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Comparison of the Thermal-Hydraulic Characteristics of Optimised Fuel Assembly with That of Standard Fuel Assembly

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최적 핵연료집합체와 표준 핵연료집합체의 열수력학적 특성비교

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Abstract

The thermal-hydraulic characteristics of the 17 × 17 OFA (Optimized Fuel Assembly) used in the KNU 7&8 are analyzed and compared with that of the 17 × 17 SFA (Standard Fuel Assembly) loaded in the KNU 5&6. The thermal-hydraulic chanacteristics analyzed are minimum DNBR, fuel centerline temperature and exit void fraction at normal operation and design over power transient. Additionally, local linear rod power, which will cause fuel centerline melting, is calculated. The DNBR sensitivity calculations are performed with respect to the reactor operating parameters. COBRA-IV-I code is used for these calculations. The modified W-3 correltion and the drift-flux model are applied for the critical heat flux calculation and the void fraction calculation, respectively. From the calculated results, it has been found that the possibility of DNB occurrence is higher in the OFA than in the SFA. The other hand, the local linear power resulting in fuel centerline melting of the OFA is nearly equal to that of the SFA.

요 약

원자력 7, 8호기에 장전된 17×17 최적 핵연료집합체의 연수력학적 특성을 원자력 5, 6호기에 장전된 17×17 표준 핵연료집합체와 비교하여 분석하였다. 분석된 열수력학적 특성은 정상상태와 과도출력상태에서의 최소 DNBR, 연료봉 중심온도, 출구 기포율등이며, 아울러 연료봉의 중심이 용융되는 국부선출력과 원자로 운전변수들에 대한 DNBR 민감도 계수도 계산하였다. 사용된 코드는 COBRA-IV-I이며, 임계열속 계산에는 R형 그리드에 대해 수정된 W-3 상관식을, 기포율계산에는 drift-flux model을 이용하였다.

계산결과, DNB가 발생할 확률은 최적 핵연료집합체가 더 높았으나, 연료봉의 중심이 용융되는 국부선출력은 표준 핵연료집합체와 거의 동일한 것으로 나타났다.

Nomenclature

C_o	=	drift-flux parameter
C_{oi}	=	fitting parameters
G	=	mass flux (lb/hr-ft²)
g	=	acceleration of gravity
g_c	=	gravitational conversion factor
K_n	=	axial grid spacing coefficient
L	=	total heated core length (ft)
P	=	system pressure (psia)
TDC	=	thermal diffusion coefficient
t	=	time
V_{gj}	=	drift velocity (ft/sec)
V_{gji}	=	fitting parameters
X	=	fitting parameter
Y	=	fitting parameter

Greek Symbol

α	=	void fraction
σ	=	surface tension
ρ_f	=	liquid density
ρ_g	=	vapor density
χ	=	quality

I. Introduction

The current design on the pressurized water reactor (PWR) fuel assembly is based on a slightly under-moderated lattice of the fuel rods. The optimization of this design was taken during the period of relatively low uranium ore price. Presently, significant efforts are being made to improve the uranium utilization.

In the neutronic aspect of this problem, an increase of the water content in the core was expected to increase the reserved reactivity, and consequently increase the discharge burnup.[1] For this purpose the Westinghouse Optimized Fuel Assembly (OFA) with smaller fuel rod diameter and substitution of Zircaloy for inconel as grid material, has been loaded in the Korean Nuclear Unit(KNU) 7&8.

The nuclear calculation for unit assembly of OFA revealed some attractive aspects such that the neutron flux depression effect in the grid regions and the axial power peaking factor was reduced.[2]

In this study, the thermal-hydraulic analysis of the OFA, loaded in the KNU 7&8, was performed and compared with the thermal-hydraulic characteristics of the Standard Fuel Assembly(SFA), loaded in the KNU 5&6. The criterion of the fuel integrity for the ANS Condition I&II categories was under consideration, and the DNBR sensitivity analysis was also performed for various reactor operating parameters. These calculations were based on the subchannel analysis by the COBRA-IV-I code.

