COBRA-TF EVALUATION AND APPLICATION FOR PWR STEAMLINE BREAK DNB ANALYSIS

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ABSTRACT

COBRA-TF (CTF) is a thermal hydraulic (T/H) subchannel code capable of calculating reversed flow, countercurrent flow and cross-flow using either three-dimensional (3D) Cartesian or subchannel coordinate formulations for fluid flow and heat transfer solutions. Recent software improvements on the code capability under the Consortium for Advanced Simulation of Light Water Reactors (CASL) program include software optimization, new closure models, and parallelization for modeling full reactor core T/H responses under PWR normal operating and accident conditions. Under the CASL development program, the CTF code is used for predicting the core T/H responses for both reactivity feedback and margin to departure from nucleate boiling (DNB) during a Pressurized Water Reactor (PWR) main steamline break (SLB) accident. An evaluation of the CTF code capability was performed for such application in comparison to experimental data from rod bundle tests simulating PWR fuel and SLB conditions. The CTF subchannel predictive capability was also compared with results from subchannel codes used in the industry. The CTF modeling capability was further evaluated using a full reactor core subchannel model to predict fluid and fuel conditions at the DNB-limiting time step of a PWR SLB case without offsite power on the Oak Ridge National Laboratory (ORNL) high-performance computing platform. Results of the capability evaluation indicate that CTF could potentially be applied to PWR accident analysis with respect to the DNB acceptance criterion.

KEYWORDS
COBRA-TF, Subchannel Modeling, DNB, Steamline Break

1. INTRODUCTION

COBRA-TF (Coolant Boiling in Rod Arrays – Two Fluid) or CTF is a thermal hydraulic (T/H) subchannel code based on a two-phase two-fluid model that formulates the conservation equations of mass, energy, and momentum for three fields of vapor, continuous liquid, and entrained liquid droplet. The code is capable of calculating reversed flow, countercurrent flow and cross-flow using either three-dimensional (3D) Cartesian or subchannel coordinate formulations for fluid flow and heat transfer solutions. Recent software improvements on the code capability under the Consortium for Advanced Simulation of Light Water Reactors (CASL) sponsored by the U.S. Department of Energy (DOE) include software optimization, new closure models, and parallelization for modeling full reactor core T/H responses under PWR normal operating and accident conditions.

During a postulated PWR main steamline break (SLB) event initiated from the hot zero power (HZP) condition, increased steam flow rate from the broken main steam pipe on one of the steam generators would result in significantly reduced primary coolant temperature and an increase in the reactor core average power and the peak fuel rod power, thus imposing a challenge to the Departure from Nucleate Boiling (DNB) criterion. Although it is classified as a Condition IV event that allows fuel cladding failures for radiological dose evaluation, many plant safety analyses conservatively show no fuel rod failure by meeting the DNB Ratio (DNBR) limit with a 95% probability at a 95% confidence level. The DNB analysis is performed for the more limiting SLB case either with offsite power available and the reactor coolant pumps in operation, or without offsite power and the reactor core is cooled through natural circulation.

Under the CASL program, a new code system, Virtual Environment for Reactor Applications-Core Simulator (VERA-CS), has been developed to couple the CTF code with a neutron transport code for PWR core modeling and simulation. Within VERA-CS, CTF simulates the core T/H responses for both the reactivity feedback and margin to DNB at the limiting time step of the SLB accident, with the core boundary conditions from a system transient code. An evaluation of the CTF capability was performed for such application in comparison to experimental data from the rod bundle tests simulating the PWR fuel and SLB conditions, including turbulent mixing test, DNB test, and buoyancy flow distribution test. Moreover, The CTF modeling capability was evaluated using a full reactor core subchannel model to simulate several DNB-limiting cases of a PWR SLB event with and without offsite power on a high-performance multiprocessor computing platform.

2. CTF COMPARISONS WITH DATA AND BENCHMARK WITH VIPRE-W

COBRA-TF was originally developed by the Pacific Northwest Laboratory and has been updated over several decades by several organizations. CTF [1] is the version of COBRA-TF being jointly developed and maintained by the Penn State University and the Oak Ridge National Laboratory as part of the CASL VERA multi-physics software package for reactor simulation, VERA-CS. Code improvements on CTF include:

- Improvements to user-friendliness of the code through creation of a PWR preprocessor utility [2],
- Code maintenance, including source version tracking, bug fixes, and transition to modern Fortran,
- Incorporation of an automated build and testing system using TriBITS [3],
- Addition of new code outputs for better data accessibility and simulation visualization,
- Extensive source code optimizations and full parallelization of the code [2, 4],
- Improvements to closure models, including boiling heat transfer model and grid heat transfer enhancement model for PWR applications,
- Addition of consistent set of steam tables from IAPWS-97 standard [5],
- Application of extensive automated code regression test suite to prevent code regression during development activities,
- Code validation study with experimental data [6].

The CTF preprocessor is an independent utility designed to generate input decks from a reduced set of user-input data in a quick and less error-prone manner. The utility produces models of PWR rod-bundle geometry, based on input of basic characteristics of the fuel assembly including rod pitch and fuel rod dimensions. The CTF output parameters such as subchannel flow rate, temperature, enthalpy, and pressure and fuel rod temperatures are projected onto a user-specified or pre-processor generated mesh and written to files in a format suitable for visualization.

The CTF code capability for PWR DNB analysis was evaluated based on comparisons of the CTF predictions with rod bundle test data and subchannel code VIPRE-W [7], which is the Westinghouse version of the VIPRE-01 code originally developed by the Pacific Northwest National Laboratory under the sponsorship of the Electric Power Research Institute (EPRI) [8]. The rod bundle test data selected for the evaluation included turbulent mixing data, buoyancy data and DNB data. .

2.1. Comparison with Rod Bundle Mixing Test Data

Mixing and DNB test data from the PWR Subchannel and Bundle Test (PSBT) performed by the Nuclear Power Engineering Corporation (NUPEC) of Japan were made available for thermal-hydraulic modeling and benchmark through the Organization for Economic Cooperation and Development (OECD). The benchmark database and benchmark problem specifications [9] were prepared jointly by the Pennsylvania State University (PSU) and the Japan Nuclear Energy Safety Organization (JNES) with support from the U.S. Nuclear Regulatory Commission (NRC) and OECD Nuclear Energy Agency (OECD/NEA).

