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Study on Models of Thermal-Hydraulic Design Parameters for Research Reactors using Plate-type fuels

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1. Introduction

There are several ways to show quantitatively how safe research reactors(RRs) are and what characteristics research reactors are expected to have in terms of thermal-hydraulics [1,2]. Those parameters of great importance to heat transfer in the core of RRs are onset of nucleate boiling (ONB), onset of significant voids (OSV), onset of flow instability (OFI), and critical heat flux (CHF).

It is not ready to use those design parameters for RRs in the Safety and Performance Analysis Code (SPACE) [3], which is a state-of-the-art system code recently licensed for the safety analysis of nuclear power plants in Korea.

Since research reactors are so different from power reactors as shown in table I, models of design parameters applicable to RRs should be carefully investigated considering the differences in operating conditions and geometries [2,4].

In this paper, the models of ONB, OSV, and OFI are studied and presented for application of the SPACE code to RRs with plate-type fuels.

Table I. Comparison of Characteristics between Research Reactors and Power Reactors

	Research Reactors	Power Reactors
Purpose	Production and R&D using neutron - High neutron flux	Electricity generation - High power production
Reactor type	Mostly Tank in pool type - Large coolant inventory for ultimate heat sink	Loop type - Small coolant inventory for heat sink
Power	0~30 MW_{th} for general RRs 40~250 MW_{th} for MTR	~ 3000 MW_{th}
Core size	Small (dia: ~ 50 cm, leng: 50~70cm) - Small source term	Large (dia: ~ 300 cm, leng.: 450 cm) - Large source term
Fuels	Metallic with high conductivity - Fuel alloy particles dispersed in Al Plate, tubular, finned rod type	Ceramic fuel - High heat capacity Rod type
Operating condition	Low pressure and low temperature - 1~10 bar, 20~50 degree in Celsius	High pressure and high temperature - 150 bar, ~ 300 degree in Celsius

2. Review of thermal-hydraulics design parameters of research reactors

A set of representative parameters of RRs considered in this study are as shown in table II: flow directions of upward and downward, power up to several tens of MW, maximum heat flux up to 1.3 MW/m^2, mass flux up to 6,000 kg/m^2-s under low pressure and temperature conditions.

Table II. Representative parameters of the KJRR and other typical research reactors

	KJRR	**Other RRs**
Reactor type	Open-tank-in-pool Downward flow	Open/Closed-tank-in-pool Downward/Upward
Power	15 MW_{th}	Up to hundreds MW_{th}
Max. Heat flux	~ 1300 kW/m^2	Depends on designs ~ 3000 kW/m^2
Mass flux	~ 6000 kg/m^2-s	Depends on designs
Fuels	Metallic U-7Mo/Al-5Si 19.75 wt% LEU Plate type	Depends on designs Metallic U ~ 19.75 wt% LEU Plate or curved type
Operating condition	Low pressure and low temperature - ~4 bar, 5~35℃	Depends on design Low/Medium pressure and low temperature - ~ tens bar, ~50℃

Research reactors are designed such that 1) Onset of nucleate boiling(ONB), at which nucleate boiling starts locally, does not occur during normal operation because of operational stability and 2) critical heat flux (CHF) resulted from departure from nucleate boiling or other mechanisms never happen in fuels since the fuel integrity is the first essential barrier for nuclear safety.

Onset of flow instability (OFI) had been used as a conservative criterion instead of CHF because OFI may act as a triggering mechanism to accelerate the CHF under a certain condition depending on the configuration of the core cooling channels of RRs [5]. Onset of Significant Void (OSV) was often selected as OFI since both are very close [6].

The relationship between limits and settings in RR design can be explained in Fig. 1 [7]: 1) normal operating condition is the lowest level of thermal-hydraulics demands; 2) operating limit and trip settings are followed by ONB considering instrumentation delay and uncertainties; and 3) safety limit to avoid CHF or excessive cladding temperature is so far from the limiting safety system setting.

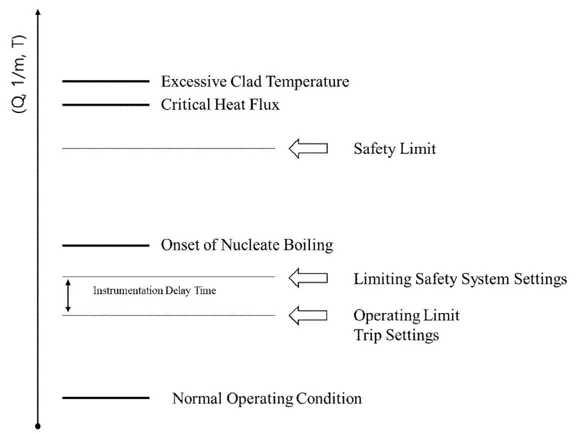

Fig. 1. Conceptual relationship between limits and settings in research reactors.

The actual safety margin, a quantitative way to show how safe research reactors is in terms of thermal-hydraulics design parameters, such as ratios of ONB, OFI and CHF of at normal operating condition of 10 MW RR [8], are shown in Fig. 2.

