The 14th International Topical Meeting on Nuclear Reactor Thermalhydraulics, NURETH-14
Toronto, Ontario, Canada, September 25-30, 2011

NURETH14-558

DEVELOPEMENT OF CHF CORRELATION "MG-NV" FOR LOW PRESSURE AND LOW VELOCITY CONDITIONS APPLIED TO PWR SAFETY ANALISIS

T. Yumura, T. Yodo, Y. Makino and T. Suemura
Mitsubishi Heavy Industries, LTD., Kobe, Hyogo, Japan

Abstract

The Critical Heat Flux (CHF) is one of the important parameters in the safety analysis of Pressurized Water Reactor (PWR). If the CHF is reached, an abrupt drop occurs in the heat transfer between the fuel rod cladding and the reactor coolant, which may induce a large temperature excursion of fuel cladding and a subsequent fuel failure. Therefore, accurate prediction of CHF is required in order to assure a sufficient safety margin in the PWR core.

Mitsubishi Heavy Industries, ltd (MHI) is developing a new series of CHF correlations which covers various fuel designs and wide range of fluid conditions with sufficient reliability. In this paper, a new CHF correlation, MG-NV (Mitsubishi Generalized correlation for Non-Vane grid spacers) is presented. This correlation is one of the basic components of the new correlation series and was developed to cover low pressure and low velocity conditions where the rod bundle CHF data are limited. The CHF correlation was developed based on open CHF database and provides conservative but more reliable rod bundle CHF predictions compared with the conventional CHF correlations used in safety analyses at low pressure condition, such as Main Steam Line Break event.

Introduction

For the reactor safety of Pressurized Water Reactor (PWR), Departure from Nucleate Boiling (DNB) must be prevented during the normal operation and Anticipated Operational Occurrences (AOOs). This important design basis is mainly realized by protection systems such as Over Temperature ΔT trip, which prevents the reactor core from DNB by maintaining the plant operating condition described by the combination of reactor power, primary loop pressure and temperature within a predetermined allowable area. In addition, other protection systems safely maintain the key parameters such as primary pressure, flow rate and local heat flux.

To determine and to verify the protection systems via safety analysis, the design correlation is utilized to predict Critical Heat Flux (CHF). Therefore, the design CHF correlation generally covers the allowable range of operating conditions and reflects the fuel performance based on many DNB tests, which are carried out using the test rod bundles that represent the actual fuel design including grid spacers.

On the other hand, the several anticipated events initiated at the sub-critical condition go through the condition outside of above mentioned operating conditions, and therefore, are not covered by the design CHF correlation. In the Main Steam Line Break (MSLB) event, pressure decreases due to excess cooling at a steam generator with failed steam line, and it results in lower pressure condition than the applicable range of the design CHF correlation. In addition, the primary coolant flow also decreases significantly, if the power supply is lost.

Historically, the W-3 correlation [1] has been widely used for this area. This correlation was developed from the single tube CHF data of wide pressure range and has been verified by many researchers and PWR vendors. Although this correlation does not take account for the effect of fuel rod bundle design precisely, it can conservatively predict CHF in various fuel rod bundles which have grid spacers with mixing-vanes.

While W-3 has been conservatively applied to the safety analysis where the design CHF correlation is not applicable, the design margin is unclear due to the large DNBR uncertainty comes from the scattering in its database. In addition, the narrow applicable range for thermal equilibrium quality disturbs its application to the low flow rate conditions where CHF likely occurs at high quality conditions.

MHI is developing a new series of CHF correlations, which are applicable to the present and next generation fuels. One of those correlations, MG-NV, has been developed to cover such low pressure and low flow conditions, resolving the above problems of W-3.

1. Overview of existing data and correlations

1.1 CHF data

Prior to developing a new correlation, the available open CHF databases were surveyed. Generally, CHF depends on the geometrical configuration of flow channel. Therefore, the many rod bundle CHF tests have been conducted by fuel vendors, reflecting their various fuel designs. However, the data acquired at low pressure conditions are very limited, because the CHF tests generally focus on the design operating conditions as described above.

The most popular database of rod bundle CHF was published by EPRI [2], which includes the many fuel vendors' test data acquired at Heat Transfer Research Facility of Columbia University (HTRF-CU). Several low pressure data for the Combustion Engineering's (CE's) and Westinghouse's fuel designs are included in it. Although CE fuel rod bundle design is different from that of MHI fuels, some of those test sections use vane-less grid spacers, therefore, the result is expected to be conservatively applicable to MHI fuel designs which features mixing-vane grid spacers. The available data in the pressure range of interest are summarized in Table 1.

In order to compensate the insufficiency of the rod bundle data, our attention was turned to the abundant single tube data in the world. While the old single tube data tend to show large scattering, there is a substantial number of data at the desired low pressure and/or low flow conditions due to the flexibility of the test conditions. It is expected that those data can illustrate the fundamental behaviour of CHF for developing a correlation.