The NUPEC test facility was similar to other rod bundle test facilities, such as the Columbia University Heat Transfer Research Facility [10]. The test section contained an electrically heated rod bundle. The electrically heated test rods (or tubes) were made of Inconel 600 with the alumina insulator in the interior. There were three types of grid spacers along the bundle length: mixing vane (MV) grid, non-mixing vane (NMV) grid, and simple support (SS) grid. The diagrams of the test section and axial locations of the grid spacers and pressure taps are shown in Figure 1.

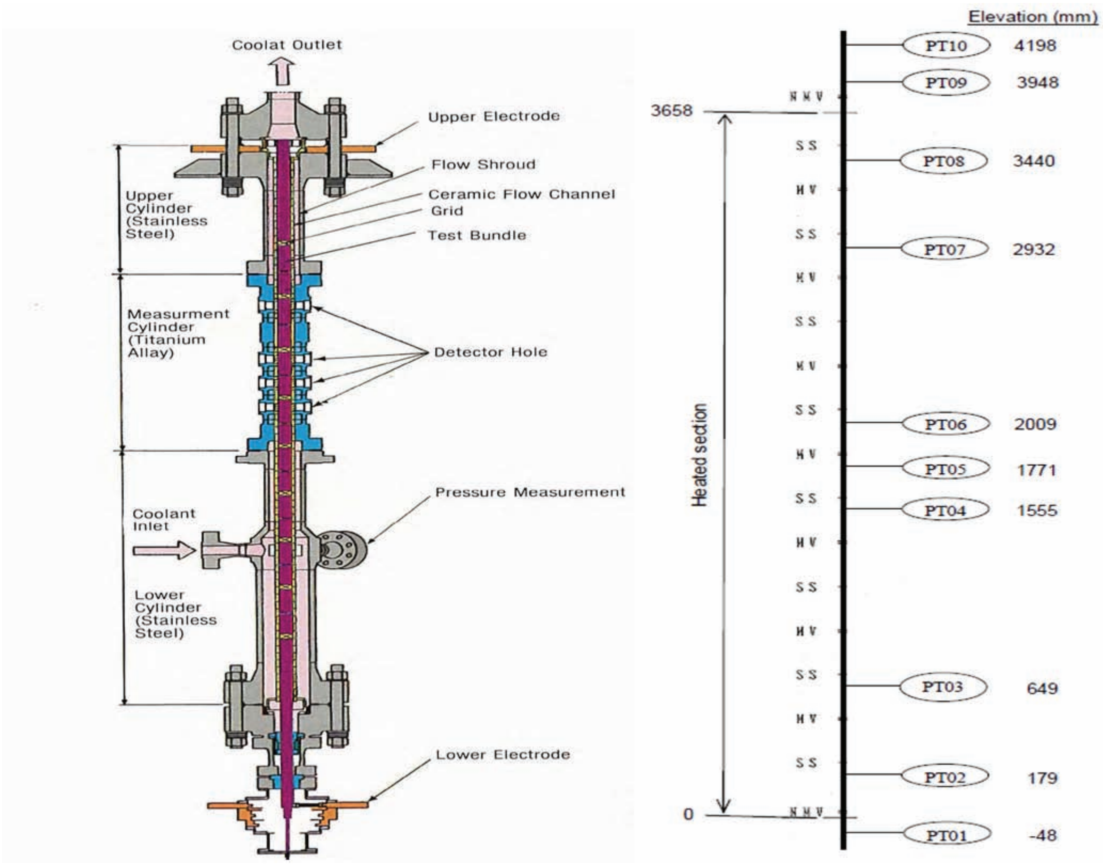

Figure 1. PSBT Test Section and Axial Diagram of Test Bundles [9].

A mixing test is designed to measure turbulent mixing effect in a rod bundle with a relatively large radial power gradient between hot and cold rods. For the PSBT mixing test (A1), the cold and hot rods were arranged in two columns on each side of the 5x5 test bundle, with three cold rods and two hot rods arranged alternatively in the central column. The hot to cold rod power ratio was four to one. Thirty-six thermocouples were placed at each subchannel of the 5x5 test bundle for measuring the fluid exit temperature. The mixing test was conducted mainly under the single-phase flow conditions with a uniform axial power distribution.

The turbulent mixing in subchannels was modeled using the following empirical correlation:

$$\Delta Q = -w' \times \Delta h \times \Delta X \quad (1)$$

where ΔQ = energy exchange due turbulent mixing (W or Btu/hr)
w' = lateral turbulent flow per unit length (kg/s-m or lbm/hr-ft)
Δh = enthalpy difference between two subchannels (J/kg or Btu/lbm)
ΔX = axial nodal length (m or ft)

$$w' = ABETA \times G_{AVG} \times S \quad (2)$$

where A*BETA* = empirical coefficient,
G_{AVG} = average axial mass flow in the connected channels (kg/s-m^2 or lbm/s-ft^2)
S = rod-to-rod gap width (m or ft)

The ABETA value was found to be about 0.07 based on the PSBT mixing test data with the VIPRE-W code [11]. The ABETA value of 0.07 was then used as input to the CTF calculations. The CTF model accounted for mixing exchange in mass, momentum and energy. The radial geometric model for the PSBT mixing and DNB tests is shown in Figure 2. The axial nodal length was set to be 24.5mm (1.0inch).

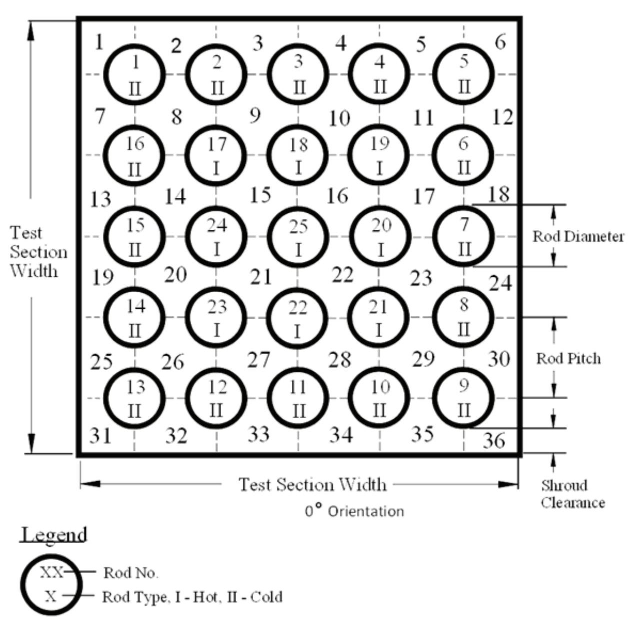

Figure 2. CTF Modeling of PSBT Test Bundles.

