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YGN 3&4 Reactor Flow Model Test

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영광 3,4호기 원자로 유동 모델 시험

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Abstract

Experimental studies were conducted on a 1/5.03 scale reactor flow model of the Yonggwang Nuclear Units 3 and 4. The purpose of the flow model test was to estimate the hydraulic effect in the reactor vessel due to the relative size difference between the ABB–CE's System 80 and the YGN 3&4 reactors. The flow model was designed according to the principle of similarity. Obtained from the test were the core inlet flow distribution, the core exit pressure deviations, and the segmental and overall pressure losses across the flow path from the reactor vessel inlet to outlet nozzle. These data will be used to provide input data for the core thermal margin analysis and to verify the analytical hydraulic design method.

요 약

1/5.03 축소 원자로 모델을 이용하여 원자력 발전소 영광 3,4호기를 위한 유동 시험을 수행하였다. 이 유동 시험의 목적은 ABB-CE사의 System 80과 영광 3,4호기 원자로 크기의 상대적인 차이로 인해 발생하는 원자로 용기내의 수력학적 영향을 평가하는 것이다. 유동 모델은 상사성 원리에 따라 설계하였다. 이 시험에서 얻은 결과는 노심 입구 유량 분포, 노심 출구 압력 분포, 원자로 입구 노즐에서 부터 출구 노즐까지 유동로를 따른 부분 구간및 전체 압력 손실이다. 이데이타들은 노심의 열적 여유도 분석에 필요한 입력 자료 제공과 해석적 수력설계 방법의 검증에 이용하게 된다.

I. Introduction

In the absence of a mathematical model for predicting the hydraulic characteristics in complex geometries, one must revert to experimental techniques. At present, there exists no mathematical model or at least some uncertainties in predicting hydraulic characteristics of the reactor internals, especially for the pressure drops across the complex reactor internals and the flow distributions in the core inlet and exit so that experimental technique should be employed to obtain the hydraulic characteristics and verify the analytical hydraulic design method.

Small scale models are valuable for obtaining quantitative data for use in prototype design where design problems are so complex that analytical method is not available. The selection of a model scale factor is influenced by a number of requirements or restrictions, such as the necessity for maintaining turbulent flow within the model, the capacity of the water supply facility, the model physical size and attendant cost, the effect of the use of normal manufacturing tolerances on the flow area variation, and the relative size of model parts and model instrumentation.

The model should be designed based on the laws of similitude ; strict compliance with the laws of similitude is theoretically necessary to assure that the model is a precise representation of the reactor. Geometric similarity requires that the model and the reactor be the same shape, and that all linear dimensions of the model be related to the corresponding dimensions of the reactor by a constant scale factor. Dynamic similarity between reactor and model is attained when the forces acting on similar volume elements have the same ratio. An investigation into the equations of fluid flow reveals that all requirements for dynamic similarity are met if the reactor and model are geometrically similar and their dimensionless force ratios such as the Reynolds number, Euler number, Froud number and Prandtl number are equal. While all these requirements must be met for strict dynamic similarity on a theoretical basis, in practice, only in rare instances are all of the force ratios significant and it is seldom feasible, economically and technically, to achieve exact geometric and dynamic similarities. Consequently, compromises are necessary in areas where previous experience indicates that violation of the laws of similitude will not seriously impair the value of the test.

It should be noted that every modeling effort is inherently an approximation and should be treated as such, since assumptions for the model may be proved invalid for the prototype and many unexpected difficulties could arise in the actual fabrication, assembling and testing of the model. Therefore, proper interpretation of the model test data is required when applying them to the reactor. The model for the YGN 3&4 reactors was designed to perform the test by maintaining the geometric similitude between model and prototype in the main coolant flow paths with a scale ratio of 1/5.03, except for the core region. Water at room temperature was used as test fluid, which is considered the most feasible and reliable.

The objectives of Yonggwang Nuclear Units 3&4 reactor flow model test were ;

(1) to measure the pressure distributions at the core inlet and exit, and then to determine the core inlet flow distribution to provide input for the thermal margin analysis, and

(2) to measure the segmental and overall pressure losses along the main coolant flow paths in the model to verify the analytical hydraulic design method.

