Journal of the Korean Nuclear Society
Volume 27, Number 4, August 1995

Evaluation of the Thermal Margin in a KOFA-Loaded Core by a Multichannel Analysis Methodology

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(Received January 12, 1995)

다수로해석 방법론에 의한 국산핵연료 노심 열적 여유도 평가

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(1995. 1. 12 접수)

Abstract

A study has been performed to investigate the thermal margin increase by replacing the single-channel analysis model with a multichannel analysis model. A new critical heat flux(CHF) correlation, which is applicable to a 17×17 Korean Fuel Assembly(KOFA)-loaded core, was developed on the basis of the local conditions predicted by the subchannel analysis code, TORC. The hot subchannel analysis was carried out by using one-stage analysis methodology with a prescribed nodal layout of the core. The result of the analysis showed that more than 5% of the thermal margin can be recovered by introducing the TORC/KRB-1 system(multichannel analysis model) instead of the PUMA/ERB-2 system(single-channel analysis model). The thermal margin increase was attributed not only to the effect of the local thermal hydraulic conditions in the hot subchannel predicted by the code, but also to the effect of the characteristics of the CHF correlation.

요 약

단일수로 해석 모형을 다수로 해석 모형으로 대체할 경우 얻을 수 있는 열적 여유도 향상에 대한 연구를 수행하였다. 이를 위하여 17×17 국산핵연료 장전 노심에 적용할 수 있는 새로운 임계열속 상관식을 개발하였으며, 여기에 사용된 부수로 국부 조건은 다수로 해석 코드인 TORC로 계산하였다. 그리고, 고온부수로 DNBR 분석을 위하여 전 노심에 대한 단일단계 해석 모형을 개발하였다. 분석 결과 다수로 해석 모형인 TORC /KRB-1 체제를 사용할 경우 단일수로 해석 모형인 PUMA /ERB-2 체제에 비하여 약 5% 이상의 열적 여유도를 회복할 수 있는 것으로 나타났다. 이러한 열적 여유도의 증가는 두 코드 간의 고온부수로 국부조건 예측 성능 차이와 임계열속 상관식의 특성 차이에서 기인한 것이다.

1. Introduction

According to the current thermal hydraulic design criteria, the reactor core and associated coolant, control, and protection systems should be designed with appropriate margins to assure that the specified acceptable fuel design limit(SAFDL) will not be exceeded at any condition of normal operation including the effects of anticipated operational occurrences (AOOs). The most limiting parameter for the core thermal power of a nuclear reactor is the departure from nucleate boiling(DNB) since it is followed by a sudden increase in fuel cladding temperature which ultimately results in the fuel failure. To ensure that the DNB ratio(DNBR) SAFDL will not be violated, CHF is calculated using empirical correlations based on the local thermal hydraulic conditions in the hot subchannel such as pressure, heat flux, mass velocity, enthalpy, and quality at the CHF location. The thermal margin is defined as the distance from the nominal operating condition to the DNBR SAFDL. Reserving an appropriate thermal margin is very important not only to increase the confidence in the fuel integrity but also to allow the potential of power uprating capability.

Many efforts have been made to improve the thermal margin by increasing the nominal DNBR ($DNBR_N$) or decreasing the design limit DNBR ($DNBR_L$). The improvement in the fuel assembly configurations, such as adoption of mixing-vaned spacer grids or intermediate flow mixers, is a promising way to accomplish this goal, but it requires expensive thermal hydraulic performance tests. These devices may produce more turbulences near the DNB location which suppress the occurrence of DNB and consequently increase the $DNBR_N$.

An alternative way is to improve the thermal hydraulic analysis methodology. This includes the improvement in the treatment of DNBR uncertainties comprising various uncertainties of system/operating parameters[1－4]. A detailed thermal hydraulic subchannel analysis of the core is necessary to calculate the flow rate and fluid enthalpy in various regions of a fuel assembly at normal operation and transient conditions, and thus to provide the local thermal hydraulic conditions in the hot subchannel used in the CHF calculation. It has been known that an open channel approach [5, 6] gives more thermal margin than a closed channel approach since the latter contains various conservative factors to compensate for its simplified thermal hydraulic modeling. Most subchannel codes apply the open channel approach in modeling the core in order to take into account the interchannel diversion crossflow and turbulent mixing effects, which results in more realistic thermal hydraulic conditions in the hot subchannel compared with those calculated by the closed channel approach. Improvements have been made in the crossflow and turbulent mixing models, the numerical solution techniques, and the two-phase flow constitutive relations such as subcooled void model and two-phase pressure drop model, etc. Improvements have also been made in the CHF correlations, which have reduced the DNBR limit while maintaining the assurance of not experiencing DNB with a 95% probability at a 95% confidence level[7－10]. All of these improved correlations have their own DNBR limits lower than the limits imposed on the other correlations such as the W-3 correlation. The improvements have resulted from a better understanding of CHF phenomena and, therefore, a better formulation of CHF correlation. Since a CHF correlation is developed for a specific fuel geometry in conjunction with a particular code for calculating the local thermal hydraulic conditions, the design DNBR calculation is restricted to the use of the same code or more conservative one.

