Preliminary analysis of iSMR fuel for regulatory audit preparation

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1. Introduction

Many advanced nations have actively attempted to develop their own SMRs to sustain their nuclear industries, export SMR designs, and supply electricity to remote regions. South Korea plans to submit the Standard Design Approval (SDA) to the regulatory body in 2026 for the iSMR design, an integral system with four modular reactors.

The Korea Institute of Nuclear Safety (KINS) has been preparing regulatory criteria for the safety review. It is essential to first identify safety issues related to the iSMR system and its fuel.

This study also aims to develop a regulatory audit technique for iSMR using the CTF code.

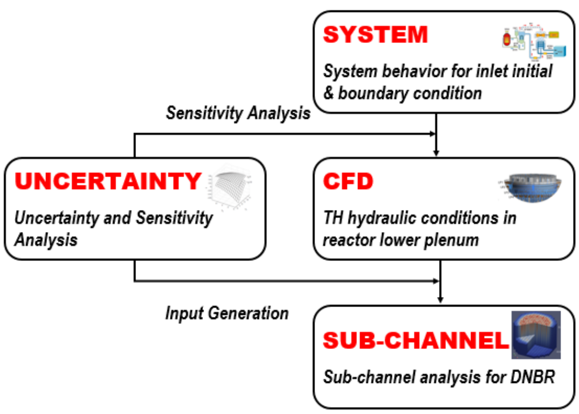

Figure 1. Code system for DNBR [1]

2. Preliminary analysis of iSMR fuel

Although the detailed design of the iSMR fuel has not yet been completed, it is well known that most designs are based on 17ACE7 fuel. The total axial length of the iSMR fuel is 2,400 mm, and an upper cutback is used to mitigate the bottom-skewed power phenomenon due to the absence of a boron system. The location of the water hole for inserting the control rod has been slightly modified compared to the 17ACE7 fuel. The lattice structure remains 17×17, the same as in the 17ACE7 fuel. However, the detailed design and the number of grid spacers to enhance heat transfer capability are not yet known.

The modeling of iSMR fuel in this study has been performed within the limits of the available information. However, the primary objective of this study is to develop a subchannel analysis system for iSMR as part of the regulatory audit assessment.

Figure 2 illustrates the schematic diagram of iSMR fuel using CTF.

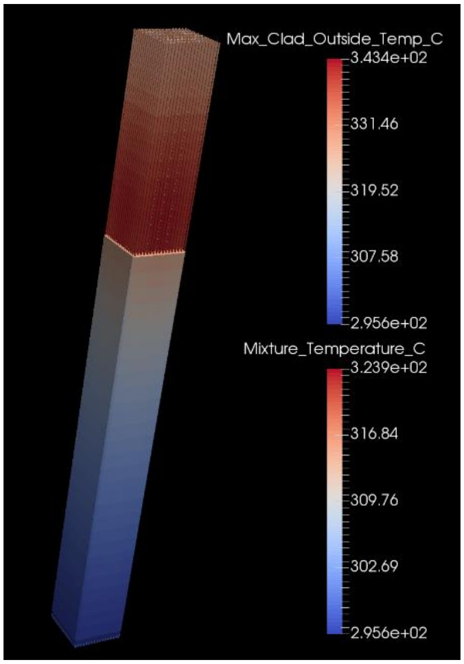

Figure 2. The schematic diagram of iSMR fuel assembly

Table I shows the operating parameters under normal conditions to provide CTF with boundary conditions, considering the power shape at BOC, MOC, and EOC, respectively.

Table I. Operating parameters at normal condition [2]

Power [MWt]	520
No. of Assembly	69
RCS Pr. [kg/cm^2]	158.19
Linear Power [kW/m]	12.077
Lattice Type	17 X 17
No. of Fuel Rod	260
Water Hole per Assembly	29
Height [mm]	2400

The CTF code adopts a two-phase, three-flow-field mathematical and physical model to analyze the thermal-hydraulic characteristics of coolant in each component under transient and accident conditions in nuclear power plants. The three flow fields are vapor, liquid, and entrained droplets. CTF solves two energy conservation equations, three momentum conservation equations, and four mass conservation equations for each spatial dimension.[3] Figure 3 presents the results of the subchannel analysis from BOC to EOC. The cladding and coolant temperatures increase along the axial length, while the mass flux exhibits fluctuations, particularly in regions with grid spacers.

