Development of PWR Integrated Safety Analysis Methodology Using Multi-Level Coupling Algorithm

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ABSTRACT

The sub-channel code COBRA-TF has been introduced for the evaluation of the thermal margins on the local pin-by-pin level in PWR. The coupling of COBRA-TF with TRAC-PF1/NEM is performed by providing the axial and radial boundary conditions and the relative pin power profiles obtained with the pin power reconstruction. Efficient algorithm for coupling of the sub-channel code COBRA-TF with TRAC-PF1/NEM in PVM environment was developed addressing the issues of time-synchronization, data exchange, spatial overlays, and coupled convergence. The local feedback modeling on the pin level was implemented. The update of the local form functions and the recalculation of the pin powers after obtaining the local feedback parameters were introduced. The coupled TRAC-PF1/NEM/COBRA-TF code system was tested on the REA and MSLB benchmark problems. In both problems the local results are closer to the correspondent critical limits. The maximum value of fuel enthalpy reached after the power spike during REA is around 48 cal/g compared to 43 cal/g obtained with the assembly average model. The minimum departure from nucleate boiling ratio does not drop below 3.5 during the MSLB transient (1.3 is the critical value). The possibility of local on-line refine safety evaluation is demonstrated and the obtained results demonstrate the importance of undertaken efforts.

1. INTRODUCTION

As the computer technology advances, the performing of coupled three-dimensional (3D) kinetics/thermal-hydraulics calculations for best-estimate safety calculations is becoming a common practice. However, the best-estimate evaluation should be performed on the local pin level to operate with more realistic safety margins. The thermal margins that include the departure from the nucleate

boiling ratio and the centerline fuel temperature are the critical parameters that determine the safety of a PWR. The excess of one of these values leads to the failure of fuel rod during the transient and ultimately will result in accident situation with the release of the fuel and fission materials in the primary coolant system. Therefore, the location of any potential fuel rod that could fail and monitoring of the key safety parameters of that rod are important to prevent any kind of fuel rod failure. Most of the currently applied safety methodologies use the average fuel bundle power multiplied by the appropriate nuclear and engineering hot channel factors (HCFs) to estimate the temperatures and DNBRs. Usually, HCFs are precomputed in static detailed pin-by-pin calculations. However, the dynamically computed *local pin* values of the fuel temperature, cladding, and DNBR are the values that are required to make more accurate safety evaluation of a particular fuel assembly. To accomplish this, the scale of modeling for the best-estimate code should be downsized to the fuel pin level and the sub-channel safety analysis should be performed, i.e., the capability of fine sub-channel analysis on the pin level should be integrated into the calculation scheme of the main coupled 3D neutronics/thermal-hydraulics code. This is not a trivial task since these codes are already quite numerically intensive and the on-line introduction of an additional sub-channel code may increase the computational cost to such extent, that the local evaluations of the safety parameters in reasonable time will become impossible. One of the objectives of this research is to introduce an efficient refined methodology for hot channel analysis integrated into the coupled PWR safety code, so that the necessity of the expensive core-wide pin-based coupled calculations could be avoided. Methodologies for local safety parameters evaluation in conjunction with coupled 3D kinetics/thermal-hydraulics system code calculations have been studied elsewhere [1,2]. This paper presents the development and implementation of an on-line refined technique in the framework of a multi-level coupling algorithm, as well as results of applications for the rod ejection accident (REA) and the main steam line break (MSLB) analysis.

2. MULTI-LEVEL COUPLING ALGORITHM

The sub-channel analysis is performed by a sub-channel code that uses the boundary conditions calculated by the main neutronics/thermal-hydraulics system code. The coupling of COBRA-TF [3,4] is introduced by implementing a multi-level algorithm through parallel virtual machine (PVM) [5] with the PSU system code TRAC-PF1/NEM [6]. The PVM environment is a unique tool that makes it possible to couple large codes written in FORTRAN, so that the calculations can be performed on both single and multiple-processor architecture. The parallel approach implemented for coupling greatly reduces the amount of the coding work and provides calculation speedup.

First, the coupling was performed for a single isolated fuel bundle with the axial boundary conditions (inlet flow and enthalpy and outlet pressure) provided by TRAC-PF1 and the local pin powers provided by NEM. A fine sub-channel analysis is performed by COBRA-TF with the assumption that there is a zero cross flow with the neighboring assemblies. The cross flow effects are generally insignificant for the analysis of rod ejection accidents (REA) during the first few seconds of the transient. TMI-I 15×15-fuel assembly flow channel model was developed to run COBRA-TF and it is shown in Fig. 1. COBRA-TF model consists of 256 flow channels, 208 fuel rods, 16 thimble tubes and one instrumentation tube. One may see that this type of analysis is inaccurate, since the existing in the open PWR lattice cross flow is not included. Nevertheless, it demonstrates the feasibility of the coupling at the initial level and provides reasonable results for REA.

The next step is the involvement of the sub-channel cross flows into the calculational scheme via the radial boundary conditions provided from TRAC-PF1. The difficulties arise from the fact that a PWR

fuel assembly is an open bundle, so intensive cross flows between the neighboring bundles exist. The existing cross flow must be accounted for during sub-channel calculations via the radial boundary conditions provided in addition to the axial boundary conditions from the main code. The possibility to independently provide the radial boundary conditions for the sub-channel code is the crucial point of this option. With the option with radial cross flow modeling implemented, the newly recalculated safety parameters are more accurate and the obtained results are treated as the best estimate hottest pin safety margins predictions.