In this study, we investigated the thermal margin improvement quantitatively by replacing the current closed channel approach(the ERB-2 CHF correlation[11] with the PUMA code[12]) for the KOFA-loaded core with an open channel approach based on the TORC code system. The TORC is a subchannel analysis code being used in the core thermal hydraulic design of the CE-type fuel assemblies, but has not been applied to the KOFA-loaded core. The applicability of TORC code to the non-CE type fuel assemblies has been studied previously[13]. So a multichannel thermal hydraulic analysis methodology by the TORC code has been developed for the analysis of the KOFA-loaded core. This includes the development of a new CHF correlation system and a hot subchannel analysis method. The thermal margin of a reference 17×17 KOFA-loaded core has been evaluated with the new methodology and compared with the current margin calculated by the closed channel approach using the PUMA code.

2. Development of a Multichannel Analysis Model for the KOFA-Loaded Core

2.1. CHF Correlation

First of all, the CHF correlation system should be established for the analysis of thermal margin. The ERB-2 correlation is presently used in the analysis of the KOFA-loaded core with the aid of the PUMA code in calculating the hot subchannel parameters. A new CHF correlation was developed, in this study, on the basis of the local conditions predicted by the multichannel analysis code, TORC, since the CHF correlation system with the TORC code appropriate to the KOFA-loaded core was not available. The appropriate functional form of the CHF correlation applicable to the KOFA has been studied[13]. The CHF correlation, KRB-1 by name, has the form of

$$q''_{CHF} = a_1L + a_2(P/1000) + a_3\tanh[a_4(d_g + g_{sp}) + a_5] + a_6D_{he} + a_7D_{hy} - [a_8(P/1000) + a_9L(P/1000) + a_{10}L + a_{11}(P/1000)^2]G\chi + a_{12}G + a_{13},$$

where q''_{CHF} = critical heat flux in MBtu/hr/sq-ft
L = heated length in ft
P = pressure in psia
d_g = distance from the last upstream spacer grid to the CHF location in inch
g_{sp} = grid spacing in inch
D_{he} = heated equivalent diameter in inch
D_{hy} = hydraulic diameter in inch
G = mass velocity in Mlbm/hr/sq-ft
x = thermodynamic quality

The procedure of the KRB-1 CHF correlation development is shown in Fig. 1.

The rod bundle CHF data base was compiled from the Westinghouse and SIEMENS 5x5 CHF test bundles[14] including:

- bundle configuration : 5×5 square array,
- channel type : typical and thimble channels,
- axial heat flux profile : uniform and cosine shape,
- grid spacing : 272 to 660mm,
- heated length : 2. 44 to 4.27m,
- rod diameter/pitch : 9.5/12.6 to 10.8/14.3mm, and
- mixing vane : none and 5 different types.

The geometry of each CHF test bundle is presented in Table 1.

The local thermal hydraulic conditions at CHF locations were analyzed by the TORC code. The KRB-1 CHF correlation coefficients were optimized by the

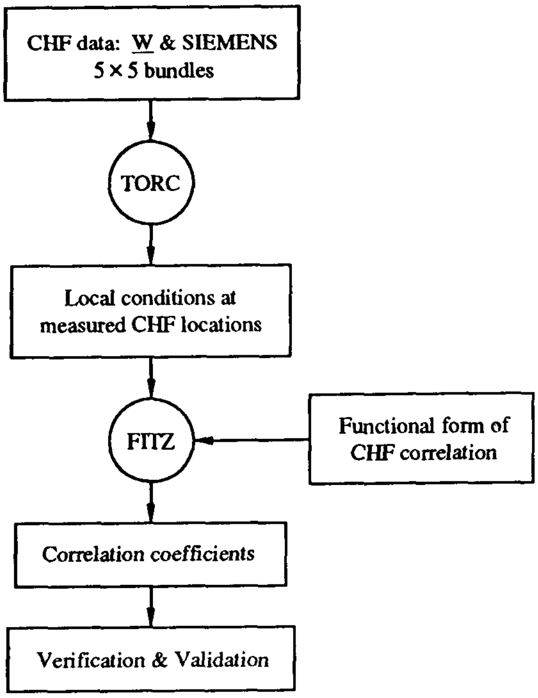

Fig. 1. Procedure of the KRB-1 CHF Correlation Development

FITZ code[15] which uses the Marquardt method in the least squares fitting of a nonlinear function. The KRB-1 CHF correlation coefficients and its applicable ranges are given in Table 2.

The CHF prediction capability of the KRB-1 correlation can be represented by the mean of 0.999 and the standard deviation of 10.7% for 1750 measured-to-predicted CHF ratios(M/Ps). The validity of the KRB-1 correlation was examined by analyzing the parametric trends of M/P for various thermal hydraulic parameters such as pressure, mass velocity, thermodynamic quality, heated length, and hydraulic diameter. As shown in Fig. 2, the KRB-1 correlation is well correlated for these thermal hydraulic parameters.

