

# Evaluation of CHF correlation Model Performance in MATRA for Various Rod Bundle Geometries

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## 1. Introduction

- **Critical Heat Flux (CHF)** is a key thermal design parameter determining heat transfer and thermal margin in a reactor core. To predict this, various CHF correlations have been developed based on experimental data.
- However, because CHF in a fuel assembly is governed by multiple variables, **current CHF correlations are highly dependent on specific geometries** and radial/axial power distributions [1].
- To develop an enhanced CHF prediction model, **the performance of pre-existing CHF correlations and their influencing factors** were analyzed using the MATRA code according to fuel assembly designs and thermal-hydraulic conditions.

### MATRA Code Simulation

CHF Correlation

Performance by Rod Bundle Geometry

Performance influencing factors

## 2. Method

### CHF Correlation

### Rod array

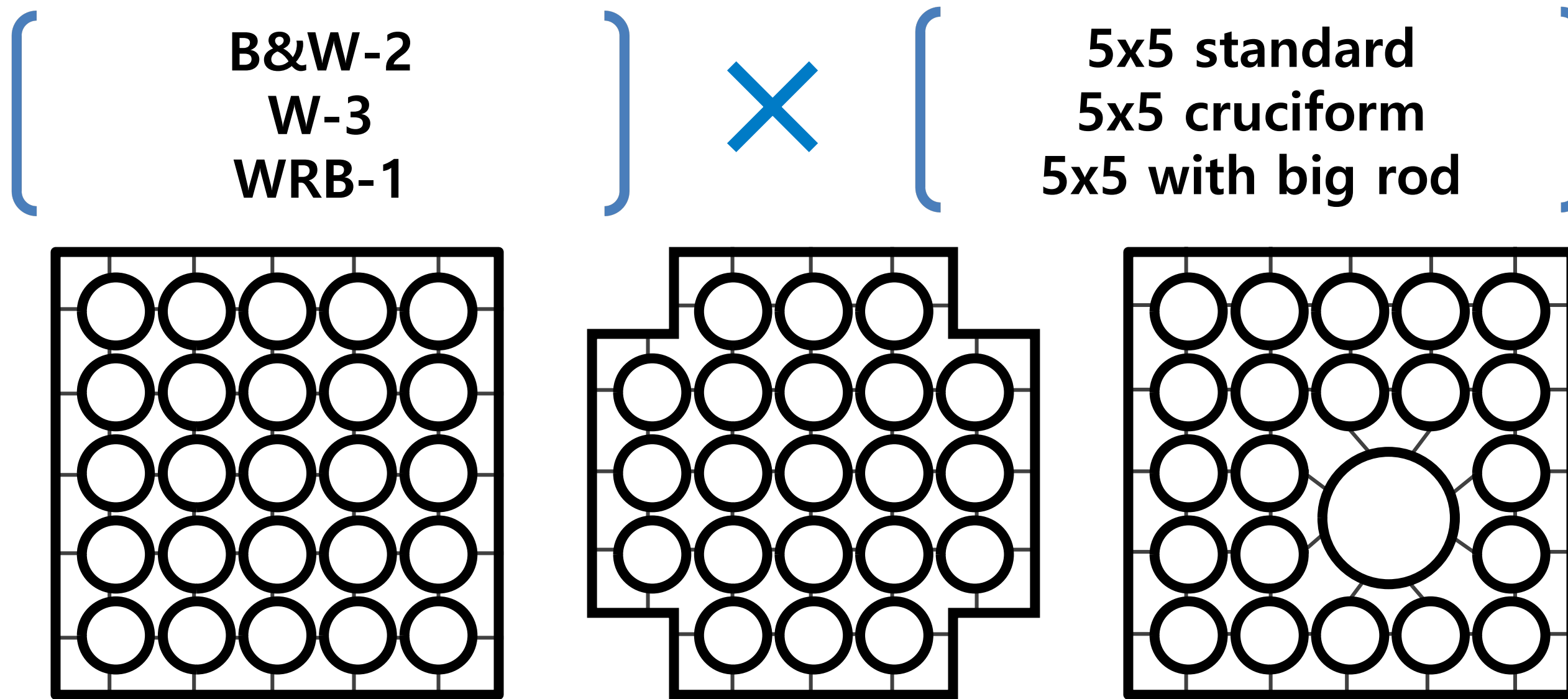


Fig 1. Geometries of rod configuration

- CHF database collected by EPRI [2], total 3,104 experimental data of three fuel assembly geometries were analyzed by MATRA.
- The simulations were performed using the iteration on power function.

