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《Original》

DNBR Sensitivities to Variations in PWR Operating Parameters

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가압경수로의 운전변수 변화에 대한 DNBR의 민감도

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Abstract

Analyzed are the the DNBR(Departure from Nucleate Boiling Ratio) sensitivities to variations in various PWR operating parameters utilizing the 'Korea Nuclear Unit 1(KNU-1) design and operating data. Studied parameters in the analysis are core power level, system pressure, core inlet flow rate, core inlet temperature, enthalpy rise hot channel factor, axial power peaking factor and axial offset. The calculations are performed using the steady state and transient thermal-hydraulics computer program, COBRA-IV-K, which is the revised version of COBRA-IV-i that has been adapted, partially modified and verified at KAERI. A reference case is established based on the design and operating condition of the KNU-1 reactor core, and this provides a basis for the subsequent sensitivity analysis. From the calculation results it is concluded that the most sensitive parameter in the DNBR thermal design is the coolant core inlet temperature while the axial power peaking factor is the least sensitive.

요 약

한국원자력 1호기(KNU-1)의 설계 및 운전자료를 이용하여 가압경수로 운전변수들의 변화에 대한 DNBR의 민감도를 분석하였다. 본 민감도 분석에는 원자로 출력, 압력, 냉각수 주입유량, 냉각수 주입온도, 반경방향 및 축방향 출력분포 그리고 축방향 출력편차 등의 운전변수가 고려되었다.

민감도 분석을 위하여는 노심의 열수력 해석용 전산코드인 COBRA-IV-K를 사용하였는데 본 코드는 COBRA-IV-i의 수정판으로써 한국에너지연구소에서 일부 프로그램을 수정하였고 또한 신뢰도도 확인하였다. 민감도 분석을 수행하기 전에 KNU-1 원자로심의 설계 및 운전조건을 근거로 하여 기초계산을 수행하고 이 결과를 본 민감도 분석의 기본자료로 삼았다. 민감도 분석결과 원자로의 DNBR 열설계에 있어서 가장 민감한 운전변수는 냉각수 주입온도이고 가장 둔감한 변수는 축방향 출력분포라는 것이 밝혀졌다.

1. Introduction

The thermal and hydraulic design of pressurized water reactor (PWR) maintains core thermal protection during normal operations and incidents of moderate frequency by avoiding departure from nucleate boiling (DNB). The DNB ratio (DNBR), that is defined as the ratio of critical heat flux (CHF) to the actual heat flux at the point of interest, is calculated and compared with the corresponding safety criterion to assure the fuel integrity.

The objectives of the DNBR sensitivity analysis in this paper are to look into the effect of changes in PWR core parameters on DNBR, and to apply the analysis results for the following purposes:

a) investigation of the most and least sensitive (important) parameters in the core thermal and hydraulic design and reactor operation,

b) approximate but quick estimate of variations in core thermal margin due to changes in PWR design or operating parameters, and

c) establishment of a basis for statistical core thermal design in the future.

An extensive sensitivity analysis is performed utilizing the steady state and transient thermal-hydraulics computer program, COBRA-IV-K. that has been adapted and partially modified at the Korea Advanced Energy Research Institute (KAERI).

The COBRA-IV-K code has been verified against its application to the steady state and transient analyses. A reference case is established based on the design and operation condition of the Korea Nuclear Unit 1 (KNU-1) reactor core and provides the basis of the subsequent sensitivity analysis. Studied parameters for the sensitivity analysis are core power level, system pressure, core inlet flow rate, core inlet temperature, enthalpy rise hot channel factor, axial power peaking factor and axial offset. A concept of sensitivity factor is adopted to represent the sensitivity of DNBR to changes in each parameter, and the calculated sensitivity factors are compared with each other to decide the relative importance of each parameter in the PWR DNBR thermal design.

2. Thermal-Hydraulic Computer Program; COBRA-IV-K

2.1. Description of COBRA-IV-K

COBRA-IV-K[1] is a subchannel thermal-hydraulic analysis computer program which determines coolant density, mass velocity, enthalpy, static pressure and flow distribution under both steady state and transient conditions. The basic programming approach in the COBRA-IV-K code is to divide a bundle into several computational cells, where variables like fluid enthalpy, pressure and velocity are appropriately averaged. Balance equations of mass, momentum and energy are then numerically solved to find local properties of fluid such as enthalpy, density, axial flow rate, lateral flow rate and static pressure. COBRA-IV-K is also capable of predicting DNBR and fuel temperature. Through a significant progress in numerical capability incorporated into the code, also possible are analyses of transients like blowdown, flow blockage, coolant expulsion, reentry and recirculation.