Ⅱ. Model Design

Ⅱ-1. Theoretical Consideration (the Principle of Similarity)

The Buckingham π theorem defines a model as a physical system with characteristics uniquely related to those of the model which accurately predict the characteristics of the prototype. The validity of scale model experiments is based on the supposition that the same physical laws govern the phenomena under consideration in the prototype as in the model. Suppose that these laws are described by a general relationship of the form for the prototype :

$$\Psi(X_1, X_2, X_3, \cdots, X_n)=0 \tag{1}$$

If this equation is unique, then the need for dimensional homogeneity leads to the requirement for an equivalent relation :

$$\Phi(\pi_1, \pi_2, \pi_3, \cdots, \pi_n)=0 \tag{2}$$

among a set of independent dimensionless groups π_i of the original parameters x_i. It is always desirable to attain a model for which the dimensionless groups π_i are equal to those of the prototype.

$\pi_i(\text{Model}) = \pi_i(\text{Prototype})$

For a reactor vessel hydraulic model, the parameters of principal importance are included in the equation.

$$\Psi(P, V, \rho, \nu, L, De, \varepsilon)=0 \quad (3)$$

where

P : system pressure

V : *flow velocity*

ρ : density of test fluid

ν : dynamic viscosity of test fluid

L : characteristic length of flow path

De : hydraulic diameter of flow path

ε : roughness of flow path surface

These seven parameters can be reduced to four dimensionless groups, i.e.,

$$\Phi(L/De, \varepsilon/De, VDe/\nu, \Delta P/(\rho V^2/2))=0 \quad (4)$$

These dimensionless groups are related to the geometry, the relative roughness, the Reynolds number and the Euler number respectively.

Achieving complete similarity between the dimensionless groups of the model and prototype is sometimes impractical. It is seldom feasible to satisfy exact geometric and dynamic similarities, and all the dimensionless groups are important only in rare instances. For example, as long as flow in both the model and reactor is well into the turbulent flow regime where Reynolds number is larger than 5,000 except within the boundary layer, the error resulted from the Reynolds number difference is very small and not significant, regardless of the working fluid being used. The Reynolds number in the outlet nozzle of the YGN 3&4 model is in the order of 10^6, which is lower by the ratio of 1/37 than the corresponding Reynolds number for the actual reactor. But flow in the reactor and model, being well into the turbulent regime, assures satisfactory agreement between the model and the reactor. The relative roughness was considered of importance in the downcomer region. But the pressure loss in this region is very small compared with the total pressure loss in the reactor vessel and so its effect is negligible. Therefore, the Euler number should be similar in the model and the prototype to satisfy the principle of the similarity.

In the investigations reported in references 1 and 2, geometrical similarity between the model and prototype was maintained wherever possible, and a careful attention was given to the effect of scale factors. Large scale factors led to compact models, which were relatively inexpensive to fabricate and test; small flow passages, however, were likely to produce a flow field which has the characteristics different from the one encountered in the prototype. Large models were expensive and required extensive testing facilities. Reactor flow tests with 1/7 scale model[1] and ABB-CE's 1/5 scale model flow tests for 3410 MWt and 3800 MWt class reactors had been successful. Those scaling ratios are generally accepted preferable. Thus it was decided to adopt 1/5 as the scaling factor for the characteristic length for the YGN 3&4, and 1/5.03 was finally settled. The scale factor was the ratio of the cross section dimension

Table 1. Design Data for Prototype and Mdel Reactors

Parameters	Prototype	Model
Folw Rate, Kg/sec	15309	659.9
Water Density, Kg/m^3	701.0	996.2
Inlet Region		
Re(at inlet nozzle)	6.866×10^7	1.719×10^6
E	1.67	1.67
Core		
Re	5.028×10^5	1.268×10^5
E	2.31	2.44
Outlet Region		
Re(at outlet nozzle)	1.120×10^8	2.455×10^6
E	3.58	3.58

* Note : Re–Reynolds Number E–Euler Number

of one model core tube (1.625 inch) to that of the YGN fuel bundle (8.18 inch). Design parameters for the prototype and model reactor are provided in Table 1.

Ⅱ-2. Model Description

The YGN flow model consisted of a vessel with inlet and outlet nozzles, a core support barrel, a

