Journal of the Korean Nuclear Society
Volume 32, Number 2, pp.108~127, April 2000

An Experimental Study on the Mass and Energy Release for a Hot Leg Break LBLOCA During Post Blowdown

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(Received May 4, 1999)

Abstract

Hot leg break LBLOCA(Large Break LOCA) had a potential to be a containment maximum pressure accident in YGN3&4, which was induced from excessive conservatism in the CE analysis methodology of mass and energy release. This study conducted mass and energy release experiment for the hot leg break LBLOCA during post blowdown with an integral test facility, SNUF(Seoul National University Facility). This facility simulated YGN 3&4 with volume ratio of 1/1140 based on Ishii' s three level scaling. Experiment showed that SI(Safety Injection) water refilled cold leg first and core later. SI water was vaporized in the core, which resulted in the repressurization of reactor. This increase of pressure drove the water in cold leg to flow up half height of U tubes. However, since the water was drained back soon, the release through the SG side broken section by evaporation was negligibly small. This study also provided experimental assessment of RELAP5 results by KAERI for the release through the SG side broken section.

Key Words : LOCA, hot leg, post blowdown, release, experiment, scaling

1. Introduction

Containment maximum pressure accident was differently reported in PSAR(Preliminary Safety Analysis Report) and FSAR(Final Safety Analysis Report) of YGN3&4(Young Kwang Nuclear Power Plant Unit 3&4) which is the base of Korean Standard Nuclear Power Plant[1]. DEDLSB (Double Ended Discharge Leg Slot Break) was reported to be such an accident in PSAR, while DEHLSB(Double Ended Hot Leg Slot Break) in FSAR. This change was caused by the limitation of CE design code for the hot leg break analysis. As shown in Table 1, in PSAR, DEDLSB was analyzed by CEFLASH-4A and FLOOD3 codes in blowdown and post blowdown period, respectively. However, for DEHLSB, only blowdown period was analyzed by CEFLASH-4A code because of the limitation of FLOOD3 code for hot leg break analysis, which consists of hydraulic circuits applicable to only the cold leg break analysis. Thus, in FSAR, the CONTRANS code was applied to estimate the mass and energy release during post blowdown for DEHLSB, which

Table 1. Comparison of Mass/Energy Release Analysis Methodology in PSAR and FSAR for YGN3&4

PSAR				FSAR			
DEDLSB		DEHLSB		DEDLSB		DEHLSB	
Blowdown	Post Blowdown	Blowdown	Post Blowdown	Blowdown	Post Blowdown	Blowdown	Post Blowdown
CEFLASH-4A	FLOOD3	CEFLASH-4A	-	CEFLASH-4A	FLOOD3	CEFLASH-4A	CONTRANS

is based on Fort Calhoun methodology[1].

However, it has been criticized that the BOIL-OFF model in the CONTRANS code, where whole heat in SGs(Steam Generators) contributes as a heat source, is excessively conservative and thus results in unrealistically higher containment pressure. In this code, after reflood phase, all decay heat and metal heat make the SI water saturated steam, and this steam passes through the SG, which makes the steam thermal equilibrium with SG secondary side. Moreover, all coolant in reactor vessel is assumed saturated water. In addition to the methodology difference, another consideration was attributed by the specific design characteristics of YGN3&4 having larger 1 hot leg and smaller 2 cold legs in each loop.

KINS(Korea Institute of Nuclear Safety) reviewed the mass and energy release analysis methodology and pointed out the excessive conservatism of this methodology[1]. KAERI(Korea Atomic Energy Research Institute) has also implemented the analysis of LBLOCA using RELAP5 for YGN3&4 and concluded that there was no physical mechanism to support the treatment of SGs as a heat source after EOB(end of blowdown) and that more systematic RELAP5 analysis and careful experiments could release the excessive conservation in mass and energy release methodology[3].

In world-wide studies, in spite of many analytical and experimental studies on cold leg break LOCA, there have been rare studies on the hot leg break LOCA and especially on the difference between the two accidents. Also there have been limited studies about the mass and energy release analysis even for cold leg break.

In this study, experiments were conducted with integral test facility, SNUF, in order to understand the detailed phenomena in hot leg break LBLOCA during post blowdown phase and to assess the RELAP5 analysis by KAERI for such accident which had showed the excessive conservatism in mass and energy release analysis. The SNUF was set to simulate YGN3&4 and scaled down with volume ratio 1/1140, based on Ishii' s 3 level scaling. The broken hot leg was designed to have two broken sections, which represented a reactor side broken section and a SG side broken section. The mass releases were measured separately in both sides during the post blowdown period because the phenomenological difference of hot leg and cold leg break LOCA would apparently exhibit after EOB.

Through the series of experiments, the detailed release phenomena during post blowdown in hot leg break LBLOCA were observed and the role of SGs was especially examined. The sensitivity studies for SI conditions were also conducted. More realistic heat source mechanism of SG was discussed rather than direct comparison of experimental results with excessively simplified BOIL-OFF model. The more realistic heat source mechanism of SG could be considered as followings; (1) a case that the SI water directly passes through the U tubes, (2) a case that the SI water flows up to some height of U tubes and is evaporated by the heat of SG. Thus, the coolant behavior in cold leg suction part was carefully investigated. The results were also discussed to assess the RELAP5 analysis of YGN3&4 conducted by KAERI for some important parameters using scaling analysis.

2. Description of LOCA Phases

In postulated LBLOCA, coolant release continues over several phases like as blowdown and post blowdown phase(refill, reflood and post reflood phase). The blowdown period extends from time zero until the primary system is essentially depressurized to containment pressure. The refill phase is the duration that the emergency core cooling system(ECCS) refills from the bottom of the reactor vessel to the bottom of the core. The reflood phase is the period that the ECCS water refloods the core. It is ended when the liquid level in the core is 2 feet below the top of the active core. The post reflood phase is the residual period of the accidents. In this phase, the dominant process is the continued cooling of the SGs by the ECCS water leaving the core.

During blowdown phase, most of the initial primary coolant is released to containment. However, during post blowdown, the evaporation of SI water can contribute to the containment pressure rise and thus generate the second peak pressure, which will be the criteria of containment design pressure in most nuclear plants. During that phase, the SGs heat removal has different characteristics depending on break locations. For cold leg break, the steam and entrained water carried out of the core pass through the SG, where the entrained water is evaporated or can be superheated to nearly the temperature of the SG secondary system temperature, while for hot leg break, majority of coolant from the core is expected to be directly discharged to the containment without passing the SG since the broken leg provides a direct release path[4].

3. Test Facility and Measurement Instruments

3.1. Scaling and Design of Test Facility

In order to establish the experimental facility, SNUF, Ishii' s scaling law was applied. Ishii' s scaling law was recently known as three level scaling, which is a kind of top-down and bottom-up approach concept; global scaling, boundary flow scaling and local phenomena scaling[5-11].

Prior to the discussion of scaling method, initial conditions of experiment should be considered because the experiments in this study are simulated to start from EOB. The EOB conditions of RELAP5 analysis for YGN3&4 were used as reference initial conditions of experiments. The RELAP5 is generally known to well predict the thermo-hydraulic behaviors in blowdown phase. The RELAP5 analysis did not show such EOB point when the primary pressure became same with the containment pressure[3]. Thus, in this study the EOB was assumed to be about 15 seconds elapsed from the initiation of accident. At this time, the primary pressure was decreased close to containment pressure and the core void fraction was nearly 100%. Resultantly, the collapsed liquid level of the reactor fell to minimum. The comparison of EOB conditions for important parameters is presented in Table 2.

