LOCA Analysis for the Basic Design of **HANARO** Fuel Test Loop

Dae-Young CHI, Chung-Young Lee, Bong-Shick Sim, Jun-Yun Kim,
Kook-Nam PARK, Cheol PARK

HANARO Center, Korea Atomic Energy Research Institute
P.O.Box 105, Yuseong, Daejeon, Korea, 303-305
e-mail : cpark@kaeri.re.kr

ABSTRACT

Basic design of HANARO FTL is almost done, and this paper presents the preliminary LOCA analysis for the basic design of a fuel test loop using MARS code, which is being developed by KAERI for the simulation of a wide variety of PWR system transients. The focal points are on the thermal-hydraulic behavior of fuel as well as the pressure response in a shielded room where the process equipment are located. Through the analysis, the integrities of fuel and shield room were confirmed.

1. Introduction

HANARO(High-Flux Advanced Neutron Application Reactor)[1] is a light water cooled, heavy water reflected, open-chimney-in-pool type research reactor with a maximum thermal power of 30MW. Since the successful operation of HANARO in 1995, the utilization demands have been increased continuously together with installation of utilization facilities. According to the increasing demands on irradiation tests to develop new fuels in Korea, a fuel test loop (FTL) facility is now under design to conduct in-core fuel performance test in PWR and CANDU operating conditions, which will be installed at HANARO.

In general, a large LOCA is one of the design basis accidents (DBA) to evaluate the adequacy of various structures, systems, and components used to protect public health and safety. This is the case in HANARO FTL facility design. The primary purpose of LOCA analysis is to demonstrate the satisfactory performance of the emergency cooling water (ECW) system and accumulators. To confirm integrities of a shield room and reactor hall integrity due to the pressure build-up following a LOCA is another concern in this analysis.

Hence, the objective of this work is to perform the analysis for the thermal-hydraulic behavior of fuel and system of FTL following large break LOCA as well as the pressure response in a shielded room where the process equipments of FTL are located.

2. HANARO Fuel Test Loop [2]

HANARO FTL will serve for the irradiation of test fuels and materials at high pressure and high temperature conditions of PWR and CANDU reactors. It consists of an in-pile test section (IPS) and an out-file system (OPS), as shown in figure 1. IPS, located in IR hole of HANARO core at figure 2, provides a pressure boundary and an environment in the HANARO core for carrying out fuel performance tests of fuel pins under typical commercial power reactor conditions. Hence, it is specially designed to separate the high pressure and high temperature coolant from the low pressure and low temperature coolant of the reactor. It can adopt 1 to 3 fuel pins of 70 cm length for irradiation test, as shown in figure 3. The IPS is connected by rigid piping to the OPS which provides cooling water at the required pressure, temperature, flow rate and chemistry. The coolant comes in through the lower connection on the IPS head, flows down between the inner pressure tube and the flow divider tube which is part of the inner assembly, then it comes up past the test fuel, up the inside of flow divider tube and out through the upper connection on the IPS head.

The out-file system, which provides the proper test conditions and safety functions, is designed to have process equipments such as pumps, pressurizer, cooler, and other control instruments. These process equipments are housed in a shielded room separate from the HANARO process equipment. Piping connects the loop frame to the IPS and OPS. Major systems of OPS are a main cooling water (MCW) system, an emergency cooling water (ECW) system, a component cooling water (CCW) system, let down, makeup and purification (LMP) system, a radiation monitoring (RMS) system, and so forth. The MCW establishes and maintains the design process condition to the IPS during normal test condition. The ECW system provides emergency cooling water to the IPS subsequent to anticipated operational occurrences(AOO) and design basis events (DBE).

Major design and operating conditions of FTL for PWR and CANDU fuel test are shown in table 1.

3. Description of Code and Model

3.1 MARS Code

The MARS (Multi-dimensional Analysis of Reactor Safety) code[3], which is being developed and verified by KAERI, is a realistic system transient analysis code that can be used for the simulation of a wide variety of PWR system transients. This code is a unified version of 1 D reactor system analysis code, RELAP5/MOD3 and 3D reactor vessel analysis code, COBRA-TF coupled with 3D reactor kinetics code, MASTER and containment code, CONTEMPT4. Through the validation calculations for various experimental results, the capability of MARS code has been verified as a multi-dimensional system analysis code [4].