For this study, the W-3 correlation modified for R grid and the drift-flux model are incorporated in the COBRA-IV-I code. The analysis showed that the OFA did not have significant shortcomings compared with the SFA.

II. Methods and Procedures

2.1 Basic Concept of COBRA-IV-I code

COBRA-IV-I code utilizes the basic concepts of subchannel analysis in which the flow area of a nuclear fuel bundle is divided into subchannels whose boundaries are defined by the adjacent fuel rod surfaces. The subchannels are divided axially into discrete control volume for which the equations of continuity, energy and momentum are written. The integral balance for the mass, energy and momentum of the mixture can be expressed as follows:

$$\frac{\partial}{\partial t}\int_V \rho \, dV + \int_A \rho (\vec{u} \cdot \vec{n}) \, dA = 0$$

$$\frac{\partial}{\partial t}\int_V \rho \, e \, dV + \int_A \rho \, e (\vec{u} \cdot \vec{n}) \, dA$$

$$= \int_V [\rho (\vec{f} \cdot \vec{u}) + \rho r] dV + \int_A [(\vec{\vec{T}} \cdot \vec{u}) - \vec{q}] \vec{n} \, dA$$

$$\frac{\partial}{\partial t}\int_V \rho \, \vec{u} \, dV + \int_A \rho \, \vec{u} \, (\vec{u} \cdot \vec{n}) \, dA$$

$$= \int_V \rho \, f \, dV + \int_A (\vec{\vec{T}} \cdot \vec{n}) \, dA$$

where $\vec{f}$ is the sum of all body force acting on the fluid and r is the rate of integral heat generation per unit mass from all sources. The surface stress is denoted by the tensor $\overleftrightarrow{T}$ and the heat flux vector by q.

2.2 Critical Heat Flux Correlation

The modified W-3(R) correlation[3] was employed in KNU 5&6 design and the WRB-1 correlation[4] in KNU 7&8 design by Westinghouse. However, in this paper, the R correlation was used for DNBR calculation for both the SFA and the OFA because the consistency in DNBR calculations on both assemblies is essential in order to compare the assemblies. The use of the R correlation for the calculation of DNBR of OFA can be justified because the thermal-hydraulic characteristics of mixing vane grid for OFA are similar to that of mixing vane grid for SFA.[5]

The R correlation was established by adopting a modified spacer factor, which has been developed to incorporate the R type mixing vane grid benefit for both the typical and the cold wall channels, to the W-3 correlation.

2.3 Void Fraction Calcualtion

For the calculation of the void fraction, the Zuber/Findly void-quality correlation[6] based on an one-dimensional drift-flux model is introduced. The drift-flux parameters are calculated by the correlation suggested by Ohkawa/Lahey.[7] The empirical drift-flux parameters are given by:

$$V_{gi} = V_{gj1}, \qquad \alpha < X,$$
$$\quad = \min(V_{gj1},\ V_{gj2}), \qquad \alpha < X,$$

and,

$$C_0 = C_{01}, \qquad \alpha < X,$$
$$\quad = \min(C_{01},\ C_{02}), \qquad \alpha \geqq X,$$

where,

$$V_{gj1} = 2.9\frac{[gg_0\sigma(\rho_f - \rho_g)]^{1/4}}{\sqrt{\rho_f}}$$

$$V_{gj2} = Y\frac{[gg_0\sigma(\rho_f - \rho_g)^{1/4}}{\sqrt{\rho_f}}\left[1-\left[\frac{\alpha - X}{1 - X}\right]\right]$$

$$C_{01} = \left[1.2 - 0.2\left[\frac{\rho_g}{\rho_f}\right]^{1/2}\right]\ [1-\exp(-18\alpha)]$$

$$C_{02} = 1.0 + 0.2\left[1-\left[\frac{\rho_g}{\rho_f}\right]^{1/2}\right]\left[1-\left[\frac{\alpha - X}{1 - X}\right]\right]$$

$$X = 0.5881164 - 1.81701\Psi + 2.00025\Psi^2 - 3.34398\Psi^3$$

$$Y = \max(Y_1,\ 3.136)$$

and,

$$Y_1 = 4.72085 - 17.26736\Psi + 56.14883\Psi^2 + 113.216\Psi^3 - 1250.603\Psi^4 + 3039.767\Psi^5 - -2431.823\Psi^6,$$

where,

$$\Psi = (\rho_g/\rho_f)^{1/2}$$

At subcooled boiling region, the flow quality, x, is evaluated by the Levy's profile fit model.[8]

