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# A ONE-DIMENSIONAL NEUTRON KINETICS MODEL FOR THE THOR CODE

## FINAL REPORT

Prepared by
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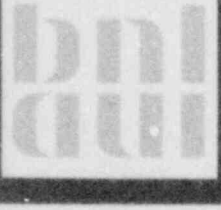

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## ABSTRACT

The neutron kinetics model developed for the THOR code is described. The kinetics model calculates the energy deposition in the reactor core due to fission and radioactive decay. This calculation is performed using one of three user-selected options: (1) Table lookup of total power as a function of time, (2) point kinetics, or (3) 1-group, 1-dimensional (axial) space-dependent kinetics. Feedback effects due to changes in fuel temperature, coolant temperature, and coolant density are included in both the point and space-dependent options. The space-dependent option also includes the capability to calculate the initial critical axial power distribution, and to represent explicitly the movement of control rods in and out of the core.

NOMENCLATURE

| | |
|---|---|
| $C_i(t)$ | Density of delayed neutron group i (MW) |
| D | Diffusion coefficient (cm) |
| $E_n$ | Effective energy fraction of decay heat group n |
| $f^k(t)$ | Control density in mesh interval k |
| $H_n(t)$ | Decay power of decay heat group n (MW) |
| I | Number of delayed neutron groups |
| J | Net current ($n/cm^2$-sec) |
| $K_{eff}$ | Effective multiplication factor |
| P(t) | Instantaneous fission power (MW) |
| $P_T$ | Total power (MW) |
| R(t) | Total reactivity |
| $R_{T_f}(t)$ | Fuel temperature reactivity |
| $R_{T_m}(t)$ | Moderator temperature reactivity |
| $R_{\rho_m}$ | Moderator density reactivity |
| $R_{ext}(t)$ | External perturbation reactivity |
| t | Time (sec) |
| TOS | THOR operating system |
| $\overline{T}_f$ | Average fuel temperature (°K) |
| $\overline{T}_f^*$ | Reference average fuel temperature (°K) |
| $\overline{T}_m$ | Average moderator temperature (°K) |
| $\overline{T}_m^*$ | Reference average moderator temperature (°K) |
| V | Neutron speed (cm/sec) |
| Z | Axial position (cm) |

GREEK

| | |
|---|---|
| $\alpha_{T_f}$ | Fuel temperature reactivity-feedback coefficient ($°K^{-1/2}$) |

| | |
|---|---|
| $\alpha_{T_m}$ | Moderator temperature reactivity-feedback coefficient (°F$^{-1}$) |
| $\alpha_{\rho_m}$ | Moderator density reactivity-feedback coefficient [(g/cm$^3$)$^{-1}$] |
| $\bar{\alpha}_m$ | Average moderator void fraction |
| $\bar{\alpha}_m^*$ | Reference average moderator void fraction |
| $\beta$ | Total delayed neutron fraction |
| $\beta_i$ | Delayed neutron fraction of delayed neutron group i |
| $\lambda_i$ | Decay constant of delayed neutron group i (sec$^{-1}$) |
| $\lambda_n^H$ | Decay constant of decay heat group n (sec$^{-1}$) |
| $\Lambda$ | Prompt neutron generation time (sec) |
| $\phi(z,t)$ | Neutron flux (n/cm$^2$-sec) |
| $\psi(z,o)$ | Normalized steady-state axial power shape |
| $\Sigma^a$ | Macroscopic absorption cross-section (cm$^{-1}$) |
| $\Sigma^f$ | Macroscopic fission cross-section (cm$^{-1}$) |
| $\Sigma_c^k$ | "Controlled" cross-section (cm$^{-1}$) |
| $\Sigma_{uc}^k$ | "Uncontrolled" cross-section (cm$^{-1}$) |
| $\nu$ | Average number of neutrons per fission |

# 1 INTRODUCTION

## 1.1 Background

The THOR (THermohydraulic Of Reactors) computer code[1] is presently under development in the Thermal Hydraulic Development Division at Brookhaven National Laboratory under the sponsorship of the Reactor Safety Research Division of the U.S. Nuclear Regulatory Commission. The code is designed as a fast-running "best estimate" computer program to predict normal and hypothetical accident-induced, thermohydraulic transients in light-water cooled reactor systems. The purpose of this report is to describe the neutron kinetics model which has been developed for incorporation into the THOR code.

## 1.2 Basic Description of the Models

The neutron kinetics model calculates the energy deposition in the reactor core due to fission and radioactive decay during a reactor transient. This calculation is performed using one of the following options: (1) Table lookup of reactor power as a function of time, (2) point kinetics, or (3) 1-group, 1-dimensional (axial) space-dependent kinetics. The table lookup and point kinetics options are both formulated under the assumption that the power shape does not change throughout the transient. In the table lookup option, the total power is simply obtained from a user-specified table of reactor power as a function of time. Linear interpolation is used to obtain values lying between the table entries. The point kinetics option calculates the total power using the standard point-reactor kinetics equations, and requires that the user specify a table of reactivity values as a function of time. While the table lookup method does not account for thermohydraulic

feedback, the point kinetics option does calculate reactivity contributions due to (core-averaged) changes in the fuel temperature, moderator temperature, and moderator density. These contributions are calculated under the assumption that the reactivity varies linearly with changes in these thermohydraulic properties. The linear coefficients (e.g. the Doppler coefficient) are required as input data. The point kinetics model is described further in Section 2.

The space-dependent kinetics option solves the 1-group neutron diffusion equation in 1 space-dimension with up to six groups of delayed neutrons. A set of 1-group cross sections must be supplied by the user for each nuclear region. These cross sections are generally obtained from a more detailed multigroup calculation and must be appropriately averaged in the x-y plane to yield the required cross sections for the 1-dimensional axial equations. The initial condition is calculated by the solving the steady-state eigenvalue problem for the critical flux shape and effective multiplication factor. Space-dependent feedback effects are included in both the steady-state and transient calculations, and are modeled under the assumption that the cross sections vary linearly with both the square root of the fuel temperature and the moderator temperature, and quadrtically with the coolant void fraction. These feedback coefficients must be supplied by the user. Control rod movement due to rod withdrawal and scram is represented explicitly. Details of the space-dependent kinetics model are given in Section 3.

# 2. THE POINT KINETICS MODEL

## 2.1 The Point Kinetics Equations

The standard point kinetics equations are solved in conjunction with an additional equation describing the deposition of decay heat in the core. These equations take the form

$$\frac{d}{dt} P(t) = \left[\frac{R(t) - \beta}{\Lambda}\right] P(t) + \sum_{i=1}^{I} \lambda_i C_i(t) \tag{2.1}$$

$$\frac{d}{dt} C_i(t) = \frac{\beta_i}{\Lambda} C_i(t) - \lambda_i C_i(t), i=1,\cdots,I \tag{2.2}$$

$$\frac{d}{dt} H_n(t) = E_n P(t) - \lambda_n^H H_n(t), \; n=1,\cdots,N \tag{2.3}$$

where

$P(t)$ = amplitude function (often interpreted as instantaneous fission power)

$C_i(t)$ = density of delayed neutron group i

$H_n(t)$ = decay power of decay heat group n

$R(t)$ = total reactivity

$\beta$ = total effective delayed neutron fraction

$\Lambda$ = prompt neutron generation time

$\lambda_i$ = decay constant of delayed neutron group i

$\beta_i$ = effective delayed neutron fraction of delayed neutron group i

$E_n$ = effective energy fraction of decay heat group n,

$\lambda_n^H$ = decay constant of decay heat group n.

