MULTI-LEVEL COUPLED METHODOLOGY FOR LOCAL EVALUATION OF SAFETY PARAMETERS IN PWRs

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ABSTRACT

A multi-level methodology for local evaluation of safety parameters utilizing a sub-channel thermal-hydraulic analysis, based on the coupled TRAC-PF1/NEM/COBRA-TF code system is presented. The pin power reconstruction techniques and the coupling schemes of TRAC-PF1/NEM and COBRA-TF in parallel computing environment are discussed. A dynamic intranodal pin power algorithm in two versions has been implemented and tested on the Main Steam Line Break TMI-1 benchmark model. The consistency of the obtained realistic physical results is demonstrated. Further, the coupling options, implemented in the TRAC-PF1/NEM/COBRA-TF code system are presented. Finally, the ongoing work and planned future activities in this area are outlined.

1. INTRODUCTION

Thermal margins, which include the departure from nucleate boiling ratio (DNBR) and the maximum fuel temperature, are the critical parameters that determine the safety of a pressurized water reactor (PWR). The excess of the values of these parameters in a hot channel results in failure of a fuel rod and ultimately leads to an accident. To determine the thermal margins of a reactor core at given conditions, a hot channel analysis is usually employed that implies the location and calculation of fluid, thermal, and neutronics conditions in a most limiting fuel assembly (FA)/node (part of FA). Prior the determining the thermal margins of a reactor core, certain boundary conditions are necessary to be provided for the hot channel model. These boundary conditions include the detailed (pin-by-pin) power distribution in the hottest fuel assembly, as well as the thermal-hydraulics (T-H) data for the channel. To determine the detailed intranodal power distribution, a pin power reconstruction procedure that utilizes the calculated coarse mesh (nodal) neutronics data is used. In addition, enhancements have been implemented in the pin power reconstruction model for MOX and other highly heterogeneous PWR core environments applications. The T-H boundary conditions are calculated with the system code TRAC-PF1. In this paper, the developed methodology for a refined hot channel analysis using the coupled TRAC-PF1/NEM/COBRA-TF system is discussed.

TRAC-PF1/NEM is a best-estimate coupled core/system code that allows evaluating the typical design basis accidents occurred in PWRs, such as a loss of coolant accident (LOCA) and reactivity insertion accidents (RIA).[1] The three-dimensional (3D) neutronic module employed in the code is based on the semi-analytical NEM that solves for the coarse-mesh (nodal) results, i.e., volume- and surface averaged nodal neutron fluxes and surface currents.[2]

COBRA-TF employs a two-fluid, three-field representation of the two-phase flow that occurs in PWRs during some transients and in BWRs during normal operation and transients.[3] Each of the fields is treated in three dimensions and is compressible. The three fields are continuous vapor, continuous liquid and entrained liquid drops. The code features extremely flexible noding for both the hydrodynamic mesh and the heat transfer solution. Because of the flexibility, one is able to model nearly all the complex geometry’s encountered in the vertical components of a nuclear reactor vessel. Unlike earlier versions of COBRA (COBRA-IIIC, COBRA-IV), COBRA-TF has a full boiling regime to calculate post critical heat flux (CHF) transfer as well as the heat momentum transfer between the phases. 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. For PWR hot-channel analysis applications channel may represent:

a) Several regions lumped together inside a fuel bundle (assembly) – two approaches are possible, one exploring the detailed pin-based sub-channel layout, and the other using a coarser few-zones sub-channel layout.
b) A lumped region of the core.

2. PIN POWER RECONSTRUCTION METHODOLOGY

Two user dependent options are introduced into the calculational scheme of intranodal pin powers. As the first step, the conventional interpolation technique[4] is used for reconstruction of local power profiles in the nodes of interest. These nodes axially construct the assembly associated with the hot channel. The peak power node is determined based on the maximum nodal relative power value, calculated with the nodal method corewise. For each of these nodes the two step procedure is employed to determine the corner values of fluxes and currents for this node. Then the forth-order polynomial interpolation is performed to determine the flux/power variation within the node. This method is widely used,[3] but for the first time was implemented in TRAC-PF1/NEM. Several disadvantages are known for this scheme, the most severe is being inaccurate to predict the correct intranodal flux/power profiles in highly heterogeneous reactor media (such as MOX-fueled cores), in the vicinity of strong neutron flux changes (such as near the reflector), and during some transients and accidents. However, at this point, the conventional UO_2-fuelled PWR model is chosen for verification of the implemented pin power reconstruction algorithm. The predicted steady-state and transient results are being compared using the two implemented options.

