PTS Thermal-hydraulic Response
during a postulated MSLB accident of Kori Unit 1

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Abstract

The present study primarily aims to evaluate the system thermal hydraulic and thermal mixing behavior in downcomer of a postulated Main Steam Line Break (MSLB) as one of the major PTS initiators in PTS Evaluation of Kori Unit 1. For this purpose, the MSLB event sequences were reviewed and the most severe event sequence was selected based on the peer review on design and operational features. Based on the present design and operating condition of Kori Unit 1, a base calculation of the most severe sequence of MSLB events is conducted using RELAP5 and the sensitivity of the thermal-hydraulic mixing in downcomer was also investigated. The current result shows an overall downcomer fluid cooling from 558 ^{o}K to 436 ^{o}K. From the sensitivity analysis results, it is found that the thermal mixing could be affected by the downcomer modeling.

1. INTRODUCTION

Rapid cooling of the reactor pressure vessel (RPV) during a transient or accident accompanied by high coolant pressure has the potential of producing severe thermal stresses in the vessel wall and challenging the vessel integrity. This phenomenon is called overcooling or Pressurized Thermal Shock (PTS). The U.S. NRC (United States Nuclear Regulatory Committee) issued the Regulatory Guide (RG) 1.154, which specified the standard format and contents to evaluate the total risk from the PTS events in nuclear power plants [1].

The PTS evaluation is one of the most important issues in the Periodic Safety Review (PSR), which is effective regulatory rule for the existing nuclear power plants (NPP). Korean NPP utilities and vendors to propose the life extension of the old NPPs including Kori Unit 1 [2] have studied the PTS risk analysis. Among the events led to PTS risk, the MSLB event is one of the most contributing accidents to the PTS risk due to the excessive heat removal and significant PTS loading. Even though the occurrence frequency of MSLB is 6.4×10^{-3}/RY, according to the preliminary study, the degree of overcooling in MSLB could be severe and it could have a great impact on the total PTS risk. According to the PTS risk evaluation

procedure in RG 1.154, the result from the thermal-hydraulic analysis should be used in the analysis by the deterministic and probabilistic fracture mechanics, and the total risk should be determined with the occurrence frequency.

During the PTS event analysis, the cold emergency core cooling (ECC) water is forced to enter into the reactor vessel downcomer through cold legs and is mixed with hot water. When the RCS loop flow is low or stagnant, those hot-and-cold water may not be well mixed; i.e. thermal stratification. Particular attention to the potential of thermal stratification should be given to the thermal-hydraulic analysis. The present study primarily aims to evaluate the system thermal hydraulic behavior and thermal mixing behavior in downcomer of a postulated Main Steam Line Break (MSLB) as one of the major PTS initiators in PTS Evaluation of Kori Unit 1.

For this purpose above, the MSLB event sequences were reviewed and the most severe event sequence was selected from a previous work [3] on design and operational features of Kori Unit 1. The RELAP5/MOD3.2.2 beta [4], the most recent version code was used for the thermal-hydraulic calculation of the selected MSLB sequence. The version has improved capabilities such as reduction of mass error, time step control, which was believed as effective in predicting a stable calculation. The multiple channels modeling for downcomer was adopted for the simulation of the potential multi-dimensional hydrodynamic behavior affecting the thermal mixing. The effect of downcomer modeling schemes on the thermal mixing was also investigated to determine the mixing characteristics

2. EVENT SEQUENCE

The delineated event-tree is shown in Figure 2-1. The sequence in bold line is analyzed one in this study. The first heading of the event tree is "safety injection (SI) signal generated on demand," where the direct "demand" may be either high steam-line differential pressure or high steam flow coincident with either low steam pressure or low T_{avg}. The high steam flow signal will close the main steam isolation valves (MSIV), while the high differential pressure signal will not. If the steam-line break is upstream of the MSIVs, the only function of the MSIVs is to isolate the break from the other steam lines.

The next heading, "steam generators (SG) blow down," addresses the action of the possible closing of the MSIVs. This branch considers whether an MSIV closure signal would be generated owing to the break and whether the MSIVs would close if the signal is given. The net system response to the break and MSIV closures is presented in terms of the number of steam generators blowing down.

