Assessment of the Subchannel Code CTF for Single- and Two-Phase Flows

As Published: 10.1080/00295450.2018.1507221

Publisher: Informa UK Limited

Persistent URL: https://hdl.handle.net/1721.1/136127

Version: Author's final manuscript: final author's manuscript post peer review, without publisher's formatting or copy editing

Assessment of the Subchannel Code CTF for Single- and Two-Phase Flows

Xingang Zhao,[1*] Aaron J. Wysocki,[2] Koroush Shirvan,[1] and Robert K. Salko[2**]

[1]Department of Nuclear Science and Engineering, Massachusetts Institute of Technology, 77 Massachusetts Ave, Cambridge, MA 02139, USA

[2]Oak Ridge National Laboratory, P.O. Box 2008, Oak Ridge, TN 37831, USA

[*]Primary author (Email: xgzhao@mit.edu)

[**]Corresponding author (Email: salkork@ornl.gov)

# of pages: 43

# of tables: 4

# of figures: 14

Abstract

As part of the Consortium for Advanced Simulation of Light Water Reactors (CASL), the subchannel code CTF is being used for single- and two-phase flow analysis under light water reactor operating conditions. Accurate determination of flow distribution, pressure drop, and void content is crucial for predicting margins to thermal crisis and ensuring more efficient plant performance. In preparation for the intended applications, CTF has been validated against data from experimental facilities comprising the General Electric (GE) 3 × 3 bundle, the boiling water reactor (BWR) full-size fine-mesh bundle tests (BFBTs), the RISO tube, and the pressurized water reactor (PWR) subchannel and bundle tests (PSBTs). Meanwhile, the licensed, well-recognized subchannel code VIPRE-01 was used to generate a baseline set of simulations for the targeted tests, and solution parameters were compared to CTF results.

The flow split verification problem and single-phase GE 3 × 3 results are essentially in perfect agreement between the two codes. For two-phase GE 3 × 3 cases, flow and quality discrepancies arise in the annular-mist flow regime, yet significant improvement is observed in CTF when void drift and two-phase turbulent mixing enhancement are considered. The BFBT pressure drop benchmark shows close agreement between predicted and measured results in general, although considerable over-prediction by CTF is observed at relatively high void locations of the facility. This over-estimation tendency is confirmed by RISO cases. While overall statistics are satisfactory, both BFBT and PSBT bubbly-to-churn flow void contents are markedly over-predicted by CTF.

The issues with two-phase closures such as turbulent mixing, interfacial and wall friction, and subcooled boiling heat transfer need to be addressed. Preliminary sensitivity studies are presented herein, but more advanced models and code stability analysis require further investigation.

Keywords: CTF, subchannel analysis, flow mixing, pressure drop, void fraction.

I. INTRODUCTION

Subchannel analysis has been used extensively to evaluate core-wide thermal-fluid behavior in light water reactors (LWRs). It provides a higher degree of physical modeling and more detailed local information than lumped analysis approaches. Although it does not offer resolution as fine as a high-fidelity computational fluid dynamics (CFD) solver, subchannel analysis allows for a relatively fast, reasonable estimation of the flow, enthalpy, and void distributions. Jointly developed and maintained by Oak Ridge National Laboratory and North Carolina State University, CTF is an improved version of the legacy subchannel code COBRA-TF[1] adopted for use in the Consortium for Advanced Simulation of Light Water Reactors (CASL) for aiding in addressing its challenge problems.[2] CASL is the first innovation hub initiated in 2010 by the US Department of Energy (DOE) to bridge research, engineering, and industry.[a] Since then, activities related to CTF have included implementing software quality assurance measures, implementing new features and models, performing validation and verification testing, establishing and supporting a CTF user group, and developing the code for use in coupled applications.[3] CTF is applicable to single- and two-phase flows in LWR geometries at normal and accident operating conditions.

Development of CTF in CASL has been focused on improving the code for normal reactor operating conditions in both pressurized water reactors (PWRs) and boiling water reactors (BWRs), as well as for departure for nucleate boiling (DNB) margin analysis and crud-induced power shift (CIPS). CTF relies on a two-fluid, three-field (liquid film, entrained droplets, and vapor core) modeling approach.[4] It solves discretized forms of six equations for mass, momentum, and energy conservation for the liquid film and vapor fields, mass and

[a] CASL website: www.casl.gov

momentum equations for the droplet field, and an energy equation for solid structures, for a total of nine equations.

Accurate determination of flow distribution, pressure drop, and void content is crucial for predicting margins to thermal crisis (i.e., DNB for PWRs) and ensuring more efficient plant performance. In preparation for the intended applications, CTF has been validated[5] against data from experimental facilities comprising the General Electric (GE) 3 × 3 bundle,[6] the BWR full-size fine-mesh bundle tests (BFBTs),[7] the RISO tube,[8] and the PWR subchannel and bundle tests (PSBTs).[9]

To address issues with limited experimental data and their ineluctable uncertainties, code-to-code verification and benchmarking provide an effective and reliable approach to assess the two-phase capabilities of CTF. The well-established subchannel code VIPRE-01, licensed by the US Nuclear Regulatory Commission (NRC), was selected to generate a baseline set of simulations for the targeted tests. The solution parameters are compared to CTF results. Originally developed to help evaluate LWR core T/H parameters and safety limits in normal and assumed accident scenarios, the up-to-date MIT version of VIPRE-01 was benchmarked for both PWRs and BWRs[10] and has been selected for this work. Unlike CTF, VIPRE-01 solves conservation equations (3-equation model) by assuming incompressible thermally expandable mixture flow with empirical models, accounting for vapor/liquid slip in two-phase flow.

This paper presents CTF validation and benchmarking work, while highlighting and addressing the closure terms in the greatest need of improvement that will drive future CASL development activities. Flow mixing models are assessed via single-phase (including a two-channel flow split verification problem and single-phase GE 3 × 3) and two-phase (GE 3 × 3) flow cases. Single- and two-phase BFBT pressure drop tests are simulated, as well as RISO tube cases, to illustrate CTF's capability to predict pressure gradients. Finally, the focus is shifted to predicted void fraction results against BFBT and PSBT void data.

II. FLOW MIXING PREDICTION ASSESSMENT

II.A. Two-Channel Single-Phase Flow Split Problem

This preliminary verification problem consists of two unheated channels connected by a gap of 3.1 mm wide, corresponding to the typical gap value of a PWR channel. The goal was to verify the single-phase flow redistribution length in CTF. The two channels have the same boundary conditions and differ in flow area only, which leads to a lateral pressure gradient: channel 2 had a hydraulic diameter twice that of channel 1 (see Fig. 1). The turbulent mixing (inter-channel natural flow mixing due to turbulence, usually assumed to cause a lateral enthalpy flux without any associated net mass-flow[11]) model was disabled so that wall friction became the only cross-flow driver. The McAdam's friction correlation[4] is used here and throughout this paper. An analytical solution was derived to determine the theoretical flow split considering the two channels to be in mechanical equilibrium (at which point the frictional pressure drop is the same in both channels and the cross-flow ceases). The CTF vs. VIPRE-01 (and analytical equilibrium) axial flow profiles are compared in Fig. 2. The normalized mass flux is defined as the relative difference between local and inlet mass flux. As can be seen, the agreement is essentially perfect between the two codes: both predict the ideal flow split at ~7 m from the inlet. Salko et al.[12] addressed the concern of the flow redistribution length being excessively long for this verification case, yet this result showed that the prediction by CTF is well in line with a licensed subchannel tool.

II.B. GE 3 × 3 Benchmark

The GE 3 × 3 is a classic BWR-like test facility that is used for assessing interchannel flow mixing since exit mass flux and quality measurements were performed for individual channel types (inner, side, and corner). Detailed bundle geometry (Fig. 3) and operating conditions are provided in the original Lahey et al. report.[6] The bundle was uniformly heated both axially and radially (for diabatic tests); power level, mass flow rate, and inlet subcooling varied from case to case (4 test points for single-phase flow and 13 for two-phase flow). Six sets of pin type spacer grids were employed to hold the rods.