3.2 Model Description

For the analysis, the modeling of HANARO FTL facility was developed and input for MARS has been prepared. The nodalization of FTL facility used in this analysis is shown in Figure 4. Input data on volumes, junctions, and heat structures were obtained from the facility drawings and documents. The IPS is represented by 2 channels in parallel, i.e., hot and average channel. Fuel is modeled to have 5 axial node and 11 radial meshes with a heat structure component. Heat structure component was also used to consider the heat transferred from the high temperature FTL coolant piping to the low temperature HANARO pool water. For piping and component outside reactor pool, heat structure is not attached since heat loss is very small. 1.0 was used as a break discharge coefficient at ruptured location. The shield room and reactor hall model for pressure build-up calculation is shown in figure 5.

3.3 Major Input and Assumptions [5]

1) Heat generation in fuel and IPS

Heat generated in the IPS during the irradiation test consists of fuel heat by fission and IPS structure heat by γ heating. According to physics calculation [6] for 3% enrichment test fuel in

70 cm length, average linear heat generation rate of test fuel in IR hole of HANARO operating at 30 MW_{th} was estimated as 29.37 kW/s with the axial power distribution in figure 6. Gamma heating rates of flow tube and inner pressure tube were also evaluated as 3.805 kW/kg and 6.036 kW/kg, respectively. Heat generation in outer tube is not considered because it is removed by the HANARO pool water.

2) Volumes of shield room and Reactor hall

The pressure response at a shield room depends on the mass and energy released following a LOCA as well as room volume. The gross volume of a shield room was assumed to 20 % decrease to account for equipment and other components volumes within the shield room since the final equipment selection has not been made. The volumes used in this analysis are as below.

- Room #1 volume : 118.45 m^3
- Room #2 volume : 198.21 m^3
- Reactor Hall volume : 31000 m^3

This analysis considered the following 3 cases.

- Case 1 : Break flow released only room 1
- Case 2 : Break flow released room 1 and room 2
- Case 3 : Break flow released directly to reactor hall

3) Trip parameters and set points

Various thermal hydraulic parameters are continuously monitored at specified locations to protect FTL and HANARO. They generate signals for reactor trip and system/components actuations if the measuring values reach their set points. Those important parameters and set points used in the analysis are presented in table 3.

4) Initial operating conditions

The assumed initial conditions are listed in table 4, which represent the conservative operating conditions before the postulated large break LOCA occurs.

4. Analyses and Discussions

4.1 Accident Sequences

A large break LOCA at room 1 postulated in this analysis is a double ended rupture of the cold leg piping downstream of the main coolant pump. The rupture was assumed to occur instantly and the coolant starts to be discharged to room 1. It results in a rapid depressurization of the FTL system, and immediate coolant flow stagnation through the IPS vessel. FTL low flow signal is generated to trip HANARO reactor, and the power actually starts to decrease a little bit later due to trip delay. FTL low-low flow isolation set point reaches in 0.762 seconds, and the actuations of valves such as FTL isolation valves and accumulator valves were assumed to be finished 1 second later. By the injection of ECW cooler flow and the trip of reactor power, the fuel cladding temperature sharply decreases just after reaching its peak at 3.2 seconds.

Table 2 presents a chronological sequence of the system responses and the event times following a large break LOCA at room 1.

4.2 Safety parameters

Large break LOCA is generally used as the design basis accident for determining the performance requirements of ECCS and containment. Here, the following safety parameters were used to confirm the integrities of fuel, shield room and reactor hall

1) Peak cladding temperature

Most important one among the acceptance criteria for ECCS for US light water commercial power reactors is that the calculated maximum fuel cladding temperature should not exceed 1204 ^{o}C (2200 ^{o}F). The uncertainties of 167 ^{o}C and 100 ^{o}C have been assumed for TRACPF/MOD1 and RELAP5/MOD3 respectively. So the analysis limit of 1037 ^{o}C (1310 ^{o}F) has been conservatively used in this analysis.

2) Pressure of shield room and reactor hall

Rapid pressurization resulted in the shield room and reactor hall following a large break LOCA. Hence, their integrities should be estimated during the transient. Here, the pressures below have been used as an allowable limit against their integrity.