The scaling analysis in this study focuses on the refill and reflood phase in the view point of mass and energy release. At first, in the global scaling(first level), the governing equations are

Table 2. Scaling of Initial Conditions

	SNUF	YGN3&4	Ratio
Core Pressure [MPa]	0.35	0.5	$p_R = 0.7$
Core Temperature	Saturated	Saturated	$T_{sat_R} = \dfrac{T_{sat}(p_m)}{T_{sat}(P_p)} = 0.9695$
Mean Power [kW]	60.0	60.0E3	$N_{pchR} = 1$ + Energy Inventory Scaling
Core Exit Void Fraction	100%	100%	$N_{d_i} \equiv 1$
Liquid Rise Velocity [m/s]	0.01111	0.276	$u_{0R} = 4.025E\text{-}2$

nondimensionalized with following geometric scaling factors;

$$\begin{aligned} &\textit{Axial Length Scaling}: L_i \equiv l_i/l_0 \\ &\textit{Flow Area Scaling}: A_i \equiv a_i/a_0 \end{aligned} \tag{1}$$

For SNUF, the axial length scaling ratio and the flow area scaling ratio were set as 1/6.4 and 1/178.4, respectively. Thus, the volume scaling ratio was 1/1144.5.

The geometric scaling criterion requires following relations to be satisfied for all components of system;

$$\begin{aligned} A_{iR} &= (a_i / a_0)_R = 1 \\ L_{iR} &= (l_i / l_0)_R = 1 \end{aligned} \tag{2}$$

Based on these requirements of Eq. (2), the important design parameters of SNUF are obtained as shown in Table 3 for SNUF and YGN3&4.

The nondimensionalization of the governing equations produces the important dimensionless numbers such as kinematic similarity parameters, dynamic similarity parameters, and energetic similarity parameters;

Phase Change No. :

$$N_{pch} \equiv \left[\frac{4\dddot{q}_0\,\delta l_0}{d u_0 \rho_f i_{fg}}\right]\left[\frac{\Delta\rho}{\rho_g}\right] = N_{Zu} \tag{3}$$

Subcooling No. :

$$N_{sub} \equiv \left[\frac{\Delta i_{sub}}{i_{fg}}\right]\left[\frac{\Delta\rho}{\rho_g}\right] \tag{4}$$

Froude No. :

$$N_{Fr} \equiv \left[\frac{u_0^2}{g l_0 a_0}\right]\left[\frac{\rho_f}{\Delta\rho}\right] \tag{5}$$

Drift-Flux No. :

$$N_{di} \equiv \left[\frac{V_{gi}}{u_0}\right]_i \tag{6}$$

$$(\textit{or Void-Quality Relation})$$

Time Ratio No. :

$$T_i^* \equiv \left[\frac{l_0/u_0}{\delta^2/\alpha_s}\right]_i \tag{7}$$

Thermal Inertia Ratio No. :

$$N_{thi} \equiv \left[\frac{\rho_s c_{ps}\delta}{\rho_f c_{pf} d}\right]_i \tag{8}$$

Friction No. :

$$N_{fi} \equiv \left[\frac{fl}{d}\right]_i \left[\frac{1+x(\Delta\rho/\rho_g)}{(1+x\Delta\mu/\mu_g)^{0.25}}\right]\left[\frac{a_0}{a_i}\right]^2 \tag{9}$$

Orifice No. :

$$N_{oi} \equiv K_i[1+x^{3/2}(\Delta\rho/\rho_g)]\left[\frac{a_0}{a_i}\right]^2 \tag{10}$$

Here, the reference velocity, u_0, is the mean rising velocity of core liquid level, which seems to be very reasonable because the core liquid level is

Table 3. Design Parameters and Scaling Ratio of SNUF

		SNUF	YGN3&4	Scaling Ratio
Vessel				
Height	[mm]			l_R =1/6.4
Flow Area	[mm^2]	1975	12.7E3	a_R =1/178.4=(1/13.4)^2
Volume		74748	13.3E6	V_R = 1/1144.5
Hot Leg				
Flow Length	[mm]	683	4.38E3	l_R =1/6.4
Flow Area	[mm^2]	4990	0.89E6	a_R =1/178.4
Suction Leg				
Flow Length	[mm]	1172	7.52E3	l_R =1/6.4
Flow Area	[mm^2]	2579	0.46E6	a_R =1/178.4
Discharge Leg				
Flow Length	[mm]	916	5.88E3	l_R =1/6.4
Flow Area	[mm^2]	2579	0.46E6	a_R =1/178.4
Fuel Hydraulic Diameter	[mm]	14.167	3.0	d_R =4.7223
Fuel Conduction Depth	[mm]	2.5	2.425	δ_R =1.0309

the most important factor that governs the system behaviors during LBLOCA post blowdown. This fact is identified in experimental results.

For the sake of perfect similarity, all of parameters in Eqs. (1) ~ (10) should satisfy the following requirement;

$$\Psi_R \equiv \frac{\Psi_m}{\Psi_p} = \frac{\Psi_{for\ model}}{\Psi_{for\ prototype}} = 1 \tag{11}$$

In designing the experimental facility, however, it is not only impossible but also ineffective to meet the required condition of Eq. (11) for all parameters in the system, depending on the purpose of the experiment. Thus, the following considerations were taken.

Under the same fluid property conditions, the similarity of Froude number in Eq. (5) leads to the following velocity-length relation;

$$(u_0)_R = (l_0)_R^{1/2} \tag{12}$$

Referring the RELAP5 analysis for YGN3&4 [3], this requirement produces too fast model velocity and thus the model core would be refilled within a few seconds, which makes detailed system observation almost impossible. By the way, the Froude number, generally, is a dimensionless number that reflects the effect of gravitational force and free surface behaviors. The gravitational force and free surface behaviors do not play an important role in this experiment because the system behavior is mainly governed by the steam generation in core. And, although the Froude number is a parameter that decides the core two phase flow regime[12], the dependency of overall system behaviors on Froude number was revealed not much, as shown in sensitivity studies in this papers and reference 14. Moreover, the COMET analysis shows little dependency of mass and energy release on CRF(Carryover Rate Fraction)[3]. Therefore, the reference velocity should be decided as the value that can make sure the detailed observation of the system behavior rather than that can satisfy the similarity of Froude number. Resultantly, the selected reference velocity is 0.01111 m/sec, as shown in Table 2.

According to Ishii et al.[5-8], the similarity of drift flux number requires following void relation;

$$(\alpha_e)_R\left[\frac{\Delta\rho}{\rho_f}\right]_R=1 \quad or \quad (\alpha_e)_R\fallingdotseq 1 \qquad (13)$$

The core exit void fractions are same as 100 % in both prototype and model system. And thus this requirement is satisfied.

Friction number and orifice number were considered. At first, the orifice number in Eq. (10) is composed of loss coefficient K_i, fluid properties including quality x and geometrical area ratio. Since K_i is only the function of shape(d_i /d_j), the K_{iR} will be unity when (d_i /d_j)$_R$ goes to unity under the condition of a_{iR} = a_{jR}, which is already satisfied in geometrical similarity condition. Futhermore, as the K_i does not show large variation with the variation of diameter ratio from fluid handbook, small distortion in geometrical similarity requirement does not produce large difference in K_i. And, the other terms in Eq. (10) lead to unity from the condition of the same fluid property and the geometrical similarity. Thus, the orifice number similarity is easily satisfied.