2.2. Modification of the Program

Due to the large number of subchannels in a reactor core it is impractical, if not impossible, to apply the subchannel analysis technique to the whole core. Therefore, partial modification of modeling in the program has been incorporated into the COBRA-IV-i code[2] to add capability of corewide thermal-hydraulic analysis under steady state and transient conditions. Employed are the following modeling schemes for the

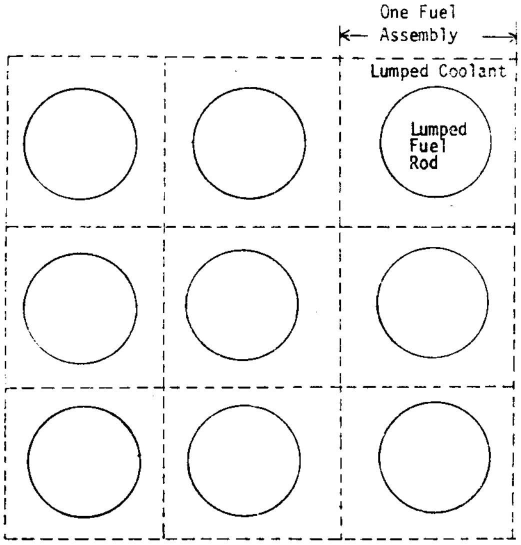

Fig. 1. Fuel Assembly Modeling for Corewide Thermal-Hydraulic Analysis Based on Zero Shear Stress Assumption between Assemblies

corewide calculation:

a) Each fuel assembly is represented by a single channel with a rod-centered geometry as shown in Figure 1.

b) Each channel is given lumped parameter values and uniform fluid properties.

c) No Pressure gradient is assumed to exist between fuel assemblies.

The assumption of no pressure gradient across each fuel assembly leads to the modification of the lateral momentum balance equation into the following form:

$$\frac{\partial(\rho uv)}{\partial x}+\frac{\partial}{\partial y}(\rho v^2)=-K\rho|v|v$$

where x : axial direction

y : lateral direction

ρ : density of fluid

u : axial velocity of fluid

v : lateral velocity of fluid

K : frictional resistance

The boundary conditions employed for the corewide analysis are as follows:

a) variable core inlet temperature or enthalpy

b) variable inlet flow and pressure distribution

c) uniform core outlet pressure distribution

2.3. Code Verification

Verification calculations of COBRA-IV-K for the corewide application are performed by investigating steady state thermal-hydraulic parameters of the KNU-1 reactor core adapting 1/8 core model. shown in Table 1 are the calculated results and corresponding KNU-1 FSAR[3] values for cycle 1, BOL under ARO (all rods out) eguilibrium xenon condition, and they are in good agreement with each other. Also, investigated is a assembly-wise enthalpy distribution in the core assuming lnlet flow maldistribution. Figure 2 shows the assumed core inlet flow di-

Table 1. COBRA-IV-K Calculated KNU-1 Reactor Thermal-Hydraulic Parameters Compared to the FSAR Values for Cycle 1, BOL Under ARO Condition

Parameter	COBRA-IV-K	FSAR
Core Temperature Rise (°F)	68.1	68.5
Core Pressure Drop (psi)	23.9	22.4
Core Average Temperature (°F)	575.3	576.9
Average Coolant Velocity (ft/sec)	15.2	14.9
Hot Assembly Exit Temperature (°F)	618.4	N/A

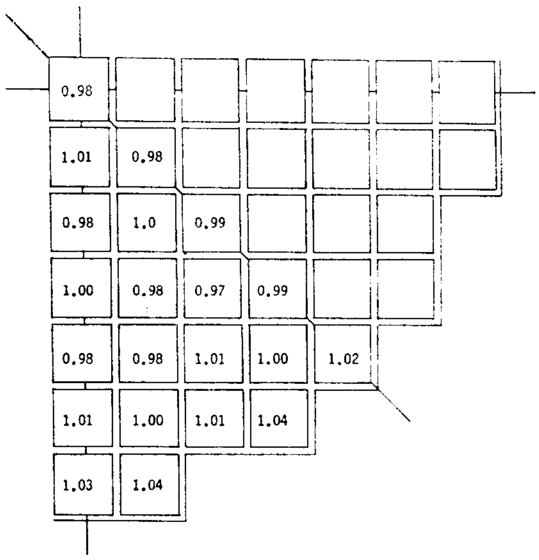

Fig. 2. Core Inlet Flow Distribution Assumed for Core Inlet Flow Maldistribution Analysis

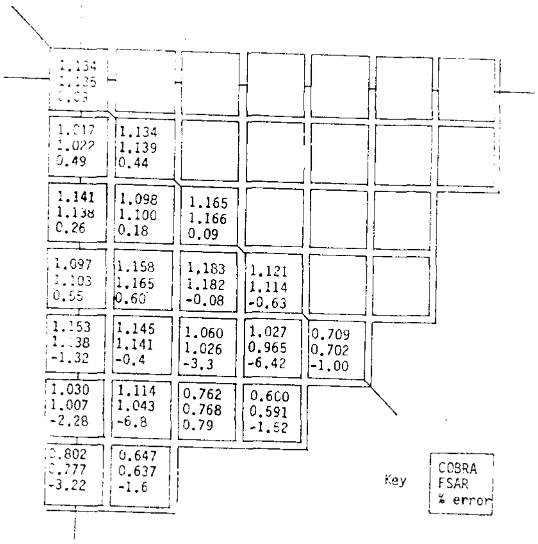

Fig. 3. Normalized Enthalpy Distribution in 1/8 Core of KNU-1

stribution used for the KNU-1 flow maldistribution calculation. Figure 3 shows the normalized enthalpy distributions; one by the COBRA-IV-K calculation and the other from the FSAR, and they are in reasonably good agreement with the largest error under 7%. Through the above mentioned verification calculations, the COBRA-IV-K code is proven capable of reliable corewide analysis.