In order to overcome the aforementioned difficulties, several analytical or semi-analytical power reconstruction methods have been proposed and implemented by various authors.[5-]

[7] All these methods solve the diffusion equation within the node of interest. These methods are regarded as superior ones as compared with the interpolational methods, which utilize power series expansion[5] of the flux within the node.

Such option is being developed for TRAC-PF1/NEM. The pin power reconstruction module introduced into TRAC-PF1/NEM is characterized by more rigorous interpolation method of calculation of corner flux values using the modification of the method of successive smoothing.[5] Analytical flux reconstruction method, the linear combination of analytical solutions of two-dimensional (2D) neutron diffusion equations in the most limiting node, is used to determine the peak power pin within the node.[8] The boundary conditions are obtained from the global nodal results and include volume average flux, the surface fluxes, and currents. The lattice code is employed to determine the 2-group pin-power form-functions. The spectral re-homogenization effects are included into 2D-assembly cross-section profile.

The method of successive smoothing (MSS) is implemented to reconstruct the flux corner values in the node of interest. The solution of a diffusion equation in a corner point of the node is singular, therefore the corner value of flux has to be obtained in a different way such as based on smoothing of the nodal values of volume averaged flux, surface fluxes and currents. The three-step MSS includes linear extrapolation, smoothing, and continuos smoothing procedures to determine the flux corner values. Having the corner values of flux and average surface fluxes obtained, the minimum set of boundary conditions is established and the intranodal fluxes can be reconstructed with the analytical flux reconstruction method. Weak-element approximation is used further for matching discrete points and/or average fluxes on the boundary of a fuel assembly. Two-dimensional inhomogeneous diffusion problem is solved for each node based on the Helmholtz equation of type:

$$\Delta\eta_g + k_g^2\eta_g = 0$$

where: g – number of energy groups

The solution within the fuel assembly of interest is constructed via a linear combination of the elementary functions obtained. The resulting analytical approximation is a superior to polynomial interpolation, particularly for the cases with steep flux gradients at the assembly interfaces, such as MOX-fueled cores or during transients. However, for the MSLB benchmark, the results do not differ significantly from the previously implemented interpolational procedure.

3. COUPLING OF TRAC-PF1/NEM WITH COBRA-TF

The coupling of TRAC-PF1/NEM with the sub-channel code COBRA-TF is provided in parallel virtual machine (PVM) environment. TRAC-PF1/NEM is modified to allow passing the necessary data, which are thermal-hydraulic boundary conditions and axial power distributions for the hottest channel, to COBRA-TF. COBRA-TF calculations are performed in parallel to the main TRAC-PF1/NEM calculation flow and activated when the evaluation of thermal margins is necessary. Three different ways of coupling of the two codes are developed as alternative options. For a fixed hot channel location COBRA-TF core calculations are performed and subsequently the COBRA-TF core T/H data optionally can be used as feedback parameters for the cross-section update. The location of hot channel can change during the transient calculation. A special algorithm to locate and recalculate the hottest channel is developed. The hot channel is identified through the evaluation of thermal data and power peaking factor. More flexible approach is utilized when one assumes the possibility of multi-assembly location of a hypothetical hot channel in a PWR. The linear heat rate of the channel (kW/ft) and axial variation of relative powers are determined within TRAC-PF1/NEM and then transferred as input to COBRA-TF. The later then performs the calculations and ultimately provides the thermal margins for the chosen assembly.

The two types of geometrical refinement, one based on detailed pin-by-pin sub-channels, and the other on few-zone sub-channels can be applied following the approach presented in Ref. 9. The geometry should be selected next, either pin-based layout using pin-by-pin axial reconstruction or sub-channel reduction with the construction of the hottest pin axial power distribution. The procedure of the integrated local safety margin evaluation is given on Fig.1. COBRA-TF is activated then with thermal-hydraulics re-calculation of the hottest channel following an evaluation of the local thermal margins.