Next, the similarity of friction number under the same fluid property and geometrical similarity leads to following requirement;

$$N_{fiR} = 1$$

Thus,

$$f_R \fallingdotseq \frac{l_R}{d_R}=2.1 \qquad (14)$$

The prototypic Reynolds number and the ratio of Reynolds numbers can be calculated like as (saturated water condition is used in calculation);

$$Re_p = 4.2215\times 10^6$$

$$Re_R = \frac{\rho_R u_{0R} L_R}{\mu_R} \fallingdotseq 2.2891\times 10^{-3} \qquad (15)$$

Thus,

$$Re_m = 9.6634\times 10^3$$

Since the friction factors, f's, are the function of Reynolds number and wall roughness, corresponding friction factors can be obtained from Moody chart as followings(smooth wall condition is used for the sake of simplicity);

$$f_m \fallingdotseq 0.03$$
$$f_p \fallingdotseq 0.01$$

Thus,

$$f_R \fallingdotseq 3.0 \qquad (16)$$

Although this shows small difference comparing with the similarity requirement by Eq. (14), such a small gap is not expected to lead large difference in overall system behavior. Thus, under the same fluid properties and the same steam quality condition, the similarity of friction number is approximately met.

Thermal inertia number in Eq. (8) means the ratio of structure thermal capacity and fluid thermal capacity. This structure thermal capacity can play a role of heat source to the fluid and it will be discussed in second level scaling.

The phase change number in Eq. (3) is the most important dimensionless number because the phase change governs the steam generation in core and pressure behaviors. Such a pressure behavior controls the ECCS water distribution and its behavior. Setting the phase change number ratio as unity leads to

$$[\ \dot{q}_o'''\ ]_R=\left[\frac{du_0}{\delta l_0}\right]_R=1.0825 \qquad (17)$$

Since the ratios of flow area and flow path are 1/178.4 and 1/6.4 respectively, the total heat generation ratio can be calculated as(such a calculation method can be seen in chapter 7 of reference 8);

$$[\dot{q}_0]_R = [\ \dot{q}_0'''\ ]_R V_R = 9.4583\times 10^{-4} \qquad (18)$$

Therefore, since the decay heat generation rate in YGN3&4 is about 60 MW near the EOB, the heat generation required in the experiment is

$$\begin{aligned} \dot{q}_{0m} &= [\dot{q}_0]_R \, \dot{q}_{0p} \\ &= 56.75 \ kW \end{aligned} \tag{19}$$

The subcooling number is automatically satisfied because this number is composed of only fluid properties.

The time ratio number is deduced from the solid energy equation, and thus this counts for the solid temperature[6]. However, the solid temperature is beyond the interest of this study, and the important is the total heat addition to coolant. For the given geometry in Table 3, the time ratio number ratio is evaluated as;

$$T^*_{i\,R} = 3.6528 \tag{20}$$

Finally, the time scale can be determined like following;

$$\tau_R = \frac{l_{0R}}{u_{0R}} = 3.882 \tag{21}$$

This means that the chronology of the model lags about 3.9 times than that of the prototype.

Second level scaling is inventory and boundary flow scaling. In order to meet the mass inventory similarity, following boundary area condition is required[8];

$$\left[\frac{a_{in\ or\ out}}{a_0}\right]_R = (l_{0R})^{1/2} \tag{22}$$

This equation requires the smaller boundary area than system flow area.

There are two boundary flows; SI flow and break flow. Phenomena of SI flow are not the concerns of this study and the SI flow is adjusted appropriately to obtain suitable core liquid rise velocity. This makes the model transient slow and makes possible the detailed observation of system behaviors. The selected core liquid rise velocity, which is the result of SI flow, is presented in Table 2.

Break flow in this study is the case of sudden expansion. For subsonic break flow, following requirement should be satisfied[8];

$$\left[\frac{a_{break}}{a_0}\right]_R = (l_{0R})^{1/2}$$

or

$$\begin{aligned} [a_{break}]_R &= (l_{0R})^{1/2}[a_0]_R \\ &= 0.395[a_0]_R \end{aligned} \tag{23}$$

This requirement is easily acquired by using ball valves in break locations because the valve flow area is smaller than pipe area by less a half times.

In the energy inventory scaling, the following equation is considered;

$$\begin{aligned} \frac{dE}{dt} &= \dot{q} - w \\ &\quad + \sum \dot{m}_{in} i_{in} - \sum \dot{m}_{out} i_{out} \end{aligned} \tag{24}$$

In above equation, the first term in right hand side, $\dot{q}$, is heat added. Noting that the heat is transferred from both fuels and structures to fluid , it can be seen from Eq. (24) that only the total heat, not the heat addition mechanism, is important. Thus, the shortage of heat addition by the structure heat, if ever, can be compensated by electrical heaters. This fact also resolves the thermal inertia number similarity in global scaling. Eventually, the necessary model power can be determined here;

Required Power
= Core Power + Structure Heat
= 56.75 + Structure Heat
= 60 kW

Third level scaling is local phenomena scaling. This is a bottom-up approach concept.

The break simulation was already discussed in the second level scaling and the break size was determined.

The important driving mechanism in mass and energy release experiment is the core heat transfer. Such heat transfer would be governed by the phase change rather than convective heat transfer. Thus, the behaviors induced by the phase change can be simulated by the phase change number scaling, which is already satisfied in first level scaling.

Pressure behavior is determined by scaled inventory of mass and energy in the equation of state. Such an inventory scaling was discussed in second level scaling and the remaining problem is the initial condition. Thus, the initial pressure of system and fluid inventory distribution should be considered. The prototypical pressure at EOB in RELAP5 analysis[3] shows about 0.5 MPa and the model initial pressure was set 0.35 MPa. Such a small gap in two pressures is thought not to lead much difference of system behaviors. Furthermore, the EOB point of experiment has ambiguity and thus the exact EOB point of experiment is thought to be the time after a few seconds from experiment commencement.

Pressure distribution, which is the driving force of fluid movement, was also reviewed. In both systems, the high pressure part is the core and the low pressure part is break location. This pressure difference results in the pressure gradient. Thus, if two systems are geometrically similar, the pressure gradient is also expected to be similar.

The reverse minor loss coefficient of RCP (Reactor Coolant Pump) was reviewed since the prototypic reverse coefficient is very large and SNUF' s is small. However, since the reverse flow in both YGN3&4 and SNUF is extremely small or almost stagnant, the effect of the loss coefficient on pressure drop is very small, noting that the pressure drop is given by;

$$\Delta p = K\frac{1}{2}\rho u^2 \tag{25}$$

Moreover, since the reverse loss coefficient of SNUF is smaller than that of YGN3&4, it is more likely to happen that the coolant in cold leg suction part rises up to U tubes conservatively.

The initial fluid distribution should be scaled. Specially the liquid water distribution, which is potentially evaporated to steam, is important. It was assumed that all liquid water would be discharged through break until EOB and there would be no water in RCS except in reactor lower plenum. Thus, in experiments, all the water in the loop was drained as an initial condition except in reactor lower plenum.

Finally, the conditions of secondary system were review. The secondary pressure at EOB in YGN3&4 is as high as 6 MPa[3]. Thus, the scaling of secondary system is basically impossible in this study. However, fortunately such a scaling is not prerequisite if all discussions on the coolant behaviors in U tubes would be bounded within nothing but hydraulics. Thus, in this study the pressure of secondary system is maintained as high as possible in the state of saturated water.

3.2. Test Facility

Based on the above scaling study, the test facility SNUF was established to simulate YGN3&4 which has one hot leg and two cold legs in broken loop. The flow path length ratio and the flow cross sectional area ratio are 1/6.4 and $(1/13.4)^2$ respectively, and the total volume ratio is 1/1140. Detailed design parameters are summarized in Table 3 and the system configuration is designed like as Figs. 1 and 2.