II.B.1. Single-phase flow benchmark

This study was conducted to evaluate the capability of CTF to correctly predict single-phase flow distribution. The four unheated cases were simulated with CTF and VIPRE-01. Mass flow rate is the only parameter that varies (increases) from case 1B to 1E. Predicted exit mass fluxes are compared against measurements in Fig. 4. The turbulent mixing (TM) cross-flow model in both codes is a simple turbulent diffusion approximation for which the user is required to enter the single-phase mixing coefficient (β). A constant value of $\beta = 0.007$ is used here (and throughout this paper unless otherwise specified) in accordance with previous assessments[13] using the Kumamoto University 2 × 3 facility data.[14] It is noticeable that VIPRE-01 and CTF solutions substantially agree with each other, and both closely match experimental results for inner and side channels, having the root-mean-square of relative error (rRMSE) values very close to or less than the measurement uncertainty (± 2%). Corner channel predictions deviated more from measured data (rRMSE is close to 10% for both codes).

II.B.2. Two-phase flow benchmark

Based on the single-phase flow benchmark, this study evaluated the impact of unique CTF modeling features comprising the two-phase TM enhancement and void drift (VD, introduced to describe the tendency of a vapor phase within liquid phase along inter-connected channels to redistribute itself according to a certain equilibrium void distribution[11]) mechanism. VD models are not currently available in VIPRE-01, and the code user manual[15] suggests that the same TM model used in single-phase flow be used in two-phase flow for lack of a thorough understanding of the underlying physics. Unlike VIPRE-01, CTF includes the Lahey–Moody model[16] for estimating VD,[a] in which the default scaling weight factor $K_a = 1.4$[b] is recommended.[4] In addition, experimental evidence showed higher TM rate in two-phase flow compared to the rate in single-phase flow.[4,19] The TM rate was found to first increase with increasing steam quality (due to the increased turbulent fluctuations as a result of the liquid-bubble interaction) until reaching its maximum around the slug-annular regime transition point and then decrease dramatically to the level of single-phase flow TM in annular-mist flow. Faya et al.[20] modeled this enhancement effect with a two-phase multiplier Θ_M based on the Beus model,[21] and this parameter was implemented in CTF for this work. Its maximum value was set to be equal to 5.0.[4]

A total of 13 two-phase flow measurement points made publicly available are compared with simulation results. Initially, cases were run with VD disabled and a constant TM coefficient

---

[a] The Lahey-Moody VD model cannot be implemented in VIPRE-01 directly because the model can only be applied with an equal-volume TM model (whilst VIPRE-01 uses an equal-mass TM model). It was recently modified and made applicable to VIPRE-01, showing reasonable validation results.[17]

[b] The optimal (default) scaling factor (K_a) was picked using the "view-graph norm" by fitting data from three experiments including the GE 3 × 3. It was also found[18] that the calculated results had little sensitivity to K_a as long as this parameter was in the range from 1.2 to 1.6.

($\beta = 0.007$) in CTF (no two-phase enhancement). The same grid loss coefficients as provided in the original reference[6] were applied in both codes.[13] Fig. 5a shows that under this configuration, CTF and VIPRE-01 mass fluxes were visually and statistically similar, closely matching experimental data for inner and side channels (rRMSE ≈ 5%), although with larger scatter than single-phase flow. The corner channel was poorly predicted (rRMSE > 20%), as most data points were substantially underestimated, implying an over-prediction of void content in this channel type. Discrepancies between the two codes and with measured results became more dramatic when the flow at the bundle outlet was in the annular-mist regime.[a]

When both the two-phase TM enhancement and VD were taken into account, CTF solutions were significantly improved and agreed better with measurements (see Fig. 5b), in particular for corner channels (rRMSE value is half of that with the noVD configuration). However, most corner mass fluxes were still considerably underpredicted, requiring search and implementation of more robust (e.g., flow regime–dependent) TM and VD models. Furthermore, the corner channel hydraulic diameter (~7 mm) was significantly smaller than a typical BWR bundle hydraulic diameter (11–12 mm), hinged on which standard interchannel flow mixing models were developed. Therefore, models derived from the data of a tight lattice bundle (or a small diameter tube) are deemed better qualified for this channel type.

The comparison in Fig. 5b is also performed by employing a Reynolds number–related TM (ReTM) model in VIPRE-01, as recommended by Brynjell-Rahkola et al.[22] ReTM was included in VIPRE-W (Westinghouse version of VIPRE-01), and the code package was validated against BFBT data. Figs. 5a and 5b show that while VIPRE-01 results with ReTM slightly outperformed

[a] according to the void-based flow regime map in CTF[4]

those with constant TM (cTM), ReTM on its own was unable to improve corner predictions to a more significant extent for these cases.

Similar results were generated for the exit quality inside different channel types with an inverse trend as compared to mass flux comparison: corner channel quality tended to be over-estimated by both codes.

III. PRESSURE DROP PREDICTION ASSESSMENT

III.A. BFBT Pressure Drop Benchmark

These experiments were conducted by the Nuclear Power Engineering Corporation (NUPEC) at the BFBT facility in Japan, where a BWR-type 8 × 8 fuel assembly design was adopted. Benchmark specifications are documented in Neykov et al.[7] Tests from Phase II (critical power benchmark) Exercise 0 (steady-state pressure drop) of the BFBT were modeled for this study. They covered both single-phase (series P7, 10 published points) and two-phase (series P6, 22 published points) pressure drop measurements. The assembly type C2A, having a non-uniform axial (cosine shape) and radial power distribution, was used for all test cases. Its cross-sectional view among other assembly types, as specified in Section IV.A, is depicted in Fig. 6, showing 60 heater rods and 1 large central guide tube with no water. Pressure tap locations are shown in Fig. 7.

III.A.1. Single-phase flow benchmark

In an unheated vertical upflow bundle, the pressure drop comprises wall friction, gravity, and form loss (introduced by the presence of spacer grids). Note that the reported BFBT experimental results did not account for gravitational pressure loss; therefore, this term was subtracted from the predicted total loss for different pressure tap locations. This study assessed the wall friction (McAdam's correlation[4]) and grid form loss (Shiralkar and Radcliffe's approach[13]) models in both codes. Fig. 8 shows that the predicted and measured pressure gradients agreed well (VIPRE-01 slightly outperforming CTF), although most pressure taps were underpredicted except for T1 and T3. The relative discrepancy between measured and predicted results (i.e., rRMSE) for dpT9, which covered the entire bundle with 7 grids, was smaller than all tap data. The taps with the largest sources of error were T2, T5, T6 (1 grid span) and T7 (3 grid spans). The reported measurement error for bundle-averaged pressure drop is 1%, and predictions fell outside this uncertainty range. Sensitivity studies on single-phase friction and form loss correlations may help further improve the statistics.

III.A.2. Two-phase flow benchmark

The traditional means of modeling the effect of two-phase flow on the frictional pressure loss is the use of a two-phase friction multiplier. In VIPRE-01, the default and recommended Electric Power Research Institute (EPRI) correlation was selected for this study. The EPRI correlation was based on the analytical homogeneous model and incorporated mass flux dependence observed in some adiabatic steam-water vertical upflow data.[10,23] In CTF, the Wallis multiplier, defined as one over liquid volumetric fraction squared, was applied on the liquid wall drag.[24] Form loss due to spacer grids was calculated with the Romie multiplier[10] in VIPRE-01, while it was computed separately for each phase in CTF.[4]

Table I summarizes pressure drop rRMSEs using VIPRE-01 and CTF for all nine pressure taps. Results were much more scattered than single-phase cases (Fig. 8). Overall, the two codes resulted in similar statistics, while CTF performed slightly better. However, discrepancies at certain tap locations were significant. The largest disagreement with experiments was from taps T1–4, which were located at the upper part of the bundle (i.e., relatively high void). Separate tap pressure drops were rearranged, and the two most representative comparisons are plotted in Fig. 9, with Fig. 9a presenting entire bundle (T9), and Fig. 9b illustrating the bare (no grid) upper part bundle section (T3-T1). On one hand, VIPRE-01 consistently underestimated one or more grid spans and overestimated the bare bundle part. This observation is consistent with that presented in Brynjell-Rahkola et al.[22] and Le Corre et al.[25] On the other hand, CTF tended to overpredict two-phase pressure drop, in particular for tap T3 and the two bare bundle sections—T3-T1 (as shown in Fig. 9b) and T4-T2—which were severely overestimated by CTF. The two-phase friction multiplier and the liquid-vapor interfacial drag model were called into question and are addressed in the next subsection.