There are two aspects of coupling of two codes - spatial and temporal. When performing coupling under PVM, special attention is paid to the proper time synchronization of TRAC-PF1/NEM and COBRA-TF, since both codes use different time-step size selection algorithms. The synchronization multi-level coupling algorithm of the main neutronics/T-H code and a subchannel code proceeds as follows. TRAC-PF1/NEM is restarted first under PVM with the converged pre-calculated steady state solution. The converged solutions assumes that for a chosen hottest fuel assembly (See Fig. 2), the converged values of coolant mass flow rate (in kg/s) and enthalpy (in kJ/kg) at the bottom of the assembly, coolant pressure at the fuel assembly top (in kPa), as well as assembly average linear heat power ratio (in kW/m) and reconstructed relative pin-power distributions at the each axial layer of the fuel assembly are available from the VESSEL component of TRAC-PF1 and from the efficient pin-power reconstruction module of NEM. COBRA-TF is initialized at the beginning of TRAC-PF1/NEM calculating sequence and started with certain input parameters, which are given in COBRA-TF input file and assumed to be close to the converged average fuel channel parameters at the steady state (Fig. 3). After 1 second, the first data pass from TRAC-PF1/NEM to COBRA-TF is performed with the channel boundary conditions and detailed channel pin-power distribution mentioned above. In addition to T-H boundary conditions and pin powers, the value of the current time step from TRAC-PF1/NEM is also sent to COBRA-TF. TRAC-PF1/NEM time step is then used as a bounding time domain for COBRA-TF simulation. COBRA-TF, in turn, has its own minimum and maximum time steps within the received time domain. Generally, it is observed that the average time step of TRAC-PF1/NEM is around 0.1 - 0.5 seconds, while this value for COBRA-TF is approximately several orders of magnitude lower and less than 10^{-2} seconds. Therefore usually it takes at least dozen steps for COBRA-TF to proceed within a given time domain that corresponds to a time step of TRAC-PF1/NEM. The synchronization is performed in such a way that after a data passes TRAC-PF1/NEM proceeds further with next time step and new boundary conditions and powers and then stays idle until COBRA-TF calculations are done and it is ready to receive the new data of boundary conditions and powers from TRAC-PF1/NEM.

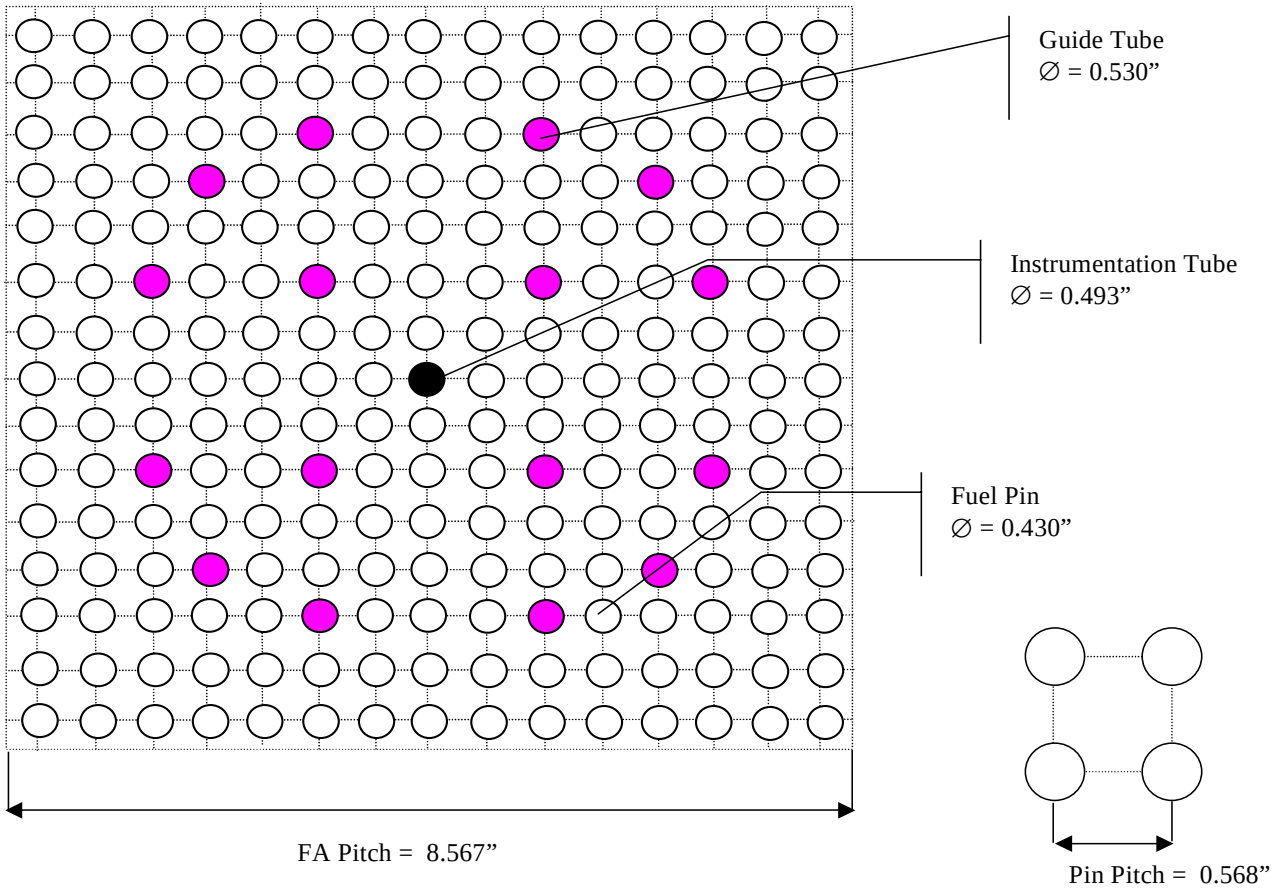

Figure 1. TMI-1 Fuel Assembly Layout. 225 Fuel Rods, 16 Thimble Tubes, 1 Instrumentation Tube.

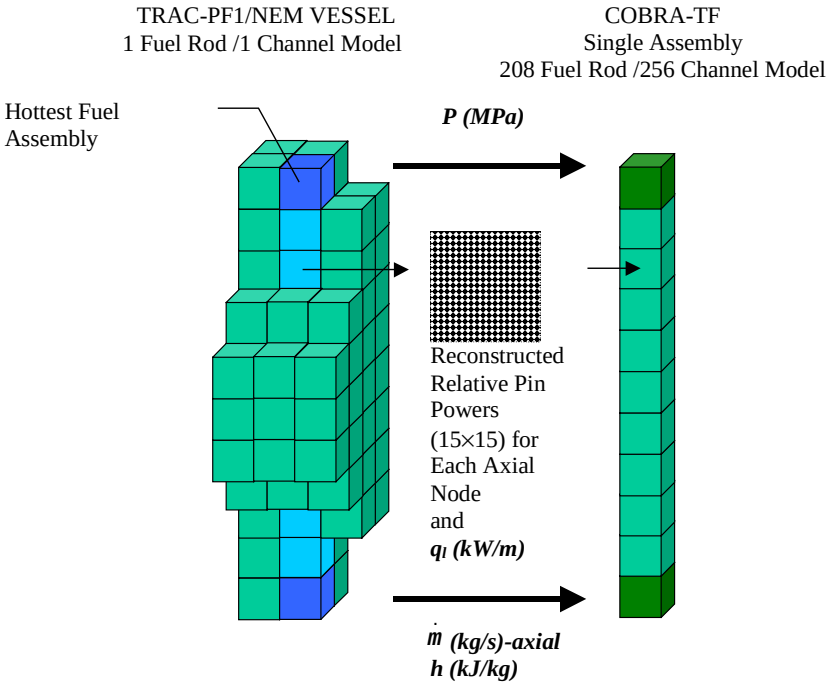

Figure 2. TRAC-PF1/NEM and COBRA-TF Spatial Coupling Scheme.