To verify COBRA-IV-K against transient analysis, studied are transients like uncontrolled control bank withdrawal from power, partial loss of coolant flow and complete loss of coolant flow for KNU-1. The calculation results (DNBR vs time) are shown in Figures 4, 5 and 6, and

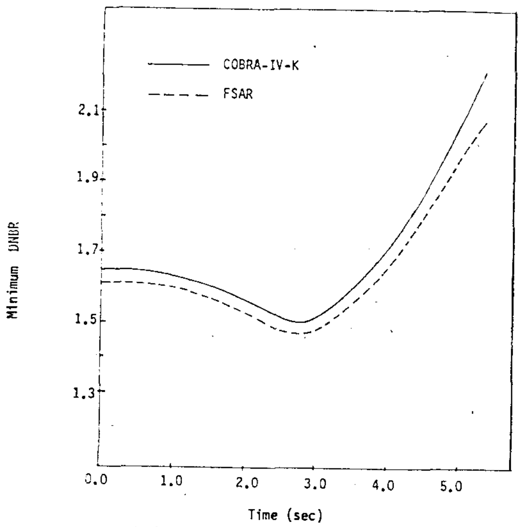

Fig. 4. Minimum DRBR vs Time for Uncontrolled Control Bank Withdrawal from Power at 75p cm/sec for KNU-1

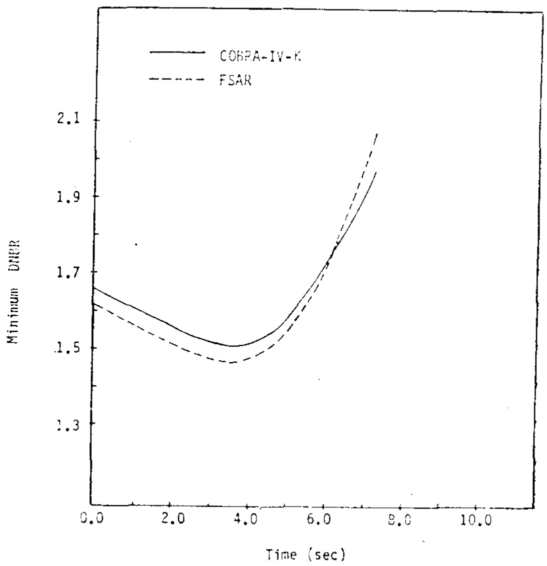

Fig. 5. Minimum DNBR vs Time for Partial Loss of Flow of KNU-1

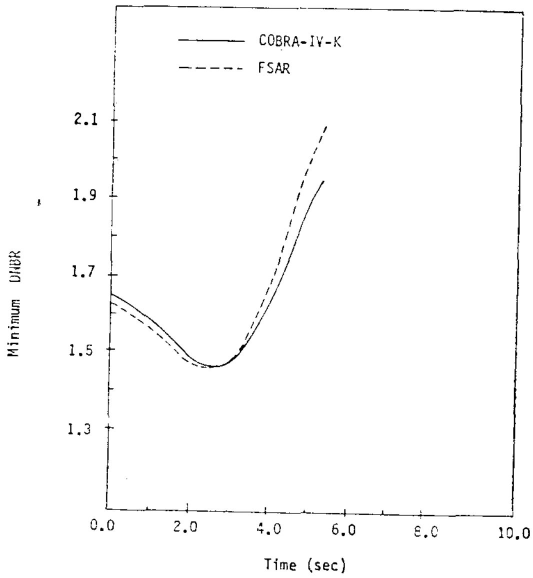

Fig. 6. Minimum DNBR vs Time for Complete Loss of Coolant Flow of KNU-1

compared with those from the FSAR. From these Figures, the COBRA-IV-K is evaluated to be reliable for core thermal transient analysis including the DNBR sensitivity analysis. Minor disagreements as shown in the Figures are suspected to result from the differences initial condition and trip reactivity insertion rate between the COBRA-IV-K and FSAR Calculations.

3. Sensitivity Analysis

Preceding the extensive study on sensitivities ofDNBR to variations in PWR operating parameters, a concept of sensitivity factor is introduced to represent the importance of each parameter in the thermal-hydraulic design. The sensitivity factor is defined as the percentage change in DNBR resulting from one a percent change in each parameter, all other parameters being held constant. And, the value of the sensitivity factor associated with each parameter is obtained by averaging the calculated values within the plus and minus 10 percent about the reference point for that parameter. For the sensitivity study it is also required to establish the reference case that provides the basis for various parametric studies.

3.1 Reference Case

The reference case for the PWR DNBR sensitivity analysis is established by setting all input variables to their conservative values obtained from the KNU-1 full power steady state operating condition. The calculation is carried out on a nine rod-bundle geometry in the hottest region of the core. Figure 7 describes the nine rod pitch and diameter. Here, the typical cell represents a coolant channel cornered by four fuel rods, while the thimble cell is surrounded by three fuel rods and one guide tube. Listed in Table 2 are the values of COBRA-IV-K input parameters for the reference calculation as well as corresponding nominal values. For the sake of conservatism, the radial power distribution in the bundle is assumed to be uniform at the normalized power of 1.55 except for the guide tube. The axial power distribution is of a chopped cosine shape with its normalized peak power at 1.55.