- Allowable pressure of Room 1 concrete wall : 0.2185 MPa
- Lifting pressure of Room 1 shield plug : 0.1272 Mpa
- Design pressure of reactor hall : 25 mmWg

4.3 Calculation Results

As soon as the guillotine break occurs at the cold leg piping down stream of the main coolant pump, the coolant is immediately lost to a shield room. Subsequently, the FTL system pressure rapidly depressurized and the IPS experienced blowdown including flow reversal, as shown in figure 7. Due to the low-low flow signal, the isolation valves are fully closed and the accumulator valves are fully open at 1.762 seconds. Figure 7 presents the variation of IPS flow during the LOCA transient. As shown in the figure, an accumulator can provide emergency cooling water of around 0.6 kg/s in a stable manner except short period of blowdown just after initiation of the accident. The fuel heats up to its maximum over the blowdown period. Then the cladding temperature sharply decreases by the injection of ECW coolant flow and the reactor trip. The analysis results indicated that the maximum fuel peak cladding temperature was predicted as 929 K (656 ^{o}C) in case 1 as shown in figure 8, which may be less than the typical allowable limit 1037 ^{o}C (1310 K) even if the calculation uncertainty of the code is included.

Another concern is the pressure build-up in a shield room and reactor hall. Due to the released mass and energy of break flow, the shield room and reactor hall pressures increase. Figure 9 shows the break flow into the shield room # 1, which indicates that most of discharged flow is poured within several seconds after accident. Pressure responses in the shielded room during transient are shown in figure 10 and 11, which presents the comparisons with the allowable pressure of room 1 concrete wall and the lifting pressure of room 1 shield plug, respectively. It can be seen in figure 10 that the room #1 concrete wall maintains its integrity during all cases of transients because the room 1 pressure is far below the allowable limit of the strait line in the figure. However, shield room pressure may exceed the lifting pressure of shield plug in case 1 of transient, as shown in figure 12. That implies the necessity of pressure release to additional space. The pressure behavior in case 2 and 3 support this fact. The pressure response of the reactor hall during transient is shown in figure 13, which indicates that the reactor hall ceiling can maintain integrity during any case of LOCA transient.

5. Concluding Remarks

Using MARS code for the best-estimate thermal hydraulic system analysis code, LOCA analysis for the basic design of HANARO FTL facility was carried out. For the analysis, the modeling of HANARO FTL facility was developed and input for MARS has been prepared. The analysis is focused on the thermal-hydraulic behavior of fuel and ECW system as well as the pressure response at a shielded room where the process equipment are located.

From the analysis results, it is expected that the integrities of test fuel and shield room are maintained during the postulated LOCA accident at main pipe in the shield room. That is, in the basic design stage, the fuel test loop facility seems to be properly designed against postulated LOCA. This analysis will be updated

This analysis should be updated through the detail design process reflecting the geometry, design and operating conditions of related systems. And the pressure response in the shield room also needs more detail estimation because this calculation is a little coarse.

Acknowledgements

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References

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[2] B. S. Sim et. al., “Design of the Fuel Test Loop in the HANARO”, Proceedings of 6th IGORR, Daejeon, Korea, April, 1998.

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Table 1. Design and Operating Conditions of Fuel Test Loop

	Parameter	PWR	CANDU
System Design Parameter	Pressure	17.5 MPa	17.5 MPa
	Temperature	350 ^{o}C	350 ^{o}C
IPS Operation Condition	Generated Heat	112 kW	116 kW
	Inlet Temperature	300 ^{o}C	277 ^{o}C
	Outlet Temperature	312 ^{o}C	290 ^{o}C
	Pressure	15.5 MPa	10 MPa
	Mass Flow Rate	1.6 kg/s	1.63 kg/s

Table 2. Accident Sequences of Large Break LOCA at Room #1

Time (s)	System Action
0.0	Room #1 Cold Leg Large Break Initiation
0.008	HANARO Trip (low flow)
0.012	Isolation Trip (low-low flow)
0..758	Start to power decrease (due to 0.75 s trip delay)
0.762	FTL Isolation Valves Begin Closing
0.762	FTL Accumulator Valves Begin Opening
0.762	
1.276	FTL Vent Valves Begin Opening
1.762	
1.762	Room #1 Pressure Signal (high pressure)
1.762	FTL Isolation Valves Finish Closing
3.2	FTL Accumulator Valves Finish Opening
	FTL Vent Valves Finish Opening
	Peak Clad Temperature Reached at 656 ^{o}C