The CTF calculations were made for the two test runs shown in Table 1. The predicted fluid temperature distributions at the channel exit were compared with the measured temperature values and with the VIPRE-W results. The temperature differences are shown in Figure 3.

Table 1. Testing Conditions of PSBT Mixing Test (A1) for Comparison with CTF Predictions.

PSBT Mixing Test (A1) Run #	Pressure (bar)	Mass Flux (10^3kg/m^2-s)	Inlet Temperature (˚C)	Bundle Power (MW)
1	98.1	1.283	166.1	0.98
13	147.4	3.053	255.7	2.68

Case 1 Delta

1	2	3	4	5	6	Channel number
-0.43	1.62	1.14	4.78	4.98	2.39	Measured -COBRA-TF (˚C)
-0.35	1.69	1.21	4.85	5.04	2.45	Measured - VIPRE (˚C)
0.08	0.07	0.07	0.07	0.06	0.06	COBRATF-VIPRE (˚C)
7	8	9	10	11	12	
0.49	1.42	0.06	4.76	5.42	4.87	
0.58	1.51	0.13	4.84	5.50	4.92	
0.09	0.09	0.07	0.08	0.08	0.05	
13	14	15	16	17	18	
-1.74	-3.27	-0.42	0.06	2.79	3.44	
-1.65	-3.17	-0.35	0.14	2.85	3.49	
0.09	0.10	0.07	0.08	0.06	0.05	
19	20	21	22	23	24	
-3.63	-5.37	-6.12	-4.84	1.58	1.85	
-3.55	-5.27	-6.05	-4.76	1.65	1.89	
0.08	0.10	0.07	0.08	0.07	0.04	
25	26	27	28	29	30	
-3.81	-4.78	-5.54	-6.25	-1.38	-0.73	
-3.72	-4.69	-5.47	-6.16	-1.30	-0.68	
0.09	0.09	0.07	0.09	0.08	0.05	
31	32	33	34	35	36	
-3.13	-2.08	2.24	-0.23	-0.43	-2.51	
-3.05	-2.01	2.31	-0.15	-0.36	-2.45	
0.08	0.07	0.07	0.08	0.07	0.06	

Case 13 Delta

1	2	3	4	5	6	Channel number
1.65	1.19	5.88	5.8	6.89	4.44	Measured -COBRA-TF (˚C)
2.01	1.43	5	5	6.13	3.67	Measured - VIPRE (˚C)
0.36	0.24	-0.88	-0.8	-0.76	-0.77	COBRATF-VIPRE (˚C)
7	8	9	10	11	12	
-0.82	-0.95	4.87	6.87	7.76	6.22	
-0.28	-0.13	3.92	6.09	7.01	5.45	
0.54	0.82	-0.95	-0.78	-0.75	-0.77	
13	14	15	16	17	18	
-1.47	-2.45	3.68	4.53	3.84	5.09	
-1.08	-1.78	2.79	3.73	3.08	4.32	
0.39	0.67	-0.89	-0.8	-0.76	-0.77	
19	20	21	22	23	24	
-2.57	-4.85	-2.12	-3.87	0.84	0.99	
-2.18	-4.18	-3.01	-4.67	0.08	0.22	
0.39	0.67	-0.89	-0.8	-0.76	-0.77	
25	26	27	28	29	30	
-2.02	-3.15	-0.03	-1.33	-1.54	-1.88	
-1.48	-2.33	-0.98	-2.11	-2.29	-2.65	
0.54	0.82	-0.95	-0.78	-0.75	-0.77	
31	32	33	34	35	36	
0.76	-1.31	-5.31	-2.1	-3.31	-4.36	
1.11	-1.07	-6.2	-2.9	-4.07	-5.13	
0.35	0.24	-0.89	-0.8	-0.76	-0.77	

Figure 3. Comparisons of CTF Exit Temperature with PSBT Mixing Data and VIPRE-W Results.

The VIPRE-W and COBRA-TF code predicted temperatures were generally in very good agreement, although the turbulent mixing formulation and solution method were quite different between the two codes. The differences between the code predictions and the measured values were larger, possibly due to the uncertainties in the bundle and grid spacer orientations and calibration errors in the thermocouple measurements

2.2. Comparison with VIPRE-W for DNB Predictions

A DNB test is designed to measure the rod bundle power when DNB occurs on one or more test rods. For the PSBT DNB tests, DNB occurrences were detected from thermocouples positioned in the test rods at the thermal limiting elevations. A bundle radial power gradient was applied to ensure that most DNB occurrences on the interior (hot) test rods away from the test housing. A DNB data point was recorded together with the conditions under which it occurred, such as the measured bundle average power, pressure, mass flow rate and inlet temperature. Two DNB test conditions with a uniform axial power distribution from a PSBT test (A0) are listed in Table 2. CTF modeling of the DNB test bundle was similar to that used for the mixing test in Section 2.1. CTF and VIPRE-W predictions of subchannel flow rates, equilibrium quality and void fractions are plotted as a function of the axial elevation in Figures 4 and 5 for the two test cases in Table 2.

Table 2. Testing Conditions of PSBT DNB Test (A0) for CTF Benchmark.

PSBT DNB Test (A0) Run #	Pressure (bar)	Mass Flux (10^3kg/m^2-s)	Inlet Temperature (˚C)	Bundle Power (MW)
1	98.5	1.411	243.2	2.70
10	147.4	2.251	253.1	3.22

No measurement from the DNB test could be easily compared with the code predictions. For reactor design applications, Departure from Nucleate Boiling Ratio (DNBR) calculations are performed using an empirical DNB correlation based on the bundle test data and local fluid conditions predicted from a subchannel code. The VIPRE-W code has been used for developing many DNB correlations. Therefore, for the code capability evaluation, a code-to-code benchmark between VIPRE-W and CTF predictions was made on the local fluid conditions (flow and quality) that were used as input to an empirical DNB correlation for the DNBR calculation. The code predictions are generally in good agreement despite the differences in the code solution methods and modeling of turbulent mixing and two-phase flow. The good agreement in the local fluid predictions indicates that CTF can be used with an empirical DNB correlation for PWR DNBR margin prediction.