The analysis of variance, F-test, was used to ident-

Table 1. CHF Test Bundle Geometry Data

TS number	Configu-ration	Axial profile	Grid spacing [mm]	Heated length [m]	Rod dia./ ptich [mm]	Mixing vane type
1	THM*-5×5	Uniform	534	3.0	10.8/14.3	None
2	TYP*-5×5	Uniform	534	3.0	10.8/14.3	None
3	TYP-5×5	Uniform	545	3.0	9.5/12.7	None
4	THM-5×5	Uniform	534	3.0	10.8/14.3	P
5	THM-5×5	Uniform	534	3.0	10.8/14.3	None
6	THM-5×5	Uniform	534	3.0	10.8/14.3	None
7	TYP-5×5	Uniform	545	3.0	9.5/12.7	None
8	TYP-5×5	Uniform	545	3.0	9.5/12.7	S
9	TYP-5×5	Uniform	545	3.0	9.5/12.7	None
10	TYP-6×6	Uniform	545	3.0	9.5/12.7	None
11	TYP-5×5	Uniform	272	3.0	9.5/12.7	S
12	TYP-5×5	Uniform	545	3.0	9.5/12.7	S
13	TYP-5×5	Uniform	545	3.0	9.5/12.7	F1
14	TYP-5×5	Uniform	545	3.0	9.5/12.7	Swirl
15	TYP-5×5	Uniform	534	3.0	10.8/14.3	F1
16	TYP-5×5	Uniform	545	3.0	9.5/12.7	Swirl
17	TYP-5×5	Uniform	545	3.0	9.5/12.7	Swirl
18	TYP-5×5	Uniform	545	3.0	9.5/12.7	F1
19	THM-5×5	Uniform	545	3.0	9.5/12.7	Swirl
20	THM-5×5	Uniform	545	3.0	9.5/12.7	Swirl
21	TYP-5×5	Uniform	545	3.0	9.5/12.7	Swirl
22	TYP-5×5	Uniform	545	3.0	9.5/12.7	Swirl
23	THM-5×5	Uniform	545	3.0	9.5/12.7	Swirl
24	THM-5×5	Uniform	545	3.0	9.5/12.7	Swirl
25	TYP-5×5	Uniform	660	4.27	9.5/12.6	R
26	TYP-5×5	Uniform	660	2.44	9.5/12.6	R
27	THM-5×5	Uniform	660	2.44	9.5/12.6	R
28	TYP-5×5	Uniform	559	2.44	9.5/12.6	R
29	TYP-5×5	Uniform	559	4.27	9.5/12.6	R
30	THM-5×5	Cosine	559	4.27	9.5/12.6	R
31	TYP-5×5	Cosine	559	4.27	9.5/12.6	R

*) THM : with GT(guide tube), TYP : without GT

Table 2. KRB-1 CHF Correlation

$$q''_{CHF} = a_1 L + a_2 (P/1000) + a_3 \tanh[a_4 (d_g + g_{sp}) + a_5] + a_6 D_{he} + a_7 D_{hy} - [a_8 (P/1000) + a_9 L(P/1000) + a_{10} L + a_{11} (P/1000)^2] G\chi + a_{12} G + a_{13}$$

Correlation coefficients

$a_1 = -0.019278$	$a_2 = -0.17253$	$a_3 = 0.1396$
$a_4 = 0.05461$	$a_5 = -1.97$	$a_6 = -0.38082$
$a_7 = -0.054003$	$a_8 = 1.89870$	$a_9 = 0.047388$
$a_{10} = -0.10821$	$a_{11} = -0.67613$	$a_{12} = 0.134698$
$a_{13} = 1.25103$		

Applicable ranges

$99.3 < P < 171.7$ bar	$2.1 < L < 4.3$m
$1220.6 < G < 5018.1$kg/m^2-s	$0.28 < g_{sp} < 0.66$m
$-0.2 < \chi < 0.3$	$9.4 < D_{hy} < 13.2$mm

Table 3. Results of F-Test for CHF Test Bundles

Description	Calculated F value	Upper limit of F at 99% confidence	Hypothesis
All data	3.18	2.18	not satisfied
R-grid mixing vane	17.55	19.32	satisfied
Swirl-type mixing vane	2.53	5.95	satisfied
No mixing vane	3.03	7.08	satisfied
17×17 geometry	2.84	2.98	satisfied
14×14 geometry	3.45	19.32	satisfied
Without GT, Uniform	2.85	2.87	satisfied
With GT	3.52	5.19	satisfied
3m heated length	2.72	2.48	not satisfied
4.27m heated length	20.10	41.83	satisfied

ify systematic variations among the test series at a 95% confidence level. The F value is calculated by

$$F = S_1^2/S_2^2,$$

where S_1^2 = variance of the data within the test series, and

S_2^2 = variance of the means of the test series.

The result of F test for various data groups are presented in Table 3. A value of calculated F above the upper limit of F indicates that there is too much variation among the means. As indicated in Table 3 most of the F-tests except two result in values below the upper limits. Especially the CHF data for the KOFA fuel geometry(no mixing vane), which is the basis of the correlation limit DNBR calculation, are come from the same population.

The normalities of the residual and M/P distributions were also examined by the D'-test[16] and it has revealed that the populations have normal distributions with a 1% significance level. So the Owen's one-side-