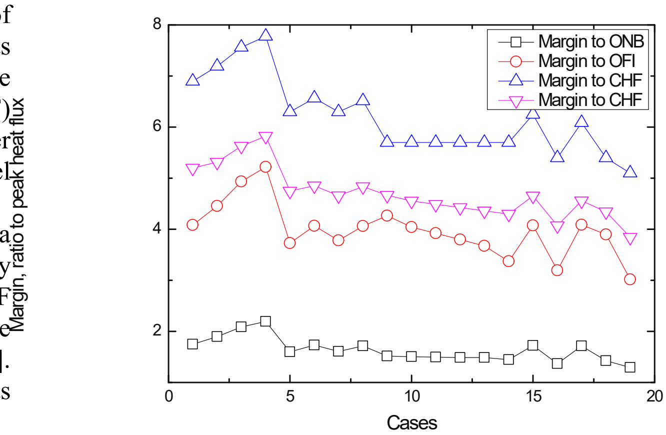

Fig. 2. Margin to actual peak heat flux [5].

3. Models

The methodology applied to select the models of ONB and OFI including OSV for RRs is as follows: 1) firstly, review of the conventional system codes such as RELAP5/MOD3.3 [9], SPACE [10], TRACE [11]; 2) search of references, publicly open, for phenomenological understanding, phenomena prediction models, applicable ranges of every model with comparison to operational conditions of RRs in interest and 3) selection of a model or a set of models expected to be appropriate for the RRs with an acceptance criteria (the model should be widely used and sufficiently validated and the model does not have any proprietary issue.)

There was not used any correlation for predicting ONB and OFI in the system codes such as RELAP5, SPACE, TRANCE [9,10,11].

Bergles and Rohsenow correlation [12] was selected in this study among 6 ONB correlations, Bergles and Rohsenow, Sato-Matsumura [13], Marsh and Mudawar [13], Jens-Lottes [14], CNNC [15], Basu [16] since it has been widely accepted over the world [8] and has the simplest relation between local parameters such as temperature and pressure only.

OFI in RRs (so-called flow excursion) is categorized as static instability such as Ledinegg instability [13,17]. The conventional S-shaped curve shown in Fig. 2 is used to explain what OFI is: as mass flow through cooling channel reduces, the flow pattern changes from single-phase liquid to two-phase liquid and flow and pressure does not show proportional relation any more due to change in flow structure.

In practice, 4 different methods can be applied to determine whether OFI happens or not: 1) ONB correlations; 2) Net vapor generation (or onset of significant void); 3) onset of fully developed boiling (FDB); 4) global variables. Since ONB is too conservative as shown in Fig. 1 and Fig.2, and FDB is not well established, OSV correlation [19] and global variable approaches [20, 21, 22, 6] were studied and only 3 models are presented in table III.

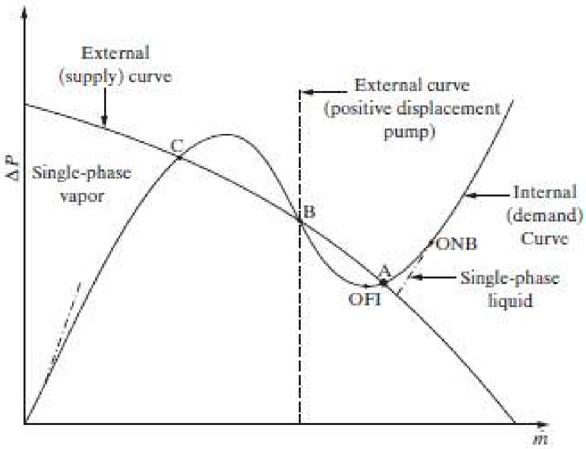

Fig. 2. Internal and external pressure drop and flow characteristics during OFI.

Table III: Selected models for OFI prediction in RRs

					RRs			W-F [20]	A-J [6]	S-Z [19]
					JRTR	KJRR	Others			
channel	geometry				plate	plate	plate	plate, tube	plate	plate, tube
channel	plate	L	wet	mm			600	400 ~ 609.6	566	
channel	plate	L	heat	mm	640	640			300	
channel	plate	B	wet	mm				25.4	54	
channel	plate	B	heat	mm	60	60	63		50	
channel	plate	Gap	mm		2.35	2.35	2.1~2.9	1.4 ~ 3.225	2.35	2.2 ~ 6.3
channel	tube	D	mm		-	-	-	6.452		7 ~ 24
Flow direction (down:d, up:u)					d	d	-	d	u	
p	bar				~ 1	~ 2	~ 32	1.17~ 1.86	1 ~ 1.4	1.01 ~ 138
Subcooling, Inlet	°C						-	25.2 ~ 65		
Temperature (in/out)	°C				5~35 /	5~35 /	~ 38 /	/	35~65 /	/
Heat flux	kW/m^2				177.5	436.4	~ 3,000	420~ 3,400	50 ~ 650	
Mass flux	kg/m^2s				~ 2,700	~ 6,300	~ 10,000	600 ~ 9,000	118 ~ 1,400	
velocity	m/s				2.7	6.3	~ 10	0.6 ~ 9.14		

4. Conclusions

Onset of nucleate boiling (ONB) and onset of flow instability (OFI) correlations were reviewed and selected to implement them into the SPACE code in order to show how safe research reactors are during the design and to support the licensing of research reactors.

