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

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Abstract — As part of the Consortium for Advanced Simulation of Light Water Reactors, the subchannel code CTF is being used for single-phase 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 full-size fine-mesh bundle tests (BFBTs), the Risø tube, and the pressurized water reactor 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 the CTF results.

The flow split verification problem and single-phase GE 3 × 3 results are essentially in perfect agreement between the two codes. For the 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 overprediction by CTF is observed at relatively high void locations of the facility. This overestimation tendency is confirmed by the Risø cases. While overall statistics are satisfactory, both BFBT and PSBT bubbly-to-churn flow void contents are markedly overpredicted 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.

Note — Some figures may be in color only in the electronic version.

I. INTRODUCTION

Subchannel analysis has been used extensively to evaluate corewide 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 COBRA-TF legacy subchannel code1 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 U.S. Department of Energy (DOE) to bridge research,
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 a single-phase flow case (including a two-channel flow split verification problem and single-phase GE 3 × 3) and a two-phase flow case (GE 3 × 3). Single- and two-phase BFBT pressure drop tests are simulated, as well as Risø 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 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 (interchannel natural flow mixing due to turbulence, usually assumed to cause a lateral enthalpy flux without any associated net mass-flow) model was disabled so that wall friction became the only cross-flow driver. The McAdams friction correlation 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

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*a CASL website: www.casl.gov.*

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![Fig. 1. Model of two-channel flow split problem.](image)
flow ceases). The CTF versus 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. 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. General Electric 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. 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 test points 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 \( \beta \). A constant value of \( \beta = 0.007 \) is used here (and throughout this paper unless otherwise specified) in accordance with previous assessments using the Kumamoto University 2 × 3 facility data. It is noticeable that the VIPRE-01 and CTF solutions substantially agree with each other and both closely match the experimental results for the inner and side channels, having the relative root-mean-square error (rRMSE) values very close to or less than the measurement uncertainty (±2%).
The corner channel predictions deviated more from the 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 a liquid phase along interconnected channels to redistribute itself according to a certain equilibrium void distribution\(^1\)) mechanism. Void drift models are not currently available in VIPRE-01, and the code user manual\(^1\) 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,\(^8\) in which the default scaling weight factor \(K_V = 1.4^4\) is recommended.\(^4\) In addition, experimental evidence showed a 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 \(\Theta_M\) 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 (Ref. 4).

A total of 13 two-phase flow measurement points made publicly available are compared with the simulation results. Initially, cases were run with VD disabled and a constant TM coefficient (\(\beta = 0.007\)) in CTF (no two-phase enhancement). The same grid loss coefficients as provided in the original reference\(^6\) were applied in both codes.\(^13\) Figure 5a shows that under this configuration, the CTF and VIPRE-01 mass fluxes were visually and statistically similar, closely matching the experimental data for the inner and side channels (rRMSE \(\approx 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 overprediction of void content in this channel type. Discrepancies between the two codes and with the measured results became more dramatic when the flow at the bundle outlet was in the annular-mist regime.\(^d\)

When both the two-phase TM enhancement and VD were taken into account, the CTF solutions were significantly improved and agreed better with the measurements (see Fig. 5b), in particular, for the 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 to 12 mm) and 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. Figures 5a and 5b show that while the VIPRE-01 results with ReTM slightly outperformed 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 the mass flux comparison: The corner channel quality tended to be overestimated 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 by Neykov et al.\(^7\) Tests from Phase II (critical power benchmark) Exercise 0 (steady-state pressure drop)

\(^d\)According to the void-based flow regime map in CTF (Ref. 4).
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 nonuniform axial (cosine shape) and radial power distribution, was used for all test cases. Its cross-sectional view among other assembly types, as specified in Sec. IV.A, is depicted in Fig. 6, showing 60 heater rods and one large central guide tube with no water. Pressure tap locations are shown in Fig. 7.

Fig. 5. Predicted versus measured exit mass flux for GE 3 × 3 two-phase cases with (a) cTM and noVD and (b) ReTM and VD. [cTM: constant TM coefficient (0.007); ReTM: Reynolds number–dependent TM model; VD: VD and two-phase TM enhancement enabled in CTF (1-ф TM coefficient = 0.007); noVD: VD and two-phase TM enhancement disabled in CTF (1-ф TM coefficient = 0.007)].