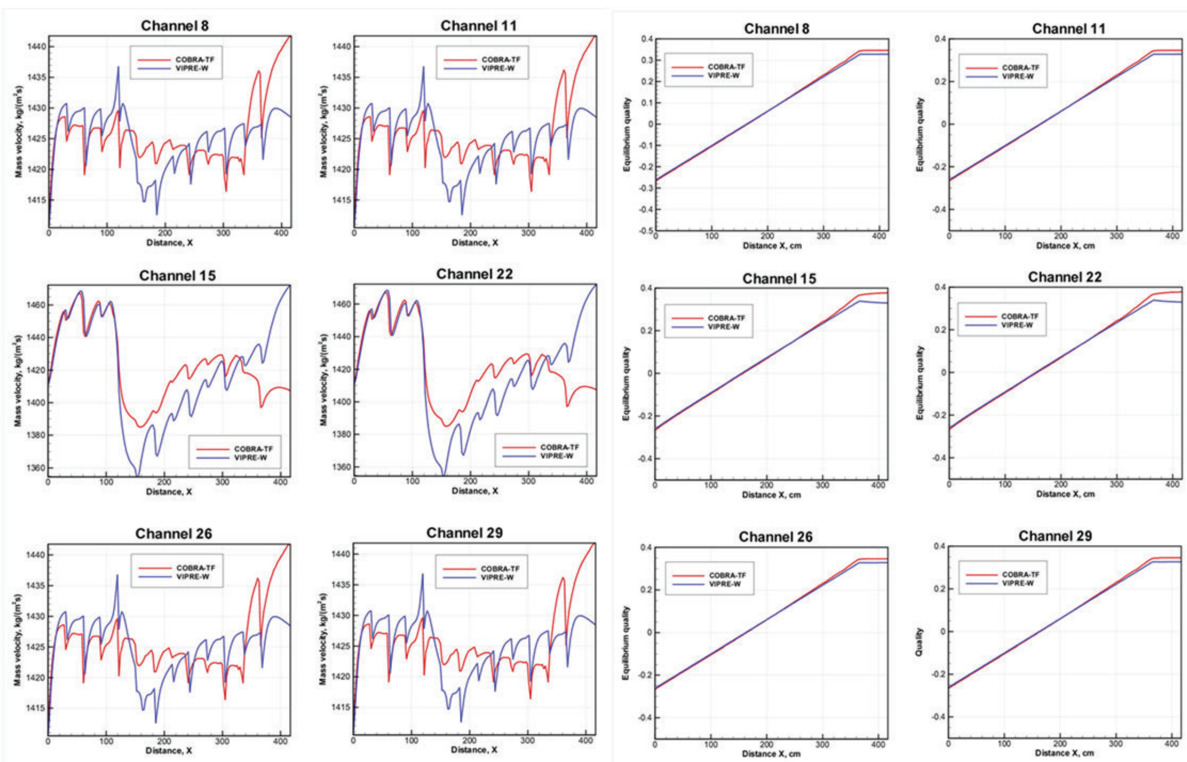

Figure 4a. Comparison of CTF and VIPRE-W Local Fluid Predictions – PSBT DNB (A0) Case 1.

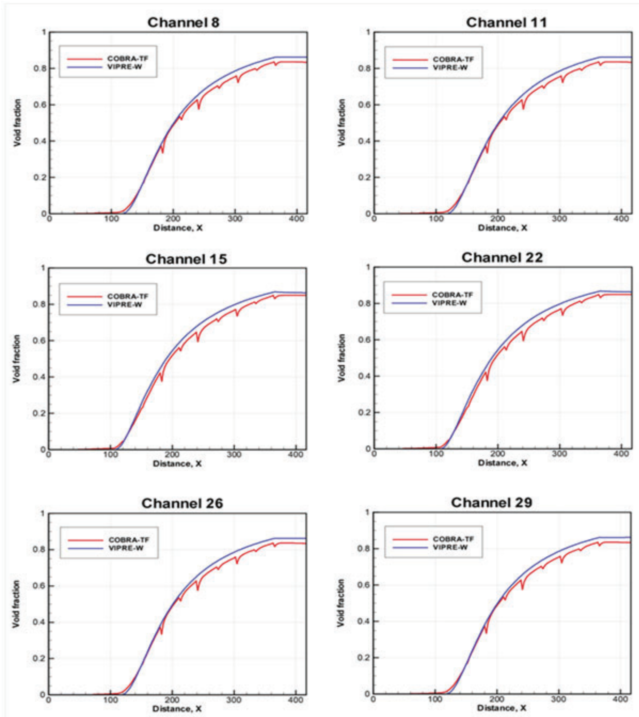

Figure 4b. Comparison of CTF and VIPRE-W Void Predictions – PSBT DNB (A0) Case 1.

