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Applicability of CHF Correlations Relevant to External Vessel Cooling for In-Vessel Corium Retention

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ABSTRACT

In-Vessel Corium Retention through External Vessel Cooling (IVR-EVC Concept) has been suggested as one of the most effective measures for the interruption of severe accident progression. Through the considerable experiments and analytical devotes, the real applications of this concept into operating and advanced nuclear power plant have been discussed and investigated. In the evaluation of the IVR-EVC concept, one of the most important items in the heat transfer from reactor vessel to flooded water is the critical heat flux at the outer vessel surface. Therefore, experimental works considering various scales have been conducted to identify the critical heat flux and the underlying mechanism. In this paper, the suitable critical heat flux equation for the evaluation of the IVR-EVC concept has been identified considering the Korean Next Generation Reactor and large-scale experimental conditions.

1. Introduction

The prevention and mitigation measures for severe accident have been intensively investigated since the TMI-2 and Chernobyl accident. Due to these considerable works, some kinds of cooling methodologies for the molten corium have been suggested, i.e. gap cooling, in-vessel corium retention through external vessel cooling (IVR-EVC) concept, corium cooling at reactor cavity and core catcher. Among these candidates, the corium cooling at the reactor cavity has been highlighted as the realistic strategy for the corium cooling. Moreover, several design requirements relating corium cooling at reactor cavity for next generation reactor have been suggested. However, the corium ejected from vessel breach result in very complicated phenomena, i.e. the molten core-coolant interaction inducing the steam explosion and corium-concrete interaction (CCI). The ejected corium also induces threats to the containment, i.e. containment over-pressurization due to the non-condensable gases resulted from the CCI. In addition, the coolability of the molten core, which is relocated at reactor cavity, is raised as one of the most uncertain issues. Therefore, these items result in difficulties of the analysis for the corium cooling at reactor cavity.

Recently, to overcome the above concerns and exclude complicated ex-vessel phenomena, the IVR-EVC concept has been suggested as an accident management strategy. For the real application of the IVR-EVC concept into nuclear power plants, relatively large-scale experiments and analyses have been accomplished to evaluate the feasibility of the concept. Specially, the feasibility of this concpet for Louviisa in Finland and AP600 in U.S. has been investigated and the real applications have been adapted in both plants. Due to the exclusion of the complicated ex-vessel phenomena and simplification of the accident management, the applicability of the IVR-EVC concept into Korean Next Generation Reactor (KNGR) is discussed and investigated. However, the effectiveness of this concept is plant specific and depends on accident scenarios and the real application is very difficult without detail feasibility analyses: thermal analysis and mechanical analysis, etc. Especially, thermal analysis can be accomplished by the comparison between the heat flux through reactor vessel wall from corium pool and the critical heat flux (CHF) at that position. If the heat flux at any position is smaller than the CHF, the integrity of reactor vessel can be maintained. In this point of view, CHF is one of very important parameters for evaluating the integrity of reactor vessel and various kinds of experiments have been conducted to identify the CHF value and underlying mechanism.

In this paper, to assist the feasibility evaluation of the IVR-EVC concept for KNGR, recent status of the art related to CHF are described and the most suitable CHF correlation is identified considering the KNGR and experimental conditions.

2. Overview of Relevant Experimental Works

2.1 Small-scale Experiments

Small-scale experiments have been accomplished not to apply experimental results to real situations, but to identify the effect of inclination angle and size on CHF. However, the understandings about underlying CHF mechanism in small-scale experiments are helpful to get basic information for large-scale experiment. Therefore, the simple descriptions are described as follows:

Author	Test condition	Conclusion
Ishigai et al. [1]	Heater dia. (d): 25 & 50 mm Surround material dia. (D): 50, 100 & 200 mm Saturated water Atmospheric pressure	CHF for 0^o decreases with increase of the diameter of the heated surface and decrease of the diameter ratio (d/D). Complete dryout was not occurred at the heated surface, even if the vapor blanket covered the all of the heated surface.
Githinjii & Sabersky [2]	Heater: 1/8×4 inch Chromax Subcooled isopropyl alcholol	Heat transfer characteristics for 0^o was somewhat different from those of 180^o and 90^o and CHF was minimal at the 0^o. CHF for 90^o is somewhat larger than that for 180^o (Subcooling Effect).
Anderson & Bova [3]	Heated: 2, 6 and 12 inch flat plate for 0^o Freon-11 pool Indirect heating	CHF decreases with increase of characteristic length, i.e. diameter, of the heated surface. The effect of inclination angle on CHF seemed to be small till 15^o. Melt through of the plate was visible due to the low CHF.
Vishnev et al [4]	Heater: 96×10.4×0.062 mm Stainless steel Helium pool Inclination angle: 180, 150, 145, 90, 60, 30 and 0^o	As the inclination increases, heat transfer coefficient increases under fully developed nucleate boiling. CHF decreases with increase of the inclination angle due to the bubble distortion, decrease of bubble escape frequency, etc.
El-Genk [5, 6, 7]: Fig. 1	<u>Flat plate test section</u> (dia.: 50.8mm): copper <u>Curved test section</u> (dia.: 50.8, curvature radius: 148, thickness: 12.8, 20 & 30 mm): copper <u>Curved test section</u> (dia.: 75, curvature radius: 218.5, thickness: 20 mm): copper & stainless steel Saturated Water Atmospheric pressure	<u>*Experiment with flat plate :*</u> CHF increases with increase of inclination angle. <u>*Experiment with curved test section :*</u> CHF is not affected by the thickness of the heated surface over 20 mm. CHF decreases with increase of inclination angle and the behavior is somewhat different from those of large-scale experiments. <u>*Experiment with curved test section (different material);*</u> For the inclination angles of 0^o and 7.91^o, the CHFs for stainless steel are 12% and 40 % lower than those for copper. CHF occurred at the lower most position, 0^o, and moved to upper position along the inclination angel. This is also different from the SBLB experiment performed by F.B. Cheung and need some verification.

* Horizontal downward facing (0^o), vertical position (90^o), horizontal upward facing (180^o)

In addition to small-scale experiments, large-scale experiments, ULPU-2000, SBLB and SULTAN, have been also conducted to identify the CHF. Specially, the correlation resulted from the ULPU-2000 configuration-II is used for the safety evaluation of the AP600. Their detail descriptions are described as follows:

2.2 ULPU-2000 Experiment

To define the coolability limit on the real-scaled hemispherical vessel, T.G. Theofanous et al [8] in UCSB (University of California, Santa Barbara) have performed ULPU-2000 experiment, which is large scale experiments with 3 different kinds of configuration. Schematics of each configuration are shown in fig. 2. Here, two-dimensional curved surface made by thick copper has been used in ULPU-2000 experiment.

Using configuration I, they have performed an experiment to identify the CHF behaviors at the lower positions (between 0^o and 30^o). The test section is divided into 8 areas (number 8 is the lowermost area, from 0^o to 3.75^o, and 1 is uppermost area, from 26.25^o to 30^o). Here, experiments have been conducted using two kinds of heating method, i.e. uniform heating method (UF) and peak heating method to occur the CHF at special position (SF). The number following the signs, UF and SF, means the area, at which uniform heat flux is maintained or CHF is occurred, respectively. A series of UF experiments is to simulate the CHF at the bottom position; some positions are heated up uniformly and the other positions are maintained at different heat flux. For example, UF-6-50% means that 6, 7 and 8 areas are maintained at uniform heat flux and the other areas (from 1 to 5) are maintained at 50% of uniform heat, which are applied at areas of 6, 7 and 8. A series of SF experiments is to simulate CHF at specified position. For example, SF-5 means that CHF is occurred at area 5, between 11.25^o and 15.0^o. Based on the experiments for the configuration I, they suggested following CHF correlation:

$$q_{CHF} \quad = 300 \qquad kW/m^2 \qquad \theta < 5^o$$
$$= 300 + 12.6 \cdot \theta \qquad kW/m^2 \qquad 5^o < \theta < 30^o$$