The total power deposited in the core due to fission and radioactive decay is given by

$$P_T(t) = \left[1 - \sum_{n=1}^{N} E_n\right] P(t) + \sum_{n=1}^{N} \lambda_n^H H_n(t) . \qquad (2.4)$$

The number of delayed neutron groups (I) and number of decay heat groups (N) are presently fixed at 6 and 11, respectively. With the exception of R(t) and $\Lambda$, which must be provided by the user, the constants appearing in Eqs. (2.1) to (2.3) are built into the code, and are identical to those used in the TRAC[2] and RELAP[3] codes.

## 2.2 Solution of the Point Kinetics Equations

Equations (2.1) to (2.3) are solved using simple implicit techniques. Equation (2.1) is discretized in the following manner:

$$\frac{1}{\Delta t_j}\left[P(t_{j+1}) - P(t_j)\right] = \frac{\bar{R}^j - \beta}{\Lambda}\left[\theta^j P(t_{j+1}) + (1-\theta^j)\,P(t_j)\right] + \sum_{i=1}^{I} \lambda_i \left[\theta_d^j\, C_i(t_{j+1}) + (1-\theta_d^j) C_i(t_j)\right], \qquad (2.5)$$

where $\Delta t_j \equiv t_{j+1} - t_j$, $\theta^j$ and $\theta_d^j$ are parameters discussed below, and $\bar{R}^j$ denotes the average reactivity over the time step. Equation (2.2) is formally integrated over a time step, i.e.

$$C_i(t_{j+1}) = C_i(t_j)e^{-\lambda_i \Delta t_j} + \frac{\beta_i}{\Lambda}\int_{t_j}^{t_{j+1}} dt\, P(t) e^{-\lambda_i (t_{j+1}-t)} . \qquad (2.6)$$

The integral in this equation is approximated by assuming that P(t) varies linearly over the time interval $\Delta t_j$; the following equation is thus obtained:

$$C_i(t_{j+1}) = C_i(t_j)e^{-\lambda_i \Delta t_j} + \frac{\beta_i}{\Lambda}\left\{\left[\frac{1-e^{-\lambda_i \Delta t_j}}{\lambda_i \Delta t_j} - e^{-\lambda_i \Delta t_j}\right] P(t_j) - \left[\frac{1-e^{-\lambda_i \Delta t_j}}{\lambda_i \Delta t_j} - 1\right] P(t_{j+1})\right\} . \qquad (2.7)$$

The same procedure is used to approximate Eq. (2.3). Using Eq. (2.7), the $C_i(t_{j+1})$ can be eliminated from Eq. (2.5), and the resulting equation readily solved for $P(t_{j+1})$. Once $P(t_{j+1})$ is known, the $C_i(t_{j+1})$ and $H_n(t_{j+1})$ are updated separately using Eq. (2.7) and the analagous equation for the $H_n(t_{j+1})$.

The parameters $\theta^j$ and $\theta_d^j$ are chosen in order to improve the accuracy of the solution and to insure numerical stability. Currently $\theta_d^j$ is fixed at ½, while $\theta^j$ is chosen as follows

$$\theta^j = \begin{cases} 1/2 & , \ \bar{R}^j \geq 0.5\ \beta \\ 1 & , \ \bar{R}^j < 0.5\ \beta\ . \end{cases} \tag{2.8}$$

This formula is used since $\theta^j = ½$ is more accurate for less stiff problems (i.e. transients approaching prompt critical), while $\theta^j = 1$ is necessary for an accurate representation of the stiff modes present in a delayed supercritical or subcritical transient. The neutronics time step $\Delta t_j$ controlled such that the power does not change by more than 10% over a time step.

The power distribution returned to the heat conduction segment of THOR is calculated from

$$P\ (Z,t) \ = \ P_T(t)\ \psi\ (Z,0) \quad , \tag{2.9}$$

where $\psi(z,0)$ is the normalized steady-state axial power shape. [Equation (2.9) also used in the table lookup option, with $P_T(t)$ obtained directly from the user-supplied power versus time curve.]

## 2.3. The Point Kinetics Feedback Model

A table of reactivity as a function of time must be supplied by the user. This reactivity can be either the total reactivity R(t), or the

reactivity contribution $R_{ext}(t)$ due solely to external perturbations such as control rod movement. In the latter case, the total reactivity is calculated as the sum of components due to thermohydraulic feedback and external perturbations:

$$R(t) = R_{T_f}(t) + R_{T_m}(t) + R_{\rho_m}(t) + R_{ext}(t) \quad , \tag{2.10}$$

where $R_{T_f}(t)$, $R_{T_m}(t)$, and $R_{\rho_m}(t)$ are the reactivity components due to changes in the fuel temperature, moderator temperature, and moderator density, respectively. These components are calculated in terms of the core-averaged thermohydraulic properties:

$$R_{T_f}(t) = \alpha_{T_f}\left(\sqrt{\bar{T}_f} - \sqrt{\bar{T}_f^{\,\star}}\right) \tag{2.11a}$$

$$R_{T_m}(t) = \alpha_{T_m}\left(\bar{T}_m - \bar{T}_m^{\,\star}\right) \tag{2.11b}$$

$$R_{\rho_m}(t) = \alpha_{\rho_m}\left(\bar{\rho}_m - \bar{\rho}_m^{\,\star}\right) \tag{2.11c}$$

Here the starred quantities denote core-averaged reference (i.e. steady-state) properties. The coefficients $\alpha_{T_f}$, $\alpha_{T_m}$, $\alpha_{\rho_m}$ are generally given in a Safety Analysis Report (SAR), and must be supplied by user with reactivity given in terms of $\Delta k/k$.

# 3. THE SPACE-DEPENDENT KINETICS MODEL

## 3.1 The Time-Dependent Neutron Diffusion Equation

The space-dependent neutron kinetics model solves the time-dependent 1-group, 1-dimensional neutron diffusion equation:

$$\frac{1}{V}\frac{\partial}{\partial t}\phi(Z,t) = \frac{\partial}{\partial Z} D(Z,t) \frac{\partial}{\partial Z}\phi(Z,t) + \Sigma(Z,t)\phi(Z,t) + \sum_{i=1}^{I} \lambda_i C_i(Z,t) \quad , \tag{3.1}$$

where

$$\Sigma(Z,t) \equiv (1-\beta)\,\nu\Sigma^f(Z,t) - \Sigma^a(Z,t) \tag{3.2}$$

and

$\phi$ = neutron flux

$C_i$ = density of the $i^{th}$ delayed neutron precursor

$D$ = diffusion coefficient

$\nu\Sigma^f$ = nu, the number of neutrons per fission, times the macroscopic fission cross section

$\Sigma^a$ = macroscopic absorption cross section

$\beta$ = total delayed neutron fraction

$\lambda_i$ = decay constant of delayed neutron group i

$I$ = number of delayed neutron groups

$z$ = axial position.