The next heading on the event tree, "main feed water (MFW) isolated on demand," comes from the main feedwater and condensate system state tree and is concerned with stopping the main feedwater flow. Among other things, the SI signal will send a signal to trip the main feedwater pumps, runs back the MFW control valves," close the MFW pump discharge valves, and prevent the MFW bypass valves from opening. A second important signal is high water level in any steam generator, which will do all of the above except close the MFW pump discharge valves. The final signal is reactor trip coincident with low average temperature T_{avg}, which only closes the MFW control valves.

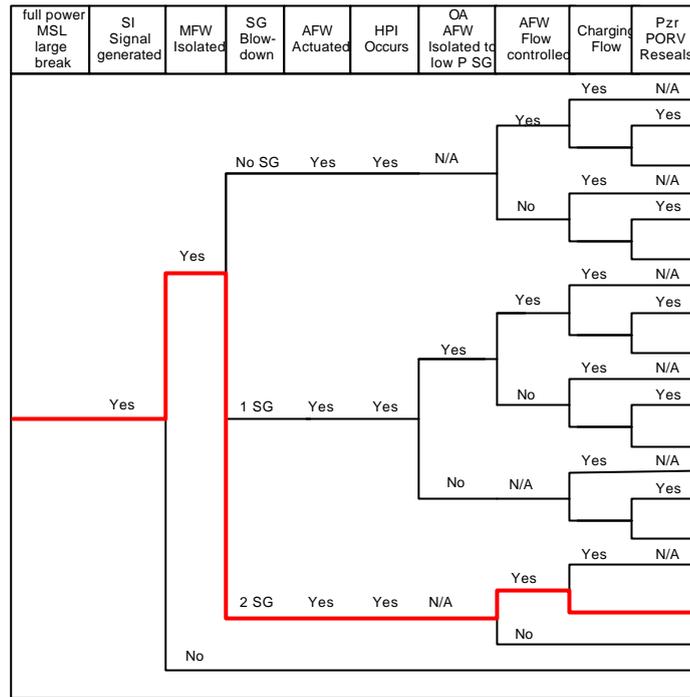| full power MSL large break	SI Signal generated	MFW Isolated	SG Blow-down	AFW Actuated	HPI Occurs	OA AFW Isolated to low P SG	AFW Flow controlled	Charging Flow	Pzr PORV Reseals

Figure 2-1. Sequence of Events for MSLB Transient

The next three headings are associated with defining auxiliary feedwater flow conditions. The first, "auxiliary feed water (AFW) actuates on demand," defines whether the auxiliary feedwater system is initiated. Once initiated, two potential conditions are considered under the heading "AFW flow automatically controlled": (1) flow controlled at a nominal flow rate or (2) a failure to automatically control, resulting in abnormally high flow rates (overfeed). The third heading, "AFW isolated to low-pressure SG," identifies whether auxiliary feedwater flow is isolated from the depressurized steam generator. It should be noted that this requires an operator action and is very important in minimizing the RCS overcooling.

The next branching, "high pressure injection (HPI) occurs on demand," addresses the initiation of SI flow as a result of an SI signal or an operator action.

Under the next heading, "Charging flow runs back on demand," control of repressurization via charging pump flow runback is addressed. Charging flow is run back automatically when the pressurizer water level is restored. Failure to run back automatically would result in challenging the PZR PORVs. Because the charging flow is controlled on pressurizer level rather than on pressure, it is conceivable that overpressurization could occur with resultant opening of the pressurizer PORVs. At this point, the operator can shut off the charging flow and monitor the repressurization caused by the thermal expansion of the primary system water, but because this sequence is extremely unlikely, no operator action was considered.

The final tree heading, "pressurizer PORV reseats on demand," is required because if the repressurization is not controlled (charging flow does not run back), the high pressure is assumed to lead to a PORV lift. Thus, the potential for a PORV failure to close must be examined. This failure to close includes mechanical failures to close and the failure of the operator to block the PORVs in a short period of time.

This case is one of the MSLB transients in which the secondary side is depressurized. The initiating event is a 0.3363 ㎡ break in the steamline of a SG. The selected transient scenario included the failure of closing the MSIVs, is the most severe one in a viewpoint of overcooling.