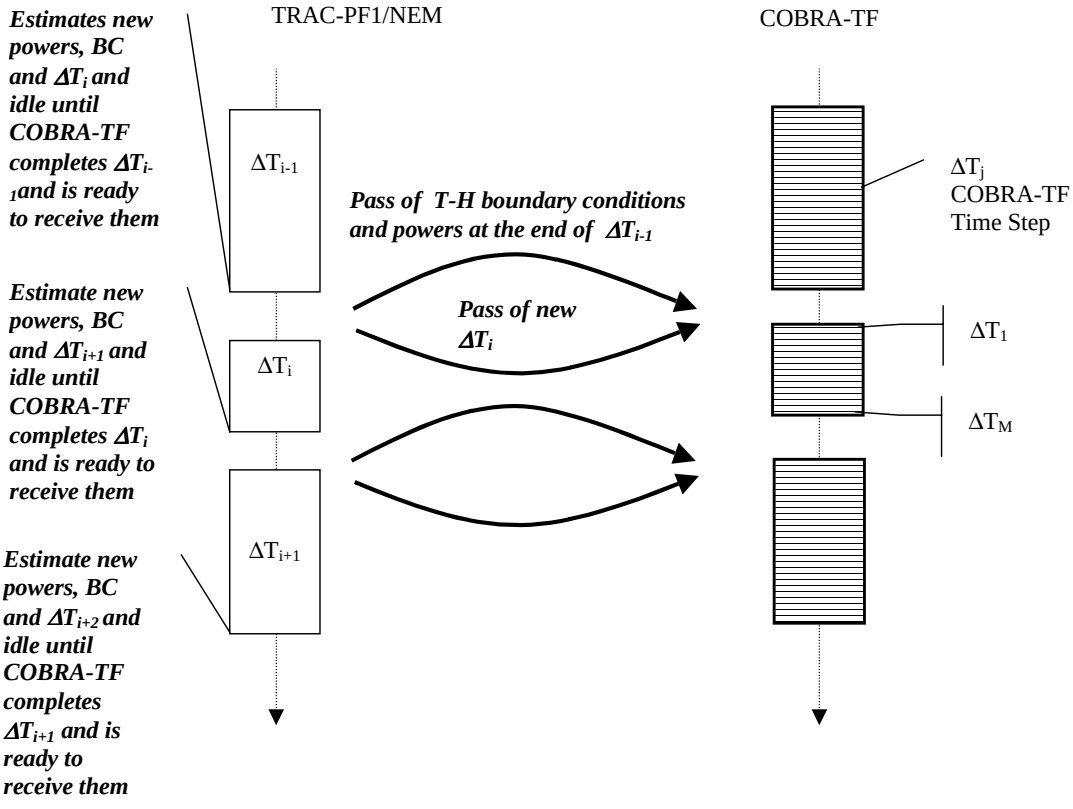

Figure 3. TRAC-PF1/NEM and COBRA-TF Temporal Coupling Scheme.

In the developed multi-level coupling algorithm the T-H solution obtained with COBRA-TF on local level does not influence the TRAC-PF1/NEM core calculations. COBRA-TF solution is used only to evaluate the local safety parameters in the hottest channel. However, one may account for COBRA-TF solution in TRAC-PF1/NEM pin-power reconstruction and use the local values of pin-based cell fuel temperatures, moderator temperatures and densities, as feed back parameters for the recalculation of determining of form function values. The node (assembly) average T-H feed back parameters are used in the first evaluation. The local T-H data can successfully be used in this step as a second iteration for a correction of form functions and update of reconstructed pin powers. Since, COBRA-TF calculations are more time consuming compared to TRAC-PF1/NEM, COBRA-TF calculations need be repeated only if the relative pin powers computed with the local T-H conditions deviate significantly from those ones computed with the assembly average T-H conditions.

2. VERIFICATION OF TRAC-PF1/NEM/COBRA-TF COUPLING ALGORITHM

The described PWR safety integrated methodology was implemented in the TRAC-PF1/NEM/COBRA-TF neutron kinetics/thermal-hydraulics coupled code system and verified on two established benchmarks, both characterized by uncontrolled reactivity insertion, namely, Rod Ejection Accident (REA) and Main-Steam-Line-Break (MSLB) accident.

2.1 TMI-1 REA ANALYSIS

Rod ejection is the rapid ejection of a single rod from the core region during the operation. The core power rises so rapidly that heat transfer does not proceed out of the fuel to the coolant during first two-three seconds; and the rod ejection process is almost adiabatic in nature, especially if the ejected rod worth is large enough (about the delayed neutron fraction). The power rise heats fuel and the Doppler feedback almost immediately causes the power reduction before the scram due to overpower. The rapid power rise may lead to fuel failures accompanied by a significant release of fuel and fission products into the coolant.

Fuel may fail by different modes. For example, the fuel failure due to relatively low worth rod ejections may be associated with the localized fuel fragmentation, local pin pressure increase and cladding weakening. Then DNB may lead to fuel failure. As it was shown [7], the beginning of uranium oxide fuel fragmentation occurs when the fuel enthalpy is equal or greater than 280 cal/g (it takes 1 cal of energy to raise the temperature of 1 g of water by 1°C at normal conditions). The chosen acceptance criterion for the analysis of REA accident is that the fuel enthalpy should remain below the 280-cal/g threshold point. However, this criterion was established based on experimental data obtained using fresh fuel. Recent experiments [8] showed that for high burnup fuel this criterion is equal to approximately 100 cal/g.

