Preliminary Assessment of Existing Sodium-Cooled Experimental Data for Validation of the SLTHEN Code

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1. Introduction

The Korea Atomic energy Research Institute (KAERI) has performed a conceptual SFR design with the final goal of constructing a prototype plant by 2028. The main objective of the SFR prototype plant is to verify TRU metal fuel performance, reactor operation, and transmutation ability of high-level wastes.

The core thermal-hydraulic design is used to ensure the safe fuel performance during the whole plant operation. Compared to the critical heat flux in typical light water reactors, nuclear fuel damages in SFR subassemblies are arisen from a creep induced failure. The creep limit is evaluated based on both the maximum cladding temperature and the uncertainties of the design parameters. Therefore, an accurate temperature calculation in each subassembly is highly important to assure a safe and reliable operation of reactor systems. The current core thermal-hydraulic design 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 [1]. In this work, the SLTHEN code is validated for a single subassembly evaluation using existing sodium-cooled experimental data.

2. SLTHEN Code

To enhance the computational efficiency, a simplified energy equation called the ENERGY model was developed in the middle of the 1970s specifically for liquid metal-cooled reactors. To describe the cross-flow by the wire wrap of the fuel pin, a two-region model is employed. The axial velocities in the internal and wall regions of a subassembly can be obtained from the flow split method. This two region approximation enables the momentum equations to be decoupled from the energy equations. Once the flow is split, the temperature and pressure drops are calculated along the axial node with the finite difference equations using a one-pass procedure instead of an iterative one. This simplification significantly reduces the computer storage and computing time.

The SLTHEN code employs two region approximations, which enable the momentum equations to be decoupled from the energy equations. In the central region, the mean flow oscillates around each rod as it progresses along the axial direction. In the outer region near the wall, the flow pattern is quite different. This difference in the outer and inner regions of the assembly suggests that the subassembly flow can be divided into two regions.

The resulting energy transport equations for the two regions are then calculated by

$$\rho C_p U_{zI} \frac{\partial T}{\partial z} = (\rho C_p \varepsilon_I + \zeta k)\left(\frac{\partial^2 T}{\partial x^2} + \frac{\partial^2 T}{\partial y^2}\right) + Q \qquad (1)$$

$$\rho C_p U_s \frac{\partial T}{\partial s} + \rho C_p U_{zII} \frac{\partial T}{\partial z} = (\rho C_p \varepsilon_n + \zeta k)\frac{\partial^2 T}{\partial n^2} + (\rho C_p \varepsilon_s + \zeta k)\frac{\partial^2 T}{\partial s^2} + Q \qquad (2)$$

where the left and right terms represent convective heat transfer and conduction by the enhanced eddy diffusivity, respectively. Q, k and ζ are the volumetric heat source, coolant thermal conductivity coolant and conductivity enhancement ratio from the geometrical factor.

3. Code Validation

A code validation was conducted based on four types of experimental data [2]. The detailed characteristics of the test subassemblies are displayed in Table I where P/D and H/D are the pitch-to-diameter ratio and height-to-diameter ratio, respectively. The cosine distribution in the axial power shape is calculated based on the maximum to average value.

These validation experiments were tested in liquid-sodium environments with electrically heated fuel pins. Steady-state temperature distributions were measured using the thermocouples located around the subchannels, cladding outer walls, and wire wraps. The heated fuel pins revealed cosine power shapes to resemble the actual profile in nuclear reactors. The ORNL 19 pin test only utilized a uniform axial heating. For all tests, temperatures at the end of the heated zone are measured. The ORNL 61 pin and WARD 61 pin tests located thermocouples in three different axial elevations. The radial peak is assumed to be uniform.

The heat transfer with a SFR subassembly reveals the single phase characteristic and thermo-physical property variation is generally very small. Therefore, the validation tests used a smaller heating power than that of actual reactors, and simply accessed a relative temperature distribution to the inlet/outlet temperature difference.

Table I: Subassembly Specification for Sodium-Cooled Experiments

	Pin number	Diameter (inch)	Wire dia. (inch)	P/D	H/D	Heated Length	Axial power shape*
ORNL 19 Pin	19	0.23	0.056	1.244	52.2	21	uniform
ORNL 61 Pin	61	0.23	0.056	1.244	52.2	36	cosine 1.38/1
Toshiba 37 Pin	37	0.256	0.052	1.21	47.2	36.6	cosine 1.21/1
WARD 61-Pin	61	0.519	0.037	1.082	7.7	45	cosine 1.4/1

*Maximum to average

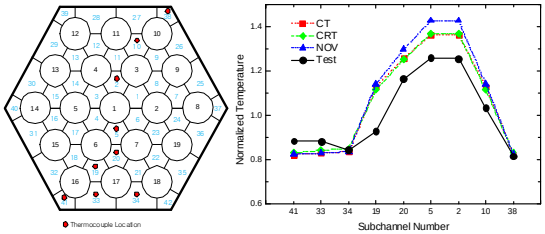

Fig. 1. ORNL 19 pin results.