III.B. RISO Round Tube Benchmark

The RISO experimental facility[8] enabled study of upward flow through a vertical cylindrical pipe, either unheated or with a constant heat flux applied over a specified section of the pipe. Previous work performed at CASL[26] concluded that the effect of heating and tube diameter on CTF's pressure drop overprediction was negligible. Hence, only the adiabatic (preheated to a desired constant thermodynamic quality throughout the pipe) round tube test section RISO #10 (tube inner diameter = 10 mm, length = 9 m, exit pressure = 7 MPa) was selected for this work to serve as a separate effect benchmark. Impacts of flow mixing and grids were eliminated, and most cases fell within the annular-mist flow regime. A wide range of mass fluxes (from 500 up to 3,000 kg/m^2-s) was covered by a total of 26 data points. Pressure gradients were measured by electric differential pressure transducers with a claimed accuracy of ± 0.5 kPa.

Fig. 10 confirms the dramatic overprediction tendency of CTF found in the BFBT bare bundle section. Moreover, as quality (void fraction) increases, such overprediction becomes more substantial. It is important to note that VIPRE-01 also predicts higher-than-measured pressure gradients, but to a much smaller extent than CTF (at high quality in particular).

The void-based Wallis wall friction multiplier used in CTF was derived from horizontal laminar annular flow[24] and showed reasonable approximation for turbulent annular flow. However, no assessment was available for vertical flow, and no formulation for other flow regimes was provided in Wallis.[24] In the TRACE[a] V5.0 theory manual,[27] the Wallis multiplier is validated against and agrees well with vertical air-water data when the void fraction is higher than 82%. Another set of data covering void fraction ranging from 10 to 90%—the adiabatic steam-water data of Ferrel-McGee—was used in the TRACE V5.0 theory manual,[27] and a modified multiplier was derived to better fit measurements. As compared to Wallis, the modified formulation has a lower exponent (1.72 vs. 2) in the denominator and has been applied in TRACE for the bubbly/slug flow regime. The RISO #10 high void content (> 80%) cases were recently modeled in TRACE,[28] and close agreement was found between CTF and TRACE, significantly overpredicting pressure drop in annular flow where both codes use the Wallis multiplier. Besides the two-phase wall friction multiplier, pressure drop overprediction may also result from the use of an inadequate interfacial drag model. In the current version of CTF,[4] the interfacial drag in annular-mist flow regime depends on the nature of the liquid film (stable or unstable). An unstable film will have large waves, which increase the pressure drop and cause a higher friction factor. The interfacial

[a] The TRACE/RELAP Advanced Computational Engine (TRACE) is the latest in a series of advanced best-estimate reactor systems codes developed by the NRC for analyzing steady-state and transient neutronic-thermal-hydraulic behaviors in LWRs.

friction factor is taken as the maximum of the unstable film friction factor (Henstock & Hanratty model) and five times the stable film friction factor (Wallis model). Such a criterion would not only trigger discontinuity at the stable–unstable film boundary, but also would tend to overestimate interfacial drag and therefore pressure drop. As illustrated in Fig. 11, by disabling the Henstock & Hanratty model (which yields higher friction and questionable applicability[a]) and switching the denominator exponent of the Wallis wall friction multiplier from 2 to 1.72 (denoted as *CTF-new* in Fig. 11), the predicted pressure drop's agreement with data was considerably improved for high void cases in which stable liquid film was established. However, it has been shown[28] that applying the Wallis interfacial friction model only would lead to a significant overprediction of the film flow fraction.

A more physics-based interfacial drag modeling package is being implemented in CTF. Based on Lane's approach,[29] this package relies on local closure terms such as film thickness (in the case of annular flow), a two-zone interfacial friction factor, an entrainment inception point, an entrainment suppression point (serving as demarcation between stable and unstable film regimes in annular flow), and modification of the transition criterion between annular-mist and churn-turbulent flow regimes. More details regarding the theory and implementation of the package can be found in Salko et al.[28] and Lane.[29] According to Figs. 28-29 of Salko et al.,[28] the accuracy of pressure drop and liquid film thickness predictions for RISO tests is clearly improved with the new package, although further improvement on flow regime detection logic and extended validation test matrix is required.

IV. VOID FRACTION PREDICTION ASSESSMENT

[a] The Henstock & Hanratty model was developed using low-pressure air-water data from 30–35 mm diameter tubes.

IV.A. BFBT Void Benchmark

Tests from Phase I (void distribution benchmark) Exercise 1 (steady-state subchannel grade benchmark) of BFBT were modeled for this study. These tests measured 15 sets of published void data referring to five assembly types (0-1, 0-2, 0-3, 1 and 4) as presented in Fig. 6. While assembly types 0-1, 0-2, and 0-3 had uniform axial and radial heating, they differed in that some heater rods were shut off in assembly types 0-2 (2 non-heated pins) and 0-3 (7 non-heated pins). Assembly type 1 employed the same geometry as assembly type 0-1, but it applied a non-uniform axial (cosine shape) and radial power profile. Assembly type 4 (uniform axial power distribution) had the same geometry as assembly type C2A, which was used in Section III.A for pressure drop benchmark.

Table II recapitulates VIPRE-01 and CTF outlet void solutions in terms of mean error (ME = measured - predicted) and standard deviation of error (σ) for all data, as well as for each assembly and channel type (corner, side, inner, near non-heated [nNH], in non-heated [iNH]). Both codes applied the same wall friction correlation (modified McAdam's) and form loss coefficient (0.94) as recommended in VIPRE-W[22] since the latter code fits data within the reported experimental uncertainty (± 0.02). It can be concluded that VIPRE-01 generally outperformed CTF and matched measurements closely, just like VIPRE-W. Bundle-averaged data were slightly overestimated by both codes, as well as individual channel types, except for corners (which were underestimated by CTF). For inner channels and channels surrounded by one or more unheated pins, CTF behaved poorly as compared to VIPRE-01, especially in bubbly and slug flow where CTF overprediction became more remarkable; and for assembly types 0-2 and 0-3 in which large enthalpy and mass flux gradients were expected around non-heated rods. For these cases, CTF seemed to struggle more with non-peripheral channels when void drift was enabled. Further analyses on interfacial drag modeling in small and small-to-large bubble flow regimes are necessary.

IV.B. PSBT Void Benchmark

The Nuclear Power Engineering Corporation (NUPEC) PWR subchannel and bundle tests (PSBTs) data were released following the success of the BFBT benchmark. Specifications are detailed in Rubin et al.[9] Phase I (void distribution benchmark) steady-state test cases were selected for this study. Data from both Exercise 1 (single subchannel) and Exercise 2 (5 × 5 bundle) were modeled. Reported void measurement uncertainties were 3% for a single subchannel[a] and 4% for a region-averaged bundle.

IV.B.1. Single subchannel benchmark

The object was to evaluate void modeling using CTF with no bias from inter-channel flow mixing and spacer grids. Four uniformly heated channel types are included in this benchmark: corner, side, inner, and inner next to a guide tube (denoted as test series S1–S4, respectively). Channel length is equal to 1.555 m. Outlet pressure ranges from 4.9 to 16.6 MPa, mass flux from

---

[a] The masking effect of CT measurement technique on subchannel void fraction was studied by the Organisation for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA).[30] Due to low fidelity of the current CT scan technology, some near-wall void is masked, leading to lower experimental void fraction than is reality. Depending on the assumptions, the CT scanner will not show any void until the actual void content is 3.8–7.8%. Along with concerns on shifting of the CT image, it is therefore suggested[30] that a total experimental uncertainty of ~6.2% would be deemed more reasonable.

494 to 3,072 kg/m^2-s, and experimental channel average void (at 1.4 m from the channel bottom where void fraction was measured by means of CT scanner) from 0.003 to 0.83.