Variables	Range	Thermal-hydraulic analysis	Correlation models
Inlet mass velocity	67.81 ~ 6046 [kg/m <sup>2</sup> · s]	Subcooled void fraction	Homogeneous model
Inlet temperature	303.1 ~ 445.2 [K]	Bulk void fraction	Chehal-Lellouche model
Exit pressure	1.345 ~ 16.79 [MPa]	Two-phase friction multiplier	Homogeneous model
Rod diameter	9.5 ~ 11.18 [mm]	Type of void draft correction factor	KAERI (Independent of flow regime)
Rod length	1219 ~ 4267 [mm]		
Rod to rod gap	1.75 ~ 4.06 [mm]		
Axial geometry slice gap	29 ~ 101.6 [mm]		
Axial power profile (ratio)	Uniform, Non-uniform (0.296 ~ 1.63)		

Table II. Models employed to compute the parameters related to the two-phase flow

Table I. Inlet conditions of selected CHF database

## 3. Results and Discussion

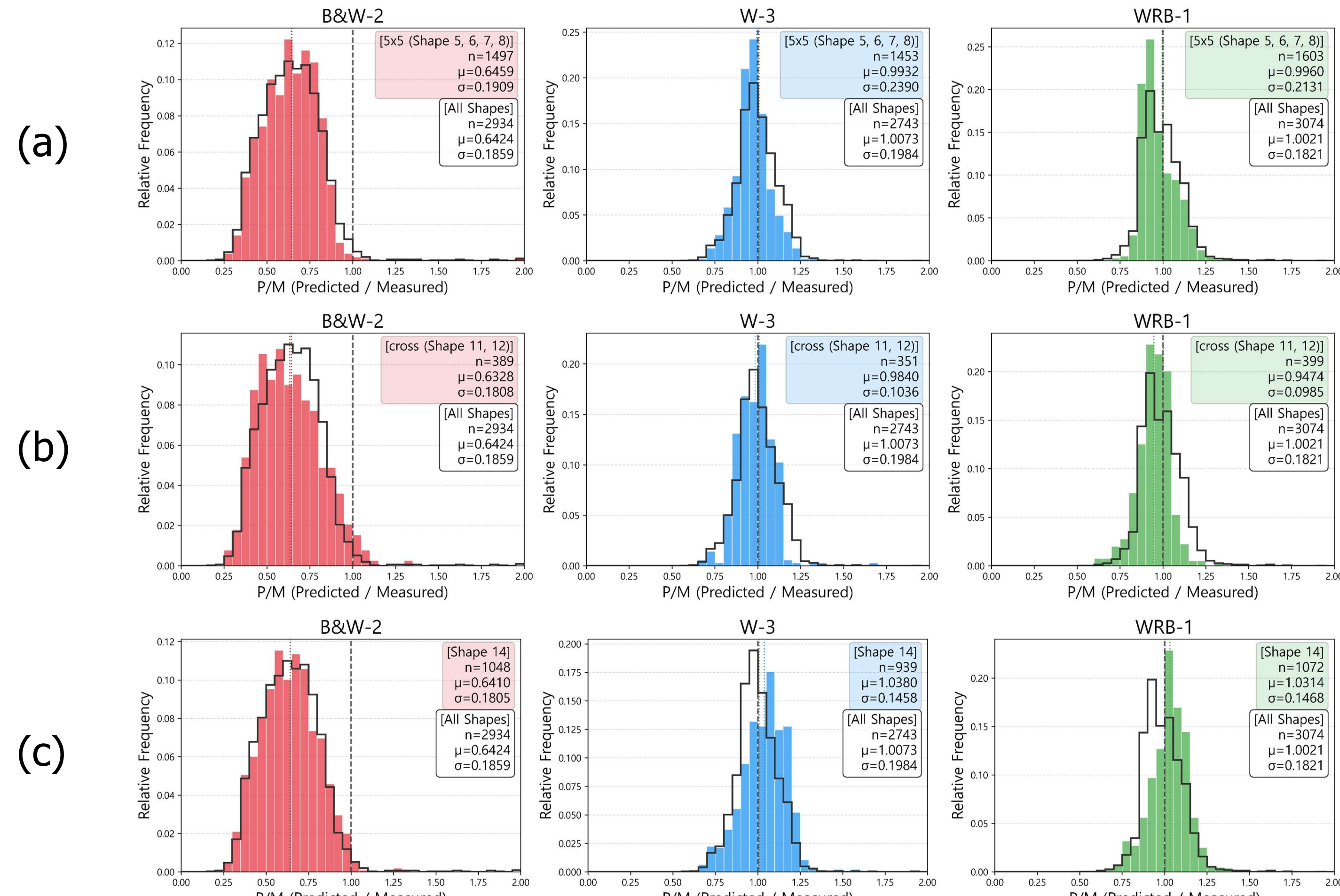


Fig 2. Prediction data histogram according to fuel rod assembly geometries (a) 5x5 standard, (b) 5x5 cruciform, (c) 5x5 with big rod