The delayed neutron precursors satisfy the equation

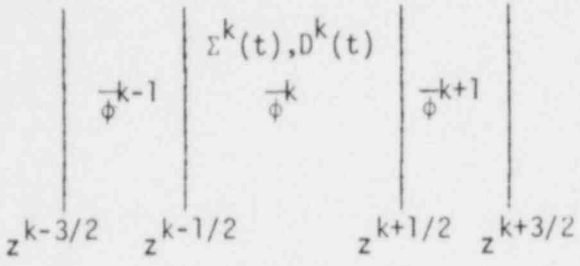
$$\frac{\partial}{\partial t} C_i(Z,t) = \beta_i \nu\Sigma^f(Z,t)\phi(Z,t) - \lambda_i C_i(Z,t), \tag{3.3}$$

where $i=1,\cdots,I$ ,

$\beta_i$ = fraction of fissions which produce an $i^{th}$ group delayed neutron precursor.

## 3.2 The Spatial Finite Difference Approximation

Equations (3.1) and (3.3) are spatially discretized using a "mesh-centered" finite difference formulation in which the unknowns are the average fluxes and precursor densities within each mesh interval (rather than the pointwise values at the mesh interval interfaces). The axial dimension is divided into K mesh cells using the following notation:




$$\Delta z^k \equiv z^{k+1/2} - z^{k-1/2}$$

The cross sections (i.e. $D^k(t)$, $\Sigma^k(t)$) are assumed to be spatially constant within a mesh cell. Integrating Eqs. (3.1) and (3.3) over the interval $[z^{k-1/2}, z^{k+1/2}]$, and then dividing by $\Delta z^k$ yields

$$\frac{1}{V}\frac{d}{dt}\bar{\phi}^k(t) = -L^k(t) + \Sigma^k(t)\bar{\phi}^k(t) + \sum_{i=1}^{I} \lambda_i \bar{C}_i^k(t) \ , \tag{3.4}$$

$$\frac{d}{dt}\bar{C}_i^k(t) = \beta_i \nu\Sigma^{f,k}(t)\bar{\phi}^k(t) - \lambda_i \bar{C}_i^k(t), \qquad i=1, \cdots, I, \tag{3.5}$$

where

$$\bar{\phi}^k(t) \equiv \frac{1}{\Delta Z^k} \int_{Z^{k-\frac{1}{2}}}^{Z^{k+\frac{1}{2}}} dZ\ \phi(Z,t) \ , \tag{3.6a}$$

$$\bar{C}_i^k(t) \equiv \frac{1}{\Delta Z^k} \int_{Z^{k-\frac{1}{2}}}^{Z^{k+\frac{1}{2}}} dZ\ C_i(Z,t) \ . \tag{3.6b}$$

The first term on the right hand side of Eq. (3.4) represents the net leakage across the surfaces of the mesh cell; this leakage can be written

in terms of the net currents $J(z^{k\pm1/2},t)$ on the mesh cell surfaces, i.e.

$$L^k(t) \equiv -\frac{1}{\Delta Z^k}\left[D(Z,t)\frac{\partial}{\partial Z}\phi(Z,t)\right]_{Z=Z^{k-\frac{1}{2}}}^{Z=Z^{k+\frac{1}{2}}}$$

$$= \frac{1}{\Delta Z^k}\left[J(Z^{k+\frac{1}{2}},t) - J(Z^{k-\frac{1}{2}},t)\right] \quad . \tag{3.7}$$

This leakage term is approximated using first order differences, i.e.

$$L^k(t) \cong -\frac{2}{(\Delta Z^k)^2}\, D^k(t)\left\{\left[\phi^{k+\frac{1}{2}}(t)-\bar{\phi}^k(t)\right] - \left[\bar{\phi}^k(t)-\phi^{k-\frac{1}{2}}(t)\right]\right\}, \tag{3.8}$$

where $\phi^{k\pm1/2}(t)$ are pointwise fluxes evaluated at the surfaces $z = z^{k\pm1/2}$. These "halfpoint" values are calculated by requiring that the next currents be continuous at $z = z^{k\pm1/2}$, i.e.

$$\frac{2D^{k+1}(t)}{\Delta Z^{k+1}}\left[\bar{\phi}^{k+1}(t) - \phi^{k+\frac{1}{2}}(t)\right] = \frac{2D^k(t)}{\Delta Z^k}\left[\phi^{k+\frac{1}{2}}(t)-\bar{\phi}^k(t)\right], \tag{3.9a}$$

$$\frac{2D^{k-1}(t)}{\Delta Z^{k-1}}\left[\phi^{k-\frac{1}{2}}(t) - \bar{\phi}^{k-1}(t)\right] = \frac{2D^k(t)}{\Delta Z^k}\left[\bar{\phi}^k(t) - \phi^{k-\frac{1}{2}}(t)\right]. \tag{3.9b}$$

Solving Eqs. (3.9) for the halfpoint values yields

$$\phi^{k+\frac{1}{2}}(t) = \left[\hat{D}^k(t)\,\bar{\phi}^k(t) + \hat{D}^{k+1}(t)\,\bar{\phi}^{k+1}(t)\right] \Big/ \left[\hat{D}^k(t) + \hat{D}^{k+1}(t)\right], \tag{3.10a}$$

$$\phi^{k-\frac{1}{2}}(t) = \left[\hat{D}^{k-1}(t)\,\bar{\phi}^{k-1}(t) + \hat{D}^k(t)\,\bar{\phi}^k(t)\right] \Big/ \left[\hat{D}^{k-1}(t) + \hat{D}^k(t)\right], \tag{3.10b}$$

where

$$\hat{D}^k(t) \equiv \frac{D^k(t)}{\Delta Z^k}$$

Substituting Eqs. (3.10) into Eq. (3.8) and then rearranging yields

$$L^k(t) \cong \gamma^{k-}(t)\,\bar{\phi}^{k-1}(t) + \gamma^{k}(t)\bar{\phi}^k(t) + \gamma^{k+}(t)\,\bar{\phi}^{k+1}(t) \tag{3.11}$$

where the coupling coefficients introduced here are defined by

$$\gamma^{k-}(t) \equiv -\frac{2}{\Delta Z^k}\,\frac{\mathcal{D}^k(t)\,\mathcal{D}^{k-1}(t)}{\mathcal{D}^k(t)+\mathcal{D}^{k-1}(t)}\quad, \tag{3.12a}$$

$$\gamma^{k+}(t) \equiv -\frac{2}{\Delta Z^k}\,\frac{\mathcal{D}^k(t)\,\mathcal{D}^{k+1}(t)}{\mathcal{D}^k(t)+\mathcal{D}^{k+1}(t)}\quad, \tag{3.12b}$$

$$\gamma^{k}(t) \equiv -\left[\gamma^{k-}(t)+\gamma^{k+}(t)\right]\quad. \tag{3.12c}$$

