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# HIGH ACCURACY MODELING FOR ADVANCED NUCLEAR REACTOR CORE DESIGNS USING MONTE CARLO BASED COUPLED CALCULATIONS

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## ABSTRACT

The present trend in advanced and next generation nuclear reactor core designs is towards increased material heterogeneity and geometry complexity. The continuous energy Monte Carlo method has the capability of modeling such core environments with high accuracy. Because of its statistical nature, Monte Carlo core calculations usually involve a considerable computer time to attain reliable converged results for both integral parameters and local distributions. This is especially true when coupling it with a thermal-hydraulics code to obtain three-dimensional (3D) power and thermal-hydraulic solutions for a reactor core. Results from feasibility studies performed at the Pennsylvania State University (PSU) on both accelerating Monte Carlo criticality calculations by using hybrid nodal diffusion Monte Carlo schemes and thermal-hydraulic feedback modeling in Monte Carlo core calculations were presented in the previous publication at the ICAPP 2007 conference. This paper presents the new developments of the hybrid Monte Carlo-based coupled core calculations at PSU, and the results of applying the developed coupled hybrid system NEM/MCNP5/CTF to a simplified Three-Dimensional (3D) 2×2 fuel-pin array from a representative Boiling Water Reactor (BWR) assembly.

*Key Words*: *MCNP5, COBRA-TF, NEM, Coupled Calculations*

## 1. INTRODUCTION

Presently, deterministic codes based on the diffusion approximation of the Boltzmann transport equation, coupled with channel-based (or sub-channel based) thermal-hydraulic codes, carry out the three-dimensional (3D) reactor core calculations of the Light Water Reactors (LWRs). These deterministic codes utilize nuclear homogenized data (normally over large spatial zones, consisting of fuel assembly or parts of fuel assembly, and in the best case, over small spatial zones, consisting of pin cell), which is functionalized in terms of thermal-hydraulic feedback parameters (in the form of off-line pre-generated cross-section libraries). The present trend in advanced and next generation nuclear reactor core designs is towards increased material heterogeneity and geometry complexity. High accuracy modeling is required for these advanced nuclear reactor core designs. Such high-fidelity methods take advantage of the recent progress in computation technology and coupled neutron transport solutions with thermal-hydraulic feedback models on pin or even on sub-pin level (spatial scale). The continuous energy Monte Carlo method is well suited for solving such core environments with the detailed representation of the complicated three-dimensional problem. The major advantages of the Monte Carlo method over

the deterministic methods are, the continuous energy treatment, and the exact three-dimensional geometry modeling. Because of its statistical nature, Monte Carlo core calculations usually involve a considerable computer time to attain reliable converged results for both integral parameters and local distributions. This is especially true when coupling it with a thermal-hydraulics code to obtain three-dimensional (3D) power and thermal-hydraulic solutions for a reactor core. Coupled Monte-Carlo calculations can serve as reference solutions for verifying high-fidelity coupled deterministic neutron transport methods with detailed and accurate thermal-hydraulic models. Results from feasibility studies performed at the Pennsylvania State University (PSU) on both accelerating Monte Carlo criticality calculations by using hybrid nodal diffusion Monte Carlo schemes and thermal-hydraulic feedback modeling in Monte Carlo core calculations were presented in the previous publication [1]. The development and verification of such reference high-fidelity coupled multi-physics scheme is accomplished at PSU on the basis of MCNP5, NEM and COBRA-TF (CTF) computer codes. The computation process is greatly accelerated by calculating the 3D distributions of fission source and thermal-hydraulics parameters with the coupled NEM/CTF code and then using coupled MCNP5/CTF code to fine tune the results to obtain an increased accuracy. The PSU NEM code employs cross-sections generated by MCNP5 for pin-cell based nodal compositions. This paper presents the new developments of the hybrid Monte Carlo-based coupled core calculations at PSU, and the results of applying the developed coupled hybrid system NEM/MCNP5/CTF to a simplified 3D 2×2 fuel-pin array from a representative Boiling Water Reactor (BWR) assembly.

## 2. COMPUTER CODES

The research studies presented in this paper utilized and modified accordingly the following computer codes.

### 2.1. MCNP5 code

The MCNP code (Monte-Carlo N-Particle), MCNP5 [2], is an advanced version of a flexible 3D Monte Carlo code with provision to compute accurate detailed analysis of complex reactor configurations. The publicly available MCNP5 and NJOY99 [3] codes have been utilized and modified at PSU for the purposes of this research. The basic version of MCNP5 used in the presented research was MCNP5 release 1.40.