3. CODE AND MODELING

As mentioned previous, RELAP5/MOD3.2.2beta code was used to calculate the thermal-hydraulic behavior following MSLB. The RELAP5 code is an internationally well recognized best-estimate system transient analysis code, based on a non-homogeneous and non-equilibrium model for one dimensional two phase flow system. Basically, this code solves six field equations including constitutive models and correlations. It uses a partially implicit numerical scheme to permit economical calculation of system transients. The RELAP5 base model for the Kori Unit 1 was illustrated in Figure 3-1. The model consisted of 174 hydrodynamic volumes and 229 junctions. The RPV wall was simulated by 238 heat structures with six meshes in radial direction. It includes a reactor pressure vessel (RPV) with single channel core and single channel downcomer (base case), two loops represented by intact loop and broken loop with pressurizer, and ECCS systems connected to cold leg. To consider the effect of three-dimensional on the thermal mixing and the resultant temperature distribution of downcomer in detail, the reactor vessel downcomer is modeled with four azimuthal channels with and without inter-channel crossflow junctions. The detailed downcomer modeling is shown in Figure 3-2.
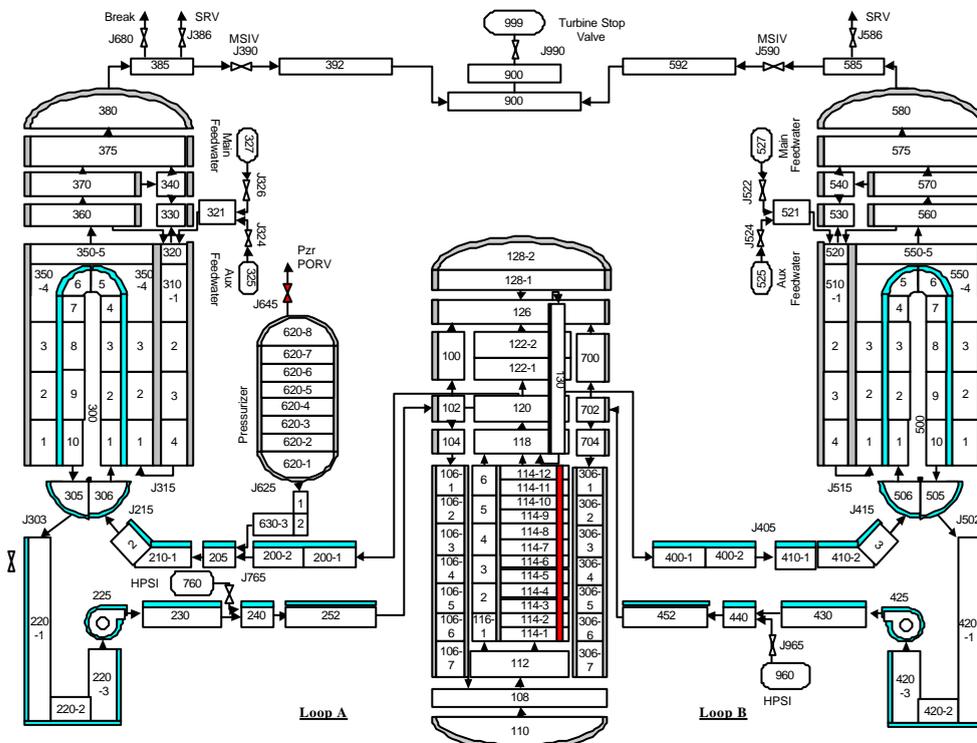

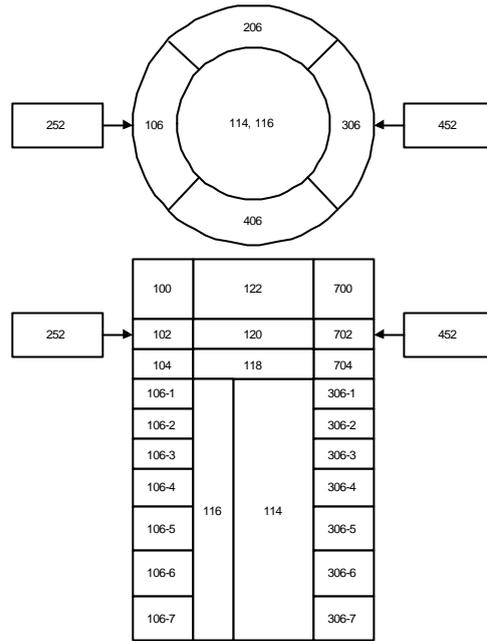Figure 3-1. RELAP5 nodalization for MSLB in Kori Unit 1

