# Parallel Coupling Methodology for Evaluation of Local Safety Parameters in BWR

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## ABSTRACT

The purpose of this work is to extend the TRAC-BF1/NEM coupled code capabilities to obtain the transient fuel rod response. An implementation of a pin power reconstruction scheme coupled to a subchannel analysis module, based on the COBRA-TF thermal-hydraulics (T-H) analysis code is performed. The objective of the hot channel analysis procedure is to locate and calculate the fluid, thermal and neutronic conditions of the most limiting node (axial region of assembly/channel) within the core. TRAC-BF1/NEM is a system-coupled code consisting of the best-estimate T-H system simulation code TRAC-BF1 and a three-dimensional transient neutron kinetics nodal expansion method (NEM) code. An advanced pin power reconstruction (PPR) method, based on an extension of NEM, was developed and implemented into the computer program CELL. A within-node flux shape is constructed in a non-separable form using the NEM global data and then modulated with heterogeneous flux and power distribution obtained from the lattice physics calculation.

The hot channel analysis module for the local safety margin evaluation is integrated into the coupled TRAC-BF1/NEM calculation procedure using parallel virtual machine (PVM) environment. COBRA-TF improved heat-transfer models and capability for modeling the cross-flow into a subchannel make the code an ideal simulation tool for performing hot-channel analysis calculations. The COBRA-TF core T-H solution is performed in parallel to the TRAC-BF1 calculation. Boundary conditions for COBRA-TF are obtained from the TRAC-BF1 system model calculation. Nuclear cross-sections are updated using the TRAC-BF1 T-H data as feedback parameters.

# 1. INTRODUCTION

The use of coupled three-dimensional (3D) neutron kinetics/thermal-hydraulic system codes is becoming increasingly common as a means to more accurately model not only reactivity insertion accidents (RIA), but also reactor operational transients. These best estimate analysis tools describe more realistically the local core effects and coupled reactor core/plant dynamics interactions, and subsequently forecast safety margins more accurately. The recent 3D nodal kinetics models usually employ planar meshes that are of the size of fuel assemblies (or parts of assemblies). However, the accident consequences (fuel rod enthalpy, departure of nucleate boiling - DNB, or peak cladding temperature) need to be evaluated in terms of a single fuel rod (pin) response rather than assembly (bundle) average response. The modeling capabilities of the above mentioned coupled codes must be further extended to include a dehomogenization or pin flux and power reconstruction scheme, coupled to a subchannel model, in order to obtain the transient fuel rod response.

Adjacent fuel assemblies affect not only neutron spectrum and burnup characteristics, but also power distribution across the assembly. Static local power peaking factors are reconstructed as a part of current core design methodology (3D steady-state and depletion calculations) for accurate thermal margin (fuel temperature, enthalpy, and DNB) prediction. The effect on the power distribution because of the adjacent assemblies becomes more pronounced during space-dependent core accidents and should be taken into account in transient simulations. For point and 1D-kinetics models, pin peaking can be handled in a variety of ways that ensure an overall conservative result. Different fuel vendor approaches are briefly described in reference 1. In summary, a bounding value for the pin to node (assembly) factor can be taken from the static reload design codes (with appropriate adjustment to accommodate uncertainties). However, the pin power peaking factor calculated in this way can be not very well predicted for highly heterogeneous and changing core environment during some transients and accidents. This fact is especially important for cases where the fuel cladding temperature can exceed 800 C. Much more realistic transient pin power distribution can be predicted by applying a dehomogenization algorithm, based on a flux reconstruction scheme, which uses nodal results at each time step. A subchannel analysis module for obtaining thermal margins supplements the calculation approach used.

The selected subchannel analysis code COBRA-TF[2] uses a two-fluid, three-field representation of the two-phase flow in which each field is treated in three-dimensions and it is compressible. The three-fields are continuous vapor, continuous liquid, and entrained liquid drops. It features extremely flexible noding for both the hydrodynamic mesh and the fuel rod geometry. The forgoing flexibility provides the capability to model the wide variety of geometries encountered in vertical components of a nuclear reactor vessel. COBRA-TF can model the hot assembly and hot sub channel within the hot assembly such that the effect of local pin power and its uncertainties can be considered for the coolant conditions. Unlike earlier versions of the COBRA codes (COBRA-IIIC, and COBRA-IV-I) and the VIPRE codes[3], COBRA-TF has a full boiling curve for heat/mass/momentum transfer between the phases included into the code calculational scheme. Therefore, it can model transients in which the critical heat flux is exceeded.

The proposed improved methodology for hot channel analysis relies on the identification and explicit pin power recalculation (PPR) of relevant hot channels in the reactor core to evaluate the local fuel response with nuclear and T-H interactions. A hot channel usually represents a fuel assembly in BWR simulations since there is no cross-flow between assemblies. The details for geometry refinement for the COBRA-TF hot channel model are a user-defined option**.** For each axial level of the hot channel the local pin powers are calculated applying the PPR method. This method uses information provided by the NEM[4] nodal code and basic cross-section data. A dynamic algorithm has been developed to identify the most limiting channel and fuel assembly (radiallly) and axial region (node) based on the current state of the core.

Te first step in the verification process of the TRAC-BF1/NEM/COBRA-TF integrated safety analysis code utilizing the parallel virtual machine (PVM[5]) interface was completed. Testing of the multi-level coupling scheme was performed to examine such issues as functionality, convergence, and execution synchronization. The obtained results are discussed in detail in the following sections of this paper. Further testing was done to ensure that consistent realistic physical results are produced. Benchmark problems are being developed for code-to-code comparisons with CORETRAN[3]. Multiple links will be developed so that several assemblies may be modeled in parallel in the event the limiting channel changes during a given transient.

