CUPID 코드를 이용한 PWR 전노심 부수로스케일 정상상태 해석

Steady-state Thermal-hydraulic Analysis of PWR Whole Reactor Core in Subchannel Scale Using CUPID

May 18, 2017

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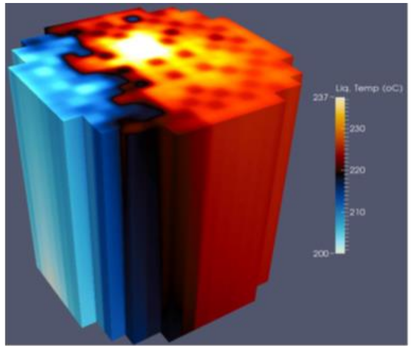
Contents

NuTHEL
Nuclear Thermal Hydraulic Engineering Lab.

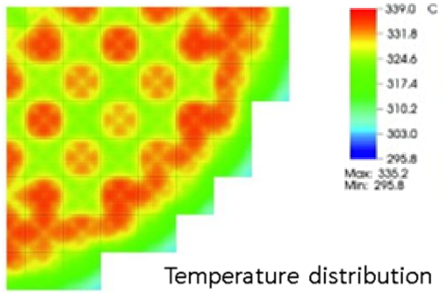
Background

- Multi-physics reactor core analysis with high fidelity thermal-hydraulic simulation tool
- Subchannel scale whole core pin-by-pin analysis
  - COBRA-TF (CTF in CASL, NURESAFE)
    - Under accident conditions with coupling multiple codes
  - COBRA-FLX (ARCADIA code system in AREVA)
    - Used for full core 3D pin-by-pin T/H analysis in PANBOX
  - SUBCHANFLOW (KIT)
    - Inhouse subchannel T/H analysis code for DYNSUB

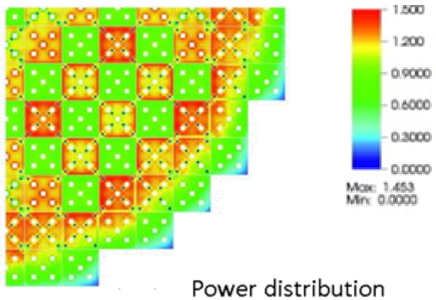


- MSLB anaylais at the HZP condition, CASL (Kucukboyaci et al. (2015))

- MATRA
  - Developed by KAERI (based on COBRA)
  - Effective for reactor core design and evaluation of DNBR margin
    - Achievement of required accuracy within reasonable time
  - Features not optimized for accident analyses
    - Homogeneous Equilibrium Model (HEM)
    - Spacial marching scheme in the axial direction



Temperature distribution



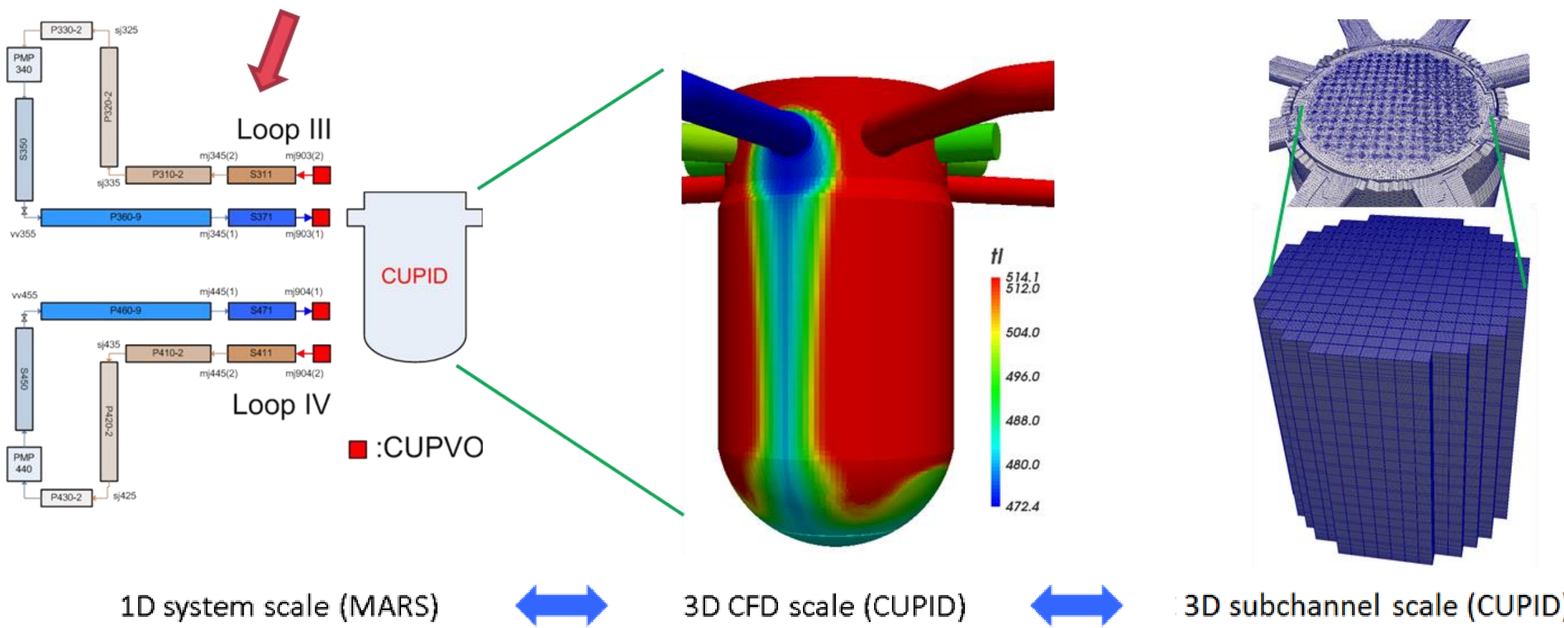
Power distribution

(Jung et al. (2013))

*) Kucukboyaci et al., COBRA-TF Parallelization and Application to PWR Reactor Core, CASL-U-2015-0167-000, 2015.
**) Jung et al., Practical reactor employing direct whole core neutron transport and subchannel thermal/hydraulic solvers, Annals of Nuclear Energy, 2013

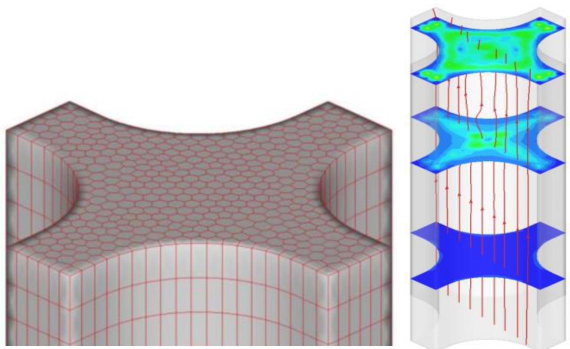
❖ CUPID

- KAERI's inhouse code for multi-dimensional two-phase flow simulation
- Physical models
  - Two-fluid model for two-phase flow
  - Porous medium approach with flow regime map and corresponding constitutive models
- Numerical solver
  - Highly parallelized, pressure correction equation for whole computational domain
  - Non-Staggered (Cell-Centered), Semi-implicit or Implicit numerical schemes



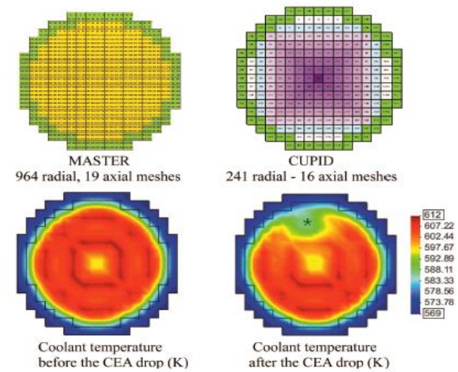
❖ Key subchannel T/H models

- CFD scale analysis for open medium model



Cho et al., 2013, KAERI

- Assembly scale analysis : CUPID-MASTER code



Jeong et al., 2010, KAERI

CFD scale | Subchannel scale | Assembly scale

Scale

❖ Objectives of the present study

- To extend the capability of CUPID to subchannel scale TH analysis
  - To implement subchannel TH models
  - To validate the extended code for various rod bundle experimental data
  - To demonstrate the whole core analysis capability

Contents

Validation of CUPID subchannel T/H models

Subchannel model

- Friction factor, form loss, turbulent mixing and void drift, and grid spacer model
- Considering lateral and axial flow direction

Model	Lateral direction	Axial direction	Note
Frictional pressure loss		O	$-\frac{1}{2}\left(\frac{f}{d_{hy}}+K'\right)\left(\frac{G_k^2}{\rho_k}\right)$
Form loss	O		$-\frac{K_G}{2}\left(\frac{W_{IJ,k}\lvert W_{IJ,k}\rvert}{l_{IJ}\rho_k s_{IJ}}\right)$
Turbulent mixing and void drift	O		• EM (Equal Mass exchange) • EVVD (Equal Volume exchange and Void Drift)
Grid spacer		O	$-\frac{K}{2}\left(\frac{G_k^2}{\rho_k}\right)$

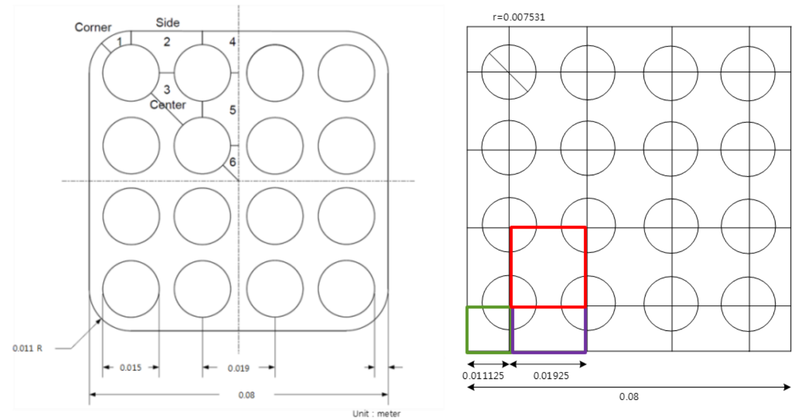
Where, $G_k = \sum_k \alpha_k \rho_k U_k$,
$W_{IJ,k} = \sum_k \alpha_k \rho_k V_{IJ,k} \times s_{IJ}$

Validation of subchannel analysis code MATRA (Kwon et al., 2014, KAERI)

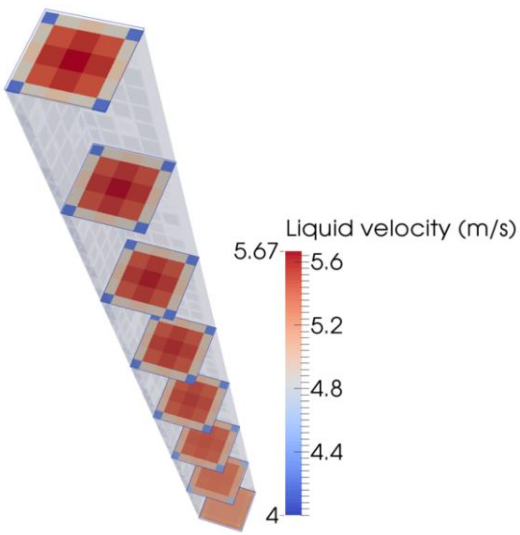
CTF Theory Manual (Robert K. Salko et al., 2014, PSU)

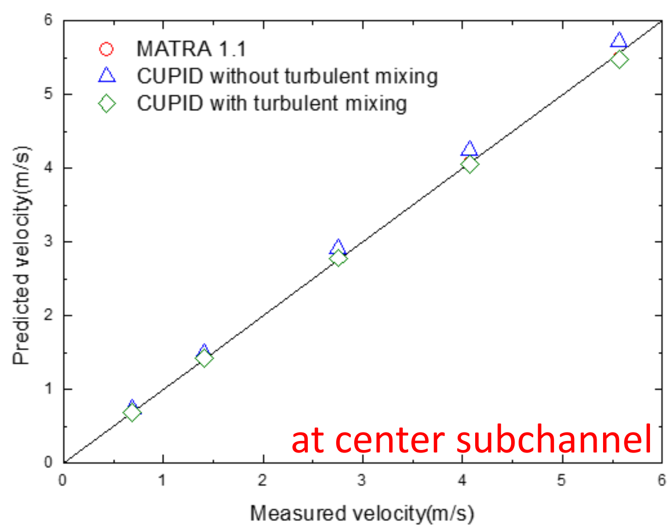
Validation of CUPID for unheated single-phase flow