2.4 Surface Heat Transfer Coefficient

Forced convection heat transfer coefficients are obtained using Dittus- Boelter correlation. After the occurrence of the onset of nucleate boiling, the outer cladding wall temperature is determined by Thom's correlation.

2.5 Power Distribution

Rodwise power distributions at the BOC, for the hot assembly of the OFA and the SFA are presented in Figure 1 and 2 , respectively. These values were obtaiend by CORE2D code.[9]. Axial power distributions at the BOC of the OFA and the SFA are presented in Figure 3. These distributions were calculated by CORE1D code.[10]

The calculations are performed for the OFA and SFA with the power distributions given in Figures 1, 2 and 3. Also the other calculations are carried out for the OFA with the power distributions for the SFA. The former calculations will show the nuclear-thermohydraulic coupled characteristics of the OFA and that of the SFA, while the latter calculations will show the difference in the thermohydraulic

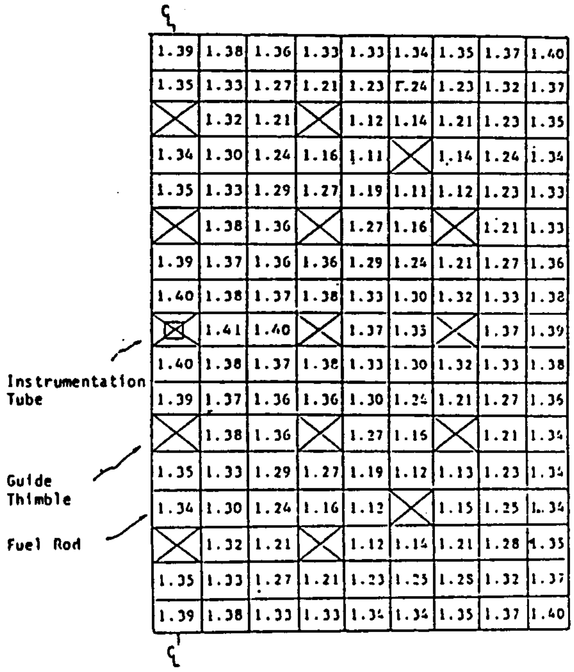

Figure 1. Rodwise Power Distributin in the Hot Assembly, KNU 7&8 (OFA), BOC, HFP, ARO Eq. Xe

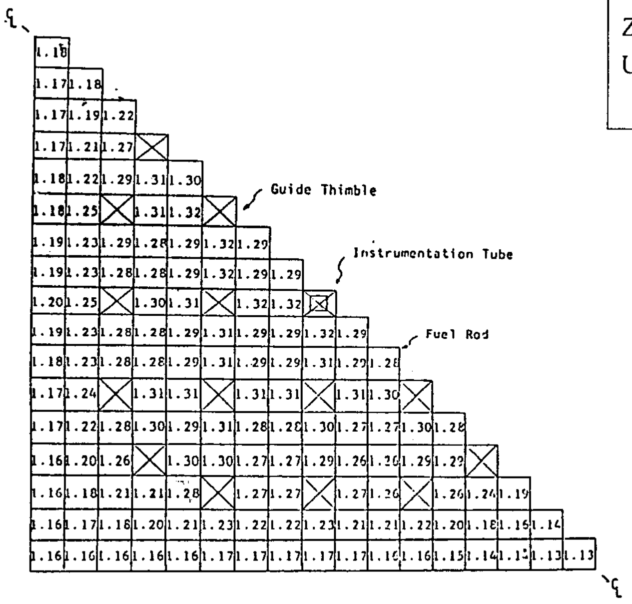

Figure 2. Rodwise Power Distribution in the Hot Assembly, KNU 5&6 (SFA, BOC, HFP, ARO Eq, Xe)

characteristics of the two assemblies.

