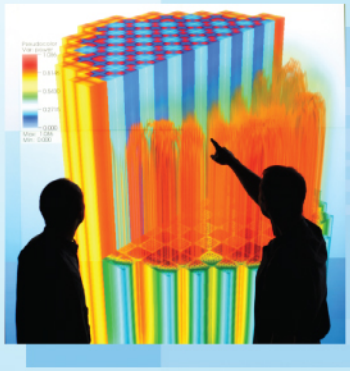
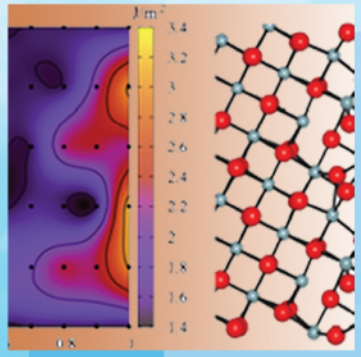
CASL-U-2014-0203-000

Power uprates
and plant life extension



Engineering design
and analysis

Science-enabling
high performance
computing



Fundamental science

Plant operational data

Consortium for Advanced Simulation of LWRs

Implementation and Validation of COBRA-TF Subcooled Nucleate Boiling Model

Liping C, Y. Sung, and V. Kucukboyaci
Westinghouse Electric Company

American Nuclear Society ICAPP 2014
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Implementation and Validation of COBRA-TF Subcooled Nucleate Boiling Model

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Introduction

- Consortium for Advanced Simulation of Light Water Reactor (CASL) is sponsored by US Department of Energy through its first Energy Innovation Hub program, aiming to provide nuclear energy industry with coupled, higher-fidelity modeling and simulation capabilities to address light water reactor operational and safety performance-defining phenomena.

- CASL's foundational technology, the Virtual Environment for Reactor Applications (VERA) is a coupled multi-physics models solved by the state-of-the-art numerical methods.

- COBRA-TF is selected by CASL as the reactor core thermal hydraulic subchannel simulation code.

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Introduction

- COBRA-TF is envisioned to couple with the reactor neutronic and fuel rod performance codes in VERA to simulate the reactor thermal hydraulic responses and determine the limiting phenomena relating to nuclear power plant operation and safety.

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COBRA-TF Overview

- COBRA-TF was originally developed in 1980's by Pacific Northwest National Lab. (PNL) of US Department of Energy. Its major applications as designed originally were nuclear power plant loss of coolant accident and other anticipated operational transients.

- In its current form, it solves the subchannel thermal fluid conservation equations in three fields of vapor, liquid film and droplets with phase change and thermal and momentum interactions at phase interfaces and fluid to solid boundary (two-phase flow two-fluid model and solid conduction model)

- The conservation equations are numerically solved together with the closure laws defining the mass, heat and momentum transfers at the phase interfaces and walls.

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COBRA-TF Overview

- Post-CHF heat transfer models are critical in simulating nuclear plant accidental transient, as fuel rod peak cladding temperature is the most important safety limiting parameters.

- In the intended applications of COBRA-TF as part of the VERA tool suite (normal operation as well as accidental transient), the nucleate boiling heat transfer is equally important, e.g. the local subcooled boiling in normal operation is related to the CRUD formation and operational economics and safety.

- The nucleate boiling model (Chen correlation) of COBRA-TF is limited to low pressure conditions and inaccurate under high pressure condition typical of nuclear plant normal operation.

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Nucleate Boiling Models

- Chen Correlation (< 70 bar or 1000 psia):

$$h_{chen} = h_{fc} + h_{nb}$$

$$h_{fc} = F \times 0.023 Re^{0.8} Pr^{0.3}$$

$$h_{nb} = S \times 0.00122 \left[\frac{{k_L}^{0.79} {c_{pL}}^{0.45} {\rho_L}^{0.49}}{{\sigma_L}^{0.79} {\mu_L}^{0.79} {h_{fg}}^{0.79} {\rho_g}^{0.79}} \right] \Delta {T_{sat}}^{0.24} \Delta {p_{sat}}^{0.75}$$