The TRAC-PF1/NEM/COBRA-TF coupling scheme is represented in Fig.2. Three coupling options are considered during the course of calculations. In the first option the three-dimensional core power distribution computed with NEM and the TRAC-PF1 T-H core boundary conditions are passed to COBRA-TF, which in turns performs the T-H calculations of the vessel. Feedback parameters calculated with COBRA can be used to update the TRAC neutronics data as option 2. Fine sub-channel calculations can be performed further at step two, if required. The feedback parameters can be computed by TRAC-PF1 itself as well. In this case, as may be seen from Fig.2, after performing of pin power reconstruction, the sub-channel analysis mode can be activated as option 3. The axial powers and T-H boundary conditions are passed to COBRA-TF in order to perform the dynamic evaluation of thermal margins for the most limiting channel.

4. RESULTS AND DISCUSSION

The developed methodology allows evaluating local safety parameters of PWR using the coupled TRAC-PF1/NEM/COBRA-TF code. The OECD MSLB TMI-1 benchmark model[10] is used for test calculations. The steady state results of pin power distribution in the hottest node agree well with the SIMULATE-3 results.[11] The same radial and axial location of the hottest node is predicted by both codes. The axial variation of maximum pin power in the hottest channel at the steady state condition, full power is given in Fig.3. As it can be seen the maximum power occurs at the upper half of the active core as expected. The peak power occurs in the axial node number 20, in the upper part of the core. The radial pin power profile in the hottest channel corresponding to the axial layer number 20 is given in Fig.4. As it can be seen the maximum intranodal relative power is equal approximately 1.16 and occurs near the center of the node, which represents an assembly in the radial plane.

The implemented different options of T-H analysis are being tested for transient. The same data after 8.7 seconds of MSLB transient initiation are shown in Figures 5 and 6, respectively. It is found that the hot channel is located next to the stuck rod, as it is predicted by PANBOX2.[9] As it can be seen from Fig.5, the axial relative power profile in the hottest node changes after 8.7 seconds of transient as compared to steady state; the peak power occurs at the lower part of the core in the axial layer number 5. However, the maximum intranodal relative power is higher and equals approximately 1.3 versus 1.16 in the steady state condition. The shape of the radial pin power profile also changes. The maximum intranodal relative power occurs at the node interface next to the stuck rod position, which can be seen in Fig.6. The coupled code system is being qualified using code-to-code comparisons with PANBOX2 code results and experimental data.[12]

REFERENCES

1. R.Bandini. (1990), "A Three-Dimensional Transient Neutronics Routine for the TRAC-PF1 Reactor Thermal Hydraulic Computer Code." PhD Thesis. The Pennsylvania State University.
2. Beam, T, et al (1999), "Nodal Kinetic Model Upgrade in The Penn State Coupled TRAC/NEM Codes." *Annals of Nuclear Energy*. **Vol. 26**, pp. 1205-1219.
3. Solis-Rodarte, J., et al.,"Development of Refined Hot Channel Analysis Capability Based on TRAC-BF1/NEM/COBRA-TF*," Trans. Am. Nucl. Soc.*, **80**, 244 (1999).

4. Moon, H. (1989), "A Multi-group, Multidimensional, and Nodal Diffusion and Depletion Method for the Analysis of Light Water Reactors." PhD Thesis. The Pennsylvania State University.

5. Finneman, H., Boer, R., and Muller, R. (1992), "Combination of Finite-difference and Finite-volume Techniques in Global Reactor Calculations." *Kerntechnik*. **Vol. 57**, No. 4, pp. 216-222.

6. Joo, H., et al., "Consistent Analytic Pin Power Reconstruction Method for Static and Transient Reactor Safety Analysis", Proceedings of International Conference: Mathematics and Computation, Reactor Physics and Environmental Analysis in Nuclear Applications. Madrid September 1999. pp.148-157.

7. Tomasevich, D. ibid. pp.782-791.

8. Boer, R. and Finnemann, H. (1992), "Fast Analytical Flux Reconstruction Method for Nodal Space-time Nuclear Reactor Analysis." *Annals of Nuclear Energy*. **Vol. 19**, No. 10-12, pp. 617-628.