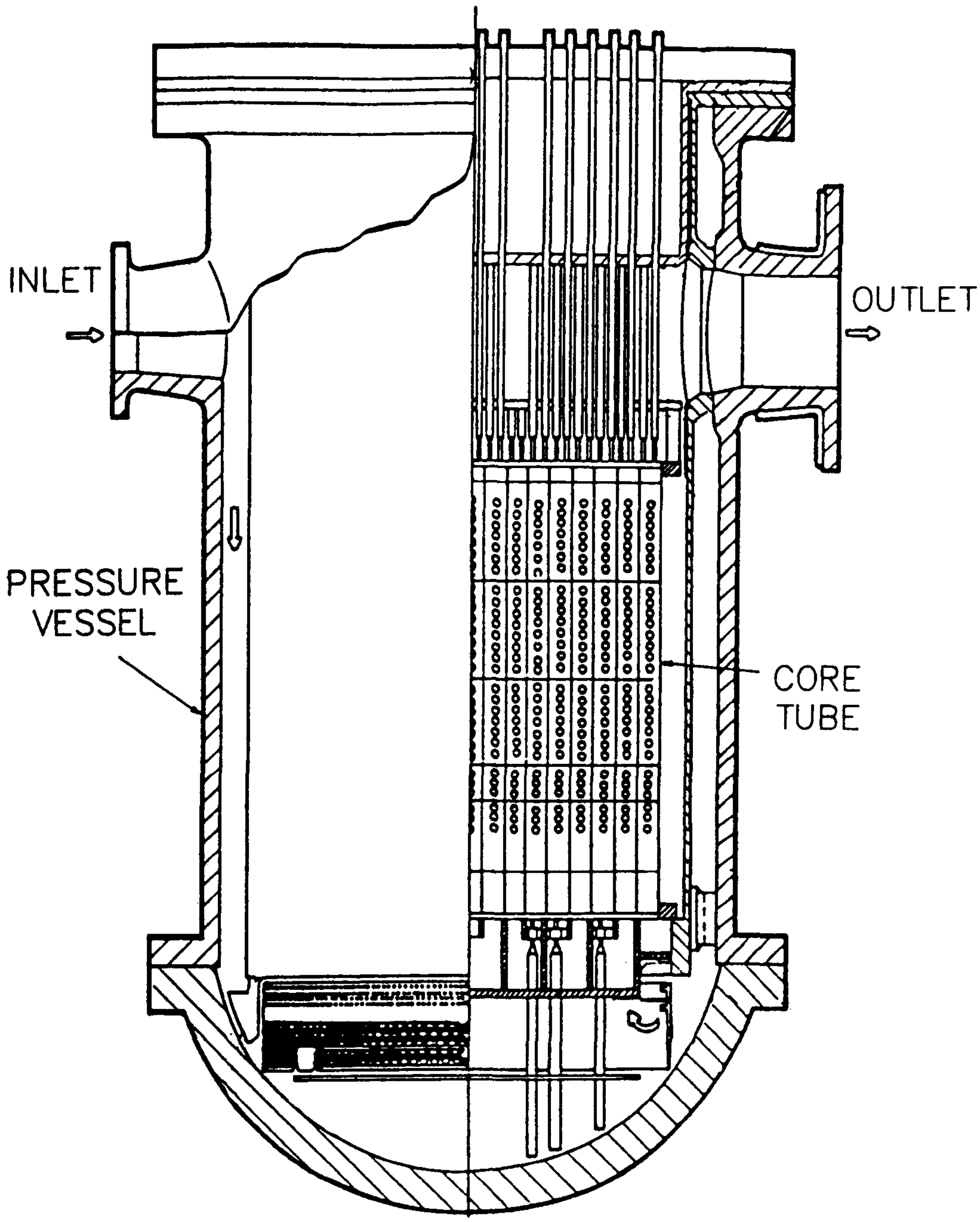

Fig. 1 Reactor Flow Model for YGN 3 and 4

hemispherical bottom head, a flow skirt, a lower support structure, a core simulation and an upper guide structure:see Figure 1. The model was designed with the primary objective of maintaining geometric similitude. All components within the model vessel were scaled geometrically except for the core region. A simplified geometry was used for the model core because the simulation of the fuel rods and spacer grids on the reduced scale is impractical. There was one square tube for each reactor fuel bundle position and each core square tube contained six resistor (orifice) plates to duplicate the axial hydraulic resistance of the reactor fuel assembly. The orifices were used to simulate the reactor core axial flow resistance and to measure the flow rate. Shown in Figure 2 is a comparison between the reactor fuel assembly and the model core tube. The Euler number for axial pressure drop was equal in both the model core