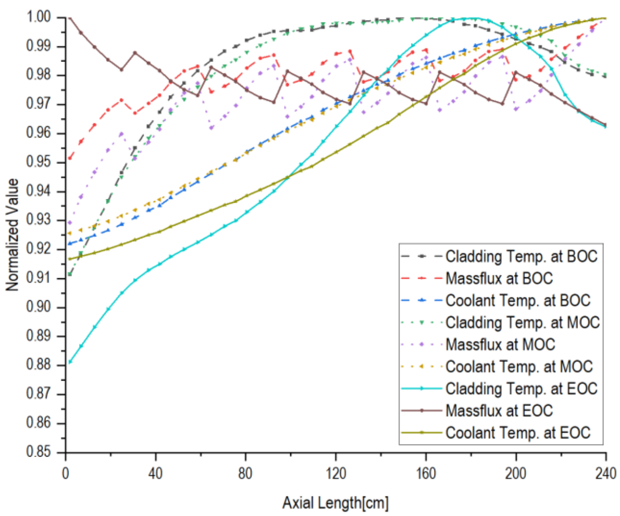

Figure 3. CTF results on the subchannel properties

Figure 4 shows the DNBR and its power distribution along the axial length. The minimum DNBR for each power shape occurs at the point of maximum power peaking. The overall DNBR trend aligns well with the power distribution.

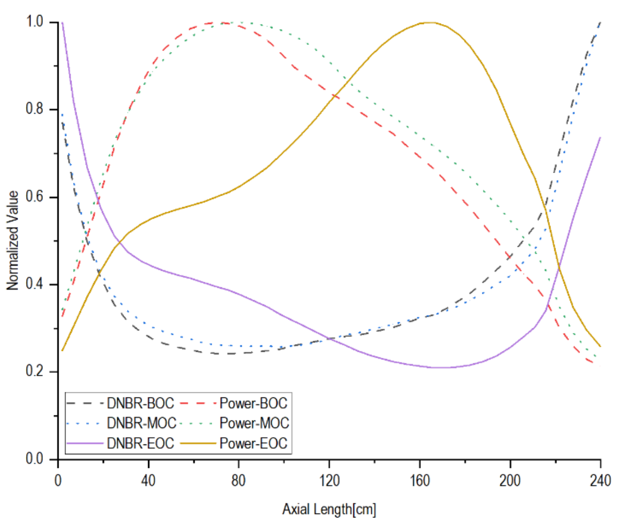

Figure 4. CTF results on the DNBR according to power

3. Scaling analysis for CHF experiment

Developing a critical heat flux (CHF) correlation requires a large dataset obtained through experiments. To reduce costs and time, nuclear designers typically perform scaling analyses to derive CHF correlations efficiently.

For iSMR operation under normal, transient, and accident conditions, the driving force is expected to be natural circulation without any electrical power. Therefore, the CHF correlation should be developed specifically under natural circulation conditions.

It is also essential to understand the buoyancy effects caused by density changes due to temperature differences between the inlet and outlet when simulating natural circulation using experimental facilities.

General scaling laws for modeling nuclear reactor systems have been proposed by Ishii and Kataoka [4], who attempted to develop scaling laws for natural circulation loops. Their method has been adopted in this study to design various modeling approaches.

