuel Rod Enthalpy (Calories Per Gram)

Four Bundle Local Peaking $(P4B_L) = 1.30$

XN-NF-80-19(NP) (A)
Volume 1, Supplement 2

Figure 7.1-1 is proprietary and therefore has been deleted.

Volume 1, Supplement 2



Figure 7.1-2 Core Average Power versus Time During Control Rod Drop Accident

7-17

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XN-NF-80-19(NP)(A)
Volume 1, Supplement

Figure 7.1-3 is proprietary and therefore has been deleted.

XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

Figure 7.1-4 is proprietary and therefore has been deleted.

XN-NF-80-19(NP)(A Volume 1, Supplement

7.2 FUEL MISLOADING ERROR

At the present time two separate incidents are analyzed as part of the fuel misloading analysis. The first incident which is termed the fuel misorientation error assumes that a fuel assembly is misoriented by rotation through 90° or 180° from the correct orientation when loaded into the reactor core. The second incident, the fuel mislocation error, assumes a fuel assembly is placed in the wrong core location during refueling. For both the fuel misorientation error and the fuel mislocation error, the assumption is made that the error is not discovered during the core verification and the reactor is operated during the cycle with a fuel assembly misloaded.

The limiting parameter of interest for the fuel misloading error is the MCPR for the misloaded fuel assembly. The fuel misloading analysis determines the difference between the MCPR for the correctly loaded core and the MCPR for the core with a misloaded fuel assembly. The resulting Δ MCPR is then compared with the Δ MCPR determined from the analysis of anticipated plant transients in order to establish the operating MCPR limit.

7.2.1 Fuel Misorientation Error

The objective of this analysis is to determine the largest Δ MCPR for a misoriented assembly during a given fuel cycle. A description of the incident and the general procedure that will be followed in evaluating the Δ MCPR are described in detail in Reference 7-1.

Utilizing the XFYRE computer mode, fuel depletion calculations are performed for a fuel assembly rotated 180 degrees. Three depletion calculations are performed in order to model the assembly at the bottom, midchannel and top of the core taking into account the variation in the width of the water gaps at each elevation. Figures 7.2-1 through 7.2-12 present local peaking factors and S factors at bundle exposures of 0.0 and 10,000 MWD/MT at the three core elevations for a typical BWR fuel assembly. In this calculation it is assumed that the XFYRE calculations of local power distribution at 0, 40, and 70 percent voids are representative of the bottom, midchannel, and top assembly conditions. The variation with exposure of peaking factor, S factor, and K-infinity at the three elevations are presented in Tables 7.2-1 through 7.2-3.

As discussed in step 2 of the procedure given in Reference 7-1, four bundle calculations are performed with the XDT diffusion theory code. A periodic boundary conditions is applied around

XN-NF-80-19(NP) (A

Volume 1, Supplement 2

the four bundle module. One fuel assembly out of the four is rotated 90 degrees or 180 degrees and the results of the misorientation are shown in Table 7.2-4. The 180 degree misorientation results in the largest peaking factor, S-factor and K-infinity for the misoriented bundle and therefore is the limiting case.

The control rod step through calculations performed with the XTGBWR code utilize cross sections, local peaking factors and S factors as a function of voids and exposure generated by the XFYRE depletion calculations. The control rod step through calculation is first performed with no misoriented assemblies and the locations of the limiting MCPR and LHGR assemblies are noted. At each point in the step through calculation, the cross sections, local peaking factors and S factors of the limiting assembly at that point are replaced by those of the 180⁰ rotated assembly and the MCPR and LHGR of the assembly recalculated using XTGBWR. The MCPR and LHGR due to the assembly misorientation are then determined at each exposure point in the step through calculation for the cycle. It is anticipated that the MCPR for the misoriented bundle will not exceed the appropriate MCPR safety limit at any point in the cycle and this verification will be reported on a cycle specific basis.
7-23 through 7-34

XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

Figures 7.2-1 through 7.2-12 are proprietary and therefore have been deleted.