- Thom Correlation (52~138 bar or 750~2000 psia):

$$q_{Thom} = e^{(2P/1260)} \times \left(T_w - T_f\right)^2 / 0.072^2$$

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Nucleate Boiling Models

- Jens-Lottes Correlation (7~172 bar or 100~2500 psia):

$$q_{Thom} = e^{(4P/900)} \times \left(T_w - T_f\right)^4 / 1.9^4$$

- Gorenflo Correlation (0.1~210 bar or 1.6~3040 psia):

$$h_{Gorenflo} = h_0 \times F_P \times \left(q^{"}/q_0^{"}\right)^n \times \left(R_P/R_{P0}^{"}\right)^{0.133}$$

$$F_P = 1.73 \times P_r^{0.27} + [6.1 + 0.63/(1 - P_r)] \times (P_r)^2;$$

$$n = 0.9 - 0.3 \times P_r^{0.15}$$

For water, the reference values are:

$$h_0 = 5600 \frac{W}{m^2 \cdot C}, \qquad q_0^{"} = 20000 \frac{W}{m^2}, \qquad R_{p0} = 0.4\ \mu m$$

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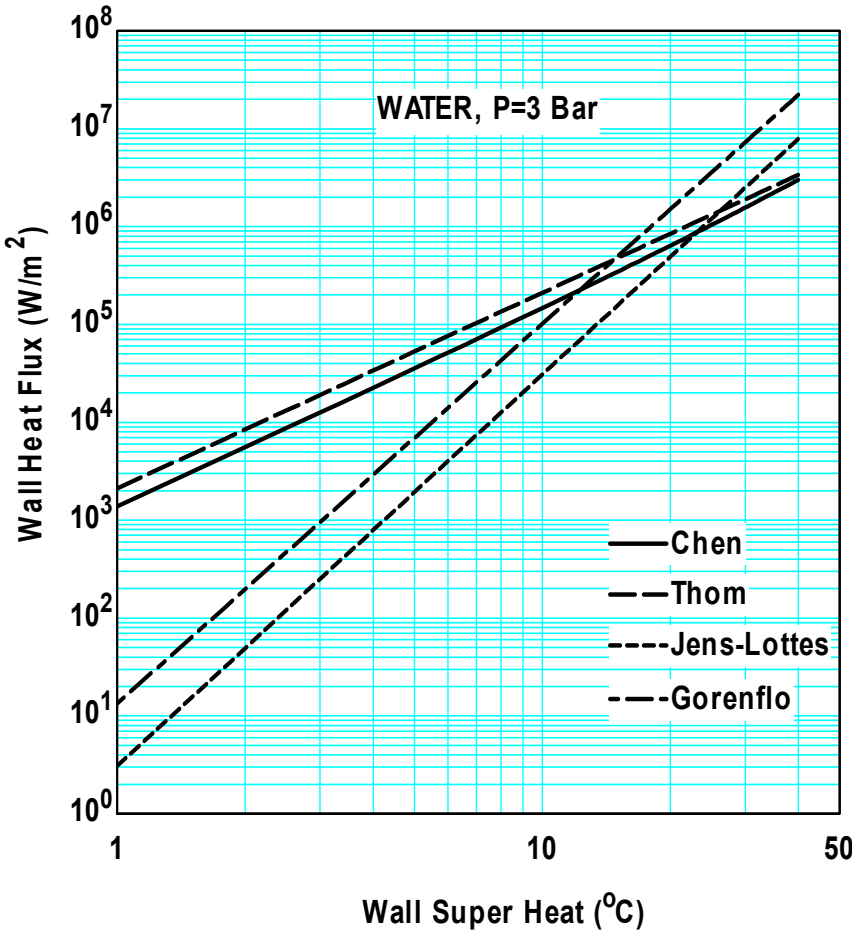
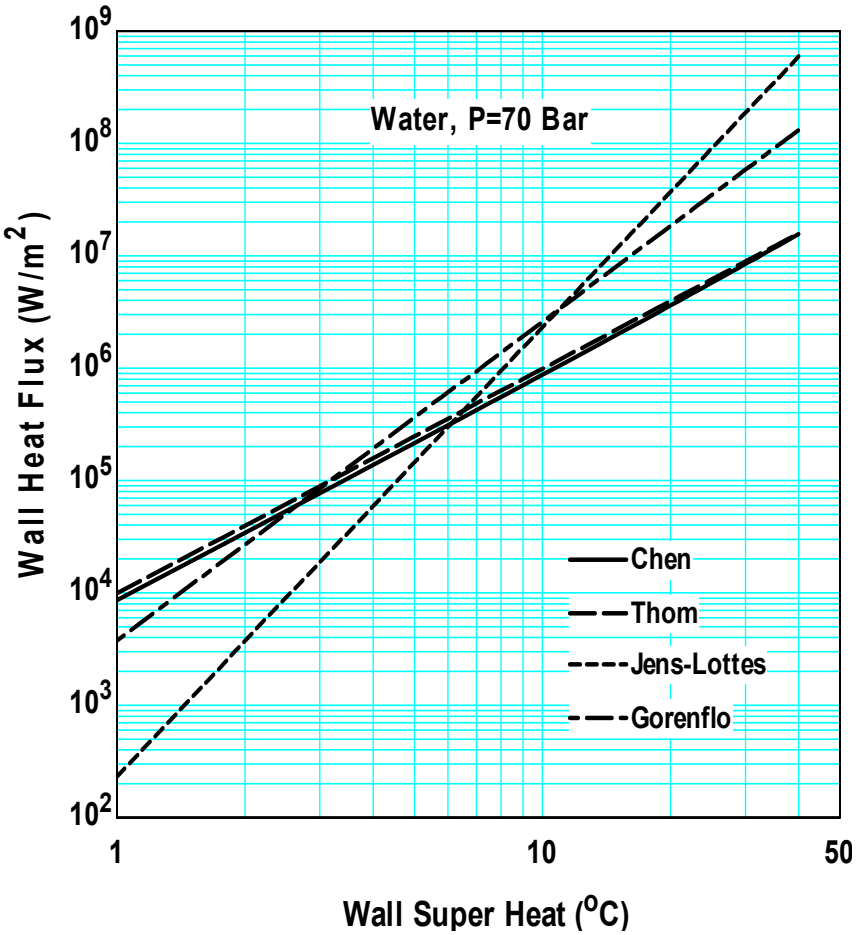
Nucleate Boiling Models