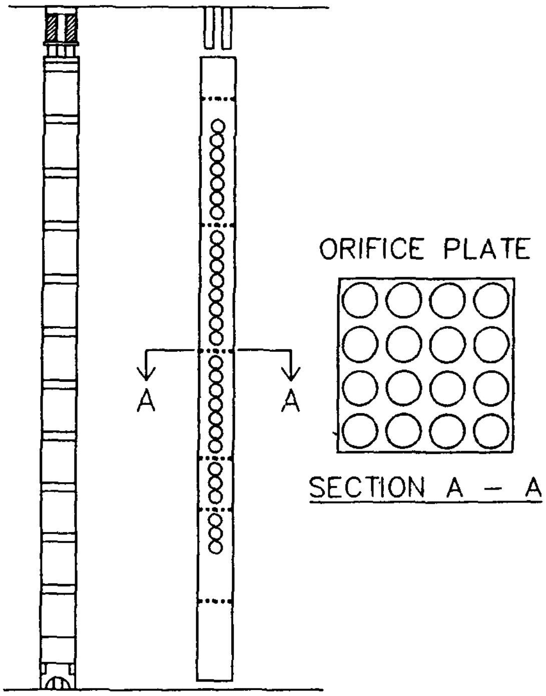

Fig. 2 Reactor and Model Fuel Assembly Layout

and the fuel assembly. The axial resistance of model core was predicted based on the reactor core average pressure loss at full power conditions. Thus, core heating effects were included in the model core design on the core wide average basis. The lateral flow resistance was simulated by a series of holes in the walls of the model core tubes. The resistance to cross flow was experimentally determined, which is created by the narrow gaps between fuel rods. The area of the holes at the walls of the model core tubes was determined based on the requirement that the transverse pressure loss coefficient in the model equal to that in the prototype. Thus dynamic similarity was assumed between the model and prototype.

The flow model was installed in the ABB–CE's large scale hydraulic test facility. The facility had the capacity to circulate up to 15,000 GPM of room temperature water. The facility had a heat exchanger to remove pump heat and to maintain stable temperature. The pressure head for the test loop was maintained by a large diameter tank connected to the pump suction. Flow to the model was controlled and monitored at each inlet by a valve and a flow meter. The flow exiting the model was returned to the suction header through two return lines, each connected to a model outlet. Each return line had a gate valve that was used to adjust the return line resistance and to balance the outlet nozzle flow splits.

II-3. Instrument and Calibration

The model instruments included the installed pressure taps and the associated pressure transmitters to indicate the pressures. Pressure taps were installed at the model inlet nozzles, the downcomer annulus, the core inlet, the core exit and the model outlet nozzles. All the pressure taps were configured as holes in a solid boundary of pipe or core tube. All the model core tubes were instrumented for measurement of flow rates. The orifice plates in the model core tubes were calibrated prior to installing the flow model in the test loop. The calibration loop and procedure were well described in *reference 3*.

The instruments on the test facility were used to determine if the basic test parameteres were properly set for the test. The instruments included the flow meters, the flow meter pressure transmitters and the temperature recording thermocouples. The flow meters were "Gentile" meters that had a venturi type of throat. The flow meters were previously calibrated by an independent laboratory. The thermocouple in the loop was calibration checked prior to testing.