### 2.2. Nodal Expansion Method (NEM) diffusion code

The Nodal Expansion Method (NEM) diffusion code has been developed, maintained, and continuously enhanced at PSU. NEM [4] is a few-group (with up to 10 energy groups) 3D steady-state and transient nodal core model with three geometry modeling options: Cartesian, Hexagonal-Z, and Cylindrical (R-θ-Z). The code is based on transverse integration procedure and it was recently updated to utilize semi-analytical transverse-integrated flux representation and improved transverse leakage approximation. The nodal coupling relationships are expressed in a partial current formulation. The time dependence of the neutron flux is approximated by a first order fully implicit finite-difference scheme (upgraded later with exponential transformation

technique), whereas the time dependence of the neutron precursor distributions is modeled by a linear time-dependent approximation. NEM is using the Response Matrix technique for inner iterations to calculate (update) ongoing partial currents for each spatial node in the framework of each energy group solution. The coarse-mesh rebalance and asymptotic extrapolation methods are used to accelerate convergence of the outer iterative solution process.

### 2.3. Advanced Thermal-Hydraulic Sub-Channel Code COBRA-TF

The PSU version of the advanced thermal-hydraulic sub-channel code COBRA-TF [5] has been recently improved and applied for stand-alone and coupled Light Water Reactor (LWR) core calculations under the name CTF. COBRA-TF is applicable to both Pressurized Water Reactor (PWR) and BWR analyses. The code features two-fluid, three-field representation of the two-phase flow. The three-fields are continuous vapor, continuous liquid, and entrained liquid drops in the annular flow region. For each space dimension, COBRA-TF solves three momentum equations, four mass conversation equations, and two energy conservation equations. The code is able to handle both hot wall and normal flow regimes maps and it is capable of calculating reverse flow and cross flow situations. The conservation equations for each of the three fields and for heat transfer from and within the solid structure in contact with the fluid are solved using a semi-implicit, finite-difference numerical technique on an Eulerian mesh. The code features extremely flexible nodding for both the thermal-hydraulic and the heat-transfer solution. This flexibility provides a capability to model the wide variety of geometries encountered in vertical components of a nuclear power vessel.

### 2.4. NJOY99 Nuclear Data Processing System

The NJOY nuclear data processing system converts evaluated nuclear data from the ENDF format into cross-section libraries for different applications including continuous-energy Monte Carlo (MCNP5). NJOY99 is the latest release of the NJOY nuclear data processing system and this release is used for the present studies.

## 3. MONTE CARLO BASED COUPLED CALCULATIONS

The initial developments performed in advance at PSU (to facilitate the Monte Carlo based coupled calculations for steady state reactor core applications) are described in detail in our previous publication [1] and summarized in the following subsections for the reader's convenience. In addition, the new developments are presented in the next subsection.

### 3.1. Modifications Facilitating Coupled Calculations

#### 3.1.1. Thermal-hydraulics feedback modeling in Monte Carlo calculations

The thermal-hydraulic feedback information (densities and temperatures) is transferred through the interfacing code (described in detail in the next sections).

The temperature effects on continuous cross-section data include Doppler broadening of resolved resonances; Doppler broadening of unresolved resonances; changes in S(α,β) thermal scattering kernel; and elastic scattering, which is Doppler broadening over the entire energy range. In order to consider the above-mentioned effects, continuous temperature dependent cross-section libraries were developed for MCNP5 at PSU along with automated cross-section temperature interpolation capability (which is also described in detail in the next sections).

### 3.1.2. Convergence diagnostics of Monte Carlo calculations

Two methodologies, the Shannon Entropy [6] and the Relative Entropy [7], have been proposed to measure the convergence of the fission source distribution. Since in the MCNP5 release 1.14 code version (this version was used in the first feasibility studies performed at PSU) a diagnostic feature to track the fission source convergence in MCNP5 criticality calculations was not available, such feature was developed at PSU based on the Relative Entropy calculations. This feature was applied to the Monte Carlo method as posterior graphical stationary diagnostic tool to measure the convergence of the fission source distribution.

### 3.1.3. Hybrid nodal diffusion/Monte Carlo calculations

A new methodology to speed up the MCNP5 criticality calculations was devised as part of the development of the Monte Carlo based coupled calculations. The algorithm utilizes a hybrid nodal diffusion/Monte Carlo calculation scheme. The primary concept of this methodology is to pre-generate an initial source distribution of the Monte Carlo method by a nodal diffusion method. In the coupled nodal diffusion/Monte Carlo calculation scheme, the fast nodal diffusion calculation is used to provide the fission source distribution as an initial source distribution for the Monte Carlo calculation. The fission source distribution simulated in the nodal diffusion calculation is practically closer to the true source distribution of the Monte Carlo calculation than the typical initial guess. Therefore, the coupled methodology avoids utilizing large number of inactive cycles in the Monte Carlo calculation. The implementation of this hybrid methodology employs NEM as a nodal diffusion solver.