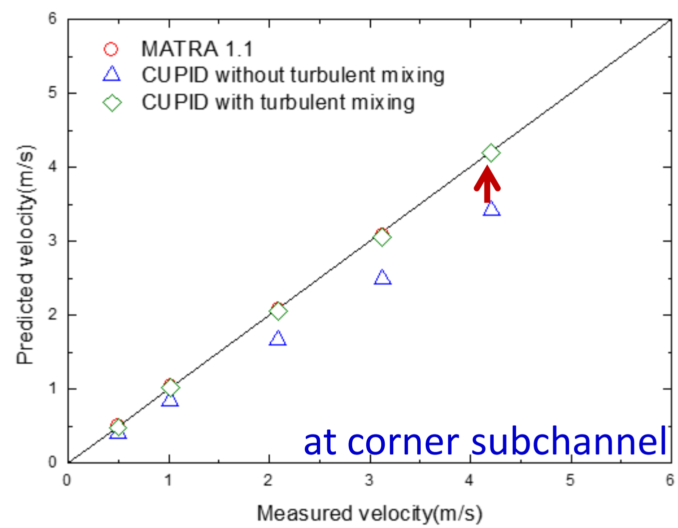
- CNEN 4x4 mixing test
  - Liquid velocity and pressure drop
  - Effect of turbulent mixing model
  - Errors in liquid velocity
    - Corner: -2.6 ~ 0.5 %
    - Center: -1.8 ~ 0.8 %



- Cross-sectional views of the test section and CUPID model







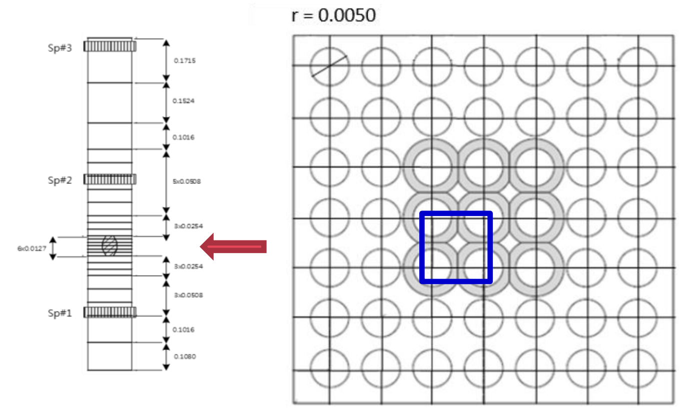
Marinelli V., Pastori L., Kjellen B., "Experimental Investigation of Mass Velocity Distribution and Velocity Profiles in an LWR Rod Bundle", Trans. ANS 15, pp 413-415, 1972.

Hwang, Dae Hyun, Kyung Won Seo, and Hyouk Kwon. *Development of Validation System for Subchannel Analysis Codes under Steady-State PWR Conditions*. No. KAERI/TR--4143/2010. Korea Atomic Energy Research Institute, 2010.

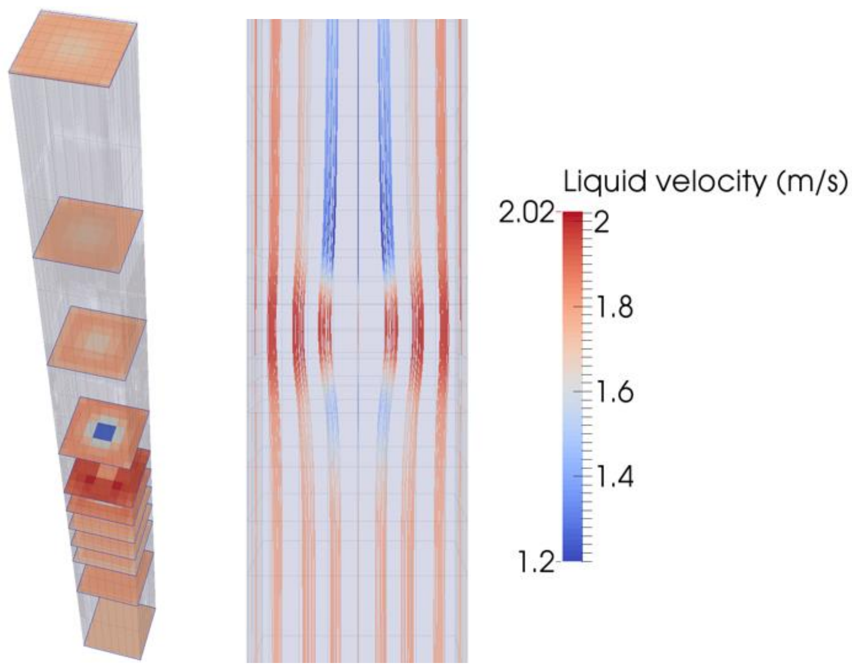
Validation of CUPID subchannel T/H models

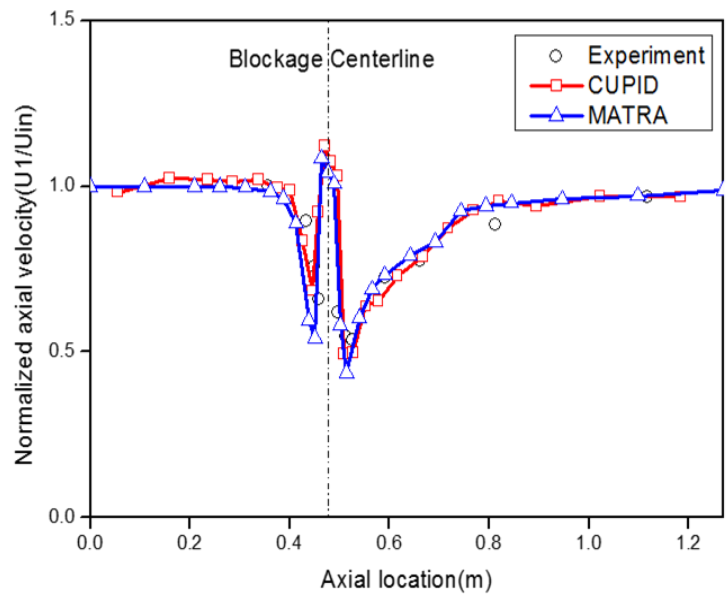
Validation of CUPID for unheated single-phase flow

- PNL 7x7 flow blockage test
  - 70 % area reduction at the center subchannels
  - Calculation results
    - Bypass flow in front of the blockage
    - Jet effect at blockage
    - Flow recovery by turbulent mixing after blockage



- Cross-sectional views of the test section



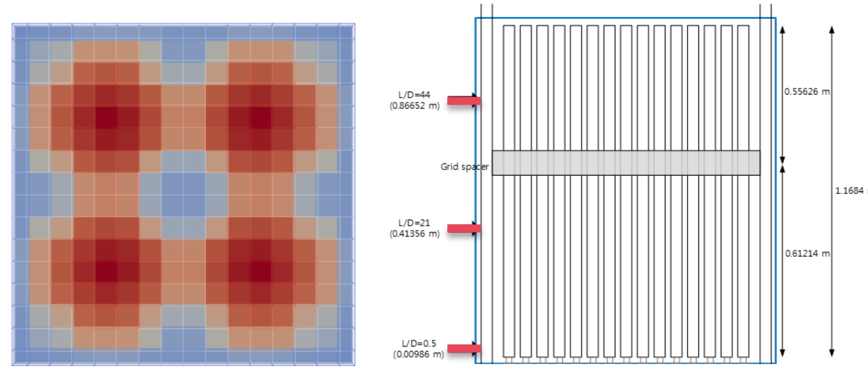


Creer J. M., Rowe D. S., Bates J. M., Sutey A. M., Effects of Sleeve Blockages on Axial Velocity and Intensity of Turbulence in an Unheated 7x7 Rod Bundle, Pacific Northwest Laboratories, BNWL-1965, 1976.

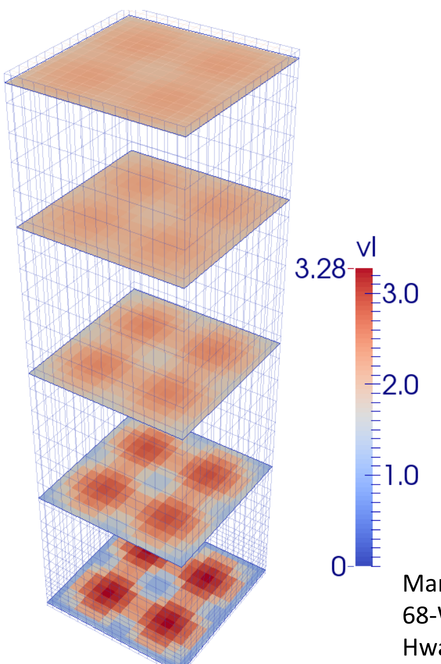
Hwang, Dae Hyun, Kyung Won Seo, and Hyouk Kwon. *Development of Validation System for Subchannel Analysis Codes under Steady-State PWR Conditions*. No. KAERI/TR--4143/2010. Korea Atomic Energy Research Institute, 2010.

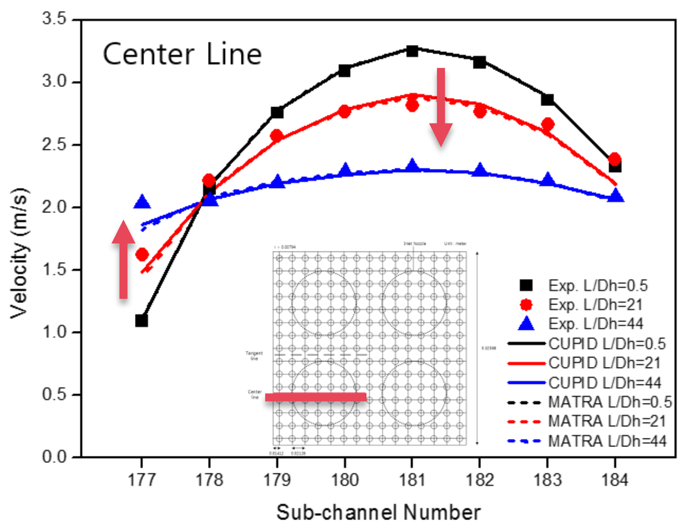
❖ Validation of CUPID for unheated single-phase flow

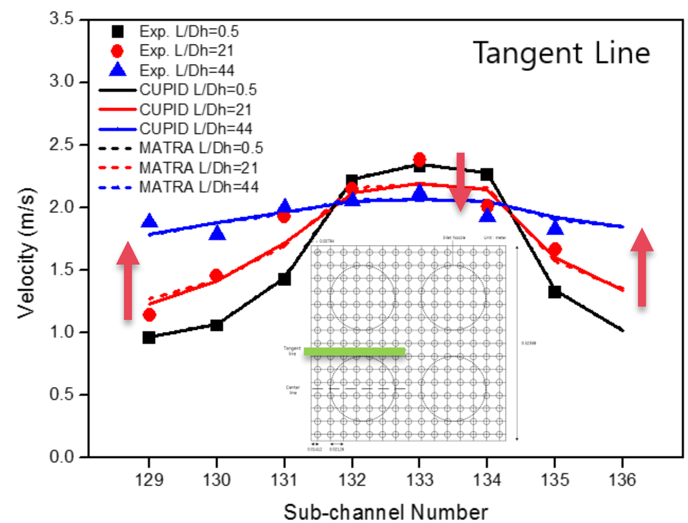
- CE 15x15 inlet jetting test
  - Effect of non-uniform inlet velocity
    - Flattening velocity distribution
    - Verifying the effect of cross flow and turbulent mixing
  - Errors in liquid velocity prediction
    - Along the center line: 8.2 %
    - Along the tangent line: 9 %



- Inlet velocity distribution and measurement elevations







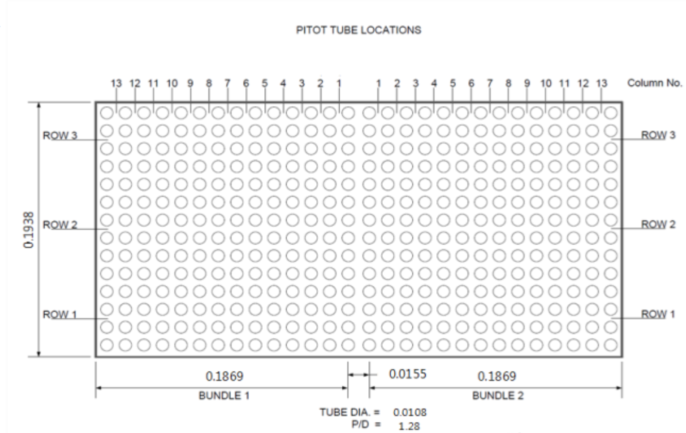
Marshall R. C., Letendre R. P., “Influence of Inlet Geometry on Flow in the Entrance Region of a Nuclear Reactor Rod Bundle”, ASME, 68-WA/HT-34, pp.1-8, 1969.

Hwang, Dae Hyun, Kyung Won Seo, and Hyouk Kwon. *Development of Validation System for Subchannel Analysis Codes under Steady-State PWR Conditions*. No. KAERI/TR--4143/2010. Korea Atomic Energy Research Institute, 2010.