Zero flux boundary conditions are specified on the outer surfaces of the reactor. Letting K denote the mesh cell adjacent to an outer surface such that $z^{k+1/2}$ denotes the outer surface and $\phi(z^{K+1/2},t) \equiv 0$, the coupling coefficients for this cell are given by

$$\gamma^{k-}(t) \equiv -\frac{2}{\Delta Z^k}\,\frac{\mathcal{D}^k(t)\,\mathcal{D}^{k-1}(t)}{\mathcal{D}^k(t)+\mathcal{D}^{k-1}(t)}\quad, \tag{3.13a}$$

$$\gamma^{k+}(t) \equiv 0 \tag{3.13b}$$

$$\gamma^{k}(t) \equiv -\left[\gamma^{k-}(t)+2\,\mathcal{D}^k(t)\right]\quad. \tag{3.13c}$$

Substituting Eq. (3.11) into Eq. (3.4), and then rewriting the result in matrix form yields

$$\frac{1}{V}\,\frac{d}{dt}\,\underline{\phi}(t) = \left\{\left[-L(t)\right]+\left[\Sigma(t)\right]\right\}\underline{\phi}(t)+\sum_{i}^{I}\gamma_i\underline{C}_i(t), \tag{3.14}$$

where

$$\underline{\phi}(t) \equiv \mathrm{Col}\left[\bar{\phi}^1(t),\cdots,\bar{\phi}^k(t),\cdots,\bar{\phi}^K(t)\right]\quad, \tag{3.15a}$$

$$\underline{C}_i(t) \equiv \mathrm{Col}\left[\bar{C}_i^1(t),\cdots,\bar{C}_i^1(t),\cdots,\bar{C}_i^K(t)\right]\quad, \tag{3.15b}$$

$$\left[\Sigma(t)\right] \equiv \text{diag}\left[\Sigma^1(t),\cdots, \Sigma^k(t),\cdots, \Sigma^K(t)\right] , \qquad (3.15c)$$

and the entries of the tridiagonal matrix [L(t)] are the coupling coefficients defined in Eqs. (3.12) and (3.13). Equations (3.14) and (3.5) can be combined to yield

$$\frac{d}{dt}\ \underline{\Psi}(t) = \left[\ A(t)\ \right] \underline{\Psi}(t) \quad , \qquad (3.16)$$

where

$$\underline{\Psi}(t) \equiv \text{Col}\ \left[\ \underline{\Phi}(t),\ \underline{C}_1(t),\cdots,\ \underline{C}_i(t),\cdots,\ \underline{C}_I(t)\ \right] , \qquad (3.17)$$

and the matrix [A(t)] is defined by comparison with Eqs. (3.5), (3.14), and (3.16). The solution of Eq. (3.16) is discussed in the following section.

## 3.3 Solution of the Time-Dependent Equations

Equation (3.16) is time-differenced as follows:

$$\frac{1}{\Delta t_j}\left[\ \underline{\Psi}(t_{j+1}) - \underline{\Psi}(t_j)\ \right] = \theta\left[\ A(t_{j+1})\right]\ \underline{\Psi}(t_{j+1}) \qquad (3.18)$$

$$+ (1-\theta)\left[A(t_j)\right]\ \underline{\Psi}(t_j),$$

where $\Delta t_j \equiv t_{j+1} - t_j$, and $\theta$ is an input parameter. Note that setting $\theta = 1$ results in a fully implicit scheme, while the Crank-Nicholson method is obtained with $\theta = 1/2$. The use of $\theta < 1/2$ is not allowed since numerical stability can no longer be guaranteed in this case.

Equation (3.18) is solved by first eliminating the precursor densities at the current time step from the flux equation, and then inverting the remaining tridiagonal matrix (with dimension K) using a standard LU decomposition with forward elimination-backward substitution to solve for the current flux values. Once the fluxes are known at the current time step, it is a simple matter to update the precursor densities.

The instantaneous fission power generated in mesh cell k at time $t_j$ is given by

$$P^K(t_j) = N \Delta Z^K \Sigma^{t,K} \bar{\phi}^K(t_j) \quad , \tag{3.19}$$

where N is a constant relating fissions/sec to power in MW. In the event of a reactor scram, the decay power $P_D^k(t_j)$ is also calculated:

$$P_D^K(t_j) = P_T^K(t_s)\, F_D(t_j - t_s) \quad , \tag{3.20}$$

where $P_T^k(t_j)$ is the total power at time $t_s$ at which the scram occurs, and $F_D(t_j - t_s)$ is the decay heat fraction calculated from the ANS standard decay heat curve.[4] The total power returned to heat conduction segment of THOR is calculated as the sum of the contributions due to fission and decay heat.

The neutronics time step is controlled such that the pointwise power does not change over any time interval by more than a user-specified input parameter. The time step itself is always smaller than or equal to the TOS* time step and is independently adjusted to accommodate important neutronic events such as prompt criticality.

## 3.4 Calculation of the Initial Condition

The initial condition for the transient calculation is obtained by solving the steady-state neutron diffusion equation written in the form

$$-\frac{\partial}{\partial Z} D(Z) \frac{\partial}{\partial Z} \phi(Z) + \Sigma^a(Z)\,\phi(Z) = \frac{1}{\lambda} \nu\Sigma^f(Z)\phi(Z) \quad , \tag{3.21}$$

where $\lambda$ is an eigenvalue which will be unity to a physically critical reactor. Equation (3.21) is discretized as in Section (3.2). The resulting equations can be written in the matrix form

$$[M]\,\underline{\phi} = \frac{1}{\lambda}\,[F]\,\underline{\phi} \quad , \tag{3.22}$$

where $\underline{\phi}$ is a vector containing the discrete valves of the flux, [M] is a tridiagonal matrix, and [F] is a diagonal matrix of $\nu\Sigma^{f,k}$ values.

*TOS ≡ THOR operating system

Equation (3.22) is solved using the power method:[5]

$$\underline{S}^{(n)} = [F]\, \underline{\Phi}^{(n-1)} \tag{3.23a}$$

$$\underline{\Phi}^{(n)} = \frac{1}{\lambda^{(n-1)}} [M]^{-1} \underline{S}^{(n)} \tag{3.23b}$$

$$\lambda^{(n)} = \lambda^{(n-1)} \frac{||[F]\, \underline{\Phi}^{(n)}||_1}{||\underline{S}^{(n)}||_1} , \tag{3.23c}$$

where n denotes the fission source iteration index, and $||\cdot||_1$ is the $L_1$ vector norm. The indicated matrix inversion is performed using a direct matrix factorization technique;[5] a forward elimination, backward substitution procedure is then used to calculate the fluxes at each iteration. The iterative procedure given by Eq. (3.23) is accelerated using a simple asymptotic source extrapolation technique.[6] (A Wielandt iteration procedure applied to Eq. (3.22) may be more efficient than this accelerated power method; the use of the Wielandt method to solve Eq. (3.22) should be investigated as part of future work.) The converged eigenvalue is equal to the effective multiplication factor ($k_{eff}$). To insure that the reactor is critical, the $\nu\Sigma^f$ valves are divided by $k_{eff}$ prior to beginning the transient calculation.