### 3.1.4 Generation of few-group diffusion cross-sections by MCNP5

It is important that the cross-section data used in Monte Carlo and nodal diffusion calculations are consistent. The generation of the diffusion cross-section library by the Monte Carlo method would produce the most consistent cross-section data to be used in the hybrid nodal diffusion/Monte Carlo calculation. MCNP5 has capabilities to calculate several types of reaction rates, e.g., fission, absorption or total reaction rate. Consequently, the fission, production and cross-sections can directly be obtained using the spatial and energy collapsation scheme. The calculation of Discontinuity Factors (DFs) for NEM requires a node-surface flux tally. Since the few-group cross-sections and DFs are calculated by MCNP5 for single pin-cell models with reflective boundary conditions (with exception of non-fuel cells when color-set calculations are employed) there was not any difficulty to obtain accurate surface flux values from MCNP5. The standard MCNP5 cannot directly calculate the group-to-group scattering reaction rate (it can only calculate group scattering reaction rate) and the transport reaction rate (which is used to determine the diffusion coefficient). Subsequently, a methodology to generate the group-to-group

scattering cross-section and the diffusion coefficient by the Monte Carlo method was developed and implemented. The MCNP5 was modified at PSU by adding routines calculating few-group node-wise homogenized diffusion parameters. This innovative feature reinforces the consistency of the hybrid nodal diffusion/Monte Carlo calculation process and further accelerates the MCNP5 calculation.

## 3.2. New Developments Improving Coupled Calculations

The new developments to improve the Monte Carlo based coupled calculations for steady state reactor core applications are described in the subsections below.

### 3.2.1. Automated methodology for generation of temperature dependent MCNP5 continuous energy cross-section libraries from different evaluated nuclear data files

The NJOY code is a nuclear data processing system which produces point-wise and group-wise cross sections from the Evaluated Nuclear Data File (ENDF). It has the capability to generate continuous energy MCNP5 cross sections. Since the generation process for cross-section libraries is tedious and involves a lot of data manipulation, an automated tool has been developed to process all data and manage processing requirements during the generation process. This automated methodology has been developed as part of the hybrid Monte Carlo-based coupled core calculations at PSU.

The duration of the cross-section generation procedure has been significantly reduced. This automated tool performs the most time consuming tasks of the cross-sections generation process. The final output of automated tool is a cross-section library together with the directory of cross-section files readily useable by MCNP5.

This tool is used together with the automated cross-section temperature interpolation capability for intermediate points. The most accurate technique to update the cross-section libraries, taking into account the temperature feedback, would be to update the libraries with NJOY, for each nuclide, in their corresponding region, at the temperature of that region. This technique would not be practical because of the vast computational time that it would take. Investigations carried out at PSU have exemplified that the most appropriate and practical technique, with high accurate results, would be to use a pre-generated cross-section library with temperature increments of 50K, and then, an interpolation method would be used to approximate the cross sections at the provided by COBRA-TF temperatures [1]. The square root of temperature, $\sqrt{T}$, interpolation method is used in the studies presented in this paper. This $\sqrt{T}$ interpolation method has demonstrated the correct trend and excellent agreement with the actual cross-sections [1, 8]. Considerable amount of computational time is saved by not having to re-generate the cross section libraries.

Currently, studies are being performed at PSU to include the possibility of generating/interpolating temperature-dependent thermal scattering cross-section libraries as well.

### 3.2.2. Internal coupling

The coupling between the MCNP5, NEM and COBRA-TF codes requires a great deal of data exchange; hence, manual coupling between the codes is very tedious and error-prone. An interfacing code is therefore foreseen and developed as a comprehensive tool for the coupling work.

In general, codes can be coupled in an external manner (creating an automated tool to read/write/manage the feedback files) or in an internal (integrated) manner (creating an automated tool to control the flow of actions directly in the source files of the coupled codes). The external coupling is, typically, easier to perform, but slower because of the time it takes to transmit the feedback data between the codes. On the other hand, the internal coupling transfers the feedback data in a direct way avoiding any extra transfer action that will increase the time of the calculation. Internal coupling represents a faster and more efficient way of passing the feedback information.

The NEM/MCNP5/CTF code system includes an interfacing code that provides thermal-hydraulic feedback with hybrid nodal diffusion capabilities to the LANL Monte Carlo code, MCNP5 by coupling in an internal manner with the thermal-hydraulic sub-channel code COBRA-TF and the NEM diffusion code. The NEM/MCNP5/CTF code is developed such that it automates the sequences needed between the MCNP5, NEM and COBRA-TF codes. The NEM/MCNP5/CTF code is entirely written in standard Fortran 90/95 language, like the latest versions of MCNP5 and COBRA-TF.

## 4. COUPLING SCHEME

### 4.1. Multi-level NEM/MCNP5/CTF Scheme

The coupling scheme presented in Fig. 1 was devised at PSU [1], and has similarities and differences with coupling schemes reported elsewhere [9, 10].

The computation process can be greatly accelerated by calculating the 3D distributions of fission source and the thermal-hydraulics parameters with the coupled NEM/CTF code and then using coupled MCNP5/CTF calculations to fine tune the results to obtain an increased accuracy of the transferred parameters. The PSU NEM code employs cross-sections generated by MCNP5 for pin-cell based nodal compositions. As mentioned above, the sub-channel code COBRA-TF is used for thermal-hydraulics analyses in individual flow sub-channels of a fuel assembly. Initially, the nodal diffusion code NEM is utilized to calculate the complete solution of the core configuration and the fission source distribution in order to accelerate the MCNP5 calculations. NEM is coupled with COBRA-TF in serial integration manner to provide this solution (in fact NEM is incorporated as a subroutine in the COBRA-TF calculation flow). Once this is achieved, the results are transferred to the MCNP5 code for further fine-tuning analysis. The interface code transforms and incorporates the NEM/CTF output into the MCNP5 modeling and vise versa. The multilevel coupling scheme is shown in Fig.1.