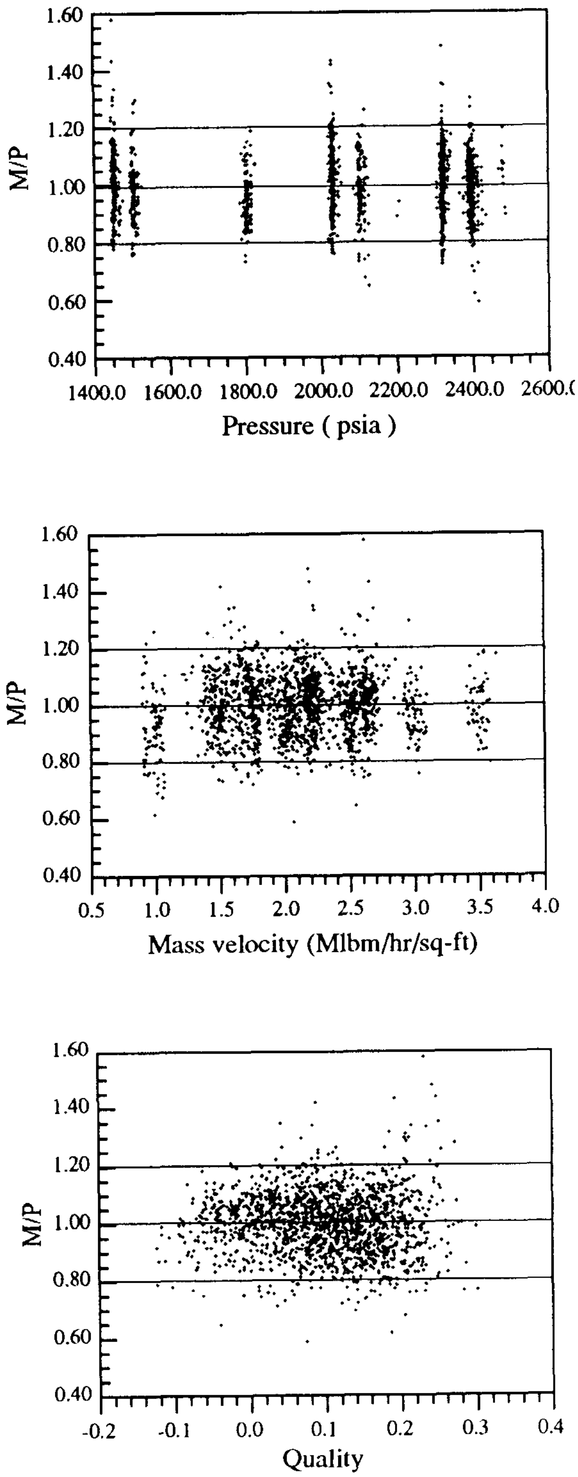

Fig. 2. Parametric Trend of M/P Calculated by KRB-1 Correlation

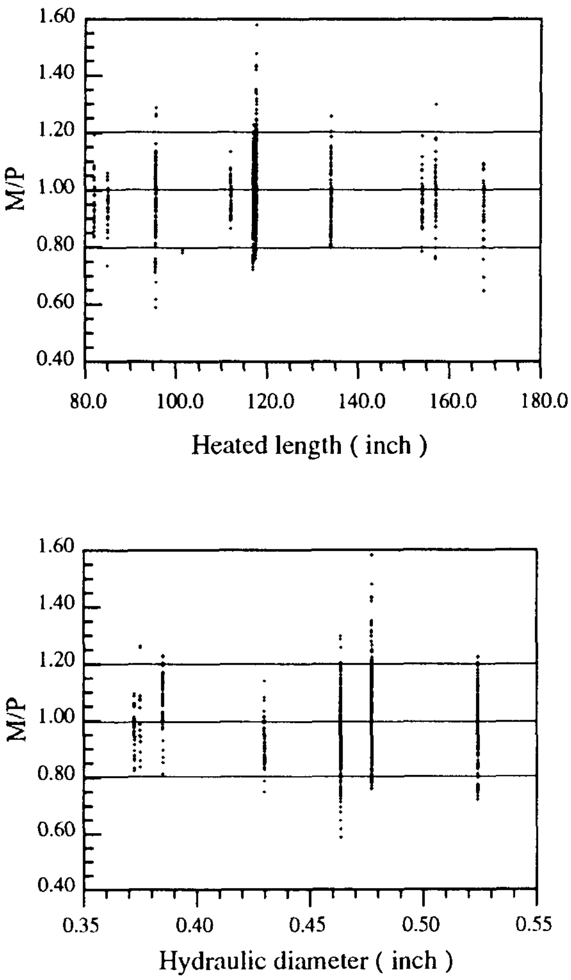

Fig. 2. Parametric Trend of M/P Calculated by KRB-1 Correlation(Continued)

d tolerance limit factor[17] with a 95% probability at a 95% confidence level could be applied to determine the CHF correlation limit DNBR. As shown in Fig. 3, the number of CHF data below the tolerance limit line is less than 5% of the total number of data points.

The KRB-1 CHF correlation limit DNBR was calculated as 1.23 which is slightly higher than 1.22 of the PUMA/ERB-2 system.

The next step is to determine the design limit DNBR on the basis of the statistical thermal design

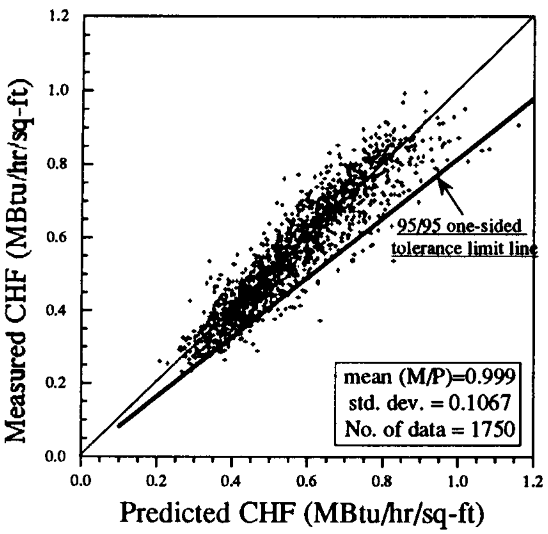

Fig. 3. Comparison of CHF Predicted by the KRB-1 Correlation with the Measured Data

procedure[1]. The DNBR sensitivity factors for the KOFA-loaded Yonggwang-1(YGN-1) core were calculated using the TORC/KRB-1 system and the results are given in Table 4.