Parameter	Set point	Delay time
Reactor trip		
High temperature	320 ^{o}C	0.75 sec
Low flow rate	1.28 kg/s	0.75 sec
High flow rate	1.84 kg/s	0.75 sec
Low pressure at IPS outlet	14.1342 Mpa	0.75 sec
High pressure at IPS outlet	17.2368 Mpa	0.75 sec
High pressure at room #1	0.10825 Mpa	0.75 sec
Isolation valve, Accumulator		
High-high temperature	326 ^{o}C	0.95 sec
Low-low flow rate	0.4 kg/s	0.95 sec
Low-low pressure	13.4447 MPa	0.95 sec

Table 3. Trip parameters and set points

Table 4. Initial operating conditions

IPS Operation Parameters	Values(PWR 3%)
Generated Heat (kW)	87.8
Inlet Temperature (¡ É)	303
Outlet Temperature (¡ É)	312
Outlet Pressure (MPa)	15.5
Mass Flow Rate (kg/s)	1.6
IPS Up Channel Flow Area (1 ¡ ¿ 10^{-3} § ³)	0.4939
IPS Fluid Velocity in Fuel Region (m/s)	4.6
Specific Heat Capacity (kJ/kgK)	6.0540 @307.45 ¡ É

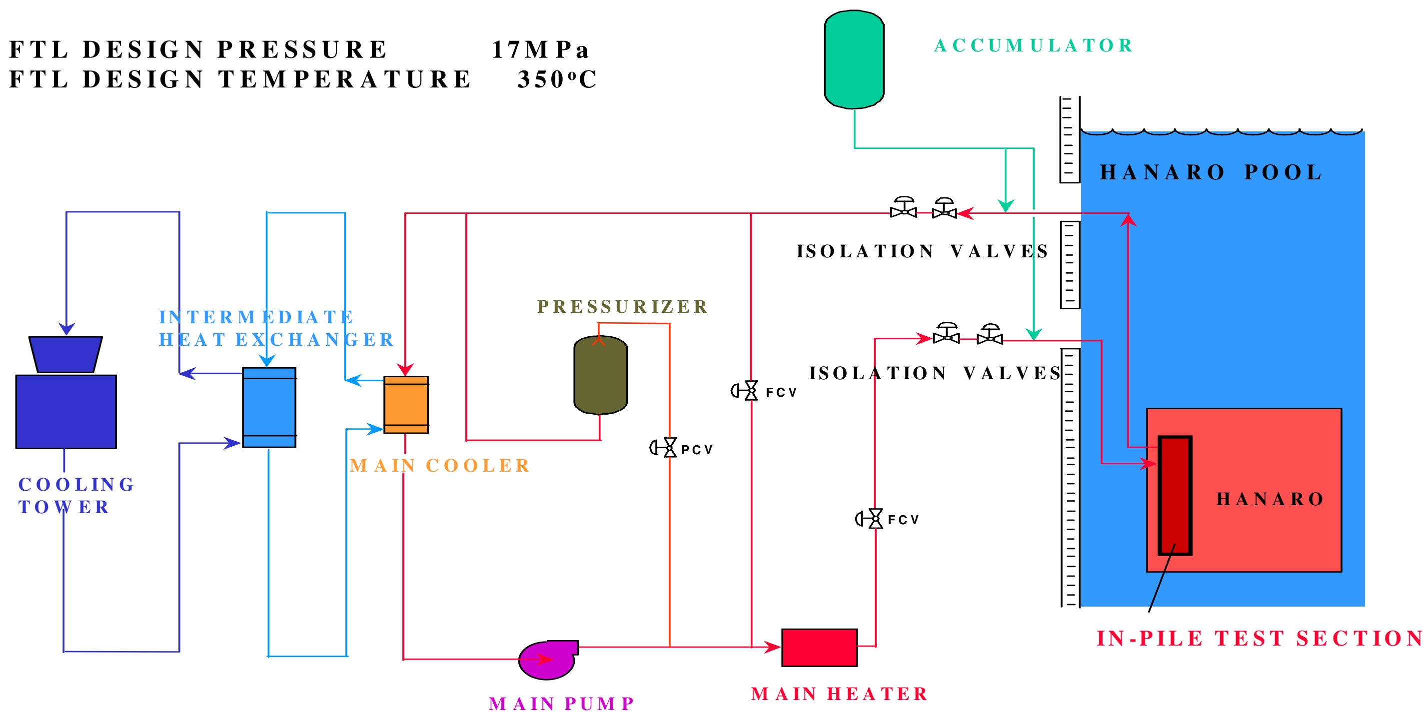

Fig. 1 Schematics of HANARO Fuel Test Loop

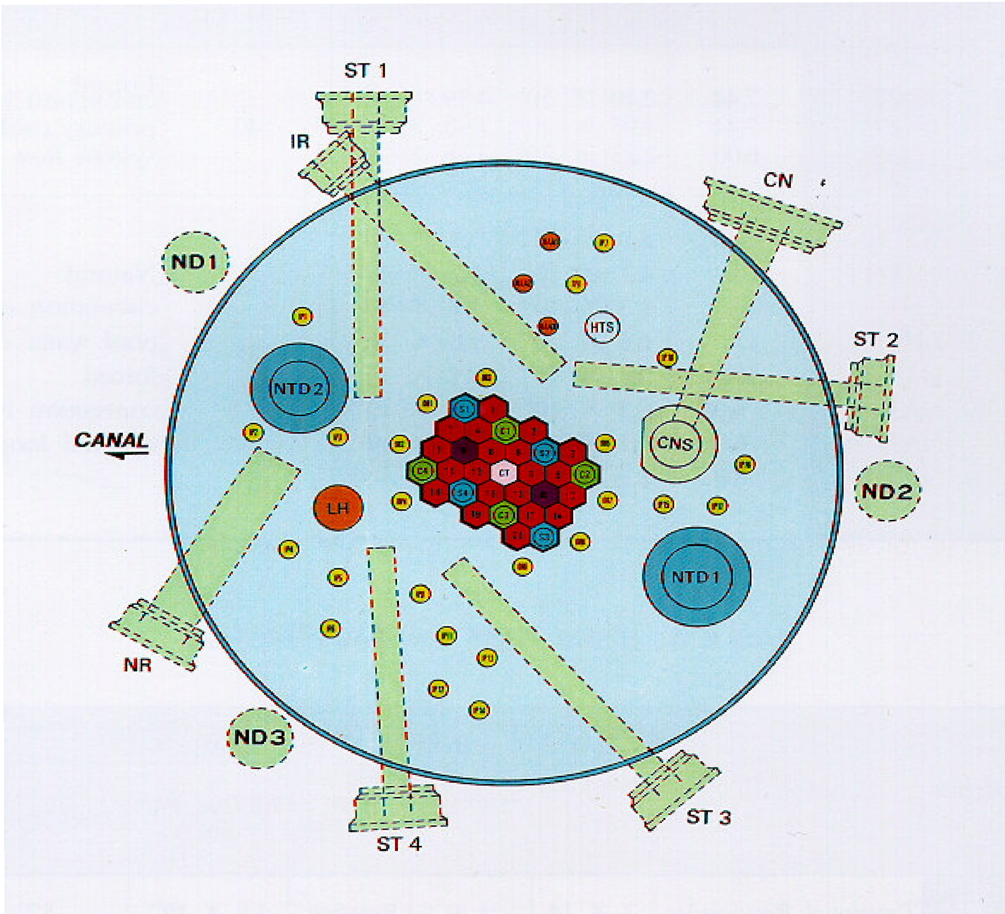

Fig. 2 Reactor Core of HANARO

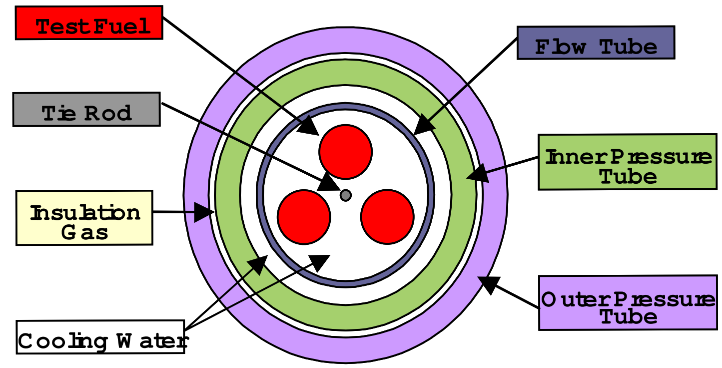

Fig. 3 Cross-sectional view of IPS

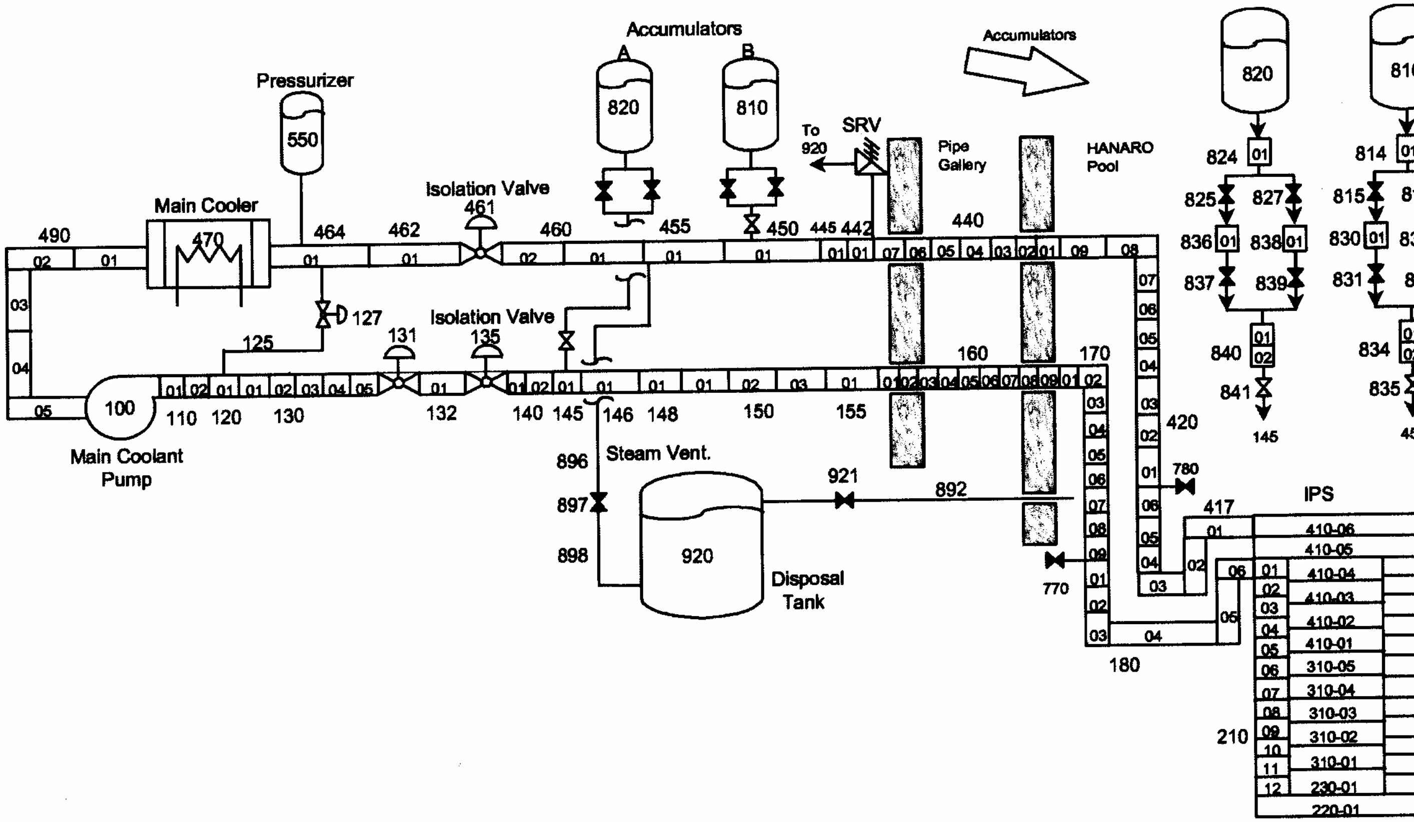