## 3.5 The Space-Dependent Feedback Model

The feedback model used in THOR is the same as that used in the BNL-TWIGL code.[7] Cross section variations with average fuel temperature ($\bar{T}_f$), average moderator temperature ($\bar{T}_m$), and average moderator void fraction ($\bar{\alpha}_m$) in each axial heat conduction node are represented in the form

$$\Sigma(\bar{T}_f, \bar{T}_m, \bar{\alpha}_m) = \Sigma^* (\bar{T}_f^*, \bar{T}_m^*, \bar{\alpha}_m^*) + \frac{\partial \Sigma}{\partial \sqrt{\bar{T}_f}} (\sqrt{\bar{T}_f} - \sqrt{\bar{T}_f^*})$$

$$+ \frac{\partial \Sigma}{\partial \bar{T}_m} (\bar{T}_m - \bar{T}_m^*) + \frac{\partial \Sigma}{\partial \bar{\alpha}_m} (\bar{\alpha}_m - \bar{\alpha}_m^*) + C(\bar{\alpha}_m - \bar{\alpha}_m^*)^2 \qquad (3.24)$$

where $\Sigma$ represents the absorption cross section, the fission cross section, or the diffusion coefficient, and the starred quantities denote reference values. A set of feedback coefficients ( $\frac{\partial \Sigma}{\partial \sqrt{\bar{T}_f}}$, $\frac{\partial \Sigma}{\partial \bar{T}_m}$, C, etc. ) and reference cross sections must be supplied by the user for each nuclear composition. The reference thermohydreulic properties are taken to be the converged THOR steady-state properties. However, if there is iteration between the neutronics and thermohydraulic segments during the steady-state calculation, the thermohydraulic properties (on which the reference cross sections are based) must be supplied by the user as well.

At each time step of the transient calculation, the thermodydraulic properties $\bar{T}_f$, $\bar{T}_m$, and $\bar{\alpha}_m$ are calculated using the axial power distribution from the previous time step. These new properties are used in Eq. (3.24) to update the cross sections, which are then used to solve for the new axial power distribution as described in Section 3.3. A similar transfer of information can be used during the steady-state calculation in order to insure that the final converged critical power distribution is consistent with the final steady-state thermohydraulic properties.

## 3.6 Representation of Control Rod Motion

Control rod movement as a result of a reactor scram or rod withdrawal is modeled in the same manner[8] as in the BNL-TWIGL code.[7] A control density $f^k(t)$ is defined as the fraction of control rods present in mesh interval k as a function of time such that $0 < f^k(t) < 1$. The cross sections in each mesh interval are then calculated as weighted averages of

the "controlled"($\Sigma_c^k$) and "uncontrolled" ($\Sigma_{uc}^k$) cross sections, i.e.

$$\Sigma^k(t) = \Sigma_{uc}^k + f^k(t)\left[\Sigma_c^k - \Sigma_{uc}^k\right] . \qquad (3.25)$$

(The time-dependence of the cross sections $\Sigma_c^k$ and $\Sigma_{uc}^k$ due to feedback (Eq. (3.24)) has been suppressed here.) The initial controlled and uncontrolled cross sections for each nuclear composition and the initial control densities $f^k(0)$ must be supplied by the user. Control rod movement is represented by shifting the control density distribution at a constant (user-specified) rod velocity. The control rods can enter from either the top or the bottom of the core.

## 4. RESULTS AND DISCUSSION

The point kinetics and space-dependent kinetics options were initially debugged by solving simple problems for which analytical solutions can be obtained. The neutron kinetics model has been successfully linked with the coding for the core thermalhydraulics. A detailed description of the input data required for the neutron kinetics model is given in Appendix A.

The input data and output for a sample problem is given in Appendix B. This problem represents a control rod withdrawal from an initial critical condition, and was solved using the space-dependent kinetics option. All feedback coefficients for this sample problem were set to zero. The reactor power is initially 3400 MW, and remains constant during the time interval $0 < t < 0.3$ s. At $t = 0.3$ s, a control rod is withdrawn and the power increases gradually until $t = 2.9$ s. A reactor scram occurs at this time since the total power has reached the user-specified trip value of 3500 MW. Following the scram, the total power decreases rapidly until it is only 390 MW at $t = 4.0$ s. This sample problem (with 28 spatial mesh points) required only 0.1 s on the BNL CDC 7600 computer to calculate the initial (critical) condition and the transient solution shown in Appendix B.

In conclusion, the flexibility of the neutron kinetics model described in this report allows the accurate calculation of the axial power distribution for a wide range of reactor transients. Future work should include the extension of the 1-group space-dependent kinetics model to include 2 prompt neutron energy groups. The 2-group model would thus eliminate the need to collapse the often available 2-group cross section and feedback data in order to obtain the 1-group constants. Furthermore, the 2-group model would also provide a more accurate

representation of the significantly different effects on the fast and thermal cross sections due to changes in the core thermohydraulic properties.

## 5. REFERENCES


1. "THOR-1 (PWR): A Computer Code for Predicting the Thermal Hydraulic Behavior of Nuclear Reactor Systems," BNL-NUREG-24760, Brookhaven National Laboratory (1978).

2. "TRAC-P1: An Advanced Best-Estimate Computer Program for PWR LOCA Analysis," Los Alamos Scientific Laboratory (1978).

3. "RELAP 4/MOD 5: A Computer Program for Transient Thermal Hydraulic Analysis of Nuclear Reactors and Related Systems," ANCR-NUREG-1335, Idaho National Engineering Laboratory (1976).

4. "RETRAN - A Program for One-Dimensional Transient Thermal Hydraulic Analyses of Complex Fluid Flow Systems," EPRI-NP-408, Electric Power Research Institute (1977). See also "ANS Standard Decay Energy Release Rates Following Shutdown of Uranium-fueled Thermal Reactors.

5. E. L. Wachpress, Iterative Solution of Elliptic Systems, Prentice-Hall, Inc., Englewood Cliffs, N.J. (1966).

6. M. R. Wagner, "GAUGE-A Two-Dimensional Few Group Neutron Diffusion-Depletion Program for a Uniform Triangular Mesh," GA-8307, Gulf General Atomic (1968).

7. D. Diamond, ed., "BNL-TWIGL, A Program for Calculating Rapid LWR Transients," BNL-NUREG-21925, Brookhaven National Laboratory (1976).