Similarity groups can be derived from the mass, momentum, and energy conservation equations based on the Ishii and Kataoka method. The Richardson number and friction number are defined below.[4][5]

$$\mathrm{Ri} = \frac{\mathrm{g}\beta\Delta\mathrm{T}\Delta\mathrm{L}}{U^2} = \frac{Buoyancy}{Inertia}$$

And

$$\mathrm{F} = f\frac{l}{d} + k = \frac{Friction}{Inertia}$$

In additional to the above-defined physical similarity groups, several geometrical similarity groups are obtained. They are

$$l_R = \frac{l_{model}}{l_{prototype}}$$

And

$$a_R = \frac{ql^{-1/2}{}_{model}}{ql^{-1/2}{}_{prototype}}$$

Finally, we can get CHF ratio below

$$CHF_R = \left(\frac{\varphi \cdot q \cdot l}{d \cdot u}\right)_R$$

Table II lists the results of the natural circulation capability for iSMR and the experimental facility. The total pressure drop needs to be calculated along the flow path to determine the total loss coefficient. However, this study assumes consistency in the friction number between iSMR and the experimental facility, as the focus is on the fuel assembly rather than the system loop.

Table II. Results of scaling analysis

Properties	17 X 17	2 X 2	3 X 3	5 X 5
Power[kWth]	7,536.23	119.94	231.88	666.67
Pressure[kg/cm^2]	158.19			
Height[mm]	2,400			
Mass Flow[kg/s]	54.18	0.83	1.67	4.79
Core temp. [C]	295.5 / 320			
No. of Rod	260	4	8	23
No. of water hole	29	0	1	2
Distance Between source and sink	Same distance			
CHF Correlation	AECL Lookup Table			

Figures 5 to 8 present the calculation results for each model. The 3×3 and 5×5 models closely match the original assembly in terms of cladding and coolant temperature, whereas the 2×2 model does not. The mass flux trend of the 5×5 model aligns most accurately with the original assembly. Among all models, the 5×5 model also shows the closest DNBR trend to the original assembly. In contrast, the 2×2 model fails to accurately simulate the 17×17 assembly in all aspects based on the results of this calculation.