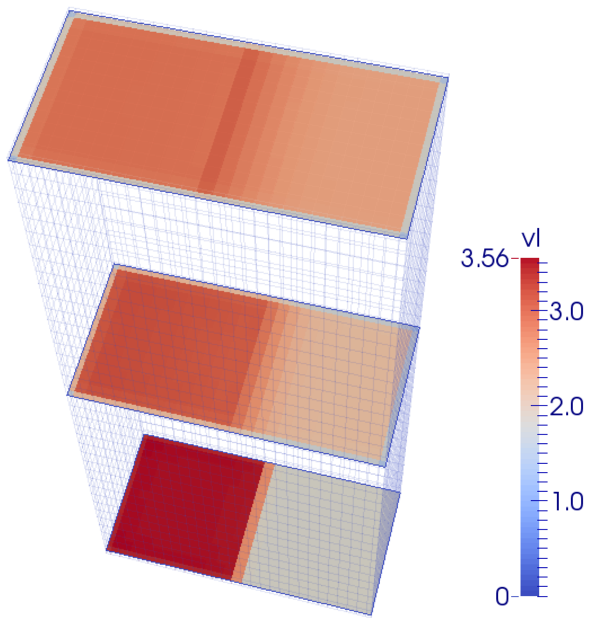
Validation of CUPID subchannel T/H models

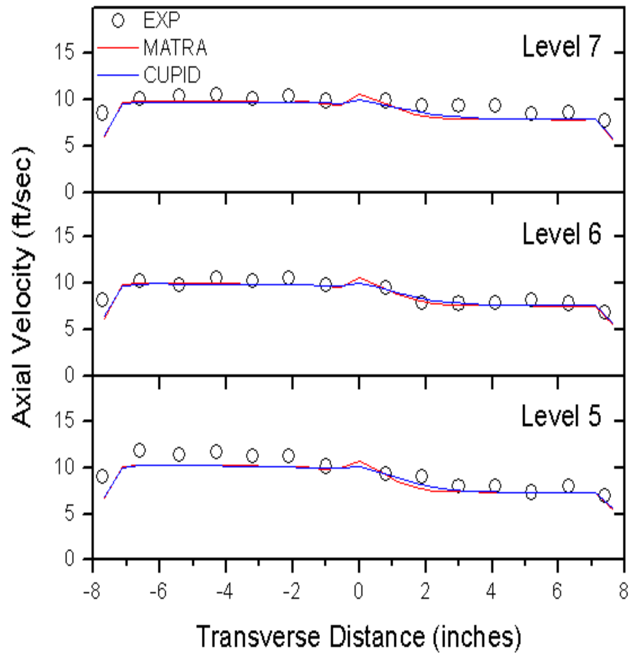
Validation of CUPID for unheated single-phase flow

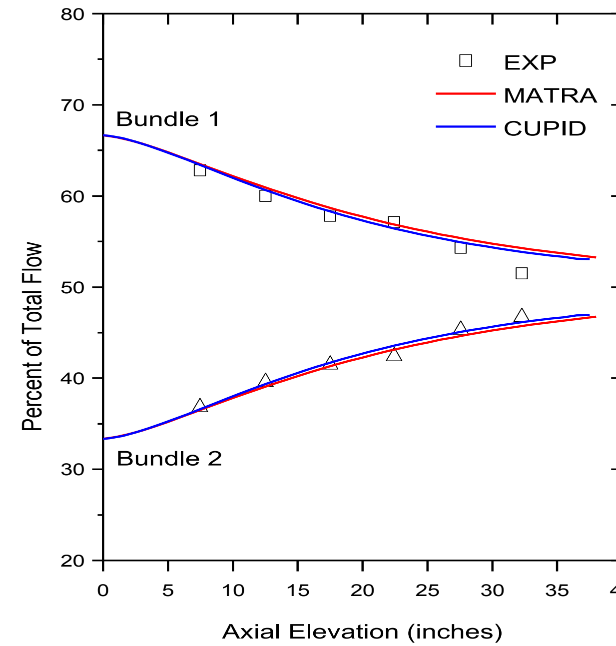
- Weiss et al.'s 14x14 two-assembly inlet blockage test
  - Partial blockage case (1100/550 gpm)
  - Flattening velocity distribution due to cross flow
  - Reasonable prediction of local velocity and fraction of flow at each assembly



- Cross-sectional view of the test section







Chelemer H., Chu P. T., Hochreiter L. E., THINC-IV: an Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores, WCAP-7956, Westinghouse Electric Corp., 1973.

Hwang, Dae Hyun, Kyung Won Seo, and Hyouk Kwon. *Development of Validation System for Subchannel Analysis Codes under Steady-State PWR Conditions*. No. KAERI/TR--4143/2010. Korea Atomic Energy Research Institute, 2010.

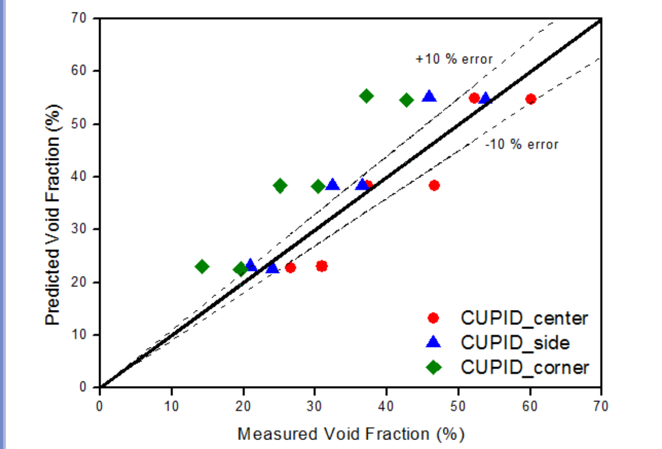
Validation of CUPID subchannel T/H models

❖ Validation of CUPID for unheated two-phase flow

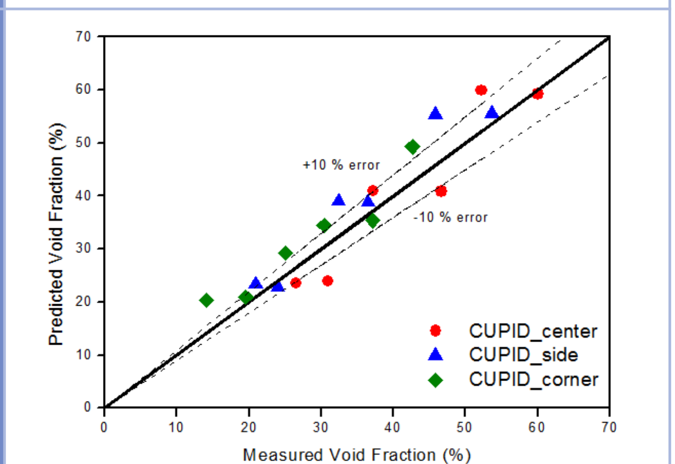
- RPI 2x2 air-water mixing test
  - Void drift phenomena under various flow regimes
    - Bubbly, slug and churn-turbulent flow

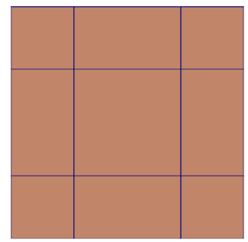
EVVD model
: Equal Volume exchange turbulent mixing and Void Drift model
: $[(\alpha_i - \alpha_j) - (\alpha_i - \alpha_j)_{EQ}]$

W/O EVVD

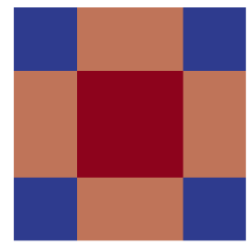


With EVVD

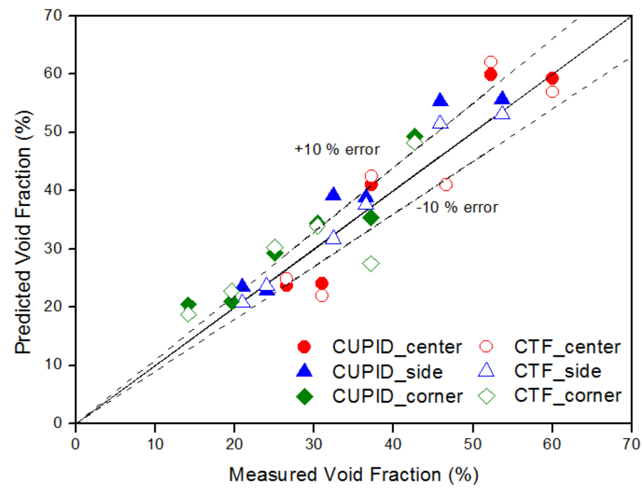




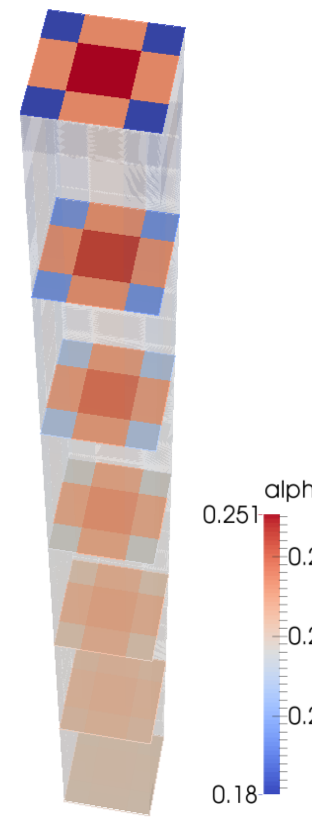
- W/O EVVD



- With EVVD



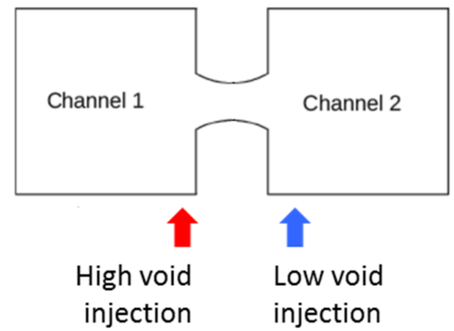
- CUPID vs. CTF

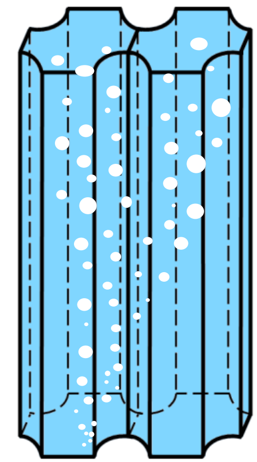


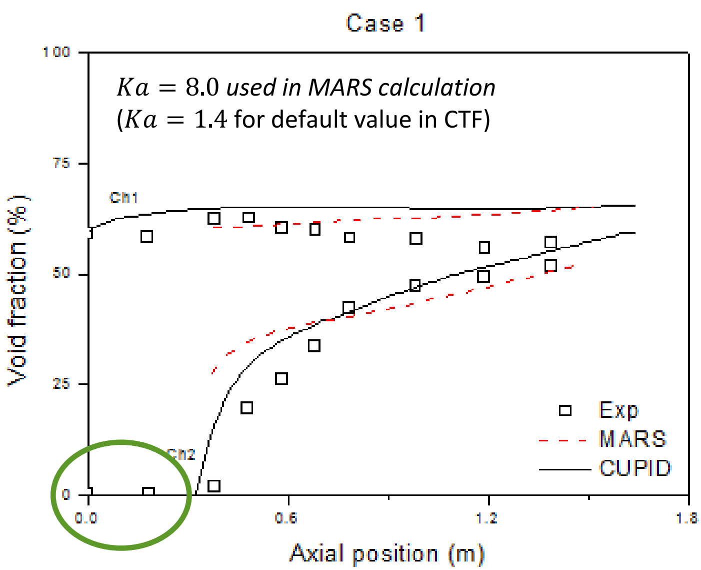
Sterner R. W., Lahey R. T. Jr., Air-Water Subchannel Measurements of the Equilibrium Quality and Mass Flux Distribution in a Rod Bundle, NUREG/CR-3373, 1983.