206

252 | 106 | 114, 116 | 306 | 452

406

100	122	700
252 →	102	120	702	← 452
104	118	704
106-1			306-1
106-2			306-2
106-3			306-3
106-4			306-4
106-5	116	114	306-5
106-6			306-6
106-7			306-7

Figure 3-2. Schematic Illustration of Downcomer Model

4. RESULTS AND DISCUSSION

4.1 Base Case

Steamline breaks are characterized as cooldown events due to the increased steam flow rate, which causes excessive energy removal from the steam generators and the reactor coolant system (RCS). The scenario of this transient is shown in Table 4-1.

In this transient the primary side loses energy to the steam generators, mainly to the affected steam generator. The depressurization of the steam generator caused a reduction in the saturation temperature, which increased heat transfer from the primary side and resulted in a larger vapor generation. The failure of the operator to close the MSIVs provided additional fluid to the steam generator for vaporization, and overcooling of the primary side.

Based on the modeling scheme described above, a RELAP5 steady state calculation was performed to initial condition over the system. The calculated condition indicated the RCS pressure of 15.5 MPa, fluid temperature at cold leg of 558 $^{\circ}$K, which were identical to the operating system value. The initial conditions used in the calculation are represented in Table 4-2. A base case transient from this steady state condition was calculated up to 8000 seconds.

Table 4-1. Sequence of Events for MSLB Transient

Time(sec)	ITEM	Reference
0.0	Break Initiation	Area : 0.3363 m²
3.0	SIAS signal High-high steam flow MSIV Isolation fail RCP & MFW Trip Reactor & Turbine Trip	Setpoint : 3.55 Mpa (low SG pressure) Setpoint : 521.9 kg/sec
7.0	SI flow initiation time	SIAS + 4.15s delay
63.0	AFWP actuated	60s delay after SIAS
107	Pzr empty time	
4016	AFW to SG-B terminated	Setpoint : level 96%
4480	AFW to SG-A terminated	Setpoint : level 96%

Table 4-2. Initial Conditions Used in Calculation

Major Parameters	Nominal Value	Calculated Value
Core Power	1730.2 MWt	1730.2 MWt
RCS Pressure	15.5 MPa	15.5 MPa
Hot Leg Temperature	592.6 °K	597.0 °K
Cold Leg Temperature	555.3 °K	558.2 °K
Feedwater Temperature	491.6 °K	491.6 °K
Steam flow rate	455.0 kg/s	471.0 kg/s

■ Plant Thermal Hydraulic Behavior

Figure 4-1 presents the primary and secondary pressure response. After SLB occurs, reactor coolant temperatures and RCS and steam generator pressures decrease. The low RCS pressure leads to reactor trip, SI injection signal, MFW isolation and RCP Trip. As the depressurization of the S/G reached to 0.14 MPa at 100 sec, the energy release to break became almost zero and the RCS pressure began to increase and reached to 15.2 MPa at 500 sec. After that, as RCS pressure increased, SI injection flow rate decreased and pressure increase terminated. The successive supply of AFW recovered S/G level and it led to RCS pressure decrease. As RCS pressure decreases, SI system was reactivated and RCS pressure was in the stable state at high pressure. In the primary system, the pressure decreased as the break-induced heat transfer to the SG A. As SG A emptied and the tubes became surrounded by high-quality fluid, the heat transfer to the SG degraded. The primary pressure decreased about 7.0 MPa. At 100 sec, the pressurizer began refilling, and the primary pressure began to increase and stabilized around 15.2 MPa. The pressure in SG A decreased continuously, until

reaching near atmospheric conditions (0.14 Mpa, 20 psia) at 100 sec. Unaffected steam generator (SG B) experienced the same depressurization as SG A, because of the failure of its MSIV closure. The pressure of SG A decreases faster than one of SG B because of a larger steam flow to break.

