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Assessment of Subchannel Temperature Distributions in the WARD 61-Rod Heat Transfer Experiment Using the SLTHEN Code

Sun Rock Choi, Jonggan Hong, Jaehyuk Eoh

Korea Atomic Energy Research Institute

Introduction

- The core thermal-hydraulic design is used to ensure an appropriate margin for fuel safety limits.
- In a sodium-cooled fast reactor (SFR), DNBR is not a concern because of the high thermal conductivity and high boiling temperature of sodium coolant, and nuclear fuel damage commonly arises from a creep induced failure.
- The creep limit is evaluated based on the maximum cladding temperature considering the uncertainties of the design parameters. An accurate temperature calculation in each subassembly is highly important to assure a safe and reliable operation of reactor systems.
- The core thermal-hydraulic design in the KAERI is performed using the SLTHEN (Steady-State LMR Thermal-Hydraulic Analysis Code Based on ENERGY Model) code, which calculates the temperature distribution based on the ENERGY model.
- In this work, the SLTHEN code is validated using subchannel temperature distributions in the WARD 61-rods heat transfer experiments.

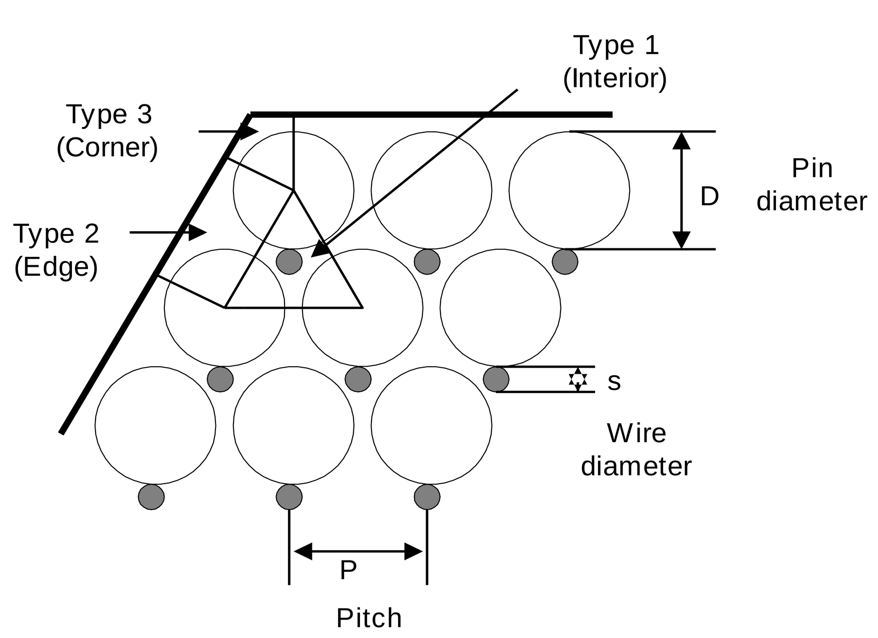
SLTHEN Code

- **S**teady-State **L**MR core **T**hermal-**H**ydraulic Analysis code based on **En**ergy Model
  - ✓ T/H analysis of wire-wrapped assemblies in LMRs
  - ✓ Simplified governing equation called ENERGY model to enhance the computational efficiency
  - ✓ Empirical correlations to describe the subchannel flow distribution and the radial flow mixing characteristics
- Two region model
  - ✓ Central region: Enhanced eddy diffusivity by wire-wraps