The Organisation for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) released benchmark results of PSBT Phase I[30] in which various existing subchannel, system, and CFD codes were evaluated, including VIPRE-01 by Computer Simulation & Analysis, Inc. (CSA, current code custodian for VIPRE-01, recently acquired by Zachry Group) and F-COBRA-TF by AREVA (French version of COBRA-TF). Void results extracted from NEA's work, along with replicated VIPRE-01 and CTF solutions, were post-processed and plotted in Fig. 12. Replicated VIPRE-01 uses closure models consistent with CSA instructions.[30] Note that 15 out of 43 CTF cases struggled to fully converge, regardless of axial noding, conduction model, and channel geometry options. Fig. 12 shows that CSA and replicated VIPRE-01 results agreed closely with each other. CTF yielded higher void as compared to other codes, and such overestimation was even more dramatic for test series S1 and S3. In VIPRE-01, the subcooled steam quality was calculated using a profile-fit model (Levy or EPRI[10]), and the effect of the phase slip was accounted for in the void model (drift-flux by default). In CTF, a different methodology was adopted in which bubbles were assumed to exist so long as the wall surface temperature exceeded the bulk saturation temperature, and it relied on the near-wall condensation model of Hancox–Nicoll[31] to recondense away the generated vapor during subcooled boiling.

To address the concern of void overprediction, a sensitivity study was conducted. In CTF, the calculation of heat flux used for net vapor generation (q''_{SCVG}) and related heat transfer coefficients required further assessment. The term q''_{SCVG} is basically the heat flux that results in vapor generation due to nucleate boiling (q''_{NB}) subtracted by the heat flux available for vapor condensation (q''_{cond}). Table III lists three different ways of interpreting the latter two terms in CTF (including the baseline version) and their respective performances with regard to PSBT single subchannel cases. Special attention should be paid when the Thom correlation[32] is used for calculation of nucleate boiling heat flux (and heat transfer coefficient) in subcooled boiling region,[a] as it seems to be more appropriate to subtract off the convective component of heat transfer (versions #2 and #3 in Table III, also consistent with VIPRE-01 if the EPRI drift-flux model is selected). As for the vapor condensation component, whether or not the convective heat flux should be further subtracted from the heat flux calculated from the Hancox–Nicoll correlation is still debatable.[b] Based on results presented in Table III and Fig. 13, version #3 led to the closest agreement with data, and all 43 test points converged. More validation work and code stability analysis will be performed.

IV.B.2. Rod bundle benchmark

The PSBT 5 × 5 rod bundle test consisted of three assembly types: S5 (uniform axial heating), S6 (cosine shape axial heating), and S7 (cosine shape axial heating with one central thimble rod). Gamma-ray transmission measurements were made at three axial locations: the lower at 2,216 mm, the middle at 2,669 mm, and the upper at 3,177 mm. Seventeen spacer grids were used on each bundle. CSA VIPRE-01 solutions were collected from the NEA report[30], and

---

[a] The Chen correlation has already taken into account both nucleate boiling and convective heat transfer components in its original formulation.[33]

[b] The key here is the interpretation of the Hancox–Nicoll correlation. The authors of the original paper focused on heat transfer at the point of net vapor generation (NVG) or bubble departure from the wall, and this could be translated into condensation heat transfer in the case of subcooled boiling. This correlation is also used in the EPRI drift-flux model in VIPRE-01.[10]

predicted vs. measured region-averaged (average of four central subchannels) void fraction results for each test series are presented in Fig. 14. While most high void (> 0.5) data points fell within measurement uncertainty (± 4%), both codes tended to overpredict bubbly/slug flow void content.

V. CONCLUSIONS

This paper assessed the single- and two-phase capabilities of CTF by validating with an extended database and benchmarking with the US NRC licensed code VIPRE-01. Relevant solution parameters were compared between the two codes and against targeted measurements. A comparison of fluid flow and heat transfer key closure models between CTF and VIPRE-01 within the scope of this project is summarized in Table IV. This study found that CTF prediction of single-phase flow redistribution is consistent with results from an analytical solution and VIPRE-01. Single-phase and low-void pressure drop predictions tend to compare favorably with experimental data. Two-phase flow distribution was also well predicted with the inclusion of two-phase turbulent mixing and void drift models in the code. However, this study has revealed that CTF tends to consistently overpredict overall void content and largely overpredict two-phase pressure drop in medium-to-high void regions. These observations have led to the conclusion that the closure models in greatest need of improvement comprise:

- interfacial drag in low (bubbly/slug flow) and high (churn/annular flow) void regions;
- two-phase wall friction multiplier;
- subcooled boiling heat transfer mechanism;
- turbulent mixing and void drift in channels having a small hydraulic diameter.

A more physics-based interfacial friction modeling package is being implemented, and promising pressure drop solutions are generated. Future work will include implementing other advanced/mechanistic closures in CTF to help improve its accuracy and robustness.

Acknowledgments

This research was supported by the Consortium for Advanced Simulation of Light Water Reactors (www.casl.gov), an Energy Innovation Hub (http://www.energy.gov/hubs) for modeling and simulation of nuclear reactors under U.S. Department of Energy Contract No. DE-AC05-00OR22725.

References

1. M. THURGOOD, J. KELLY, T. GUIDOTTI, R. KOHRT, and K. CROWELL, "COBRA/TRAC - A Thermal-Hydraulics Code for Transient Analysis of Nuclear Reactor Vessels and Primary Coolant Systems Equations and Constitutive Models," NUREG/CR-3046, PNL-4385, Pacific Northwest National Laboratory (1983).
2. P. TURINSKY, D. KOTHE, and D. BURNS, "Update on Capabilities Development at CASL," *Proceedings of the International Congress on Advances in Nuclear Power Plants* (*ICAPP*), Nice, France, May 3-6 (2015).
3. R. SALKO, A. WYSOCKI, B. COLLINS, M. AVRAMOVA, and C. GOSDIN, "Development and Assessment of CTF for Pin-Resolved BWR Modeling," *Proceedings of the International Conference on Mathematics & Computational Methods Applied to Nuclear Science & Engineering* (*M&C*), Jeju, South Korea, April 16-20 (2017).
4. R. SALKO and M. AVRAMOVA, *CTF Theory Manual* (2017).
5. X. ZHAO, R. SALKO, A. WYSOCKI, and K. SHIRVAN, "Validation and Benchmarking of CTF for Single- and Two-Phase Flow," *Proceedings of the 17th International Topical Meeting on Nuclear Reactor Thermal Hydraulics* (*NURETH-17*), Xi'an, Shaanxi, China, September 3-8 (2017).
6. R. LAHEY, B. SHIRALKAR, and D. RADCLIFFE, "Two-Phase Flow and Heat Transfer in Multirod Geometries: Subchannel and Pressure Drop Measurements in a Nine-Rod Bundle for Diabatic and Adiabatic Conditions," GEAP-13049, AEC Research and Development Program (1970).
7. B. NEYKOV, F. AYDOGAN, L. HOCHREITER, K. IVANOV, H. UTSUNO, F. KASAHARA, E. SARTORI, and M. MARTIN, "NUPEC BWR Full-size Fine-mesh Bundle

Test (BFBT) Benchmark, Volume I: Specifications," NEA/NSC/DOC(2005)5, ISBN 92-64-01088-2 (2006).

8. J. WURTZ, "An Experimental and Theoretical Investigation of Annular Steam-Water Flow in Tubes and Annuli at 30 to 90 bar," Riso Report No. 372, Copenhagen, Denmark, Riso National Laboratory (1978).
9. A. RUBIN, A. SCHOEDEL, M. AVRAMOVA, H. UTSUNO, S. BAJOREK, and A. VELAZQUEZ-LOZADA, "OECD/NRC Benchmark Based on NUPEC PWR Subchannel and Bundle Tests (PSBT), Volume I: Experimental Database and Final Problem Specifications," NEA/NSC/DOC(2010)1 (2010).
10. C. STEWART, J. CUTA, S. MONTGOMERY, J. KELLY, K. BASEHORE, T. GEORGE, D. ROWE, G. GOSE, and J. WESTACOTT, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores. Volume 1: Mathematical Modeling," NP-2511-CCM-A, Volume 1, Revision 4.3 (2011).
11. X. ZHAO, "Critical Power Characteristics of Axially Heterogeneous Tight Bundle Designs," M.Sc. thesis, Massachusetts Institute of Technology (2016).
12. R. SALKO, M. GERGAR, C. GOSDIN, and M. AVRAMOVA, "CTF Void Drift Validation Study," CASL Tech. Rep., CASL-U-2015-0320-002 (2015).
13. R. SALKO, T. BLYTH, C. DANCES, J. MAGEDANZ, C. JERNIGAN, J. KELLY, A. TOPTAN, M. GERGAR, C. GOSDIN, M. AVRAMOVA, S. PALMTAG, and J. GEHIN, *CTF Validation and Verification* (2017).
14. M. SADATOMI, A. KAWAHARA, K. KANO, and Y. SUMI, "Single- and Two-Phase Turbulent Mixing Rate between Adjacent Subchannels in a Vertical 2 × 3 Rod Array Channel," *International Journal of Multiphase Flow*, **30**, 481 (2004).