## 2. PIN POWER RECONTRUCTION ALGORITHM

In order to have a detailed flux distribution at the pin level, the pin power reconstruction method developed and implemented in the CELL code[6] was verified and implemented into the NEM code. Two sequential steps are performed into the CELL subroutine for obtaining the rod power as a part of the internal NEM neutronics calculations. The first step involves the determination of the corner values (namely, neutron flux and current). The second step involves the determination of power distribution at the pin level.

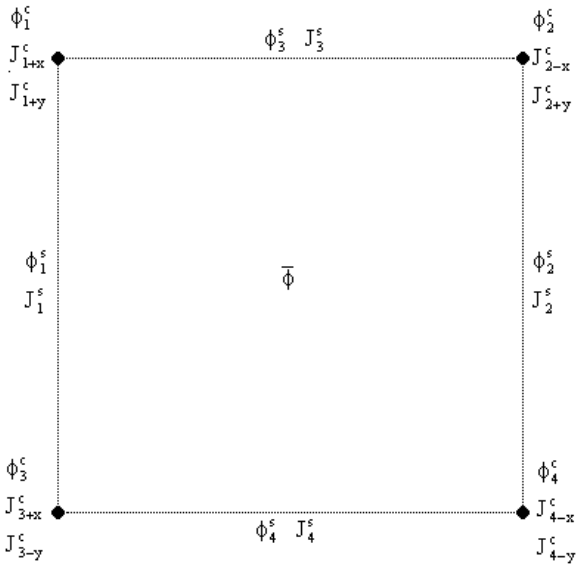


Figure 1. Flux and current data needed for a high order interpolation

One of the major changes introduced in the NEM PPR scheme is how the average flux expansion coefficients are obtained. Originally, the CELL algorithm was based on the nodal expansion coefficients, obtained in the framework of net current formulation. Subsequently, the algorithm was modified to match the partial current formulation of the NEM code. The CELL code is transformed into a NEM subroutine and implemented into the NEM source code and the power reconstruction is performed at both steady state and transient levels.

The first step in the pin power reconstruction involves the calculation of the corner fluxes and currents for the node of interest. All data shown in Figure 1, except the corner values, are directly available from the global nodal calculation. The notation used in this figure is as follows:

$\bar{\phi}$ = average nodal flux

$\phi_j^s$ = average surface flux for surface j

$\phi_j^c$ = flux at the corner j

$J_j^s$ = average surface net current for surface j

$J_{j\pm u}^c$ = net currents in the u-direction at the corner j (u = x,y)

For the central corner surrounded by the four nodes shown in Figure 2, the available nodal values shown there are used. A corner value is determined based on the four-node geometry shown in Figure 3. The flux distribution within each of these nodes is described by a bi-quadratic polynomial:

$$\phi_i(x,y) = \sum_{n=0}^{2} \sum_{m=0}^{2} a_{mn} \xi^m \rho^n \qquad (1)$$

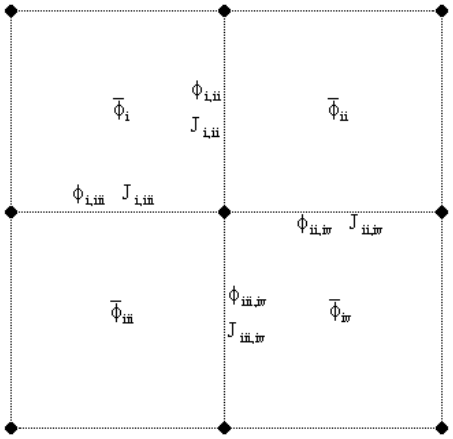


Figure 2. Nodal values for the four nodes joining at a corner

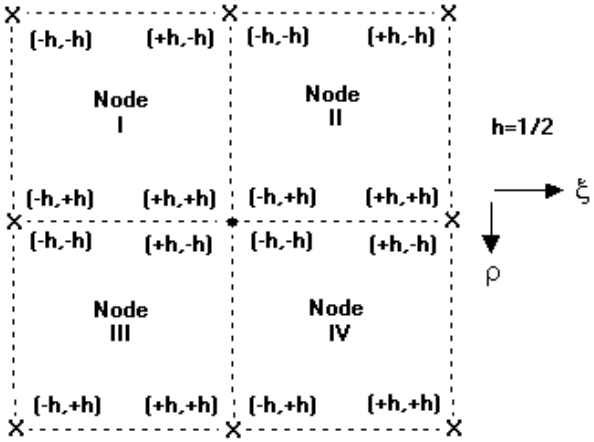


Figure 3. Four nodes joining at a corner and use of intrinsic coordinates

After applying a series of constraints imposed on the system, corner values are obtained. These corner values are still considered as homogeneous corner fluxes because their calculation is based on homogenous nodal values. A new approach is introduced by using heterogeneous

CASMO-3 calculations for obtaining heterogeneous corner fluxes. According to Moon[6], the flux continuity at the central corner can be described as follows:

$$\phi_{i}^{c} = \phi_{ii}^{c} = \phi_{iii}^{c} = \phi_{iv}^{c} \tag{2}$$

The average surface flux discontinuity, used in global nodal calculations, is extended here to corner flux discontinuity condition. After inserting form flux factors Equation (2) becomes:

$$H_{i}\phi_{i}^{c} = H_{ii}\phi_{ii}^{c} = H_{iii}\phi_{iii}^{c} = H_{iv}\phi_{iv}^{c} \tag{3}$$