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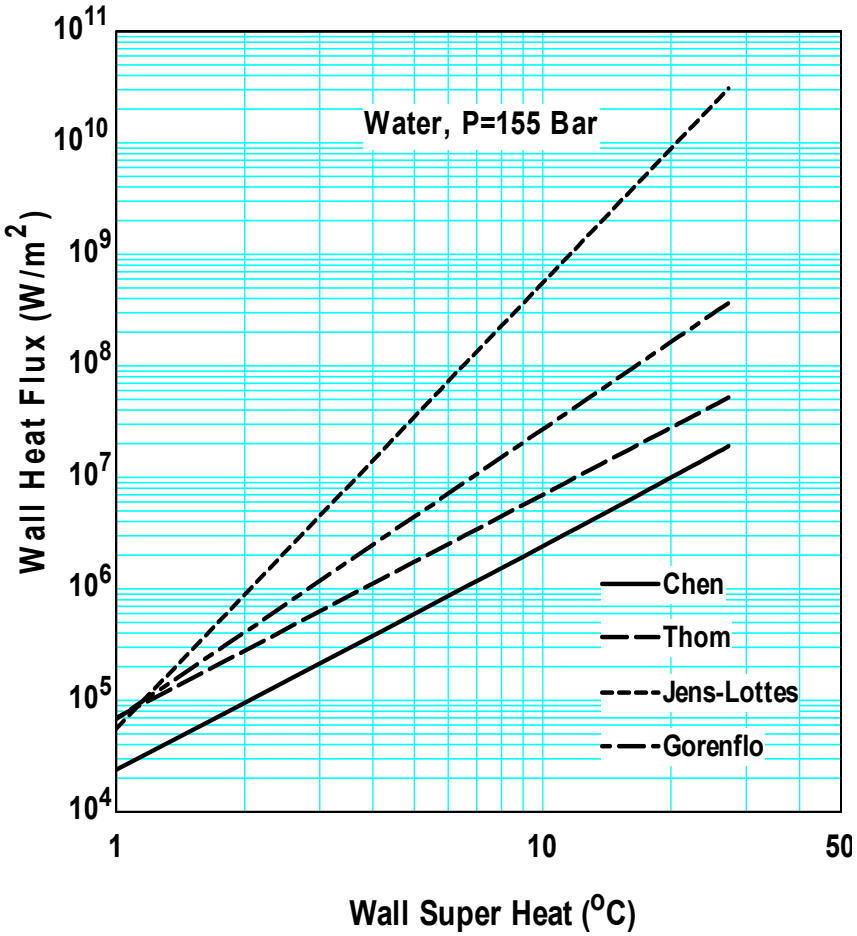
Nucleate Boiling Models