9. Boer, R. et al (1992), "The Coupled Neutronics and Thermal Hydraulics Code System PANBOX for PWR Safety Analysis." *Kerntechnik*. **Vol. 57**, No. 1, pp. 49-54.

10. Ivanov, K. et al, (1997), "PWR MSLB Benchmark, Final Specifications," NEA/NSC/DOC(97)15, OECD Nuclear Energy Agency.

11. "SIMULATE-3 – Advanced Three-Dimensional Two-Group Reactor Analysis Code", STUDSVIK/SOA-92/01.

12. Watson, J. et al, "Verification and Validation of Coupled 3D Core Models and System Codes", Proc. of German Nuclear Society Meeting, Rossendorf, Germany, January 2000.

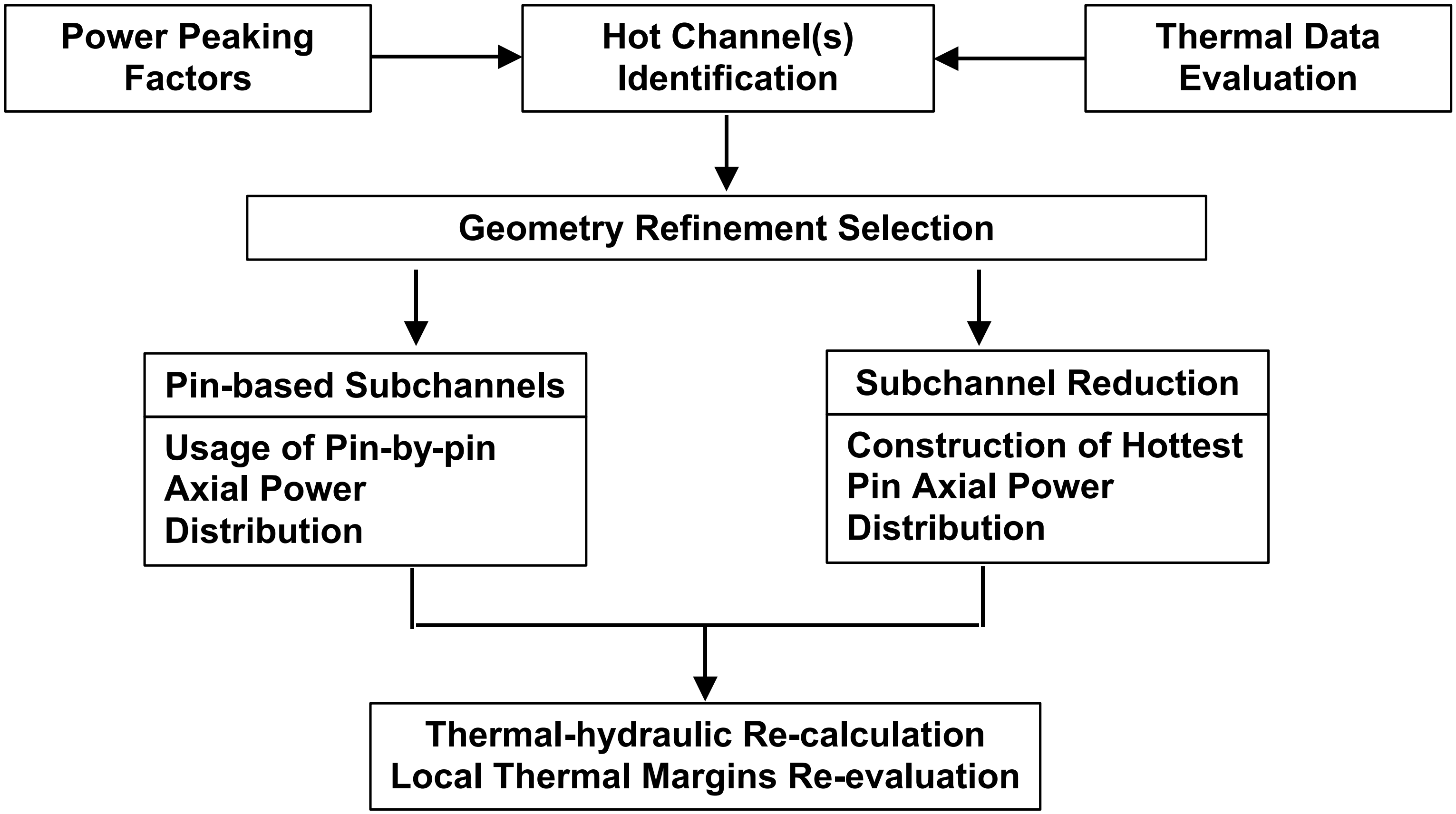

Fig 1. Procedure of the Integrated Local Safety Margin Evaluations