They also performed experiments using the configuration II to clarify the CHF behaviors for the overall inclination angles. The test section of configuration II is composed of three curved surfaces, one of that has the same geometry and material as that used in configuration I. The experimental methodologies are very similar to those of configuration I, except the power limitation above certain position. This limitation is resulted from the possibility of test section failure without control of heating power. For example, UF-8-145% means that the 8 areas from the lowermost are maintained at uniform heat flux and the heat fluxes at the other areas are calculated considering the power shaping principle. However, the heat flux at the uppermost area is limited to the 145% of the uniform heat, because of the possibility of the CHF at downstream. Therefore, the heat fluxes at the other areas are re-calculated considering the heat flux limitation. The major observations from configurations II are as follows: 1) higher CHF compared to that of configuration I (~ 50%) resulted from the effects of induced water flow and subcooling, 2) existence of a transition angle (~15^o), at which the increase rate of CHF changes. Based on the results of configuration II, they suggested following CHF correlation:

$$q_{CHF} \quad = 500 + 13.3 \cdot \theta \qquad kW/m^2 \qquad \theta < 15^o$$
$$= 540 + 10.7 \cdot \theta \qquad kW/m^2 \qquad 15^o < \theta < 90^o$$

To clarify the insulation effect, they have also performed another experiment using the configuration III facility. To simulate the real situation as real as possible, 15 cm riser and 10 cm exit restriction have been installed in experimental loop. In addition, the configuration III experiments are aimed to find out the following items: effect of the change in the flow geometry (riser and exit restriction) and composite effect of the change in the flow geometry and the material of the test section (copper & ASTM Stand Class 3 Steel). In the experiment of configuration III, CHF is occurred in the case of copper, however, CHF is not occurred in the case of the steel due to the failure possibility of the test section. However, they suggest that the nucleate boiling is maintained at the vicinity of the CHF predicted by the correlation of configuration II.

Finally, they suggested that the CHF is considerably higher than heat flux transferred from the corium pool through the reactor vessel and IVR-EVC concept has large thermal margin in the case of AP600.

2.3 SBLB Experiment

SBLB (Subscale Boundary Layer Boiling) experiment has been conducted by F.B. Cheung et al [9] in Pennsylvania State University, to clarify following items: local behavior of the CHF, subcooling effect of the working fluid on CHF, physical characteristics in the dynamic behavior of the 2-phase boundary layer. Experimental apparatus is consisted of three part, pressurized water tank with condenser, hemispherical and toroidal test section and data acquisition system, as shown in fig. 4. Two kinds of experiments, quenching and steady state heating methods, have been performed with hemisphere test sections, which have 8 inch and 12 inch diameter, under subcooled and saturation condition. The quenching experiment is accomplished to find out the CHF behavior, in the other hand, the steady state heating experiment is conducted to clarify the underlying mechanisms of CHF and the dynamic behavior of the boundary layer. Detail experimental results on the subcooling and inclination angle effect on the CHF are shown in fig. 5. As shown in fig. 5, the CHF increases with increase of the liquid subcooling and inclination angle. Particularly, the increase rate of CHF with inclination angle changes at the special position, between 30^o and 50^o.

Based on the SBLB experimental results, they evaluated the thermal margin, the difference between the CHF and heat flux through the reactor vessel lower head, of AP600 (100% core relocation, 3.99 m diameter of the reactor vessel). According to the analysis, the volume and height of corium pool are about 13.9 m^3 and 1.9 m, respectively. Considering the typical decay heat level, 16MW, the volumetric heat generation rate is calculated as 1.15 MW/m^3. According to evaluation results, the IVR-EVC concept has sufficient thermal margin in the case of AP600.