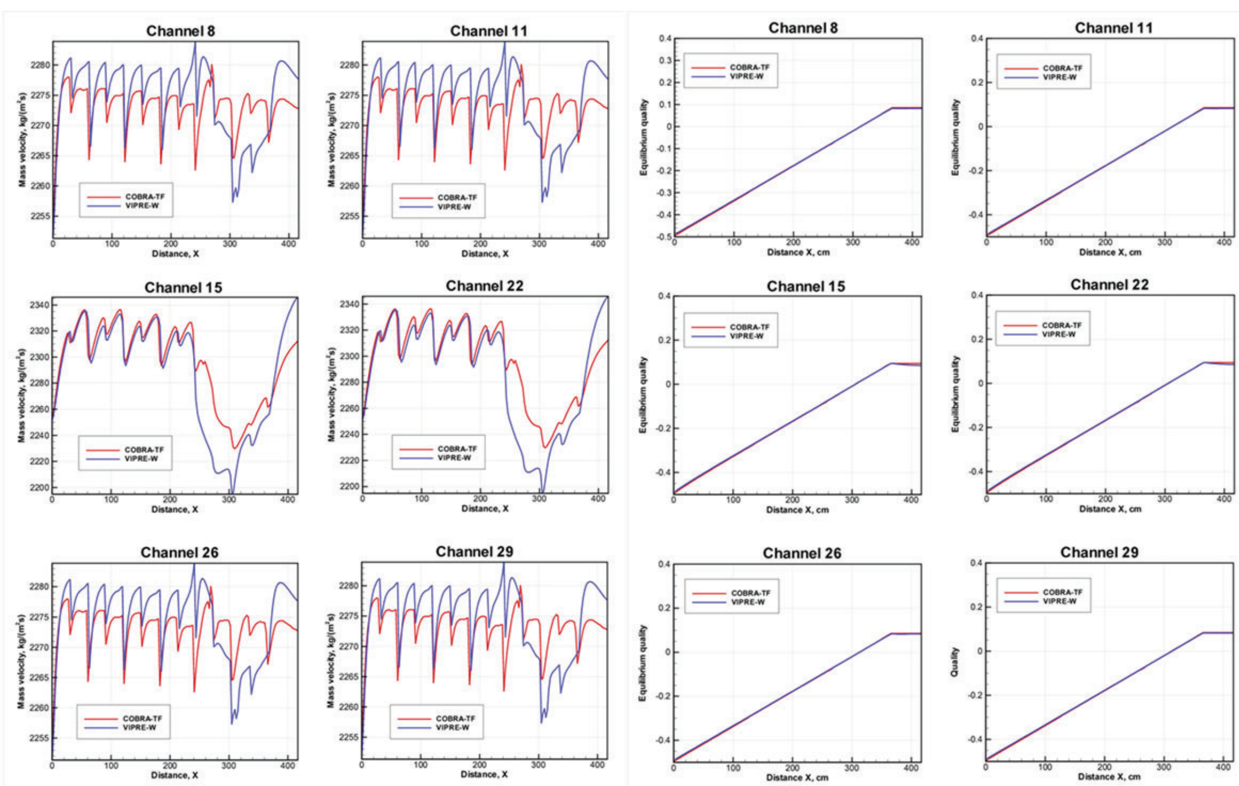

Figure 5a. Comparison of CTF and VIPRE-W Local Fluid Predictions – PSBT DNB (A0) Case 10.

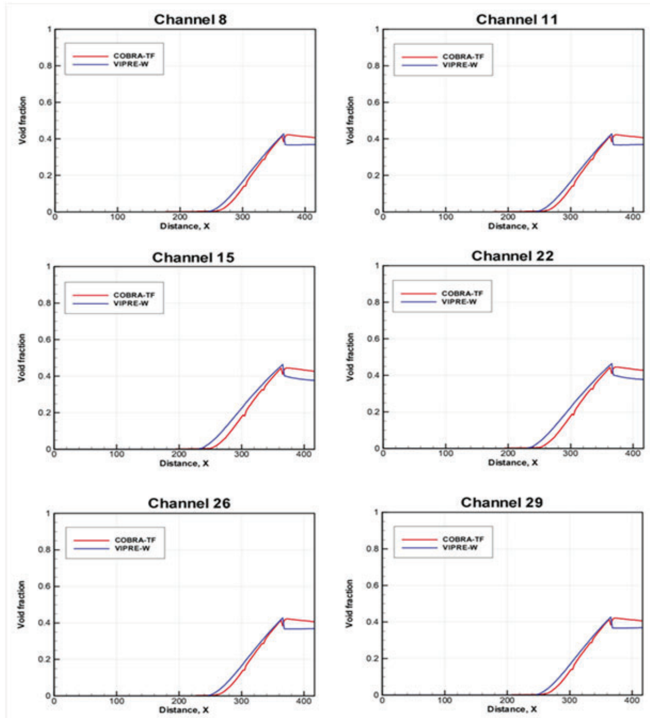

Figure 5b. Comparison of CTF and VIPRE-W Void Predictions – PSBT DNB (A0) Case 10.

2.3. Comparison with Buoyancy Test Data

Tests were performed at the Pacific Northwest Laboratories to investigate combined free and forced convection heat transfer in rod bundles with large radial power gradients [12]. The test section was a 2x6 rod bundle in vertical flow housing. The rods were electrically heated stainless steel tubes with an active length of 48 inches (121.92 cm) and an outside diameter of 0.475 inch (1.207 cm). The rod-to-rod pitch was 0.575 inch (1.461 cm). The flow housing dimensions were 1.275 inch (3.239 cm) by 3.575 inch (9.081 cm). Windows for Laser Doppler Anemometry (LDA) velocity measurements were spaced 6 inches apart along the heated length. The LDA measurements of local velocity were made along three radial positions, Y = -0.581 inch (1.48cm), 0.0 and 0.581 inch (1.48 cm). However, the LDA measurements at Y=+/-0.581 were not representative of the actual PWR fuel assembly design, since they were affected by the flow housing walls. Local fluid temperatures were also measured using thermocouples located at the center of the flow channels in the middle row (Y=0.0). At the windows where the data were collected, the LDA measurements were made in fine increments across the bundle, and the thermocouple measurements were taken only at the subchannel center.

Figure 6 provides a cross-section view of the 2x6 test section and the CTF model. An axial schematic of the test and the CTF axial meshing are shown in Figure 7. The CTF model consisted of twelve rods and twenty-one channels. Rods 1 through 6 were the high-power rods, and Rods 7 through 12 were the unheated rods. In addition to the four-foot (1.22 m) heated length, six inches (15.24 cm) of unheated length at the inlet and the outlet were included. Since the test condition was under single phase flow, options for two-phase flow, void drift, and liquid entrainment were not used in the CTF model. The following test condition was input as the boundary condition to the CTF calculation:

Pressure, psi (Kpa)	-	60 (413.7)
Inlet Temperature, F (ºC)	-	60.6 (15.9)

Outlet Temperature, F (ºC)	-	93.1 (33.9)
Flow Rate, gpm (cm^3/s)	-	1.25 (78.9)
Power per Rod (kW)	-	0.91
Hot-to-Cold Rod Power Ratio	-	1:0

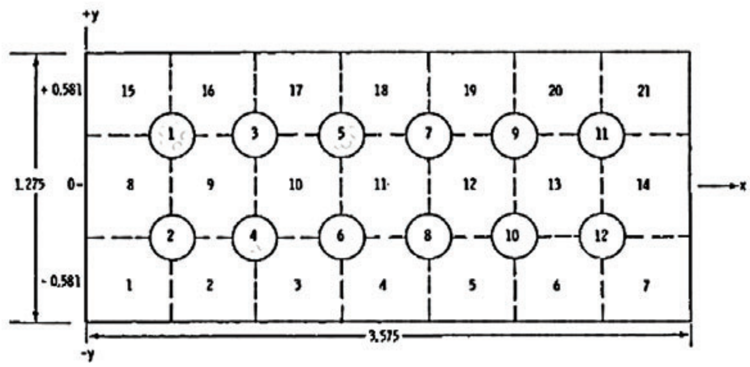

Figure 6. CTF Radial Modeling of PNNL 2x6 Test Section [12].