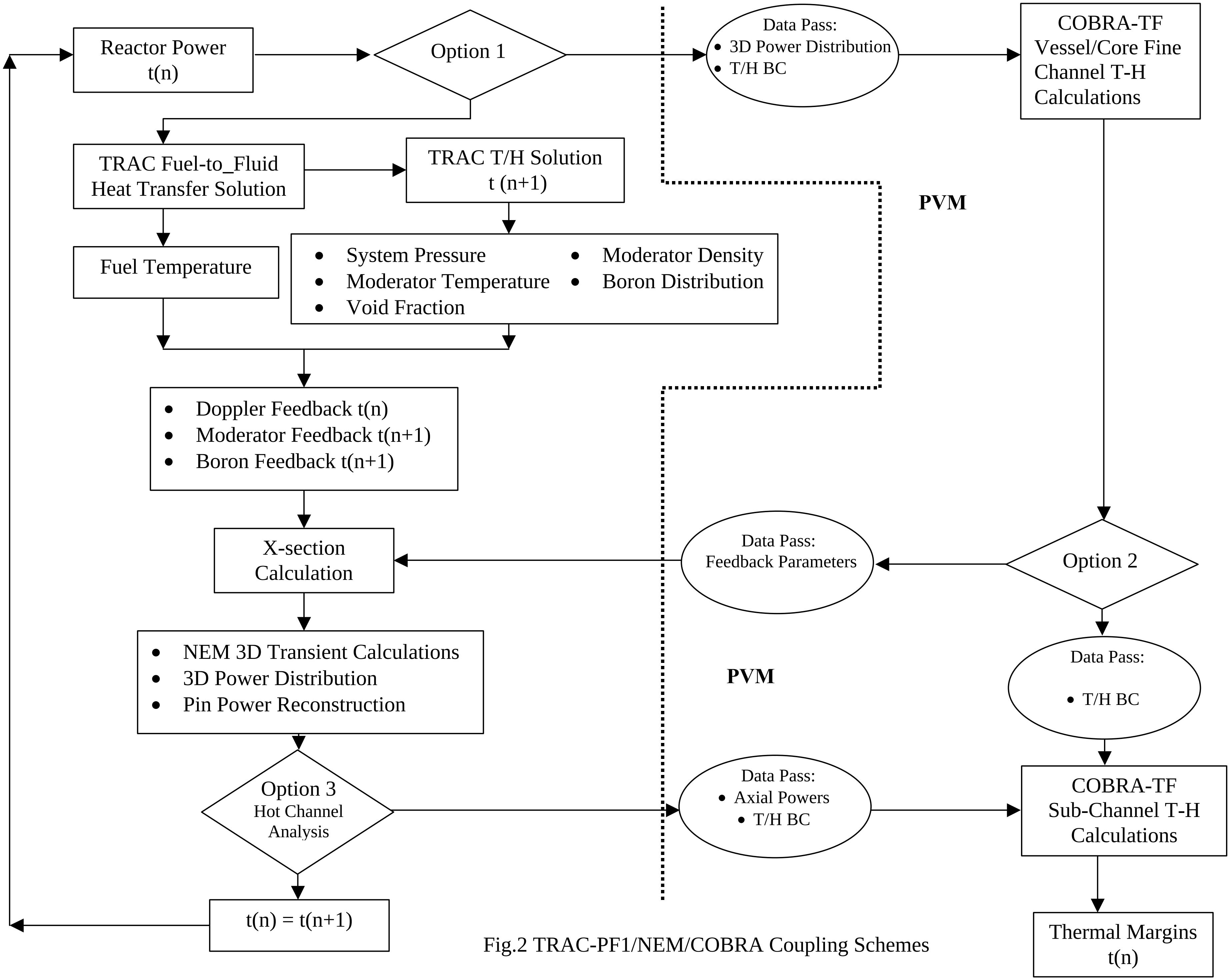

Fig.2 TRAC-PF1/NEM/COBRA Coupling Schemes

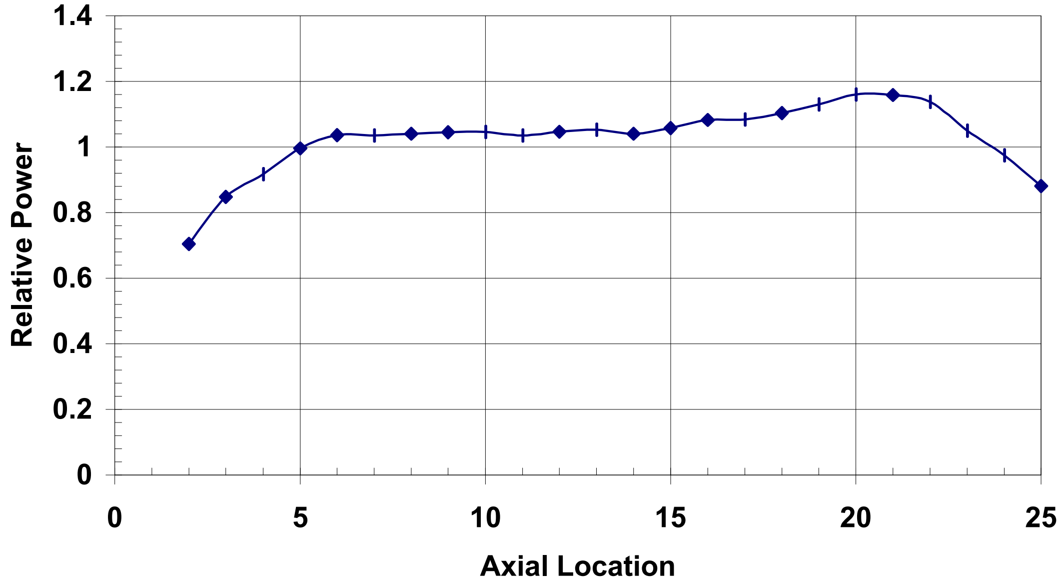

Fig 3. Axial Peak Pin Relative Power Profile in the Hottest Assembly for Steady State MSLB TMI Model

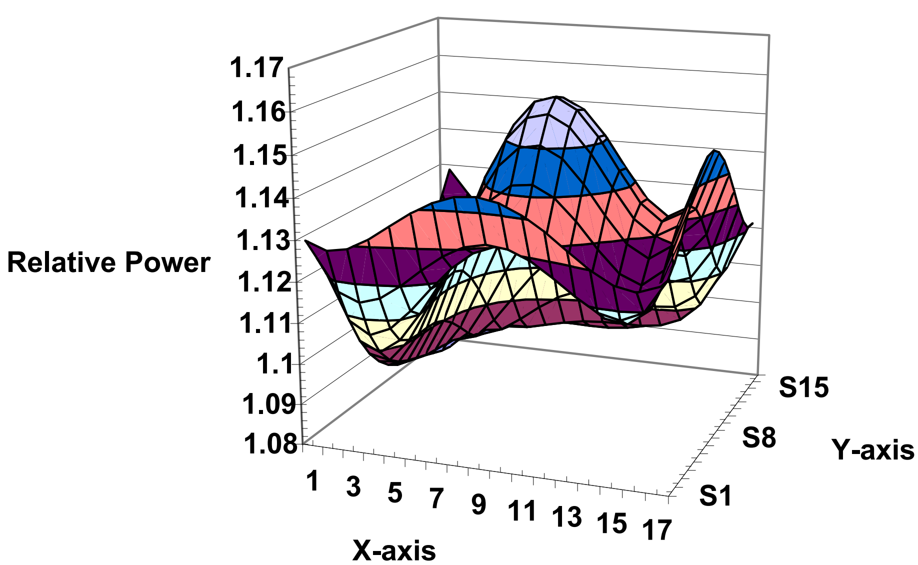

Fig 4. Radial Pin Relative Power Profile in the Hottest Node for Steady State MSLB TMI Model