2.6 Gap Conductance

The values of the gap conductance are obtained by Westinghouse model. For comparison, the linear powers resulting in fuel centerline melting are searched additionally by modified Ross-Stoute model and Battelle Northwest model.

2.7 Miscellaneous Input Data

1) Drag coefficient of grid is selected from Reference[11], i.e., to be 0.9.

2) The physical properties of UO2 pellet and Zircaloy-4 are obtained from Reference[12], i.e., as follows:

property material	thermal conductivity Btu/hr-ft-F	specific heat Btu/lb-F	density lb/ft3
Zr-4	12.00	0.122	489.0
UO2 sintered	1.96 (at 1470 F)	0.080	649.7

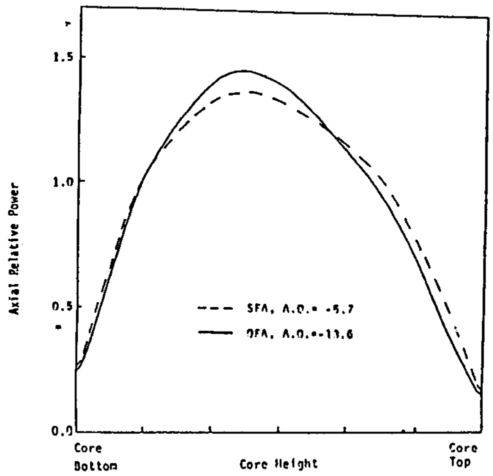

Figure 3. Axial Relative Power Distribution BOC, HFP, ARO, Eq. Xe

3) Thermal conductivity for UO2 is represented by Westing-house model.

2.8 Treatment of Uncertainties

In this study, the deterministic thermal design procedure is employed. This design procedure treats the uncertainties in the reactor design parameters very conservatively. The uncertainties are treated such as; the average heat flux (i.e., the reactor power) is increased to 102%, the system pressure is decreased to 2220 psia, the coolant temparature at the core inlet is increased by 4 F and the hot channel pitch is reduced by 7.3 mil.

A design basis of 5% reduction in coolant flow to the hot assembly should be used in the corewise thermal hydraulic analysis for the consideration of inlet flow maldistribution. But a study showed that the flow recovery at the hot assembly is so fast that the effect on DNBR is not significant. In this study, since only 1/8 of the assembly is under calculation, the 5% flow reduction in the assembly inlet will get a too conservative results. For these reasons the flow reduction is not considered in this study.

2.9 Operating Parameters

In the design of KNU 7&8, the statistical thermal design procedure was introduced while in the design of KNU 5&6, the deterministic procedure was applied thus even though the NSSS characteristics of KNU 7&8 is nearly same as that of KNU 5&6, the thermal design parameters for the DNBR calculation are different each other as shown in Table 1. Since the key ojbect of this study is the comparison of the thermal-hydraulic characteristics of the OFA in KNU 7&8 with that of the SFA in KNU 5&6, the operating parameters should be re-established.

The system pressure, the core inlet temperature and the core power level of the KNU 7&8 are adjusted to be equal to those of the KNU 5&6. The inlet mass flux of the KNU 7&8 is proportionally calculated from

Table 1. Reactor Design Parameters for KNU 5&6 and KNU 7&B

Heat Transfer	KNU 5&6	KNU 7&8
10. Average Hat Flux, Btu/hr-ft	189800	197200
11. Average Linear Power, kW/ft	5.44	5.44
12. Peak Linear Power Resulting from Overpower Transients/Operaror Errors (Assuming a Maximum Overpower of 118%), kW/ft	18.0	18.0
Fuel Assembly Design	**KNU 5&6**	**KNU 7&8**
13. Number of Fuel Assembly	157	157
14: Fuel Rods per Assembly	264	264
15. Rod Pitch, inch	0.496	0.496
16. Rod Outer Diameter, inch	0.374	0.360
17. Rod Diametral Gap, inch	0.0065	0.0062
18. Cladding Thickness	0.0225	0.0225
19. Pellet Diameter, inch	0.3255	0.3225
20. Active Fuel Height, inch	144	144