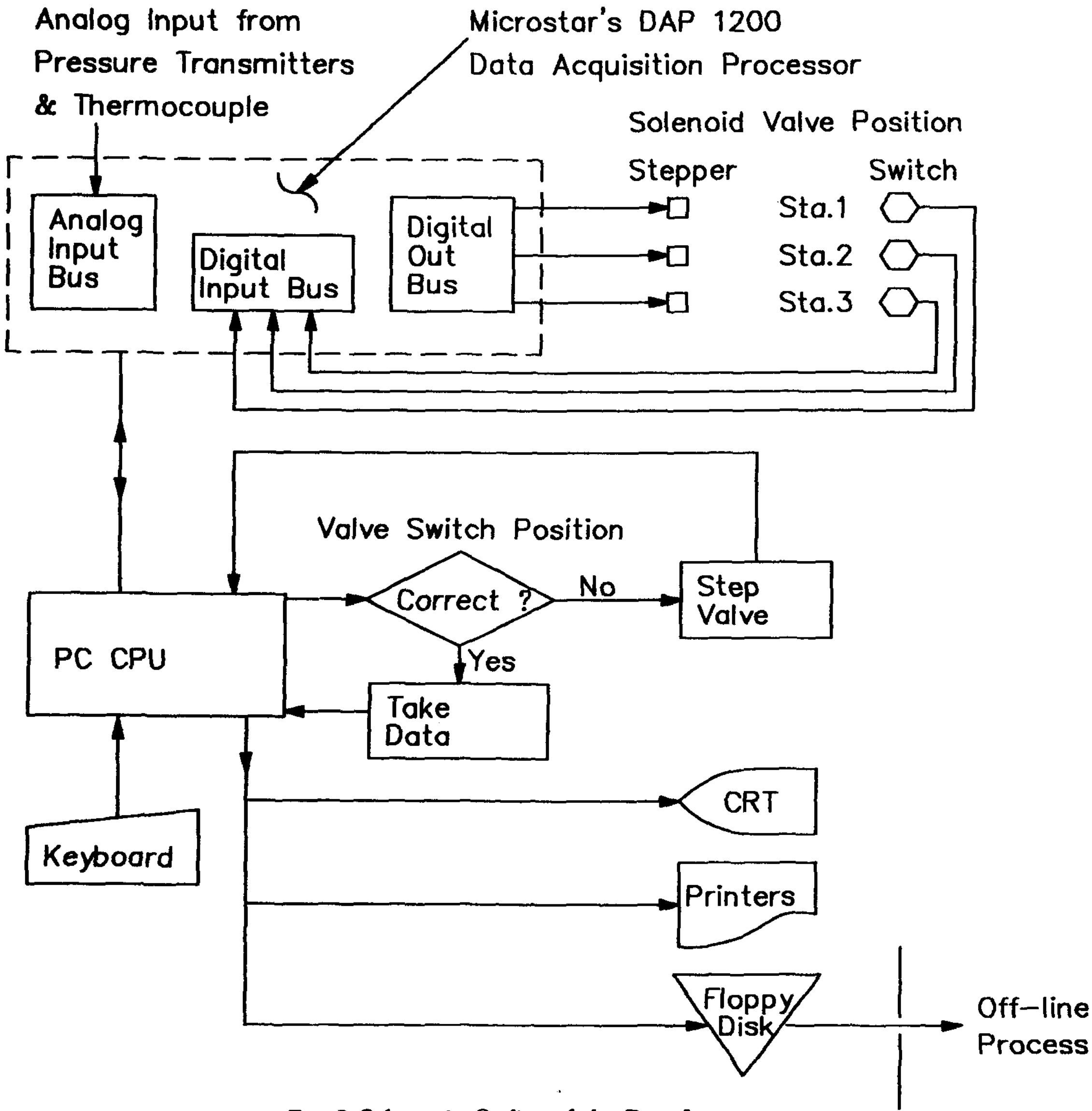

Fig. 3 Schematic Outline of the Data Logger

Ⅱ-4. Data Acquisition System

The test data logger system consisted of analog--to-digital input channels, digital input channels, digital pulsed output channels, CPU (personal computer), a CRT display and printers. The CPU was a personal computer that had been expanded with a data acquisition board. The data acquisition board added the ability to communicate with instruments and controls. Figure 3 is schematic outline of the data logger.

The software directed the actuation of the solenoid valve system and performed the functions of obtaining transmitter performance data, of analyzing the performance data (to generate equations to convert input instrument voltages into engineering units), of controlling the selection of solenoid valves to open, of logging the identification of valves that were instructed to open, of recording instrumentation voltages to disk storage, and of printing a temporary record of the data.

Ⅲ. Test Requirements and Procedures

The requirements of this test were to measure the pressure field at the core inlet and exit, segmental and overall pressure losses from the reactor vessel inlet to outlet, and to determine the core inlet flow distribution. As per the test requirements, three series of tests were conducted. The first series was for the 4-pump balanced operation ; the second series for the 4-pump unbalanced operation and the third series for the 3-pump operation. The term, "balanced", here refers to equal flow rates in each of the cold leg pipes. The criterion for determining the acceptability of the test data was the degree of the test data scatter for key measured parameters. The maximum allowable bands on data scatter for the various measured parameters are given in Table 2.

The prerequisites for making the formal test runs consisted of the following activities. The reactor model was installed in the test loop and all instrumentations were hooked up. The flow model, piping and instrumentation were leak checked. All instrumentation lines were checked to verify correct hookup and identification. The data acquisition system was also verified as being operational. Pretest calibrations of all instrumentation were made and shown to be acceptable. Each formal test run was performed according to the determined test steps. The 4-pump balanced flow tests were performed first and the unbalanced 4-pump tests next. Then the test loop was modified to perform the 3-pump simulation tests. Modifications to the loop piping arrangement were re-

Table 2. Maximum Allowable Data Scatter

Measured Parameter	Allowable Scatter(1 σ)
Core inlet pressure distribution	$s_i=2.0\%$ *
Core exit pressure distribution	$s_i=2.0\%$ *
Flow path pressure loss	$s_i=4.0\%$ **
Flow rate balance	$\varepsilon_i=\pm2.0\%$ ***

$$* \quad s_i=100\left[\frac{1}{N-1}\sum_{j=1}^{N}(E_{i,j}-E_i)^2\right]^{1/2}$$

$$** \quad s=100\left[\frac{1}{N-1}\sum_{j=1}^{N}\left(\frac{K_j-K}{K}\right)^2\right]^{1/2}$$

$$*** \quad \varepsilon=\left(\sum_{k=1}^{4}W_k-\sum_{l=1}^{2}W_l\right)/\sum_{k=1}^{4}W_k$$

where i : Core Tube Number
j : Sequential Number of Test Run
E : Euler Number
K : Pressure Loss Coefficient
N : Total Number of Tset Run
W : Flow Rate
k : Inlet Nozzle Number
l : Outlet Nozzle Number

quired to simulate the reverse flow in the inlet pipe with the non-operating pump.