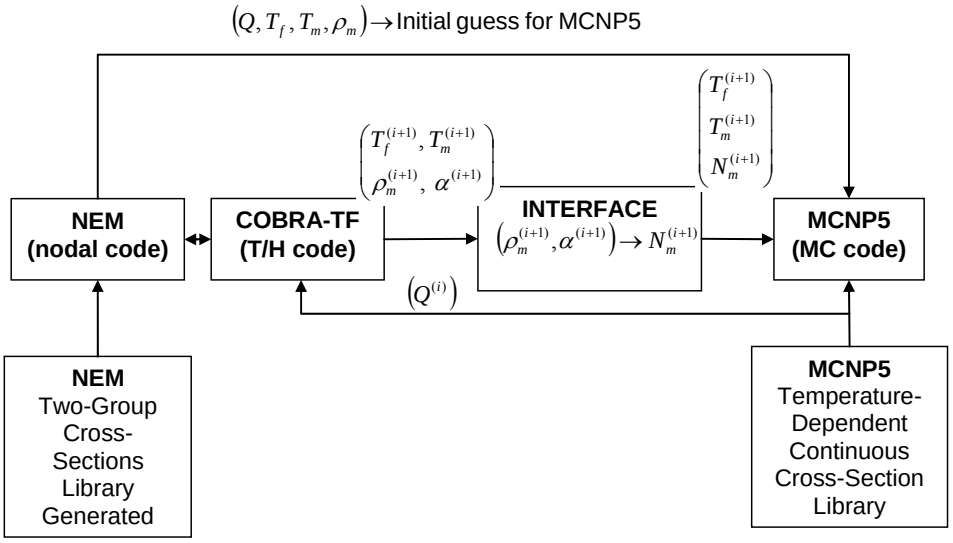


**Figure 1. Multi-level NEM/MCNP5/CTF coupling scheme**

The coupled convergence criteria of $\varepsilon=10^{-4}$ are set up on feedback parameters and power distribution between coupled iterations as depicted in the convergence scheme of Fig. 2.

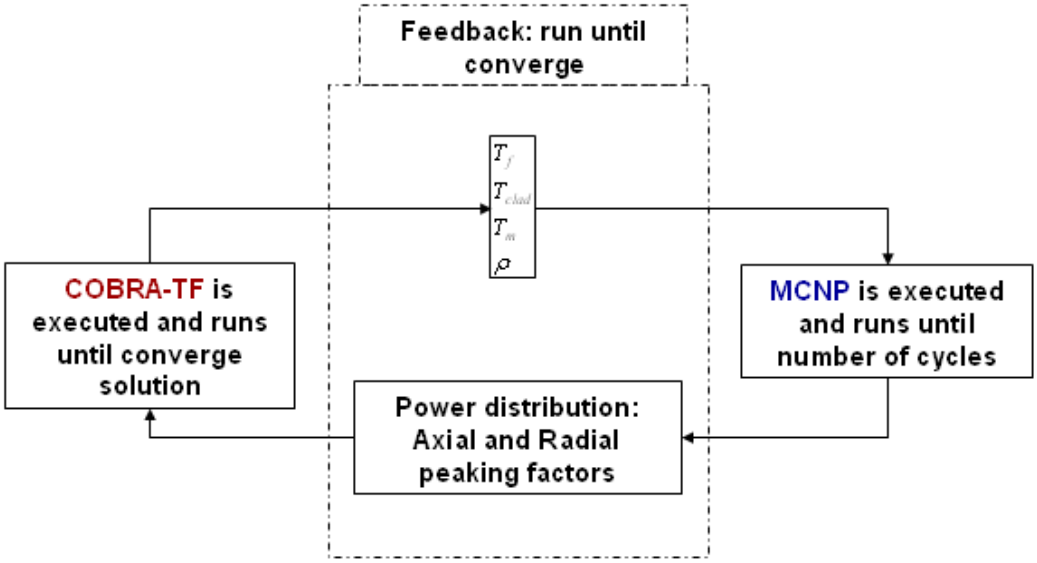


**Figure 2. Convergence scheme for the coupled calculation**

In the case of a MCNP5/CTF calculation, COBRA-TF is first run in stand-alone mode (initialized) using an initial power distribution assumption and after achieving equilibrium steady state conditions the coupled iterations with MCNP5 are started. Then the MCNP5 code is run until the number of stated cycles is reached and the power feedback is passed to COBRA-TF. Finally, the couple iterations stop when the convergence criteria are satisfied after evaluating the feedback parameters and power distribution between coupled iterations.