The latest world wide CHF database for uniformly heated single tube has been issued by Purdue University-Boiling and Two-Phase Flow Laboratory (PU-BTPFL) [3]. The single tube data which cover following ranges were selected from this database for development of the new correlation:

Pressure (P):	2 through 12.5 MPa
Mass velocity (G):	290 through 5,000 kg/m^{2}s
Thermal equilibrium quality (x):	-0.38 through 0.84
Homogeneous void fraction (α):	0 through 0.98
Tube diameter (D):	3.8 through 30 mm
Heated length (L):	5 through 500 D

Finally, it was determined that the correlation was originally developed by analysing single tube CHF data, and then was corrected and validated by comparison with the limited but conservative rod bundle CHF data.

1.2 CHF correlations

There are many correlations applicable to low pressure and/or low flow conditions. Several important correlations are reviewed here for the reference of developing new correlation.

Tong developed several CHF correlations applicable to the PWR analysis [4]. The W-2 correlation set consists of two types of correlations, namely, a "CHF correlation" for subcooled condition and a "critical enthalpy correlation" for the saturated condition. This approach is reasonable to cover wide range of thermal equilibrium quality. However, Tong subsequently developed the W-3 correlation for more precise prediction at the mid-quality region where CHF is interpolated between two correlations of W-2, because most of the conditions of concern in PWR analyses belong to such mid-quality region.

This history limited the applicable quality range of W-3 to narrow region. Figure 2 shows the applicability of W-3 to the single tube data described in 1.1, which includes the data outside of the applicable range of the correlation. The left side figure shows that the Measured to Predicted ratios (M/Ps) of CHF by W-3 are significantly reduced after the thermal equilibrium quality exceeds 15%, which is the upper limit of the applicable range of W-3. The right side figure implies that this narrow range of quality disturbs the application to low flow conditions. (This does not mean that W-3 is not at all applicable to the low flow condition. The under predictions are due to high quality conditions.)

Macbeth's correlation [5] and EPRI-1 [6] are widely used CHF correlations applicable to low pressure and low flow condition. The wide range applicability of those correlations comes from the basic configuration of the equation, which is based on the system parameter concept in contrast of the local parameter concept adopted in W-3. The correlation of system parameter concept predicts CHF from the inlet flow conditions instead of the local flow conditions. It means that the correlation internally assumes the heat balance within the flow channel. In this concept, the critical condition is described by the channel power or channel exit quality, which are convertible to each other. It is beneficial to predict the CHF at high equilibrium quality condition, where the boiling transition is dominated by the flow conditions rather than the local heat flux. However, this type of correlation cannot be applied generally to the open channel configuration of PWR core, where the channel configuration and heat balance are different from that of the CHF test.

The widest range prediction based on local parameter concept is achieved by the look up table (LUT-2006) developed by Groeneveld et al [7]. This table is based on the world wide single tube CHF data, and provides good prediction throughout the wide parameter range, because its prediction does not restricted by mathematical form like the CHF correlations. While the look up table is hard to be dealt with in a design work due to the difficulty of maintenance and software verification, it provides the useful reference for developing new correlation. In order to develop the new correlation based on local parameter concept, its mathematical form was carefully selected so as to reflect the behaviour in the wide quality range described by LUT-2006.

Table 1 Test Specification of Rod Bundle CHF data

Test Section No.	Channel Layout	Heated Length (m)	Axial Power Shape	Type of Grid Spacers*1	Grid Spacing (m)	No. of Data	Pressure (MPa)	Inlet Mass Velocity (kg/m^{2}s)	Inlet Temperature (°C)	Remarks
9	Fig.1(b)	2.13	uniform	NV	0.41	53	1.5 - 12.5	390 - 4030	166 - 309	
13	Fig.1(a)	2.13	uniform	NV	0.41	51	2.8 - 12.5	390 - 4090	174 - 318	
18	Fig.1(b)	2.13	uniform	NV	0.41	32	6.1 - 12.5	1320 - 4140	223 - 314	
19	Fig.1(b)	2.13	uniform	NV	0.48	28	6.2 - 12.5	1330 - 4080	212 - 317	
21	Fig.1(a)	2.13	uniform	NV	0.41	36	6.2 - 12.5	1300 - 4090	228 - 317	
33	Fig.1(c)	2.13	uniform	NV	0.46	35	6.2 - 12.6	1290 - 4130	228 - 314	
36.1	Fig.1(c)	2.13	uniform	NV	0.46	49	6.2 - 12.5	1340 - 4110	208 - 316	
37	Fig.1(c)	2.13	uniform	NV	0.46	39	6.2 - 12.5	1330 - 4140	208 - 318	
38	Fig.1(c)	3.81	uniform	NV	0.44	16	4.1 - 12.5	680 - 1410	175 - 308	
43	Fig.1(d)	2.13	uniform	NV	0.36	37	9.6 - 12.5	1340 - 5480	181- 315	
47	Fig.1(d)	3.81	uniform	NV	0.36	22	6.1 - 12.5	670 - 1400	183 - 288	
162	Fig.1(e)	4.27	cosine	MV	0.56	56	5.1 - 12.5	700 - 4790	172 - 306	*2

*1 NV: Non-Vane grid spacers, MV: Mixing-Vane grid spacers
*2 Data used only for validation of correlation applicability to MHI type fuel design.

