NUPEC THERMAL HYDRAULIC TEST TO EVALUATE POST-DNB CHARACTERISTICS FOR PWR FUEL ASSEMBLIES (1. GENERAL TEST PLAN AND RESULTS)

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In the present thermal hydraulic design of Pressurized Water Reactor (PWR), a departure from nucleate boiling (DNB) under anticipated transient conditions is not allowed. However, it is recognized that the DNB dose not cause a fuel rod failure immediately, and a suitable reactor trip can prevent the core from severe damages. If the fuel rod temperature under the post-DNB conditions can be accurately evaluated, the potentially existing margin in the present design method will be quantitatively assessed . To establish the heat transfer evaluation method on post-DNB event for PWR thermal hydraulic design, Nuclear Power Engineering Corporation (NUPEC) started a program, NUPEC Thermal Hydraulic Test to Evaluate Post-DNB Characteristics for PWR Fuel Assemblies (NUPEC-TH-P), in 1995 (hereinafter the year means fiscal year) under the sponsorship of Ministry of Economy, Trade and Industry (METI). This program is now under going until 2001. This paper is to show the overall plan and the status of NUPEC-TH-P.

Introduction

The reliability and safety of nuclear fuel is one of the most important issues for nuclear power plant design and operation.

Extensive experience of fuel usage has been piled up in Japanese nuclear power plant, but in order for enhancing the national consensus of nuclear power plant operation and construction, further demonstrations of core and fuel safety are indispensable through the in-pile and/or out-of-pile proven tests for acquiring the fuel irradiation behavior and the thermal hydraulic characteristics.

A series of the fuel assembly reliability proving test sponsored by METI are planned to prove the comprehensive fuel reliability in Japanese PWR and BWR various operating conditions.

These tests consist of two kinds of test, one is irradiation test which aims to get the irradiation behavior of in-core fuel and the other is thermal-hydraulic test which aims to get the thermal and hydraulic characteristics of fuel assembly.

As for the proving test on the thermal-hydraulic design reliability of PWR fuel assembly, which is one of the constituents of the proving tests, critical heat flux test (DNB test) and in-bundle void fraction test have been performed from 1983 till 1994. These tests appealed the reliability of fuel thermal design procedure by making the fuel thermal characteristics clear within current design operating condition. Current thermal design procedure assumes that the fuel on which DNB occurs are identified to be instantaneously failed, and consequently DNB occurrence is not accepted in normal and anticipated operational transient reactor conditions.

On the other hand the data of fuel behavior after the DNB occurrence has been accumulated recently and it has become clear that the fuel clad suffered DNB dose not always lose its integrity as far as the fuel rod is not exposed in high temperature condition long time.

In Japan incidentally the introduction of high burn-up and/or MOX fuel into the nuclear reactors are planned for the purpose of saving uranium resource and reducing fuel cycle cost, and these operation might result the aggravation of power peaking and might reduce the flexibility of core operations.

Under the circumstances mentioned above, the post-DNB test started in 1995 succeeding the in-bundle void fraction test to quantify the potential margin which current design (namely, not to allow DNB occurrence) has and to develop the rational and realistic evaluation procedure.

Historical summary of implemented test and planning test

In 1995 the test scheme and basic test method were studied and preliminary test was performed to confirm the durability of test rod, the thermo couple installation method and so on. From 1996 to 1998 small scale model test was performed using Freon as working fluid and the existing DNB test apparatus was remodeled to conform post-DNB test for the same period. Freon has, as is well known, much less latent heat and saturation temperature than those of water, and hence the Freon test is more simple and flexible than the water test. Taking advantage of these simplicity and flexibility, before performing mock-up water test, we studied the effect and/or sensitivity of widely ranged parameters such as grid span length, grid type, rod bow and so on. by the Freon test.

From 1999 to 2001 a series of mock-up water tests are planned to perform. The purpose of the mock-up water tests is to get the post-DNB heat convection data by using 5x5 rod bundle, the length of which is same as that of existing nuclear fuel assembly, and to evaluate the design margin quantitatively hidden in the current thermal design procedure.

The tests consist of "base test", "17x17 test", "thimble cell test", "cosine power shape test" and "rod bow test".