- WRB-1 and W-3 demonstrated high accuracy (mean P/M ≈ 1.0), whereas B&W-2 consistently underpredicted CHF across all geometries.
- The consistent underprediction of B&W-2 is attributed to its inherent conservatism (lack of a turbulence correction factor), rather than pressure data limits [4].
- W-3 slightly outperformed WRB-1 in the cruciform geometry. This is attributed to the fact that W-3 incorporates the cold-wall factor  $F_{CW} \propto \left(1 - \frac{D_{wetterd}}{D_{heated}}\right)$  [5], which accounts for changes in geometric structure.

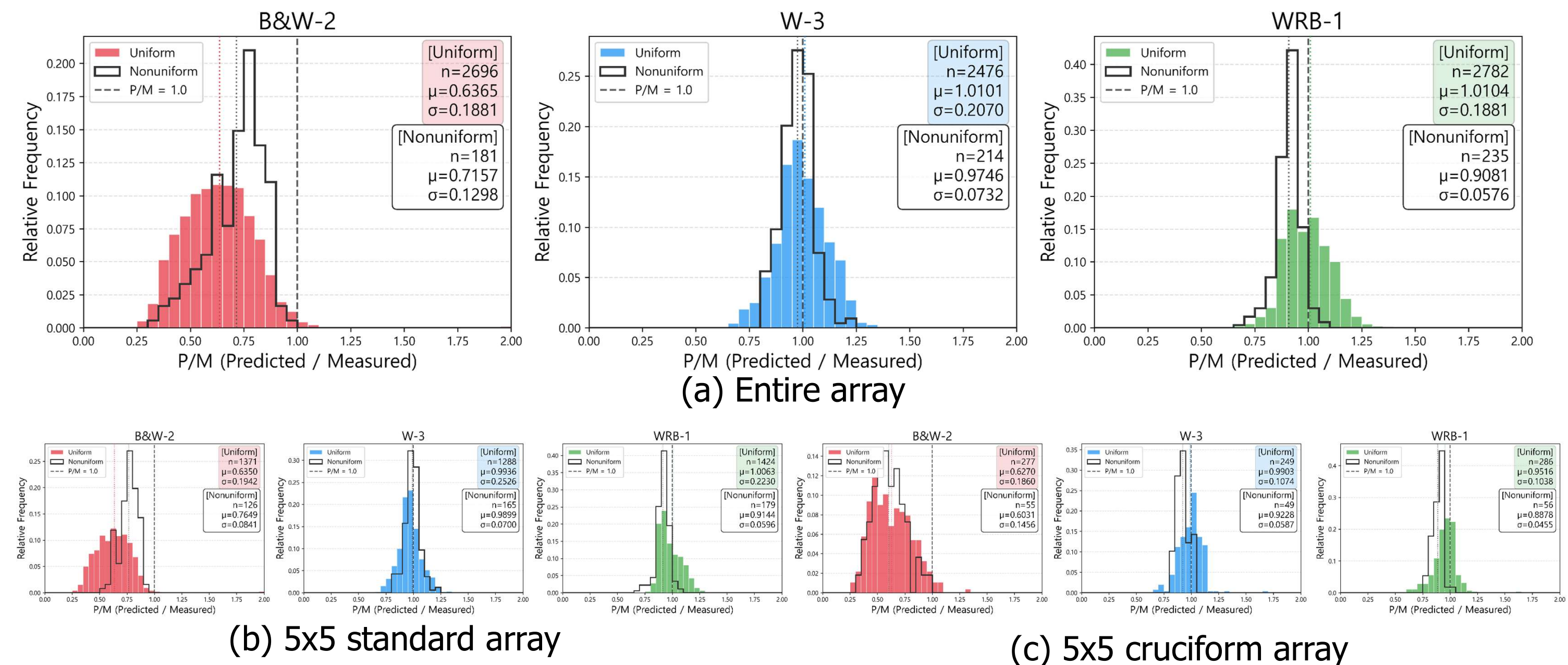


Fig 3. Prediction data histogram according to axial power distribution (vs Nonuniform)

- Nonuniform power distributions significantly improved prediction consistency for all models.
- Notably in B&W-2, geometric differences from atypical structures completely dominated over the effectiveness of axial power correction ( $F_{APk}$ ) [3].