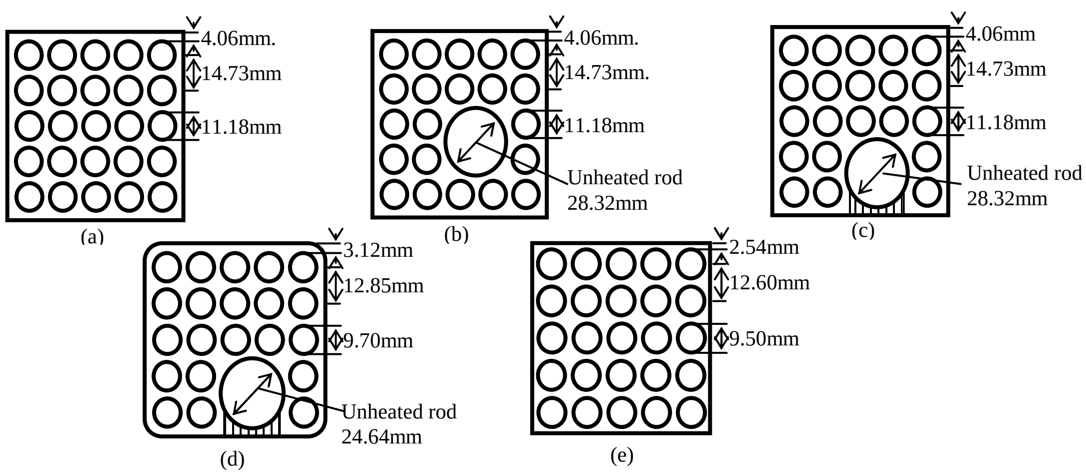

Figure 1 Channel Geometry of Rod Bundle CHF Data

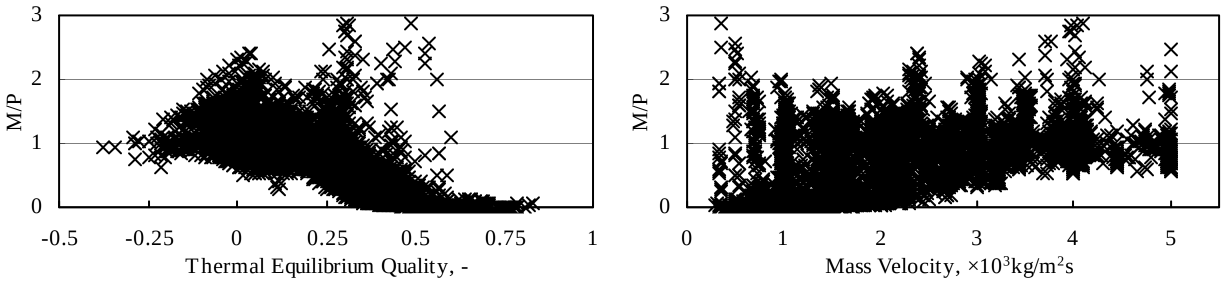

Figure 2 Applicability of W-3 to wide range data

2. Development of correlation

2.1 Formulation of the new correlation

The basic concept of the new correlation is to predict CHF for wide range of thermal equilibrium quality which covers low pressure and low flow conditions expected in PWR MSLB event. The correlation was developed based on available published data from single tube tests and rod bundle tests with non-vane grid spacers. As the result, it can be conservatively applied to any fuel bundles with mixing-vane grid spacers. The correlation is named "MG-NV", which stands for Mitsubishi Generalized correlation for Non-Vane grid.

Thermal equilibrium qualities at CHF conditions are widely distributed if the low flow data are included. The relationship between thermal equilibrium quality and mass velocity shown in Figure 3 denotes that the specifically high quality conditions are to be covered for the CHF prediction at low flow rate. In such a wide quality range, CHF is no longer a linear function of quality, as illustrated in Figure 4. In order to correlate the wide quality range data, the same formula as MHI's new standard CHF correlation, MG-S [8], is adopted for MG-NV.

$$CHF = \frac{A}{1 + exp\left[K(x - C)\right]} \quad (1)$$

This formula provides non-linear relationship between CHF and thermal equilibrium quality, x. The coefficients, A, K and C, are determined as the functions of local fluid conditions and flow channel configurations.

Figure 4 also indicates that the fast decrease of CHF that occurs in high quality region where the CHF is around 1 MW/m^2. This quality region is corresponding to the so called "Limiting Quality Region (LQR)", and the CHF mechanism is supposed to change from the break of thick liquid film to the dry out of thin liquid film [7]. Although this discontinuous change affects the CHF prediction at high quality region, it is not very important in the rod bundle CHF prediction in 2.3.

2.2 Correlation based on single tube CHF data

The coefficients in Equation (1) were determined so as to provide a similar behaviour with LUT-2006 and to fit the single tube data from PU-BTHFL CHF database. Unfortunately, large scattering is observed in the database, because it consists of worldwide data acquired by many independent researchers. The data fitting was done so as to correlate the "average behaviour" of the data and not to remain dependency on any parameters.