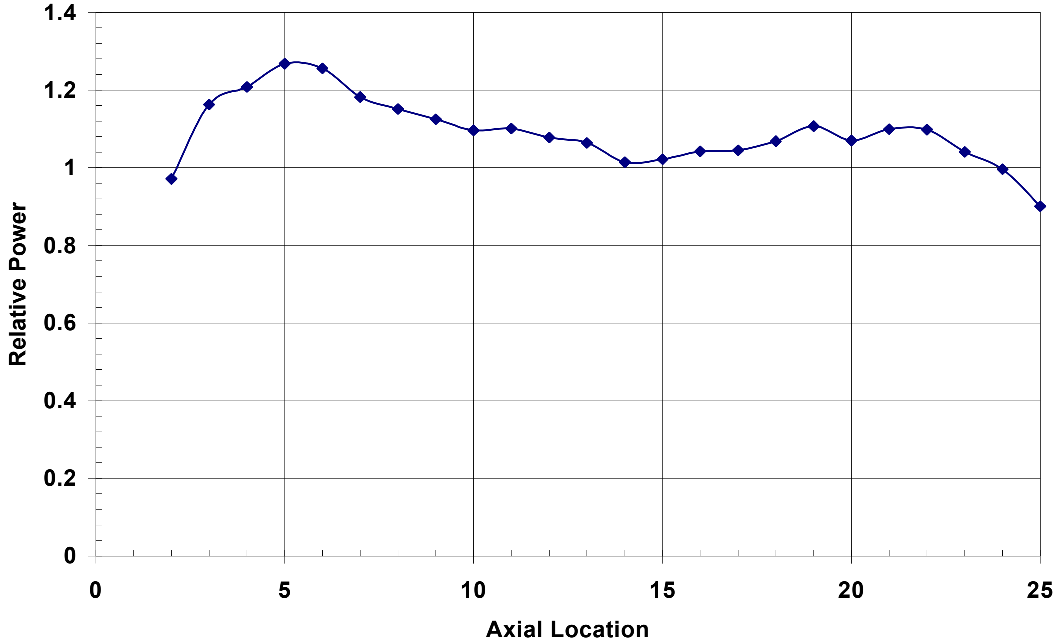

Fig 5. Axial Peak Pin Relative Power Profile in the Hottest Assembly at 8.7 sec in the Transient of MSLB TMI Model

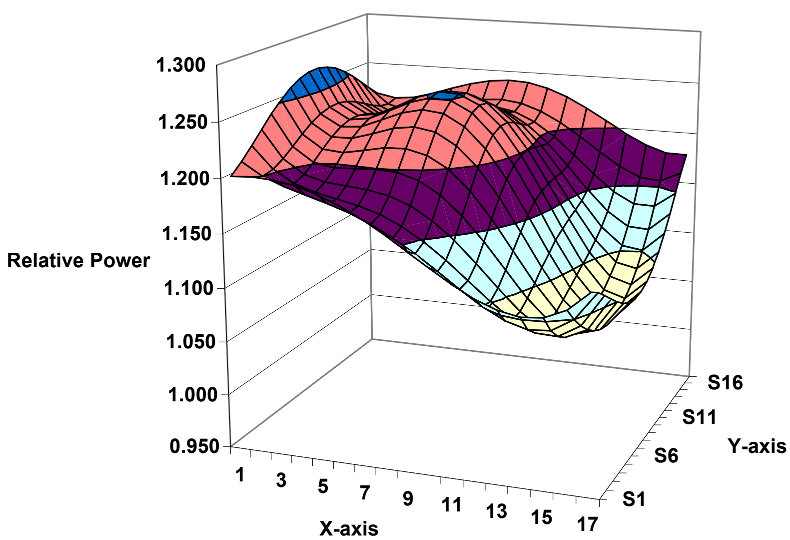

Fig 6. Radial Pin Relative Power Profile in the Hottest Node at 8.7 sec in the Transient of MSLB TMI Model