Another case is the hybrid multi-level coupled code system NEM/MCNP5/CTF simulations. In this case again first COBRA-TF is initialized with an initial power distribution assumption, then coupled NEM/CTF calculation is converged to provide initial distributions of fission source and feedback parameters to MCNP5, and finally coupled MCNP5/CTF iterations are carried out to obtain final high-accuracy converged results.

The NEM/MCNP5/CTF code system presents the possibility of executing the Monte-Carlo criticality calculation either in sequential or parallel mode. The execution mode depends on the number of processors available. The OpenMP model available in MCNP5 (release 1.40), shared-memory parallelism (threads) model, is used in the parallel execution mode.

### 4.2. Spatial Mesh Overlays

In the latest BWR type study, the coupled hybrid Monte Carlo based code system is applied to a simplified 3-D 2×2 fuel-pin array extracted from a representative BWR assembly [11]. The extracted simplified 2×2 pin array model preserves the actual heated length and uses only one material composition for each pin cell over the axial length. Each pin cell contains fresh fuel with different enrichment. In radial plane reflective boundary conditions are used, while in axial direction at the end of bottom and top reflectors vacuum boundary conditions are utilized.

The MCNP5, NEM and COBRA-TF models are consistent. The axial direction over the heated length is divided into 50 equidistant nodes. The radial plane consists of four sub-channels in COBRA-TF, one node per pin cell in NEM and a lattice array in MCNP5. The radial and axial planes of the models are shown in Fig. 3 below. The detailed material and geometrical specification of the model is given in the previous publication [1].

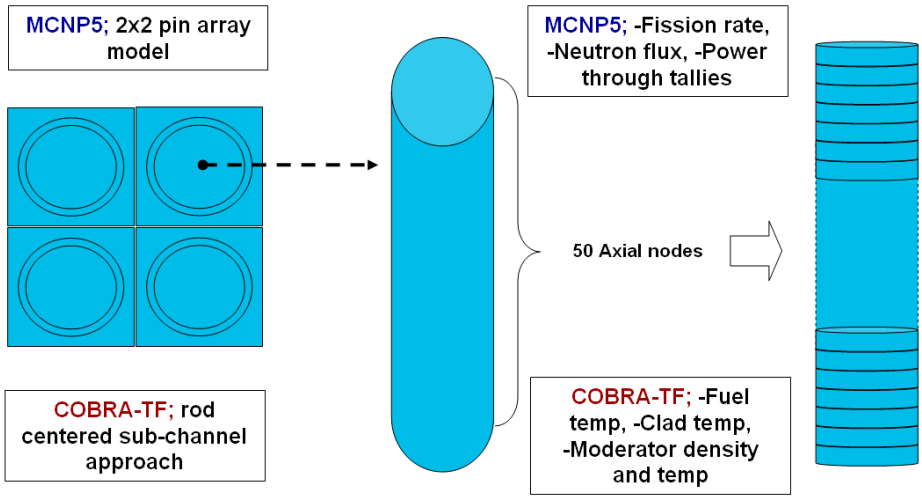


**Figure 3. Spatial mesh overlays**

The NEM model is one node per pin cell in radial plane, the same 50 nodes as MCNP5 in axial direction over the heated length, plus 5 nodes of 10 cm each for bottom reflector and another 5

nodes of 10 cm each for the top reflector. The PSU NEM code employs cross sections generated by MCNP5 for pin-cell based nodal compositions in order to be consistent with the MCNP5 model.

The COBRA-TF model is consistent with the MCNP5 and NEM models. It consists of four sub-channels defined by the flow area of the fluid surrounding each fuel pin (rod centered sub-channel approach), as illustrated in the left side of Fig. 3. The pins are modeled as $UO_2$ fuel rods with cladding material of Zircaloy-2. Only the heated length of the fuel pins is considered and it is divided into 50 equidistant nodes. The axial reflector nodes in the MCNP5 and NEM models do not receive feedback from the COBRA-TF model. Their conditions (and corresponding cross-sections) are fixed at the bottom and top reflector thermal-hydraulic conditions respectively and are not updated during the coupled iterations.

The feedback information, thermal-hydraulic data (moderator density and temperatures of the fuel, clad and moderator) as well as power data (axial and radial power factors), is transferred per node as depicted on the right side of Fig. 3. The feedback transferred per node permits a detailed representation of the behavior of the modeled parameters.

## 5. RESULTS

The results demonstrating the implementation and validation of the code modifications facilitating coupled calculations using the PSU Breazeale Research Reactor (PSBR), which is a TRIGA mark III research reactor manufactured by General Atomics, were presented in the previous publication [1]. The results of the current publication focus on the demonstration of the Monte Carlo based coupled calculations utilizing the described previously multi-level scheme in the simplified BWR 3-D 2×2 fuel-pin array mentioned above.