8. Hsiang-Shou Cheng, "Scram Bank and Rod Drop Simulations," BNL-20059, Brookhaven National Laboratory (1975).

# APPENDIX A: PREPARATION OF INPUT DATA

The following data for the neutron kinetics calculation must be supplied by the user:

CARD 1

Format: 1615

ICALC Calculation Option:

= 1 for table look-up method

= 2 for point kinetics

= 3 for space-dependent kinetics

CARD 2 (omit if ICALC ≠ 1)

Format: 1615

NTPTS No. of time points at which either total power (ICALC = 1) or reactivity (ICALC = 2) are specified.

CARD 3 (omit if ICALC ≠ 1)

Format: 8E 10.0

TPT(1), POWT(1), ...., TPT(NTPTS), POWT(NTPTS)

A sequence of time (sec) intervals and corresponding total powers (MW).

CARD 4 (omit if ICALC ≠ 2)

Format: I5,3E10.0, I5

NTPTS No. of time intervals

GENT Prompt neutron generation time (sec)

TBETA Total delayed neutron fraction

POWER Initial total power (MW)

IREAC Option of reactivity input

= 0 for table look-up

= 1 for feedback calculation

CARD 5 (omit if ICALC ≠ 2)

Format: 8E10.0

A sequence of time (sec) intervals and corresponding reactivities ($).

TPT(1), POWT(1), ...., TPT(NTPTS), POWT(NTPTS)

CARD 6 (omit if ICALC ≠ 2 or IREAC ≠ 1)

Format: 8E10.0

Point kinetics feedback coefficients [Eq. (2.11)]

PKFBC(1) $[°K^{-1/2}]$ with respect to changes in the fuel temperature

PKFBC(2) $[°K^{-1}]$ with respect to changes in the moderator temperature

PKFBC(3) $[(g/cm^3)^{-1}]$ with respect to changes in the moderator density

CARD 7 (omit if ICALC ≠ 3)

Format: 16I5

| | |
|---|---|
| NOVL | No. of overlay regions ($\leq 6$) |
| NCP | No. of different compositions |
| NLP | No. of mesh points in lower plenum |
| NCORE | No. of mesh points in core |
| NUP | No. of mesh points in upper plenum |
| NDNG | No. of delayed neutron groups ($0 \leq$ NDNG $\leq 6$) |
| IMOVE | Control rod motion |
| | = -1 for withdrawal |
| | = 0 for no motion |
| | = 1 for scram |
| INIT | Initial power shape |
| | = 0 for read |
| | = 1 for calculation |

CARD 8 (omit if ICALC ≠ 3 or INIT ≠ 1)

Format: 2I5, 3E10.0

ITERSS Iterate between neutronics and thermalhydraulics at steady-state.

= 0 yes

= 1 no

IMAX Maximum No. of steady-state iterations

ERRSS Pointwise convergence criterion at steady-state

ERRSE Source extrapolation criterion (use 0.1)

CARD 9 (omit if ICALC ≠ 3)

Format: I5, 3E10.0

NSMAX Maximum No. of neutronics time steps per thermohydraulics time step

EPS Error test constant

THET Time-differencing theta

CARD 10 (omit if ICALC ≠ 3)

Format: 8E10.0

VEL Neutron velocity (cm/sec)

XNU Average number of neutrons per fission

XNORM Fissions to power conversion ($3.203 \times 10^{-17}$)

POWER Initial power (MW)

CARD 11 (omit if ICALC ≠ 3)

Format: I5, 3E10.0

NTER Control rods enter from:

= -1 for top

= 1 for bottom

PTRIP Overpower trip value (MW)

VSCRAM Control rod scram velocity (cm/sec)

CARD 12 (omit if ICALC ≠ 3 or IMOVE = 0)

Format: 8E10.0

TMOVE Time when scram or withdrawal is initiated (sec).

CARD 13 (omit if ICALC ≠ 3 or IMOVE ≠ -1)

Format: I5, 3E10.0

IRE Control bank where a rod or a group of rods is withdrawn

FRE Fraction of control density of the withdrawn rod(s)

VRE Withdrawn rod velocity (cm/sec)

DRE Duration of rod withdrawal (sec)

CARD 14 (omit if ICALC ≠ 3)

Format: 16I5

NPP No. of piecewise constants of initial control density profile. [$f^k(0)$, Eq. (3.25)].

A sequence of pairs of mesh points bounding each different piecewise constant.

IZ1(1), IZ2(1), ..., IZ1(NPP), IZ2(NPP).

CARD 15 (omit if ICALC ≠ 3)

Format: 8E10.0

FPP(1), ..., FPP(NPP)

Piecewise constants for initial control density profile.

CARD 16 (omit if ICALC ≠ 3)

Format: 3I5, 7D10.0

NLPT Left hand boundary of region

NRPT Right hand boundary of region

NCOMP(NR) Nuclear composition in each region

DZ(NR) Width of each mesh interval for each region (cm)

CARDS 17-24 contain the nuclear cross sections and feedback data [
Eq. (3.24)]. The second subscript in each array denotes the cross section type: 1 ≡ diffusion coefficient

2 ≡ absorption cross section

3 ≡ nu time fission cross section.

Hence, for example,

REFXS(NC,1) ≡ Diffusion coefficient for composition NC (cm)

REFXS(NC,2) ≡ Absorption cross section for composition NC ($cm^{-1}$)

REFXS(NC,3) ≡ Nu times fission cross section for composition NC($cm^{-1}$).

CARDS 17 (omit if ICALC ≠ 3)

Format: 8E10.0

Cross sections at zero void (uncontrolled)

1 card for each nuclear composition (NC)

REFXS(NC,1), REFXS(NC,2), REFXS(NC,3)

CARDS 18 (omit if ICALC ≠ 3)

Format: 8E10.0

Cross sections of zero void (controlled)

1 card for each nuclear composition (NC)

REFXS(NCP + NC,1), REFXS(NCP + NC,2), REFXS(NCP + NC,3)

CARDS 19 (omit if ICALC ≠ 3)

Format: 8E10.0

Linear coefficients for void effect (uncontrolled)

1 card for each nuclear composition (NC)

VOID(NC, 1, 1), VOID(NC, 2, 1) VOID(NC, 3, 1)

CARDS 20 (omit if ICALC ≠ 3)

Format: 8E10.0

Linear coefficients for void effect (controlled)

1 card for each nuclear composition (NC)

VOID (NCP + NC, 1,1), VOID(NCP + NC, 2,1), VOID(NCP + NC, 3,1)

CARDS 21 (omit if ICALC ≠ 3)

Format: 8E10.0

Quadratic coefficients for void effect (uncontrolled)

1 card for each nuclear composition (NC)

VOID(NC,1,2) VOID(NC,2,2), VOID(NC,3,2)

CARDS 22 (omit if ICALC ≠ 3)

Format: 8E10.0

Quadratic coefficients for void effect (controlled)

1 card for each nuclear composition (NC)

VOID(NCP + NC, 1,2), VOID(NCP + NC, 2,2), VOID(NCP + NC, 3,2)

CARDS 23 (omit if ICALC ≠ 3)

Format: 8E10.0

Fuel temperature coefficients ($cm^{-1}\ °K^{-1/2}$)