The analysis of the control rod ejection transient was performed based on the Three Mile Island – 1 reactor model with a high-burnup core at the end of cycle (EOC). The detailed T-H modeling has been accomplished using the Cartesian geometry vessel model option. The vessel is divided into 241 T-H radial cells (177 fuel and 64 reflector cells) and 26 layers axially (24 active fuel layers and 2 layers for bottom and top reflector). Each neutronics node is represented as a separate thermal-hydraulic cell in both – radial and axial directions. The heat-structure model has 177 rods corresponding to the number of fuel assemblies in the core (Figure 4). The Cartesian geometry vessel model matches exactly

geometrically the neutronics core model, because every neutronics node is directly coupled to a thermal-hydraulic cell and a heat-structure. The ejected rod position is H8 in Fig. 4.

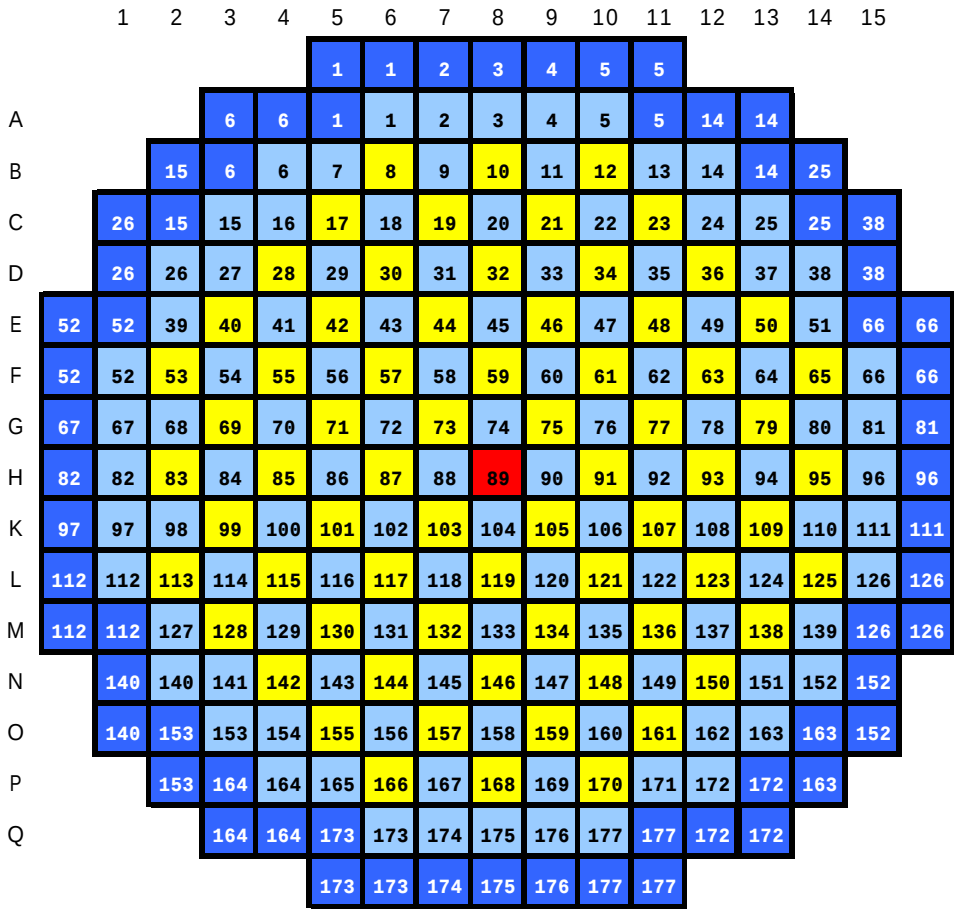

Figure 4. Radial Thermal-Hydraulic/Neutronics Mapping for TRAC-PF1/NEM Vessel Model in Cartesian Geometry. Ejected CR is in the Core Center (H8).

COBRA-TF solves the field equations for the hot channel/assembly with the axial boundary conditions (mass flow rate and enthalpy at the bottom, pressure at the top) provided from the TRAC-PF1 VESSEL component. The average linear heat generation rate (kW/ft) and the reconstructed pin power distributions for each of 208 fuel rods (in the framework of one assembly) are provided by the 3D neutronics module NEM. It was already mentioned that there is no significant heat transfer between fuel and coolant during a rapid REA. Therefore, one may expect that there is no significant cross flow between adjacent fuel assemblies. This assumption greatly simplifies the modeling with zero cross flow at the fuel assembly boundaries. The results of REA transient calculated with the computer code system TRAC-PF1/NEM/COBRA-TF are shown in the Figures 5 - 7.The local pin powers were calculated with the pin power reconstruction module and shown on Figure 5 for the axial node 22 at the time when the peak total power is reached. The values of local pin powers are given in Figure 6. One may see that the hottest pin is located in the row 12 from the side of the ejected CR. It was also observed that the locations of the hottest assembly and the hottest pin did not change during the transient, therefore the fuel temperature and enthalpy increment were monitored for that particular fuel pin in COBRA-TF.

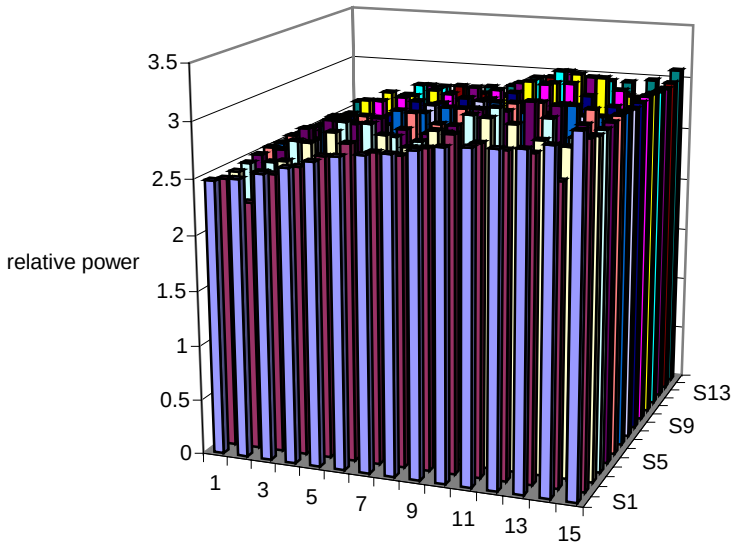

Figure 5. Relative Radial Power Distribution in the Hottest Node at the Time of Maximum Total Power (0.211 sec) during REA Simulation.