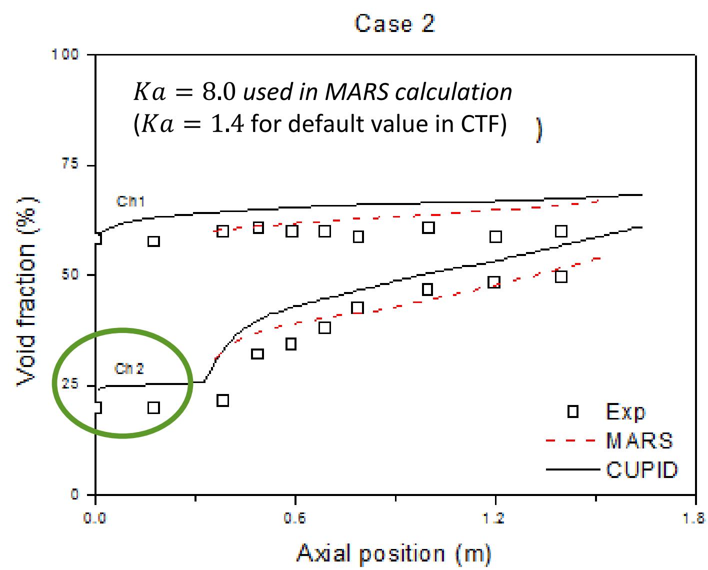
Validation of CUPID for unheated two-phase flow

- Tapucu two channel experiment
  - Asymmetric inlet flow conditions
  - Observed flow regime
    - Annular flow
  - CUPID vs. experiment vs. MARS-3D vessel







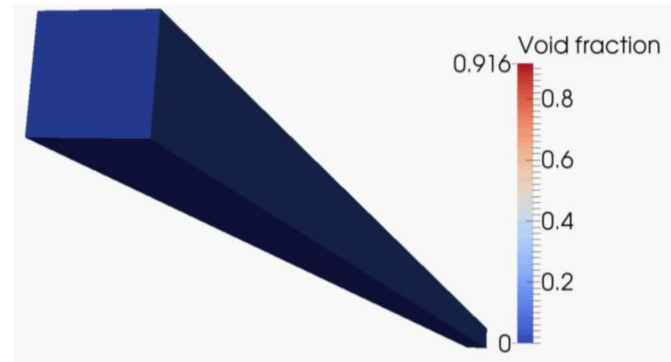


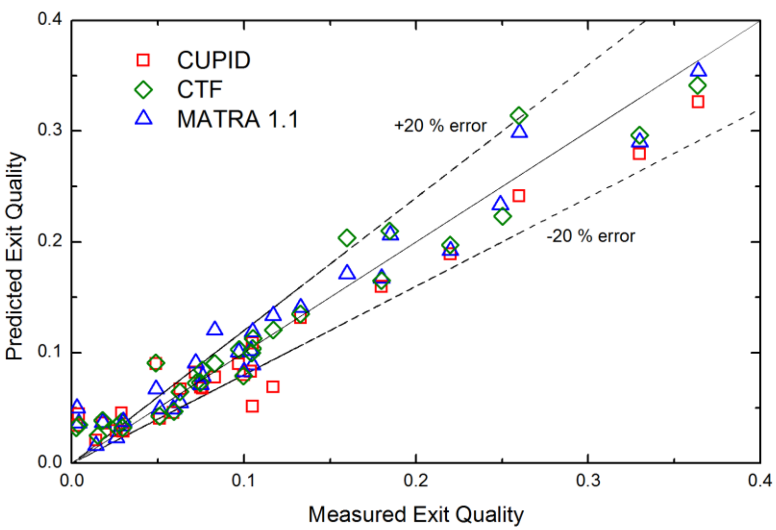
Tapucu A., Teyssedou A., Tye P., Troche N., "The effect of turbulent mixing models on the predictions of subchannel codes", Nuclear Engineering and Design, Vol. 149, pp. 221-231, 1994
Jeong J. J., Hwang D. H., Bae S. W., Chung B. D., "Assessment of the Inter-channel Mixing Model of the MARS Code", Transactions of the Korean Nuclear Society Autumn Meeting, 2005.

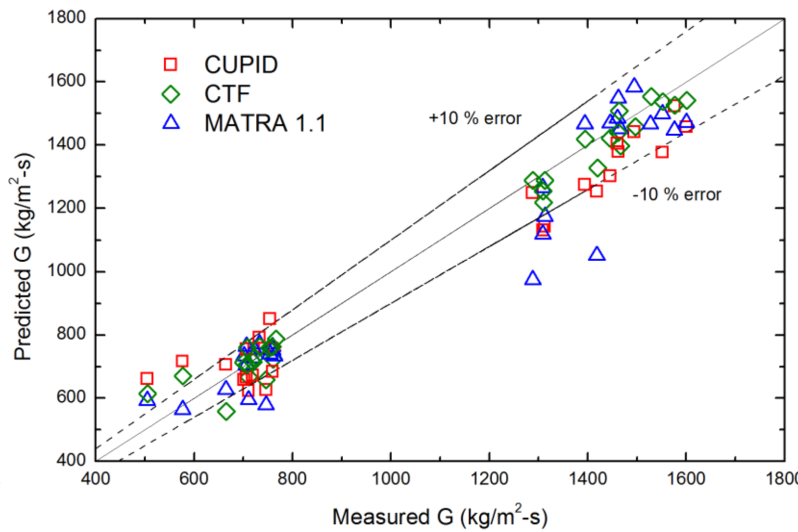
Validation of CUPID subchannel T/H models

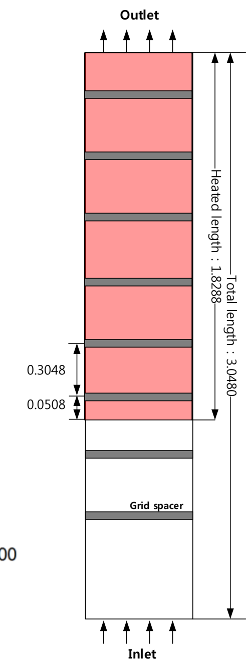
Validation of CUPID for heated two-phase flow

- GE 3x3 mixing test
  - Continuous change of flow regime conditions
    - Single-phase → two-phase (annular flow)
  - Errors in quality
    - CUPID: 0.003~0.055 ; CTF: 0.007~0.054
  - Errors in mass flux
    - CUPID: 13~31 % ; CTF: 5~22%









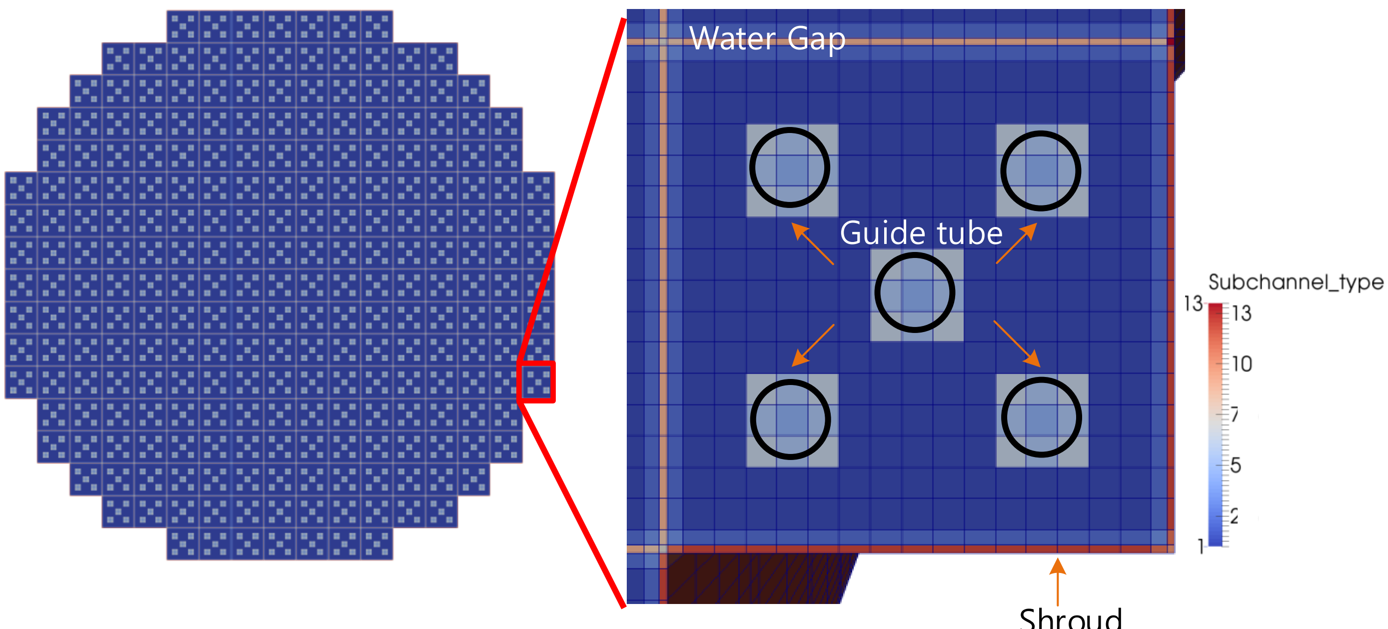
Lahey R. T. Jr., Shiralkar B. S., Radcliffe D. W., Two-phase flow and heat transfer in multirod geometries: subchannel and pressure drop measurements in a nine-rod bundle for diabatic and adiabatic conditions, GEAP-13049, General Electric, 1970.

Contents

Subchannel scale whole core T/H analysis

❖ Demonstration of the subchannel scale whole core analysis capability

- APR1400 reactor core
  - Detailed geometric information of APR1400
  - Subchannel type definition
    - Fuel rod, shroud, water gap and guide tube
  - Input geometry information following subchannel type
    - Porosity, permeability, hydraulic diameter and gap size



- Subchannel type definition

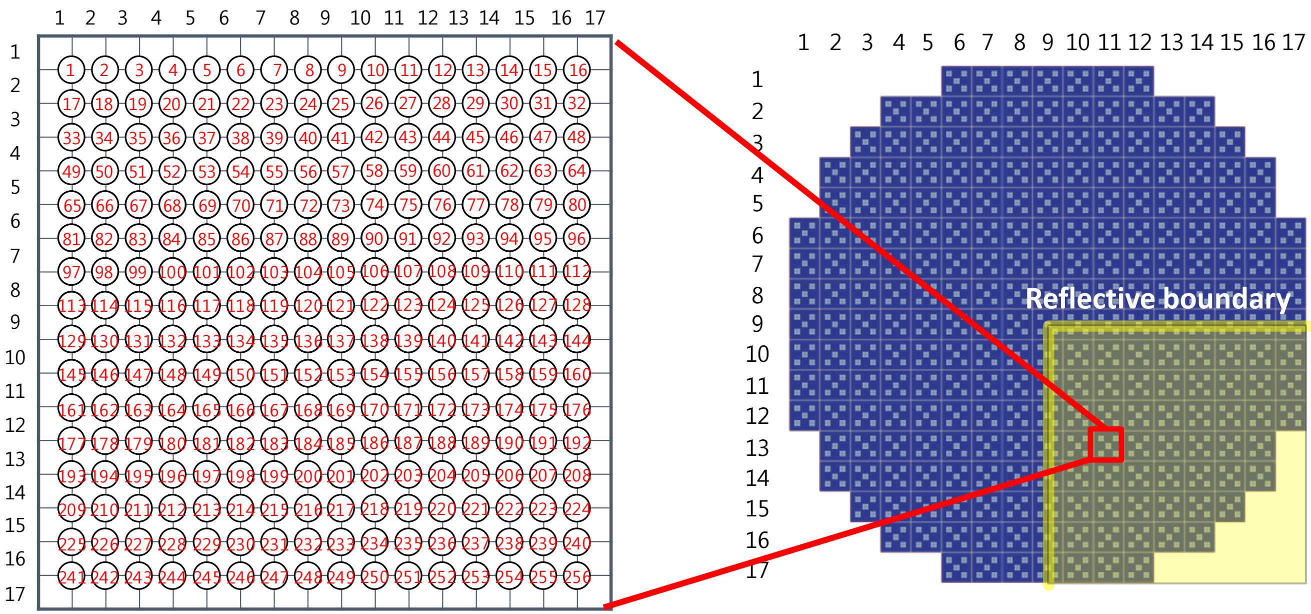
Type	Name
1	Assembly (center)
2	Assembly (side)
3	Assembly (corner)
4	Guide tube (center)
5	Guide tube (side)
6	Guide tube (corner)
7	Water gap (center)
8	Water gap (corner)
9	Water gap (side)
10	Shroud (edge)
11	Shroud (near edge)
12	Shroud (side wall)
13	Shroud (between gap)

Subchannel scale whole core T/H analysis

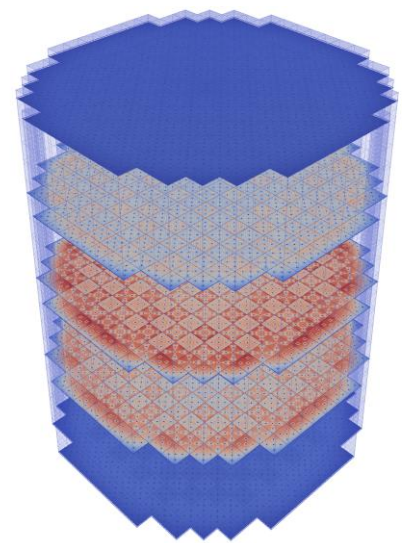
❖ Demonstration of the subchannel scale whole core analysis capability