The calculated minimum DNBRs for the reference case are 2.38 and 2.15 for the typical and thimble cells, respectively. It is noted that the minimum DNBR of the thimble cell is about 10 percent smaller than that of the typical cell

Table 2. Input Parameters to COBRA-IV-K and Corresponsing Nominal Values for KNU-1

Parameter	Unit	Nominal	Input Value
System Pressure	psia	2250	2220
Core Inlet Temperature	°F	541.2	545.2
Core Inlet Flow Rate	10^6lb/hr ft^2	2.40	2.28
Inlet Flow to Hot Assembly	% of nominal	variable	95
Core Average Heat Flux	Btu/hr ft^2	201200	205224
Power Level	%	100	102
$F^N_{\Delta H}$(rod)	—	1.435	1.55
F_z	—	variable	1.55cosine
TDC	—	0.059	0.019
Fuel Rod O.D.	inch	0.422	0.422
Thimble O.D.	inch	0.539	0.539
DNB Correlation	—	—	W-3
Core Inlet Temperature Distribution	—	variable	uniform
Core Outlet Pressure Distribution	—	variable	uniform

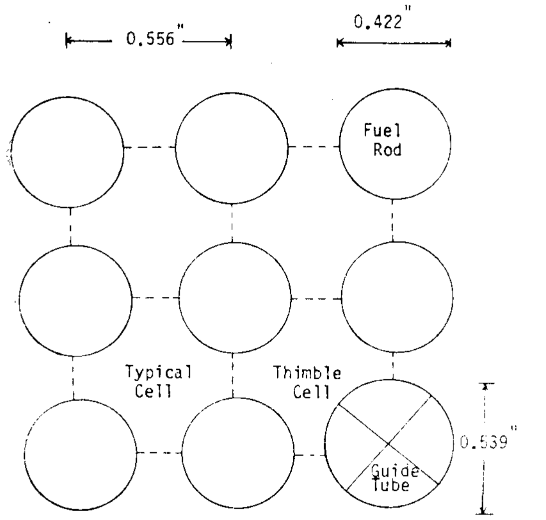

Fig. 7. Nine Rod-Bundle Model for Subchannel Analysis of the KNU-1 Reactor Core

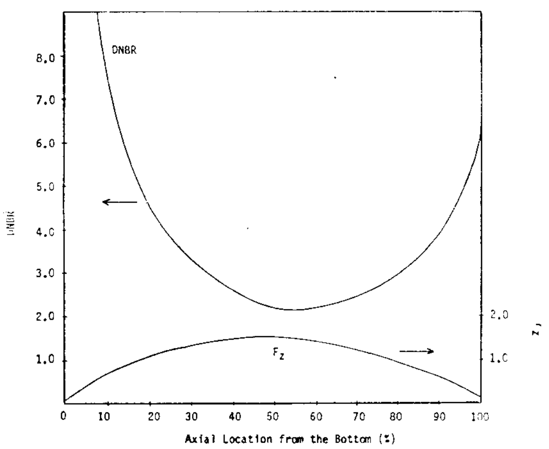

Fig. 8. DNBR and F_z along the Axial Direction on a Hot Rod for the Reference Case

when using the W-3 critical heat flux correlation, and the thimble cell is more limiting with respect to DNBR. The reduction in the minimum DNBR in the thimble cell compared to the typical cell is due to the reduction in flow area in the coolant channel and the cold wall effect. The cold wall liquid film which builds up around unheated guide tube does not contribute to cooling the heated fuel rod surface[4]. Figure 8 depicts the relationship between DNBR and normalized axial heat flux in chopped cosine shape. The axial location at which the minimum DNBR occurs is 55 percent from the bottom of the core.

3.2 Parametric Analysis

1) Core Power Level

Since low core power levels are of little concern in DNBR thermal design, power levels ranging from 70 to 120 percent are taken into consideration. Rest of the operating parameters are of the same values as those for the reference case. Shown in Figure 9 are minimum DNBRs versus core power level for both typical and thimble cells. As expected the minimum DNBR decreases with increasing power level and the thimble cell is more limiting with respect to the occurrence of DNB. The sensitivity factors of core power are calculated to be -2.17 and -1.69 for the typical and thimble cells, respectively. This also implies that DNBR in the typical cell is more sensitive to the core power variation.

2) System Pressure

The effect of system pressure change on DNBR is investigated within the pressuree range of

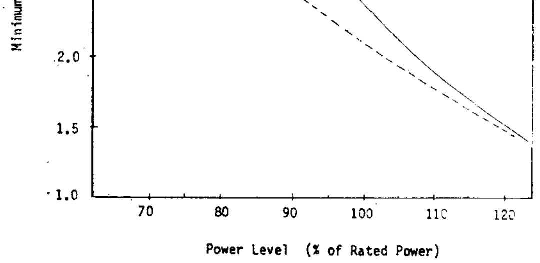

Fig. 9. Minimum DNBR vs Power Level for Typical and Thimble Cells of KNU-1

2000 to 2300 psia. Calculated minimum DNBRs versus system pressure for typical and thimble cells are shown in Figure 10. The minimum DNBR linearly increases as the pressure increases showing that more thermal margin is available at higher system pressure. The calculated sensitivity factors are 1.57 for the typical cell and 0.87 for the thimble cell.

3) Core Inlet Flow Rate

The importance of core inlet flow rate variation to DNBR is studied between 90 and 110 percent of the nominal value, and the calculation results are shown in Figure 11. The minimum DNBR increases linearly as the core inlet flow rate increases, and the sensitivity factors are 1.39 and 0.79 for typical and thimble cells, respectively.