Fig. 6. BFBT bundle channel layout. (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.)
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 (McAdams correlation\(^4\)) and grid form loss (Shiralkar and Radcliffe’s approach\(^{13}\)) models in both codes. Figure 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 the measured and the predicted results (i.e., rRMSE) for dpT9, which covered the entire bundle with seven grids, was smaller than all tap data. The taps with the largest sources of error were T2, T5, T6 (one grid span), and T7 (three grid spans). The reported measurement error for the 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. In CTF, the Wallis multiplier, defined as 1 over the liquid volumetric fraction squared, was applied on the liquid wall drag. The form loss due to spacer grids was calculated with the Romie multiplier in VIPRE-01, while it was computed separately for each phase in CTF (Ref. 4).

Table I summarizes pressure drop rRMSEs using VIPRE-01 and CTF for all nine pressure taps. The 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, the discrepancies at certain tap locations were significant. The largest disagreement with the experiments was from taps T1 through T4, 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 the 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 by Brynjell-Rahkola et al. and Le Corre et al. On the other hand, CTF tended to overpredict the 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 Sec. III.B.

### III.B. Risø Round Tube Benchmark

The Risø experimental facility 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 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 Risø #10 (tube inner diameter = 10 mm, length = 9 m, and exit pressure = 7 MPa) was selected for this work to serve as a separate-effects 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

| Tap Number | 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 |

*In units of percent.

![Fig. 9. Predicted versus measured BFBT pressure drop for (a) dpT9 and (b) dpT3-dpT1.](image)
(from 500 up to 3000 kg/m²·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.

Figure 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 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 by Wallis. In the TRACE V5.0 theory manual, 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, and a modified multiplier was derived to better fit measurements. As compared to Wallis, the modified formulation has a lower exponent (1.72 versus 2) in the denominator and has been applied in TRACE for the bubbly/slug flow regime. The Risø #10 high void content (>80%) cases were recently modeled in TRACE (Ref. 28), and close agreement was found between CTF and TRACE, significantly overpredicting the 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 (Ref. 4), the interfacial drag in the 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 friction factor is taken as the maximum of the unstable film friction factor (Henstock and Hanratty model) and five times the stable film friction factor (Wallis model). Such a criterion not only would 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 and Hanratty model (which yields higher friction and questionable applicability) and switching the denominator exponent of the Wallis wall friction multiplier from 2 to 1.72 (denoted “CTF-new” in Fig. 11), the predicted pressure drop’s agreement with the data was considerably improved for high void cases in which stable liquid film was established. However, it has been shown 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, this package relies on local closure terms such as film thickness (in the case of annular flow), a

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

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The Henstock and Hanratty model was developed using low-pressure air-water data from 30- to 35-mm-diameter tubes.

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**Fig. 10.** Predicted versus measured Risø adiabatic tube: (a) pressure drop per unit length and (b) predicted-to-measured $dP/dz$ ratio versus quality.
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 Refs. 28 and 29. According to Figs. 28 and 29 of Ref. 28, the accuracy of pressure drop and liquid film thickness predictions for the Risø tests is clearly improved with the new package, although further improvement on the flow regime detection logic and extended validation test matrix is required.

IV. VOID FRACTION PREDICTION ASSESSMENT

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 (two nonheated pins) and 0-3 (seven nonheated pins). Assembly type 1 employed the same geometry as assembly type 0-1, but it applied a nonuniform 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 Sec. III.A for the pressure drop benchmark.

Table II recapitulates the VIPRE-01 and CTF outlet void solutions in terms of the mean error (\[\text{ME} = \text{measured} - \text{predicted}\]) and standard deviation of error for all data, as well as for each assembly and channel type [corner, side, inner, near nonheated (nNH), in nonheated (iNH)]. Both codes applied the same wall friction correlation (modified McAdams) and form loss coefficient (0.94) as recommended in VIPRE-W (Ref. 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 the 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 the 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 nonheated rods. For these cases, CTF seemed to struggle more with nonperipheral channels when VD 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 NUPEC PSBT data were released following the success of the BFBT benchmark. Specifications are detailed by 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 x 5 bundle) were modeled. Reported void measurement uncertainties were 3% for a single subchannel\(^8\) and 4% for a region-averaged bundle.