15. C. STEWART, J. CUTA, S. MONTGOMERY, J. KELLY, K. BASEHORE, T. GEORGE, D. ROWE, G. GOSE, and J. WESTACOTT, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores. Volume 2: User's Manual," NP-2511-CCM-A, Volume 2, Revision 4.4 (2011).
16. R. LAHEY and F. MOODY, "The Thermal-Hydraulics of a Boiling Water Nuclear Reactor," ANS Monograph (1977).
17. J. Le CORRE, "Experimental Investigation and Modeling of Void Drift in Modern BWR Fuel Designs," *Proceedings of the 17th International Topical Meeting on Nuclear Reactor Thermal Hydraulics* (*NURETH-17*), Xi'an, Shaanxi, China, September 3-8 (2017).
18. A. FAYA, L. WOLF, and N. TODREAS, "Development of a Method for BWR Subchannel Analysis," Energy Laboratory Report No. MIT-EL 79-027 (1979).
19. M. ZIMMERMANN, "Development and Application of a Model for the Cross-Flow Induced by Mixing Vane Spacers in Fuel Assemblies," KIT (2015).
20. A. FAYA, L. WOLF, and N. TODREAS, *Canal User's Manual* (1979).
21. S. BEUS, "A Two-Phase Turbulent Mixing Model for Flow in Rod Bundles," WAPD-TM-2438, Bettis Atomic Power Laboratory (1971).
22. K. BRYNJELL-RAHKOLA, J. Le CORRE, and C. ADAMSSON, "Validation of VIPRE-W Sub-Channel Void Predictions using NUPEC/BFBT Measurements," *Proceedings of the 13th International Topical Meeting on Nuclear Reactor Thermal Hydraulics* (*NURETH-13*), Kanazawa City, Japan, September 27–October 2, N13P1080 (2009).
23. D. REDDY, S. SREEPADA, and A. NAHAVANDI, "Two-Phase Friction Multiplier Correlation for High-Pressure Steam-Water Flow," EPRI NP-2522, Research Project 813 (1982).

24. G. WALLIS, *One-Dimensional Two-Phase Flow*, Chapter 11, pp. 323-330, McGraw-Hill (1969).

25. J. Le CORRE, K. BRYNJELL-RAHKOLA, and C. ADAMSSON, "Steady-state Void, Pressure Drop and Critical Power BFBT Benchmark Analysis and Results with VIPRE-W/MEFISTO," Westinghouse Electric Sweden AB (2009).

26. A. WYSOCKI and R. SALKO, "Validation of CTF Droplet Entrainment and Annular/Mist Closure Models using Riso Steam/Water Experiments," CASL-U-2016-1080-000 (2016).

27. US NRC, *TRACE V5.0 Theory Manual*, Chapter 4, 175–184 (2012).

28. R. SALKO, M-O. DELCHINI, X. ZHAO, D. POINTER, and W. GURECKY, "Summary of CTF Accuracy and Fidelity Improvements in FY17," CASL-X-2017-1428-000 (2017).

29. J. LANE, "The Development of a Comprehensive Annular Flow Modeling Package for Two-Phase Three-Field Transient Safety Analysis Codes," PhD thesis, Pennsylvania State University (2009).

30. NEA, "International Benchmark on Pressurized Water Reactor Sub-Channel and Bundle Tests. Volume II: Benchmark Results of Phase I - Void Distribution," OECD/NEA/NSC/R(2015)4 (2016).

31. W. HANCOX and W. NICOLL, "A General Technique for the Prediction of Void Distributions in Non-Steady Two-Phase Forced Convection," *International Journal of Heat and Mass Transfer*, **14**, 1377 (1971).

32. J. R. S. THOM, W. M. WALKER, T. A. FALLON, and G. F. S. Rising, "Boiling in Subcooled Water During Flow Up Heated Tubes or Annuli Pumps," *Symposium on Boiling Heat Transfer in Steam Generating Units and Heat Exchangers*, Manchester, England, September 15–16 (1965).

33. J. C. CHEN, “A Correlation for Boiling Heat Transfer to Saturated Fluids in Convective Flow,” ASME paper 63-HT-34 (1963).

TABLE I. BFBT Two-Phase Pressure Drop Relative Root-Mean-Square Error (rRMSE) Values (in %)

Tap #	T1	T2	T3	T4	T5	T6	T7	T8	T9	All
VIPRE-01	15.7	18.9	9.0	12.7	12.4	11.7	8.4	0.6	8.6	11.9
CTF	9.5	7.8	24.1	11.6	6.9	5.6	4.9	0.7	6.4	10.6

TABLE II. BFBT Void Benchmark Results: Mean Error (ME) and Standard Deviation (σ)

Test	0-1 ME	0-2 ME	0-3 ME	1 ME	4 ME	All ME	All σ
# of points	3	3	3	3	3	15	15
VIPRE-01							
All	-0.02	-0.03	-0.03	-0.02	-0.01	-0.02	0.01
Corner	-0.03	-0.04	-0.03	-0.02	-0.08	-0.04	0.04
Side	-0.04	-0.04	-0.03	-0.02	-0.01	-0.03	0.02
Inner	-0.02	-0.02	-0.02	-0.01	0.00	-0.01	0.01
nNH	-0.03	-0.04	-0.05	-0.03	-0.05	-0.04	0.01
iNH	0.00	-0.05	0.01	0.01		-0.01	0.03
CTF (VD: on; two-phase TM enhancement: on)							
All	-0.05	-0.06	-0.06	-0.04	-0.03	-0.05	0.02
Corner	0.01	0.01	0.04	0.08	-0.01	0.02	0.05
Side	-0.04	-0.04	-0.02	0.00	0.01	-0.02	0.03
Inner	-0.05	-0.05	-0.04	-0.05	-0.04	-0.05	0.02
nNH	-0.06	-0.12	-0.13	-0.07	-0.10	-0.10	0.04
iNH	-0.02	-0.16	-0.26	-0.00		-0.11	0.12
CTF (VD: off; two-phase TM enhancement: off)							
All	-0.05	-0.06	-0.06	-0.04	-0.03	-0.05	0.02
Corner	0.03	0.03	0.05	0.07	-0.01	0.03	0.04
Side	-0.03	-0.03	-0.02	-0.01	0.00	-0.02	0.02
Inner	-0.06	-0.06	-0.06	-0.05	-0.04	-0.06	0.02
nNH	-0.05	-0.09	-0.11	-0.06	-0.08	-0.08	0.03
iNH	0.01	-0.05	-0.14	0.01		-0.04	0.07

TABLE III. PSBT Single Subchannel Void Results: Sensitivity Study

CTF version	Heat flux term interpretation[a]		Void mean error [-]				# of failed cases (out of 43 points)
	q''_{NB}	q''_{cond}	S1	S2	S3	S4	
#1 (baseline)	q''_{Thom}	$q''_{HN} - q''_{FC}$	0.08	0.02	0.09	0.05	15
#2	$q''_{Thom} - q''_{FC}$	$q''_{HN} - q''_{FC}$	0.04	0.03	0.05	0.02	3
#3	$q''_{Thom} - q''_{FC}$	q''_{HN}	0.04	0.03	0.04	0.01	0

[a] q''_{Thom} - heat flux (and heat transfer coefficient) calculated with Thom correlation; q''_{FC} - heat flux removed by forced convection calculated with Dittus-Boelter single-phase liquid correlation.