### 5.1. Convergence Diagnostic

When MCNP5 is coupled, in this particular case with the COBRA-TF code, it is crucial that the solution provided by the MCNP5 code has converged. It is needed that both, the effective multiplication factor ($k_{eff}$) and the fission source distribution, converge in order to start tallying when running criticality calculations. The $k_{eff}$ and the fission source distribution converge differently; $k_{eff}$ converges more rapidly than the source distribution. The $k_{eff}$ alone cannot be used to asses the solution convergence. In the most recent studies performed at PSU, the MCNP5 release 1.40 has been used. This release incorporates a vital tool to assess convergence[1]. The convergence of both, the $k_{eff}$ and fission source distribution (using the Shannon Entropy), for the 2×2 fuel-pin array is illustrated in Fig. 4.

In addition, the analysis of the source convergence helps to asses the initial fission source distribution from the nodal diffusion calculation. The Shannon Entropy convergence is used to observe the improvement in the reduction of cycles by using the fission source distribution simulated in the nodal diffusion calculation since it is practically closer to the true source

[1] This new tool is based on the Shannon Entropy of the fission source distribution [6].

distribution of the Monte Carlo calculation than the typical initial guess. Moreover, the convergence analysis is used to evaluate any bias in the source.

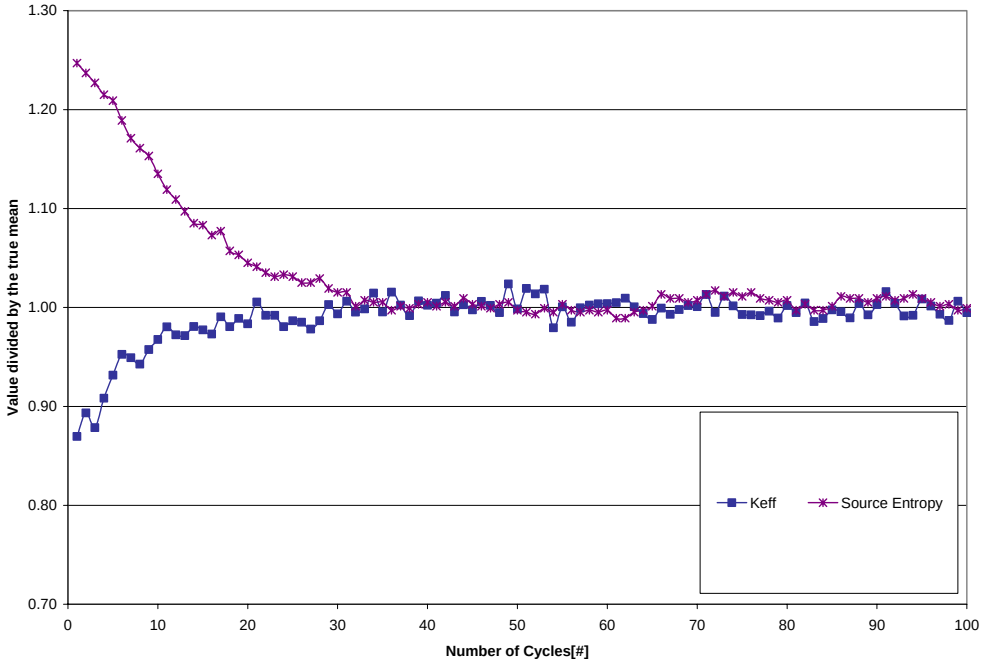


**Figure 4. $k_{eff}$ and Shannon Entropy convergence for 3-D BWR 2×2 fuel-pin array**

Through this graphical tool, the solution convergence is evaluated during each coupled iteration. The number of histories used to make the source distribution to achieve the 'exact' equilibrium state is optimized in order to reduce further the computational time. In other words, the accuracy of the technique is optimized. A larger number of histories in the criticality calculation step might improve the accuracy of the study. Alternatively, if the criticality calculation step had already reached the converged equilibrium state, at a smaller number of histories than that used previously, the computational time could be reduced further to attain the same accuracy.

### 5.2. BWR 3-D 2×2 Pin Array Results

The simplified BWR 3-D 2×2 fuel-pin array is modeled with the coupled hybrid Monte Carlo based code system. The obtained results confirmed the results from the preliminary investigations reported in [1]. The NEM/MCNP5/CTF predictions are essentially the same as the MCNP5/CTF results especially if tighter coupled convergence criteria are used (in studies reported in [1] the convergence criteria were set to 1%). In the studies reported in this paper a tighter coupled convergence criteria were set up to satisfy the requirement that the maximum relative deviations in node-wise distributions of feedback parameters and power between two successive coupled iterations simultaneously to be less than 0.01%. The results also demonstrate that the NEM-based initial fission source distribution does not bias final MCNP5 results. The proposed NEM/MCNP5/CTF multi-level coupling scheme provides about 21% savings in computation time as compared to direct MCNP5/CTF for the BWR 3D 2×2 pin array problem. It is expected by applying the proposed scheme to larger 3D problems as color set of assemblies

and core sectors the savings to rise to about 40-50 %. These savings combined with utilization of the MCNP5 parallel execution mode will make the coupled Monte Carlo calculations feasible to produce reference results at operating conditions.