Ⅳ. Data Reduction and Results

The test data, recorded during each test run, were the flow rates of 4 individual inlet nozzles, flow rates of 2 individual outlet nozzles, loop temperature, pressures at the selected vessel positions along the vessel internal flow path, and the core inlet and exit pressures of the 177 core tubes. During the YGN 3&4 reactor flow model test, 19 test runs were done for 4-pump, with 13 balanced and 6 unbalanced flow conditions.

To determine the impact of inlet nozzle flow imbalance on the core inlet pressure distribution, a statistical t-test was employed. The purpose of the t-test was to determine if the null hypothesis could be supported which says that the balanced and the unbalanced 4-pump data come from the same population (i.e., the core tubes have the same mean flow rate value for the balanced and unbalanced inlet flow rates). It was concluded that all 19 runs could be combined together to determine a final 4-pump inlet pressure distribution. The same conclusion was reached with the outlet pressure data. Next, a statistical gross error analysis was used to determine if any data which appeared too large or too small could be rejected from the data sample on a statistical basis. Data points were rejected if the probability is less than 5% that data from the same normal population would include a reading so remote from the rest of data.

To check the acceptability of the test data, the uncertainty analysis of the test data was performed. The instrumentation error analysis was based on Kline and McClintock method[6] The uncertainty was calculated by the following equation

$$\sigma Y=((\partial Y/\partial x_1)^2\ (\sigma x_1)^2+(\partial Y/\partial x_2)^2\ (\sigma x_2)^2 +\cdots+(\partial Y/\partial x_n)^2\ (\sigma x_n)^2\ )^{1/2} \tag{5}$$

where Y : measured variable

x_1, x_2, $\cdots$, x_n : independent variables

An another approach in estimating the uncertainty was to calculate the standard deviation based upon the measured variable. We compared these two uncertainty values and chose conservative one as the final uncertainty value.

The measured core inlet and exit pressure distributions were expressed in dimensionless Euler number form giving local deviations from the average planar field pressure as a fraction of the average core pressure loss.

$$E_{IN}{}^{i}=\frac{P_{IN}{}^{i}-P_{IN,AVG}}{P_{IN,AVG}-P_{OUT,AVG}} \tag{6}$$

$$E_{OUT}{}^{i}=\frac{P_{OUT}{}^{i}-P_{OUT,AVG}}{P_{IN,AVG}-P_{OUT,AVG}} \tag{7}$$

Provided in Figure 4, as an example, is the pressure distribution at core exit, expressed in Euler nuber, for the 4-pump operation. Using the pressure distributions at core inlet and exit, core inlet flow distribution was calculated and normalized. Figures 5 and 6 show the normalized reactor core inlet flow distribution and its uncertainty map for 4-pump operation. The measured pressures at several locations along the flow path from reactor vessel inlet to outlet were converted to the non-dimensional coefficients which represent the nozzle-to-nozzle pressure losses for reactor stations. The pressure loss coefficients for the 4-pump operating condition is provided in Table 3. A detailed design feedback process by the flow model test is well depicted in Figure 7

Ⅴ. Conclusions

The purposes of the YGN reactor flow model test are ;

(1) to provide the input data for the thermal margin analysis in the core, and

(2) to verify the analytical hydraulic design

(Top view)

Outlet 1

Inlet 1A

Inlet 1B

					.012	.031	.017	−.006	.012					
			−.007	.013	−.008	.003	.014	.017	.010	.019	.006			
		.033	.013	.003	−.003	.007	.016	.010	.007	.002	.007	.012		
	.020	.021	.012	.001	.006	.012	.005	−.001	.014	−.006	.011	.008	.005	
	−.016	.018	.007	−.004	.009	−.013	.000	−.003	−.001	−.002	.007	.002	.001	
.009	−.001	.008	.005	.000	.005	.000	−.008	.002	.003	.005	.001	.008	.001	.010
.025	.014	.015	.004	.006	−.009	−.009	−.007	−.016	−.028	.000	.001	.004	.010	.013
.020	.007	−.002	.003	−.001	−.007	−.013	−.005	−.016	.000	−.003	.001	−.001	.007	.001
.030	.003	.005	.009	−.007	−.003	−.019	−.017	−.012	−.008	−.009	−.004	−.002	.006	.001
.013	−.008	.000	−.002	−.005	−.011	−.018	−.011	−.014	−.010	−.011	.003	−.011	−.005	.010
	.001	−.003	−.012	−.011	−.013	−.003	−.007	−.009	−.012	.001	−.007	−.001	−.008	
	.025	−.012	−.026	−.006	−.006	−.016	−.008	−.011	−.004	−.009	.003	−.002	.001	
		.008	−.001	−.001	−.015	−.024	−.016	−.008	−.018	.001	−.016	.010		
			.004	.000	.003	−.015	−.010	−.014	−.020	.002	−.014			
					−.001	−.001	−.007	−.003	.003					