The preliminary examination for PU-BTPFL data using LUT-2006 was conducted as shown in Figure 5. The figure indicates that there are large scattering in high void fraction region and short tube length data. At high void fraction, the boiling transition occurs as the result of break or the dry-out of liquid film, in which the critical condition is strongly dominated by heat balance through the flow channel and is mostly independent of the local heat flux. It is supposed that this mechanism causes the large scattering in M/P ratio of local critical heat flux. Therefore, the basic component of the new correlation was first developed based on the data of which exit void fraction is less than 0.8.

It is generally known that CHF tends to be higher in short length tubes [9]. It was modelled as the entrance effect. Finally, single tube CHF is predicted by combination of the basic component, MG-NV0 and entrance effect f_{Inlet}. MG-NV0 was determined as a function of pressure (P), mass velocity (G), thermal equilibrium quality (x) and tube diameter (D). The f_{Inlet} was described by relative distance from inlet (L/D).

$$CHF_{SingleTube} = f_{Inlet}(L/D) \cdot MGNV0(P,G,x,D) \tag{2}$$

$$MGNV0(P,G,x,D) = \frac{A(P,D)}{1 + exp\left[K(P,G)\{x - C(P)\}\right]}$$

The prediction by Equation (2) is compared with the PU-BTPFL database in the range of void fraction less than 0.8. As confirmed in Figure 6, there is no significant dependency to the key parameters. In addition, high void fraction data are included in the comparison as shown in Figure 7. As a result, it was confirmed that the correlation can well predict average behaviour of complete database including high void fraction data

2.3 Correlation based on rod bundle CHF data

The low pressure bundle CHF data are available for CE fuel design as described in 1.1. Although the typical test sections shown in Figure 1 are different from MHI's fuel assembly design, the key parameters such as hydraulic equivalent diameter and heated length are similar to those of MHI fuel design. In addition, those data are obtained by using rod bundles with non-vane grid spacer, which is supposed to be more conservative than actual fuels.

The local conditions required by MG-NV correlation are provided by the MIDAC [10][11], which is MHI's drift flux model based two phase subchannel analysis code. The constitutive models are selected so as to provide best estimate prediction as follows.

Subcooled void detach:	modified Saha-Zuber [12]
Subcooled void generation:	Lahey [13]
Drift flux correlation:	Homogeneous for greater than 12.5 MPa
	Chexal [14] for less than 10 MPa
Two-phase flow friction:	EPRI [15]

It is known that the void drift toward the central channels occurs at low pressure condition, when flow shroud exists [16]. Since the void drift leads to high steam quality in the hot channel, it is conservative to neglect this effect in the validation of the correlation.

The subchannel diameter, D, in Equation (2) is represented by the product of the heated equivalent diameter, D_H, and the radial peak-to-average of heat flux in the subchannel, Fp ($D=D_HFp$). It means the equivalent heated diameter based on the hot rod heat flux, and well predicted the rod bundle test results including thimble cell tests.

Figure 8 shows applicability of Equation (2) to the bundle data in Table 1 except test section 162. In this preliminary analysis, the data included high void fraction case greater than 0.98, which has been finally removed in Table 1 due to the relatively large scattering of M/P. The correlation provides less conservative prediction for low flow and high void fraction conditions. This low CHF in low pressure and low flow conditions is supposed to be due to absence of the void drift modelling in the subchannel analysis and might be hard to occur in the actual core where the flow shroud does not exist, however, the correction factor for this behaviour of the bundle data was introduced into the correlation conservatively.

$$CHF_{MG-NV} = f_{Bundle}(P,G,\alpha_{\mathrm{H}}) \cdot MGNV0(P,G,x,D) \tag{3}$$

The bundle factor, f_{Bundle}, was determined so as to compensate the under predictions shown in Figure 8 and correct the slight dependency on the pressure. With this factor, MG-NV provides fairly good prediction of CHF as shown in Figure 9. The correlation limit of DNBR based on 95/95 basis [17] is determined as 1.19 for the following parameter range:

Pressure (P):	1.5 through 12.6 MPa
Mass velocity (G):	370 through 5,000 kg/m^{2}s
Thermal equilibrium quality (x):	-0.16 through 0.71
Homogeneous void fraction (α):	0 through 0.98
Heated equivalent diameter (D_H):	12 through 20 mm

The high quality data up to 0.71 is well correlated in spite of no specific consideration of LQR discussed in 2.1. It is supposed that the discontinuity at LQR is unclear or not presented in the rod bundle. The relatively lower CHF after LQR, which is modelled in LUT-2006, seems to be achieved only by the combination of long heated length and low power condition in single tube tests. It might be hard to be represented in the rod bundle test configuration.