The SNUF has appropriate components of primary system and secondary system for mass and energy release experiments. Reactor contains

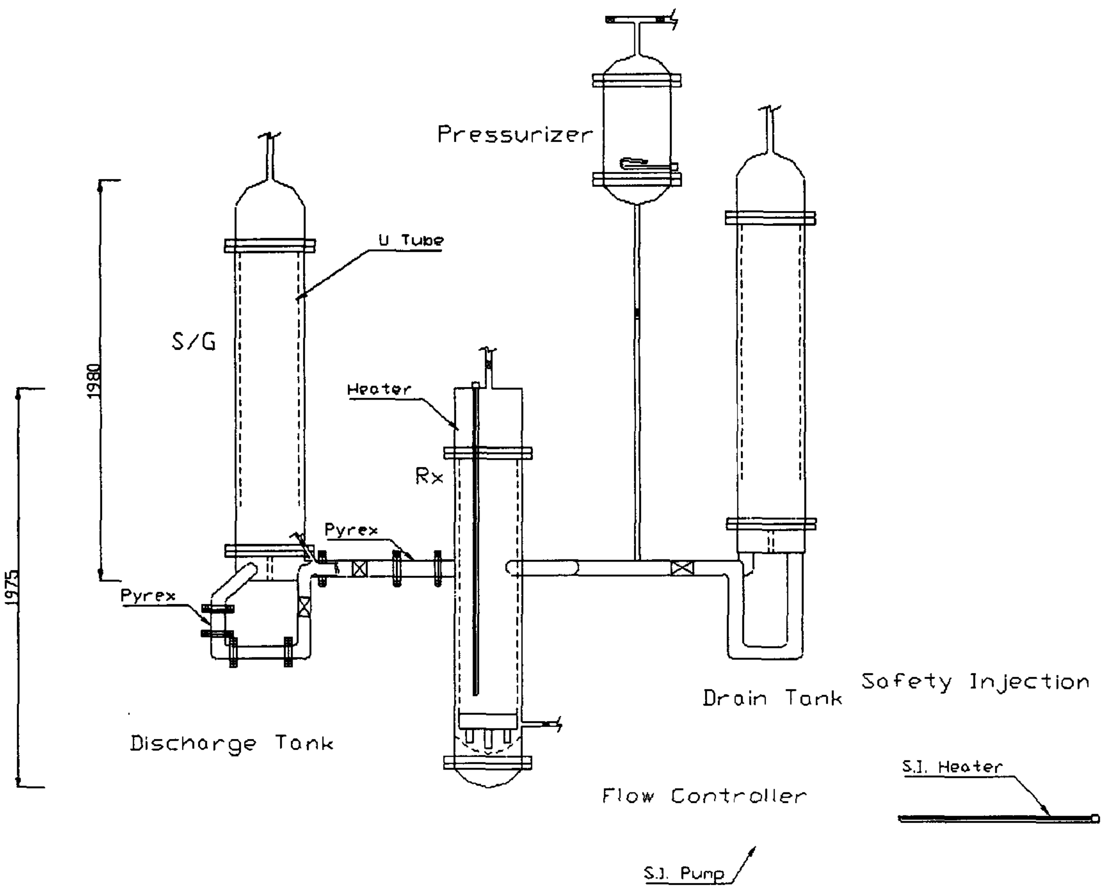

Fig. 1. Experimental Apparatus of SNUF

60 kW electrical heaters to simulate scaled 2.2% core decay heat, corresponding to decay heat at 15 seconds plus heat of inner metal structures. At the lower part of reactor vessel, the drain valve is piped outwardly for primary coolant drain during the blowdown phase.

A SG is equipped in each loop and each contains 4 U tubes with the height of 1725 mm, 6 U tubes with the height of 1643 mm and 6 U tubes with the height of 1562 mm. Inner and outer diameters of U tubes are 19 mm and 21 mm, respectively. The secondary system is composed of SG shell, steam line and feed water line.

The SI system is composed of a storage tank, a SI pump and a flow controller. The storage tank is equipped with preheaters of 20 kW in order to control the SI water temperature. SI piping is connected to cold leg.

Broken hot leg is designed to simulate double ended guillotine break, and thus composed of two broken sections, which is the most conservative case of DEHLSB. The two broken sections refer to the reactor side broken section and the SG side broken section. Each broken section is simulated by using a ball valve, and between the two broken sections a separation valve is installed. Since the separation valve is closed at the instant of beginning of experiment, independent measurements of mass release from each broken

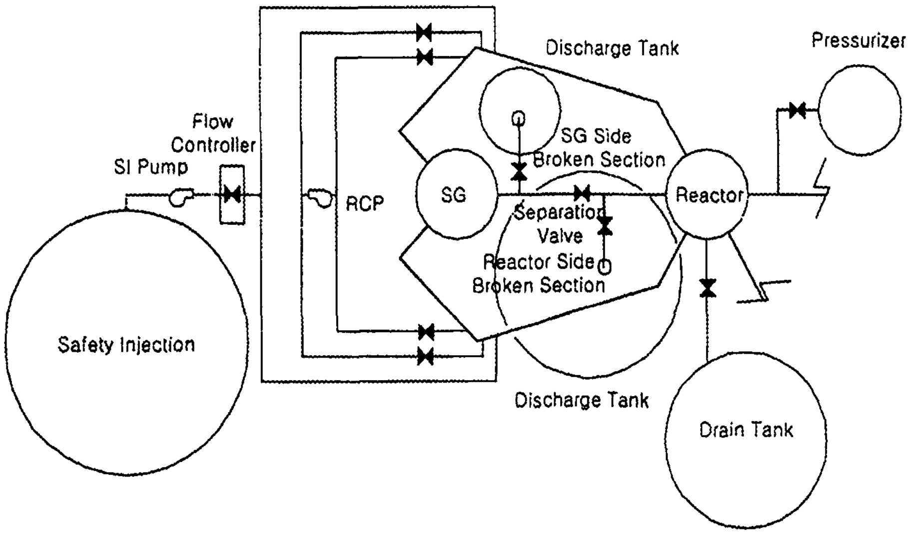

Fig. 2 Schematic Diagram of Test Facility

section are enabled. Windows are installed in hot leg for direct observation of fluid behavior. A window is also installed in cold leg suction part for the sake of direct observation of fluid behaviors in early stage of ECCS injection.

Discharge tanks connected to each broken section are the simulators of containment. Thus, the size of each discharge tank was determined to be sufficient for the expected discharge amount. The capacity of the core side discharge tank is about 150 liter with diameter of 600 mm and height of 700 mm, while that of the SG side discharge tank is 30 liter with diameter of 300 mm and height of 700 mm. Those two discharge tanks were interconnected with each other with 1/2 inch pipes in order to maintain the same pressure, but negligible steam flow through this piping was allowed. Each discharge tank contains a cooling system to condense the steam immediately and then the discharged amount can be measured by a level meter. The pressure of each discharge tank was kept atmospheric, because the atmospheric pressure is more conservative condition than RELAP5 analysis.

3.3. Measurement Instruments

The steam and water which are released through each broken section are condensed or cooled down in each discharge tank and then the water level is measured with the level meter having 10 mm resolution, which corresponds to 3.8 liter and 0.7 liter for large tank and small tank, respectively. Therefore, their accuracy is 1.43% of full scale. The discharged amount is calculated by multiplying the level by the cross sectional area of each tank.