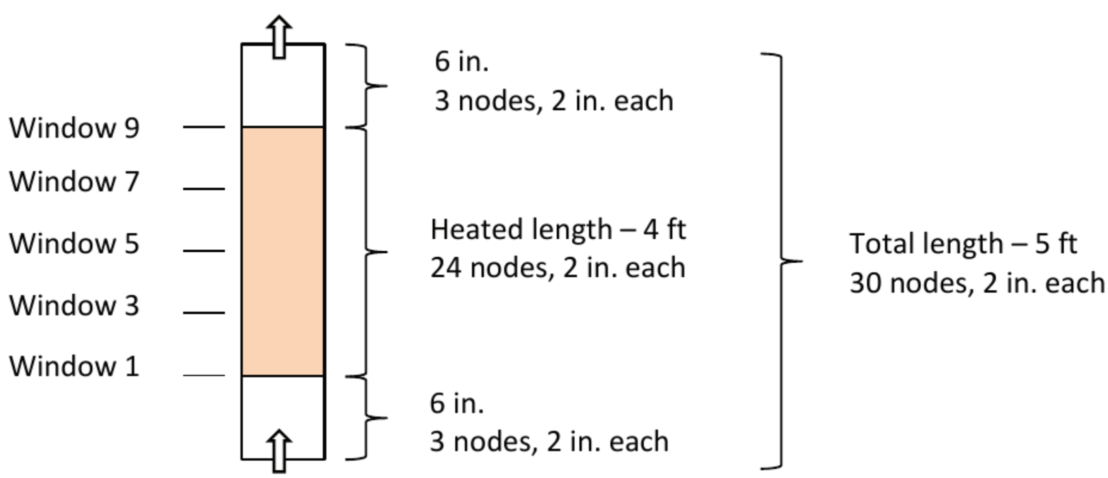

Figure 7. Axial Schematic of PNNL 2x6 LDA Measurement Windows and CTF Meshing [12].

The CTF predicted velocities in the middle row of the subchannels, Channels 8 through 14, were compared with the LDA measured values of the local velocity corresponding to Y=0.0 at different elevations in Figure 8. The CTF predicted flow velocities captured the correct trends of the flow changes due to the power gradient at different axial levels. The CTF channel-averaged velocity values showed smaller variations and were lower at the higher elevations as compared to the LDA measurements. The CTF predicted fluid temperatures were also compared to the thermocouple measurements at different elevations in Figure 8. The CTF channel-averaged temperature values again showed the correct trends of the temperature changes, although the predicted values did not account for the effect of the housing wall on the thermocouple measurements. In general, the current CTF predictive capability under the low flow and natural circulation with the high power gradient is similar to other subchannel codes such as THINC-IV and VIPRE-01 used in the reactor design applications [13,14].

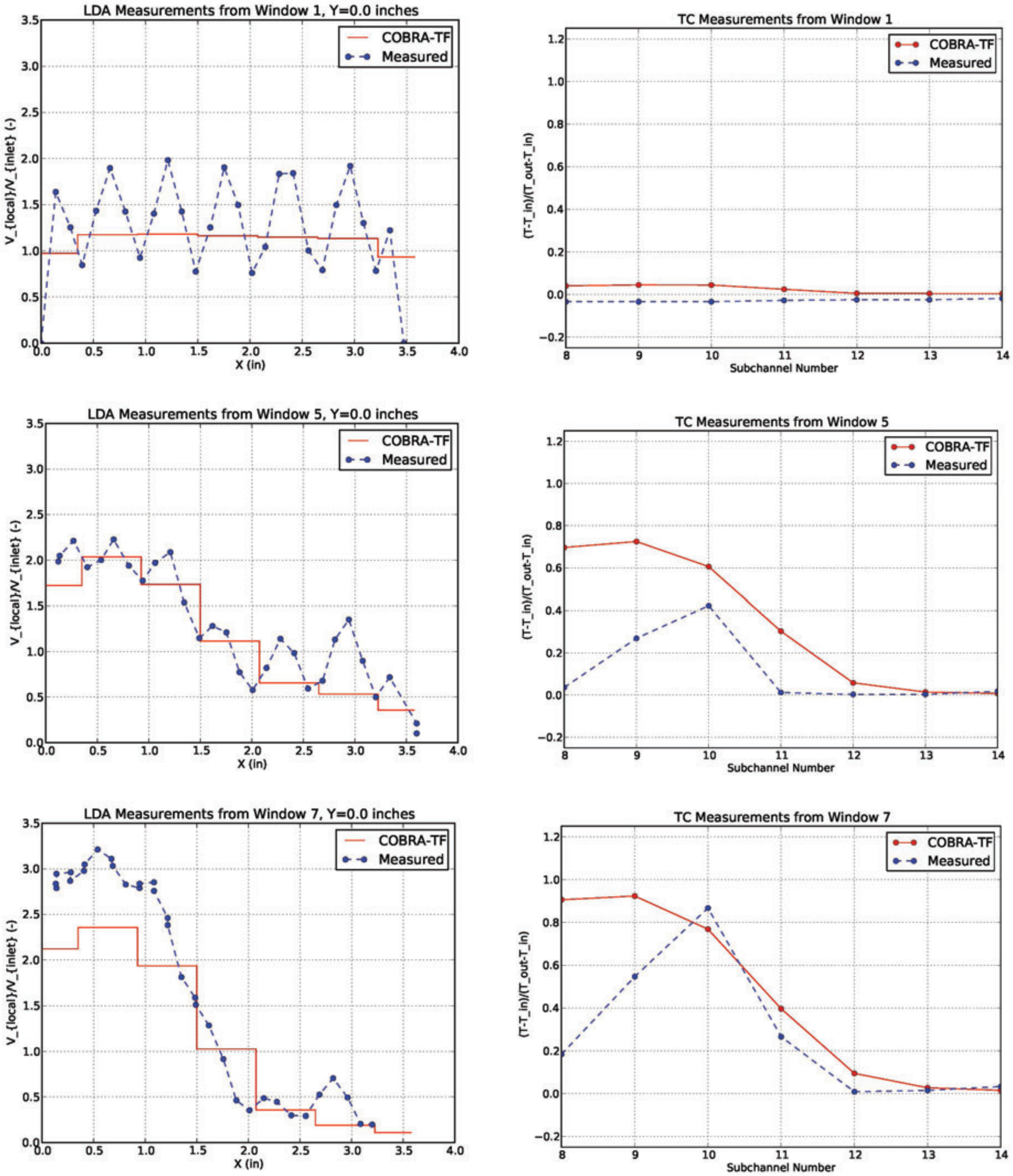

Figure 8. CTF Predicted Velocities and Temperatures Compared to PNNL Test Data.