The design limit DNBR was obtained by

$$\text{Design limit DNBR} = (\text{CHF correlation limit DNBR})/(\mu_y - 1.645\sigma_y),$$

where the parameter y is defined as

$$y = \text{DNBR}/\text{DNBR}_{ref}.$$

The overall DNBR uncertainty, σ_y, calculated by the TORC/KRB-1 system is less than that calculated by the PUMA/ERB-2 system due to the differences in the DNBR sensitivity factors of design parameters. The additional DNBR margin of 8.3% was equally taken into account to determine the safety analysis limit DNBRs of both CHF correlation systems. Consequently, the safety analysis limit DNBR of the

Table 4. Determination of the Design Limit DNBR

Parameter	σ/μ	TORC/KRB-1		PUMA/ERB-2[19]	
		S_i (typical)	S_i (thimble)	S_i (typical)	S_i (thimble)
Core thermal power	0.0075	−3.00	−2.79	−3.04	−2.86
Core inlet temp.	0.004	−9.96	−9.25	−11.77	10.79
Primary flow rate	0.013	2.53	2.63	2.03	1.80
Effective flow fraction	0.00908	2.53	2.63	1.98	1.81
System pressure	0.00737	2.54	2.32	2.70	2.29
$F^N_{\Delta H}$	0.025	−2.05	−2.01	−3.07	−2.82
$F^E_{\Delta H}$	0.0173	−1.62	−1.20	−1.17	−1.05
F^E_Q	0.0182	−1.00	−1.00	−1.00	−1.02
T/H code	0.0206	1.0	1.0	1.0	1.0
σ_y/μ_y		0.09067	0.08689	0.10582	0.09747
CHF correlation limit DNBR		1.23	1.23	1.22	1.22
Design limit DNBR		1.45	1.44	1.48	1.45
Margin		8.3%	8.3%	8.3%	8.3%
Safety analysis limit DNBR		1.57	1.56	1.60	1.58

TORC/KRB-1 system is less than that of the PUMA/ERB-2 system by about 1%. However, it should be noted that the direct comparison of the two limit DNBRs does not always indicate the difference in the thermal margin since the DNBR sensitivity of one CHF correlation system to the core operating parameters may be different from that of the other system.

2.2. Hot Subchannel DNBR Analysis Model

The hot subchannel is a hypothetical subchannel where the worst thermal hydraulic conditions in the core take place and as a consequence the minimum DNBR over the whole core occurs. In the closed channel approach such as the PUMA/ERB-2 system, the hot subchannel is characterized by power peaking factors and various engineering factors to consider the effects of the multichannel phenomena such as the inlet flow maldistribution and flow redistribution due to the lateral interactions between the neighboring channels. In the open channel approach, however, most of the engineering factors used in the closed channel approach are accommodated by the subchannel analysis. Thus it gives more realistic, i.e., less conservative results on the thermal hydraulic parameters in the hot subchannel than those from the closed channel analysis.

The ideal way to analyze a reactor core utilizing the open channel approach is to take each actual subchannel as a radial node in the analysis. This implies that, for a typical PWR core, over 30,000 radial nodes should be considered. This number is so large that there is no available computer which can handle this problem, and even if such a computer were available, the cost would be prohibitive. Therefore this approach was ruled out and two other general approaches had been developed: the multistage method[5, 6] and the one-stage method[18]. Traditionally, CE design adopts a multistage(three-stage) method for the hot subchannel analysis. In the first stage the whole core is analyzed on an assembly-to-assembly basis. From this analysis the hot assembly and its boundary conditions can be identified. In the second stage the hot assembly is divided into four regions, and in the subsequent stage, the hottest region of the four is analyzed on a subchannel basis. In this study we developed an one-stage calculation model using a prescribed nodal layout to save the computing time and efforts while maintaining the accuracy of the hot subchannel analysis results by the three-stage model. In the simplified one-stage method, the core was analyzed by only one stage using a fine mesh in a zone consisting of those subchannels with the larger peaking factors and a coarse mesh outside this zone. Through a systematic analysis for various layouts and operating conditions, the one-stage 1/8-core analysis model was developed as shown in Fig. 4 which consists of sixteen channels including the hot subchannel. The accuracy of the