TABLE IV. CTF vs. VIPRE-01: Fluid Flow and Heat Transfer Key Closure Models

Model	CTF	VIPRE-01
Turbulent mixing (TM)	• constant coefficient • two-phase enhancement (Beus)	• constant coefficient *or* Reynolds dependent correlation
Void drift (VD)	Lahey-Moody	*currently unavailable*
Single-phase pressure drop	wall friction (McAdam's *or* user-defined) & form loss coefficients	
Two-phase pressure drop	Wallis multiplier	EPRI multiplier (default)
Single-phase heat transfer	Dittus-Boelter *or* user-defined	
Subcooled nucleate boiling	• Thom *or* Chen • near-wall vapor re-condensation (Hancox-Nicoll)	• EPRI profile-fit + drift-flux (default) • Thom *or* Chen
Saturated nucleate boiling	Thom *or* Chen	

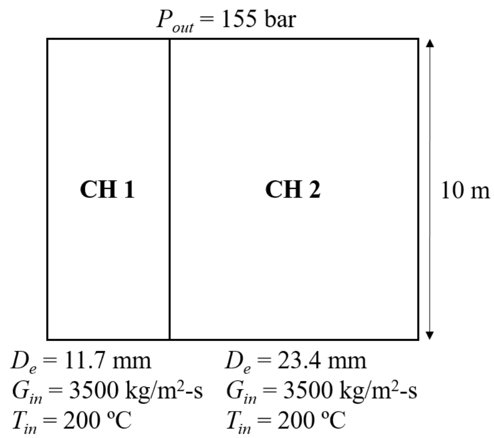


Fig. 1. Model of two-channel flow split problem.

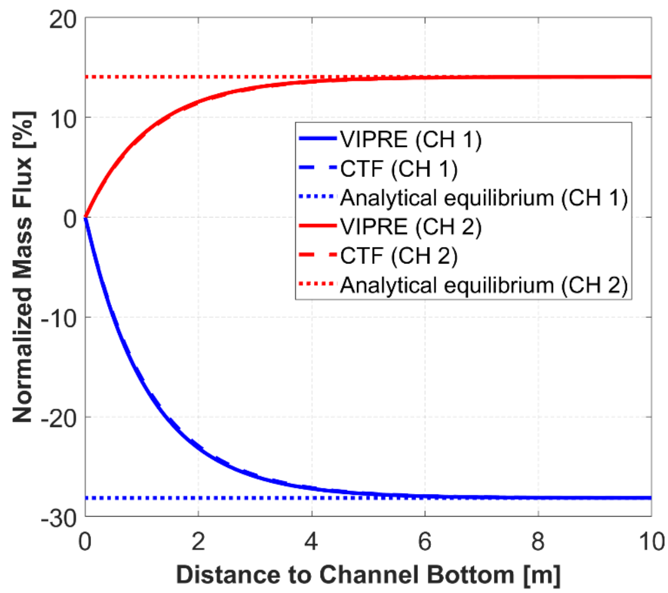


Fig. 2. Predicted normalized mass flux profile in two-channel system versus analytical solution.

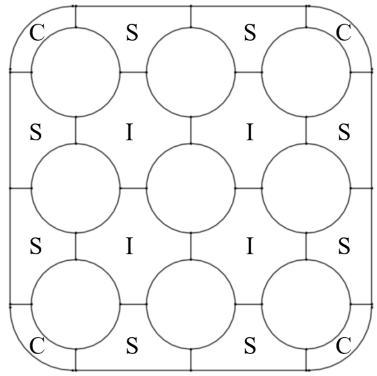


Fig. 3. GE 3 × 3 bundle channel layout (I: inner, S: side, C: corner).

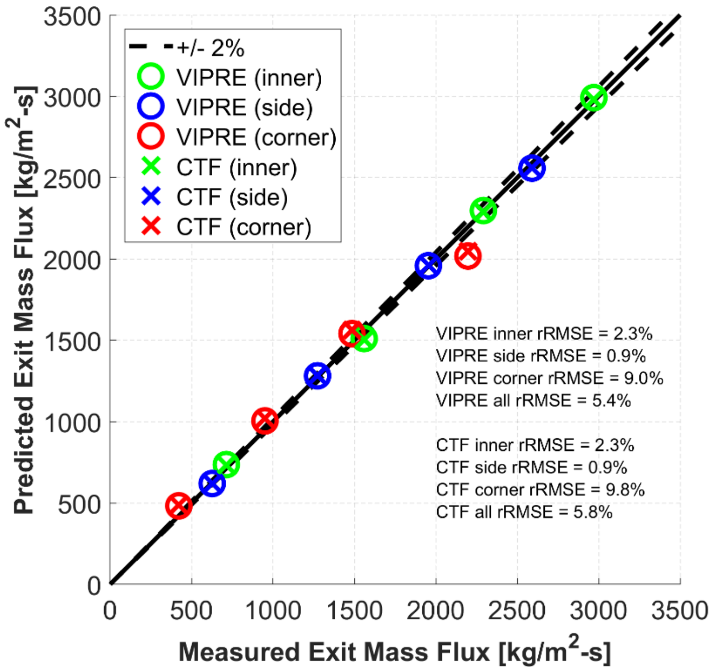
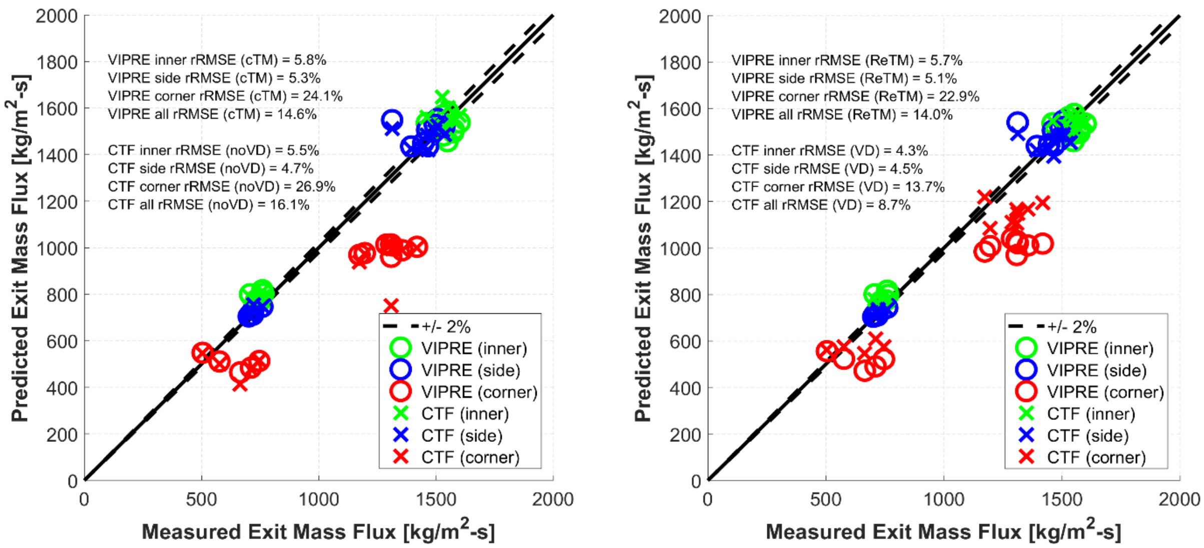


Fig. 4. Predicted vs. measured exit mass flux for GE 3 × 3 single-phase cases.



(a) (b)

Fig. 5. Predicted vs. measured exit mass flux for GE 3 × 3 two-phase cases with (a) cTM & noVD and (b) ReTM & VD.[a]

[a] *cTM*: constant TM coefficient (0.007); *ReTM*: Reynolds number dependent TM model; *VD*: VD and two-phase TM enhancement enabled in CTF (1-ϕ TM coefficient = 0.007); *noVD*: VD and two-phase TM enhancement disabled in CTF (1-ϕ TM coefficient = 0.007).