2.48	2.51	2.53	2.65	2.72	2.78	2.81	2.84	2.89	2.93	2.95	2.95	2.97	3.02	3.16	
2.45	2.25	2.53	2.61	2.72	2.85	2.79	2.78	2.86	2.99	2.94	2.90	2.89	2.67	3.05	
2.46	2.47	2.59	2.77	2.88	0.00	2.89	2.80	2.96	0.00	3.10	3.06	2.95	2.91	3.02	
2.49	2.51	2.72	0.00	2.92	2.92	2.83	2.80	2.89	3.07	3.14	0.00	3.09	2.94	3.02	
2.51	2.58	2.79	2.88	2.86	2.93	2.84	2.82	2.91	3.07	3.07	3.17	3.15	3.00	3.04	Ejected CR
2.53	2.66	0.00	2.84	2.89	0.00	2.95	2.86	3.02	0.00	3.10	3.12	0.00	3.09	3.05	here
2.52	2.57	2.72	2.71	2.77	2.91	2.91	2.97	2.97	3.05	2.97	2.97	3.06	2.98	3.04	
2.52	2.53	2.60	2.65	2.71	2.79	2.93	0.00	3.00	2.92	2.91	2.91	2.93	2.93	3.02	
2.52	2.57	2.72	2.71	2.77	2.91	2.91	2.97	2.97	3.05	2.97	2.97	3.06	2.98	3.03	
2.53	2.66	0.00	2.84	2.89	0.00	2.95	2.86	3.02	0.00	3.10	3.12	0.00	3.09	3.05	
2.52	2.58	2.78	2.88	2.86	2.93	2.84	2.82	2.91	3.07	3.07	3.17	3.14	3.00	3.03	
2.49	2.52	2.72	0.00	2.92	2.92	2.83	2.80	2.89	3.07	3.14	0.00	3.08	2.93	3.01	
2.48	2.48	2.59	2.77	2.88	0.00	2.89	2.80	2.96	0.00	3.10	3.06	2.94	2.89	2.99	
2.49	2.27	2.53	2.61	2.72	2.85	2.79	2.78	2.86	2.99	2.93	2.89	2.87	2.65	3.01	
2.57	2.56	2.60	2.66	2.72	2.78	2.81	2.84	2.88	2.93	2.94	2.94	2.95	2.98	3.09	

Figure 6. Relative Radial Pin Power Distribution in the Hottest Assembly in the Axial Node 22 at the Time of Maximum Total Power (0.211 sec). Hottest pins are Shown during REA Simulation.

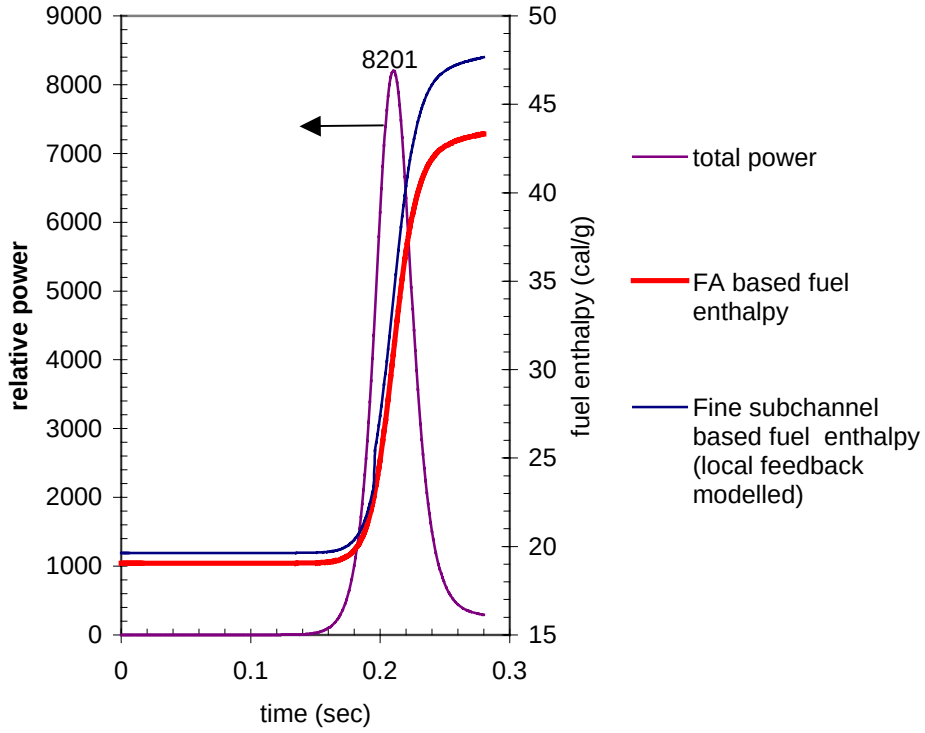

Figure 7. Maximum Total Power and Fuel Enthalpy during REA Simulation.

Figure 7 shows total core power and fuel enthalpy in the hottest pin as a function of time during the transient. Fuel enthalpy is calculated with two options – first, based on fuel assembly averaged parameters using hot channel factors (TRAC-PF1/NEM) and second, dynamically calculated with the COBRA-TF subchannel code. As one may see from Figure 7 there is a steep increment in fuel enthalpy that corresponds to a sharp power peak. COBRA-TF predicts larger temperature change during REA. As a result the fuel enthalpy rise predicted by COBRA-TF is greater than that the one based on the TRAC-PF1/NEM fuel temperatures. This result was obtained using local feed back modeling. If one does not use local feedback modeling in the pin power reconstruction, the predicted enthalpy rises higher (approximately to 54 cal/g), because the feedback effect is averaged over the assembly and thus it is underestimated. When the increase of fission power is terminated after approximately 0.28 second of transient, the flattening of fuel enthalpy vs. time curves is observed, so that the maximum fuel enthalpy predicted by online COBRA-TF calculations is not expected to get over 54-56 cal/g, which is almost 2 times less than 100 cal/g [8] – the approximate threshold point of the onset of fuel fragmentation for high-burnup fuel. These results prove that a large margin to the safety acceptance criterion exists for a PWR HZP REA at EOC.