Inlet 2B

Inlet 2A

Outlet 2

Fing. 4 4–Pump Core Exit Pressure Distribution(E^{i}_{OUT})

(Top view)

Outlet 1

Inlet 1A

Inlet 1B

					0.923	0.962	1.079	1.034	0.929					
			0.913	1.013	1.097	0.969	1.186	1.136	0.804	1.042	0.814			
		1.000	1.000	0.944	0.986	0.920	0.914	0.922	0.932	0.971	0.856	1.128		
	0.997	1.054	1.138	0.959	1.177	1.021	1.012	0.997	1.022	1.106	1.070	1.111	0.835	
	1.000	0.950	1.032	0.975	0.981	0.882	1.008	0.973	0.899	1.020	0.946	1.136	0.918	
0.995	1.115	0.959	1.023	1.005	1.098	0.944	0.946	0.992	0.986	1.001	0.972	1.133	1.081	1.011
1.050	1.111	0.972	1.018	0.938	0.980	0.844	1.007	0.949	0.978	1.043	0.901	0.890	0.997	1.061
1.013	1.080	0.877	0.946	0.914	0.939	0.951	0.992	0.950	0.940	0.963	0.911	0.911	1.046	0.967
1.051	1.015	0.845	0.966	0.978	1.056	0.955	0.961	0.979	1.130	1.034	1.006	1.025	1.111	1.055
1.005	1.071	0.990	1.028	0.942	1.090	0.923	1.008	0.995	0.962	1.058	0.953	1.002	1.099	1.027
	0.928	1.078	1.045	0.915	1.002	0.927	0.989	0.983	0.939	0.982	1.002	0.967	0.992	
	0.931	1.068	1.182	1.019	1.133	0.947	0.931	1.039	1.100	1.057	1.130	1.026	0.967	
		1.130	1.047	0.917	1.040	0.910	0.903	0.940	0.977	0.893	0.935	1.219		
			0.865	1.000	1.018	1.123	1.135	1.133	1.106	1.022	0.957			
					0.962	1.007	1.067	1.005	0.956					