- Imposed rod power from nTRACER calculation result
  - Subchannel to rod connectivity
    - Following the numbering convention of nTRACER
    - Each subchannel contains one, two or four rods.
    - A quarter rod power loaded at each subchannel

nTRACER
: Whole core neutron transport code



- Subchannel to rod connectivity



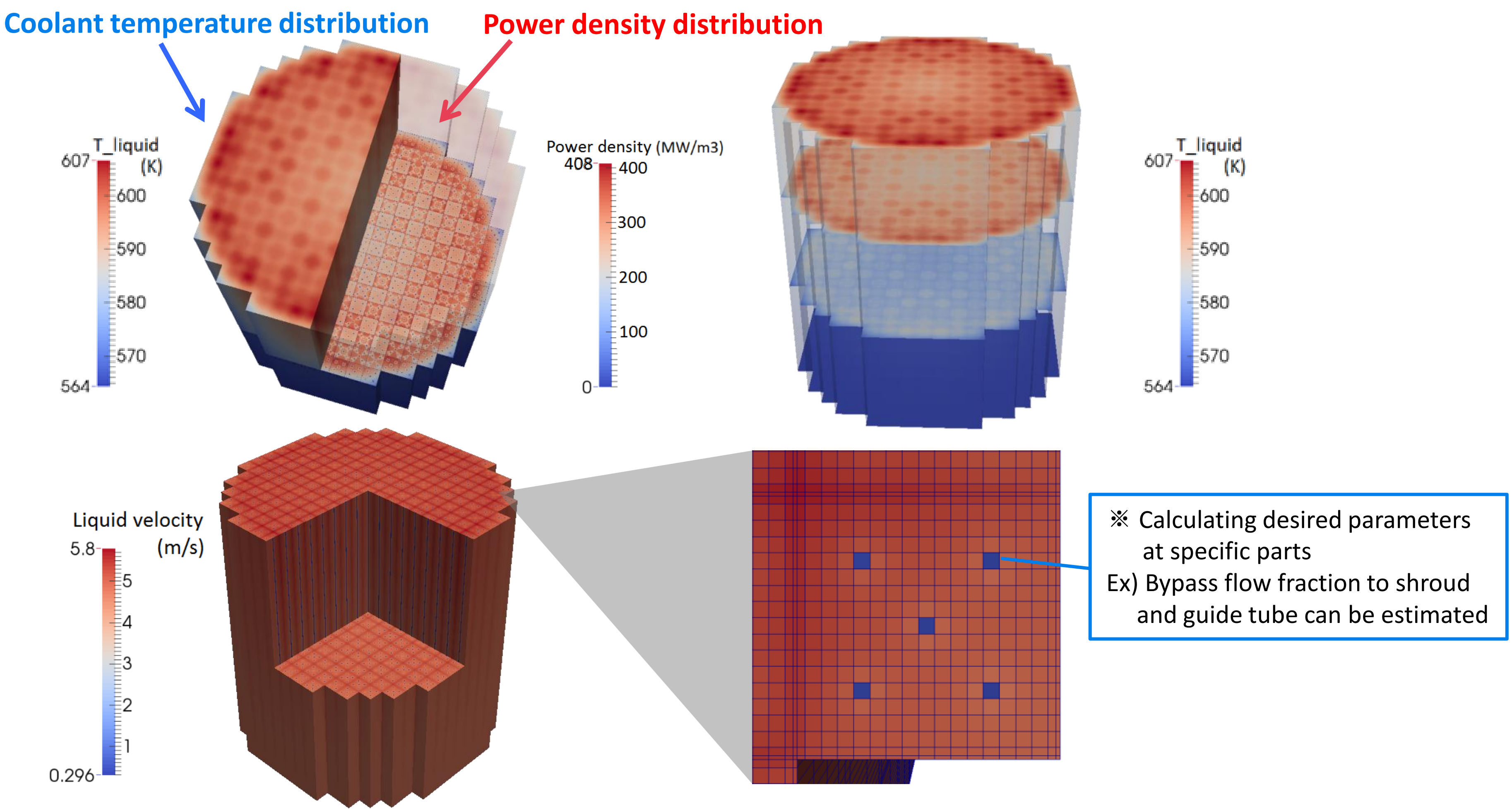
- Imposed power density distribution

Jung Y. S., Shim C. B., Lim C. H., Joo H. G., "Practical numerical reactor employing direct whole core neutron transport and subchannel thermal/hydraulic solvers", Annals of Nuclear Energy, Vol. 62, pp. 357-374, 2013.

Subchannel scale whole core T/H analysis

❖ Demonstration of the subchannel scale whole core analysis capability

- Cycle 1 hot full power state of APR1400



Subchannel scale whole core T/H analysis

❖ Demonstration of the subchannel scale whole core analysis capability

- Preliminary calculation result of whole core DNBR distribution
  - Using Biasi correlation
    - Minimum DNBR location can be found.

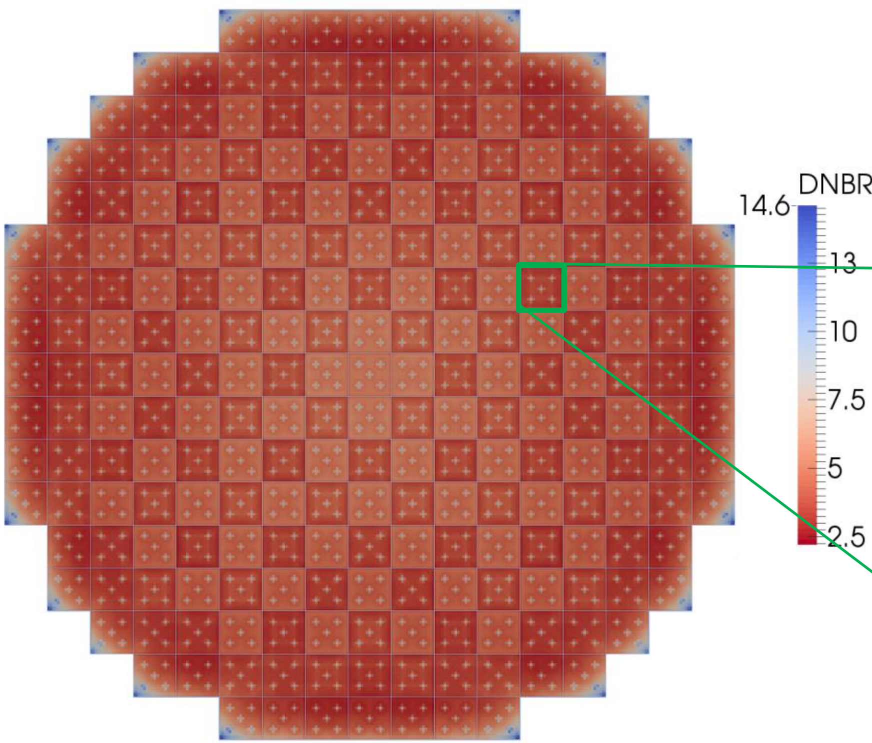
Biasi correlation

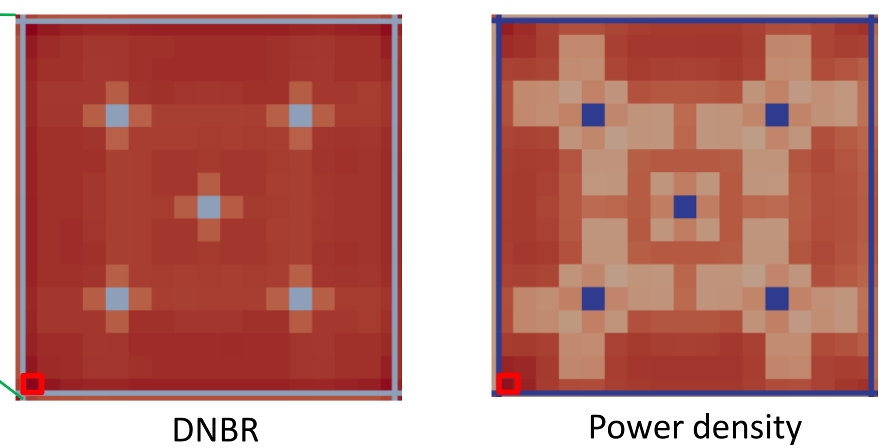
$$q''_{B1} = (5.9691 \times 10^6) G^{-1/6} \left(F(P) G^{-1/6} - X \right) D_H^{\;-n}$$
$$q''_{B2} = (11.98 \times 10^6) H(P)(1 - X) D_H^{\;-n} G^{-0.6}$$

Where,

$$F(P) = 0.7249 + 0.099 P exp(-0.032 P)$$
$$H(P) = -1.159 + 0.149 P exp(-0.019 P) + 8.99 P (10 + P^2)^{-1}$$

$$q''_{CHF} = Maximum(q''_{B1}, q''_{B2})$$



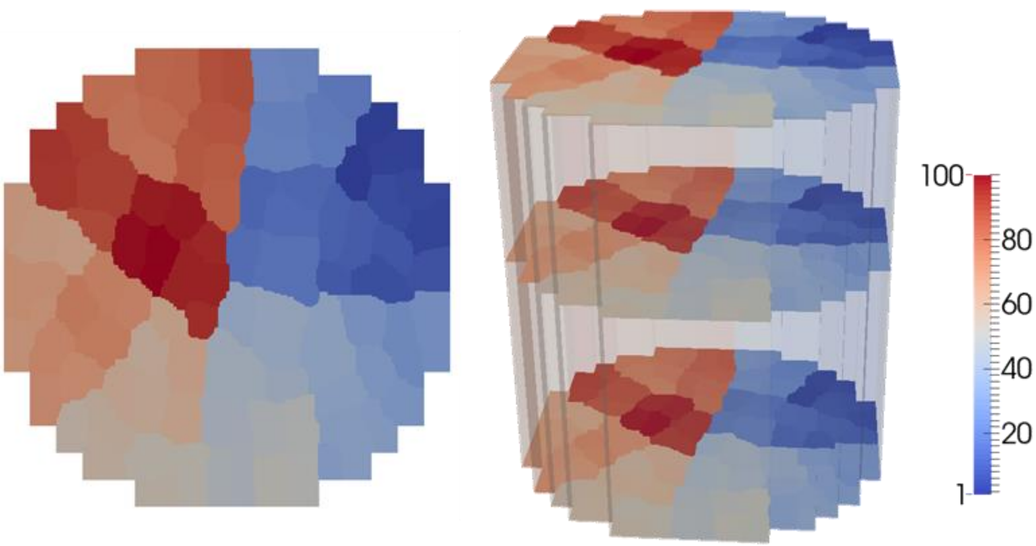


Demonstration of the subchannel scale whole core analysis capability

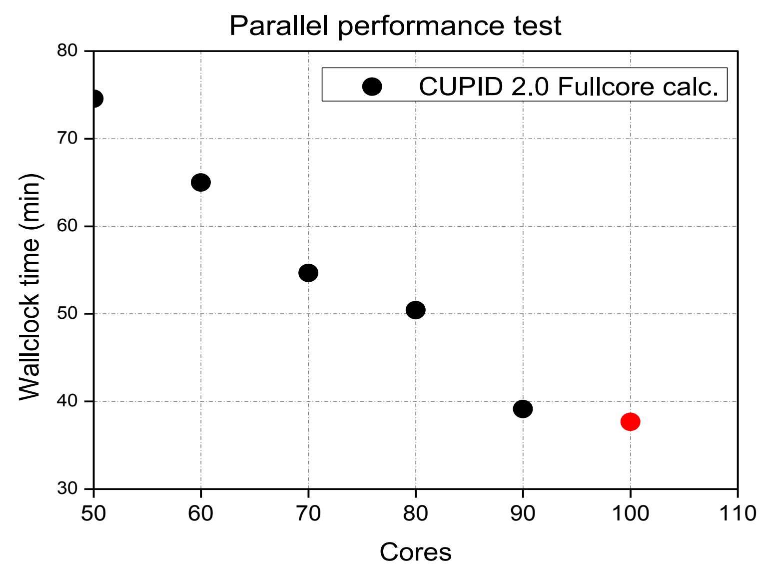
- MPI domain decomposition using METIS
  - Using 100 cores
  - Achieved good computational load balancing
  - Number of computing cells to each core: 33414~33469 (max. difference: 55)

Parameters	
Number of total cells	2,675,698
Problem time for steady-state	1.5 sec
Number of cores	100
Total wall-clock time	37.67 min

CPU : Dual Intel(R) Xeon(R) E5-2660
Dual Intel(R) Xeon(R) E5-2680



- Domain decomposition result



Contents

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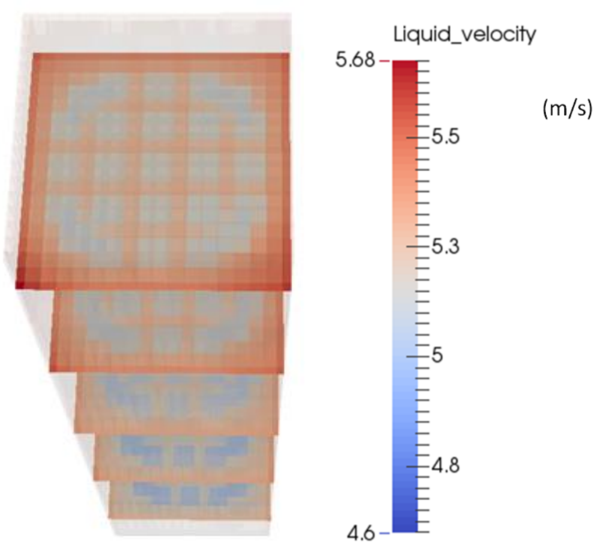
Extension of CUPID for subchannel scale TH analysis