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Validation Test

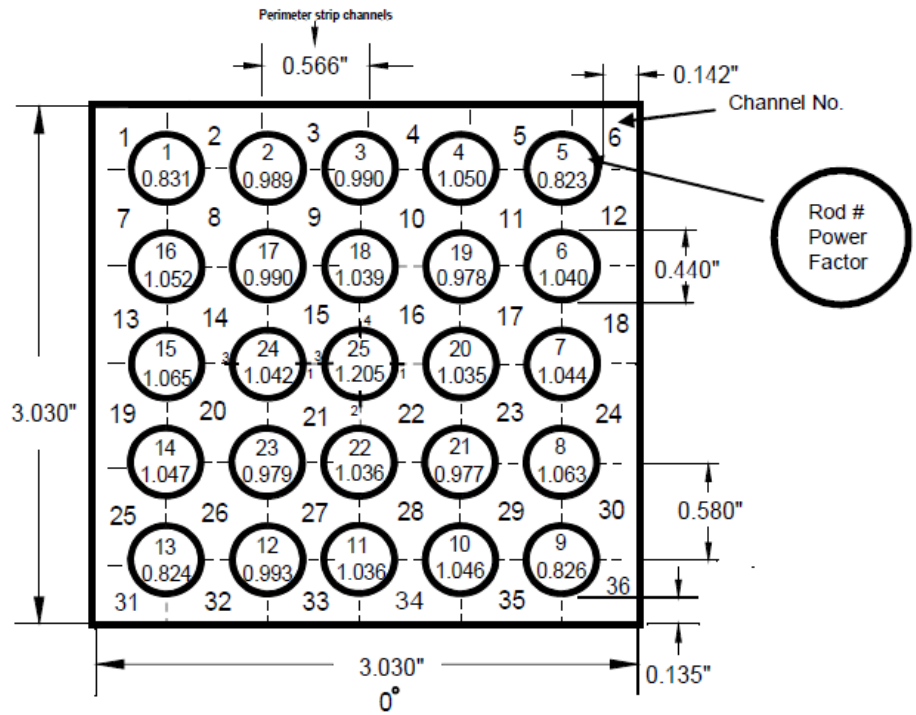
- Subcooled boiling test at the Columbia University Heat Transfer Research Facility

- Electrically heated 5 by 5 bundles simulates the peripheral region of adjacent CE 14 by 14 fuel bundles for different perimeter strip geometries (One test is selected);

- Rod O.D. = 0.440”;

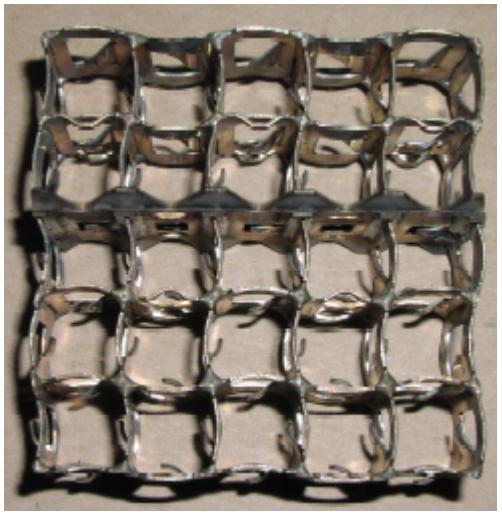
  Rod Pitch = 0.580”