1 card for each nuclear composition (NC)

FBC(NC,1,1), FBC(NC,2,1), FBC(NC,3,1)

CARDS 24 (omit if ICALC ≠ 3)

Format: 8E10.0

Coolant temperature coefficients ($cm^{-1}\ °K^{-1/2}$)

1 card for each nuclear composition (NC)

FBC(NC,1,2), FBC(NC,2,2), FBC(NC,3,2)

CARDS 25 (omit if ICALC ≠ 3 and ITERSS ≠ 1)

Format: 8E10.0

Reference thermohydraulic properties [Eq. (3.24)]

1 card for each nuclear composition

REFP(NC,1) = Fuel temperature (°K) for composition NC

REFP(NC,2) = Moderator temperature (°K) for composition NC

REFP(NC,3) = Void fraction for composition NC

CARD 26 (omit if ICALC ≠ 3)

Format: 8E10.0

Group delayed neutron fractions, $\beta_i$

BETA(1), ..., BETA(NDNG)

CARD 27 (omit if ICALC ≠ 3)

Format: 8E10.0

Group delayed-neutron decay constants, $\lambda_i$ ($sec^{-1}$)

DECAY(1), ..., DECAY(NDNG)

CARD 28 (omit if INIT = 1)

Format: 8E10.0

Initial power shape

FLUX(1), ..., FLUX(M),

where M is the total number of meshpoints.

# APPENDIX B: SAMPLE INPUT AND OUTPUT

## B.1 Sample Input Data

```
*EOR
     3
     3     2     2    24     2     6    -1     1
     0   500    1.0E-05    1.0E-01
    32         0.25     -   1.0
    2.2E+05          2.40 3.203E-17    3.4E+03
    -1  3.55E+03          150.
        0.3
     1            .2          195.           1.
     3    17    19    20    23    24    28
          .3          .4          .5
     1     2     1          12.5
     3    26     2          12.5
    27    28     1          12.5
          1.0    8.0E-02     0.0E+00
          1.0    8.0E-02     0.0E+00
          0.0         0.0         0.0
          0.0         0.0         0.0
          0.0         0.0         0.0
          0.0         0.0         0.0
          0.0         0.0         0.0
          0.0         0.0         0.0
          1.0    1.0E-01     1.0E-01
          1.0    1.5E-01     1.0E-01
          0.0         0.0         0.0
          0.0         0.0         0.0
          0.0         0.0         0.0
          0.0         0.0         0.0
          0.0         0.0         0.0
          0.0         0.0         0.0
   2.50E-04  1.64E-03  1.47E-03  2.96E-03  3.60E-04  3.20E-04
   1.24E-02  3.05E-02  1.11E-01  3.01E-01  1.14E+00  3.01E+00
#EOR
#EOF
```

## B.2 Sample Output

```
REACTOR POWER CALCULATED USING SPACE-DEPENDENT KINETICS:
CARD 1:
     NO. OF OVERLAY REGIONS                          =          3
     NUMBER OF DIFFERENT COMPOSITIONS                =          2
     NO. OF MESH POINTS IN LOWER PLENUM              =          2
     NO. OF MESH POINTS IN CORE                      =         24
     NO. OF MESH POINTS IN UPPER PLENUM              =          2
     NO. OF DELAYED NEUTRON GROUPS                   =          6
     CONTROL ROD MOTION
     (-1/0/1 = WITHDRAWAL/NO/SCRAM)                  =         -1
     INITIAL POWER SHAPE  (0/1 = READ/CALC.)         =          1
CARD 2:
     ITERATE BETWEEN NEUTRONICS AND THERMO-
      HYDRAULICS AT S-S?  (0/1 = NO/YES)             =          0
     MAXIMUM NO. OF STEADY-STATE ITERATIONS          =        500
     POINTWISE CONVERGENCE CRITERION AT S-S          =  1.00E-05
     SOURCE EXTRAPOLATION CRITERION                  =  1.00E-01
CARD 3:
     MAXIMUM NO. OF NEUTRONICS TIME STEPS
      PER THERMOHYDRAULICS TIME STEP                 =         32
     ERROR TEST CONSTANT                             =  2.50E-01
     TIME-DIFFERENCING THETA                         =        1.0
CARD 4:
     NEUTRON VELOCITY                                =  2.20E+05
     AVG. NO. OF NEUTRONS PER FISSION                =  2.40E+00
     FISSIONS TO POWER CONVERSION                    = 3.203E-17
CARD 5:
     ROD ENTERS FROM (-1/1=TOP/BOTTOM)               =         -1
     OVERPOWER TRIP VALUE (MW)                       =  3.55E+03
     SCRAM VELOCITY (CM/SEC)                         = -1.50E+02
CARD 6:
     TIME WHEN SCRAM OR WITHDRAWAL IS
      INITIATED                                      =  3.00E-01
CARD 7:
     CONTROL BANK WHERE A ROD OR A GROUP
      OF RODS IS WITHDRAWN                           =          1
     FRACTION OF CONTROL DENSITY OF
      THE WITHDRAWN ROD(S)                           =       .200
     WITHDRAWN ROD VELOCITY (CM/SEC)                 =  1.95E+02
     DURATION OF ROD WITHDRAWAL (SEC)                =  1.00E+00
```

MESH AND COMPOSITION DATA:

| OVERLAY | ZPOINT TO ZPOINT | | COMPOSITION | DELTA Z |
|---|---|---|---|---|
| 1 | 1 | 2 | 1 | 12.5000 |
| 2 | 3 | 26 | 2 | 12.5000 |
| 3 | 27 | 28 | 1 | 12.5000 |

CROSS SECTIONS AT ZERO VOID (UNCONTROLLED):

| COMPOSITION | DIFFUSION | ABSORPTION | NU FISSION |
|---|---|---|---|
| 1 | 1.0000E+00 | 8.0000E-02 | 0. |
| 2 | 1.0000E+00 | 1.0000E-01 | 1.0000E-01 |

CROSS SECTIONS AT ZERO VOID (CONTROLLED):

| COMPOSITION | DIFFUSION | ABSORPTION | NU FISSION |
|---|---|---|---|
| 1 | 1.0000E+00 | 8.0000E-02 | 0. |
| 2 | 1.0000E+00 | 1.5000E-01 | 1.0000E-01 |