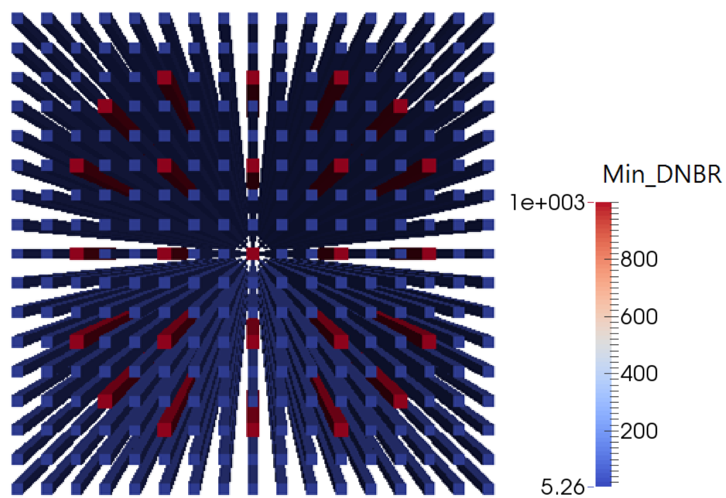
- Implementation of subchannel TH models
- Validation against experimental database
- Application of the extended code to whole reactor core TH analysis

Future works

- Systematic validation of two-phase flow models comparing with COBRA-TF results
  - Validation against two-phase flow experiment database
  - Implementation of fuel rod models



- Single assembly calculation result using CTF



- Minimum DNBR calculation result using CTF

Thank you for your attention

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Turbulent mixing models

Validation of subchannel analysis code MATRA (Kwon et al., 2014, KAERI)
CTF Theory Manual (Robert K. Salko et al., 2014, PSU)

- EM (Equal Mass exchange) model
  - Adiabatic single-phase problem
    - CUPID momentum equation

$$\frac{\partial}{\partial t}(\alpha_k\rho_k\vec{V_k})+\frac{\partial}{\partial x}(\alpha_k\rho_k u_k\vec{V_k})+\frac{\partial}{\partial y}(\alpha_k\rho_k v_k\vec{V_k})+\frac{\partial}{\partial z}(\alpha_k\rho_k w_k\vec{V_k})$$
$$=\alpha_k\rho_k\vec{g}-\alpha_k\nabla P+\nabla\cdot\left[\alpha_k\left(\tau_k^{ij}+T_k^{ij}\right)\right]+\vec{M}_k^L+\vec{M}_k^d+\vec{M}_k^T$$

Where, $\vec{M}_k^T=-\sum_J f_T w'_{ij}(U_i-U_j)$

$w'_{ij}=\beta\times s_{ij}\times\bar{G}$ (turbulent mixing rate)
β : Mixing coefficient, $f_T=1.0$

- EVVD (Equal Volume exchange and Void Drift) model
  - Diabatic single-phase and two-phase problem

$$\vec{M}_k^T=\varepsilon\frac{s_{IJ}}{z_{IJ}^T}(\rho_f v_f-\rho_g v_g)\theta\left[\alpha_{v,J}-\alpha_{v,I}-(\alpha_{v,J}-\alpha_{v,I})_{equil}\right]$$ in momentum eq.

$$\vec{M}_e^T=\varepsilon\frac{s_{IJ}}{z_{IJ}^T}(\rho_f-\rho_g)\theta\left[\alpha_{v,J}-\alpha_{v,I}-(\alpha_{v,J}-\alpha_{v,I})_{equil}\right]$$ in mass eq.

$$\vec{M}_h^T=\varepsilon\frac{s_{IJ}}{z_{IJ}^T}(\rho_f h_f-\rho_g h_g)\theta\left[\alpha_{v,J}-\alpha_{v,I}-(\alpha_{v,J}-\alpha_{v,I})_{equil}\right]$$ in energy eq.

Key subchannel T/H models (2/3)

- Turbulent mixing model – Equal Mass exchange model
  - Generally used at single-phase problem
    - MATRA axial momentum equation

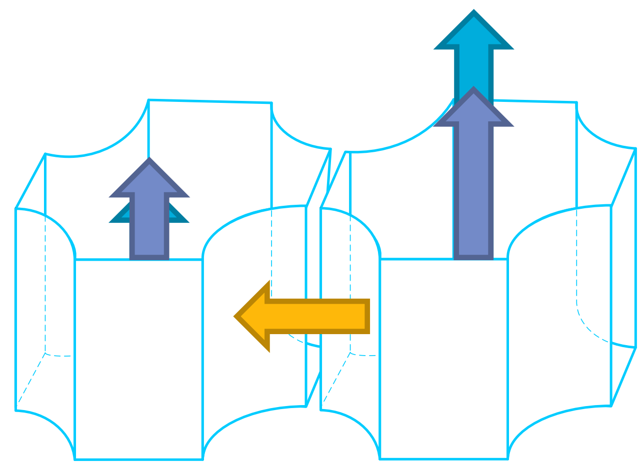
$$\frac{\partial \dot{m}}{\partial t} + \frac{\partial}{\partial x}\left(\frac{\dot{m}^2}{\rho' A_x}\right) + \sum_j \left(W_{ij} U'_{ij}\right) = -\bar{A}_x \frac{\partial P}{\partial x} - \frac{1}{2}\left[\frac{f}{d_{hy}} + K'\right]\left(\frac{\dot{m}^2}{\rho' A_x}\right) - \bar{A}_x \rho_m g \cos\theta - \sum_J f_T w'_{ij}\left(U_i - U_j\right)$$

  - Turbulent mixing rate

$$w'_{ij} = \beta \times s_{ij} \times \bar{G}$$

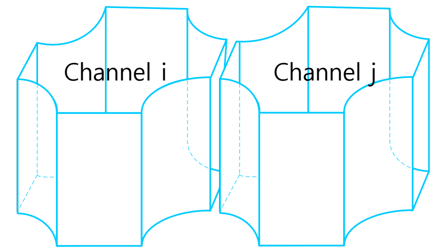
β : User input

$f_T = 1.0$



Key subchannel T/H models (3/3)

- Turbulent mixing model – Equal Volume exchange and Void Drift model
  - Generally used at two-phase problem
  - Considered void drift as an additional interchannel mixing effect to turbulent mixing

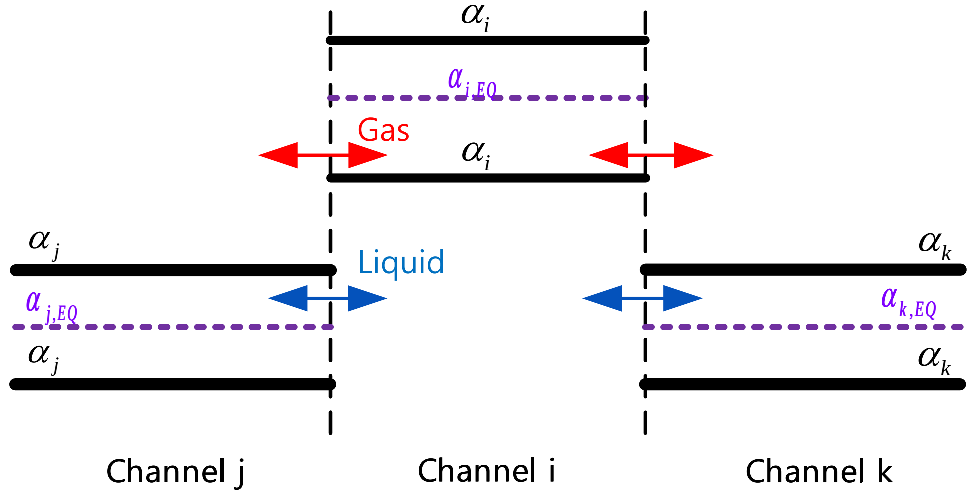


$$\left[(\alpha_i - \alpha_j) - (\alpha_i - \alpha_j)_{EQ}\right]$$

Turbulent mixing   Void drift

Lahey's derivation

$$(\alpha_{v,i} - \alpha_{v,j})_{\text{equil}} = K_a(\alpha_{v,i} + \alpha_{v,j})\frac{G_i - G_j}{G_i + G_j}$$



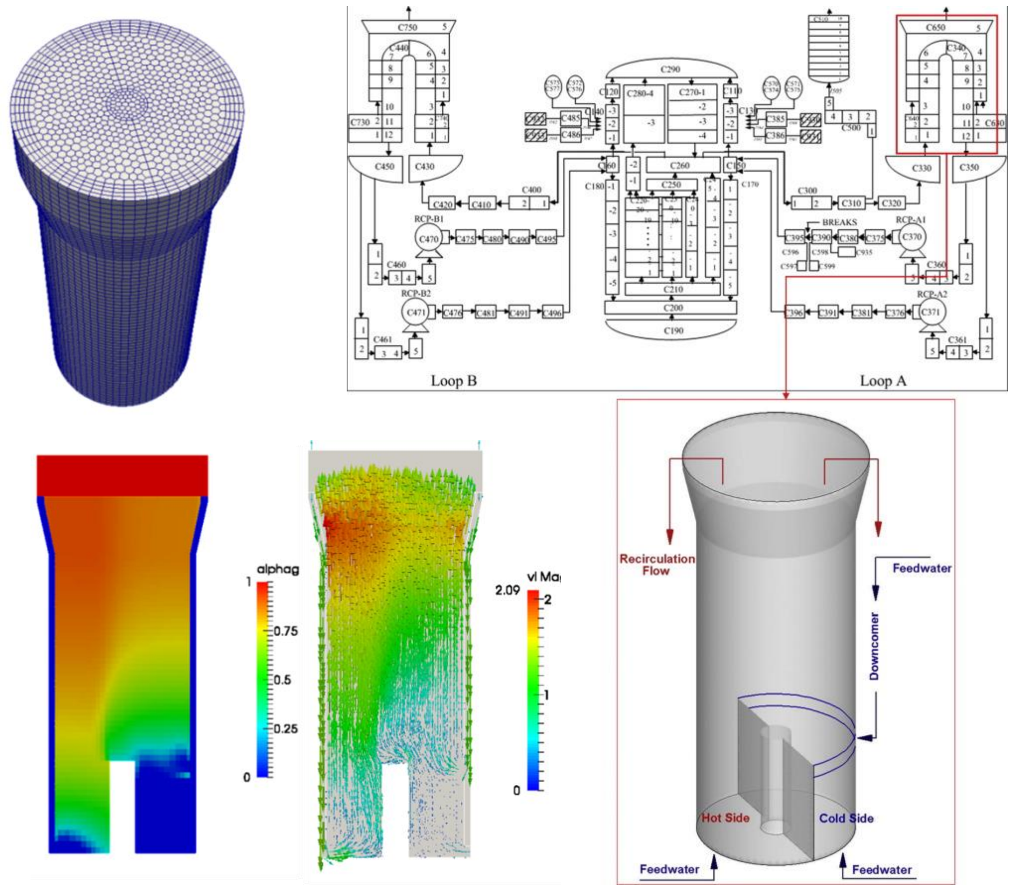
- Numerical study of void drift in rod bundle with subchannel and CFD codes (Bo pang, 2013)
- CTF Theory Manual (R.K. Salko et al., 2014)

CUPID Development Status

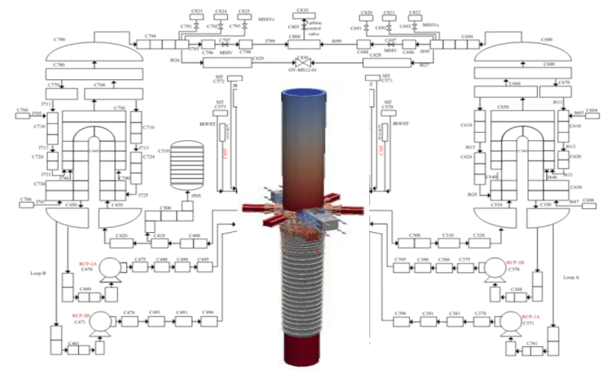
❖ Multi-scale, multi-physics analysis capability

- CUPID-MARS coupled simulation
- CUPID-MASTER, CUPID-DeCART

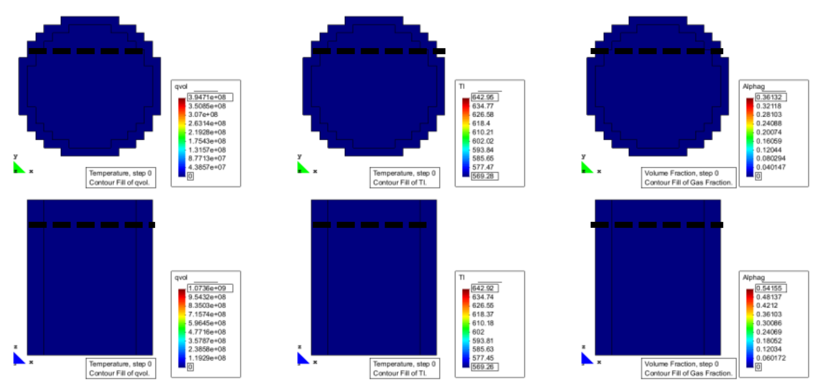
CUPID-MARS coupled simulation for PWR steam generator
(heat structure coupling)