Figure 4-2 shows hot leg and cold leg temperatures. RCS temperature behavior is similar to pressure behavior. At the earlier stage, due to excessive heat removal to the break, cold leg temperature of broken loop is lower than one of intact loop. Between 100 and 150 sec when SG A is emptied, Loop A temperature is higher than Loop B temperature. This means intact SG steam flow rate is getting larger at that time.

Figure 4-3 shows the discharged steam flow rate from the SGs. After break, the steam flow rate of SG A is the 2~2.5 times that of the SG B at early stage, and then is almost identical to that of SG B at 40 sec. After that, SG A is emptied and discharged little. However, the SG B still had a liquid inventory and continuously discharged steam beyond the 100 second time frame. From this it is found that SG B removes more heat than SG A from the primary system at 100~150 sec.

Figure 4-4 shows the comparison of the SG levels. At 4000 sec, both SG-A and SG-B are filled with by 96% narrow level, which led to AFW terminated. Following this event, the SG heat removal capabilities are reduced and the SG level decreased slightly. By this level decrease, the AFW is automatically reactivated and revert the level to stable state.

Figure 4-5 shows RCS flow rate. RCS flow decrease rapidly after RCP trip and settled down around 136.1 kg/s, i.e., natural circulation flow rate over the loop. At early behavior, intact loop flow is lower than that of broken loop. As steam flow reversed, intact loop flow rate is getting higher. This is identical to the previously mentioned on in the temperature behavior.
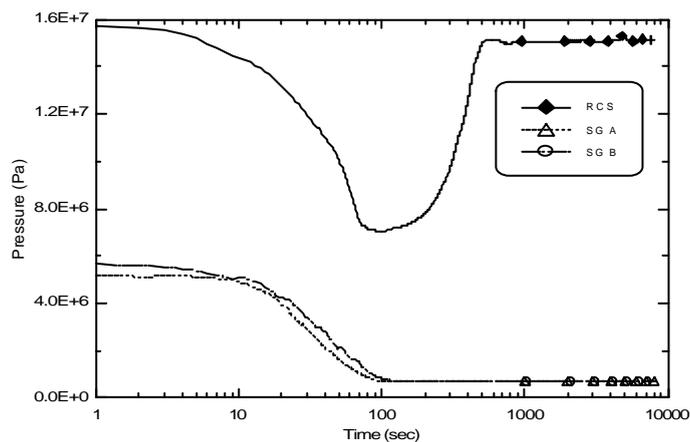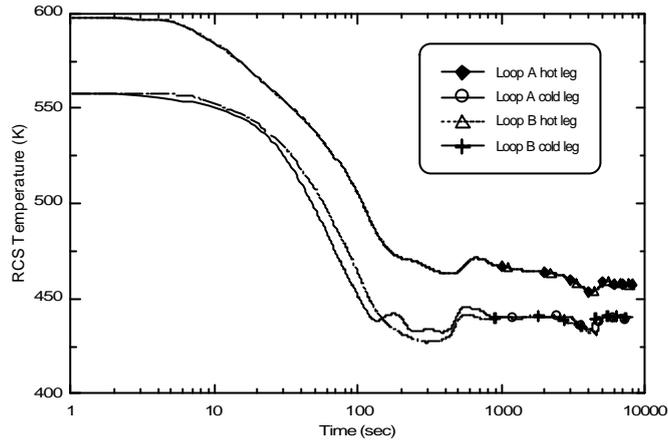