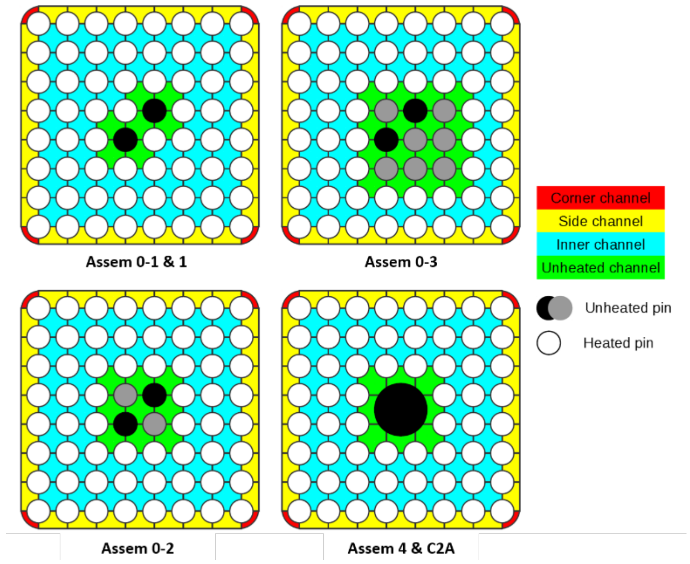


Fig. 6. BFBT bundle channel layout.[a]

[a] Within the unheated pins, black pins represent guide tubes, while gray pins represent heater rods that were shut off for particular assembly configurations. "Unheated channels" include both inside-unheated-rod and near-unheated-rod channels.

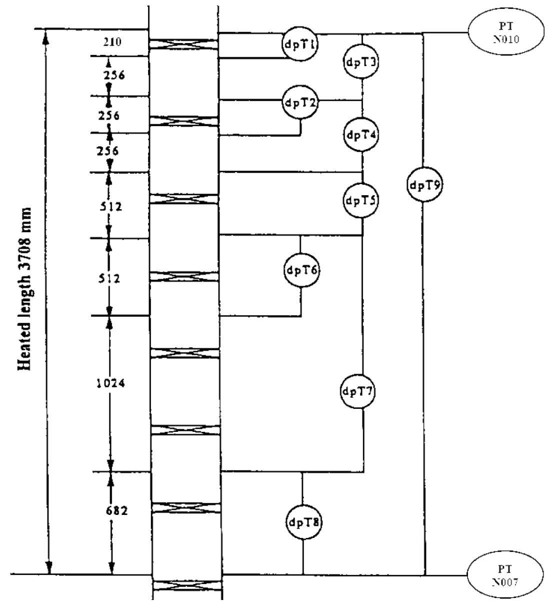


Fig. 7. BFBT bundle pressure tap axial locations.[7]

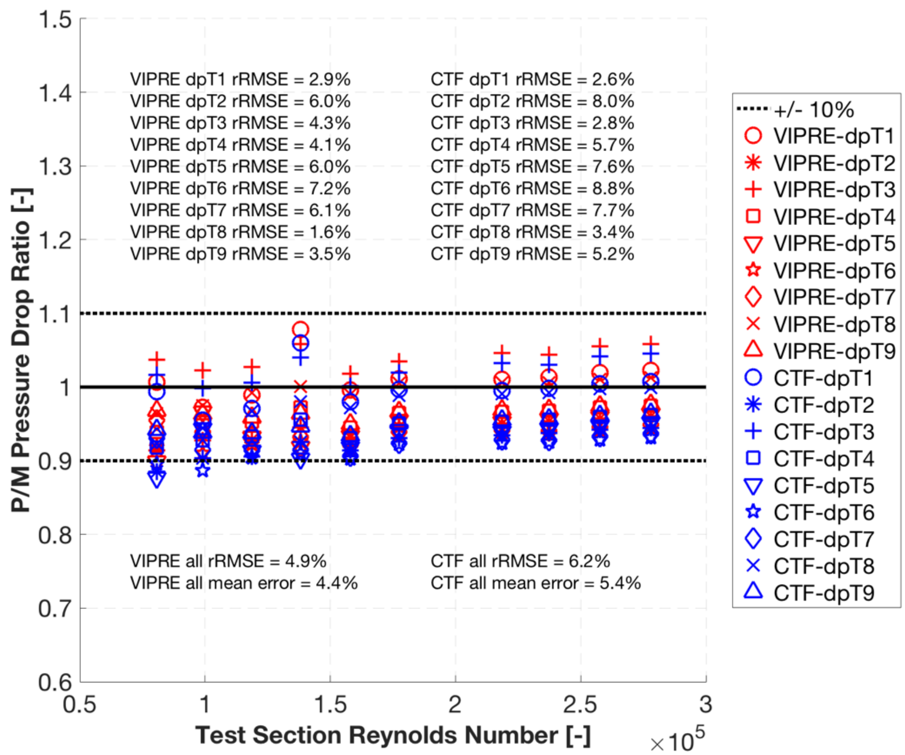
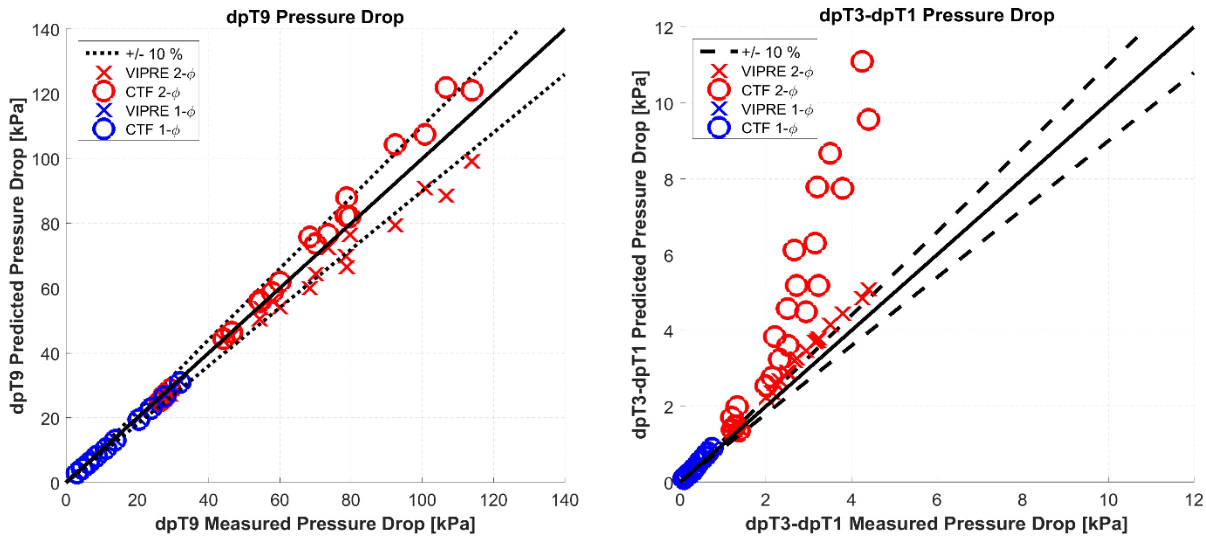
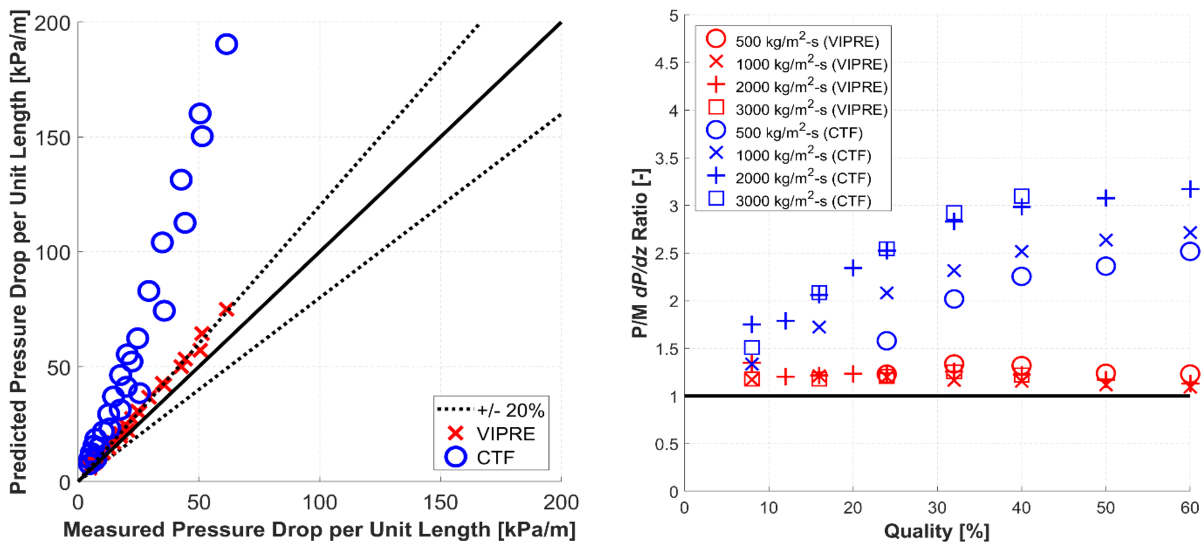


Fig. 8. BFBT predicted-to-measured single-phase pressure drop ratio.