The temperatures were measured, as shown in Fig. 2, at reactor vessel outer wall metal, fluid in intact hot leg, fluid in reactor lower plenum, fluid in reactor side broken section upper and lower part, fluid in SG side broken section upper and lower part, fluid in secondary system, and fluid in the half height and the bottom of U tube. The fluid temperatures in upper and lower parts of each broken section are measured to investigate the stratification, and the temperature at the half height of U tube is to catch up the flowing up of cold SI water into the

Table 4. Experiment Conditions

Normal Operation Conditions	Primary System Pressure Mode Core Power	: 0.8 MPa : Forced Convection : 60 kW
Initial Conditions	Primary System Pressure Secondary System Pressure SI Flow SI Temperature Core Power	: 0.35 MPa : 0.5 MPa(sat.) : 2.2 lit/sec : 60 ℃ : 60 kW

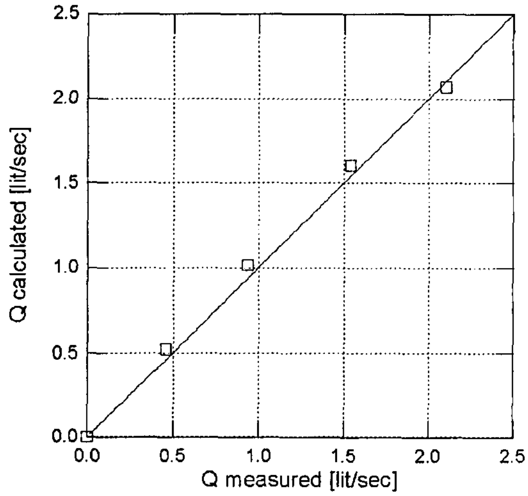

Fig. 3 Calibration Curve of Flow Rate

half height of U tube. The T type thermocouples were used, which is known to have excellent performance in the range of −200℃~400℃. Though the general accuracy of thermocouple is known to be within 1~2 ℃, the accuracy of 0.5 ℃ was maintained between 80 ~ 120 ℃ by the minute calibration.

The absolute pressure is measured at the reactor top head with DPI 260 model made in Druck Co.. Its accuracy is 0.1% of full scale according to its technical specification.

The flow rate is measured at the SI system with DP 103 wet-wet type. The volumetric flow rate correlation is

$$Q = C_d A_d \left[\frac{2\Delta p}{\rho(1-\beta^4)} \right]^{\frac{1}{2}} \tag{26}$$

where,

$$C_d = f(\beta) + 91.71\beta^{2.5} Re_d^{-0.75} + \frac{0.09\beta^4}{1-\beta^4} F_1 - 0.0337\beta^3 F_2$$

$$f(\beta) = 0.5959 + 0.0312\beta^{2.1} - 0.184\beta^8$$

$$F_1 = F_2 = \frac{1}{D(in)}$$

In the above equation, β is an important design parameter and 0.55963 is adopted in this system. The comparison of measured flow rate and calculated flow rate using above equation is presented in Fig. 3.

4. Experiments and Results

4.1. Procedure

At first, the primary system and the secondary system were maintained nearly saturated at 0.8 MPa and at 0.5 MPa, respectively. The core power was 60 kW. From this condition, the primary water inventory was drained until the water remained only at the lower plenum. This procedure is the blowdown phase. In the blowdown phase, the coolant in the cold leg

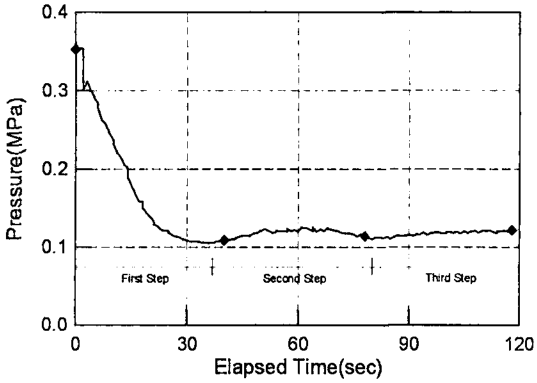

Fig. 4. Pressure Behavior in Reactor Top Head

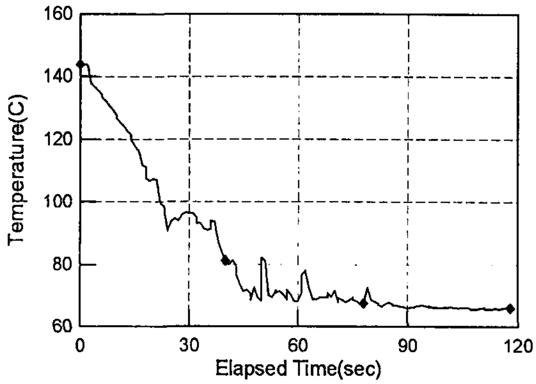

Fig. 5. Coolant Temperature in Reactor Lower Plenum

suction part did not flash up but remained as saturated or subcooled water. Thus in experiments the coolant was artificially drained. The initial temperature of secondary system was higher than that of primary system, and thus the secondary system could play a role of heat source if the injected SI water passed through U tubes.

At the commencement of experiment, the SI pump operated, each discharge valve opened to simulate the break, and the separation valve was closed at the same time. The main power was turned on earlier by 30～40 seconds for preheating than the start of the experiment. The SI temperature was maintained as 60 ℃, which was a little higher than that of prototype in order to compensate for the lack of the heat capacity of the RCS metal, and the SI flow rate was maintained constant at the rate of 2.2 lit/sec, which was correspond to the reference velocity in Table 2. The core power was set as 60 kW as previously described.

The summarized experiment conditions are shown in Table 4.

4.2. Results

As shown in Fig. 4, the primary system pressure shows three distinguished steps; first step in which the primary system pressure decreases rapidly, second step in which the pressure increases slightly and last third step in which the pressure decreases slowly toward steady state. The first step extends from 0 to 40 seconds, the second step from 40 to 80 seconds, and the thirds step from 80 seconds to the last.

The first step can be characterized by the rapid decrease of primary pressure, SI injection and EOB point. The pressure decreased until the SI water flowed into the core, which could be seen in the coolant temperature in reactor lower plenum in Fig. 5. In this figure, the temperature shows a plateau at 25～40 seconds and a sharp drop at 40 seconds, which was caused by that the coolant in lower plenum reached the bottom of core.

The SI water refilled the cold leg suction part first and then the core later because of the pressure gradient along the cold leg which connects the reactor in higher pressure and the SG side broken section in atmospheric pressure. The SI water which reached SG lower head was heated up by the cold leg metals. As a result, the temperature increased to the nearly saturation.

Here, it is important whether the SI water refills the cold leg first or the reactor first, because these two different scenarios of the refill may result in quite different accident consequence. If the SI