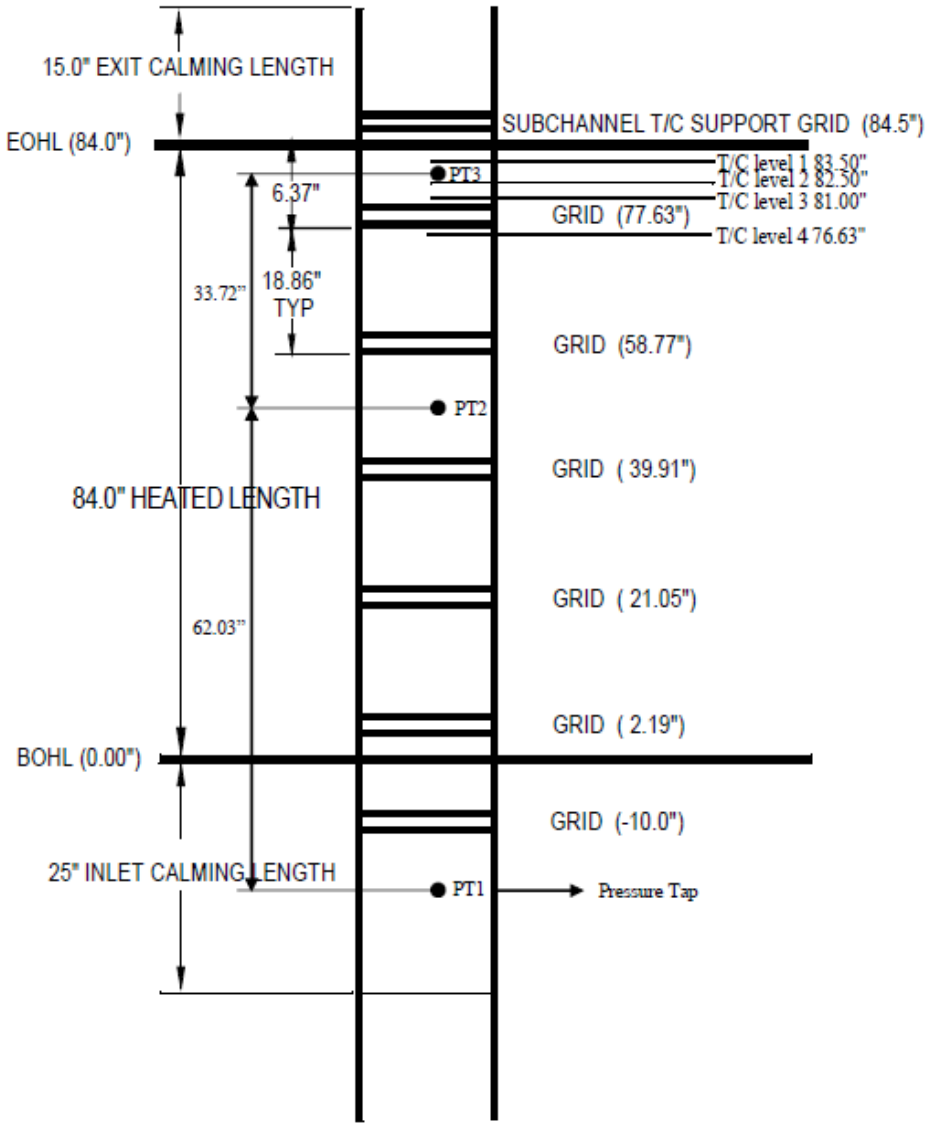
- Non-uniform lateral power distribution



Validation Test

- Uniform axial power with 7 feet of active heated length
- 6 Inconel spacer grids