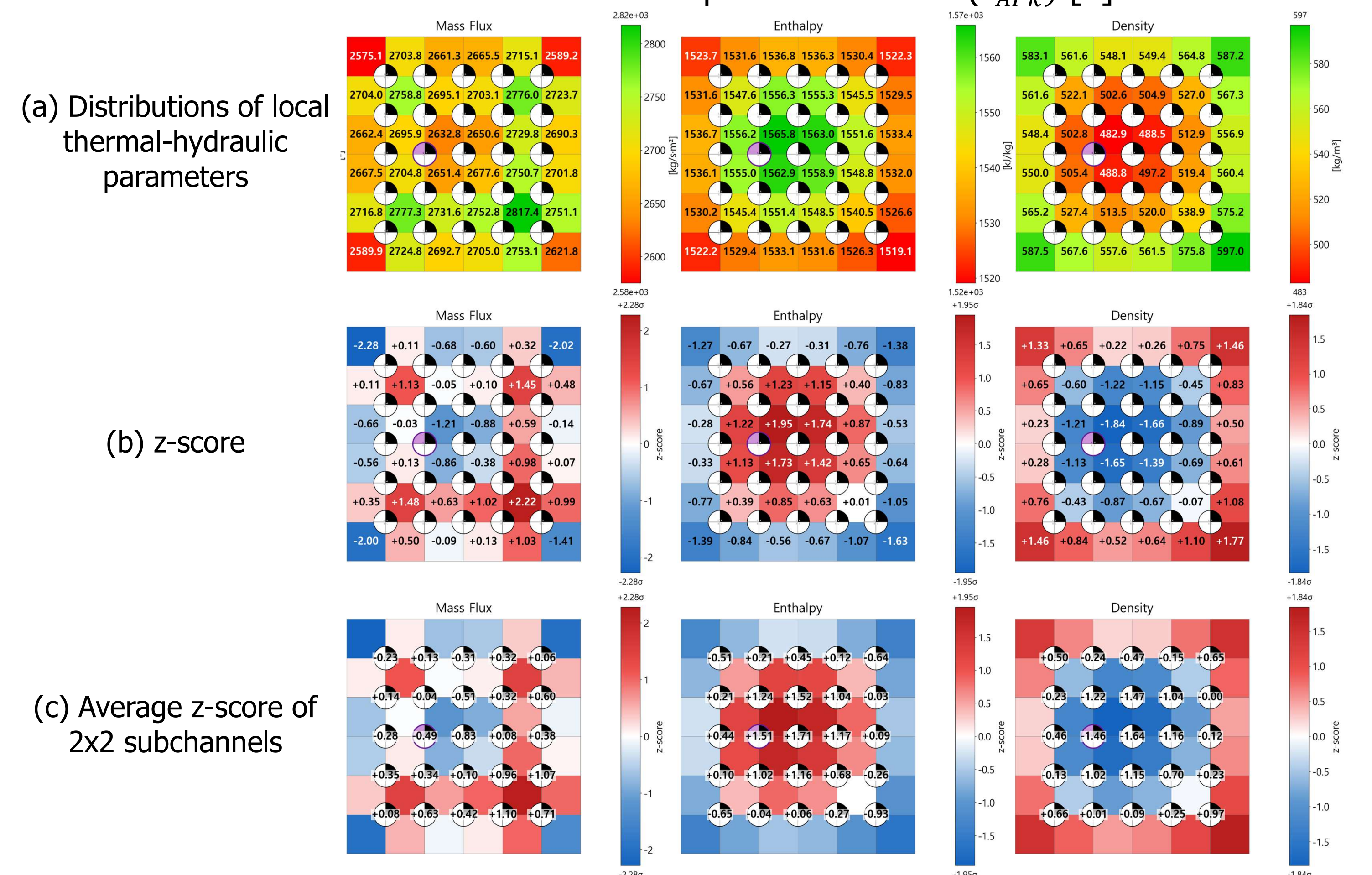


Fig 4. Specific data of thermal-hydraulic parameters prediction results on at 5x5 standard array (5x5 standard, P = 12.41MPa, T = 566.5 K, G = 2703 kg/m<sup>2</sup> · s, Purple: Actual CHF rod)