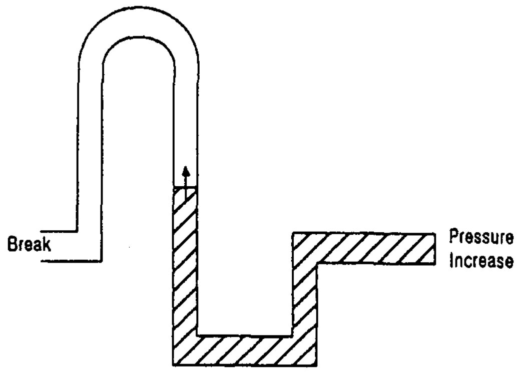

Fig. 6 Coolant Behavior in Cold Leg

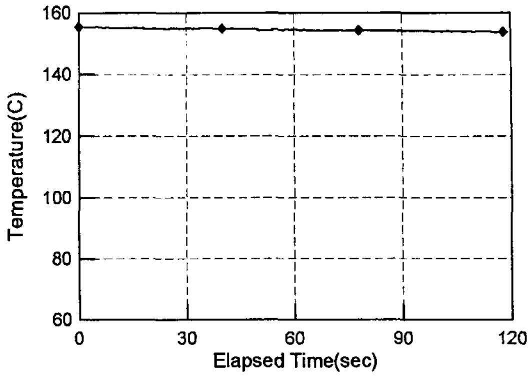

Fig. 9. Coolant Temperature in Secondary Side

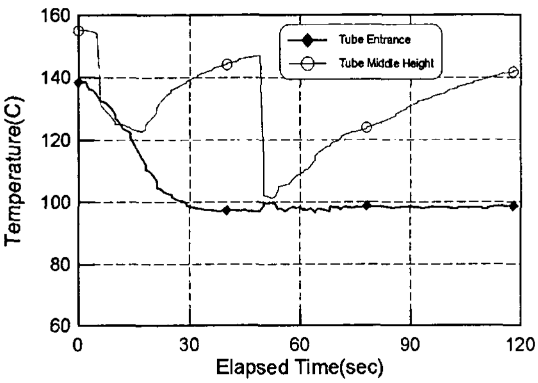

Fig. 7. Coolant Temperatures in U Tubes

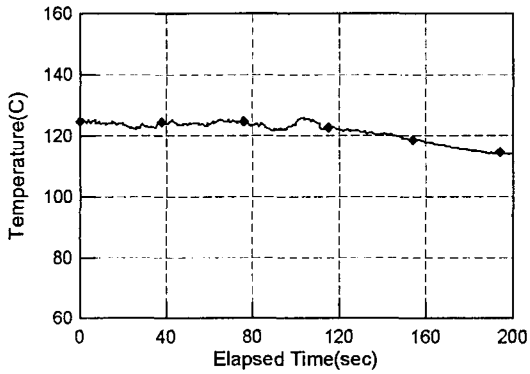

Fig. 10. Coolant Temperature in Secondary Side - Comparison Case

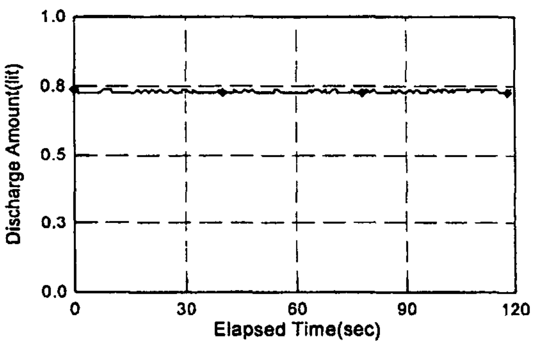

Fig. 8 Discharge Amount in SG Side Broken Section

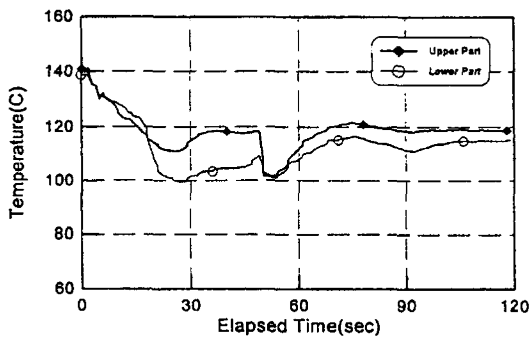

Fig. 11. Coolant Temperatures in SG Side Broken Section

water refills the cold leg first, the evaporation in core at early stage of core refill phase is important, because the evaporation gives rise to pressure increase in reactor and the pressure

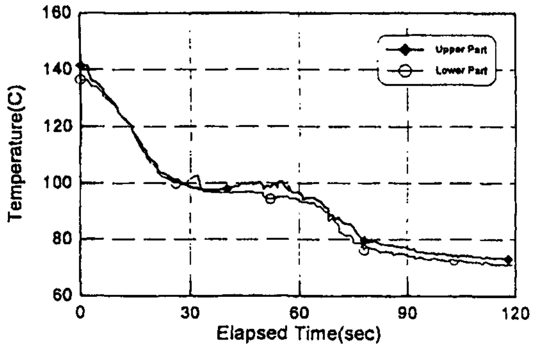

Fig. 12. Coolant Temperature in Reactor Side Broken Section

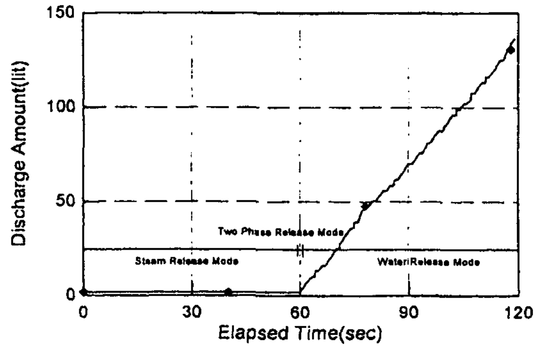

Fig. 13. Discharge Amount in Reactor Side Broken Section

increase drives the coolant in cold leg to flow up to U tubes. Whereas, if the SI water refills the core first, the cold leg can be refilled only by the flood of reactor. In this case, the evaporation in core at long term cooling period becomes important. Anyway, this experiment showed that the SI refilled cold leg first and core later, which could be a remarkable characteristic phenomenon of hot leg break.

The fact that the SI water refilled the cold leg in early stage of refill phase has another important meanings. The steam generated in core can never pass through the SG, because the water in cold leg plays a role of blockage of steam flow. Thus, the hypothesis of BOIL-OFF model that all steam is heated by SG secondary system seems far from realistic.

In this step, there exists the EOB point of this experiment. The time zero, strictly speaking, is not the EOB, because entire fluid at this point in SNUF is stagnant, differently from prototype. After the experiment commences, the flow field is formed, but the core maintains the similar flow regime as the time 0. Thus, the EOB point of this experiment lies between 0 and 40 seconds, although it is very difficult to point it out exactly.

Next, the second step is the major concerned period of this experiment in the point of mass and energy release as well as the assessment of RELAP5 analysis. When the SI water enters the core, the water is evaporated by the decay heat, and thus the generated steam increases the pressure of reactor slightly as shown in Fig. 4. As illustrated in Fig. 6, the increase of pressure drives the coolant in U tubes to flow up. This phenomenon was identified by measuring the coolant temperature at middle height of U tube as shown in Fig. 7. The sharp drop of temperature at 50 seconds means that the cold coolant reached the location where the thermocouple was installed. However, the coolant did not flow over the U tubes, and resultantly, there was no water discharge through the SG side broken section as shown in Fig. 8. Thus, the hypothesis suggested in Chapter 1, 'a case that the SI waster directly passes through the U tubes' , did not happen. Moreover, since the RCP reverse minor loss coefficient of SNUF was set lower than that of YGN3&4, the prototypic coolant rise is expected to be smaller than the scaled up rise.

The duration time in which the coolant stays in U tubes is thought about 30 seconds from 45 to 75 seconds in experimental chronology, judging from the primary pressure behavior in Fig. 4. Referring the scaling analysis discussed previously, the prototypic duration time is 7.7 seconds, since

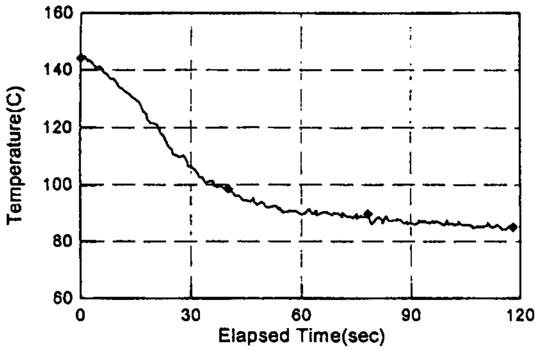

Fig. 14 Temperature at Reactor Wall

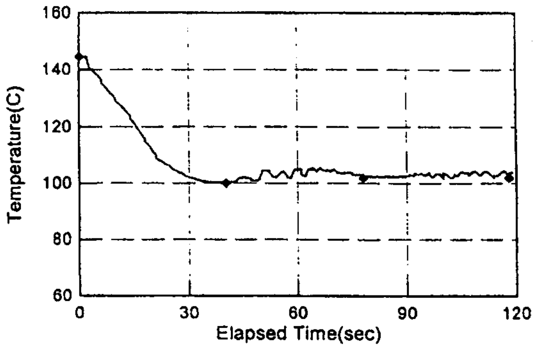

Fig. 15 Coolant Temperature in Intact Hot Leg

the chronology of the model lags 3.9 times comparing with that of prototype. The duration time of 7.7 seconds is thought to be very short time, comparing with the analysis time of the mass and energy release, 600 ~ 1000 sec.