The form flux factors used in the above equation are extracted from CASMO calculations and are included as a function of feedback parameters on the cross-section tables for a given analysis. Once the corner values are obtained, the within-node flux distribution is obtained applying a bi-quadruple polynomial interpolation. In the CELL code, the power series are used as the basis function for 1D transverse integrated flux representation:

$$\phi(u) = \sum_{m=0}^{M} C_{m}\xi^{m}, \qquad \xi = \frac{u}{\Delta u}, \quad u = x, y, z \tag{4}$$

where the CELL code uses nodal average values calculated by the nodal code, which represent the average flux in the node in term of power series. The transverse integrated flux in the NEM representation is:

$$\phi(u) = \bar{\phi}_{g}^{l} + \sum_{n=1}^{N} a_{n} f_{n}(u) \tag{5}$$

where the relationship between the $C_m$ and $a_n$ is:

$$C_{0} = \bar{\phi} + \frac{a_{4}}{80} - \frac{a_{2}}{4} \tag{6}$$

$$C_{1} = a_{1} - a_{3} \tag{7}$$

$$C_{2} = 3a_{2} - \frac{3}{10}a_{4} \tag{8}$$

$$C_{3} = a_{3} \tag{9}$$

$$C_{4} = a_{4} \tag{10}$$

and the $a_n$ nodal expansion coefficients are obtained from the global nodal expansion calculation. The non-separable within-node flux distribution is obtained using the following formulation (namely, the bi-quadruple expansion):

$$\Phi(x,y) = \sum_{m=0}^{4} \sum_{n=0}^{4} A_{mn} \xi^m \rho^n, \quad -\frac{1}{2} \le \xi \le +\frac{1}{2}, \quad -\frac{1}{2} \le \rho \le +\frac{1}{2} \tag{11}$$

The rest of the algorithm remains the same, except that after obtaining the intra-nodal flux distribution and, therefore the intra-nodal power, the detailed distribution is then transformed into a rod-wise power distribution using the rod factors obtained from lattice physics calculations.

$$P_{het}(x_i, y_j) = P_{hom}(x_i, y_j) \cdot P_{fac}(x_i, y_{j)} \tag{12}$$

$$P_{hom}(x_i, y_j) = \Sigma_{f1}(x_i, y_j) \cdot \bar{\phi}_1(x_i, y_j) + \Sigma_{f2}(x_i, y_j) \cdot \bar{\phi}_2(x_i, y_j) \tag{13}$$

where

$P_{het}(x_i, y_j) =$ heterogeneous rod power at location $(x_i,y_{j)}$

$P_{hom}(x_i, y_j) =$ homogeneous rod power at location $(x_i,y_j)$

$P_{fac}(x_i, y_j) =$ rod factor at location $(x_i,y_j)$

## 3. COUPLING DESIGN

### 3.1 PARALLEL COUPLING APPROACH

The COBRA-TF code is coupled to the TRAC-BF1/NEM[7] code using the parallel approach. PVM technology is used for coupling the codes. Because most of the information that COBRA-TF needs comes from the NEM calculation, COBRA-TF is initialized by the NEM code, which has a calculation flag for performing pin power reconstruction and hot-channel calculation. The general coupling approach is shown in Figure 4

The information transferred from NEM to COBRA-TF is the relative axial power distribution for the hot-channel on a pin-by-pin basis and the hot channel average linear heat generation rate. The T-H boundary conditions are obtained from the TRAC-BF1[8] T-H system calculation. The T-H information, that COBRA-TF needs, is dependent on the type of boundary conditions selected in the COBRA-TF calculations. These boundary conditions can be selected as described below.

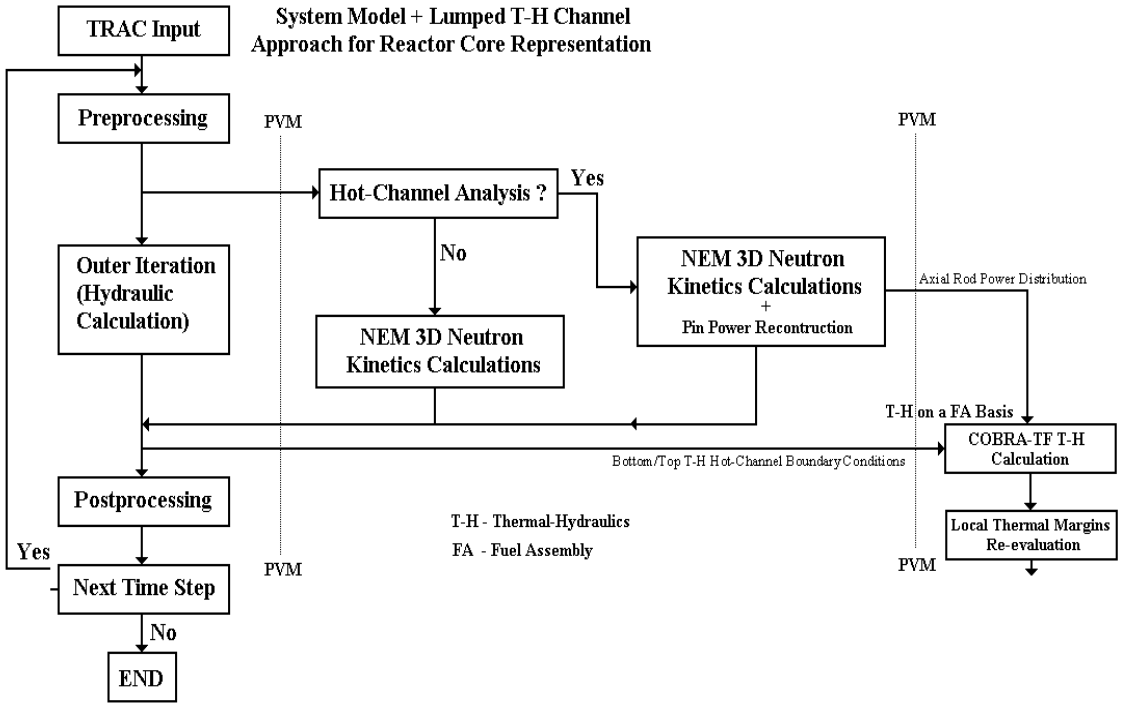


Figure 4. TRAC-BF1/NEM Hot-Channel Coupling Approach Using PVM

### 3.2 COBRA-TF BOUNDARY CONDITION MODEL

The following sets of boundary conditions are available for COBRA-TF hot-channel T-H calculation:

1) pressure and enthalpy
2) flow and enthalpy
3) flow only
4) mass source (flow rate and enthalpy)
5) pressure sink and enthalpy

The typical choice of boundary conditions for the channel inlet is to specify mass flow rate and enthalpy at the channel entrance and pressure and enthalpy at the channel outlet. One must assure about the correct use of the information obtained from the TRAC-BF1 T-H calculation and the above boundary conditions needed by the COBRA-TF code. This is done internally into the TRAC-BF1 source to assure that the units and T-H variables are consistent with the COBRA-TF T-H modeling.

All of the information transferred to COBRA-TF from TRAC-BF1/NEM is done using the PVM capabilities. The PVM environment is a unique tool that makes it possible to couple large codes written in the FORTRAN computer programming language. The calculations are then performed

on a multiple-processor basis. The PVM software was used to couple the TRAC-BF1 code and the NEM-3D code with COBRA-TF, based on the parallel approach. This approach reduces enormously the amount of new coding work and provides a calculation speedup.

## 4. HOT-CHANNEL ANALYSIS METHODOLOGY

The developed hot-channel modeling methodology explores an approach proposed for PWR[9] analysis and makes use of two basic approaches for thermal margins calculation. Figure 5 shows these two approaches. The first approach is to build the model having a degree of detail at the pin level, which should be very accurate, but also very computer time consuming. This one has the advantage that, for the case of BWR bundles, no horizontal boundary conditions are required because the can box acts basically as a zero-cross flow boundary condition. Cross-flow component is expected among the fuel pins inside the bundle can. This can be accounted for in the COBRA-TF 3D modeling capabilities.

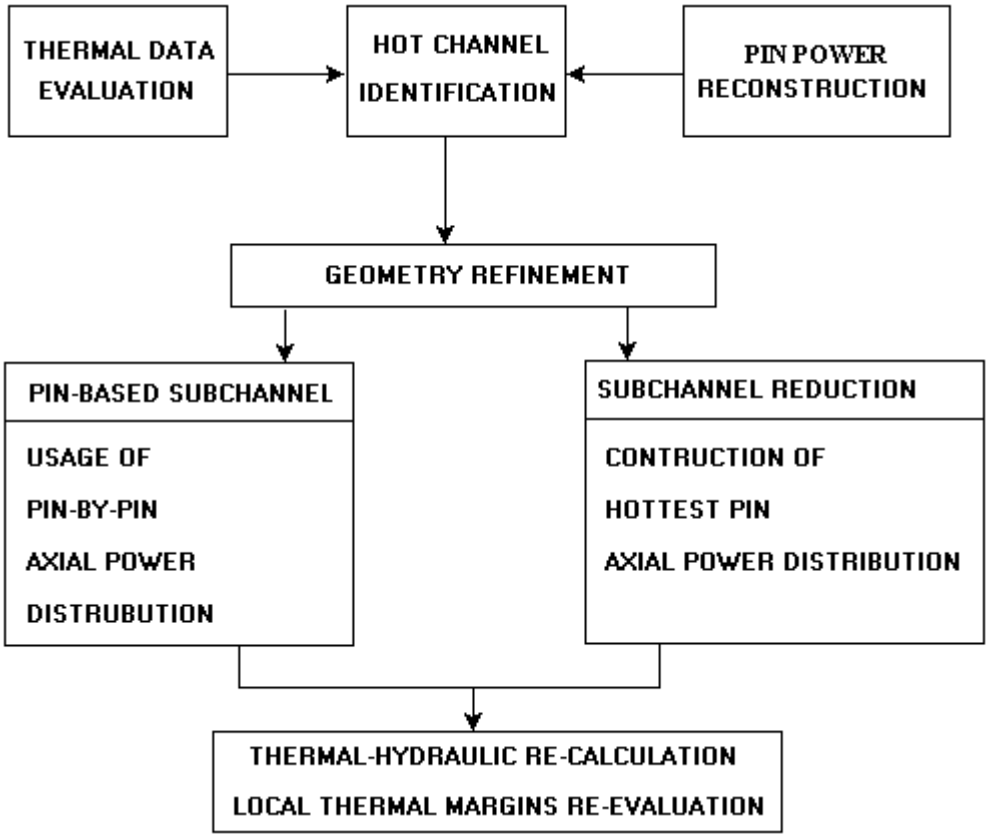


Figure 5. Scheme of the integrated local safety margin evaluation

Figure 6 shows the first approach that corresponds to a pin-based subchannel layout. Figure 7 shows an example of a few-zones subchannel layout (second approach). The later case could be built only if the proper horizontal flow boundary conditions are provided, which would require first using a refined model and once the hottest rod, and subsequently the channel of interest is

identified, the necessary boundary conditions are saved and later used in the simplified model. An obvious disadvantage of this model is that the part of the calculation has to be done off-line.