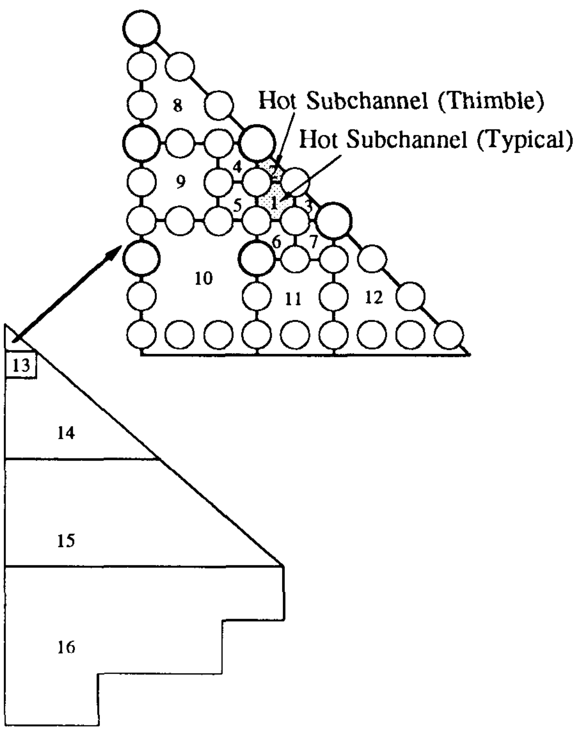

Fig. 4. One-Stage Hot Subchannel Analysis Model

one-stage analysis model was compared with that of the three-stage model at various operating conditions and it was found that the one-stage model gives slightly conservative DNBR values with the maximum deviation of 1.2% while it takes the computing time about one-tenth of that for the three-stage model.

3. Evaluation of the Thermal Margin

The thermal margin of the 17×17 KOFA-loaded YGN-1 core was evaluated by using the TORC/KRB-1 system and compared with that evaluated by the PUMA/ERB-2 system. The following three conditions were considered for this comparative study:

1) Overpower margin at normal operating condition,
2) Allowable core operating region, and
3) Thermal margin at DNB limiting accident condition.

The core thermal hydraulic design parameters required for the thermal margin evaluation are listed in Table 5[20].

The closed channel code, PUMA, considers the effects of the diversion crossflow and turbulent mixing between neighboring channels by a single engineering factor, θ_A. On the other hand, the TORC code considers these effects separately by the lateral momentum equation and the turbulent mixing parameter. The effect of the inlet flow maldistribution is inherently considered in the TORC code by reducing the hot assembly inlet flow rate in input values while it is modeled by an engineering factor in the PUMA code. The radial power distribution used in the TORC analysis is flatter, i.e. more conservative, than that used in the PUMA analysis which is considered by the engineering factor, f_{iso}.

The overpower margin at normal operating condition is calculated by increasing the core thermal power until the hot subchannel minimum DNBR reaches the safety analysis limit DNBR. The maximum power ratios at this condition were estimated as 118% and 126% for the PUMA/ERB-2 and the TORC/KRB-1 systems, respectively. The allowable core operating region is bounded by the hot leg boiling limit line and the DNB limit line. This is the basis of the design of the thermal core protection system. The role of the DNB limit line is to prevent the core from DNB at the reactivity-induced accident condi-

Table 5. Comparison of the Core TH Parameters between the PUMA/ERB-2 and the TORC/KRB-1 Systems

Parameter	PUMA/ERB-2	TORC/KRB-1
Pressure	157.2 bar	157.2 bar
Flow rate	$18.304 m^3/s$	$3365.6 kg/m^2/\text{-}s$
T_{in}	291.5°C	291.5°C
Power/Heat flux	2775 MWth	$614.8 kW/m^2$
Core bypass flow fraction	0.046	0.046
$F^N_{\Delta H}$	1.49	1.49
Turbulent mixing	$\theta_A = 0.7$	TDC = 0.005
Inlet flow maldist'n	$F^E_{\Delta H} = 1.03$	5% inlet flow reduction for hot assembly
Radial power dist'n	$f_{iso} = 1.05$	$F_{peak}/F_{ass'y} = 1.046$
Axial power dist'n	1.55 chopped cosine	1.55 chopped cosine
Fuel densification factor	1.002	1.002
Cross flow factors	N/A	$K_{ij} = 0.5$ $s/l = 0.5$ $f_T = 0.0$

tions. Thus the thermal margin variations for all of the possible operating conditions can be estimated by comparing the allowable core operating regions obtained from the two different CHF correlation systems. The thermal margin for the four different pressure levels are shown in Fig. 5. The TORC/KRB-1 system has enlarged the allowable core operating region by more than 8% compared with that obtained from the PUMA/ERB-2 system[21].

The complete loss of flow accident(LOFA) is one of the most limiting accidents in the aspect of DNB. This accident is initiated by an inadvertent trip of all reactor coolant pumps, and it is followed by the occurrence of minimum DNBR at about 3 seconds after the transient[22]. As a result of the accident analysis, the minimum DNBR calculated by the TORC/KRB-1 system was about 10% (about 5% in thermal power) higher than that calculated by the PUMA/ERB-2 system.

4. Results and Discussion

Through the comparative study of the thermal margins for various core conditions, it was found that the multichannel analysis model(TORC/KRB-1) yields more than 5% of thermal power margin compared to the single-channel analysis model (PUMA/ERB-2). We investigated the rationale of the thermal margin improvement in this section.

4.1. Characteristics of the CHF Correlation

The CHF correlations used in this study were developed on the basis of the local conditions at CHF locations calculated by the corresponding thermal hydraulic analysis codes. Comparing the local conditions at CHF locations in test bundles, the local mass velocities calculated by the PUMA code turned out to be larger than those calculated by the TORC code by about 7% in the mean value as shown in Fig. 6, while the mean values of the local qualities were nearly the same.