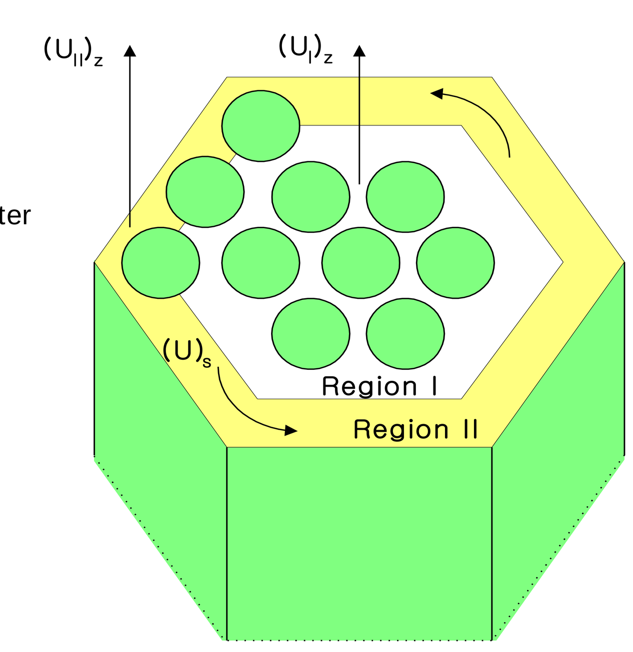
$$\rho C_P U_{zI}\frac{\partial T}{\partial z} = (\rho C_P \varepsilon_I + \zeta k)\left(\frac{\partial^2 T}{\partial x^2} + \frac{\partial^2 T}{\partial y^2}\right) + Q$$

  - ✓ Outer region: Oscillatory lateral flows by wire-wraps

$$\rho C_P U_s \frac{\partial T}{\partial s} + \rho C_P U_{zII}\frac{\partial T}{\partial z} = (\rho C_P \varepsilon_{II} + \zeta k)\left(\frac{\partial^2 T}{\partial n^2} + \frac{\partial^2 T}{\partial s^2}\right) + Q$$



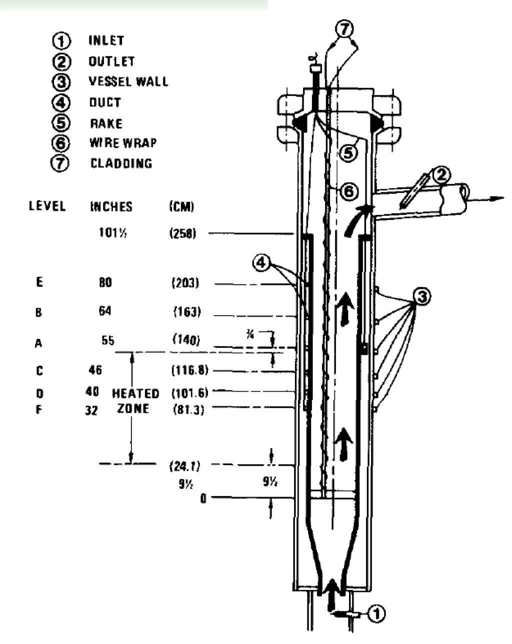
<Subchannels in a subassembly>



<Two region model>

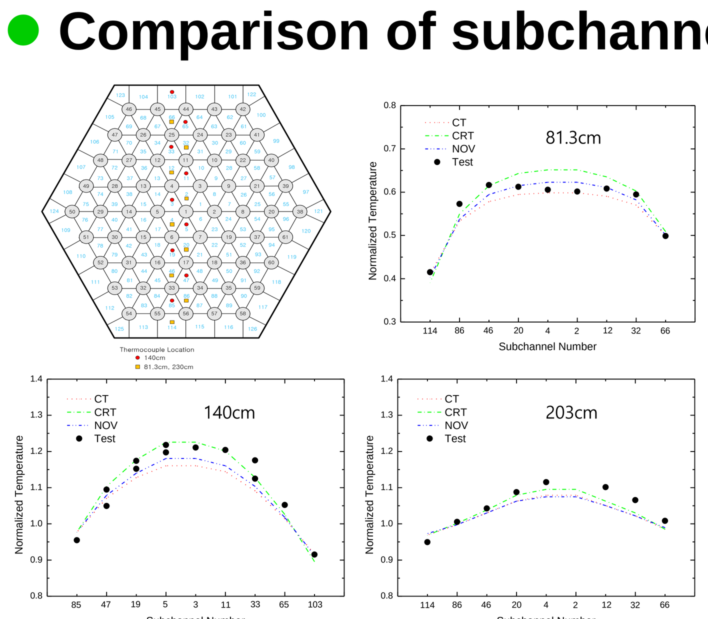
Validation

- WARD 61-rods experiment
  - ✓ Electrically heated fuel rod in flowing sodium
  - ✓ 61-rod bundle of 1.318 cm diameter
  - ✓ Pitch to diameter ratio of 1.082
  - ✓ Wrapped with a spacer wire of 0.094 cm diameter
  - ✓ Heat supply from 24.1cm to 140cm in the axial direction.