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\(^8\)The masking effect of CT measurement technique on subchannel void fraction was studied by the OECD NEA (Ref. 30). Because of low fidelity of the current CT scan technology, some near-wall void is masked, leading to lower experimental void fraction than reality. Depending on the assumptions, the CT scanner will not show any void until the actual void content is 3.8% to 7.8%. Along with concerns on shifting of the CT image, it is therefore suggested that a total experimental uncertainty of ~6.2% would be deemed more reasonable.
IV.B.1. Single-Subchannel Benchmark

The object was to evaluate void modeling using CTF with no bias from interchannel 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, S2, S3, and S4, respectively). Channel length is equal to 1.555 m. Outlet pressure ranges from 4.9 to 16.6 MPa, mass flux from 494 to 3072 kg/m²·s, and experimental channel average void [at 1.4 m from the channel bottom where the void fraction was measured by means of a computed tomography (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 (Ref. 30) in which various existing subchannel, system, and CFD codes were evaluated, including VIPRE-01 by Computer Simulation & Analysis, Inc. (CSA) (the 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 postprocessed and are 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 full converge, regardless of axial noding, conduction model, and channel geometry options. Figure 12 shows that the CSA and the replicated VIPRE-01 results agreed closely with each other. CTF yielded higher void as compared to the 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 (Ref 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 as long as the wall surface temperature exceeded the bulk saturation temperature, and it relied on the near-wall condensation model of Hancox and Nicoll31 to recondense away the generated vapor during subcooled boiling.

### TABLE II

| Test | 0-1 ME | 0-2 ME | 0-3 ME | 1 ME | 4 ME | All ME | All σ |
|------|--------|--------|--------|------|------|--------|-------|
| Number 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 | 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 | |

*ME = mean error; σ = standard deviation.
To address the concern of void overprediction, a sensitivity study was conducted. In CTF, the calculation of the heat flux used for net vapor generation (NVG) $q_{SCVG}$ and the 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 the PSBT single-subchannel cases. Special attention should be paid when the Thom et al. correlation is used for calculation of the nucleate boiling heat flux (and the heat transfer coefficient) in the subcooled boiling region, 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. Based on the results presented in Table III and Fig. 13, version 3 led to the closest agreement with the 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 $5 \times 5$ 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 2216 mm, the middle at 2669 mm, and the upper at 3177 mm. Seventeen spacer grids were used on each bundle. CSA VIPRE-01 solutions were collected from the NEA report, and the predicted versus the 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 the measurement uncertainty (±4%), both codes tended to overpredict the 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 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 the CTF prediction of single-phase flow redistribution is consistent with the results from an analytical solution and VIPRE-01. Single-phase and low-void pressure drop predictions tend to compare favorably with the experimental data. The two-phase flow distribution was also well

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$h$ The Chen correlation has already taken into account both nucleate boiling and convective heat transfer components in its original formulation.

$i$ 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 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 (Ref. 10).
predicted with the inclusion of the two-phase TM and VD models in the code. However, this study has revealed that CTF tends to consistently overpredict the 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 the following:

1. interfacial drag in low (bubbly/slug flow) and high (churn/annular flow) void regions
2. two-phase wall friction multiplier
3. subcooled boiling heat transfer mechanism
4. TM and VD 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

| CTF Version | Heat Flux Term Interpretationa | Void Mean Error | Number of Failed Cases (Out of 43 Points) |
|-------------|--------------------------------|-----------------|------------------------------------------|
| 1 (baseline) | $q_{\text{Thom}}$ | 0.08 0.02 0.09 0.05 15 |
| 2           | $q_{\text{Thom}} - q_{\text{FC}}$ | 0.04 0.03 0.05 0.02 3 |
| 3           | $q_{\text{Thom}} - q_{\text{FC}}$ | 0.04 0.03 0.04 0.01 0 |

*a$q_{\text{Thom}}$ = heat flux (and heat transfer coefficient) calculated with Thom correlation; $q_{\text{FC}}$ = heat flux removed by forced convection calculated with Dittus-Boelter single-phase liquid correlation.*

![Fig. 13. PSBT single-subchannel test point 1.1222 axial void profile: sensitivity study.](image)

![Fig. 14. Predicted versus measured region-averaged void fraction for PSBT bundle test series (a) S5, (b) S6, and (c) S7 (with CSA/VIPRE and the baseline version of CTF).](image)
implementing other advanced/mechanistic closures in CTF to help improve its accuracy and robustness.

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References

1. M. THURGOOD et al., “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,” Proc. Int. Congress Advances in Nuclear Power Plants (ICAPP), Nice, France, May 3–6, 2015.

3. R. SALKO et al., “Development and Assessment of CTF for Pin-Resolved BWR Modeling,” Proc. Int. Conf. Mathematics and Computational Methods Applied to Nuclear Science and Engineering (M&C), Jeju, Korea, April 16–20, 2017.

4. R. SALKO and M. AVRAMOVA, “CTF Theory Manual,” CASL-U-2016-1110-000, Consortium for Advanced Simulation of Light Water Reactors (2016).