In addition, they also performed an experiment to clarify the effect of thermal insulation on CHF, using the test section shown in fig. 4. According to their experiments, the CHF is 3 times higher than previous experiment, which have no thermal insulation around test vessel, at the bottom center of test vessel (over 1.2 vs. 0.45 MW/m^2, respectively). In addition, they suggest that the flow rate and CHF (~0.98 MW/m^2) are minimum at 45^o, which is smaller than that of bottom center. This is resulted from the difficulty of vapor venting at the position of minimal gap (~45^o).

2.4 SULTAN Experiment

To identify the coolability of large facility in natural convection due to the boiling, SULTAN facility is prepared by the CEA/CENG, France [10]. The program has following detail objectives: the provision of database to verify the analysis code for system, which has large thermal-hydraulic diameter under low pressure, the measurement of boundary layer thickness, the distribution of void fraction, the temperature distribution in the working fluid, pressure drop & CHF and investigation of the transition to the channeling flow regime. In addition, under the cooperation of the CEA and Department of Nuclear Engineering and Energy Conversion, CHF correlation is developed considering the overall data (192 data). The suggested correlation has about 12% RMS error and predicts the 72% and 89% of all data within ±10% and 20%, respectively.

$$q_{CHF} = \frac{Y_1 d_T^{Y_2} G^{Y_3} + Y_4 d_T^{Y_5} G^{Y_6} \Delta H_{in}^{Y_7} L^{Y_8}}{1 + Y_8 d_T^{Y_9} G^{Y_{10}} L^{Y_{11}}} \bullet \frac{[1 + (\sin\theta)^{Y_{12}}]}{1 + Y_{13}\left(\frac{\rho_g}{\rho_f}\right)}$$

Here, the overall coefficients are as follows:

Y_1	285	Y_6	0.692	Y_{11}	0.31
Y_2	0.986	Y_7	1.02	Y_{12}	2.34
Y_3	0.406	Y_8	0.247	Y_{13}	-10.84
Y_4	0.242	Y_9	3.781	Y_{14}	0.595
Y_5	1.985	Y_{10}	0.362		

3. Selection of CHF Correlations Applicable to KNGR Conditions

Though several experimental works have been performed to identify the CHF at different inclination angles, it is needed for the application of the experimental results to consider the real situations: fairly large scale & geometry and realistic heat flux level, etc. Specially, the effects of several items, i.e. size and geometry of test vessel, subcooling level, recirculation rate, in-core instrumentation, thermal insulation and test section material, should be considered for the selection of suitable CHF correlation. In this point of view, the ULPU-2000 and SBLB experiments have been minutely investigated, because those experiments consider various parameters, which affect the CHF at the external surface of reactor vessel. Details of 2 experiments and KNGR conditions are summarized in table 1.

- Geometry & Size: Although the geometry and size of test section considered in two experiments are very different, CHF levels are similar, as shown in fig. 5. Generally, CHF increase with the increase of inclination angle except for the experimental data of curved surface performed by El-Genk. ULPU Configuration-II correlation well predicts the SBLB data for subcooled condition and over-predicts SBLB data for saturation condition at upstream position. In addition, the ULPU-2000 configuration II correlation overpredicts SBLB data with increase of inclination angle. This can be induced by the strong flow through test vessel in ULPU-2000 experiment.
- Subcooling level: As shown in table 1, SBLB experiment is conducted under four subcooling conditions. ULPU-2000 experiment also considers the subcooling level by the water head, which can be representative condition of nuclear power plant, although most of experiments have been conducted at saturation condition. However, compared to above two experiments, the subcooling level in KNGR is similar or higher depending on the water height above the bottom center of lower head.
- Recirculation rate: T.G. Thoefanous suggest that the increase in CHF of ULPU configuration II compared to configuration I is resulted from the subcooling and recirculation. In this point of view, the recirculation flow rate considerably affects the CHF, however, there is no recirculation flow path in KNGR. Therefore, it is cautious to apply the configuration II correlation to the evaluation of the IVR-EVC concept for KNGR.
- The thermal insulation: To identify the insulation effect on CHF, T.G. Theofanous and F.B. Cheung

performed experiments. Based on the experimental results, the presence of insulation results in the increase of CHF. However, as there is a little information related to the thermal insulation of KNGR, it is difficult to quantitatively evaluate the effect of the insulation.