Table 2. Reactor Design Parameters for KNU 5&6 and KNU 7&8
(continued)

Thermal-Hydraulic Design Parameters	KNU 5&6	KNU 7&B
1. Reactor Core Heat Output, MWt	2775	2775
2. System Pressure, Nominal, psia	2250	2280
3. Minimum DNBR at Nominal Conditions		
— Typical Flow Channel	2.03	2.37
— Thimble flow Channel	1.72	2.21
4. Minimum DNBR for Design Transients		
— Typical Flow Channel	>1.30	>1.49
— Thimble flow Channel	>1.30	>1.47
5. DNB Correlation	R	WRB-1
Coolant	KNU 5&6	KNU 7&8
6. Nominal Inlet Temperature, F	557.0	557.7
7. Average Rise in Core, F	66.5	64.2
8. Effective Flow Rate for Heat		
Transfer, 10 lbm/hr	102.4	106.4
9. Average Mass Velocity, 10 lbm/hr-ft	2.46	2.42

that of the KNU 5&6.

III. Analysis of Dnbr Sensitivity

3.1 Reference Core

For the DNBR sensitivity study, the reference core is established by setting all input variables to their best-estimate values obtained from KNU 5&6 and KNU 7&8 full power operating condition. The calculations are carried out on a biseeted 3 x 3 rod-bundle geometry that is composed of one thimble rod and five fuel rods.

3.2 Parametric Study

The parametric study has been carried out for the reactor operating parameters such as, core power level, system pressure, core inlet flow rate, core inlet temparature and axial offset. Enthalpy rise hot channel factor is not included in the parametric study, because the six rod-bundle geometry is so small that the difference between the sensitivity factor of core power level and that of enthalpy rise hot channel factor can not be distinguished.

Calculations of the sensitivity factors are performed for the range of 90% to 100% of the reference values of the parameters.

In the axial offset analysis, the power distributions are supplemented by other axial shapes skewed to the bottom and top of the core (i.e., function $\mu \sin\mu$ with $F^N_Z = 1.55$).

4.1 Normal and Overpower Conditions

Under normal operation conditions, the minimum DNBR values are somewhat different between OFA and SFA, as shown in Figure 4. The values of minimum DNBR for OFA is 2.0 and that for SFA is 2.3. The fuel centerline temperature is 3220 F for OFA and 2870 F for SFA. As shown in Figure 5, the subcooled boiling initiated in OFA and in SFA, at the position of 40 inches and 50 inches from the bottom of