LINEAR COEFFICIENTS FOR VOID EFFECT (UNCONTROLLED):

| COMPOSITION | DIFFUSION | ABSORPTION | NU FISSION |
|---|---|---|---|
| 1 | 0. | 0. | 0. |
| 2 | 0. | 0. | 0. |

LINEAR COEFFICIENTS FOR VOID EFFECT (CONTROLLED):

| COMPOSITION | DIFFUSION | ABSORPTION | NU FISSION |
|---|---|---|---|
| 1 | 0. | 0. | 0. |
| 2 | 0. | 0. | 0. |

QUADRATIC COEFFICIENTS FOR VOID EFFECT (UNCONTROLLED):

| COMPOSITION | DIFFUSION | ABSORPTION | NU FISSION |
|---|---|---|---|
| 1 | 0. | 0. | 0. |
| 2 | 0. | 0. | 0. |

QUADRATIC COEFFICIENTS FOR VOID EFFECT (CONTROLLED):

| COMPOSITION | DIFFUSION | ABSORPTION | NU FISSION |
|---|---|---|---|
| 1 | 0. | 0. | 0. |
| 2 | 0. | 0. | 0. |

FUEL TEMPERATURE COEFFICIENTS:

| COMPOSITION | DIFFUSION | ABSORPTION | NU FISSION |
|---|---|---|---|
| 1 | 0. | 0. | 0. |
| 2 | 0. | 0. | 0. |

COOLANT TEMPERATURE COEFFICIENTS:

| COMPOSITION | DIFFUSION | ABSORPTION | NU FISSION |
|---|---|---|---|
| 1 | 0. | 0. | 0. |
| 2 | 0. | 0. | 0. |

DELAYED NEUTRON DATA:

| I | BETA(I) | DECAY(I) |
|---|---|---|
| 1 | 2.500E-04 | 1.240E-02 |
| 2 | 1.640E-03 | 3.050E-02 |
| 3 | 1.470E-03 | 1.110E-01 |
| 4 | 2.960E-03 | 3.010E-01 |
| 5 | 9.600E-04 | 1.140E+00 |
| 6 | 200E-03 | 3.010E+00 |

TBETA = .500E-03

MESH LAYOUT:

| MESH INTERVAL | INITIAL CONTROL DENSITY | COMPOSITION | HEAT CONDUCTION NODE |
|---|---|---|---|
| 28 | .5 | 1 | 8 |
| 27 | .5 | 1 | 8 |
| 26 | .5 | 2 | 6 |
| 25 | .5 | 2 | 6 |
| 24 | .5 | 2 | 6 |
| 23 | .4 | 2 | 6 |
| 22 | .4 | 2 | 5 |
| 21 | .4 | 2 | 5 |
| 20 | .4 | 2 | 5 |
| 19 | .3 | 2 | 5 |
| 18 | .3 | 2 | 4 |
| 17 | .3 | 2 | 4 |
| 16 | 0.0 | 2 | 4 |
| 15 | 0.0 | 2 | 4 |
| 14 | 0.0 | 2 | 3 |
| 13 | 0.0 | 2 | 3 |
| 12 | 0.0 | 2 | 3 |
| 11 | 0.0 | 2 | 3 |
| 10 | 0.0 | 2 | 2 |
| 9 | 0.0 | 2 | 2 |
| 8 | 0.0 | 2 | 2 |
| 7 | 0.0 | 2 | 2 |
| 6 | 0.0 | 2 | 1 |
| 5 | 0.0 | 2 | 1 |
| 4 | 0.0 | 2 | 1 |
| 3 | 0.0 | 2 | 1 |
| 2 | 0.0 | 1 | 7 |
| 1 | 0.0 | 1 | 7 |

339 FISSION SOURCE ITERATIONS
FISSION SOURCE CONVERGENCE = 9.68E-06

FINAL STEADY-STATE CONDITIONS:
POINTWISE VALUES OF POWER:

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| 1 | 0. | 2 | 0. | 3 | 7.5389E+01 | 4 | 1.4243E+02 |
| 6 | 2.5624E+02 | 7 | 2.9829E+02 | 8 | 3.2796E+02 | 9 | 3.4403E+02 |
| 11 | 3.3326E+02 | 12 | 3.0688E+02 | 13 | 2.6776E+02 | 14 | 2.1753E+02 |
| 16 | 9.2453E+01 | 17 | 2.2795E+01 | 18 | 5.6170E+00 | 19 | 1.3709E+00 |
| 21 | 5.7601E-02 | 22 | 1.1804E-02 | 23 | 2.4052E-03 | 24 | 4.2277E-04 |
| 26 | 1.2815E-05 | 27 | 0. | 28 | 0. | | |

EFFECTIVE MULTIPLICATION FACTOR = .9973515
INITIAL REACTOR POWER (MW) = 3.400E+03

```
 TRANSIENT RESULTS:
TIME = 1.000E-01      REACTOR POWER = 3.400E+03
TIME = 2.000E-01      REACTOR POWER = 3.400E+03
TIME = 3.000E-01      REACTOR POWER = 3.400E+03
TIME = 4.000E-01      REACTOR POWER = 3.489E+03
TIME = 5.000E-01      REACTOR POWER = 3.512E+03
TIME = 6.000E-01      REACTOR POWER = 3.516E+03
TIME = 7.000E-01      REACTOR POWER = 3.519E+03
TIME = 8.000E-01      REACTOR POWER = 3.521E+03
TIME = 9.000E-01      REACTOR POWER = 3.523E+03
TIME = 1.000E+00      REACTOR POWER = 3.525E+03
TIME = 1.100E+00      REACTOR POWER = 3.527E+03
TIME = 1.200E+00      REACTOR POWER = 3.528E+03
TIME = 1.300E+00      REACTOR POWER = 3.530E+03
TIME = 1.400E+00      REACTOR POWER = 3.532E+03
TIME = 1.500E+00      REACTOR POWER = 3.532E+03
TIME = 1.600E+00      REACTOR POWER = 3.533E+03
TIME = 1.700E+00      REACTOR POWER = 3.535E+03
TIME = 1.800E+00      REACTOR POWER = 3.536E+03
TIME = 1.900E+00      REACTOR POWER = 3.538E+03
TIME = 2.000E+00      REACTOR POWER = 3.539E+03
TIME = 2.100E+00      REACTOR POWER = 3.540E+03
TIME = 2.200E+00      REACTOR POWER = 3.542E+03
TIME = 2.300E+00      REACTOR POWER = 3.543E+03
TIME = 2.400E+00      REACTOR POWER = 3.544E+03
TIME = 2.500E+00      REACTOR POWER = 3.546E+03
TIME = 2.600E+00      REACTOR POWER = 3.547E+03
TIME = 2.700E+00      REACTOR POWER = 3.548E+03
TIME = 2.800E+00      REACTOR POWER = 3.549E+03
TIME = 2.900E+00      REACTOR POWER = 3.551E+03
TIME = 3.000E+00      REACTORPOWER = 3.226E+03
TIME = 3.100E+00      REACTOR POWER = 2.854E+03
TIME = 3.200E+00      REACTOR POWER = 2.457E+03
TIME = 3.300E+00      REACTOR POWER = 2.057E+03
TIME = 3.400E+00      REACTOR POWER = 1.673E+03
TIME = 3.500E+00      REACTOR POWER = 1.271E+03
TIME = 3.600E+00      REACTOR POWER = 9.531E+02
TIME = 3.700E+00      REACTOR POWER = 7.171E+02
TIME = 3.800E+00      REACTOR POWER = 5.547E+02
TIME = 3.900E+00      REACTOR POWER = 4.521E+02
TIME = 4.000E+00      REACTOR POWER = 3.897E+02
```