- The material used in experiment: There is a little effect of material on CHF according to the ULPU-2000 (usage of Copper & A 508 ASTM Standard Class 3 Steel) and SBLB experiment (in the case of heating: usage of stainless steel and Aluminum). Specially, the material used in KNGR is same as that used in ULPU-2000 configuration III. Compared to other effects, the material effect may be small.
- As the effect of in-core instrumentation nozzle is not considered in both experiments, detail analysis should be performed to identify the thermal and hydrodynamic effect of the ICI nozzles.

Table 1. Details of ULPU-2000 & SBLB Experiments and KNGR Conditions

	ULPU-2000 Experiment	SBLB Experiment	KNGR
Geometry & Size	Large-scale 2D slice Radius of curvature : 1.76 m Thickness : ~ 0.076 m	Small-scale hemisphere Radius: 0.1524 & 0.0762 m Thickness: 0.0127 m	Large-scale hemisphere Radius: ~2.4 m Thickness: 0.165 m
Subcooling Level	Saturation condition Configuration I : 1.2~1.8m above the bottom head Configuration II : ~10K at the center of bottom head	Considered 100, 97, 93 and 90^{o}C	The height from the bottom center of lower head to the center of hot/cold leg: ~ 8.5m Subcooling leevel is somewhat larger than two experiments
Recirculation Rate	Configuration I : not measured Configuration II : depend on total heat (typically, ~80 kg/s)	Maximum recirculation Rate: ~ 0.18x10^{-3} kg/s	No recirculation path
In-Core Instrumentation	Not considered	Not considered	Provision of 61 ICI (o.d.: 8.25cm, i.d.: 2 cm)
Thermal Insulation	Considered in configuration III : Increase of CHF	Use new test section: Increase of CHF at bottom center of reactor vessel CHF is minimum at 45^o	Minimum gap: ~15 cm at 45^o
Material of Test Section	Copper & A 508 ASTM Standard Class 3 Steel	Heating: stainless steel Quenching: stainless steel & Aluminum	SA-508 Class 3

Through the detail analysis, following problems are identified for two experiments:

ULPU-2000 Experiment	SBLB Experiment
• Divergence effect is not occurred in the ULPU-2000 experiment, although the scale is very similar to the real situation. • Strong flow (~80kg/s) is observed through the experimental loop. The application of configuration II correlation into the feasibility evaluation of the IVR-EVC concept for KNGR needs caution, because there is no recirculation path in KNGR. • The analysis on the effect of in-core instrumentation on CHF is insufficient.	• Although the geometry used in SBLB experiment is similar to that of KNGR, the size is very small. For the real application, the scaling methodology should be provided. • The effect of thermal inertia may be neglected in SBLB experiment, due to the thin thickness of test vessel compared to that of KNGR. • Same influence of subcooling is applied at every position through the outer surface of test vessel. However, the effect of subcooling is different at each position in real situation due to the cavity water head. • The analysis on the effect of in-core instrumentation on CHF is insufficient.

Determination of suitable correlation

- The condition of ULPU-2000 is somewhat different from that of KNGR: the flow behavior at the lower head of reactor vessel, the presence of the recirculation path. In addition, the curvature of the test vessel used in ULPU-2000 is ¾of that used in KNGR.
- The size of test vessel used in SBLB experiment is considerable smaller than that used in KNGR. Therefore, the adequate scaling methodology should be provided for the application of the SBLB data to KNGR condition.
- Basically, the usage of ULPU-2000 configuration II correlation for the evaluation of IVR-EVC concept in KNGR is reasonable, however, the effects of following items on CHF should be considered:

- Divergence effect, at the bottom center of the reactor vessel, may increase CHF.
- Depending on the condition of the nuclear power plant, induced flow effect, i.e. recirculation flow path and thermal insulation, may be occurred. This may also increase the CHF.
- As shown in SBLB data, CHF linearly increase with subcooling. Therefore, considering the subcooling effect in ULPU-2000 correlation, thermal margin will be enhanced.
- Scale effect
- The effect of in-core instrumentation nozzle is not considered in ULPU-2000 experiment. Therefore, for the enhancement of understanding on the thermal and mechanical behavior, detail analysis should be performed.