The base test simulates the 14x14 PWR fuel assembly typical cell and prepares the reference standard data for the succeeding tests. The 17x17 test is expected to show the effect of rod pitch and grid span length. The thimble cell test will show the effect of cold wall in the flow channel. The axial power distribution of the base test is the top skew type, which represents the core EOL power distribution and is estimated to be the severest condition from the view point of DNB phenomenon, on the other hand, the cosine power shape test, which represents the typical BOL power distribution and is used for the DNB estimation in standard core design, is to get the effect of the shape of axial power distribution. The rod bow test will clarify the DNB penalty and the post-DNB heat transfer behavior of bowed rod.

The NUPEC-TH-P test schedule and the content are shown in table 1 and 2 respectively.

Test Apparatus

The setup of Freon model test which is owned by Takasago Research Institute of Mitsubishi Heavy Industries, LTD. is shown in figure 1 and the principal equipment capacity is as follows.

Maximum Heating Rate	144 KW
Maximum Flow	13.6 t/h
Design System Pressure	3.5 MPa

The outline of the water mockup test loop, which was installed in Tkasago Engineering Center of NUPEC, is shown in figure 2 and the principal features of the test apparatus are as follows.

- Capacity

Maximum Heating Rate	5.6 MW
Maximum Flow	60 t/h
Design System Pressure	19.1 MPa

- Execution of unsteady state test is feasible.
- Remote operation and concentrated monitor of loop and measuring instrument is feasible.
- Automatic startup and termination of loop operation are feasible.

Water mockup test condition

Assumed PWR transient events are summarized in table 3. The slow events can expect the negative nuclear feedback by the formed void effect and the void formation will restrict the DNB outbreak. Of course the strict evaluation is needed to estimate the effect for the application to the existing core, but the fundamental method to predict the void formation was developed in the in-bundle void fraction test as a part of series of NUPEC-TH-P test.

The rapid events cannot expect the void effect and DNB might outbreak but the fast parameter changes of reactor will result the short DNB duration due to the core protection system.

The relationship between PWR transient events and the NUPEC-TH-P tests are shown in figure 3.

Two kinds of post-DNB test, namely, steady state test and unsteady state test are performed.

[Steady state test]

In the steady state test, pressure, flow, heater rod power and inlet temperature are settled to the appointed values and the inlet temperature goes up gradually and the heater rod thermo couple data are measured at DNB and post-DNB conditions. The range of each parameters are determined to cover the core condition under which DNB occurs as is shown in table 4.

[Unsteady state test]

In the unsteady state test, the test parameters, namely, power, flow and inlet temperature, are settled to the predetermined values and one of the parameter is changed to realize the DNB, the boiling transient and the film boiling.

The parameter change rates shown in table 4 are settled to cover the various transient events occurred in the PWR core.

Evaluation Items

The post-DNB test is now on progress and finishes in 1991. In the final stage of test, total evaluation of all the test including the Freon test are planned on the following items

(a) Comparison of the DNB outbreak condition with the prediction by the design DNB correlation

(b) Comparison of the test data with the prediction by the published post-DNB correlations
- Thermal conductivity of heater rod surface in boiling transient
- Thermal conductivity of heater rod surface in film boiling
- Rewet condition

(c) Preparation of the correlation for predicting the post-DNB thermal conductivity in PWR rod bundle considering following effect
- Axial power shape
- Control rod guide thimble (cold wall) effect
- Rod bowing (Channel Closure)
- Fuel assembly geometrical difference between 14x14 rod array and 17x17 rod array

(d) Re-evaluation of rod bow DNB penalty

Principal result of Freon model test

The results of "grid effect test" and "grid span length test" were shown here as the typical results of model test.

The former test was to survey the effect of grid mixing vane by comparing the results of the tests using mixing vane grid or SS (simple support without mixing vane) grid.

The latter test was to examine the effect of gird span length.

The followings are the results obtained by these tests.

(a) The mixing vane grid yielded higher heat transfer coefficient than the SS grid, hence mixing vane was confirmed to promote heat transfer. (figure 4)

(b) The shorter grid span length showed better heat transfer performance than the longer ones. (figure 5)

(c) As is commonly observed in the forced convection phenomena, the Nusselt numbers are well correlated with the Reynolds numbers in the post-DNB heat transfer phenomenon.