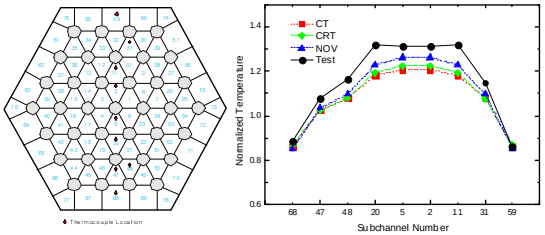

Fig. 3. Toshiba 37 pin results.

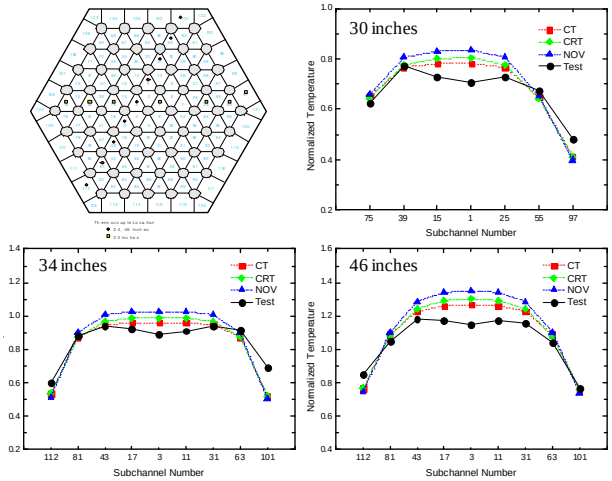

Fig. 2. ORNL 61 pin results.

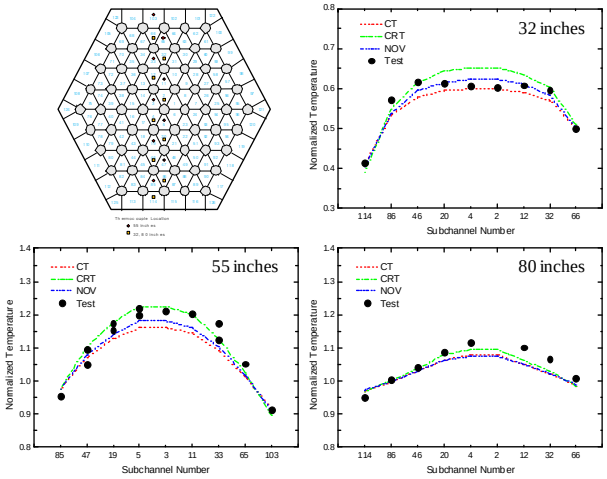

Fig. 4. WARD 61 pin results.

$$\Delta\overline{T} = \frac{T_{sub} - T_{in}}{T_{out} - T_{in}} \qquad (3)$$

Figures 1-4 compare the SLTHEN code evaluation with the experimental data. The calculation utilizes three pressure drop models such as Novendstern, Chiu-Rohsenow-Todreas, and Cheng-Todreas correlations, which were developed for the flow field induced by wire wraps. The code calculations show similar behaviors with the experimental data. However, the three friction correlation effects confirm that slight flow-split differences between the inner and outer subchannels overestimate the maximum temperature in the subassembly central regions. To increase the accuracy of temperature prediction, self-developed correlations should be developed. Figure 2 shows that temperatures increase as the thermal energy accumulates along the heated fuel pins. Figure 4 indicates a radial mixing effect through which the temperature distribution becomes flat from the end of the heated zone.

4. Conclusions

The SLTHEN code validation for the core thermal-hydraulic design has been performed based on existing sodium-cooled experimental data. The results demonstrate that the present design code appropriately predicts the temperature distributions compared with the experimental values. Major differences are observed in the central region within a subassembly owing to the uncertainty of the flow-split calculations. They indicate that self-developed correlations for friction coefficients are required for the prototype SFR in development.

REFERENCES

[1] W. S. Yang, An LMR Core Thermal-Hydraulics Code Based on the ENERGY Model, Journal of the Korean Nuclear Society, Vol. 29, pp. 406-416, 1997.
[2] S. R. Choi, SFR-DR1400000-TE-01-2012Rev.00, KAERI, 2012.