<WARD 61-rod assembly>

- Comparison of subchannel temperature distribution



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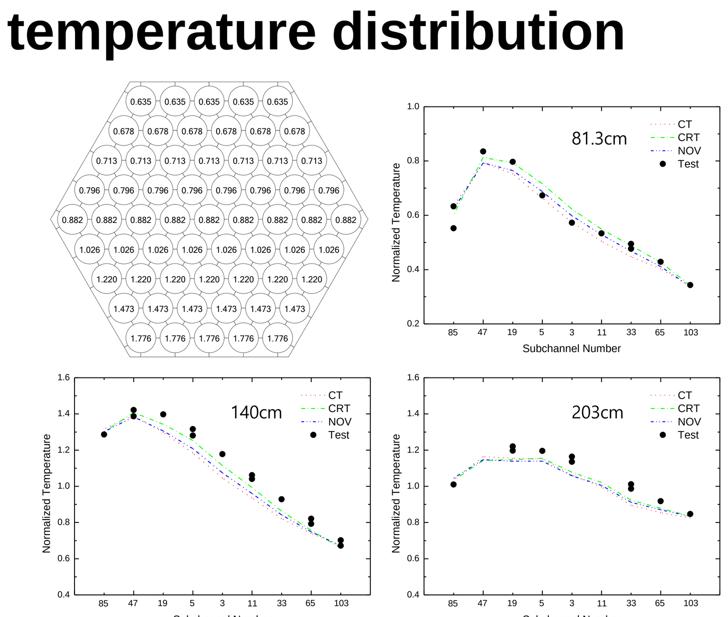

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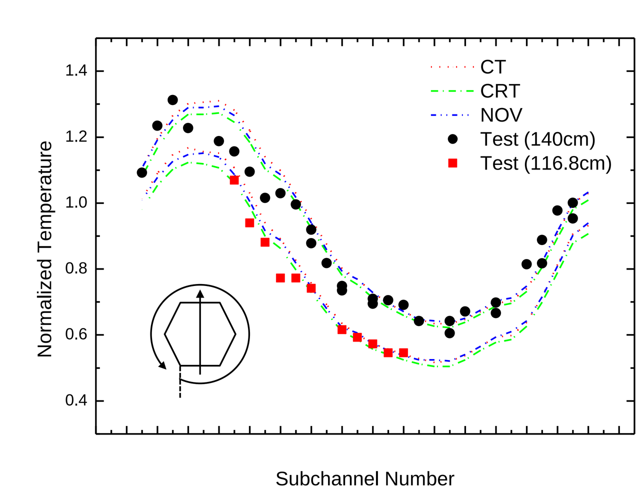
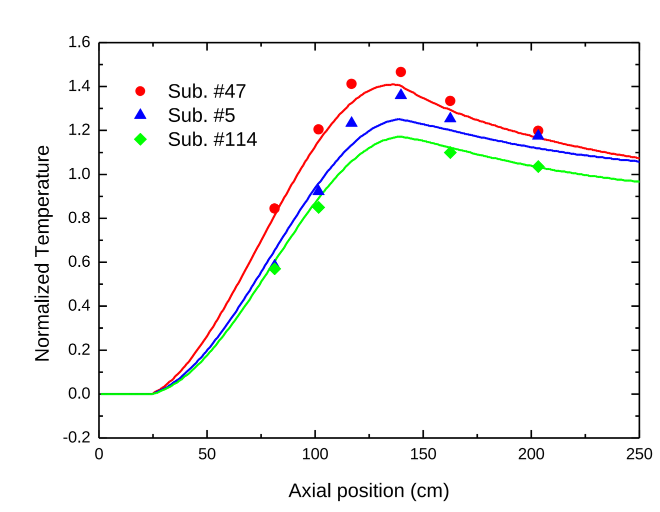


<Edge temperature>



<Axial temperature>

Conclusion

- The SLTHEN code validation for the core thermal-hydraulic design has been performed using subchannel temperature distributions in the WARD 61-rods heat transfer experiments.
- The results indicate that the SLTHEN code appropriately predicts the temperature distributions of the WARD 61-rod experimental values.
- Major discrepancy is observed at the maximum temperature in the central region.

ACKNOWLEDGEMENT

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