Volume 1, Supplement 2

Exposure	0.0 Voids		0.4 Vo	0.7 Voids		
(MWD/MT)	Non-Rotated 1800		Non-Rotated	Non-Rotated 18		
0.00 250.0 1000.0 2000.0 4000.0 6000.0 8000.0 10000.0 15000.0 20000.0 25000.0						

Table 7.2-1	Maximum	Local	Peaking	Factor	versus	Exposure
for	Non-Rota	ted ar	nd 1800	Rotated	Bundle	

XN-NF-80-19(NP≬A)

Volume 1, Supplement 2

Table 7.2-2 Maximum S-Factor versus Exposure for Non-Rotated and 180° Rotated Bundle

Exposure	0.0 Voids		0.4 Vo	ids	0.7 Voids	
(MWD/MT)	Non-Rotated	180	Non-Rotated	1800	Non-Rotated	1800
0.00 250.0 1000.0 2000.0 4000.0 6000.0 8000.0 1000.0 15000.0 20000.0 25000.0 35000.0						

XN-NF-80-19(NP)(A

Volume 1, Supplement 2

Table 7.2-3 K-Infinity versus Exposure for Non-Rotated and 180° Rotated Bundle

Exposure	0.0 Voids	0.4 Voids	800	0.7 Voids
(MWD/MT)	Non-Rotated 180 ⁰	Non-Rotated		Non-Rotated 180 ⁰
0.00 250.0 1000.0 2000.0 4000.0 6000.0 8000.0 10000.0 15000.0 25000.0 35000.0				

Table 7.2-4 Effect of Misorientation on Local Peaking, S Factors and K^∞

Orientation	Max. Local Peaking Factor	Maximum S Factor	K∞
0 ⁰			
90 ⁰			
180 ⁰			

XN-NF-80-19(NP) (A Volume 1, Supplement

Table 7.2-5	Maximum L	HGR	and	AMCPR	Due	to	Assembly
	Misloca	ation					

Cycle Exposure (MWD/MT)	Correct Loading Core MCPR	Mislocated Assembly MCPR	AMCPR	Mislocated Assembly MLHGR
0				
550				
1100				
1650		P .		
2200				
2750				
3300				
3850				
4400				
4950				
5500				
6050				
6600				
7150				

XN-NF-80-19(NP)(A) Volume 1, Supplement 2

7.2.2 Fuel Mislocation Analysis

The fuel mislocation error event and analysis procedure is described in detail in Reference 7-1. The primary objective of the analysis is to determine the largest Δ MCPR between the MCPR of the correctly loaded core and the MCPR of the mislocated assembly. A second objective is to determine the maximum LHGR for a mislocated assembly. The calculational procedure referenced above has been slightly revised and a method has been developed to determine the worst locations for a fuel misloading error. This method is / discussed below followed by a description of the revised calculation procedure. In addition, the results obtained from applying the procedure to a sample problem are presented.

Worst Location Determination Procedure

In order to determine the worst location for a fuel misloading error, each location in the core must be analyzed to determine the effects of a fuel misloading throughout the cycle. This is accomplished by performing a localized power calculation for each core location to estimate the assembly power of a misloaded assembly at the location. The misloaded assembly is assumed to have been fresh at the beginning of the cycle. The calculation is performed at each point in a control rod step through of the cycle. This analysis determines the core location which would have the highest misloaded assembly power at each point in the cycle.

The technique used to estimate the power of a mislocated assembly uses assembly average flux weighted cross sections and fast fluxes to perform a power calculation for a simplified model of the mislocated assembly and the four surrounding fuel assemblies. The cross sections and fluxes are

determined from a three dimensional calculation of the correctly loaded core using the XTGBWR reactor simulator code.

7-42

XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

Volume 1, Supplement 2

In this manner the relative assembly power of a mislocated assembly at every core location (with the exception of locations on the core periphery) can be estimated at each point in a control rod step through of the cycle using only the assembly average data calculated for the correctly loaded core. The locations which have the largest estimated misloaded power at each point in the cycle are then selected for detailed analysis.