CUPID-MARS coupled simulation for ATLAS experiment
(flow coupling)



CUPID-MASTER coupled simulation for rod ejection

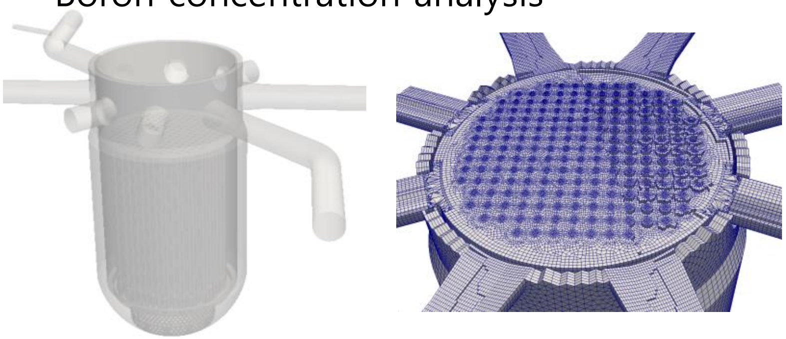


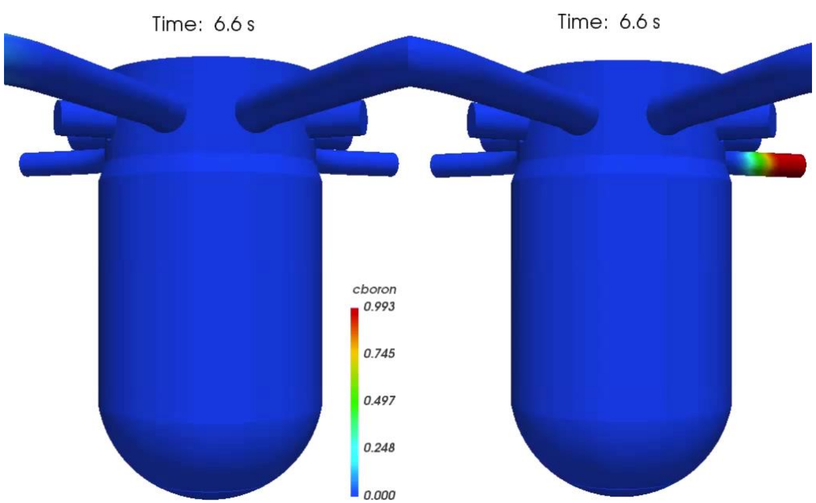
Heat generation　Liquid temperature　Void fraction

CFD scale analysis for reactor pressure vessel and core

ROCOM analysis

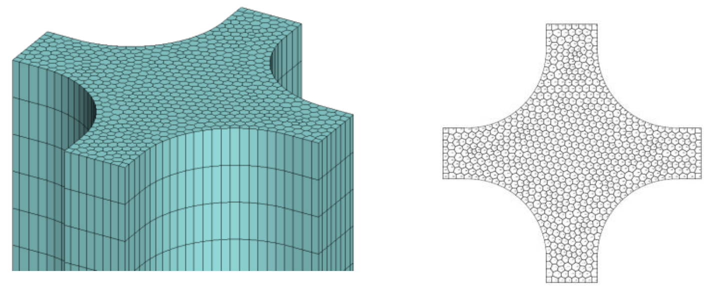
- Hexahedron + Tetrahedron: 3,434,527 cells
- Boron concentration analysis

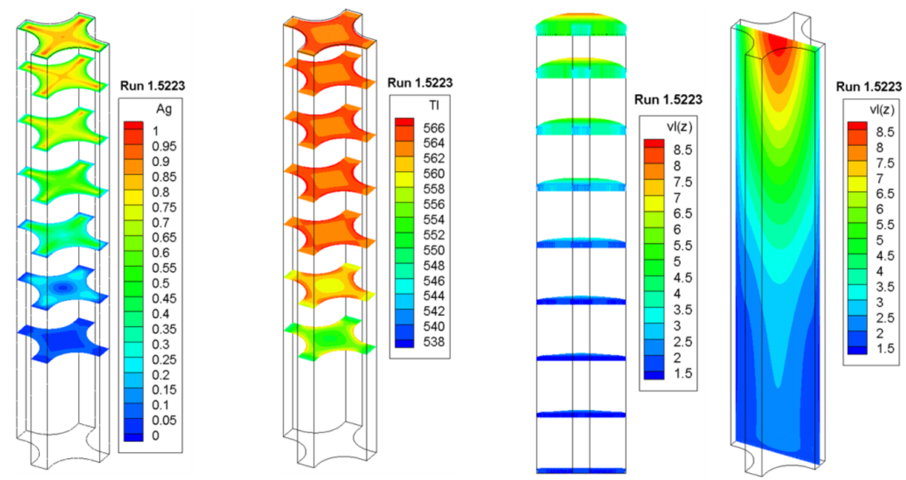




PSBT Run No. 1

- Hexahedron: 55,979 cells
- Channel averaged void fraction analysis



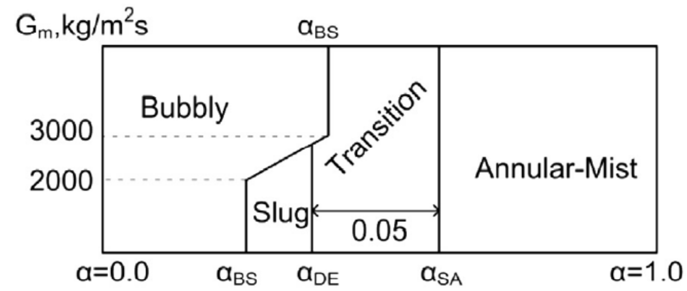


<Void fraction > <Liquid temperature> <Liquid velocity>

Constitutive models for two-phase flow in CUPID-SG

- Based on physical models in the system analysis code, MARS
  - Flow regime map (for vertical channel)
  - Interfacial area concentration
  - Interfacial drag
  - Interfacial heat transfer
  - Wall heat transfer (for pre-CHF heat transfer)
  - Heat partitioning

Ex) Flow regime map

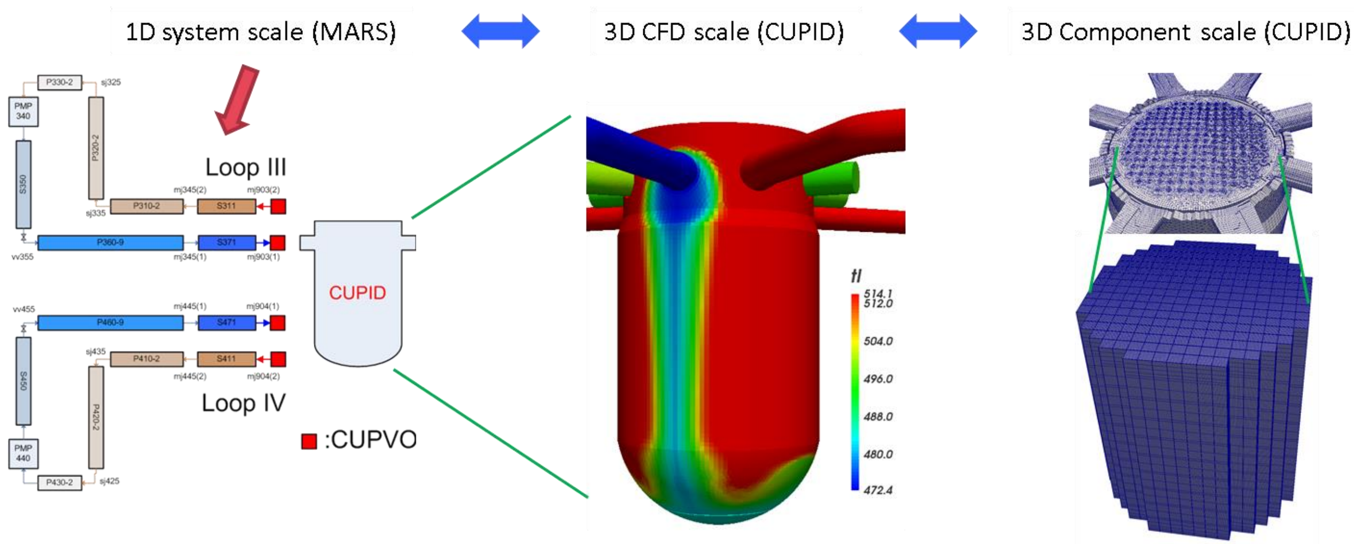


Ex) Interfacial drag

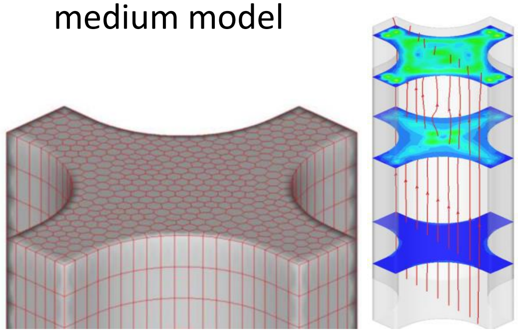
	Small bubbles in bubbly or slug flow	Taylor bubbles in slug flow	Annular flow
Drag force per unit volume	$F_{i,SB} = \frac{1}{8}\rho_c \lvert v_R \rvert v_R C_{D,SB} a_{i,SB}$	$F_{i,TB} = \frac{1}{2}\rho_f \lvert v_R \rvert v_R C_{D,TB} \frac{\alpha_{TB}}{L}$	$F_{i,ANN} = \frac{1}{2}\rho_g \lvert v_R \rvert v_R f_{i,ANN} a_{i,ANN}$
Drag coefficient	$C_{D,SB} = \frac{24}{Re_b}\left(1.0 + 0.1 Re_b^{0.75}\right)$ where, $Re_b = \frac{\rho_l d_b \lvert v_R \rvert}{\mu_{mb}}, \mu_{mb} = \frac{\mu_l}{1-\alpha_g}$	$C_{D,TB} = 10.9\alpha_{TB}^{0.5}(1-\alpha_{TB})^3$	$f_{i,ANN}$ is given by Churchill, Fore, Asali for flow regime.

Development of CUPID-SG for the analysis of two-phase flows in PWR steam generators, Kim et al., (2014)

Plan for multi-scale analysis using CUPID and need for subchannel scale analysis



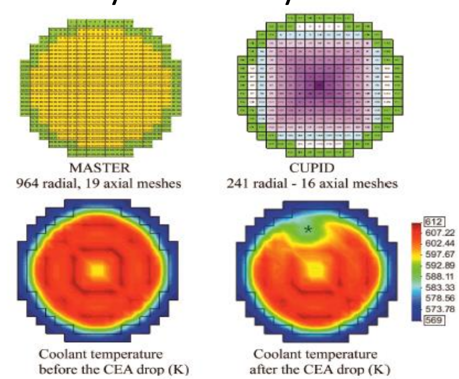
- CFD scale analysis for open medium model



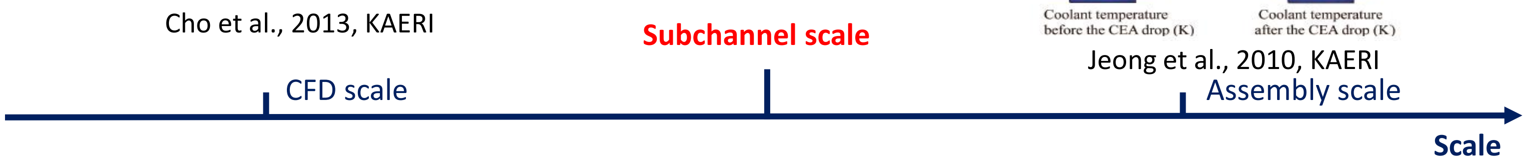
Cho et al., 2013, KAERI

Desired for coupled simulation with endurable computational time and reasonable accuracy

- Assembly scale analysis : CUPID-MASTER code



Jeong et al., 2010, KAERI

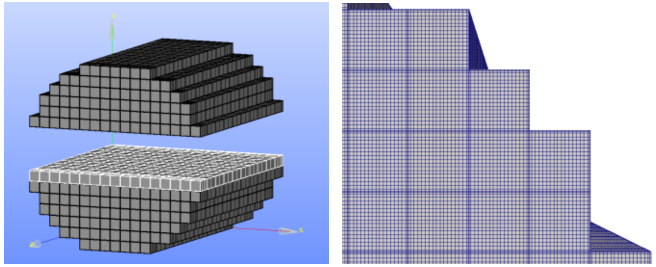


Purpose

- Three-dimensional TH analyses for nuclear reactor components
  - Reactor vessel, steam generator, water pool in passive cooling system, Calandria tank, etc.