In order to demonstrate the functionality of MCNP5/CTF coping scheme the obtained results for two cases of the BWR 3D 2×2 pin array problem are presented and discussed. In Case 1 the inlet flow rate per a subchannel is 0.16303 kg/s, while in Case 2 it is increased to 0.65212 kg/s. Tables I and II show $k_{eff}$ values and radial power distributions for Cases 1 and 2 respectively while Fig. 5 through 7 compare pin-array average axial power, fuel temperature and moderator density distributions for the two cases.

**Table I. Results for Case 1**

| **$k_{eff}$** | 1.29626 ± 0.00037 | |
|---|---|---|
| **2D Normalized Power** | 1.0664 | 1.0195 |
| | 0.9402 | 0.9739 |

**Table II. Results for Case 2**

| **$k_{eff}$** | 1.33868 ± 0.00038 | |
|---|---|---|
| **2D Normalized Power** | 1.0863 | 1.0135 |
| | 0.9329 | 0.9673 |

An initial cosine power distribution is assumed for the COBRA-TF calculation. Then, after several coupled iterations (these particular cases converged after three coupled iterations), the converged solutions are compared in the above-mentioned tables and figures. When increasing the flow rate (from Case 1 to Case 2) $k_{eff}$ is increasing also reflecting the physical fact that increased flow rate leads to higher moderator density and higher moderation resulting in higher $k_{eff}$. The axial power distributions depicted in Fig. 5 reflect the thermal-hydraulic feedback effects and are consistent with the fact that in the BWR 3D 2×2 problem the fresh fuel with the same enrichment is uniformly distributed axially (high power at the bottom - where there is a higher moderator density resulting in a higher moderation; and lower power at the top - where there is a lower moderator density resulting in a lower moderation). When increasing the flow rate (from Case 1 to Case 2) the axial moderator density distribution is shifting towards higher moderator density values (see Fig. 7), which is shifting accordingly the axial power distribution (see Fig. 5) and the fuel temperature distribution (see Fig. 6) through the moderator and Doppler feedback mechanisms.