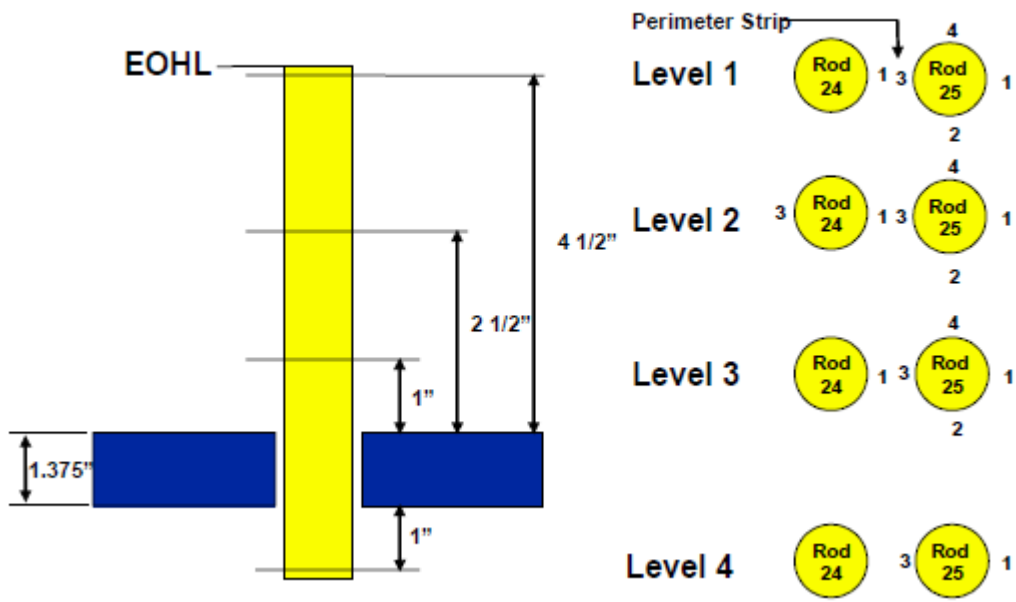




Validation Test

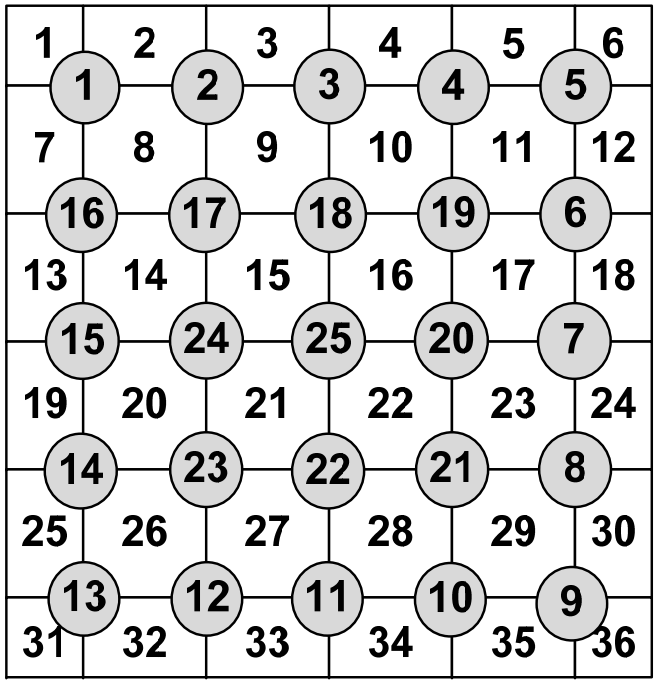
- Center rods (Rod #24 & #25) surface temperature measurement

- The average of the four thermal couple readings at level 2 is compared to the code results at the same elevation.



COBRA-TF Model

- 6 by 6 fluid channels formed by 5x5 heater rods;
- 50 equal axial nodes (7 ft. channel/rod length);
- Rod radial power distribution based on test;
- Azimuthal temperature variation is not resolved in the CTF model.