3. Full Core SLB Modeling Assessment

The CTF reactor core modeling capability was evaluated for a Westinghouse-designed 3-loop PWR under a postulated main steamline break condition. The reactor core consisted of one hundred and fifty-seven 17x17 fuel assemblies with a fuel rod outside diameter of 0.374 inch (9.5 mm). In each fuel assembly, there were 264 fuel rods, 24 guide thimble tubes and one instrument tube with an outsider diameter of 0.482 inch (12.24 mm). The assembly and rod pitches were 8.466 inches (21.5 cm) and 0.496 inch (12.6 mm), respectively. There were eleven grid spacers including six mixing-vane (MV) grid spacers and three intermediate flow mixer (IFM) grid spacers. The active fuel length was 144 inches (3.66 m).

In the plant safety analysis, the rupture of the main steam line in one of the three loops resulted in a highly asymmetric inlet coolant temperature and power distribution in the core, because the most reactive rod cluster control assembly (RCCA) was assumed in its stuck position coincident with the same region affected by the loop with steamline break. Furthermore, it was assumed that the reactor coolant pumps (RCPs) were shut down due to a loss of offsite power. Because of its relatively long period of the transient from its initiation to the minimum DNBR occurrence, the CTF simulation was performed in a quasi-steady state mode at the DNB limiting time step as determined from a system transient calculation. The following reactor core boundary condition was input to the CTF calculation:

Pressure, psia (MPa)	-	580 (4.0)
Flow Rate, lbm/s (kg/s)	-	3354 (1523) (~12% of nominal flow)
Core Average Power (MW)	-	353 (~10% of nominal power)

The core inlet temperature distribution at the MSLB DNB limiting time step is shown in Figure 9. The core and assembly radial power distributions and the axial power profile are shown in Figure 10. As seen from the figure, the core power distribution was highly asymmetric; the high power assemblies were clustered in the broken loop section of the core. The hot assembly power factor was about 7 and the hot rod power factor was about 1.3.

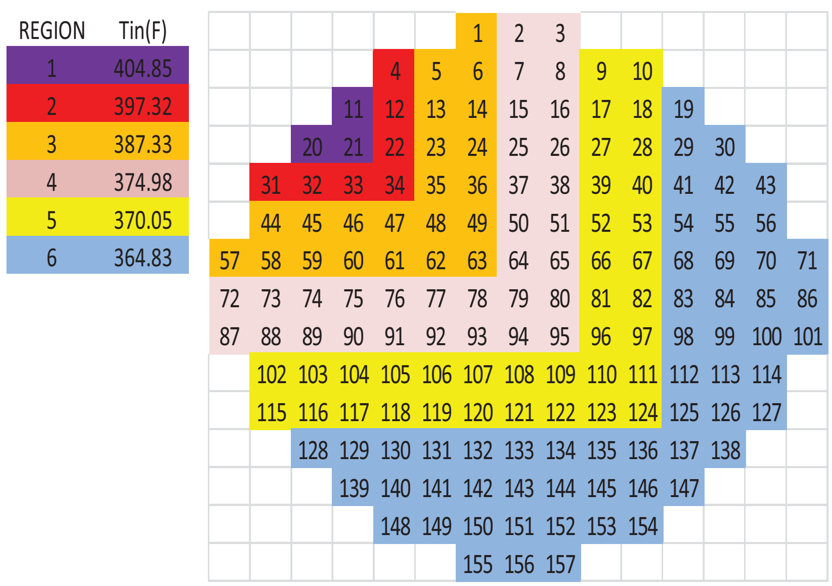

Figure 9. Reactor Core Inlet Temperature Distribution at MSLB DNB Limiting Time Step.

The CTF full 3-loop core subchannel model, constructed using the newly developed preprocessor, consisted of 45,884 flow channels, 91,256 rod-to-rod gaps, 41,448 fuel rods, and 3,925 guide thimble/instrument tubes. The two-phase flow, boiling and mixing modeling options were consistent with those used for the rod bundle data analyses. The CTF calculation was performed on a high performance multiprocessor computer platform at the Oak Ridge National Laboratory (ORNL) with 157 computing nodes allocated. The total wall-clock time to achieve solution convergence was less than 5 hours.

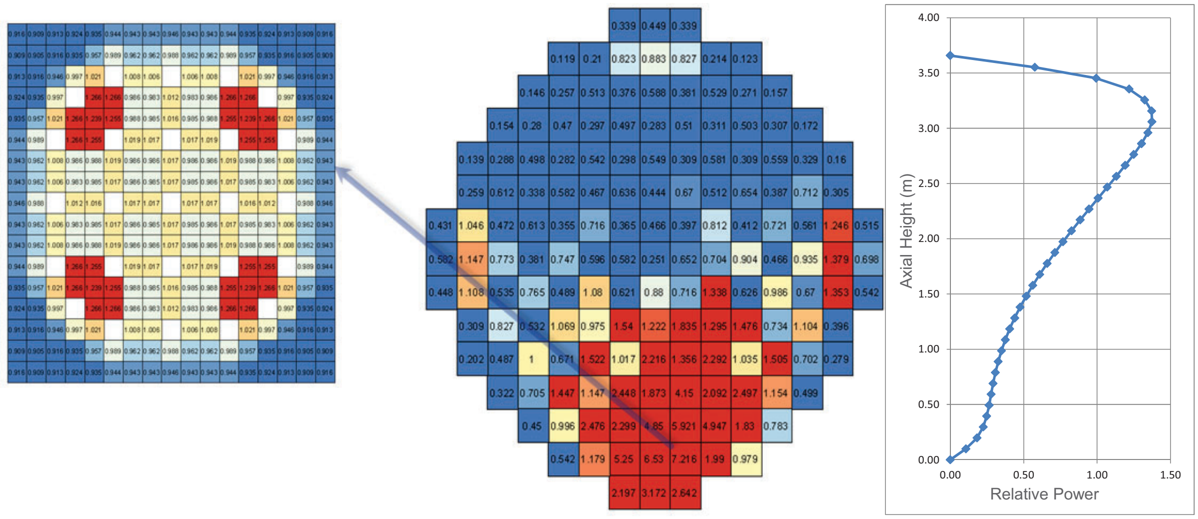

Figure 10. Reactor Core Power Distributions at MLSB DNB Limiting Time Step.