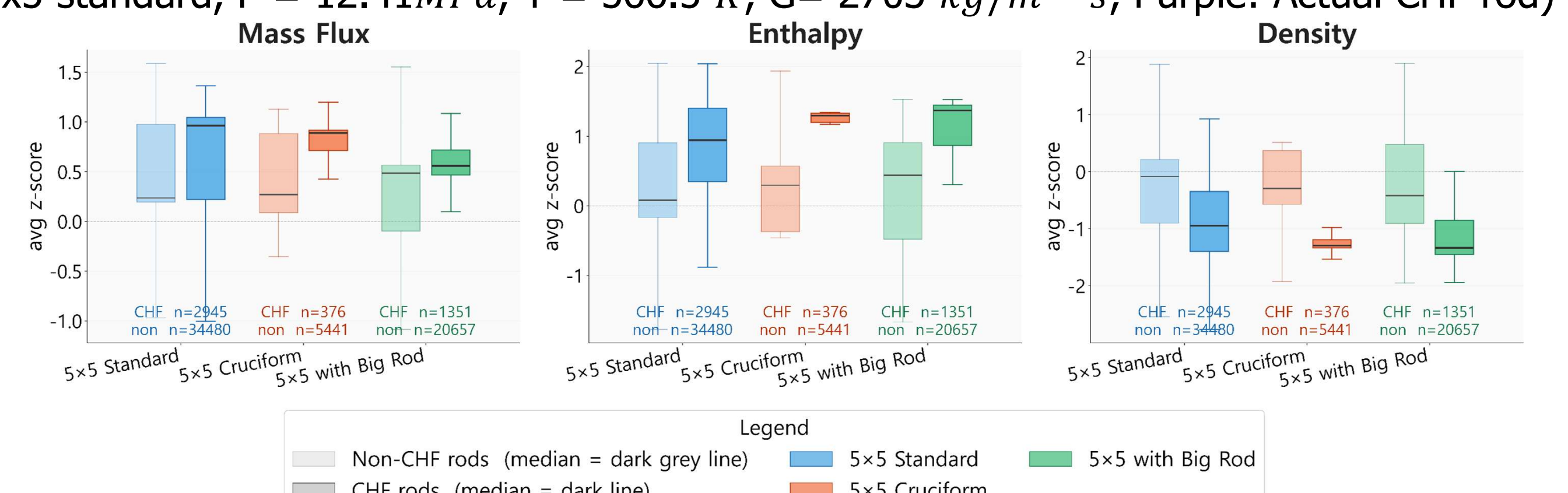


Fig 5. Impact of thermal-hydraulic parameters between CHF rods and non-CHF rods

- Enthalpy and density exhibit significant differences between CHF and non-CHF rods. Notably, in the 5x5 cruciform array, a margin of over 0.5 exists between the two distributions, indicating high sensitivity to these parameters.

## 4. Conclusion

WRB-1 and W-3 demonstrated high accuracy with mean predictions close to 1.0. Notably, in the atypical cruciform geometry, W-3 slightly outperformed WRB-1 by approximately 0.04 (mean prediction of 0.98). This is because the cold-wall factor  $F_{CW}$  is explicitly considered. For B&W-2, the nonuniform power correction led to significant improvements in the standard array but failed to do so in the atypical structure. This suggests that severe geometric distortion powerfully dominates the correlation's predictive performance in atypical geometries. Furthermore, local parameter analysis revealed that enthalpy is a primary discriminator for CHF in distorted geometries (z-score margin > 0.5). Therefore, future enhanced CHF correlations must prioritize explicit geometric corrections and local enthalpy parameters.

## References

- [1] D. H. Hwang, "Development and Application of Subchannel Analysis Code Technology for Advanced Reactor Systems", Korea Atomic Energy Research Institute, Daejeon, South Korea, Rep. KAERI/TR-3129/2006, Jan. 2006.
- [2] C. F. Fighetti, D. G. Reddy, "Parametric Study of CHF Data, Vol. 3: Critical Heat Flux Data", Electric Power Research Institute, Palo Alto, CA, USA, Rep. NP-2609, Sep. 1982.
- [3] J. S. Gellerstedt et al., "Two-phase flow and heat transfer in rod bundles," in Proc. ASME Winter Annual Meeting, Los Angeles, CA, USA, Nov. 1969, pp. 63-71.
- [4] L.S. Tong, "Boiling Crisis and Critical Heat Flux", U.S. Atomic Energy Commission, Oak Ridge, TN, USA, TID-25887. 1972.
- [5] R. C. Anderson, N. P. Wolfhope, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code", Virginia Power, Richmond, VA, USA, Tech. Rep VEP-NE-3, Nov. 1986.