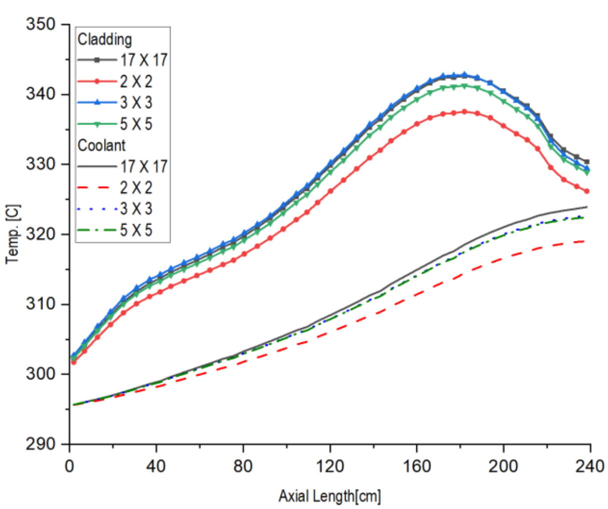

Figure 5. Temperature on each modeling

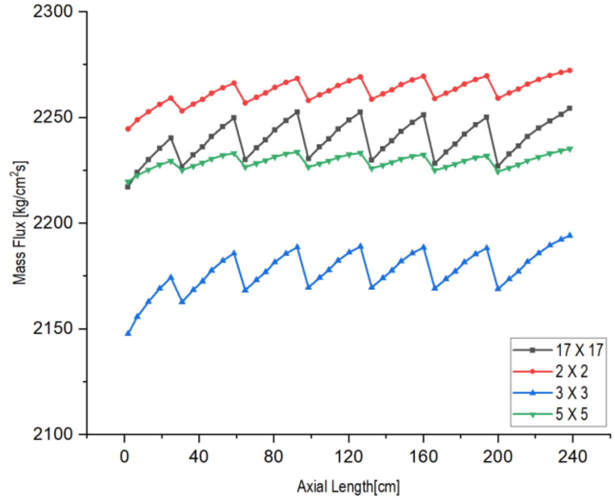

Figure 6. Mass flux on each modeling

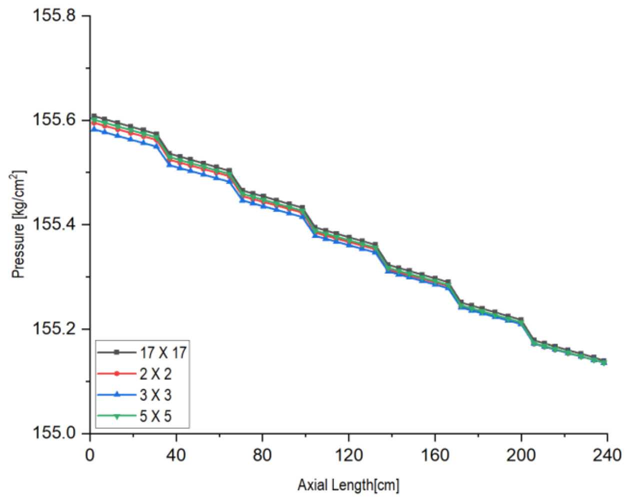

Figure 7. Pressure on each modeling

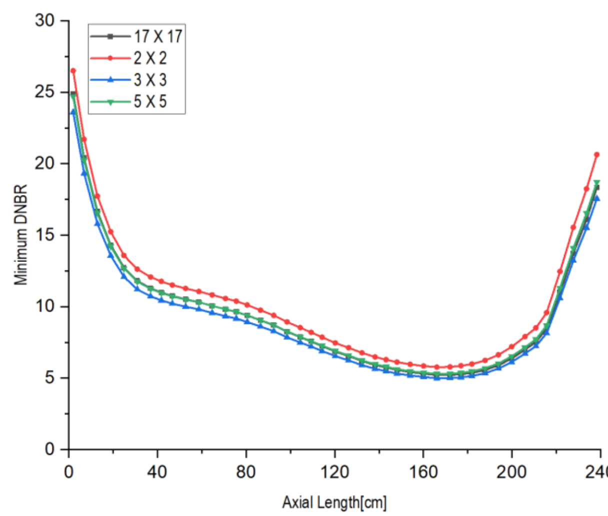

Figure 8. DNBR results on each modeling

4. Conclusions

This study presented the results of a preliminary analysis of the heat capacity of iSMR fuel under normal operating conditions. Additionally, we analyzed the nuclear fuel modeling intended for experiments to develop the CHF correlation through scale analysis. The results indicate that detailed consideration is required not only for buoyancy and resistance in natural circulation but also for the water hole arrangement.

KINS plans to continue developing regulatory audit technique necessary for iSMR licensing in the future.

Acknowledgements

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NOMENCLATURE

a	Flow area
β	Thermal expansion coefficient
d	Hydraulic diameter
f	Friction factor
g	Gravity
k	Conductivity
l	Axial length
ΔL	Length difference
q	Heat
ΔT	Temperature difference
φ	Thickness
u	Velocity

SUBSCRIPT

R	Model to prototype ratio

REFERENCES

1. Ilsuk Lee, "Assessment of Realistic Departure from Nucleate Boiling Ratio (DNBR) Considering Uncertainty Quantification of Core Flow Asymmetry", Vol. 14, 1504, Energies (2021)
2. KEPCO NF, iSMR presentation material (2025)
3. Xiaoxi Zhang, "Thermal Hydraulic Review of Light Water Reactor based on Subchannel Code CTF", Vol 413, 112482, Nuclear Engineering and Design (2023)
4. M. Ishii and I. Kataoka, "Scaling Law for Thermal-hydraulic System under Single Phase and Two-phase Natural Circulation", Vol. 81, Nuclear Engineering and Design (1984)
5. Ilsuk Lee, "Full-height Thermal-hydraulic Scaling Analysis of Pb-Bi-colled Fast Reactor PEACER", Vol. 155, Nuclear Technology (2005)