Analysis Procedures

The analysis of the fuel assembly mislocation error consists of the following steps which correspond to the procedure given in Reference 7-1 with slight revisions:

> 1. The fuel loading pattern for the cycle is developed and a control rod step through performed for the cycle

XN-NF-80-19(NP)(A) Volume 1, Supplement 2

using the reactor simulator code XTGBWR. The MCPR and maximum LHGR for the core are determined at each point in the step through calculation. Also at each point in the step through assembly average fast group flux, relative power, and flux weighted cross sections are calculated in XTGBWR for each assembly in the core.

 The assembly average data is used in the procedure described above to determine the core locations which would have the largest misloaded assembly power at each point in the step through calculation. Typically two or three core locations may be selected at each point in the cycle.
 One of the locations determined in Step 2 is selected and a high reactivity assembly is misloaded into that location at the beginning of cycle point. The misloaded core is burned (with the XTGBWR reactor simulator code) using the control rod patterns selected in Step 1 to determine the MCPR and maximum LHGR of the misloaded assembly.

XN-NF-80-19(NP)(/

Volume 1, Supplement

4. The MCPR determined for the mislocated assembly is subtracted from the MCPR of the correctly loaded core at each point in the cycle and the largest ΔMCPR determined for that core location. The maximum LHGR for the mislocated assembly is also determined.

5. Steps 3 and 4 are repeated for each location determined in Step 2 and the largest △MCPR and maximum LHGR of a mislocated assembly are determined for the cycle.
Step 6 of the procedure stated^{*} in Reference 7.1 which

describes a more detailed calculation using the ENC core monitoring method to determine a more accurate fuel mislocation \triangle MCPR has not been revised and was not used in the sample calculation discussed below.

The procedure outlined above was performed for a representative BWR core loading pattern and cycle control rod step through. The results of the worst fuel assembly mislocation found at each point in the cycle are shown in Table 7.2-5. The largest \triangle MCPR due to a mislocation was determined to be The maximum LHGR found for a mislocated assembly was Kw/ft. Figure 7.2-14 shows assembly relative powers and assembly MCPR values for the worst misloaded assembly location and surrounding locations for both the correct loading and the misloaded assembly. 7-46

XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

	Correctly Loaded Assembly K=1	
Correctly Loaded Assembly K=4	Mislocated Assembly K=0	Correctly Loaded Assembly K=2
	Correctly Loaded Assembly K=3	

Figure 7.2-13

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Assembly Arrangement Used in Estimating Mislocated Assembly Power

XN-NF-80-19(NP) (A)
Volume 1, Supplement

Figure 7.2-14 is proprietary and therefore has been deleted.

XN-NF-80-19(NP)(A) Volume 1, Supplement 2

7.3 REACTOR CORE STABILITY

Stability can be defined for an operating system as follows: a system is stable if, following an input perturbation, the transient response returns to a steady, non-cyclic state. For a time domain analysis of reactor core stability, the degree of stability is defined by the decay ratio (the magnitude ratio, X_2/X_0 , of successive transient maximum or minimum). The decay ratio is determined from the core average power response to a rapid perturbation in system pressure or control rod ' position. When the decay ratio is less than 1.0, the reactor core is stable.

7.3.1 Analysis Model and Conditions

ENC's reactor core stability analysis methodology is described in detail in Reference 7.1 and utilizes the COTRAN computer code. The one-group cross sections used in the iterative flux solution are determined from input two-group values and modified at each time step for thermal hydraulic feedback. The two-group input cross sections for COTRAN are obtained from the XTGBWR core simulator model

The reactor core conditions for the stability analysis are based on a representative Haling solution for a core loading of exposed and fresh ENC reload assemblies.

XN-NF-80-19(NP)(A Volume 1, Supplement

XTGBWR calculations were performed along the rated power-flow line and along the power-flow line corresponding to natural circulation to obtain the appropriate COTRAN input. The power-flow lines used in the example calculation are shown in Figure 7.3-1.

The COTRAN model utilized for reactor core stability analysis simulates the core average fuel design. The model is divided into

The hydraulic flow channel is modeled with the spatial detail of the neutronic calculation and extends from the inlet orifice to the upper tie plate. The modeling methodology is consistent with that utilized for the reactor core stability verification with integral plant data from Peach Bottom $2^{(7-1)}$.