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REFERENCES

[1] Cheol Park, Research Reactor Design, Management and Utilization (Chapter 2, Design of a Research Reactor), Korea Atomic Energy Research Institute, 2009.

[2] H. Kim, S.K. Park, C. Park, S.W. Lee, Study on Models of ONB, OSV, OFI for implementing to SPACE code applying to Research Reactors with plate-typed fuels, KAERI /TR-7681/2019.

[3] S.J. Ha et al., Development of the SPACE code for nuclear power plants, Nuclear Engineering and Technology, 43, 45– 62, 2011.

[4] H. Kim et al., Development of Phenomena Identification and Ranking Tables (PIRTs) to implement Research Reactor-specific capability in SPACE code, submitted in Annals of Nuclear Energy, 2019.

[5] R.B. Rothrock, Thermal-Hydraulic Bases for the Safety Limits and Limiting Safety System Settings for HFIR Operation at 100 MW and 468 psig Primary Pressure, Using Specially Selected Fuel Elements, ORNL/TM-13694, 1998.

[6] O. Al-Yahia, D. Jo, ONB, OSV, and OFI for subcooled flow boiling through a narrow rectangular channel heated on one-side, International Journal of Heat and Mass Transfer, Vol. 116, pp. 136– 151, 2018.

[7] H.T. Chae, JR-TN-CA-002, Rev.0, Design Data and Methodology for Thermal Hydraulic Analysis, 2011.

[8] IAEA-TECDOC-233, Research reactor core conversion from the use of Highly Enriched Uranium to the use of Low Enriched Uranium Fuels Guidebook, IAEA, Vienna, 1980.

[9] RELAP5/MOD3.3 CODE MANUAL VOLUME IV: MODELS AND CORRELATIONS, Division of Systems Research, Office of Nuclear Regulatory Research U. S. Nuclear Regulatory Commission, Information Systems Laboratories, Inc. Rockville, Maryland Idaho Falls, Idaho, October 2001.

[10] SPACE 3.0 manual, Volume 5, auxiliary equations manual, 2017.

[11] TRACE V5.0 Theory Manual: Field Equations, Solution Methods, and Physical Models,

[12] Bergles, A.E., Rohsenow, W.M., 1964. The determination of forced-convection surface-boiling heat transfer. Trans. ASME Journal of Heat Transfer 86, 365-372

[13] S. Mostafa Ghiaasiaan, Two-Phase Flow, Boiling and Condensation in Conventional and Miniature Systems, Cambridge University Press, New York, 2008.

[14] Jens, W.H. and Lottes, P. A., Analysis of Heat Transfer Burnout, Pressure Drop and Density Data for High Pressure Water, NL-4627, 1951.

[15] Ma, J. et al., Experimental studies on single-phase flow and heat transfer in a narrow reactangular channel, Nuclear Engineering and Design, 241, pp.2865-2873, 2011.

[16] Basu, N., Warrier, G., Dhir, V.K., Onset of Nucleate Boiling and Active Nucleation Site Density during Subcooled Flow Boiling, Journal of Heat Transfer, 2002.
[17] G. Yadigaroglu, Two-phase flow instabilities and propagation phenomena, in: M. Delhaye, M. Giot, L.M. Riethmuller (Eds.), Thermohydraulics of Two-Phase Systems for Individual Design and Nuclear Engineering, Hemisphere, Washington, DC, 1981, pp. 353– 403.
[18] Ghione, Alberto, Noel, Brigitte, Vinai, Paolo, Demaziere Christophe, Criteria for Onset of Flow Instability in heated vertical narrow rectangular channels at low pressure: An assessment study, International Journal of Heat and Mass Transfer, 105, 464-478, 2017.
[19] Saha, Pradip & Zuber, Novak, Point of net vapor generation and vapor void fraction in subcooled boiling, International heat transfer conference, Tokyo, Japan, 3 Sep 1974.
[20] R.H. Whittle and R.Forgan, A Correlation for the Minima in the pressure drop versus flow rate curves for sub-cooled water flowing in narrow heated channels, Nuclear Engineering and Design, 1967.
[21] J.E. Kennedy, G.M. Roach, Jr., M.F. Dowling, S.I.Abdel-Khalik, S.M. Ghiaasiaan, S.M. Jeter, Z.H.Quershi, The Onset of Flow Instability in Uniformly Heated Horizontal Microchannels, ASME, Vol. 122, 2000.
[22] Juhyung Lee, Heetaek Chae, Soon Heung Chang, Flow instability during subcooled boiling for a downward flow at low pressure in a vertical narrow rectangular channel, International Journal of Heat and Mass Transfer Vol. 67, pp. 1170– 1180, 2013.