Fig. 4 Nodalization of HANARO FTL facility for LOCA Simulation

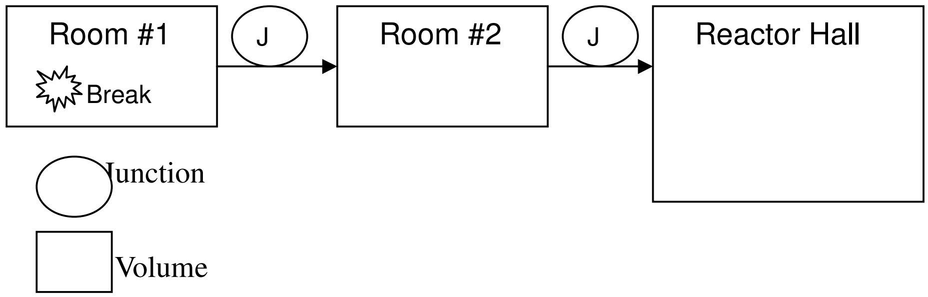

Fig. 5 Shield room and Reactor hall model

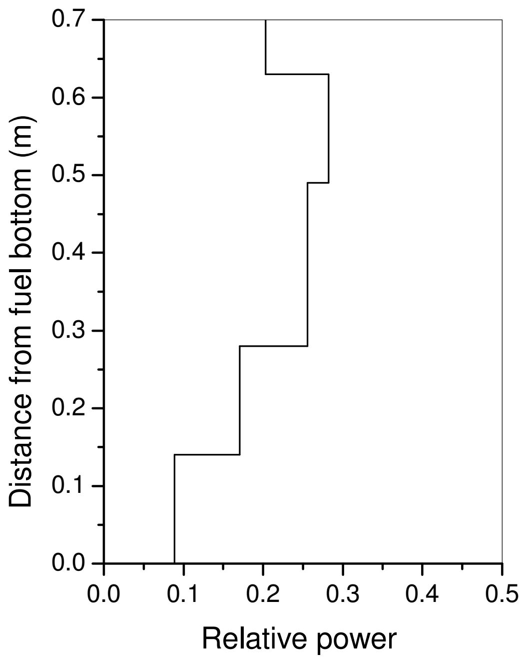

Fig. 6 Axial Power Distribution

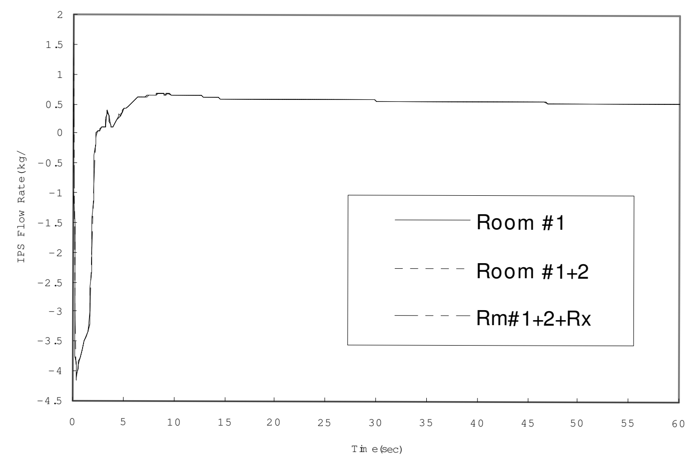

Fig. 7 IPS flow rate during LOCA

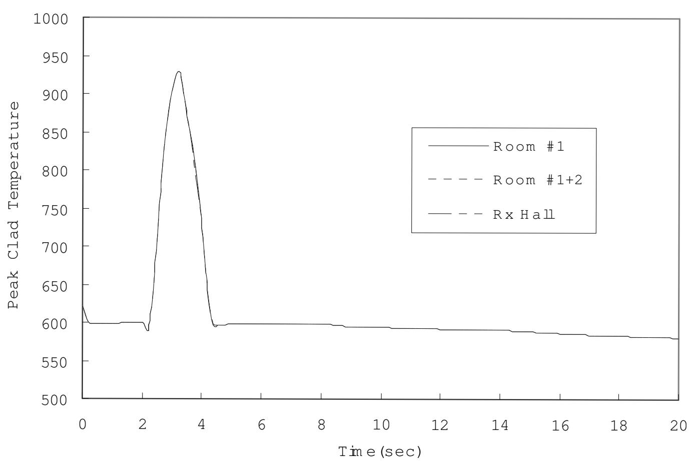

Fig. 8 Peak cladding temperature during LOCA