4. Conclusions

Considering large-scale experimental conditions and KNGR conditions, the usage of ULPU-2000 configuration II correlation for the feasibility study of IVR-EVC concept in KNGR is reasonable. More precise analysis on the feasibility study of the IVR-EVC concept, it is needed to identify following items:

- Divergence effect at the bottom center of the reactor vessel
- Induced flow effect
- Subcooling effect
- Scale effect
- The effect of in-core instrumentation nozzle

References

1. S. Ishigai, K. Inoue, Z. Kiwaki and T. Inai, “Boiling Heat Transfer from a Flat Surface Facing Downward”, *International Development in Heat Transfer*, pp. 224-229. 1961.
2. P.M. Githinji et al., "Some Effects of the Orientation of the Heating Surface in Nucleate Boiling", *ASME Journal of Heat Transfer*, Vol. 85(4), pp. 379, 1963.
3. R.P. Anderson and L. Bova, "The Role of Downward Facing in Post-Accident Heat Removal", *Transaction American Nuclear Society*, Vol. 14, pp. 294, 1971.
4. I.P. Vishnev et al., "Study of Heat Transfer in Boiling Helium on Surface with Various Orientations", *Heat Transfer-Soviet Research*, Vol. 8(4), pp. 104-108, 1976.
5. M.S. El-Genk and Z. Guo, "Transition Boiling from Inclined and Downward-facing Surfaces in a Saturated Pool", *International Journal Refrigeration*, Vol. 16(6), pp. 414-422, 1993.
6. M.S. El-Genk and A.G. Glebov, "Transient pool boiling from downward-Facing Curved Surface", *International Journal of Heat and Mass Transfer*, Vol. 38(12), pp. 2209-2224, 1995.
7. M.S. El-Genk and C. Gao, "Pool Boiling from Copper and Stainless-steel Downward-Facing Curved Surfaces", *2nd European Thermal-Science and 14th UIT National Heat Transfer Conference*, Rome, Italy, May 29-31, 1996, submitted paper.
8. T.G. Theofanous et al., “In-Vessel Coolability and Retention of a Core Melt”, *DOE/ID-10460*.
9. F.B. Cheung et al., “Critical Heat Flux Phenomenon on a Downward Facing Curved Surface”, *NUREG/CR-6507*, 1997.
10. S. Rougé, “SULTAN Test Facility: :Large Scale Vessel Coolability in Natural Convection at Lowe Pressure”, *NURETH7 Conference*, Saratoga Spring, NY, USA, Sept. 10-15, pp. 1949~1957, 1996.