2.2 OECD/NRC PWR MSLB BENCHMARK

Another limiting RIA for TMI-1 is the Main Steam Line Break (MSLB) [9] at hot full power (HFP), EOC, which is characterized by slow introduction of positive reactivity due to asymmetric addition of colder coolant into a core.

The neutronic core model contains 177 nodes in radial plane, each corresponding to 15x15 fuel assemblies pitch, except the central row of nodes, which are subdivided into two nodes. Axially, there

are 26 nodes including top and bottom reflectors. Totally, the core includes 438 unrodded and 195 rodded compositions.

The coupled 3D kinetics/T-H system model was used to calculate the Exercise 3 of MSLB benchmark and provide the core boundary conditions for the Exercise 2, in which the code is modeled in Cartesian geometry. The second exercise was used in this study, because, as it is discusses later, it is possible to account for radial mass and enthalpy transfer in COBRA-TF.

The return-to-power, re-criticality and DNB are the events that need to be monitored closely during MSLB transient. The return-to-power is associated with continuous addition of cooled water into the core and, thus, introduction of a positive reactivity. The return-to-power occurs at average 60-62 seconds after the initiation of MSLB event.

Figure 8 shows the total power response and the fuel maximum enthalpy calculated by TRAC-PF1/NEM during the MSLB transient. Depressurization and the correspondent negative reactivity insertion cause the power decrease after the beginning of the transient. Then the overcooled water starts entering the core introducing the positive reactivity (negative MTC). As a result the power reaches its maximum after approximately 6.5 seconds and the scram occurs then due to high neutron flux (112%) set point. Scram brings the core into subcritical condition and the fission power rapidly drops till the level of approximately 15% of nominal level around 10 seconds of transient. Excessive overcooling of the primary coolant in the broken steam generator continues and the overcooling of half of the core proceeds. This will eventually introduces a positive reactivity into the core, such that the core becomes critical again at around 60-second time point and the core returns to power. As a result, power increases and the return-to-power reaches the maximum of 37% of nominal level at around 67 seconds. The further dry-out of the broken steam generator causes the negative reactivity insertion into the core and further power decrease after the return-to-power highest point (after 70 seconds).

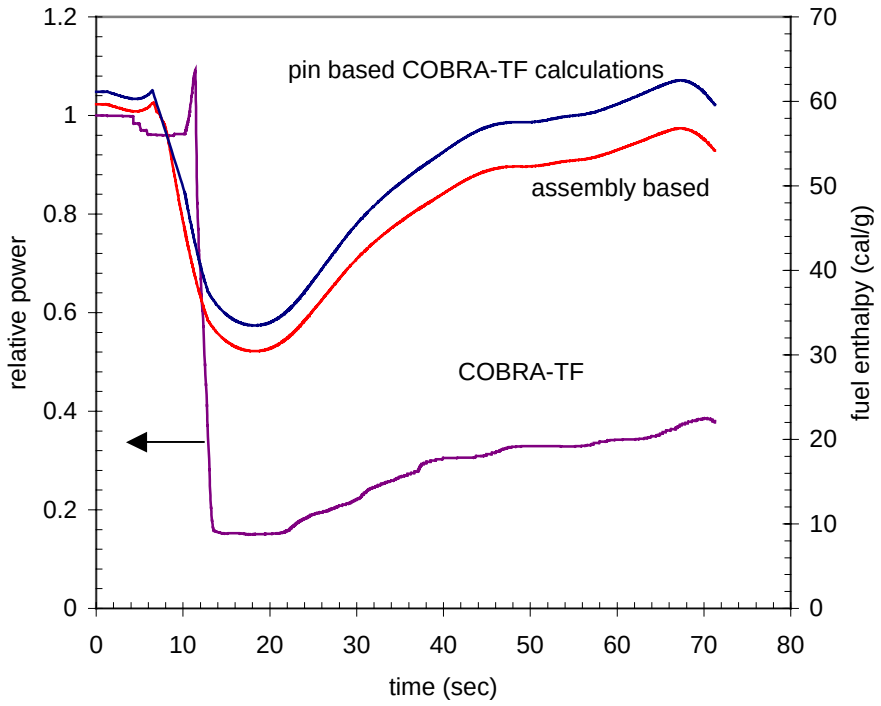

Figure 8. Relative Total Power. MSLB Exercise 2 – Return-to-Power Case.

One may see from Figure 8 that the pin-based fuel enthalpy does not exceed the value of 65 cal/g during MSLB transient span. This is less than the 100-cal/g acceptance criteria, which is used in this research. The local feedback is taken into account in this simulation. The upper value of fuel enthalpy without local feedback modeling reaches 70-cal/g value. Thus, despite the increased computational burden, the inclusion of the local feedback option provides the best estimate evaluation of the safety criteria.

The departure from nucleate boiling at the time of return-to-power (~ 70 seconds) is another major concern and it is addressed next. Actually, the necessity of DNBR monitoring is the main reason of introducing the finer subchannel analysis capabilities into TRAC-PF1/NEM/COBRA-TF code system. In contrast to REA, MSLB may be accompanied by a significant cross flows between fuel assemblies in a core. Therefore, for MSLB the cross flows between coarse T-H channels in TRAC-PF1/NEM cannot be neglected in a subchannel analysis, as it was done in REA. The cross flows contribute into mass and enthalpy transfer into/out of the outer raw of channels in COBRA-TF model. In this case, the cross flows can be considered as radial boundary conditions. To account for the radial mass and enthalpy transfer in the outer subchannel of COBRA-TF the additional row of channels was introduced into the TMI-1 fuel assembly model. These subchannels are fictitious and their only purpose is to provide the fluid momentum cells in COBRA-TF, where the values of mass flux in horizontal direction (x-y) between coarse cells for each axial level and enthalpies in the neighboring cells calculated by TRAC-PF1 can be provided.