Inlet 2B

Inlet 2A

Outlet 2

Fig. 5 4–Pump Core Inlet Flow Distribution
$(W_i/W_{avg})_{inlet}$

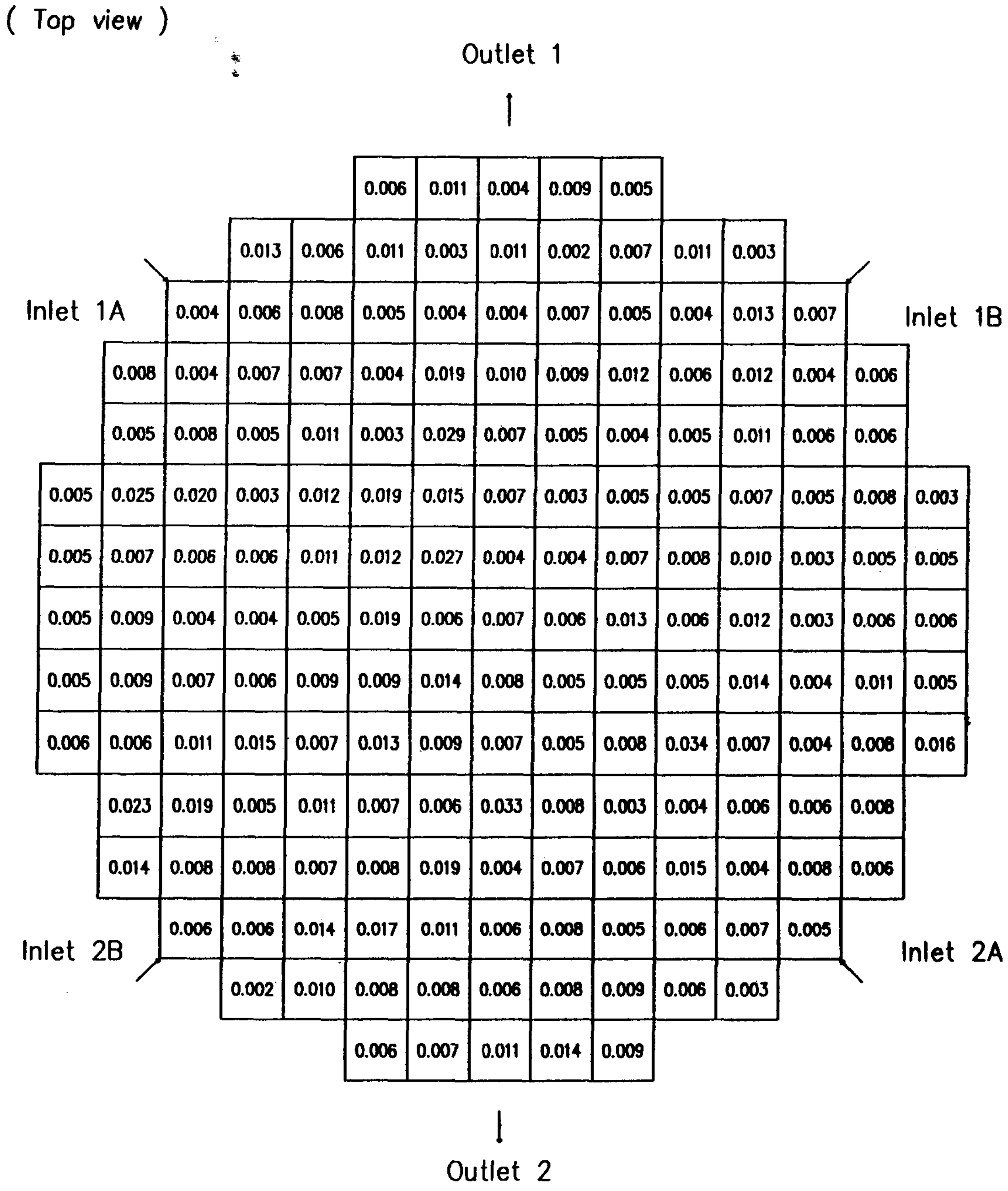

Fig. 6 Core Inlet Flow Standard Deviation for 4–Pump

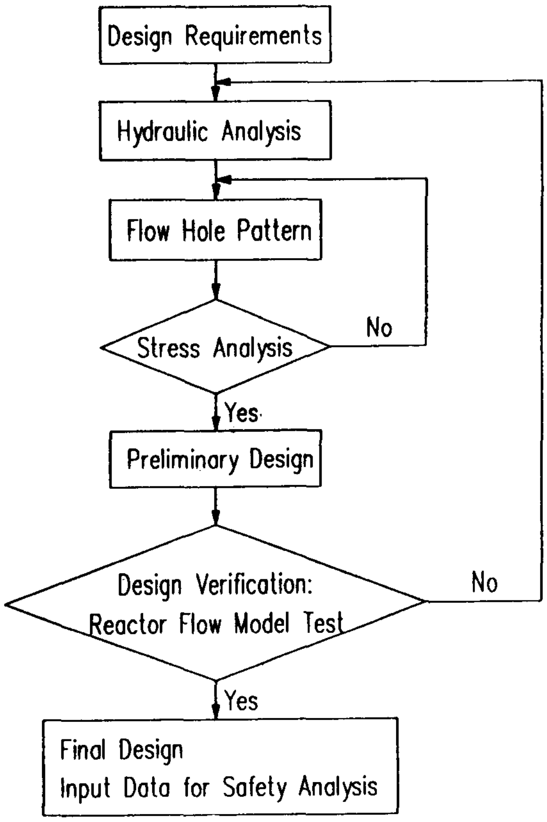

Fig. 7 Design Feedback Process for Reactor Internals

Table 3. Segmental and Overall Pressure Loss Coefficients for 4-Pump Operating Condition

Vessel flow path segment	Pressure Loss Coefficient, K_{i-j}*
Inlet region	2.59
Core	2.31
Outlet region	2.25
Overall	7.15

$$* \; K_{i-j} = \frac{P_i - P_j}{\text{Velocity Head at Inlet Nozzle}}$$

where P : Total Pressure
i,j : Flow Path Station

method.

Core inlet flow and core exit pressure distributions were fairly uniform over the entire core region, similar to those for ABB-CE's System 80 reactors. The measured data used in producing the finally reduced data met the acceptance criteria and, therefore, test results are suitable for intended use in performance and safety analyses. Design values from the analytical hydraulic method were proved to be practically the same as the model test results.

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