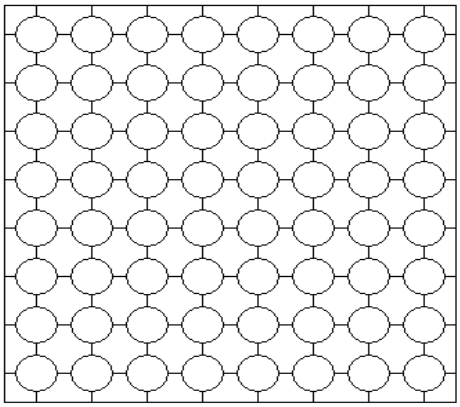

Figure 6. Pin-base subchannel layout

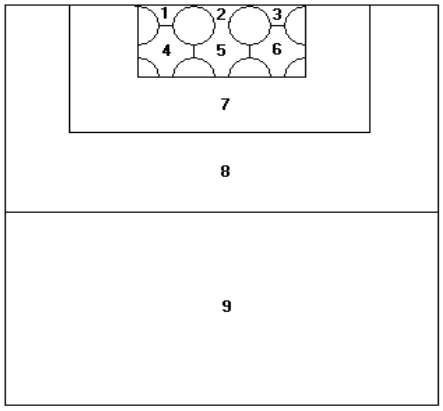


Figure 7. Few-zones subchannel layout

The COBRA-TF hot-channel model corresponding to a typical 8x8 fuel rod array is shown in Figure 8. In order to have a good level of detail (at the pin level) it is necessary to divide (at least for this specific example) the fuel bundle box into sixty-eight nuclear fuel rods and eighty-one T-H subchannels. Axially, the fuel bundle (channel) is typically divided into 24 or 25 cells of six-inches length. Using the boundary conditions provided as explained above, the fuel bundle is divided into equally spaced axial nodes (which are mapped to the neutronics channel) as it is shown in Figure 8. COBRA-TF solves then the field equations for each of these subchannels. The importance of using COBRA-TF for performing this kind of calculations is that COBRA-TF does account for the cross-flow occurring inside the fuel bundle.

For a specific application the mesh cells of COBRA-TF are defined by input in terms of CHANNELS, which are defined simply as a vertical stack of mesh cells. A channel may represent a subchannel between four fuel rods (e.g., Figure 8); several regions lumped together inside a fuel bundle, or a lumped region of the core (this case can be applied for PWR hot-channel analysis applications). The boundary conditions needed by COBRA-TF at the bottom and top part of the channels are transferred from TRAC-BF1 calculations. The boundary conditions required by COBRA-TF are flow rate and enthalpy for each channel present in the COBRA-TF model (inlet conditions) and either, pressure and enthalpy, or simply pressure at for the outlet conditions. Axial power profile data is provided by the NEM 3D calculations along with the radial pin power distribution and the average linear heat generation rate for the identified hot channel. The axial power distribution per each rod present in the hot-channel model is obtained as part of the pin power reconstruction capability implemented in NEM. A quasi-static analysis is performed by COBRA-TF using the time-varying rod power profiles and the time-varying T-H boundary conditions.

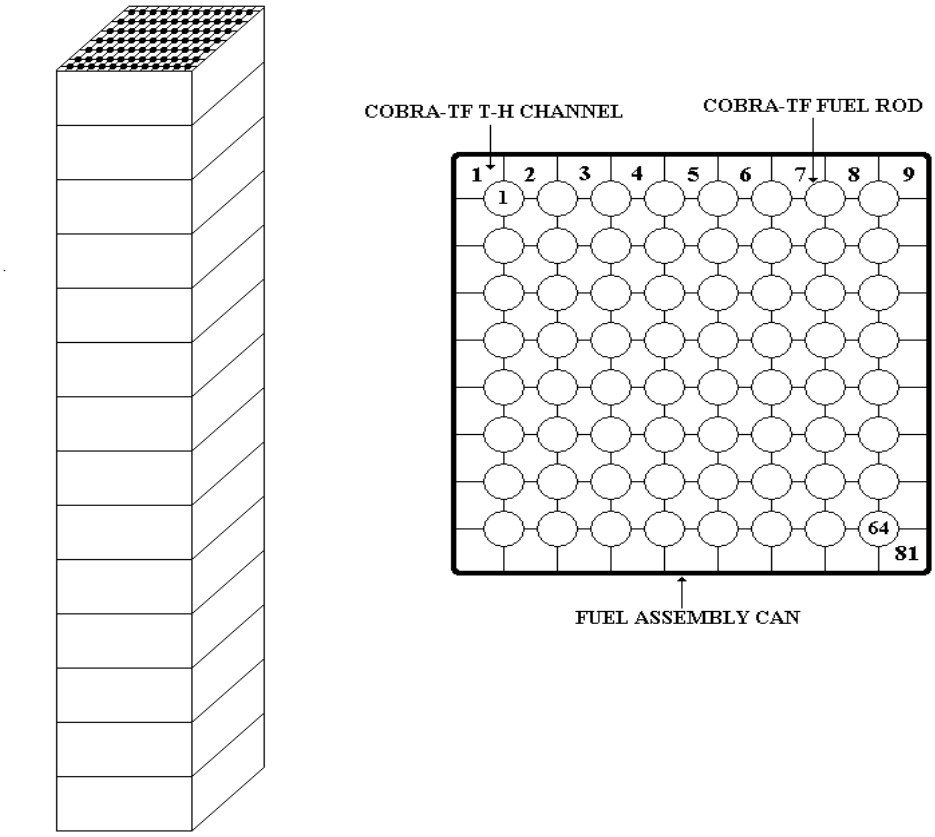


Figure 8. Typical 8x8 fuel Rod Array COBRA-TF Hot-Channel Model

## 5. STEADY STATE AND TRANSIENT RESULTS AND CALCULATION PERFORMANCE

As an illustrative example, the reconstructed homogeneous relative power radial distribution for the hottest node is presented in Figure 9. This distribution corresponds to a steady-state calculation of a control rod drop accident simulation for Laguna Verde NPP case. The rod peak power is shown there. Figure 10 shows the above distribution but with the inclusion of rod factors and form functions during the pin power reconstruction calculation (i.e., the so called heterogeneous distribution). Figures 11 and 12 correspond to the distribution obtained at the end of the transient. Transient maximum fuel temperature as obtained from COBRA-TF transient calculation is subsequently used for obtaining the fuel enthalpy using a special program designed to perform this calculation.