Figure 4-1. Primary and Secondary Pressure
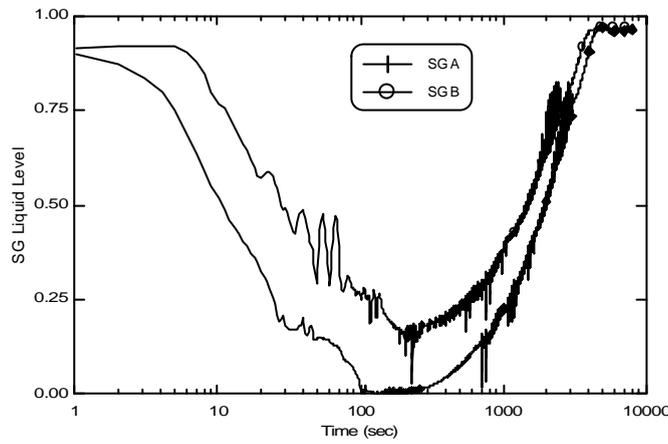
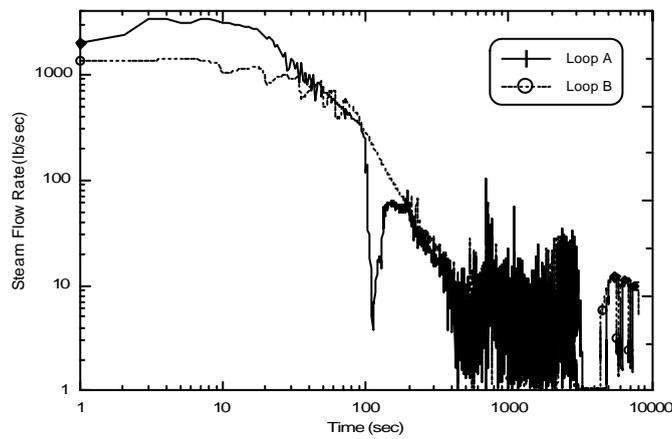Figure 4-2. RCS temperature

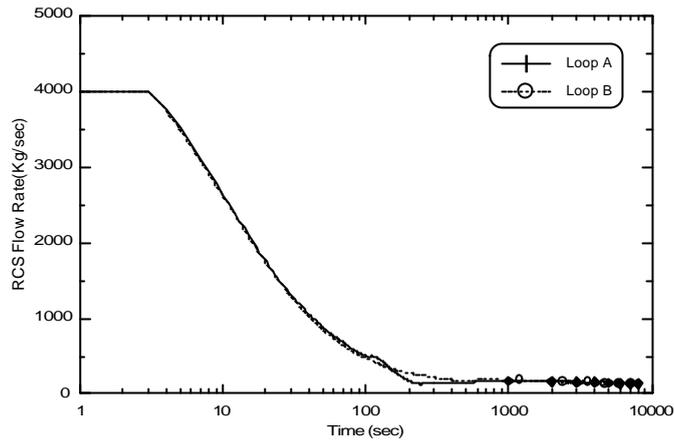Figure 4-3. Steam Flow Rate

Figure 4-4. S/G Liquid Level
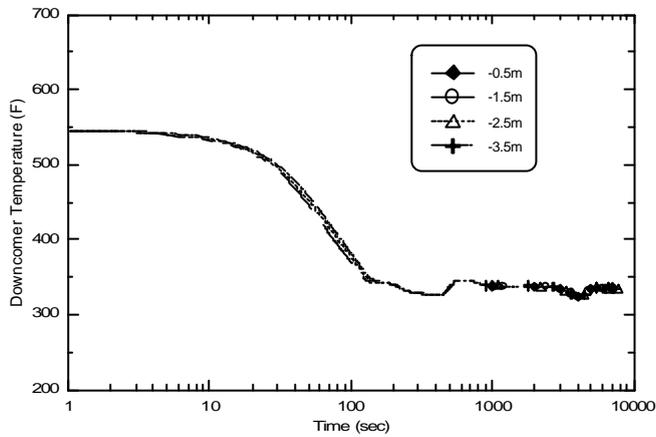
Figure 4-5. RCS Flow Rate

Figure 4-6. Downcomer Temperature

■ Downcomer Behavior

One of the most important parameters for PTS is the downcomer fluid temperature. A comparison of these temperatures at the different axial nodes as predicted is shown in Figure 4-6. The temperature behavior of downcomer with the time is similar to RCS temperature behavior as mentioned previously. Overall downcomer temperature is between 436 ^{o}K ~ 450 ^{o}K at 15.2 MPa for an extended period (longer than 7000 sec). From the calculation results, it is found that the temperature variation along the axial elevation is insignificant. After AFW injection stops, the axial temperature difference was a little observed. It is believed that these calculation results come from single channel downcomer modeling because it seems that fluid from loops A and B is fully mixed in a single channel. However, it is questionable that this instantaneous full mixing could occur in real situations.