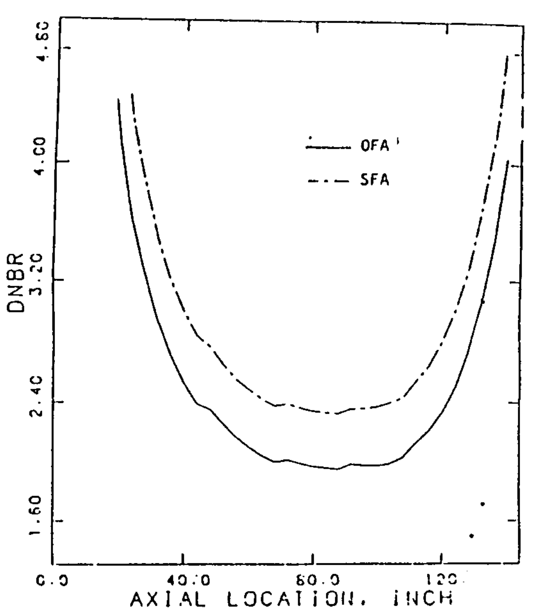

Figure 4. DNBR Distribition on Hot Rod, Normal Condition

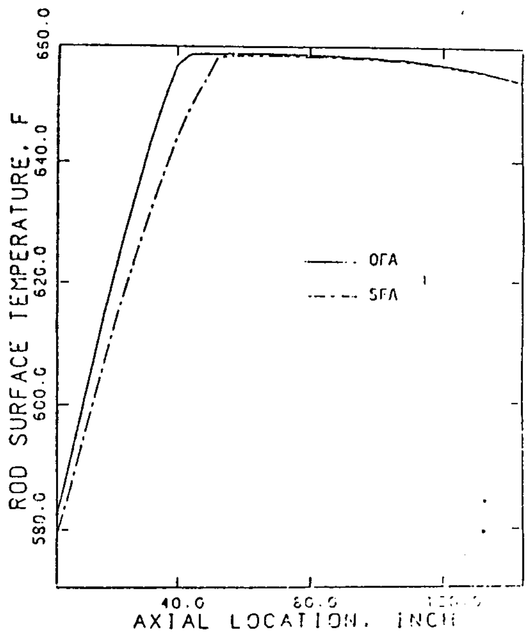

Figure 5. Hot Rod Surface Temperature, Normal Conditoin

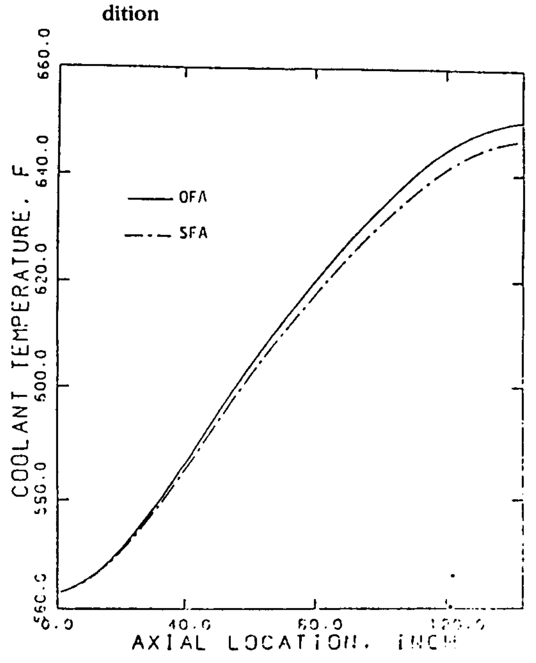

Figure 6. Coolant Temperature at Hot Channel, Normal Condition

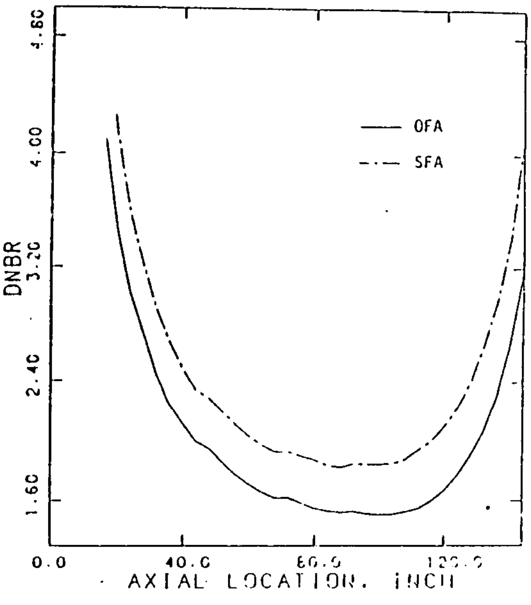

Figure 7. DNBR Distribution on Hot Rod, Overpower Condition

core, respectively. The coolant temperature rise is 87.4 F for the hot channel of OFA and 83.7 F for SFA. The results are shown in Figure 6.