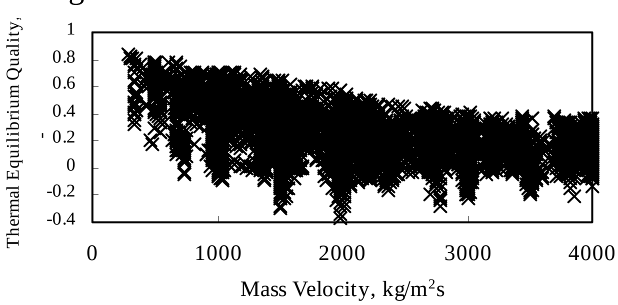

Figure 3 Relationship between quality and mass velocity from PU-BTPFL database

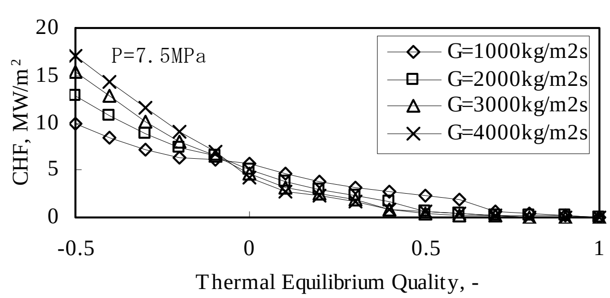

Figure 4 Typical CHF behaviour from LUT-2006

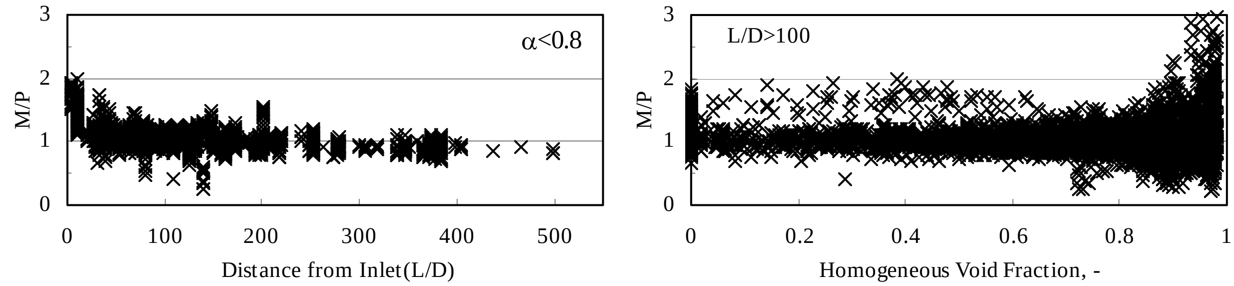

Figure 5 Comparison between PU-BHTFL data and LUT-2006