4) Core Inlet Temperature

The DNBR sensitivity to core inlet temperature change is investigated varying inlet temperature between 535°F and 555°F. and the results are in Figure 12. The minimum DNBR is a linearly decreasing function of the inlet temperature for both typical and thimble cells. The sensitivity factor of typical cell is −6.27 while it is −4.03 for thimble cell.

5) Enthalpy Rise Hot Channel Factor

Generally, radial power distribution is represented by the enthalpy rise hot channel factor ($F^N_{\Delta H}$), which may be defined as the ratio of the hottest rod power to the average rod power in the core. To investigate the sensitivity factor of $F^N_{\Delta H}$, calculations are performed using the values of $F^N_{\Delta H}$ from 1.30 to 1.70. Shown in Figure 13 are minimum DNBRs against $F^N_{\Delta H}$

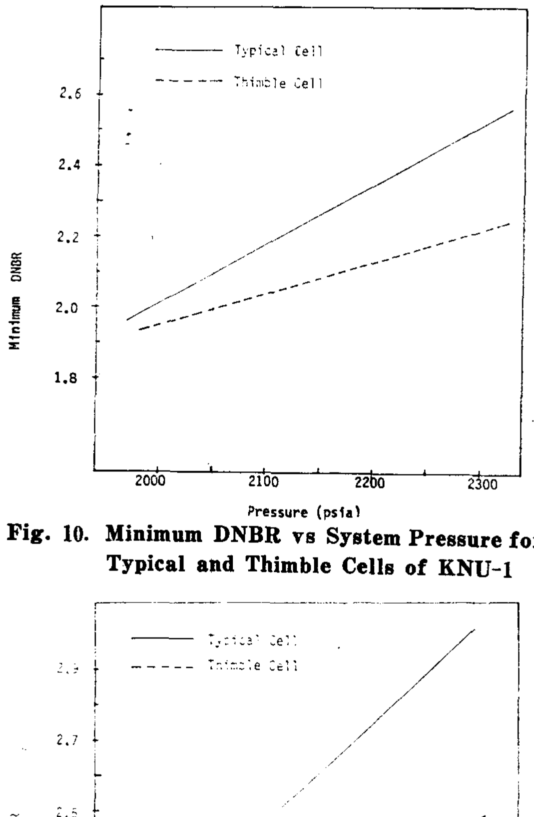

Fig. 10. Minimum DNBR vs System Pressure for Typical and Thimble Cells of KNU-1

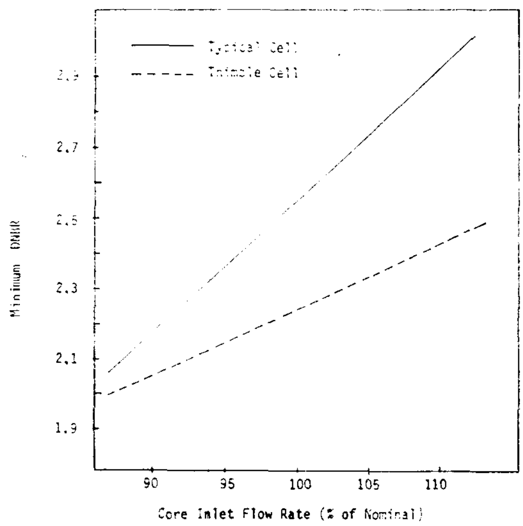

Fig. 11. Minimum DNBR vs Core Inlet Flow Rate for Typical and Thimble Cells of KNU-1

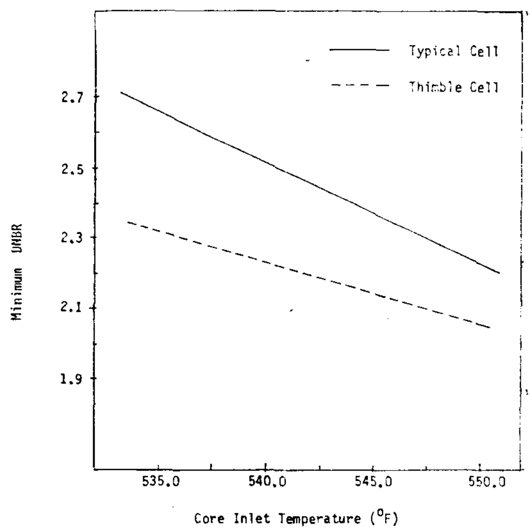

Fig. 12. Minimum DNBR vs Core Inlet Temperature for Typical and Thimble Cells of KNU-1

for both typical and thimble cells. The minimum DNBR decreases with $F^N_{\Delta H}$ increase, and the sensitivity factors for typical and thimble cells are -2.05 and -1.65, respectively.