This may be attributed to the wall effect of the test section on the subchannel velocity distribution. A fairly large portion of the total pressure loss in the test

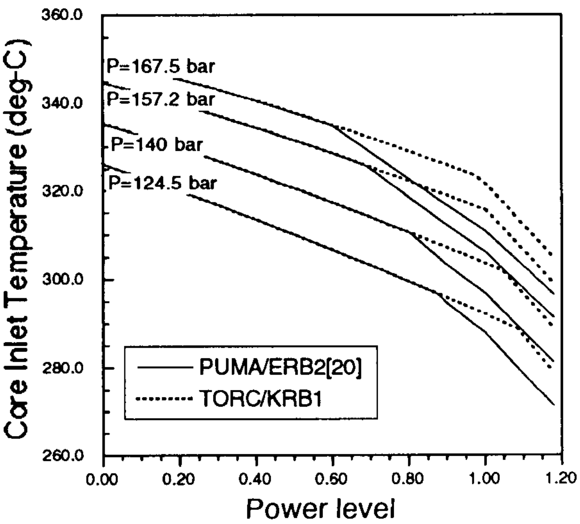

Fig. 5. Comparison of the Allowable Core Operating Regions

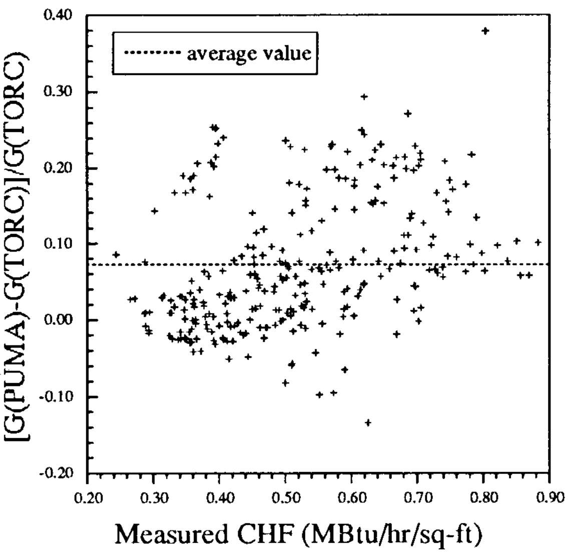

Fig. 6. Comparison of the Local Conditions at Measured CHF Locations Predicted by the PUMA and TORC Codes.

bundle is due to the frictional pressure loss at the test section wall. Thus the mass velocities at CHF locations calculated by the PUMA code are overestimated since the PUMA code cannot consider the effect of diversion crossflow which tends to make the radial flow distribution more uniform.

In general, the CHF increases with the mass velocity and decreases with the local thermodynamic quality as shown in Fig. 7.

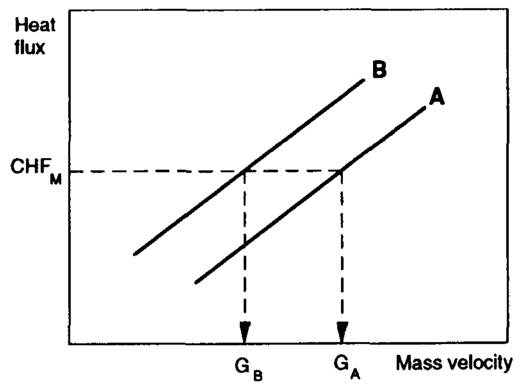

Fig. 7. Characteristics of a CHF Correlation accoding to the Predicted Local Mass Velocity

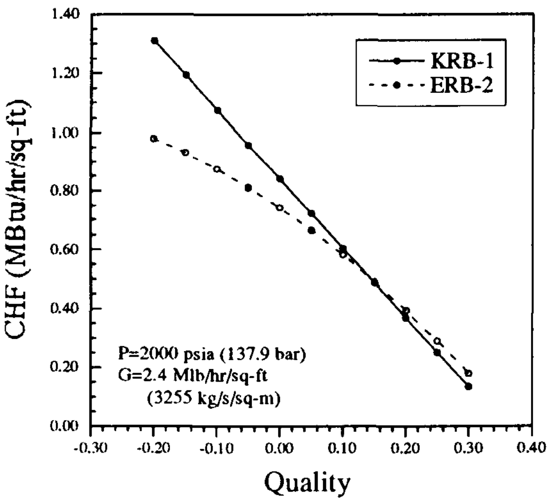

Fig. 8. Parametric Trend of the KRB-1 and ERB-2 CHF Correlations

If the local mass velocity calculated by a code A(G_A) is larger than that calculated by another code B(G_B) at the same CHF condition, then the resulting CHF correlation developed by the code A would underpredict CHF values compared with the CHF correlation developed by the code B for the same local flow conditions. This is the case of the ERB-2 and KRB-1 CHF correlations. The parametric behavior of these two correlations shows this trend obviously as in Fig. 8. In the low and intermediate quality regions, the CHF predicted by the ERB-2 correlation was lower than those by the KRB-1 correlation since the local mass velocity by the PUMA code was higher than that by the TORC code at these conditions. The trend was slightly reversed in the high quality region since the dependency of the CHF on the mass velocity in this region was also reversed(i.e., CHF decreases with increasing mass velocity). In addition, the difference of local mass velocities predicted by PUMA and TORC codes are not remarkable in the high quality region.