4.2 Sensitivity Case

As mentioned above, to simulate multi-dimensional behavior in the downcomer, the reactor vessel downcomer was modeled to 4 azimuthal nodes and cross flow junctions were modeled between the separate channels. The used cross flow junction loss coefficients (f_J) were 0 (case 1), 100 (case 2), respectively. And the fully-separated channel modeling was also investigated (case 3). In this case, the downcomer was divided into 4 axial channels and cross flow junctions were not modeled artificially between the separate channels to find out the extreme azimuthal and axial temperature distributions induced by the limiting thermal stratification expected when the communication between channels is ignored .

From the calculation results, as it was confirmed that these changes in modeling scheme do not affect global parameter such as RCS pressure, temperature, etc, those parameter behaviors was not described.

The comparison between case 1 (loss 0) and 2 (loss 100) was not presented because the case 1 is almost similar to the base case (single channel) and also to case 2. This indicated that the loss factor up to 100 did not contribute to the thermal mixing behavior in downcomer.

Figure 4.7 ~ 4.8 show comparisons of the temperature distributions and the isotherms with axial and azimuthal direction for the cases 2 and 3, i.e, the multiple channel with crossflow of loss 100 and the fully-separated channel modeling.

From the comparison, the followings are found :

1)   For cross flow channel modeling ($f_J = 0$, 100), axial downcomer fluid temperature gradient is insignificant. The maximum azimuthal spread in the upper downcomer temperature was about 12 ^{o}K after AFW trip (base case).

2)   In cross flow channel modeling case, the thermal mixing was progressed through the communication between separate channels regardless of junction loss coefficient. It can be considered as the limitation of one-dimensional code such as RELAP5.

3)   For the fully separated channel modeling, the significant axial and azimuthal gradients are appeared. The maximum axial and azimuthal spread was about 46 ^{o}K and 64 ^{o}K respectively. Those thermal gradients were also decreased as progressing time and axial direction.