Effects of variations in adjacent fuel rod power are also studied by reducing the power of those rods. For this study, the relative power of the surrounding rods is assumed to gradually decrease from 1.55 to 1.44 while the relative power of the limiting rod is held constant at 1.55. Figure 14 shows changes in minimum DNBR versus changes in surrounding rod power. The minimum DNBR increases as the surrounding rod power decreases for both cells, however, their sensitivity factors (-0.9 for typical cell and -0.3 for thimble cell) are not as large as those of $F^N_{\Delta H}$

6) Axial Power Peaking Factor

In studying the effect of axial power peaking Factor(F_z) variations on DNBR, the radial power peaking factor is assumed to be held constant at the design limit. Then the values for

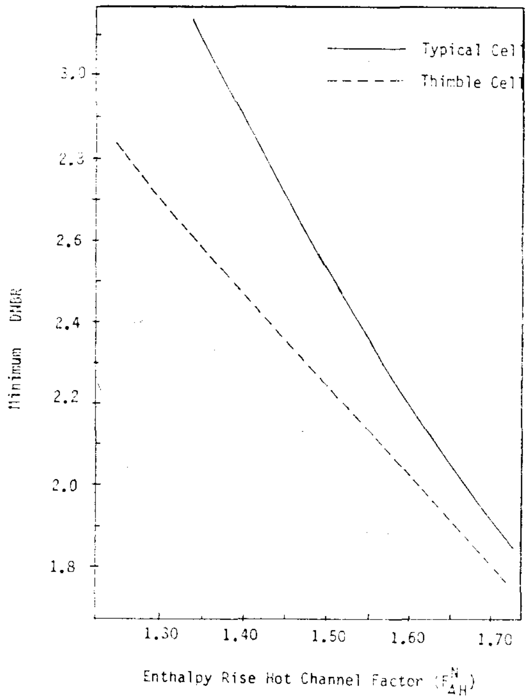

Fig. 13. Minimum DNBR vs $F^N_{\Delta H}$ for Typical and Thimble Cells of KNU-1

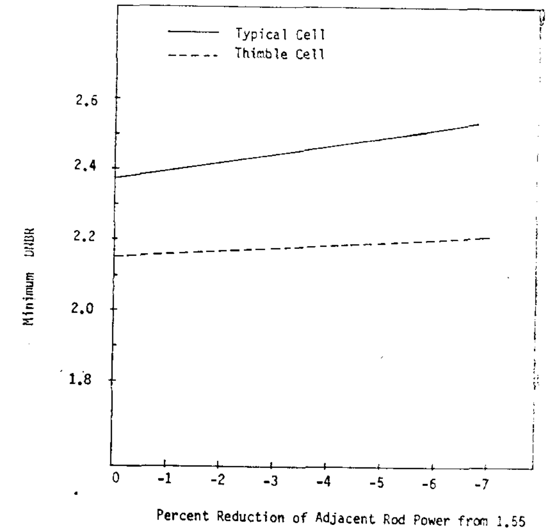

Fig. 14. Minimum DNBR vs Percent Reduction of Adjacent Rod Power from 1.55 for Typical and Thimble Cells of KNU-1

minimum DNBR are determined by the location and magnitude of the axial peaking factor. It is generally believed that axial power peaking factor can vary due to xenon axial oscillation or/and control rod movement during normal operation, and if not properly controlled it can cause fuel damage.

The sensitivity of DNBR to Fz variations is investigated using various axial power shapes with different axial peaking factors. The axial power distribution corresponding to Fz of 1.0 is assumed to be of a uniform flux shape, while rest of the calculations assume chopped cosine shapes. Figure 15 shows minimum DNBRs versus Fz for typical and thimble cells, and the minimum DNBR decreases with increasing Fz. It is relatively a slowly varying function of Fz below the value of 1.55. The sensitivity factors of axial power peaking factor are -0.77 and -0.99 for typical and thimble cells, respectively.

7) Axial Offset

Usually axial offset is used to represent the axial power distribution of the core, and defined as the difference between the fraction of full-

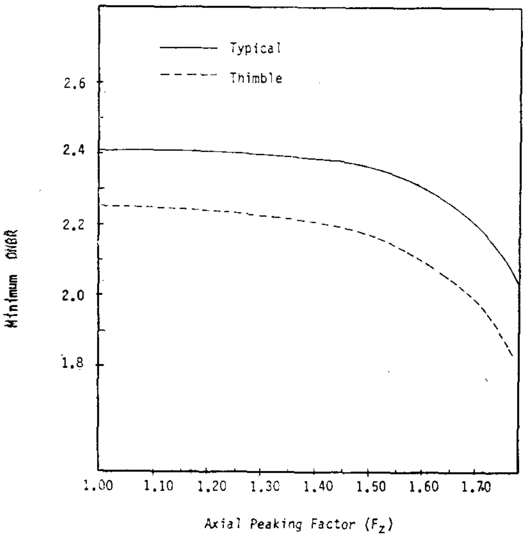

Fig. 15. Minimum DNBR vs F_z for Typical and Thimble Cells of KNU-1

rated power generated in the top half of the core and that in the bottom half divided by the relative power of the reactor[5]. It is generally noted that the minimum DNBR decreases with increasing axial offset for a given axial peaking factor. This is due to the higher coolant enthalpy in the upper region of the core that results in the lower critical heat flux and resultant lower DNBR.