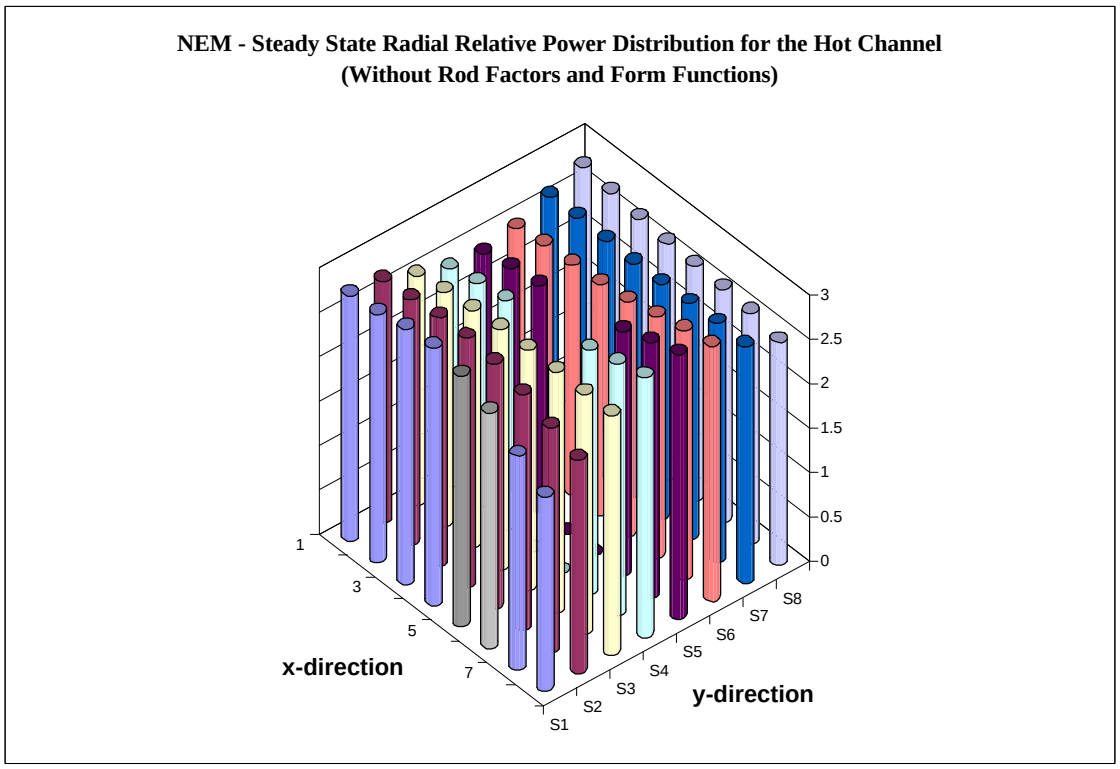


Figure 9

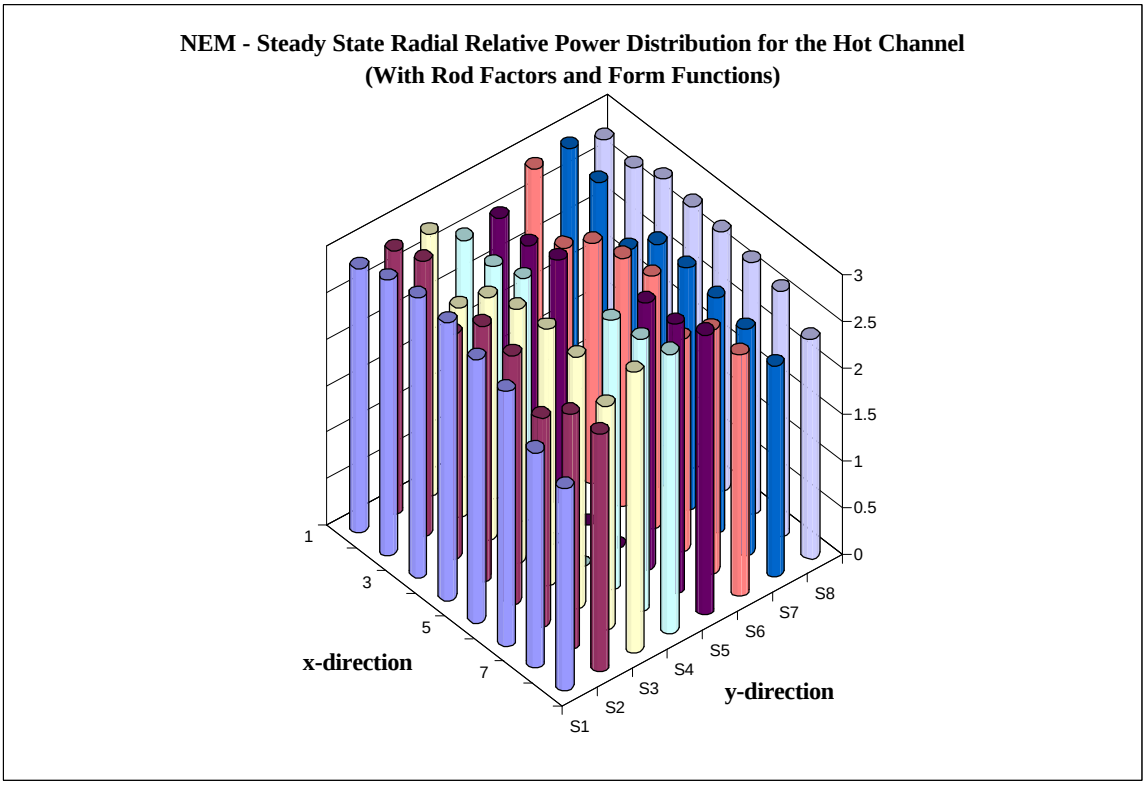


Figure 10

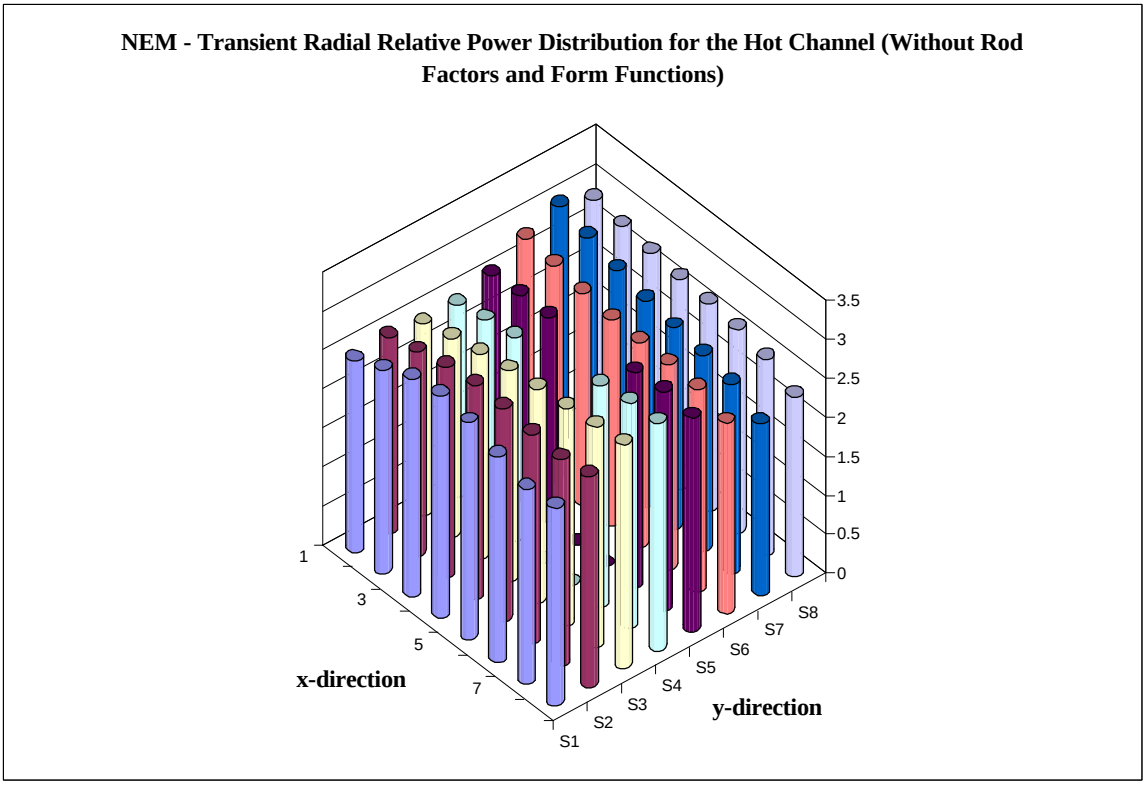


Figure 11

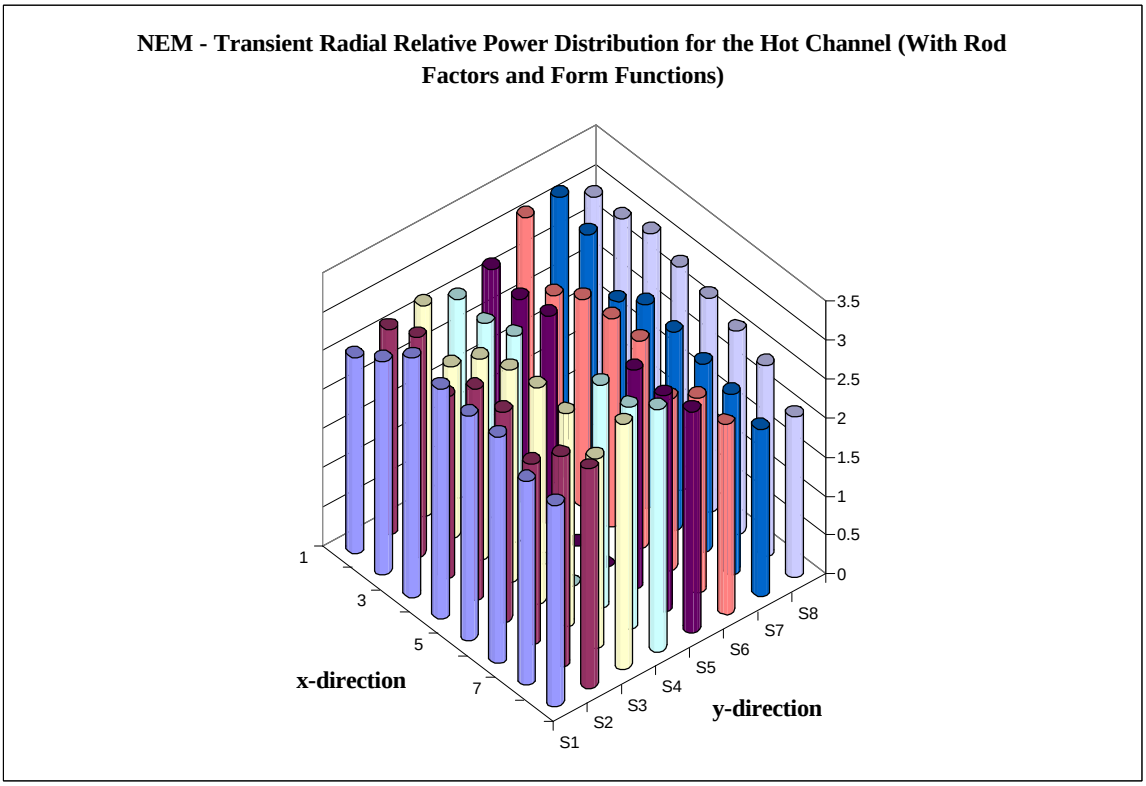


Figure 12

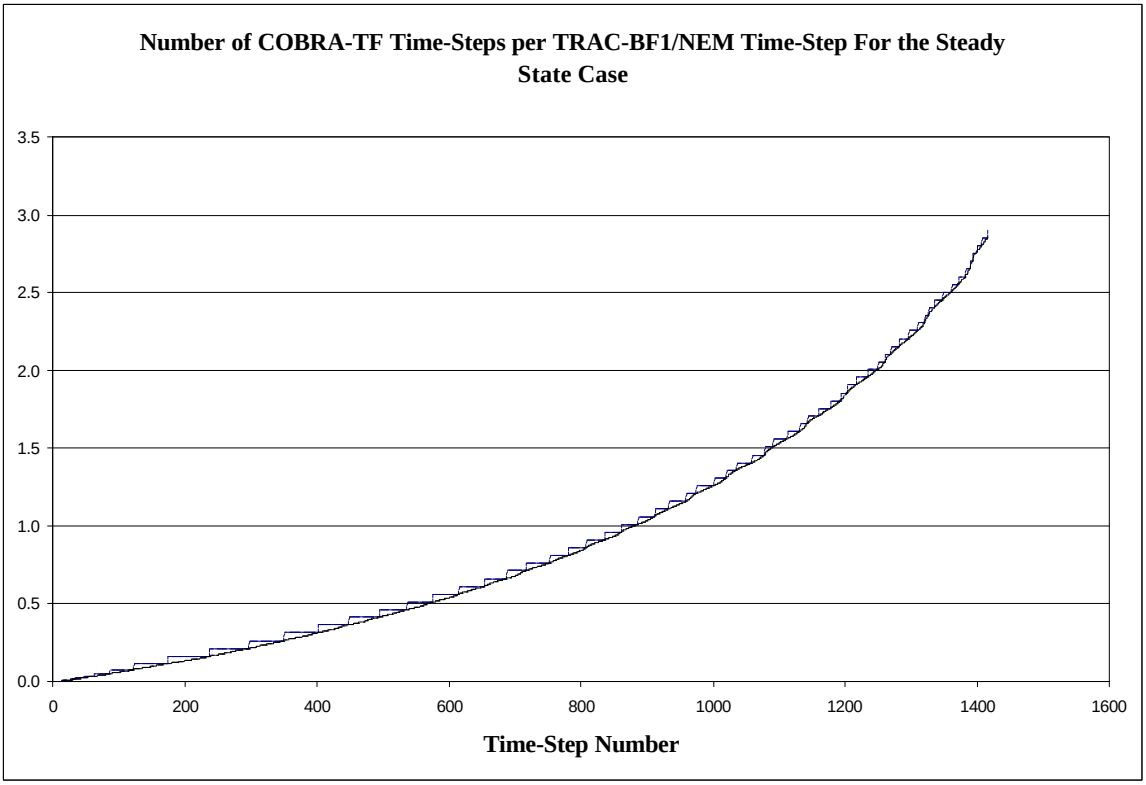


Figure 13

Figure 13 shows the number of COBRA-TF time steps required to complete the calculation of a TRAC-BF1/NEM time-step. In general, during the steady-state calculation, the number of COBRA-TF time-steps required to complete a TRAC-BF1/NEM time-step is larger than the number required in the transient case. This trend does not affect too much the calculation time for a given transient, since COBRA-TF is a transient code, and a steady state calculation is not required. Also, figure 15 demonstrates that execution synchronization is achieved during the calculation. During the transient case it was found out that the number of COBRA-TF time-steps required to complete a TRAC-BF1/NEM time-step calculation, was in the average of six.

## CONCLUSIONS

A hot-channel analysis capability, based on the COBRA-TF T-H analysis code, has been added to the TRAC-BF1/NEM 3-D neutron kinetics/T-H system coupled code. A parallel coupling methodology based on PVM is used. An improved pin power reconstruction scheme, based on the NEM calculation approach is used to obtain the relative pin power data needed for the COBRA-TF hot-channel T-H calculation. Boundary conditions are provided by the TRAC-BF1 T-H system model calculation. Nuclear cross-sections are updated using the TRAC-BF1 data as feedback parameters. Preliminary hot-channel analysis calculations have been performed to test the coupling approach used and to synchronize the coupled codes. Further testing was done to ensure that consistent realistic physical results are produced. Multiple links will be developed so that several assemblies may be modeled in parallel in the event the limiting channel changes during a given transient.

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