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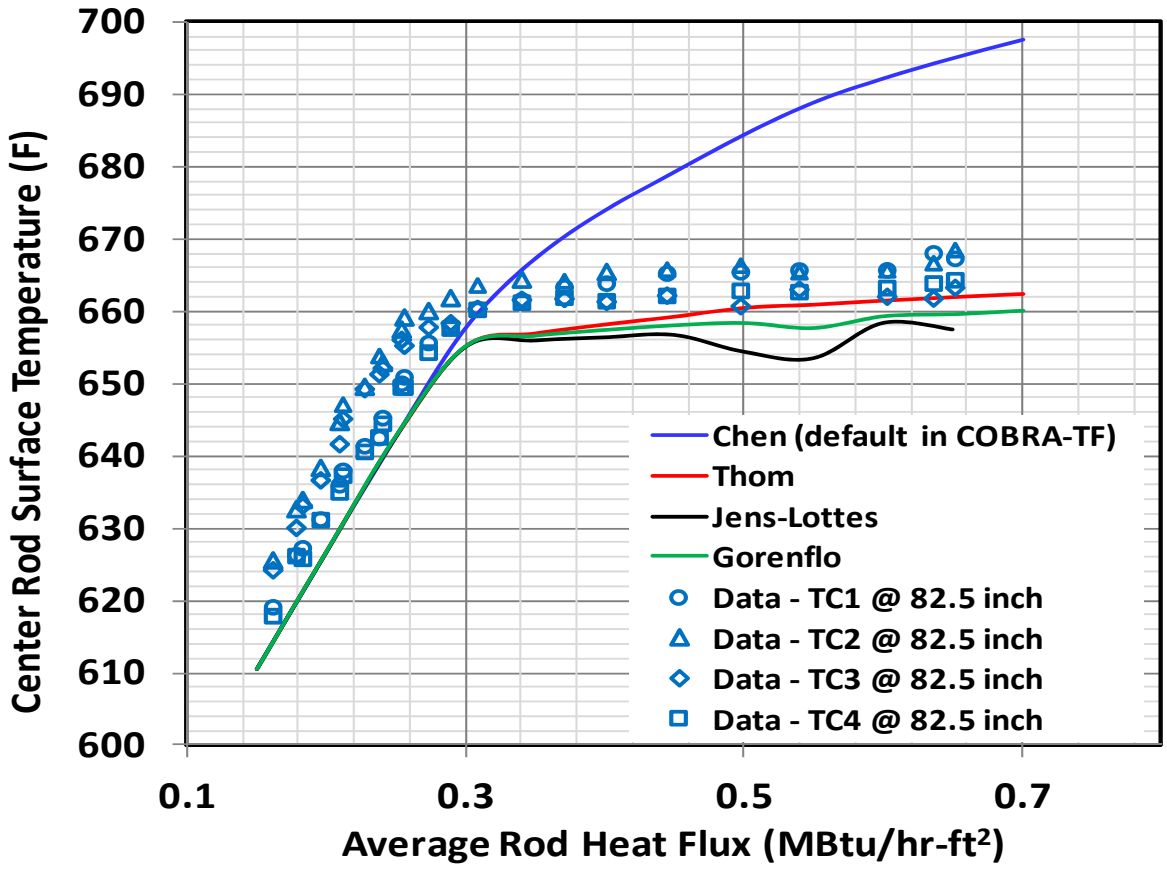
Validation Results



(P = 2255 psi, T_{in} = 563 F, G = 2.25 Mlbm/hr-ft^2)

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Conclusions

- The nulceate boiling heat transfer model built in COBRA-TF signifcantly over-predicts the wall surface temperature under high pressure conditions typical of nuclear power plant normal operation.

- The Thom correlation implemented in the COBRA-TF improves the prediction accuracy in comparison with the high pressure rod bundle tests. Jens-Lottes and Gorenflo model show similar accuracy as Thom correlation in high pressure condition.

- Both Thom and Chen correlations are numerically more stable in COBRA-TF than the Jens-Lottes and Gorenflo correlations.

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Questions?

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