Before the axial offset sensitivity factors are calculated, minimum DNBRs versus axial offset are investigated in "flyspeck" format. In generating flyspeck, axial power peaking factors correponding to axial offsets are selected from Figure 16, while the core power level is held at 100 percent of the rated power. Several axial peaking factors can be chosen from the dashed region below the solid line of the Figure. Figure 17 shows minimnm DNBRs versus axial offset in flyspeck format and this partially proves the above argument that minimum DNBR decreases with increasing axial offset. Minimum DNBRs versus axial offset are also studied at design overpower condition (118 percent of the rated power), and the results are shown in Figure

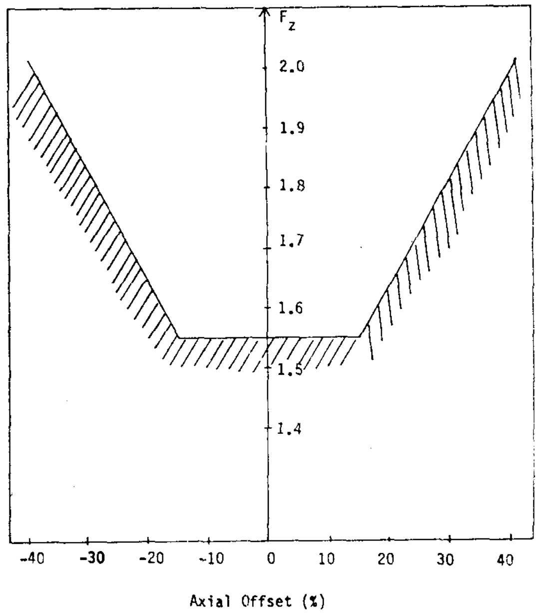

Fig. 16. Relationship between Axial Peaking Factor and Axial Offset

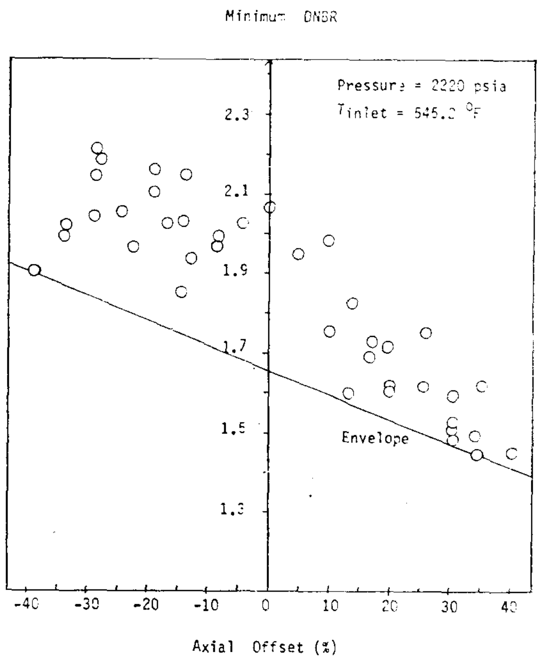

Fig. 17. Minimum DNBR vs Axial Offset at the Rated Power Condition of KNU-1 in Flyspeck Format

18. Unlike the case of the 100 percent power condition, the minimum DNBR drops below the

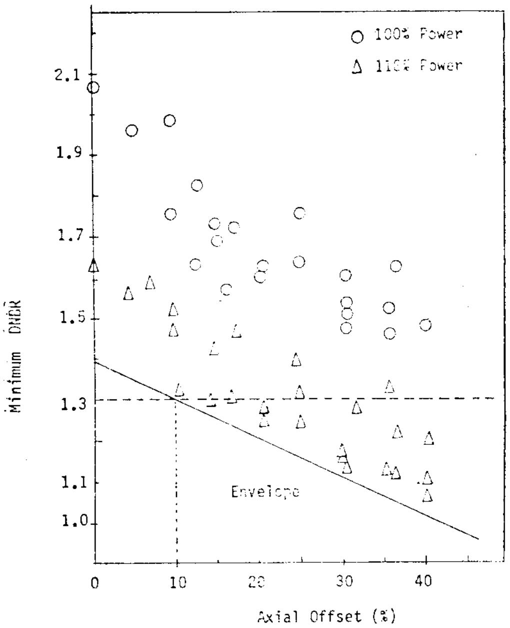

Fig. 18. Minimum DNBR vs Positive Axial Offset at the Rated and Design Overpower Condition of KNU-1 in Flyspeck Format

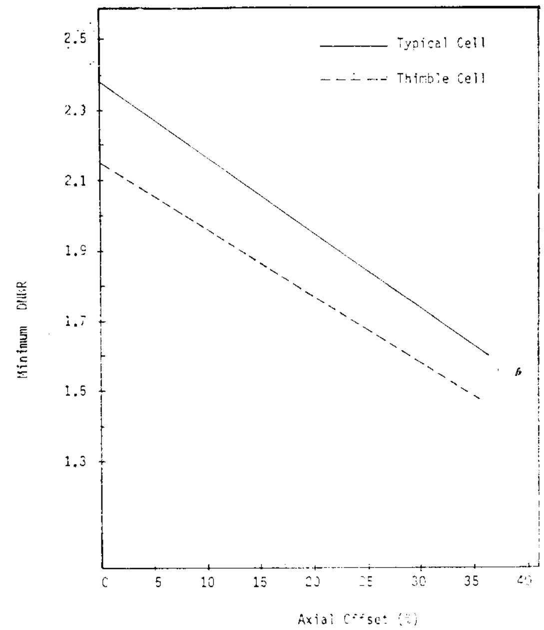

Fig. 19. Minimum DNBR vs Axial Offset for Typical and Thimble Cells of KNU-1

design limit of 1.3 above the axial offset of 10 percent. That means a reactor operation with the axial offset larger than 10 precent at the design overpower condition may lead to DNB and resultant fuel failure. Therefore, it is required to operate the reactor at the reduced power with large values of axial offsets.