(a) (b)

Fig. 9. Predicted vs. measured BFBT pressure drop for (a) dpT9 and (b) dpT3-dpT1.



(a) (b)

Fig. 10. Predicted vs. measured RISO adiabatic tube (a) pressure drop per unit length and (b) predicted-to-measured *dP*/*dz* ratio versus quality.

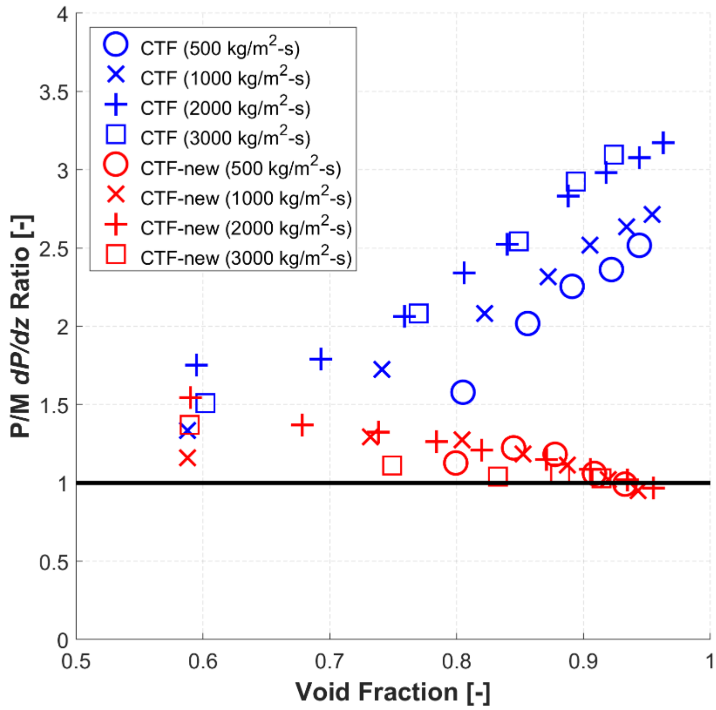
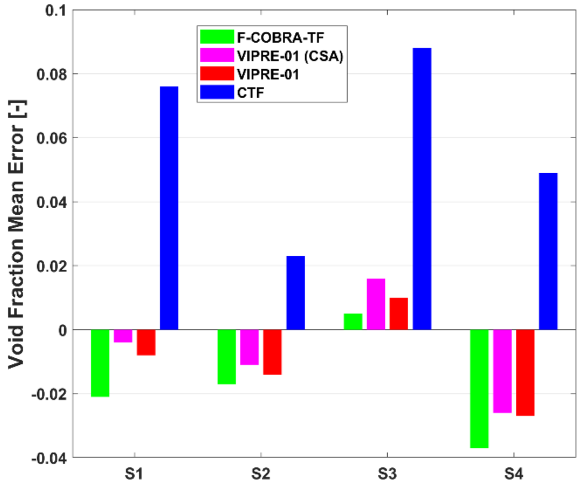
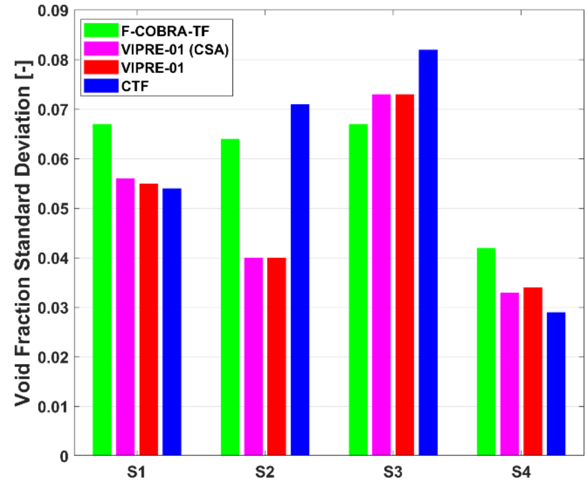


Fig. 11. RISO predicted-to-measured *dP*/*dz* ratio versus void fraction: sensitivity study.



(a)



(b)

Fig. 12. PSBT single subchannel void fraction (a) mean error and (b) standard deviation of error by test series.[a]

[a] with the baseline version of CTF

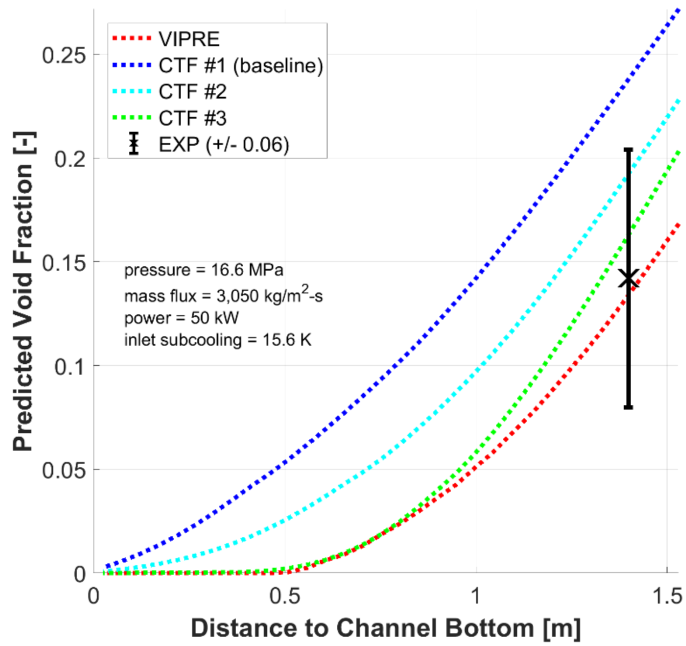


Fig. 13. PSBT single subchannel test point 1.1222 axial void profile: sensitivity study.

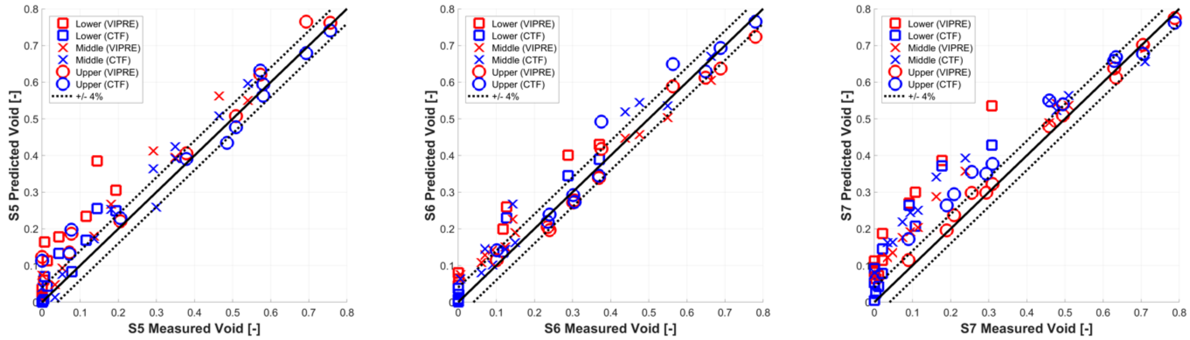


Fig. 14. Predicted vs. measured region-averaged void fraction for PSBT bundle test series S5 (left), S6 (middle) and S7 (right).[a]

[a] with CSA/VIPRE and the baseline version of CTF