4.2. Predictability of the Local Conditions in the Hot Subchannel

The local conditions at the minimum DNBR location of the YGN-1 core were analyzed for various power levels. The local quality calculated by the PUMA code increases with power faster than that calculated by the TORC code as shown in Fig. 9. The local boiling in the hot subchannel causes higher pressure drop in the subchannel. This leads to the lateral pressure difference between neighboring channels and the resulting diversion crossflow from the hot subchannel to the neighboring channels may reduce the mass velocity in the hot subchannel. The closed channel code PUMA cannot reflect the flow redistribution effect which tends to homogenize the radial distribution of axial flow, so the resulting mass velocity in the hot subchannel is less than that calculated by a multichannel code. This trend becomes more pronounced as the local quality increases.

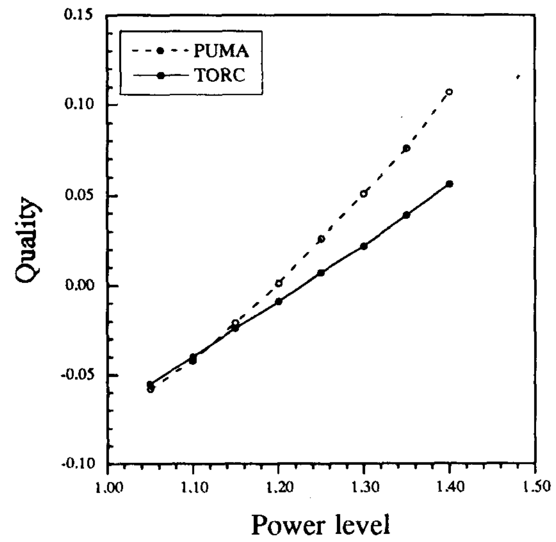

Fig. 9. Comparison of the Hot Subchannel Conditions Predicted by the PUMA and TORC Codes

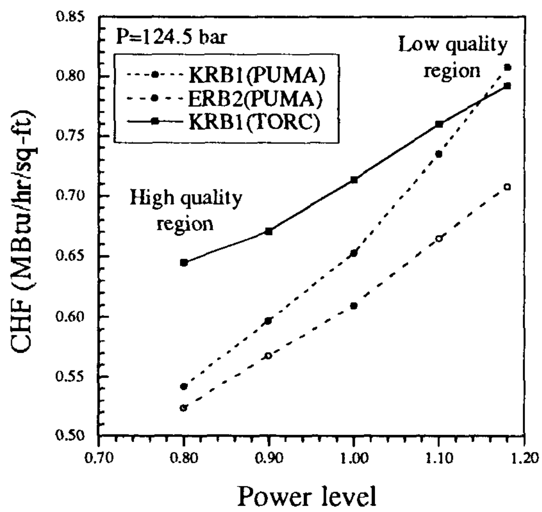

Fig. 10. Illustration of the Components of the Thermal Margin Improvement

4.3. Rationale of the Thermal Margin Increase

According to the analysis results mentioned above, we considered the following two effects which contribute to the improvement of the thermal margin employing the multichannel code system :

1) characteristics of the CHF correlation, and
2) local conditions predicted by the thermal hydraulic analysis code.

The contribution of these two effects may be changed depending on the operating conditions. Figure 10 is an example showing the portion of these two effects on the thermal margin improvement. The CHF values depicted in this figure were obtained at the conditions of DNB limit line described in Section 3 at the pressure of 124.5 bar. In this figure, the KRB1(PUMA) means the CHF values calculated by the KRB-1 correlation using the local conditions analyzed by the PUMA code, and so on. So the difference between the KRB1(PUMA) and the ERB2(PUMA) indicates the thermal margin change due to the effect of the characteristics of CHF correlation, while the difference between the KRB1(PUMA) and the KRB1(TORC) means the change in thermal margin due to the effect of the local conditions predicted by the two code systems.

At the low power level as shown in the figure, the inlet temperature should be high enough to reach the DNB limit condition, thus the local quality at the minimum DNBR location is relatively high. At this high quality region, the major part of the thermal margin improvement is ascribed to the effect of the local conditions since the hot subchannel conditions analyzed by the PUMA code is very conservative. As the local quality decreases, the portion of the effect of the local conditions decreases and shows so much as a reversed trend. In this low quality region the thermal margin improvement is mainly due to the effect of the CHF correlation characteristics. The change in the effect of the CHF correlation characteristics with respect to the local quality may be attributed to the fact that the closed channel analysis model cannot properly take into account the test section wall effects. In other words, the discrepancy of the predictability of the local conditions between the two code systems in the reactor core is inconsistent with that in the test bundle.

5. Conclusions

In this study a multichannel thermal hydraulic analysis methodology was developed for the evaluation of the thermal margin for the KOFA-loaded core. The important results are as follows :

1) A new CHF correlation, KRB-1, was developed and the correlation limit DNBR was calculated as 1.23.
2) A one-stage core analysis model with a prescribed nodal layout was developed for the hot subchannel analysis.

Through the thermal margin analysis of a reference 17x17 KOFA-loaded core, we have obtained the following results :

1) More than 5% of the thermal margin can be recovered by replacing the presently used PUMA/ERB-2 system with the TORC/KRB-1 system.
2) The thermal margin improvement is attributed to the effects of the local conditions predicted by the thermal hydraulic analysis code, and to the characteristics of the CHF correlation.

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