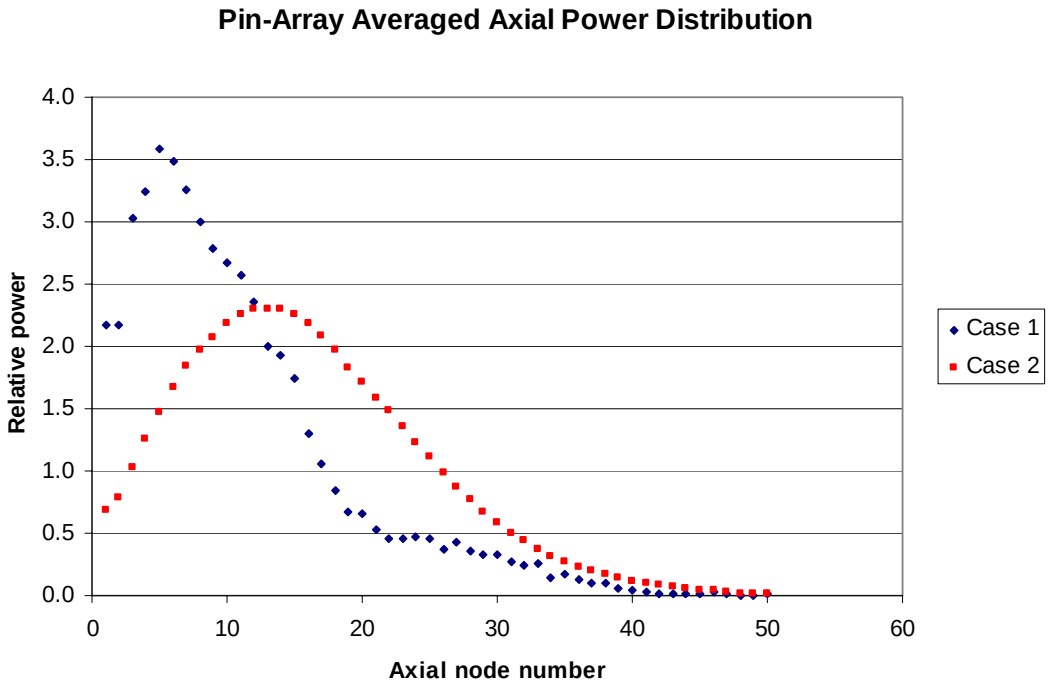


**Figure 5. Axial Power Distributions for Cases 1 and 2**

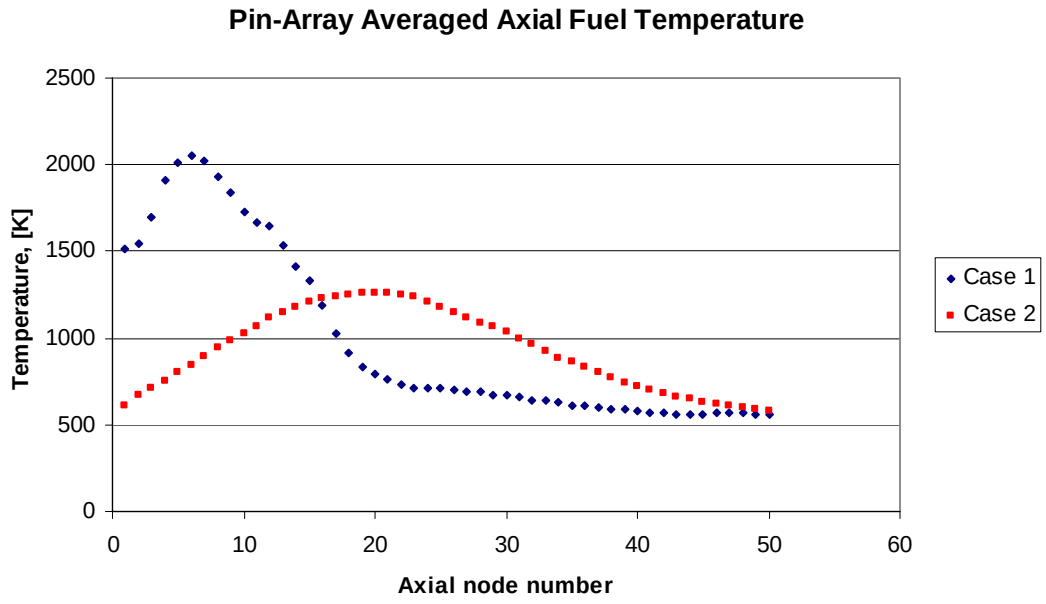


**Figure 6. Axial Fuel Temperature Distributions for Case 1 and 2**

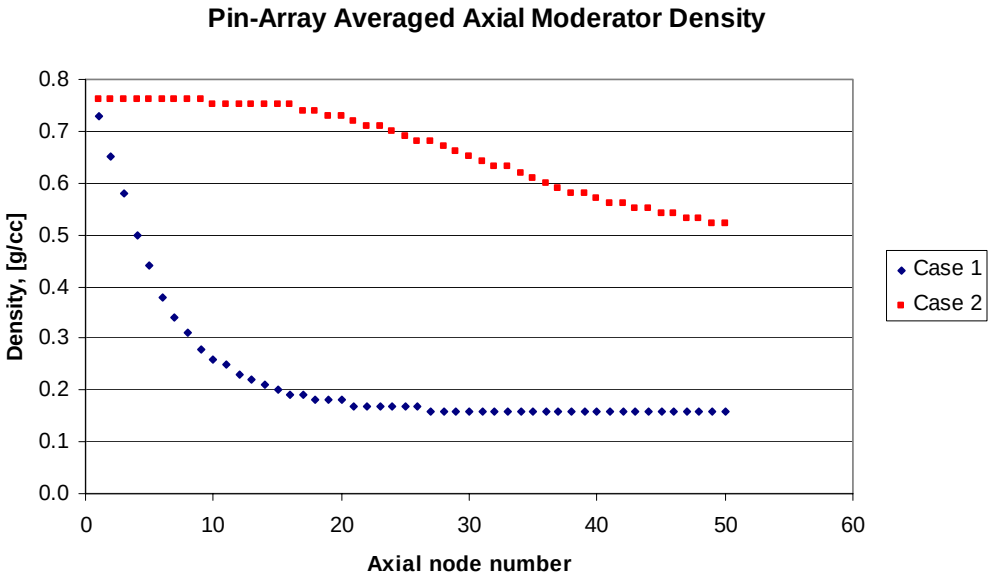


**Figure 7. Axial Moderator Density Distributions for Cases 1 and 2**

## 6. CONCLUSIONS

The investigations carried out at PSU showed that performing Monte Carlo based coupled core steady state calculations are feasible. The further developments of the coupled hybrid Monte Carlo based code system have included the novel interpolation scheme for the cross sections, an automated tool for generation of temperature dependent continuous MCNP5 cross-section libraries, and a master code that performs an internal coupling to reduce the time for reading large amount of data calculations. The square root of temperature interpolation scheme is being used as part of the master code to avoid the need of having to re-generate the cross section libraries for each isotope, of each region, at the specified feedback temperature, saving a considerable amount of time. Studies are being performed to include the possibility of generating thermal scattering cross-sections in the automated tool for generation of temperature dependent continuous cross-section libraries. The accuracy of the methodology is optimized by assuring that the correct number of cycles in the MCNP5 calculation is used. The convergence of both, the effective multiplication factor ($k_{eff}$) and the fission source distribution, of this solution provided by the MCNP5 code is checked to ensure the quality of the transferred information. In order to speed-up Monte-Carlo criticality calculations, which take the major part of the CPU time of the simulations, a combined strategy is being investigated including parallel computing, accelerating techniques, and hybrid nodal diffusion/Monte Carlo approach.

The further developments of the coupled hybrid Monte Carlo based code system are focused on 3D applications such as single PWR and BWR assemblies, and assembly color-sets to produce reference results at operating conditions for benchmarking core analysis deterministic tools.

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