Numerical scheme

- FVM (Finite Volume Method) on unstructured mesh
- Non-Staggered (Cell-Centered)
- Semi-implicit or Implicit numerical schemes



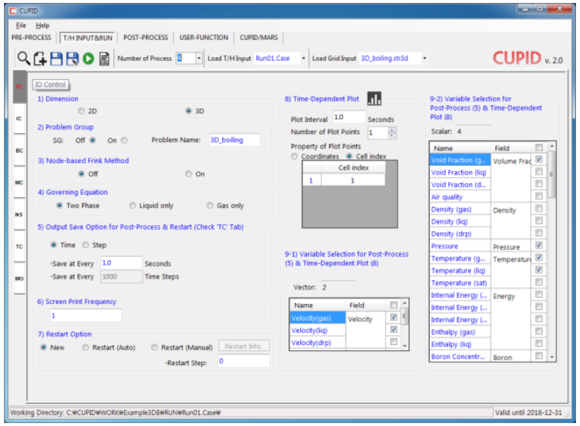
SALOME mesh

Parallelization

- Domain Decomposition, communication using MPI

Pre/Post Processing of CUPID

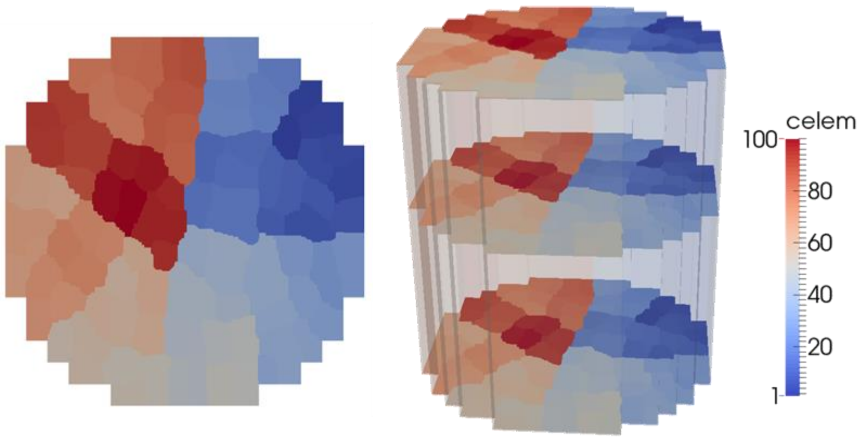
- OpenFoam input data structure
- Open source mesh generator: SALOME
- Post Processing with Paraview and GLOVE (KISTI)
- User interface workbench

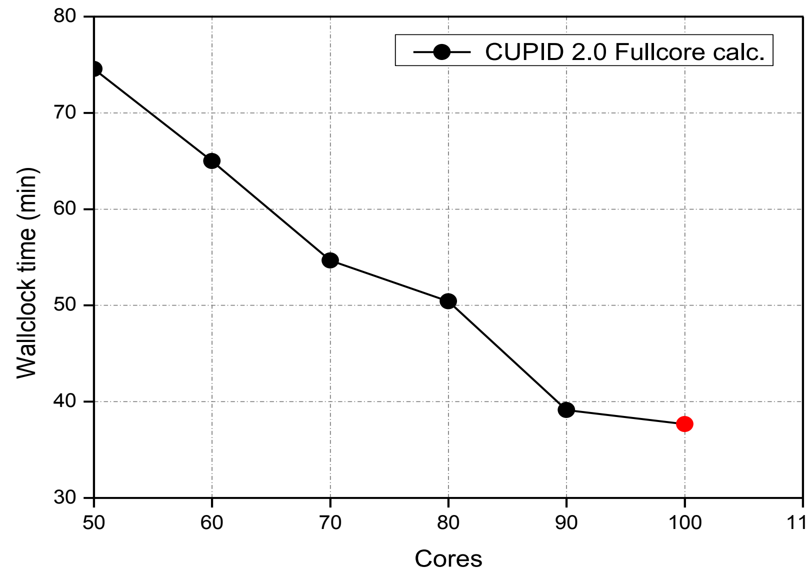


CUPID workbench

Demonstration of the subchannel scale whole core analysis capability

- MPI domain decomposition using METIS
  - Node balancing



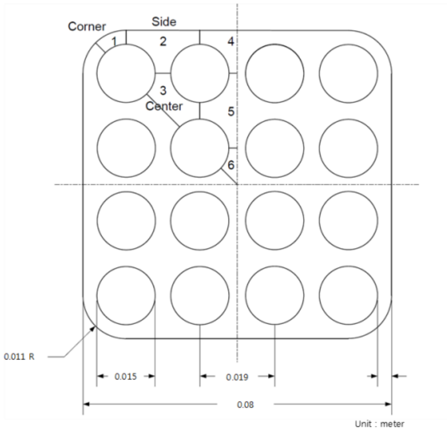


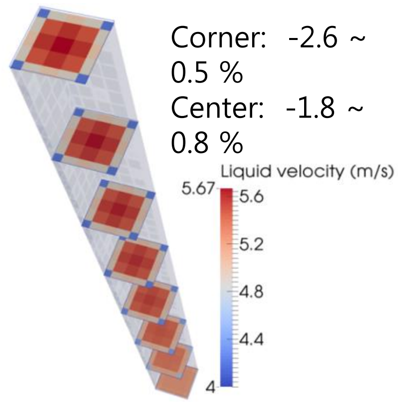
	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6
Cores	50	60	70	80	90	100
Calc. time (min)	74.57 min	64.99 min	54.66 min	50.42 min	39.14 min	37.67 min

Validation of CUPID subchannel T/H models

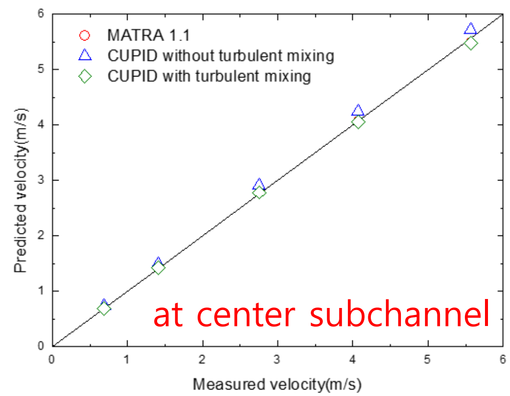
Validation of CUPID for unheated single-phase flow

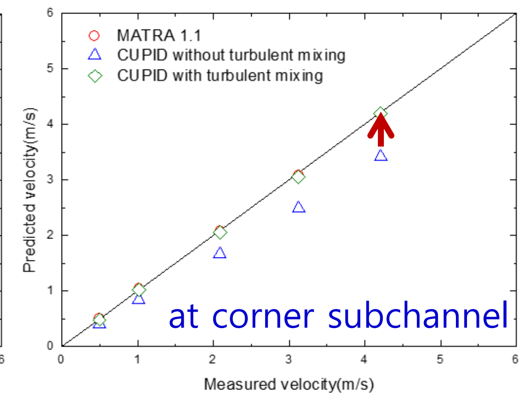
- CNEN 4x4 mixing test



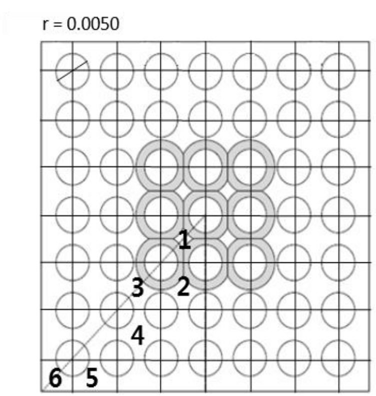


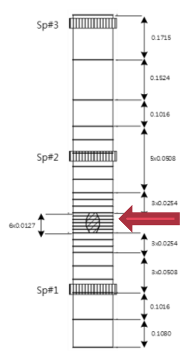
Corner: -2.6 ~ 0.5 %
Center: -1.8 ~ 0.8 %



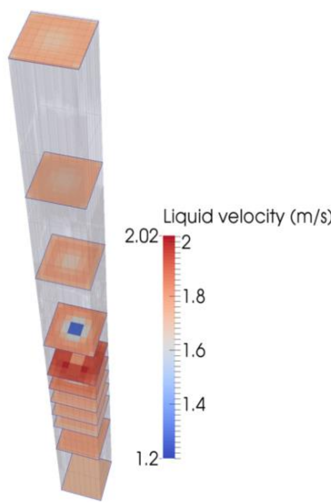


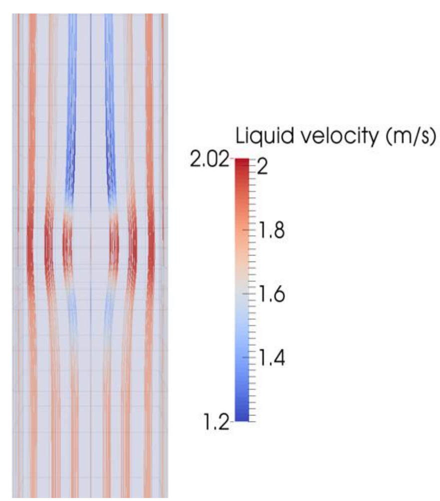
- PNL 7x7 flow blockage test

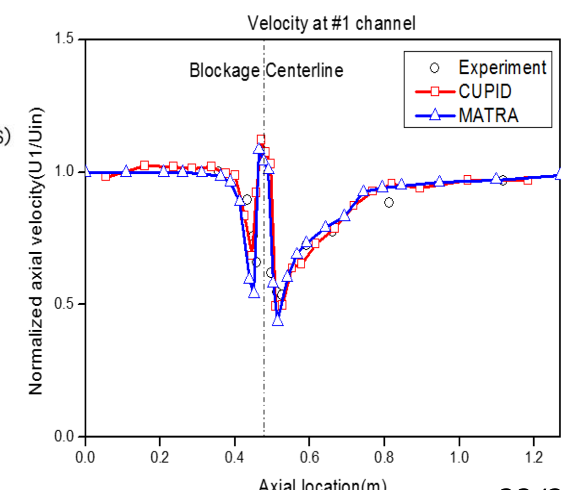




Cross-sectional views of the test section





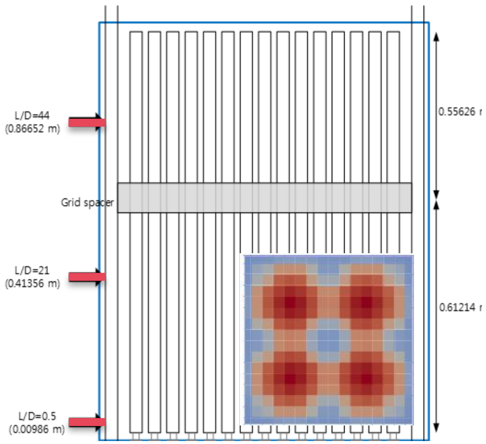


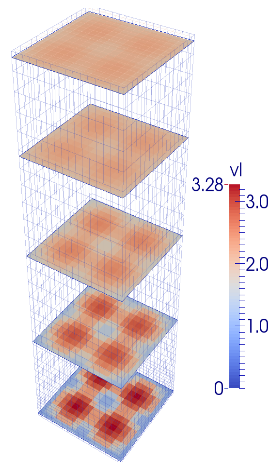
Validation of CUPID for unheated single-phase flow

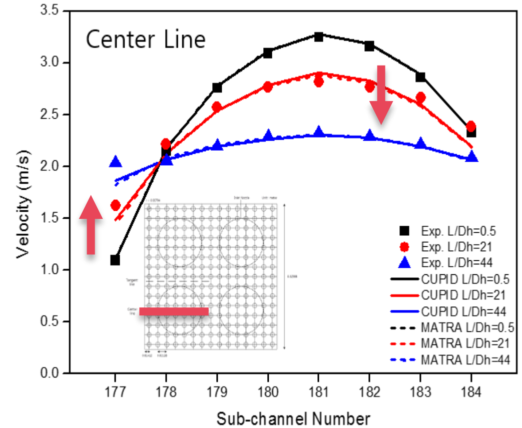
- CE 15x15 inlet jetting test

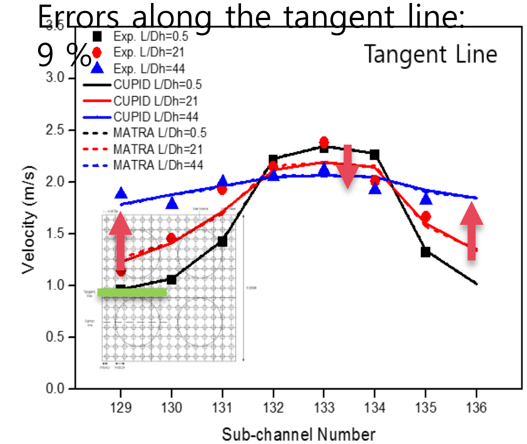
Errors along the center line: 8.2 %

Errors along the tangent line: 9 %









Inlet velocity distribution and measurement elevations

- Weiss et al.'s 14x14 two-assembly inlet blockage test