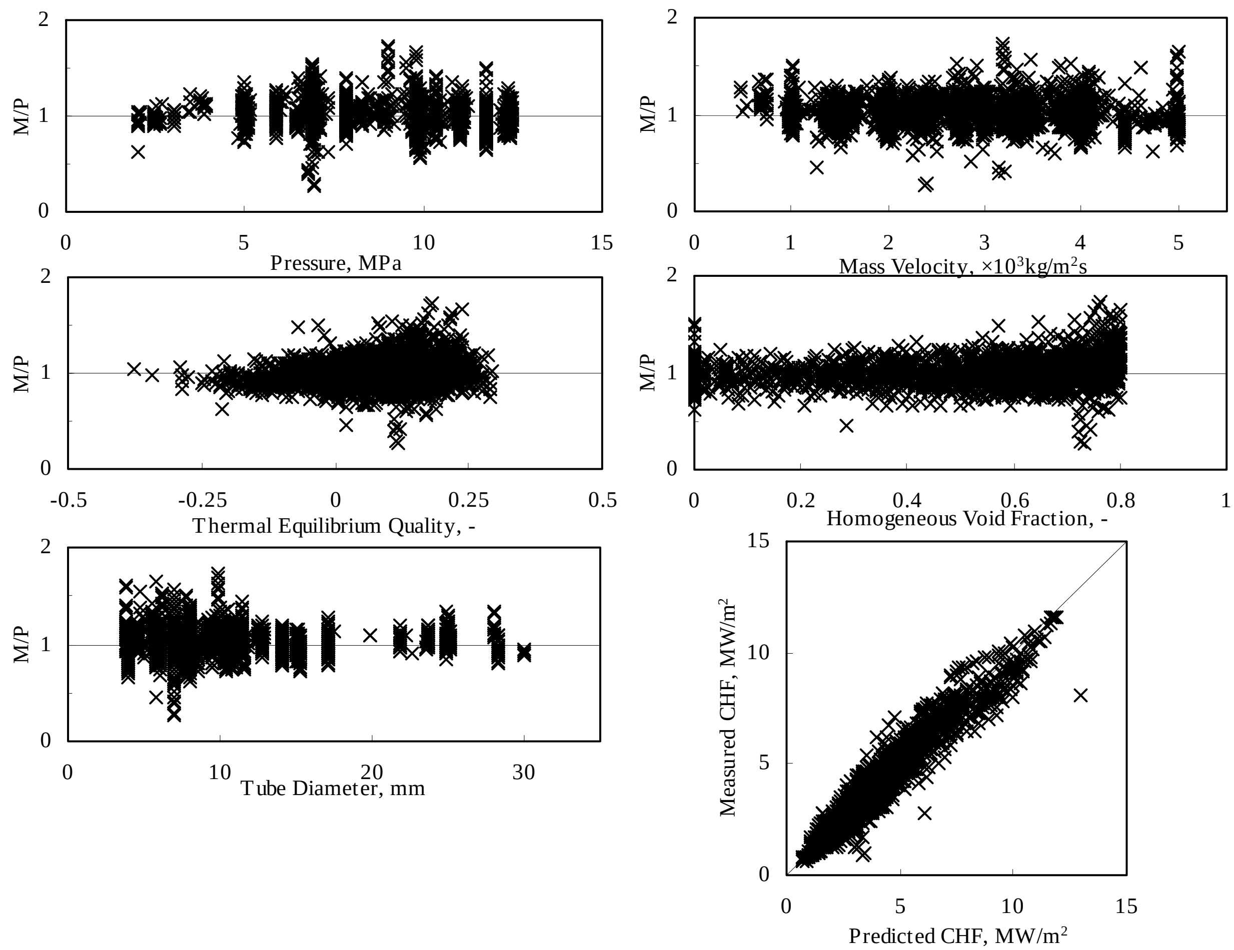

Figure 6 CHF prediction of MG-NV for the single tube data (PU-BTPFL), $\alpha < 0.8$

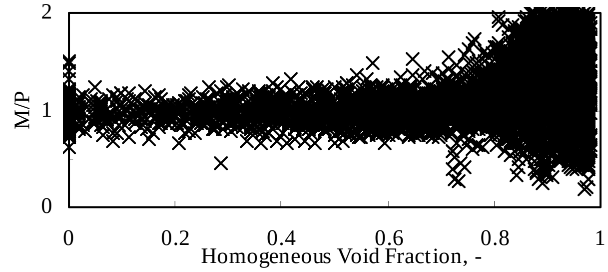

Figure 7 CHF prediction of MG-NV for the single tube data (PU-BTPFL), $\alpha < 0.98$