5. X. ZHAO et al., “Validation and Benchmarking of CTF for Single- and Two-Phase Flow,” Proc. 17th Int. Topl. Mtg. Nuclear Reactor Thermal Hydraulics (NURETH-17), Xi’an, 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 et al., “NUPEC BWR Full-Size Fine-Mesh Bundle Test (BFBT) Benchmark, Volume I: Specifications,” NEA/NSC/DOC(2005)5, ISBN 92- 64-01088-2, Organisation for Economic Co-operation and Development Nuclear Energy Agency Nuclear Science Committee (2006).

8. J. WURTZ, “An Experimental and Theoretical Investigation of Annular Steam-Water Flow in Tubes and Annuli at 30 to 90 Bar,” Risø Report No. 372, Risø National Laboratory (1978).

9. A. RUBIN et al., “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, Organisation for Economic Co-operation and Development Nuclear Energy Agency Nuclear Science Committee (2010).

10. C. STEWART et al., “VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores. Volume I: Mathematical Modeling,” NP-2511-CCM-A, Vol. 1, Rev. 4.3, Pacific Northwest National Laboratory (2011).
11. X. ZHAO, “Critical Power Characteristics of Axially Heterogeneous Tight Bundle Designs,” MSc Thesis, Massachusetts Institute of Technology (2016).

12. R. SALKO et al., “CTF Void Drift Validation Study,” CASL-U-2015-0320-002, Consortium for Advanced Simulation of Light Water Reactors (2015).

13. R. SALKO et al., “CTF Validation and Verification,” CASL-U-2016-1113-000, Consortium for Advanced Simulation of Light Water Reactors (2016).

14. M. SADATOMI et al., “Single- and Two-Phase Turbulent Mixing Rate Between Adjacent Subchannels in a Vertical $2 \times 3$ Rod Array Channel,” Int. J. Multiphase Flow, 30, 481 (2004); https://doi.org/10.1016/j.ijmultiphaseflow.2004.03.001.

15. C. STEWART et al., “VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores. Volume 2: User’s Manual,” NP-2511-CCM-A, Vol. 2, Rev. 4.4, Pacific Northwest National Laboratory (2011).

16. R. LAHEY and F. MOODY, The Thermal-Hydraulics of a Boiling Water Nuclear Reactor, American Nuclear Society, La Grange Park, Illinois (1977).

17. J. LE CORRE, “Experimental Investigation and Modeling of Void Drift in Modern BWR Fuel Designs,” Proc. 17th Int. Topl. Mtg. Nuclear Reactor Thermal Hydraulics (NURETH-17), Xi’an, China, September 3–8, 2017.

18. A. FAYA, L. WOLF, and N. TODREAS, “Development of a Method for BWR Subchannel Analysis,” MIT-EL 79-027, Massachusetts Institute of Technology Energy Laboratory (1979).

19. M. ZIMMERMANN, “Development and Application of a Model for the Cross-Flow Induced by Mixing Vane Spacers in Fuel Assemblies,” Karlsruhe Institute of Technology (2015).

20. A. FAYA, L. WOLF, and N. TODREAS, “Canal User’s Manual,” Massachusetts Institute of Technology Energy Laboratory (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,” Proc. 13th Int. Topl. Mtg. Nuclear Reactor Thermal Hydraulics (NURETH-13), Kanazawa City, Japan, September 27–October 2, 2009, N13P1080, Atomic Energy Society of Japan (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, Electric Power Research Institute (1982).

24. G. WALLIS, One-Dimensional Two-Phase Flow, Chap. 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, Consortium for Advanced Simulation of Light Water Reactors (2016).

27. “TRACE V5.0 Theory Manual,” Chap. 4, pp. 175–184, U.S. Nuclear Regulatory Commission (2012).

28. R. SALKO et al., “Summary of CTF Accuracy and Fidelity Improvements in FY17,” CASL-X-2017-1428-000, Consortium for Advanced Simulation of Light Water Reactors (2017).

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

30. “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, Organisation for Economic Co-operation and Development Nuclear Energy Agency Nuclear Science Committee (2016).

31. W. Hancox and W. Nicoll, “A General Technique for the Prediction of Void Distributions in Non-Steady Two-Phase Forced Convection,” Int. J. Heat Mass Transf., 14, 1377 (1971); https://doi.org/10.1016/0017-9310(71)90186-4.

32. J. R. S. Thom et al., “Boiling in Subcooled Water During Flow Up Heated Tubes or Annuli Pumps,” Proc. Symp. Boiling Heat Transfer in Steam Generating Units and Heat Exchangers, Manchester, United Kingdom, September 15–16, 1965.

33. J. C. Chen, “A Correlation for Boiling Heat Transfer to Saturated Fluids in Convective Flow,” BNL-6672, Brookhaven National Laboratory (1962).