.(d) Power level when re-wet started was almost same as that when DNB occurred. (figure 6)

Provisional result of water mockup test

Among a series of proving test, the standard test was first conducted in 1999. The test assembly consists of full length 25 (5x5) heater rods which simulate PWR 14x14 fuel rod bundle configuration. Followings are the summaries obtained from the standard test.

(a) Rod temperature change of 14x14 fuel in transient condition

Figure 7 shows the heater rod surface temperature change observed in the unsteady-state power increase test which simulates PWR transient condition. The power of test rod increases up to 120% of full power at the ramp rate of 30%/sec, then the power decreases after the appropriate delay time on which the trip operation is surely expected in existing core.

Figure 7 suggests the following clad temperature characteristics.
- Maximum rod surface temperature remains less than 600C and does not increase to extremes.
- The rise in temperature gradually turns to the descent due to the power down as the result of the trip operation expected in the existing core.

(b) Film boiling heat convection characteristics in transient condition

The Nusselt numbers and the Reynolds numbers acquired in this rod bundle test together with the single rod test performed in 1995 were correlated in Figure 8.

Figure 8 shows that the Nusselt numbers has positive correlation with the Reynolds numbers, and the single rod test data points are on the production of the bundle test data.

Conclusions

It is generally believed up to this test that the fuel clad temperature rise after the DNB occurrence in PWR condition is very steep and the final temperature level is pretty high, and hence the resultant instantaneous fuel rod failure is quite within the bounds of possibility. Figure 7 and other 1999 water mockup test data, however, suggest that the fuel clad temperature rise after the DNB occurrence is rather moderate and due to the reactor trip operation the temperature goes down before reaching such level that the fuel failure occurs.

Reference

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Table 1 NUPEC Thermal Hydraulic Test Schedule (PWR Post-DNB Test)

Item	Contents	'75	'76	'77	'78	'79	'00	'01
1. Fundamental Test								
(1)Mockup Proven Test Plannig	Test Method & Procedure Measuremnet Sysytem							
(2)Preparation Test	Heater Rod Specification Working Fluid : High Pressure & High Temperature Water Test Rod : Single Short Heater Rod							
(3)Small Scale Test	Wide Range Data Acquisition Working Fluid : Freon R-123 Test Rod : Single Short Heater Rod 3x3 Short Rod Bundle							
2. Remodelling of Teat Apparatus	Design, Fabrication and Built-in of Test Equippment for Water Mockup Test							
3. Fabrication of Heater Rods	5x5 Heater Rod Bundle Fabrication for Mockup Proven Test							
4. Mockup Porven Test	Five kinds of test are planned.							
(1) Test	· Trial Run of Loop · base test · 17x17 test · thimble cell test · cosine power shape test · rod bow test Working Fluid : High Temperature & High Pressrue Water Test Rod : 5x5 Long Size Rod Bundle							
(2) Evaluation	· In-Bundle Heat Transfer Behavior · Post DNB Heat Transfer Correlation · Bowed Rod DNB Penalty							

Table 2 NUPEC-TH-P Post-DNB Test Items (1/2)

Item	Content
1. Small Scale Test	
1) Working Fluid	Freon R-123
2) Test Rod	Rod Array : Single rod and 3x3 rod bundle Test Rod Length : Short (1.5m) Heater Rod Test Rod Dia./Thickness : 9.5mm / 0.57mm Heating Method : Sheeth Heaing.
3) Kinds of Test and Objectives	
a) Single Rod Test	· Study the similarity between water and Freon heat transfer behavior and between single rod test and rod bundle test
b) Standard Test	· Yield the standard data for comparing the following series of test
c) Simple Support Grid Test	· Study the effect of the grid with mixing vane.
d) Grid Span Length Test	· Study the influence of grid span length.
e) B-type Grid Test	· Study the differnce of performance between A-type grid and B-type grid.
f) Rod Bow Test	· Study the effect of channel closure due to the rod bowing.
2. Mockup Water Test	
1) Working Fluid	High Pressure & High Temperature Water
2) Test Rod	Rod Array : 5x5 Test Rod Length : Real Length (3.6m) Heater Rod Test Rod Dia./Thickness : PWR 14x14 type FA 10.7mm / 0.62mm PWR 17x17 type FA 9.5mm / 0.57mm Heating Method : Sheeth Heaing.
3) Kinds of Test and Objectives	
a) base test	· Yield the standard data for comparing the following series of mockup proven test.
b) 17x17 test	· Study the effect of 17x17 type fuel geometry
c) thimble cell test	· Study the effect of cold wall (non-heating rod)
d) cosine power shape test	· Study the effect of axial power shape
e) rod bow test	· Study the effect of rod bowing to the post-DNB heat transfer behavior · Clarify the rod bow DNB penalty