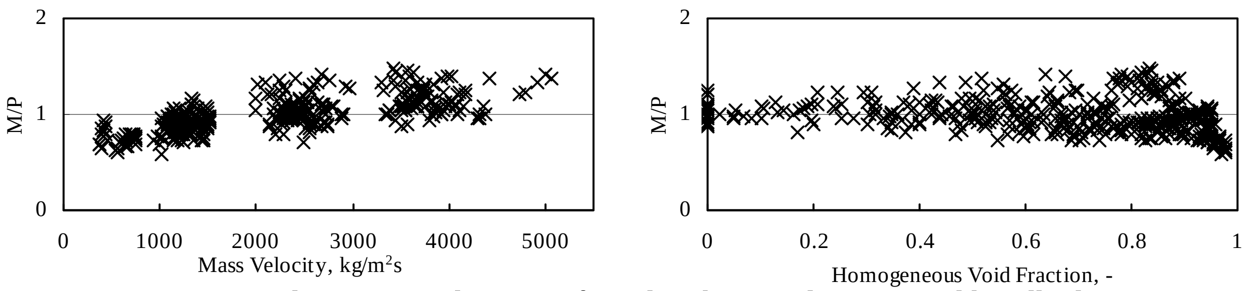

Figure 8 Preliminary application of single tube correlation to rod bundle data

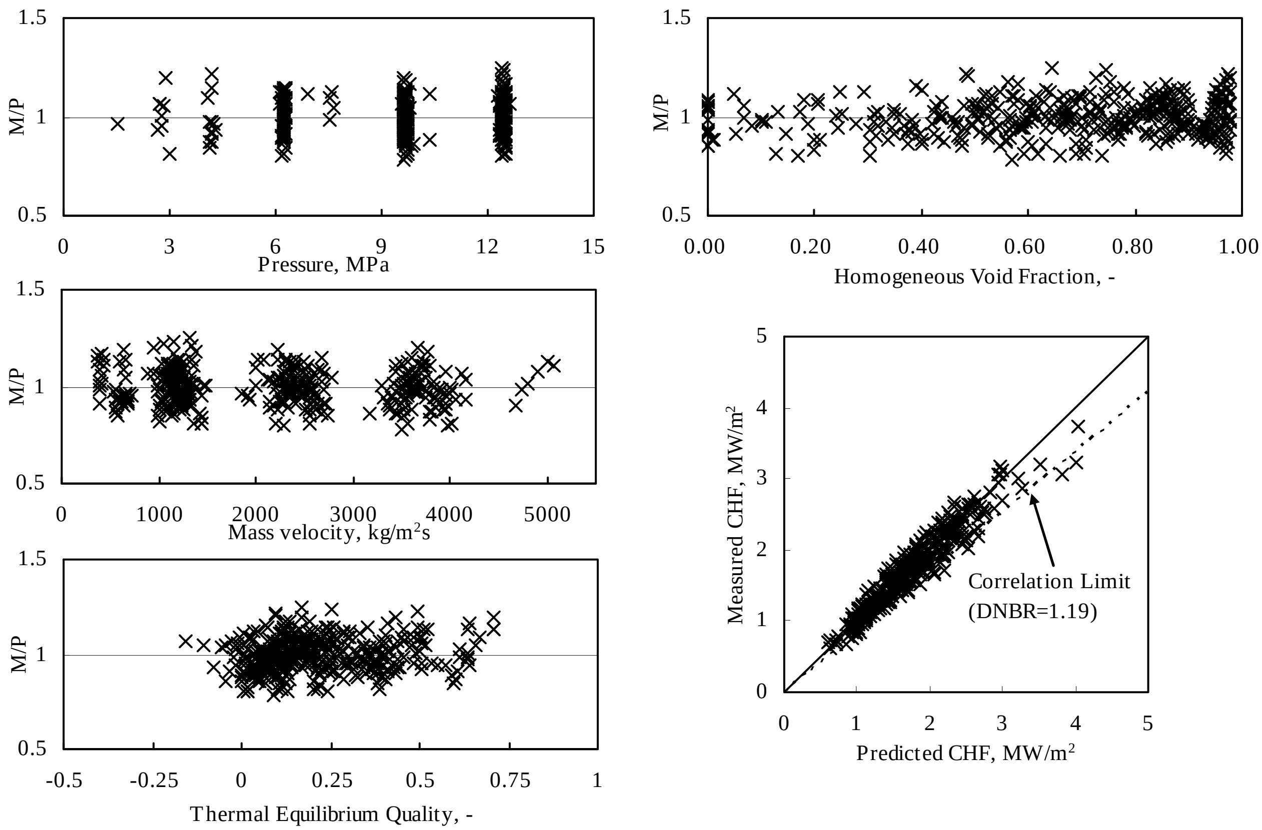

Figure 9 CHF prediction of MG-NV for rod bundle database

3. Applicability to the plant analysis

3.1 Effect of rod bundle configuration

Since the test sections used in above discussions are different from the MHI's fuel design, additional study was carried out for the applicability of MG-NV to MHI fuel.

Figure 10 shows the applicability of the MG-NV to test section 162 in Table 1, of which rod bundle geometry and grid spacer design is quite similar to MHI's standard fuel design. The mixing-vane grid spacers are located in 0.56 m spacing. The data were obtained in HTRF-CU by using the test section which is included in the database of MHI's standard design CHF correlation, MG-S [8]. In order to take into account the non-uniform axial power distribution, Tong's F-factor [1] is applied. Although the number of data within the applicable range of MG-NV is limited, MG-NV provides the specifically conservative predictions due to the mixing-vane effect of the test section. The correlation limit on 95/95 basis is 1.10 for the following parameter range:

Pressure (P):	5.1 through 12.5 MPa
Mass velocity (G):	760 through 4,790 kg/m^{2}s
Thermal equilibrium quality (x):	0.07 through 0.64
Homogeneous void fraction (α):	0.40 through 0.97
Heated equivalent diameter (D_H):	12 mm

The result endorses that the MG-NV correlation and the limit DNBR of 1.19 can be conservatively applied to MHI's mixing-vane grid spacer fuel.

In addition, the most typical rod bundle tests included in the MG-S database are compared with MG-NV as shown in Figure 11. The rod bundle configuration is basically the same as the previous test section except that the MHI's latest standard design grid spacers (Z2 grid spacer) are adopted with 0.46m grid spacing and includes thimble cell test in addition to typical cell test. Although those test conditions are covered by the applicable range of MG-NV, the CHF data including unheated control rod guide thimble effect in the MHI fuel configuration are conservatively predicted.

3.2 Comparison with the existing correlations

The comparisons with W-3 correlation and LUT-2006 at the conditions representing typical SLB conditions are shown in Figures 12 and 13. MG-NV provides similar behaviour with LUT-2006 rather than W-3, and is expected to be applicable to the higher quality conditions like as LUT-2006. MG-NV provides relatively larger CHF than W-3 in those conditions and its correlation limit is lower than that of W-3. It is expected that more design margin will be provided in the plant analysis by utilizing the new correlation.