Figure 11 shows 3D plots of coolant mass flow and temperature distributions in the reactor core from the CTF output processed using a visualization software. The CTF results accounted for the effect of increased flow in the hot channel due to the phenomenon of "thermal siphoning," or the "chimney effect." Under the low flow natural circulation condition with a skewed radial power distribution, there was significant cross-flow from surrounding core region into the hot assembly and the hot channel. The axial flow significantly increased from the channel inlet to the upper regions of the hot assembly. Figure 12 shows the liquid and vapor mass flow rates as a function of axial height. The axial flow rate increased towards the top of the hot channel where the boiling occurred. Figure 12 also shows the hot channel and hot rod temperature distributions along the heated length. The liquid temperature reached saturation at around 90 inches (2.29m) from the channel inlet. Without any neutronic power adjustment due to the reactivity feedback in the stand-alone CTF calculation, the low flow rate at the core inlet resulted in a rapid increase in enthalpy rise in the high powered assembly. Clad inner and outer surface temperatures also increased rapidly with varying slopes, depending on the heat transfer and flow regimes. The 3D distribution of the mixture temperatures in Figure 11 indicates the enthalpy rise in the high powered region and a large section of the core remained below the saturation temperature. Overall, the CTF results of the MSLB full core simulation are consistent with the previous analysis using Westinghouse subchannel code THINC-IV [14], but CTF is capable of predicting the fluid conditions in each subchannel of the core for the DNB analysis. Such capability will enable the more detailed DNBR margin prediction at different locations of the reactor core when CTF is coupled with a neutronic code.

4. CONCLUSIONS

Under the CASL development program, the CTF code is used for predicting the core T/H responses for both reactivity feedback and margin to DNB during a PWR MSLB accident, coupled with a neutron transport code and linked with a system transient code. The evaluation of the CTF code capability showed good agreements with the experimental data from rod bundle tests simulating PWR fuel and SLB conditions, and the CTF predictions are comparable to results of other subchannel codes used in the industry. The CTF modeling capability was further evaluated using a full reactor core subchannel model to predict the fluid and fuel rod conditions at the DNB-limiting time step of a PWR SLB case without offsite power on the ORNL high-performance computing platform. Results of the capability evaluation indicate that CTF could potentially be applied with a proper DNB correlation as the more robust tool for the PWR accident DNB analysis, although continuous improvements are needed for such design application. Future work under the CASL program includes a coupled VERA-CS neutronic and CTF T/H modeling and simulation of the SLB core responses with respect to DNB. Also, a viable approach is being developed to the solution verification, validation and uncertainty quantification (VVUQ) for the intended application.

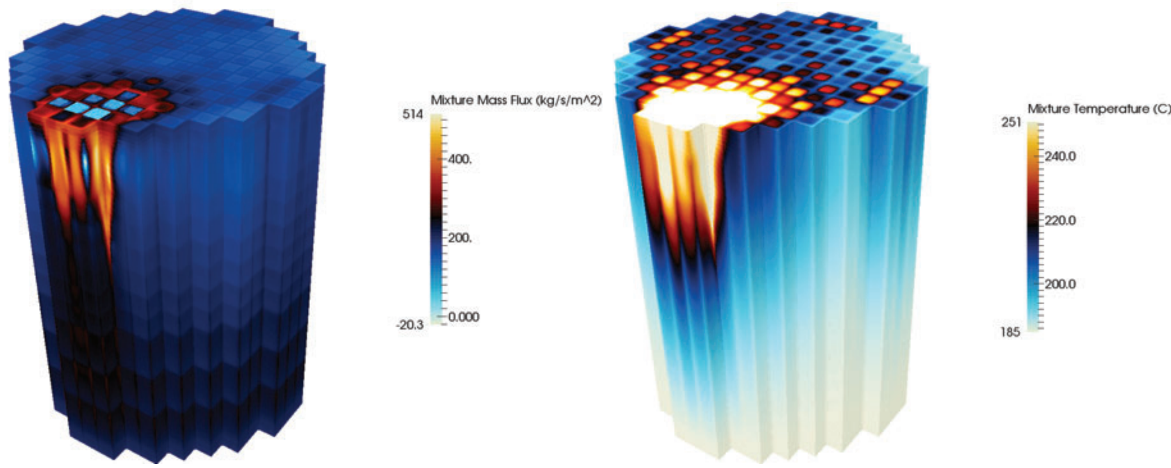

Figure 11. 3D Plots of CTF-Predicted Core Mass Flow and Temperature Distributions at MSLB DNB Limiting Time Step.

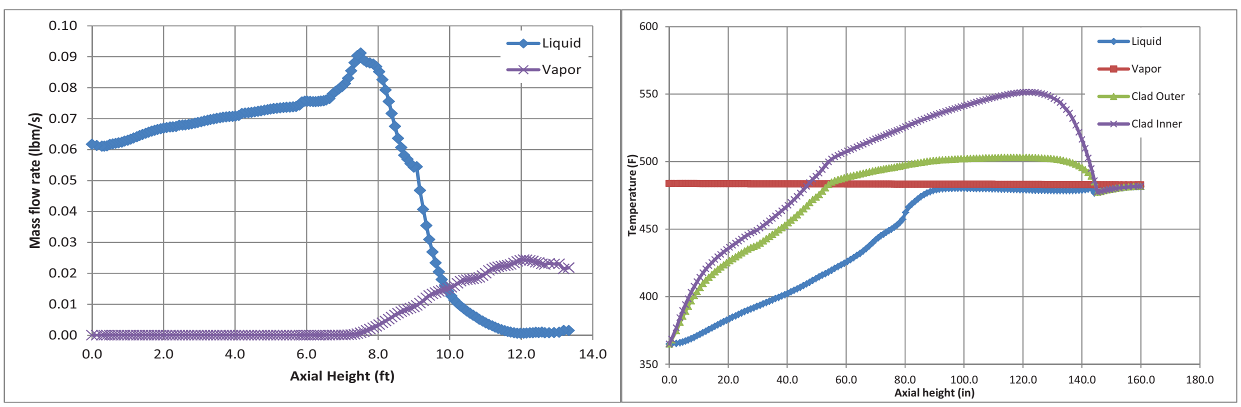

Figure 12. CTF-Predicted Mass Flow and Temperature Distributions in Hot Channel at MSLB DNB Limiting Time Step.

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