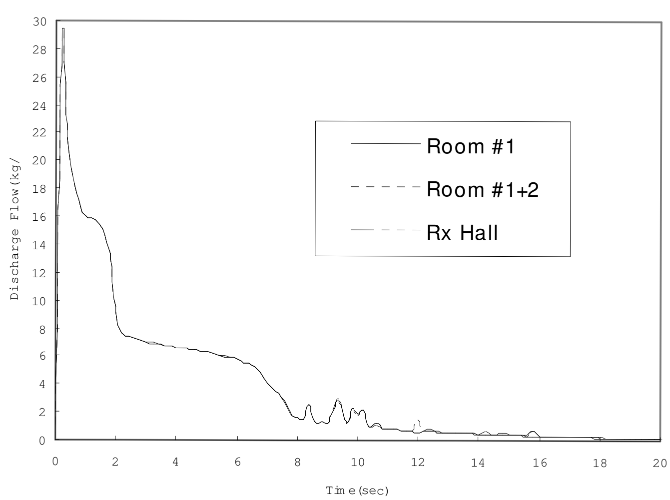

Fig. 9 Break flow rate during LOCA

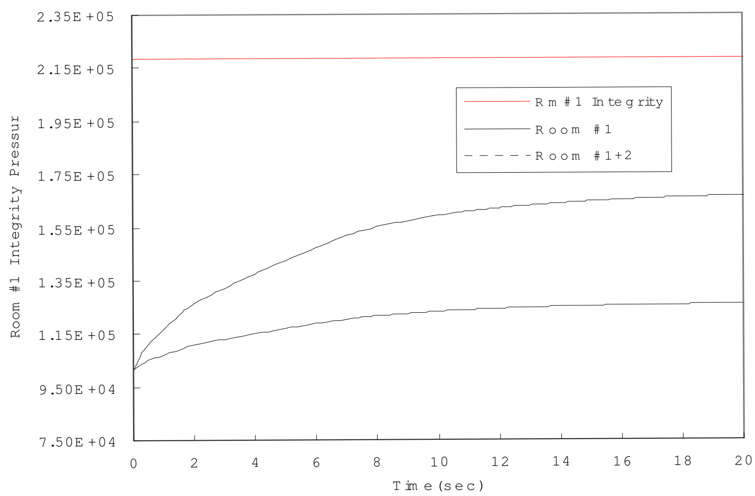

Fig 10. Pressure variation at Room #1 during LOCA

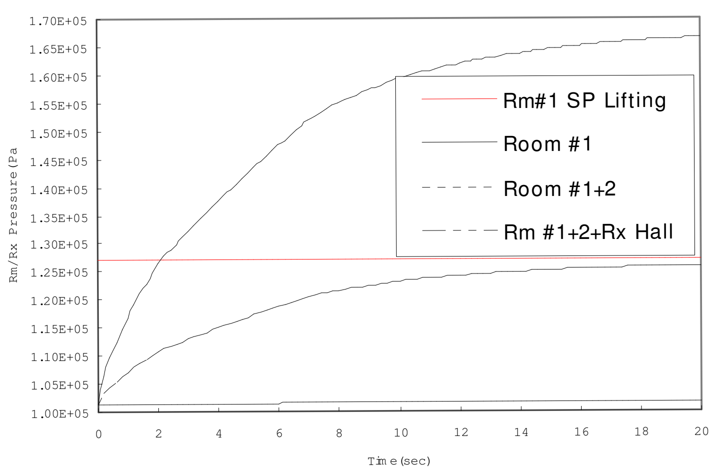

Fig 11. Pressure variation Vs lifting pressure during LOCA

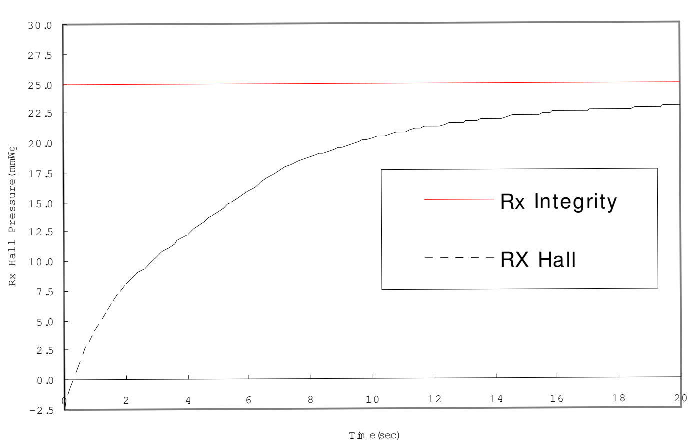

Fig 12. Pressure variation at reactor hall during LOCA