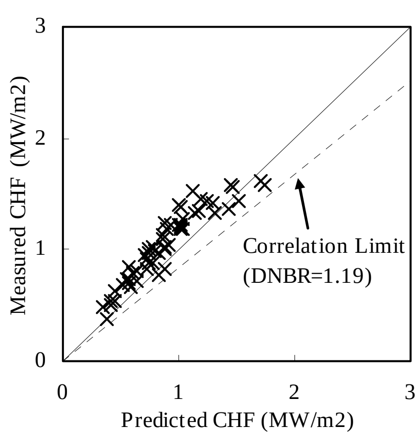

Figure 10 Comparison with low pressure data for MHI type fuel design

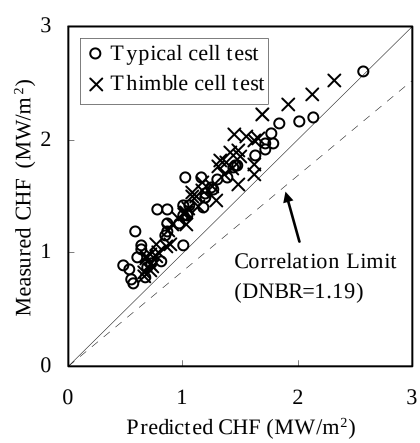

Figure 11 Comparison with high pressure data for MHI type fuel design

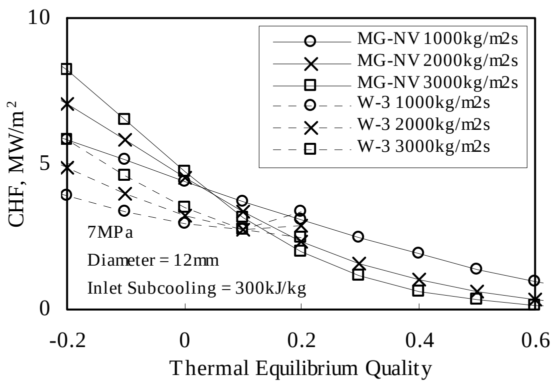

Figure 12 MG-NV versus W-3

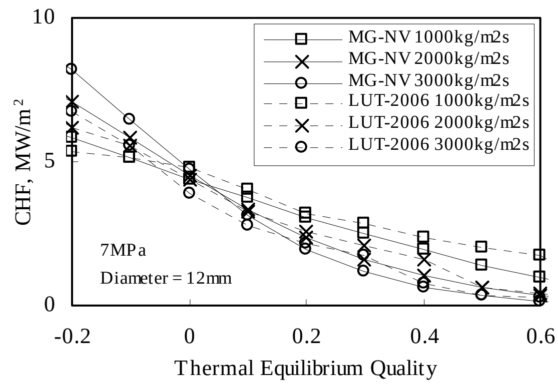

Figure 13 MG-NV versus LUT-2006

4. Conclusions

The new CHF correlation, MG-NV, has been developed in order to cover the low pressure and/or low flow conditions where the standard design correlations do not cover.

The correlation is originally developed based on the analysis of numerous single tube CHF test data, and corrected so as to correlate the rod bundle data with non-mixing-vane grid spacers, which is conservative rather than rod bundle data with mixing-vane grid spacers. The improved applicability to low flow condition is brought by adopting non-linear formula of thermal equilibrium quality. Finally, the correlation is conservatively applicable to the actual plant analysis and the limit DNBR value in the following applicable range is 1.19:

Pressure (P):	1.5 through 12.6 MPa
Mass velocity (G):	370 through 5,000 kg/m^{2}s
Thermal equilibrium quality (x):	-0.16 through 0.71
Homogeneous void fraction (α):	0 through 0.98
Heated equivalent diameter (D_H):	12 through 20 mm

The correlation will be used for MSLB analysis of PWR plant, and will be able to provide more design margin and reliability than conventional correlations.

5. References

[1] L. S. Tong, “Boiling Crisis and Critical Heat Flux”, TID-25887, Atomic Energy Commission, 1972.

[2] EPRI, “Parametric Study of CHF Data”, Vol.3: Critical Heat Flux Data, 1982.

[3] D. D. Hall and I. Mudawar, "Critical Heat Flux for Water Flow in Tubes - I. Comparison and assessment of world CHF data", Int. J. of Heat and Mass Transfer 43, pp. 2573 to 2604, 2000

[4] L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution", J. of Nuclear Energy, Vol. 21, pp. 241 to 248, 1967

[5] B. Thompson et al., "Boiling Water Heat Transfer Burnout in Uniformly Heated Round Tubes: A Compilation of World Data with Accurate Correlations", AEEW-R356, 1964.

[6] EPRI, “Parametric Study of CHF Data Vol.2: A Generalized Subchannel CHF Correlation for PWR and BWR Fuel Assemblies", EPRI NP-2609, vol. 2, 1983.

[7] D. C. Groeneveld et al., "The 2006 CHF Look-up Table", Nuclear Engineering and Design, 237, pp. 1909 to 1922, 2007

[8] T. Yodo, et al., “Development of the New Basic Correlation "MG-S" for CHF Prediction of the PWR Fuels”, NURETH-14, 434, 2011 September 25-30.

[9] G.F. Stevens, et al., “An Experimental Investigation into Forced Convection Burn-Out in Freon, with Reference to Burn-Out in Water”, AEEW-R321, 1964.

[10] Y. Makino, “Thermal Hydraulics Coupled with Core Characteristics”, Workshop of 14th International Conference on Nuclear Engineering,. 2006 July 19.

[11] S. Aoki, et al., “Analysis of the Main Steam Line Break Benchmark (Phase II) Using ANCK/MIDAC Code”, J. of Nuclear Sci. and Tech., Vol.45, No.1, 2008, pp.36-44.

[12] P. Saha and N. Zuber, "Point of Net Vapor Generation and Vapor Void Fraction in Subcooled Boiling", Proc. of 5th Int.Heat Transfer Conference., 4, pp. 151 to 157, 1974

[13] R. T. Lahey Jr. and F. J. Moody, "The Thermal-Hydraulics of a Boiling Water Nuclear Reactor", American Nuclear Society, 1977

[14] B. Chexal et al., "A Void Fraction Correlation for Generalized Applications", NURETH-4, 1989

[15] D. G. Reddy, et al., "Two-Phase Friction Multiplier for High Pressure Steam-Water Flow", EPRI-NP-2522, 1982

[16] R. T. Lahey, Jr. et al. "Mass Flux and Enthalpy Distribution in a Rod Bundle for Single- and Two-Phase Conditions", Journal of Heat Transfer, 1971.

[17] R. E. Odeh and D. B. Owen, "Tables for Nominal Tolerance Limits, Sampling Plans, and Screening", 1980.