It should be mentioned that it is not clear how to account for radial mass/enthalpy transfer occurred in the TRAC-PF1 3D Cylindrical VESSEL where, usually, several fuel assemblies are lumped into one radial channel. Even though the mass and heat transfers between macro T-H cells in TRAC-PF1 are calculated and available, there is no possibility to exactly map the macro cross flows into smaller cells of COBRA-TF. However, this mapping appears to be possible, if the 3D VESSEL is represented in Cartesian geometry with detailed one channel per one fuel assembly T-H model in TRAC-PF1. In this case, the cross flows calculated by TRAC-PF1 can be directly matched to the mass flow transfers occurred at the outer boundary of COBRA-TF fuel assembly. Mass flow variables are separated between adjacent momentum cells in COBRA-TF in each directions (x, y, z). Then the appropriate values of horizontal mass flows from TRAC-PF1 Cartesian VESSEL are fetched into only those COBRA-TF cells that compose the outer row of channels. Also, in addition to mass flow rates, the enthalpies should also be provided from TRAC-PF1 for the same COBRA-TF cells, so that altogether both of these values make the radial boundary conditions in the COBRA-TF model.

The modifications in TRAC-PF1 VESSEL are performed such that for each T-H channel in TRAC-PF1 the code calculates the mass flux (amount of mass passed through cell side surfaces in each direction per unit area per second, in kg/m^2-s) and coolant enthalpy (in J/kg) in the neighboring T-H cells. These two parameters are then passed to COBRA-TF in addition to axial boundary conditions and powers. It appears to be the most optimized way to account for the radial cross flows, if the nodalization in TRAC-PF1/NEM and COBRA-TF are the same. The minimum uncertainty in mass flows and enthalpies are introduced, when TRAC-PF1/NEM T-H model is Cartesian and each T-H channel is a fuel assembly. Obviously, if the vessel is in cylindrical geometry, the uncertainty in coarse mesh flow is greater when the mass flux and enthalpy are mapped to COBRA-TF Cartesian subchannel.

Special hot channel/assembly identification algorithm in TRAC-PF1/NEM dynamically locates the most loaded fuel assembly and reconstructs the pin powers at each axial elevation for this assembly. In MSLB analysis, this is the assembly next to the stuck out rod – N12 (see map on Fig. 4). The results are given in the form of matrix of relative radial pin powers computed for each axial node and

the average linear heat rate per one pin (in kW/ft) and then transferred to COBRA-TF via PVM at each time step.

As it was mentioned, the data needed for COBRA-TF subchannel calculations are provided from TRAC-PF1/NEM thermal-hydraulic system calculations. Figure 9 shows the time-dependent inlet coolant mass flux (kg/s/m^2) in the hottest fuel assembly provided from TRAC-PF1/NEM to COBRA-TF.

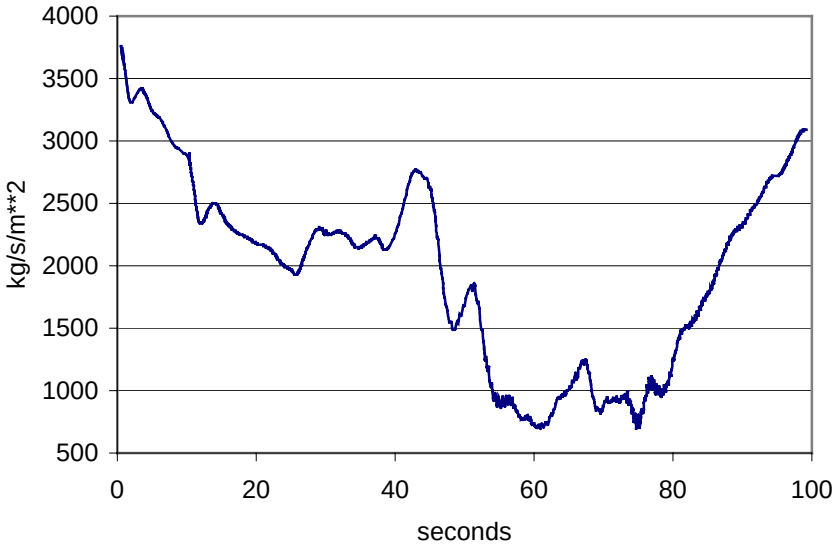

Figure 9. Time-dependent Boundary Condition – Axial Fluid Mass Flux at the Bottom of the Hottest Assembly during MSLB Transient.

The MSLB calculations with TRAC-PF1/NEM/COBRA-TF were performed and the minimum DNBR in the channels surrounding the hottest pin was calculated at each time step. Standard Westinghouse-3 CHF correlation was utilized, the same one that utilized in current COBRA-TF. The plot of minimum DNBR calculated with the Westinghouse-3 CHF correlation is given on Figure 10. One may see that the minimum local DNBR calculated with COBRA-TF is a little below of 4 and well above the allowable limit during the whole transient. These results were obtained using local feedback modeling for pin power reconstruction. Using assembly average feedback in COBRA-TF predicts lower minimum DNBR (around 3.5) but still above the critical limit.

To measure the effectiveness of the parallel implementation, the following two figures-of-merit are estimated: speedup and parallel efficiency. It was observed that during the MSLB transient it takes on average approximately 10 seconds of wall-clock time for TRAC-PF1/NEM to complete one time step, when the developed hydrodynamic solution is reached. TRAC-PF1/NEM time step is used as the bounding time domain for COBRA-TF calculations. The average time of approximately 60 seconds of wall-clock time required for COBRA-TF to complete its calculations within given time domain. Since, TRAC-PF1/NEM is mostly idle during the parallel calculations, the speedup can be estimated as:

$$Speedup \approx \frac{wall-clock\ \ time\ (COBRA-TF)}{wall-clock\ \ time\ (TRAC-PF1/NEM+COBRA-TF)} = \quad (1)$$

$$= \frac{(60s)}{(10s+60s)} \cong 0.86$$

In the current research only one processor was used, therefore the parallel efficiency is equal 86%.

It is sufficient to mention that the maximum time step in COBRA-TF is of the order of 10^{-3}. This time step observed only when the fully developed hydrodynamic solution is reached and there are no any significant perturbations. COBRA-TF typical time step is of the order of 10^{-5} during code initialization or when significant change in hydrodynamic solution occurs. On the other hand, the minimum average time step of TRAC-PF1/NEM is of the order of 10^{-3}. It can be seen that most of CPU resources are spent on solution of COBRA-TF part. Therefore, the possibility of the COBRA-TF code numerics improvements and speed-up should be considered for future analysis.