Table 3 PWR Transient Condition and Potential Margin as to DNB Evaluation

Event	Origin of Potential Margin: Void Effect	Origin of Potential Margin: Post-DNB Heat Transfer
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (rapid RCCA withdrawal)	×	○
(slow RCCA withdrawal)	○	×
Disordered Rod Cluster Control Assembly Drop	○	×
Abnormal Dilution of Boron in Reactor Coolant	○	×
Partial Loss of Forced Reactor Coolant Flow	×	○
Startup an Inactive Reactor Coolant Pump	△	○
Abnormal Increase of Steam Load	○	×
Abnormal Decompression of Secondary Cooling System	○	×
Excessive Water Feed to Steam Generator	○	△
Loss of External Power Supply	×	○
Abnormal Decompression of Reactor Cooling System	○	△
Loss of Load	△	△
Complete Loss of Forced Reactor Coolant Flow	×	○
Reactor Coolant Pump Shaft Seizure (Locked Rotor)	×	○
Feedwater System Pipe Break	△	△
Main Steam Pipe Break	○	×
Steam generator tube rupture	△	△
Rod Cluster Control Assembly Ejection Accident	△	△

○ : Potential margin is Expected

△ : Some amount of potential margin is expected

× : Potential margin is not expected

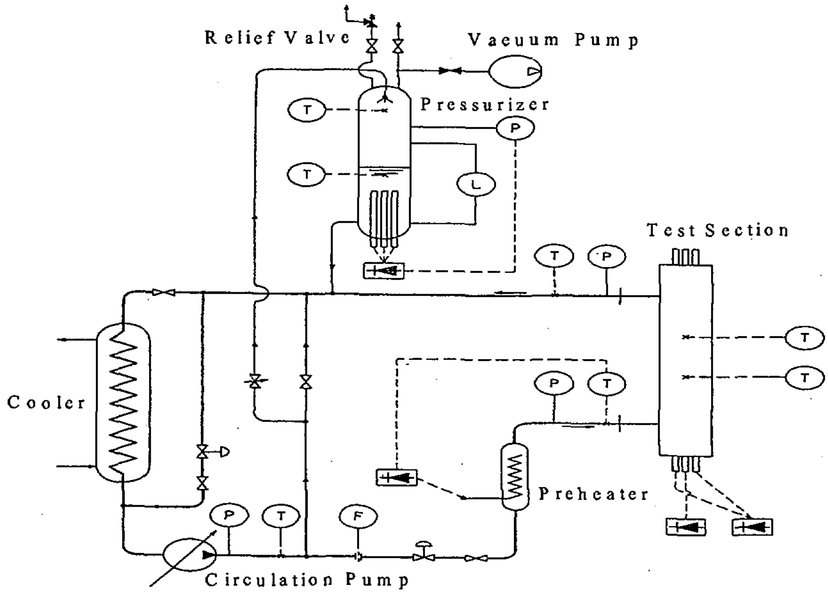

Figure 1 Freon Model Test Apparatus

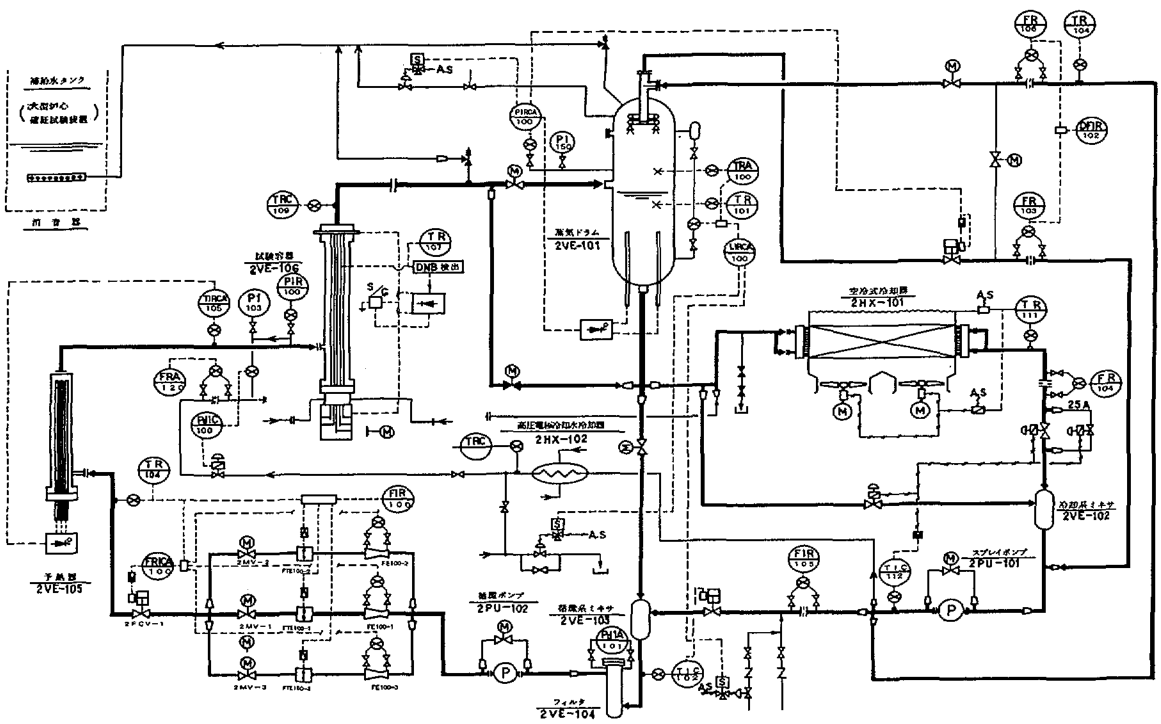

Figure 2 Mockup Water Test System

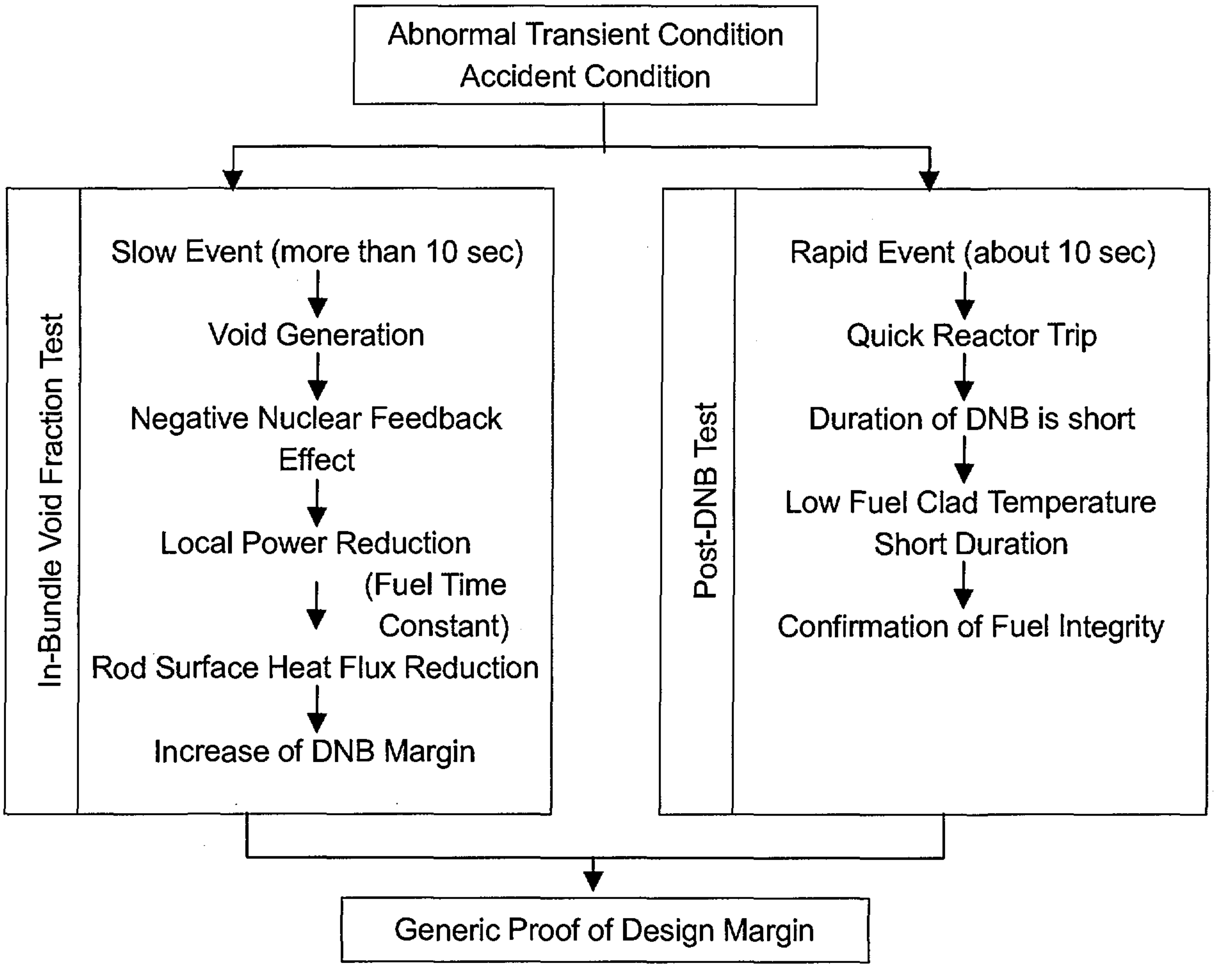

Figure 3 Event Dependency as to Current Design Potential Margin

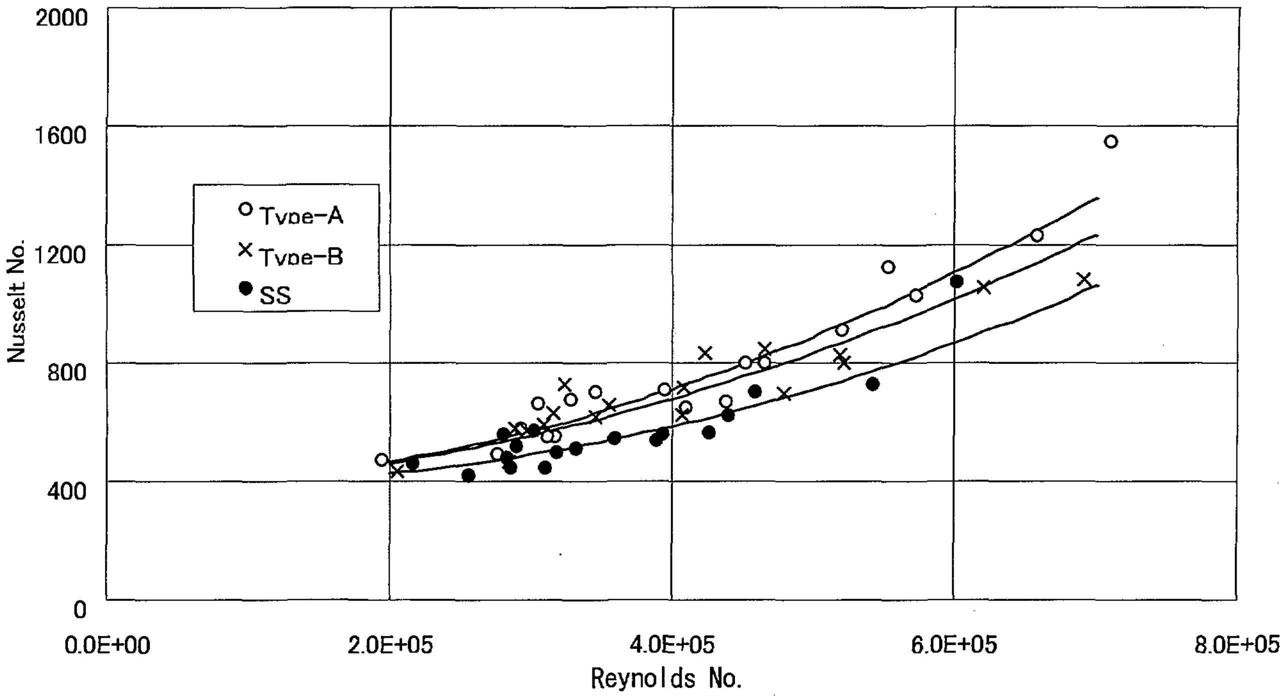

Figure 4 Effect of Mixing Vane
with mixing vane : Type-A & Type-B Grid
without mixing vane : SS Grid

