Preliminary Validation of the MATRA-LMR Code Using Existing Sodium-Cooled Experimental Data

Sun Rock Choi *, Sang-Ji Kim
Korea Atomic Energy Research Institute, 1045 Daedeok-daero, Yuseong-gu, Daejeon 305-503, Republic of Korea
**Corresponding author: choisr@kaeri.re.kr*

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. The fuel design limit is highly dependent 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 the reactor systems. The current core thermal-hydraulic design is mainly performed using the SLTHEN (Steady-State LMR Thermal-Hydraulic Analysis Code Based on ENERGY Model) code, which has been already validated using the existing sodium-cooled experimental data [1,2]. In addition to the SLTHEN code, a detailed analysis is performed using the MATRA-LMR (Multichannel Analyzer for Transient and steady-state in Rod Array-Liquid Metal Reactor) code [3]. In this work, the MATRA-LMR code is validated for a single subassembly evaluation using the previous experimental data.

2. MATRA-LMR Code

MATRA-LMR was developed specifically for an LMR analysis based on the MATRA code, which was originally developed for a water-cooled reactor based on a subchannel analysis method. For a thermal-hydraulic analysis of a core consisting of subassemblies with a subchannel of a wire-wrapped rod bundle, a subchannel analysis is widely used. It characterizes the average mass, momentum, and energy balance in every subchannel. It assumes that the axial velocity component is dominant, compared to the components in the transverse direction. Thus, a simplified model can be applied to the transverse momentum equations. A typical triangular subchannel arrangement, a control volume for an axial momentum equation, and control volumes for axial and transverse momentum equations are depicted in Fig. 1. A subchannel is a flow path designated by wire-wraps between fuel rods. There are three types of subchannels: interior, edge and corner. The flow distribution within the subchannels is calculated from the implemented flow split correlations. Since the pressure drop between the inlet and outlet ports along a subassembly must be constant, and subchannels with a small area represent a high flow velocity. Each flow area across a subchannel is also dependent on the axial position of the wire-wraps.

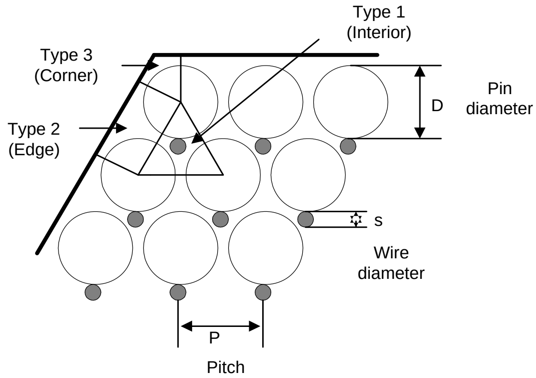

Fig. 1. Thermal-hydraulic subchannel model.

3. Code Validation

A code validation was conducted based on four types of experimental data [4]. 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

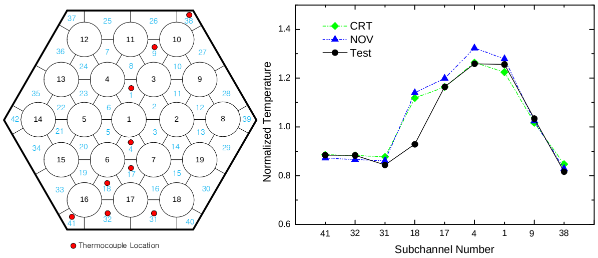

Fig. 1. ORNL 19 pin results.

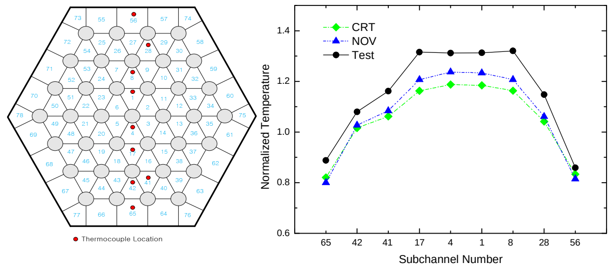

Fig. 3. Toshiba 37 pin results.

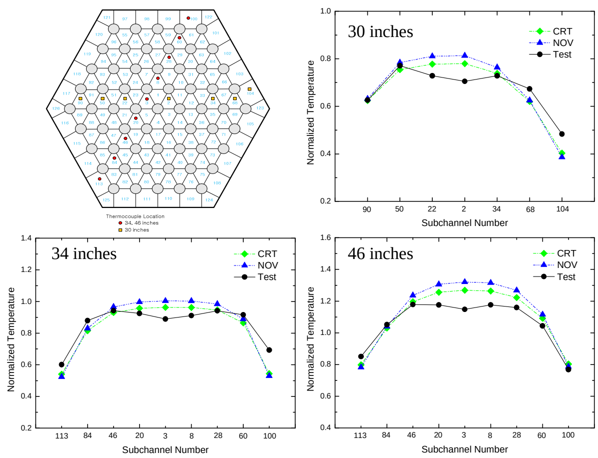

Fig. 2. ORNL 61 pin results.

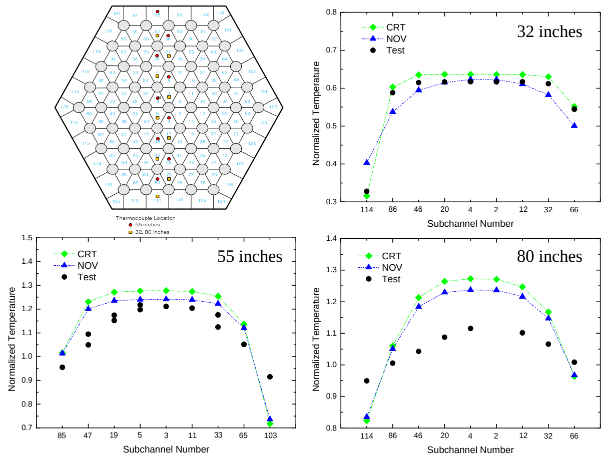

Fig. 4. WARD 61 pin results.

Figures 1-4 compare the MATRA-LMR code evaluation with the experimental data. The calculation utilizes three pressure drop models such as Novendstern and Chiu-Rohsenow-Todreas which were developed for the flow field induced by wire wraps. The code calculation reveals similar behavior with the experimental data, especially in the case of low pin number. In contrast, the SLTHEN code showed a good agreement with the experimental data of high pin number [2]. The largest difference was observed in the WARD 61 pin experiment. The maximum temperature by the MATRA-LMR code was significantly larger than the experimental data due to the radial-wise mixing difference.

4. Conclusions

The MATRA-LMR code has been validated using existing sodium-cooled experimental data. The results demonstrate that the design code appropriately predicts the temperature distributions compared with the experimental values. Major differences are observed in the experiments with the large pin number due to the radial-wise mixing difference.

ACKNOWLEDGEMENT

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