4)   The result from the case 3 was due to the artificial zero mixing between channels, therefore, it could be regarded as the limiting case of thermal stratification. Thus, it might be stated that the maximum thermal stratification under the selected MSLB event sequence was less than 46 ^{o}K in axial direction and 64 ^{o}K in azimuthal direction at maximum.
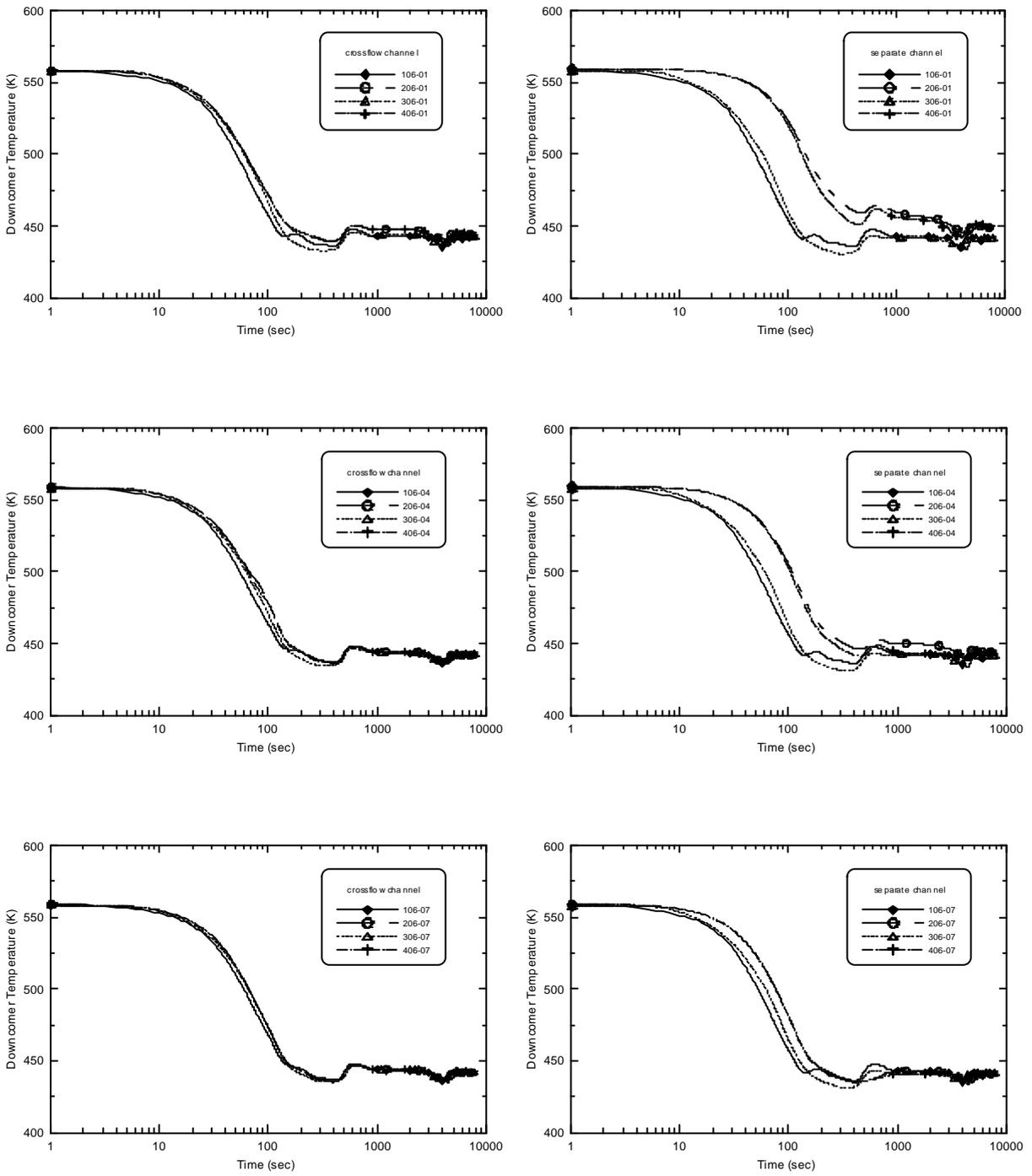
Figure 4-7. Comparisons of temperature distribution at downcomer

cross flow model(K=100.0)

At t=100 sec

At t=500 sec

separated channel model

At t=100 sec

At t=500 sec

Figure 4-8. Comparisons of the isotherm distributions at downcomer

5. CONCLUSIONS

Based on the present design and operating condition of Kori Unit 1, a base calculation of the most severe sequence of main steam line break (MSLB) events was conducted using RELAP5 code and the sensitivity of the thermal-hydraulic mixing in downcomer was also investigated. From the present analysis result, the following conclusions are obtained :

1) The MSLB event sequences specific to the Kori Unit 1 were reviewed based on the previous work [3] and the most severe event sequence was selected based on the peer review on design and operational features.

2) A RELAP5 modeling scheme, developed relevant to Kori Unit 1, could provide a thermal-hydraulic behavior reasonable to the MSLB analysis.

3) The current result shows an overall downcomer fluid cooling from 558 ^{o}K to 436 ^{o}K. Overall downcomer temperature is between 436 ^{o}K ~ 450 ^{o}K at 15.2 MPa for an extended period (longer than 7000 sec).

4) From the sensitivity analysis results, it is found that the thermal mixing could be affected by the downcomer modeling. The limiting case showed the maximum axial temperature gradient was 46 ^{o}K and the maximum azimuthal temperature was 64 ^{o}K based on the assumption of no mixing in downcomer.

5) Further investigation on the code options and cross flow modeling scheme is needed to identify the code capability in predicting the multi-dimensional thermal-hydraulic mixing phenomena. And the detailed computational fluid dynamics (CFD) analysis may be requested for simulating the real situation.

6. REFERENCES

1. US NRC, Regulatory Guide 1.154, Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors, Jan. 1987.

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3. KEPRI, Pressurized Thermal Shock Evaluation of Kori-1 Reactor Pressure Vessel, Interim Report, 1998.

4. INEL, RELAP5/MOD3.2.2Beta Code Manual, NUREG/CR-5535, Mar. 1998.