7.3.2 Analysis Methodology and Stability Margins

Stability transients are initiated by perturbing the steady-state operating conditions and applying the steady-state core average pressure drop as a boundary condition. The stability analysis results presented in this section are for a ramp decrease in pressure of \sim 4 psi in 0.10 seconds. At the end of the ramp pressure change, the system pressure was fixed for the remainder of the transient.

The resultant transient power response is analyzed in accordance with the procedures presented in Reference 7-1 to determine the operating state decay ratio.

This transient analysis procedure, when applied over the range of power/flow conditions in this example calculation, will yield a graphic representation of the core stability margins, Figure 7.3-2, at the limiting end-of-cycle operating state. As expected, the least stable operating point on these power/flow curves occurs at the intersection of the rated power/flow line and the natural circulation flow line. At this point, the decay ratio was calculated to be Increasing power along the rated power/flow line is compensated by a larger increase in inlet flow that stabilizes the system. A corresponding decrease in power along the natural circulation line results in a decrease in void content and stabilizes the system. A plot of power as a function of time at the limiting stability conditions following a pressure ramp is shown on Figure 7.3-3.



XN-NF-80-19(NP)(A)
Volume 1, Supplement 2

Figures 7.3-2 and 7.3-3 are proprietary and therefore have been deleted.

XN-NF-80-19(IIP)(/ Volume 1, Supplement

7.4 CONTROL ROD WITHDRAWAL

A control rod withdrawal analysis has been performed using the methods presented in Section 4.5 of XN-NF-80-19(P) Volume 1. A control rod pattern for the analysis was determined by starting with the projected control rod pattern for the reactor cycle and adjusting the control rod positions to place a high worth rod fully inserted with the nearby fuel at or near thermal limits. The starting control rod pattern for the control rod pattern for the starting control rod pattern for the control rod pattern for the starting control rod pattern for the control rod pattern for the starting control rod pattern for the control rod pattern for the starting control rod pattern for the control rod withdrawal analysis is shown on Figure 7.4-1. The control rod being withdrawn is the rod at 0 notches at core location 38-19.

As the control rod is withdrawn the reactor power increases resulting in a decrease in the minimum critical power ratio (MCPR) and an increase in the rod block monitor (RBM) response. A plot of the change in MCPR as a function of the control rod position is shown in Figure 7.4-2. The RBM response for the limiting channel A+C with no LPRM detector failures and with two LPRM detector strings failed is shown on Figure 7.4-3. The RBM response for channel B+D with and without LPRM failures is shown on Figure 7.4-4. The Δ MCPR, Δ MLHGR, control rod position and reactor power as a function of the rod block reading is shown on Table 7.1-1. The Δ MCPR values for the control rod withdrawal are compared to the Δ MCPR values for the other transients to determine the operating MCPR limit and rod block set point that will protect the MCPR safety limit of the reactor.

7-55

XN-NF-80-19(NP)(A) Volume 1, Supplement 2

Table 7.4-1 Control Rod Withdrawal Results

RBM Reading %	Rod Position Feet Withdrawn	∆MCPR (XN-3)	∆MLHGR, kw/ft	Reactor Power, MWt
105	3.5			
107	4.0			
109	4.5			

7-56

XN-NF-80-19(NP)(#

Volume 1, Supplement



Figure 7.4-1 Starting Control Rod Pattern for Control Rod Withdrawal Analysis

Note: * Control Rod being Withdrawn, Rod Positions in Notches, Full in = 0, Full out = blank or 48 7-57 through 7-59

XN-NF-80-19(NP)(A) Volume 1, Supplement 2

Figures 7.2-2 through 7.2-4 are proprietary and have been deleted.

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7.5 REFERENCES

- 7-1 Exxon Nuclear Methodology for Boiling Water Reactors, Volume 1, Neutronic Methods for Design and Analysis, XN-NF-80-19(P), May, 1980.
- 7-2 R. J. Tuttle, Nuclear Science and Engineering: 56, 37-71 (1975), p. 70, Table VII.
- 7-3 C. J. Paone, R. C. Stirn and J. A. Wooley, "Rod Drop Accident Analysis for Large Boiling Water Reactors", NEDO-10527, March 1972.