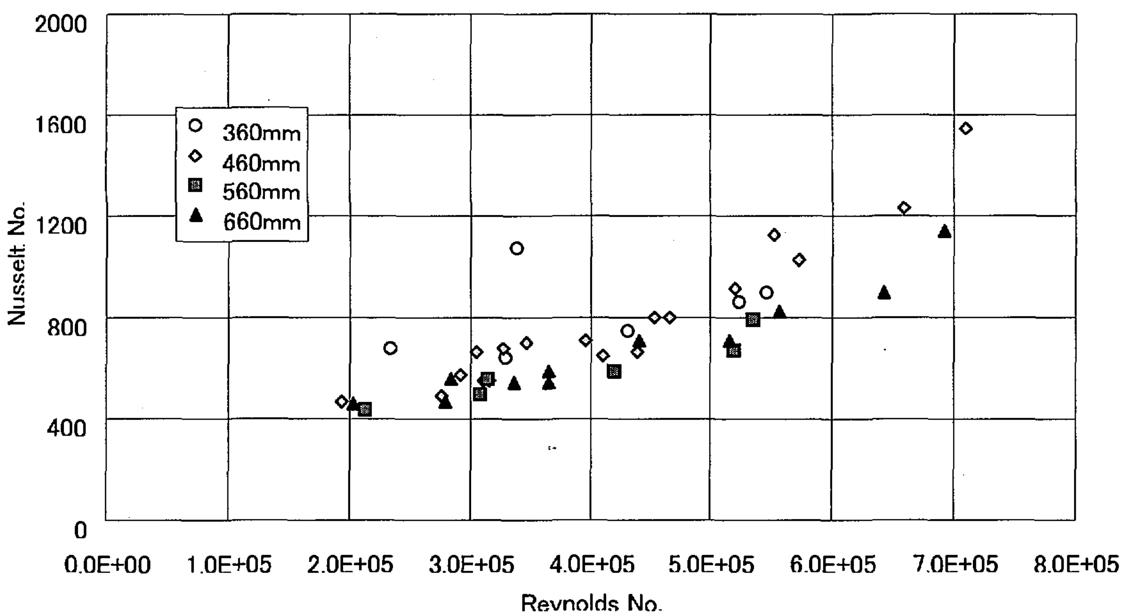

Figure 5 Effect of Grid Span Length (360～660 mm)