Although, as mentioned in scaling analysis, quantitative discussion on the amount of evaporated steam by reverse heat transfer is difficult, another experiment was conducted in order to compare qualitatively the effect of discharge amount through the SG side broken section upon the secondary temperature. In the case of comparison, the total discharge amount through the SG side broken section was artificially set about 5.0 kg in two phase form, which was considerably small amount comparing with the total discharge amount through the reactor side broken section, 145 kg[14]. The Figs. 9 and 10 show the clear difference of secondary temperature drops. This implies that once the coolant, even if it is very small amount, passes through the U tubes the temperature of SG decreases sensitively. Thus, since the temperature of secondary side did not decrease in this study, it is another evidence that the release through the SG side broken section is extremely small.

The coolant temperature in SG side broken section in Fig. 11 shows the thermal stratification. Such a stratification means that the steam flow through the SG side broken section was very slow or stagnant, noting that thermal stratification in reactor side broken section, in which comparatively higher steam velocity was formed, was very small as shown in Fig. 12.

The error for the energy release can be roughly analyzed as followings. Because the measurement of SG side release has the resolution of 0.7 liter, less than 0.7 kg of water cannot be measured. Under the case of 0.7 kg saturated steam release through SG side broken section and 150 kg saturated water release through reactor side broken section, which is the worst case of experimental error, the maximum error of energy release can be calculated as follows(for the case of atmospheric property of water);

$$i_g\,(0.1\text{MPa, Sat.}) = 0.26763 \times 10^7 \text{J/kg}$$

$$i_f\,(0.1\text{MPa, Sat.}) = 0.41985 \times 10^6 \text{ J/kg}$$

Thus, the steam total enthalpy and water enthalpy released are 1.87341×10^6 J and 6.29775×10^7 J, respectively. Therefore, the maximum energy release error through SG side broken section is 2.97% in comparison with energy release through the reactor side broken section.

Summarizing above discussions, the direct water discharge through the SG side broken section that passes through the U tubes seems not to exist. And the steam discharge which is evaporated in U tubes is carefully thought to be extremely small, since the staying time of coolant in U tubes is very short. Thus, the role of SG as a remarkable heat source for mass and energy release is expected be minor.

Discharge through the reactor side broken section can be classified as steam release mode, two phase release mode and water release mode, as shown in Fig. 13. The steam release mode extends from the start of experiment to the time at which the reactor water level reaches the bottom of hot leg, and the water release mode extends from the time at which the reactor water level reaches the top of hot leg to last. The two phase release mode lies between the two modes. The measurable amount was discharged after two phase release mode. Such different release modes are totally governed by the reactor water level.

The coolant temperature of reactor side broken section in Fig. 12 shows that the coolant is subcooled state after 60 seconds, and the coolant temperature in reactor lower plenum in Fig. 5 is also subcooled state after 20 seconds. Only the coolant temperature of U tubes entrance and middle height is saturated or superheated after 60 seconds. Thus, it is thought to be far from realistic that all coolant in reactor vessel is assumed to be saturated water in BOIL-OFF model.

In the last third step, the pressure, the temperature and so on go stable. No more transients are expected thereafter.

The metal temperature at outer wall of reactor lower plenum in Fig. 14 decreased by about 50℃ and then went to steady condition, which shows that the metal played a role of the heat source in time.

The coolant temperature in the intact hot leg decreased rapidly with the primary system

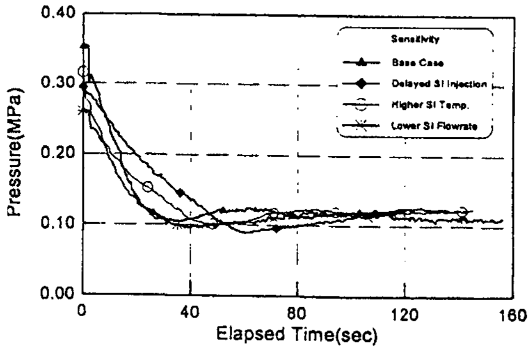

Fig. 16. Results of Sensitivity Study with SI Conditions

pressure and then went to steady in saturation temperature, as shown in Fig. 15. The fact that the intact hot leg maintained saturated means that the intact loop was stagnant, because the coolant temperature in core top part was already subcooled as shown in Fig. 12.

4.3. Sensitivity Studies and Discussion of Scaling Effects

The sensitivity studies were conducted with the variation of SI conditions. The cases of delayed SI injection(about 45 sec. delayed from the commencement of experiment), higher SI temperature(90℃) and less SI flow rate(1.59 lit/sec) were conducted. The delayed SI injection case covers the uncertainty of EOB definition and the timing of actual SI injection, the higher SI temperature compensates for the lack of metal structure in RCS, and the less SI flow rate case is to investigate the sensitivity of SI flow rate.

In any case, overall phenomena were similar with the base case and specially the pressure rise in second step appeared in any case. Fig. 16 shows the primary pressure trend with the variation of SI conditions. Under the condition of less steam generation in core according to the SI conditions, the rise of pressure was less and the

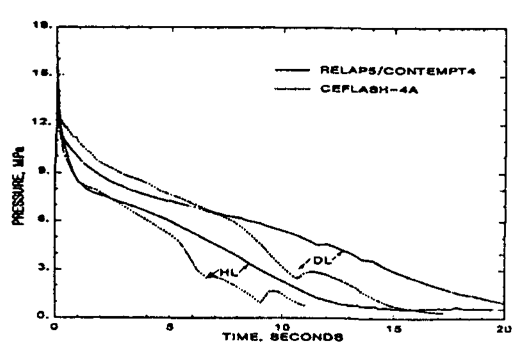

Fig. 17. Reactor Pressure by RELAP5 Analysis in Reference 13

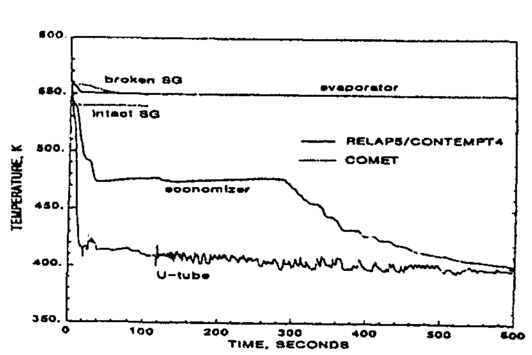

Fig. 18. Secondary Side Temperature by RELAP5 Analysis in Reference 13

water in the cold leg flowed up to less than the half height of U tube. However, the release through the SG side broken section was also negligible in all cases[14].

In addition, this sensitivity studies also show the effect of scaling distortion in Froude number and subcooling number. The variation of SI flow rate means the change of Froude number and the variation of SI temperature means the change of subcooling number. In spite of such distortion in scaling, the Fig. 16 shows similar primary pressure trend which governs overall system behavior.

The delayed SI injection means the change of the initial condition. The delay of SI injection also showed similar system behavior as can be seen in Fig.16. This fact implies that the uncertainty in initial condition does not fatally affect the system behaviors.