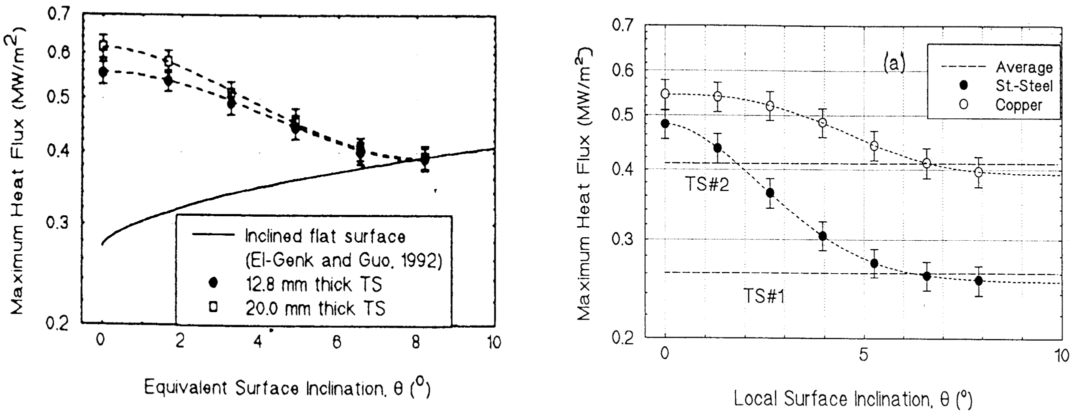

Fig. 1 El-Genk et al's Experimental Results

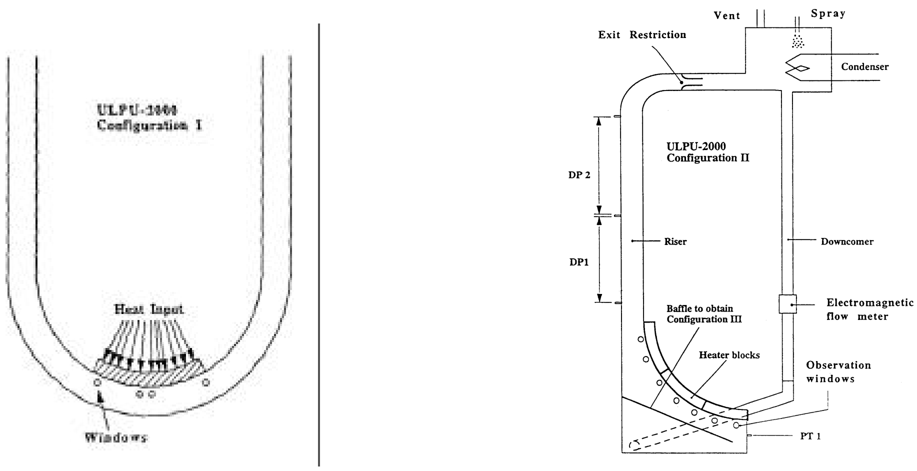

Fig. 2 Schematic of the ULPU-2000 Configuration I, II and III

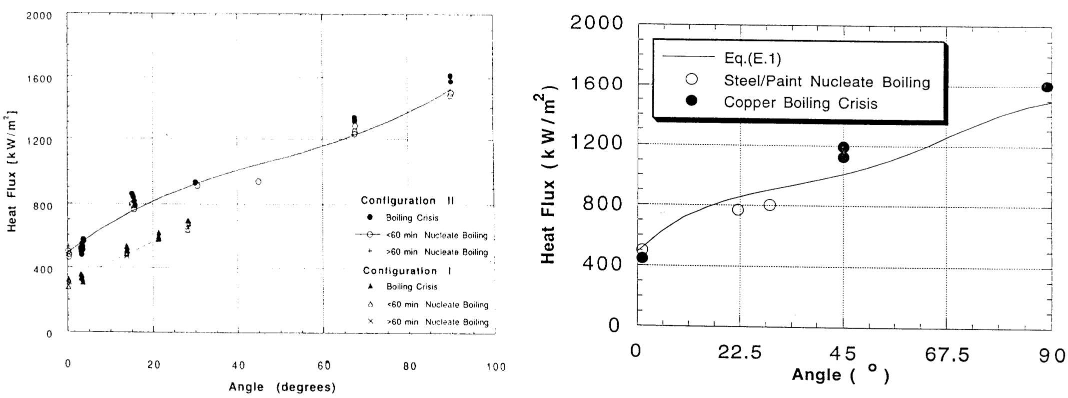

Fig. 3 Experimental Results for ULPU-2000 Experiments