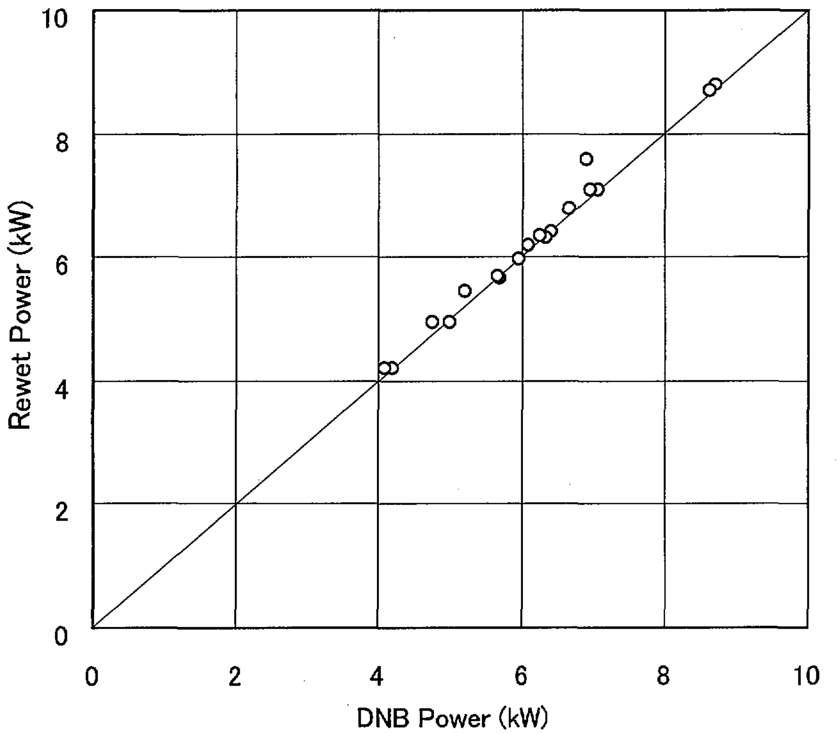

Figure 6 Comparison of DNB Power and Rewet Power

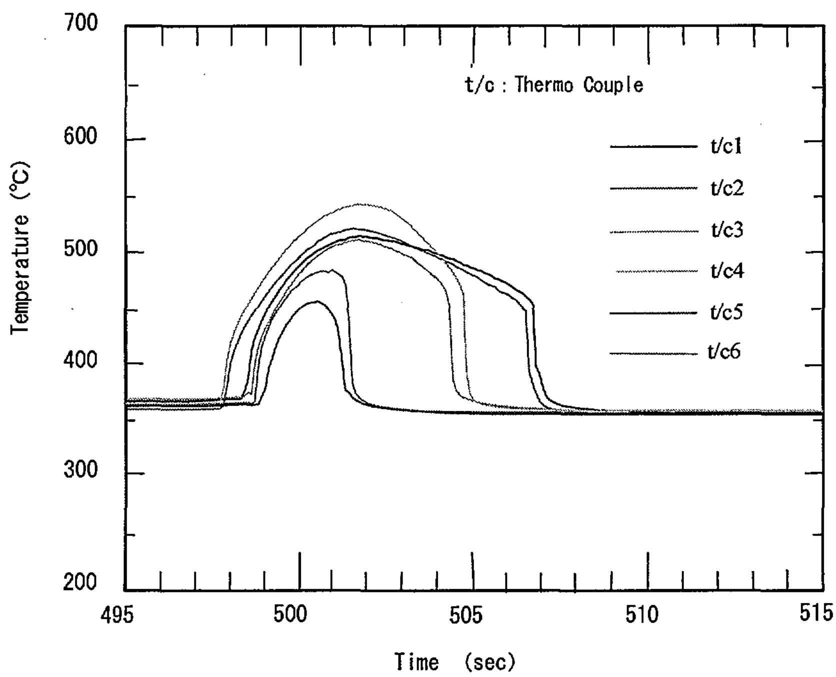

Figure 7 Unsteady-State Power Increase
(Pressure:16.6MPa; Power Ramp Rate:30%/s)

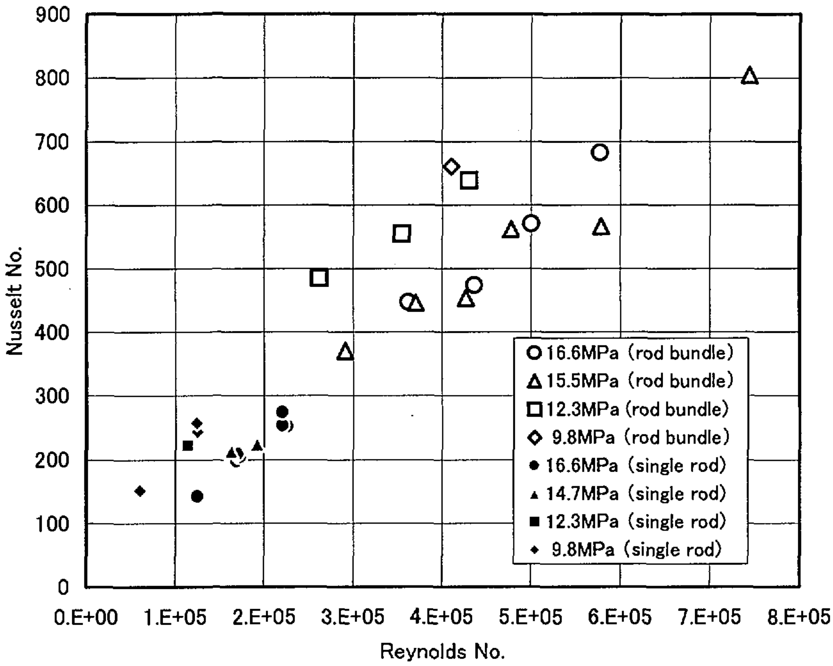

Figure 8 Correlation between Reynolds Number and Nusselt Number