5. Assessment of RELAP5 Analysis

The second purpose of this paper is to assess the RELAP5 analysis for hot leg break LBLOCA of YGN3&4 conducted by KAERI. The conclusion of RELAP5 analysis for the excessive conservatism in current mass and energy release analysis needs to be verified phenomenologically[13].

Since limited studies for the hot leg break have been implemented world widely, the comparison of experimental results and RELAP5 results for hot leg break may have important meanings, even though both results have their uncertainties. Generally speaking, perfect comparison of scaled down experimental results with prototypic RELAP5 analysis is quite difficult. However, such discrepancy can be minimized by the proper scaling for important parameters. In this study, limited comparison was carried out by scaling analysis for carefully selected parameters important to mass and energy release.

At first, the primary pressure trend was compared. The experimental result shows the increase of primary pressure in second step by the steam generation in core, which is a very important phenomenon because it governs the coolant behaviors in cold leg of broken loop. This phenomenon has been rarely reported. The intensive observation of RELAP5 analysis in Fig. 17 shows similar trend, although it is not so distinct since the pressure span of the graph is too large.

The duration time of flowing up in U tubes was compared. This experiment estimated the duration time as 30 seconds which was corresponding to

7.7 seconds in prototype. The increase of pressure in second step directly affects the behavior of coolant in cold leg of broken loop. Due to the coolant flowing up, the temperature of economizer decreases and that of U tubes increases at 25 ~ 35 seconds as shown in Fig. 18[3,13]. Judging from these facts, the duration time estimated by RELAP5 is thought to be about 10 sec.. The duration times of two results seem to be well agreed.

The height of flowing up was also compared. It should be noted that the small increase of pressure in second step can provide a sufficient driving force for the coolant in cold leg to flow up to U tubes. For example, the pressure increase of 0.1 atm would result in 1 meter flowing up to U tubes. The experiment shows that the coolant in SG lower head flows up to half height of U tubes, whereas the RELAP5 analysis shows that the coolant flows up only to the height of the economizer. The relative height of RELAP5 analysis seems to be lower than that of experiment. However, noting that the SNUF is reduced height facility and the SNUF uses the prototypic pressure or prototypic pressure difference between the reactor and the containment, the absolute heights of flowing up in both results are not much different.

6. Conclusions

This study is the first attempt of hot leg break LBLOCA analysis in the stand of mass and energy release through the experiments. Through the series of experiments, the phenomenological analysis on the mass and energy release was conducted. And the assessment of RELAP5 analysis for prototype was implemented in the view point of the release through the SG side broken section

The experiment showed that most of injected SI water was released through the core side broken section and thus the release through the SG side broken section was negligible. The remarkable feature is that the SI water refilled the cold leg first, and resultantly the steam generated in core could not be heated directly by SG. Although this experiment showed that the water in cold leg could flow up to a half height of U tubes, the duration time was not sufficiently long for the SG to be taken as a major heat source. Thus, this study demonstrated that the BOIL-OFF model is unrealistically conservative. The assessment of the RELAP5 analysis showed that the analysis for the release through the SG side broken section provided qualitatively proper results.

In further study, the analysis of the effect of secondary condition will be complementary to this study. And the more accurate quantitative analysis for hot leg break accident would be necessary to relieve the excessive conservatism of mass and energy release analysis.

Nomenclature

English

A : Flow Area or Flow Area Ratio
A_d : Cross-sectional Area of Pipe [m^2]
a : Component Flow Area [m^2]
c_p : Specific Heat [J/K-kg, kJ/K-kg]
D : Diameter of Pipe [m]
d : Hydraulic Diameter [m]
E : Energy [J, kJ]
f : Friction Factor
g : Gravitational Constant [9.8 m/sec^2]
i : Specific Enthalpy [J/m^3, kJ/m^3]
K : Loss Coefficient
L : Length or Length Ratio
l : Component Length [m]
K : Minor Loss Coefficient
$\dot{m}$: Flow Rate [kg/sec]
N : Dimensionless Number

p : Pressure [Pa, kPa, MPa]
Δp : Pressure Difference[Pa, kPa]
Q : Volumetric Flow Rate [m^3/sec]
$\dot{q}$: Heat Generation Rate [W, kW]
$\dot{q}'''$: Volumetric Heat Generation Rate [W/m^3, kW/m^3]
T^* : Time Ratio Number
t : Time [sec]
u : Velocity [m/sec]
V : Volume [m^3]
V_{gj} : Drift Velocity [m/sec]
w : Work [J, kJ]
x : Quality

Greek

α : Void Fraction
α_s : Solid Thermal Diffusivity
Δ : Difference
δ : Conduction Depth [m]
μ : Viscosity [N sec/m^2]
ρ : Density [kg/m^3]
Ψ : Parameter
τ : Time Constant

Subscript

0 : Reference
e : Exit
f : Liquid or Fluid
g : Gas or Vapor
i, j : i-th, j-th Component
j : Relative to Average Volumetric Flux j
in : Inlet
m : Model
out : Outlet
p : Prototype
R : Ratio
s : Structure
sat : Saturation
sub : Subcooling

References

1. G.C. Park, et al., "Review on the Mass and Energy Release Analysis", KINS/HR-093, (1993).
2. C.W. Kim, et al., "Development of analysis methodology for hot leg break mass and energy release", KAERI.4, (1995).
3. Jin Ho Song, Young Min Kwon, Hae Young Jeong, Tech Mo Kim and Sang Jong Lee, "An Analysis Methodology for Hot Leg Break Mass and Energy Release", KAERI/TR-746/96, (1996).
4. KEPCO, "YGN3&4 FSAR", Chap. 6.
5. M. Ishii and O. C. Jones Jr., "Derivation and Application of Scaling Criteria for Two-Phase Flow", Two-phase Flows and Heat Transfer Proc. NATO Advanced Study Institute, Istanbul, Turkey, Vol. 1, (1976).
6. M. Ishii and I. Kadaoka, "Scaling Laws for Thermal-Hydraulic System under Single Phase and Two Phase Natural Circulation", Nuclear Engineering and Design Vol. 81, pp411-425, (1984).
7. G. Kocamustafaogullari and M. Ishii, "Scaling of Two-Phase Flow Transient Using Reduced Pressure System and Simulant Fluid", Nuclear Engineering and Design Vol. 104, pp121-132, (1978).
8. M. Ishii, et al, "Scientific Design of Purdue University Multi-Dimensional Integral Test Assembly(PUMA) for GE SBWR" , NUREG/CR-6309, (1996).
9. Y. Y. Hsu, Z. Y. Wang, M. di Marzo and K. Almenas, "Scaling-Modeling for Small Break LOCA Test Facility ", Nuclear Engineering and Design Vol.122, pp175-194, (1990).
10. M. Ishii, "One-Dimensional Drift-Flux Model and Constitutive Equations for Relative

Motion between Phases in Various Two-Phase Flow Regimes", ANL-77-47, Chap. VIII, (1977).
11. G. B. Wallis, "One-Dimensional Two Phase Flow", McGraw-Hill, Chap. 4~5, (1969).
12. L. S. Tong, "Boiling Heat Transfer and Two-Phase Flow", John Wiley & Sons, Inc. (1965).
13. Young Min Kwon, Chan Eok Park and Jin Ho Song, "Comparative Mass and Energy Releases and Containment Analysis for a Large-Break Loss-of-Coolant-Accident Using RELAP5/CONTEMPT4 and Design Computer Codes", Nuclear Technology Vol. 122, pp.295-305, (1998).
14. G.C. Park et al., "A Verification Experiment on Hot Leg Break LBLOCA", KAERI, (1996).