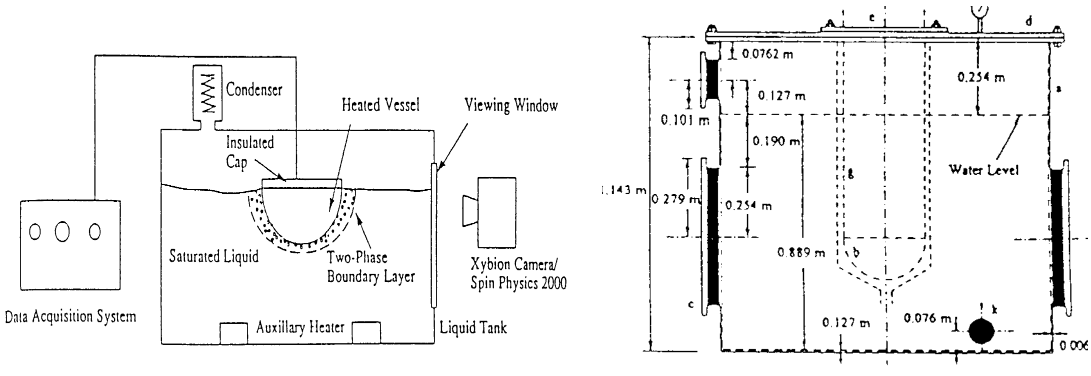

Fig. 4 Experimental Apparatus of SBLB

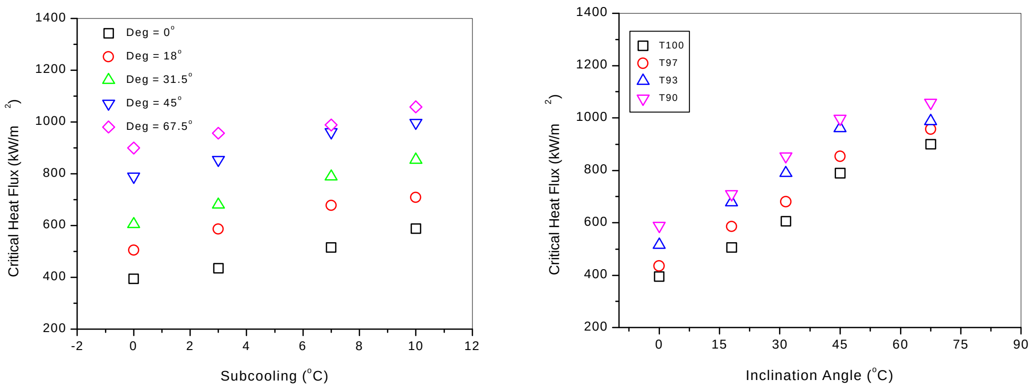

Fig.5 Experimental Results of SBLB

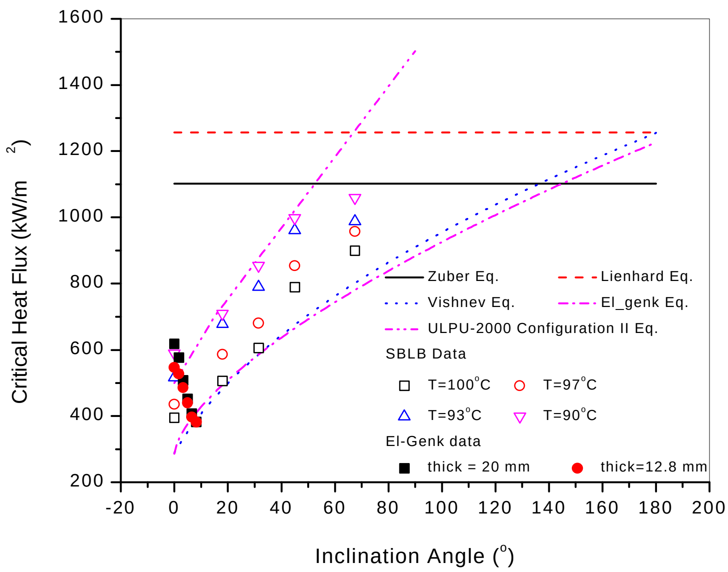

Fig. 6 CHF Behavior with Inclination Angle