During 118% overpower condition, the values of minimum DNBR for OFA is 1.5 and that for SFA is 1.8, as shown in Figure 7. The fuel centerline temperature is 3700 F for OFA and 3310 F for SFA, and the hot channel exit void fraction is 0.228 and

Table 2. Comparison of Results

Parameters		STD	OFA	OFA*
Minimum DNBR	Nominal	2.32	1.95	2.10
	Overpower	1.82	1.50	1.65
Fuel Centerline Temperature, F	Nominal	2870	3220	—
	Overpower	3310	3700	—
Peak Linear Power for Fuel Center line Melt, kW/ft	Westinghouse	20.2	20.2	—
	modified Ross-Stou	20.5	20.4	—
	Battle Northwest	19.8	19.7	—
Exit Void Fraction**	Normal	0.0 (0.00)	0.0 (0.00)	0.0 (0.00)
	Overpower	0.141 (0.169)	0.144 (0.228)	0.142 (0.191)
Bundle Average Exit Enthalpy	Normal	687.13	687.33	687.13
	Overpower	708.93	709.13	708.93

*) OFA with the same power distribution as SFA
**) void fraction of assembly average & (hot channel)

0.169 for OFA and SFA, respectively. The local boiling occures at 36 inches for the hot rod of OFA and 40 inches for SFA. The bulk boiling starts at 108 inches for the hot rod of OFA and 116 inches for SFA. The coolant temperature rise is 87.9 F for the hot channel of OFA and SFA.

The difference of gap conductances, calculated by using Westinghouse model for OFA and SFA is less than 5%.

The linear powers resulting in fuel centerline melting are 20.2 kW/ft for both OFA and SFA. For the case where the gap conductance is calculated by other models, the differences between the local linear powers resulting in fuel centerline melting for OFA and SFA is negligible.

For the clearer comparison of thermal-hydralulic characteristics, another independent calculation is carried out for the KNU 7&8 with the same power distribution as the KNU 5&6. From the calculated results, decrese in minimum DNBR of OFA compared with that of SFA is also found.

All above results are summarized in Table 2.

4.2 Parametric Study

The minimum DNBR increases as the core inlet flow rate or the system pressure increases. The minimum DNBR is a decreasing function of the inlet temperature, the power level and the axial offset. All the calculated sensitivity factors are listed in Table 3.

Table 3. DNBR Sensitivity Factors and Importance Ranking of Each Parameters

Parameter	Importance Ranking	Sensitivity Factor	
		Typical Cell OFA (SFA)	Thimble Cell OFA (SFA)
Temperature	1	-4.96 (-5.16)	-3.63 (-3.77)
Power	2	-1.45 (-1.47)	-1.24 (-1.26)
Pressure	3	1.33 (1.39)	0.87 (0.90)
Flow Rate	4	1.19 (1.24)	0.91 (0.94)
Axial Offset	5	-0.94 (-0.97)	-0.79 (-0.81)

V. Discussions and Conclusions

The DNBR values are lower in the OFA than in the SFA for all cases. The higher heat flux in the OFA which is caused dy the smaller fuel outer diameter compared with the SFA, mainly affected to DNBR.

The peak linear power resulting in fuel centerline melting in the OFA is nearly equal to that in the SFA, although the fuel centerline temperatures at normal and at overpower conditions are somewhat different.

The DNBR sensitivity factors are smaller in the OFA than in the SFA. It means that the DNBR limit by ITDP (Improved Thermal Design Procedure) will be lower in the OFA than in the SFA.

Thus it can be concluded that even though the possibility of DNB occurrence is higher in the OFA than in the SFA, the OFA satisfies the DNBR safety limit of the R correlation at the 118% overpower conditions. Fuel centerline temperature for both the OFA and the SFA is far lower than the safety limit. The local peak linear power resulting in fuel centerline melting of the OFA is nearly equal to that of the SFA and higher than the value presented in FSAR. Exit void fraction in the OFA is slightly less than that in the SFA.

The most sensitive parameter in DNBR thermal design is the coolant temperature, while the least sensitive parameter is the axial offset. The reactor operating parameters of the OFA is a little less sensitive than that of the SFA in DNBR thermal design.

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