The calculation of the sensitivity factor of the axial offset is performed varying the axial offset from zero to 35 percent with the axial peaking factor fixed at 1.55, that is, the axial power distribution is of a skewed cosine shape with the peak location switching from the bottom to the top of the core. Figure 19 shows the minimum DNBR linearly deceases as the axial offset increases for both typical and thimble cells. The calculated sensitivity factors are -0.96 for typical cell and -0.90 for thimble cell.

3.3. Discussion on Analysis Results

Summarized in Table 3 are the DNBR sensitivity factors of the KNU-1 operating parameters and the ranking of each parmeter in importance to the core DNBR thermal design. The ranking is established based on the thimble cell sensitivity factors since the thimble cell is more limiting in the DNBR thermal design of the PWR core when the W-3 correlation is utilized. The most sensitive parameter in the DNBR thermal design is the core inlet temperature followed by the core power level and the enthalpy rise hot channel factor while the least sensitive is the axial power peaking factor. The fact that the change in the axial peaking factor is less important than the enthalpy rise hot channel factor is also well implied in Figure 20 in which DNBRs are plotted against percent change in F_z and $F^N_{\Delta H}$. For this analysis, total hot channel factor ($F_Q{}^T$) is assumed to be held constant at 2.16 ($F_Q{}^T$ limit for KNU-1), while and $F^N_{\Delta H}$ are changing inverse proportionally. From the figure, the minimum DNBR is shown to decrease as $F^N_{\Delta H}$ increases if F_z is decreasing, however, it increases as F_z increases due to the

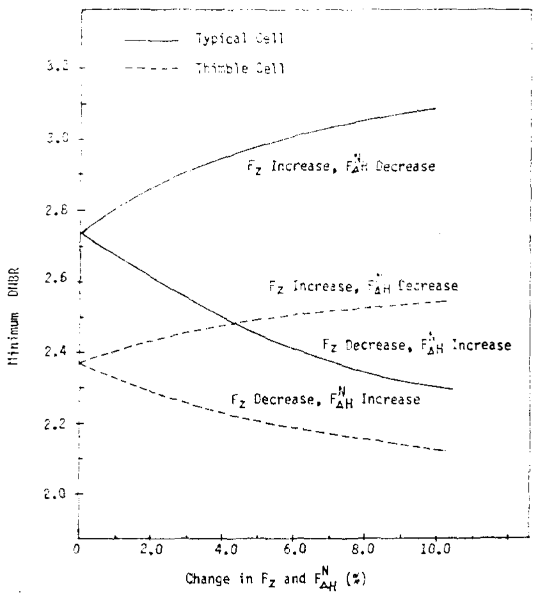

Fig. 20. Minimum DNBR Responses due to Changes in F_z and $F^N_{\Delta H}$ for Typical and Thimple Cells of KNU-1

decrease in $F^N_{\Delta H}$. This proves that the effect of $F^N_{\Delta H}$ is more pronounced than that of F_z on the PWR core DNBR thermal design.

The accuracy of all the calculated sensitivity factors can not be dirctly confirmed due to the lack of comparison data. But, the sensitivity factors of operating parameters such as enthalpy rise hot channel factor, core inlet flow rate and core inlet temperature are available for comparison purposes on conservative basis[6]. They are -1.8, 1.0 and -5.4 respectively and are also listed in Table 3, but it is not known whether these values are for the typical cell or thimble cell. From this Table it is deduced that the COBRA-IV-K calculated sensitivity factors are in reasonably good agreement with the corresponding USNRC approved values. The calculated values of sensitivity factors are found to be more conservstive than the NRC values in the case of typical cell but less conservative for thimble cell.

Table 3. DNBR Sensitivity Factors Calculated by COBRA-IV-K and Obtained From Reference 6 along with Importance Ranking of each Parameter for KNU-1

Parameter	Importance Ranking	Sensitivity Factor(%/%)		Values in Reference 6
		Typical Cell	Thimble Cell	
Core Power	2	−2.17	−1.69	N/A
System Pressure	4	1.57	0.87	N/A
Flow Rate	5	1.39	0.79	1.0
Inlet Temperature	1	−6.27	−4.03	−5.4
$F^N_{\Delta H}$	3	−2.15	−1.65	−1.8
F_z	7	−0.77	−0.90	N/A
Axial Offset	6	−0.96	−0.90	N/A

4. Conclusion

Analyzed are the sensitivities of DNBR to variations in PWR operating parameters using the subchannel analysis code, COBRA-IV-K. The code has been partially modified at KAERI, verified against its application to steady state and transient core thermal-hydraulic analysis, and found to be reliable for the sensitivity analysis purpose.

The calculated sensitivity facfors based on the KNU-1 design condition are evaluated to be reliable by the comparison with a few USNRC approved data. The most sensitive parameter in DNBR thermal design is the core inlet temperature while the axial power peaking factor is the least sensitive.

Through the sensitivity study performed herein, an insight has been gained into the degree of importance of thermal design and/or operating parameters to the PWR core thermal design. Utilizing these results it is possible to make a quick estimate of the core thermal margin changes, should changes in plant operating parameters occur. Such a quick but approximate method can be a valua ble tool in feasibility study of plant thermal margin improvement. This study also provides a basis for future development of the statistical core thermal design method which allows more thermal margin in PWR operation.

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