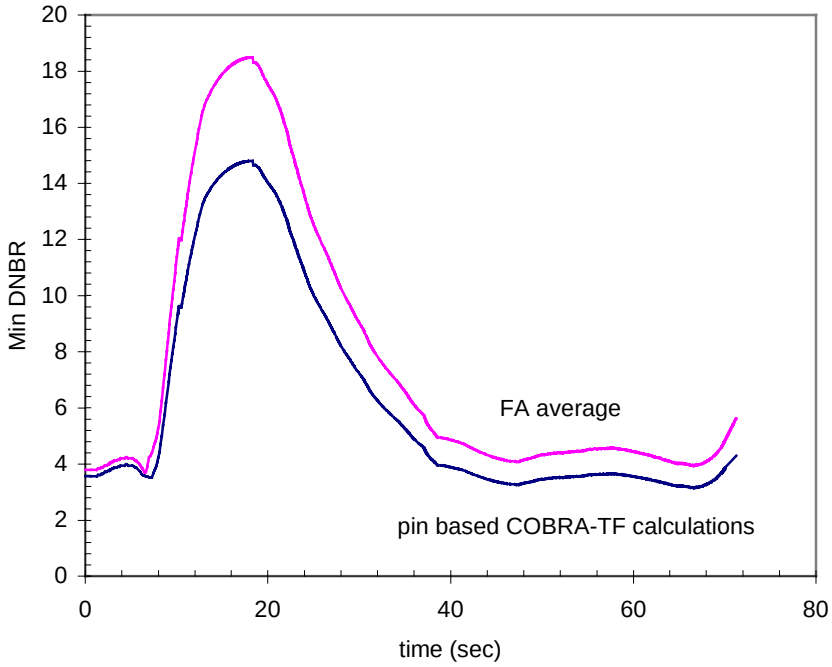

Figure 10. Minimum DNBR. MSLB Exercise 2 – Return-to-Power Case.

CONCLUSIONS

The coupling of COBRA-TF with TRAC-PF1/NEM is implemented under PVM, and the special attention is paid to the spatial coupling algorithm as well as to the temporal synchronization methodology of the two codes execution.

When implemented, the coupled TRAC-PF1/NEM/COBRA-TF code is capable to provide on0line accurate distributions of (fuel rod-based) local safety parameters. These parameters, including the local power, pellet temperature distribution, and local heat fluxes could be further employed for the evaluations of the material behavior of selected fuel rod(s) during certain transient conditions.

Coupling the system code TRAC-PF1/NEM with the subchannel code COBRA-TF allows one to perform calculations when refine evaluations of the local conditions such as the fuel enthalpy and the departure from nucleate boiling ratio are necessary. The data transfer between TRAC-PF1/NEM and COBRA-TF codes was successfully tested. The values of the pin average linear heat rate (in kW/ft) and relative pin powers (calculated for each axial elevation in the hottest fuel assembly) together with axial T-H boundary conditions (that include mass flow rate and coolant enthalpy at the bottom and coolant pressure at the top) and radial mass flow rates and enthalpies (at the adjacent cells) are transferred to COBRA-TF at each time step during the transient. PVM allows COBRA-TF to run in parallel with the main code. Two benchmark calculations, Rod Ejection Accident and Main Steam Line Break, were performed using the coupled TRAC-PF1/NEM/COBRA-TF code system based on the Three Mile Island-1 reactor model as a reference design. It was demonstrated in REA analysis that the COBRA-TF module predicted greater value of fuel enthalpy on the local level compared to the average value computed by TRAC-PF1/NEM. Still, the predicted maximum local fuel enthalpy was below the limiting value of 100 cal/g used as a failure criterion for high-burnup fuel. The second benchmark used in testing the TRAC-PF1/NEM/COBRA-TF coupled code system was the Exercise 2 of OECD/NRC PWR MSLB benchmark. The obtained values of the local minimum DNBR calculated with COBRA-TF utilizing Westinghouse-3 CHF correlation were above the allowable limit; however, again the assembly-based evaluations underestimated the accident consequences.

The obtained results indicate that the coupling of TRAC-PF1/NEM with COBRA-TF was done correctly. Online local safety evaluations are necessary, since the obtained results on assembly level tent to underestimate the accident consequences. The calculations also demonstrated that COBRA-TF detailed modeling of PWR requires a significant amount of the computational resources and COBRA-TF module is the most time consuming in the TRAC-PF1/NEM/COBRA-TF code system. The COBRA-TF code numerics improvement and more powerful computers are the proposed solutions of the mentioned problem. The investigations to improve the COBRA-TF performance are under way at Penn State, involving the implementation of new efficient matrix solver for COBRA-TF calculations in pin-by-pin resolution, and the FORTRAN 90 code version, with development of dynamic memory allocation.

REFERENCES

1. R. Böer and A. Knoll, "Sensitivity Studies for MSLB Exercises 2 and 3 with RELAP5/PANBOX", *TANSAO*, **84**, p. 20, 2001.

2. H. Joo, et al, "Significance of Refined Core Thermal-Hydraulics Nodalization in the MSLB Analysis", *TANSAO*, **84**, p. 29, 2001.

3. COBRA/TRAC – A Thermal-Hydraulics Code for Transient Analysis of Nuclear Reactor Vessels and Primary Coolant Systems. Equations and Constitutive Models. Vol. 1. NUREG/CR-3046 PNL-4385. March 1983.

4. COBRA/TRAC – A Thermal-Hydraulics Code for Transient Analysis of Nuclear Reactor Vessels and Primary Coolant Systems. Users' Manual. Vol. 3. NUREG/CR-3046 PNL-4385. March 1983.

5. A.Geist et al, "PVM Parallel Virtual Machine", A User's Guide and Tutorial for Networked Parallel Computing, MIT Press, 1994.

6. K.Ivanov et al, "Nodal Kinetic Model Upgrade in The Penn State Coupled TRAC/NEM Codes," *Annl. Nucl. Ener.*, **26**, 1205 (1999).

7. R.C.Liimatainen and F.J.Testa, "Studies in TRAT of Zircaloy-2 Clad UO_2-Core Simulated Fuel Elements", ANL-7225, 1966.

8. "Evaluation of Irradiated Fuel During RIA Simulation Tests", EPRI TR-106387 2905-06, 1996.

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