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Evaluation of Total Loss of Feedwater Accident/Recovery Phase and Investigation of the Associated EOP

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완전급수상실사고/복구과정의 평가와 관련비상운전절차의 검토

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Abstract

To evaluate the sequence of event and the thermohydraulic behavior during total loss of feedwater accident and recovery procedure, a RELAP5/MOD3 calculation is performed and compared with the LOFT L9–1/L3–3 experiment. Also, the predictability of the code for the major thermohydraulic phenomena following the accident is assessed. As a result, it is found that a pressure control using the spray until the time the water level reaches the top of the pressurizer, an overpressure protection by pressurizer PORV, a recovery of the secondary heat removal capability by refilling steam generator, and an effective cooldown by the continued natural circulation can be performed without core uncovery. It is also found that the plant-specific evaluation is necessary to confirm the effectiveness of the current symptom–oriented emergency operating procedure, especially in an overpressure protection performance and steam generator recovery performance.

요 약

완전급수상실사고 및 복구과정의 사고전개와 열수력학적 거동을 평가하기 위해 RELAP5/MOD3 계산을 수행하고 LOFT L9-1/L3-3 실험 결과와 비교하였다. 또한 본 사고의 주요 열수력 현상에 대한 코드의 예측도를 평가하였다. 본 연구의 결과로서 가압기 수위가 만수위에 도달할 때까지 살수를 이용한 압력 제어, 가압기 압력방출 밸브를 통한 과압방지, 증기발생기 재충수에 의한 이차측 열제거 능력의 재확보, 지속적인 자연순환에 의한 효과적인 일차계통의 냉각등이 이루어 질 수 있고 이 과정중 노심노출은 나타나지 않음이 밝혀졌다. 또한 현재의 현상–중심 비상운전절차서 특히 과압방지성능 및 증기발생기 회복절차 등의 유효성을 입증하기 위해서는 발전소 고유한 평가가 필요함이 밝혀졌다.

1. Introduction

The event of total loss of feedwater (TLOFW) in pressurized water reactor (PWR) is a non–design basis accident. However an attention has been moved up due to its potentiality and severity. During the TLOFW the coolant inventory in steam generator(SG) secondary side is reduced and the heat removal capability from the primary coolant system (PCS) is degraded significantly. As a result, the PCS temperature is increased and the primary coolant expands due to core decay heat, which in turn causes the PCS pressure to increase. The overpressure protection system such as a pressurizer spray system and/or a pressurizer power-operated relief valve (PORV) may be actuated and may prevent the PCS from overpressurizing for a while after following the TLOFW. Eventually, an appropriate procedure such as a primary feed-and-bleed or a secondary feed-and-bleed for a decay heat removal (DHR) should be performed to mitigate the consequence of the event without any core damage. The methodology and performance for DHR have been a major topic of the recent researches[1] since the DHR was issued as one of the unresolved safety issues USI A-45[2] in the United States Nuclear Regulatory Commission (USNRC).

According to the reference[1], a plant specific design applicable to the DHR methodology and a possibility of core uncovery during the DHR have two key problems to be resolved and the analysis of plant thermohydraulic(TH) behavior following an TLOFW as an initiating event is an essential part for the evaluations of overpressurization protection performance and of validity of the plant recovery procedure.

The purposes of this paper are to evaluate the capability of the RELAP5/MOD3 code in predicting the sequence of accident and the thermo hydraulic system behavior during TLOFW and to investigate a cooldown performance through the reactor trip, an overpressure protection performance through the pressurizer spray and PORV and an effectiveness of recovery procedure using the secondary feed-and-bleed following TLOFW based on the results from the calculation and experiment of L9-1/L3-3 in LOFT. The present study has also its aim at the investigation of emergency operation procedure(EOP) of the operating plant expected for a TLOFW accident based on the comparison with the present calculation and experiment.

2. Experiment

The LOFT L9-1/L3-3 experimental data are used in the present study, which was one of the tests simulating the TLOFW and its recovery procedure [3,4]. The LOFT facility is an experimental 50 MWt PWR, which is a scaled representation for a typical Westinghouse type PWR. The detailed design of LOFT system and its scaling features are described in reference [5]. The experiment L9-1/L3-3 was described in reference [6]. The sequence of event of the experiment L9-1/L3-3 is as follows:

1. The experiment was initiated at time 0 second with tripping off the main feedwater pump. The pressurizer heaters were deenergized prior to the test. And the reactor trip channel from SG low-low level signal was set not to actuate, which was simulated to investigate the PORV performance under high energy condition due to multiple failure, i.e, a failure of one trip channel. The reactor scram was consequently occured on high pressurizer pressure.
2. The Main Steam Control Valve (MSCV) was started to close after reactor scram under the pressure-controlled mode.
3. The pressurizer PORV held open at the time which the temperature of primary coolant reached 597°K and the RCP was tripped at the same time, which was an end of L9-1 experiment and a beginning of L3-3 recovery experiment. The PORV was set to fully open for duration of 1580 seconds (26.3 min.).
4. The secondary feed-and-bleed technique was simulated in the experiment L3-3, which used the auxiliary feedwater as 'feed' and steam

bypass valves as 'bleed.' The feed operation started at the time of PORV closure with delay of 265 seconds (4.4 min.) and continued to the secondary level 2.94 m above from the bottom of SG tube sheet.

5. After 966 seconds from the completion of SG refill, SG bleed operation started.

3. Calculation

The RELAP5/MOD3/5m5 computer code [7] was used to calculate the system behavior during the LOFT L9-1/L3-3 experiment. The code has been believed as a best-estimate code in calculating the plant transient such as a TLOFW as reported in the reference [8]. In the present calculation, the LOFT facility was modelled by 125 hydrodynamic volumes, 135 junctions and 136 heat structures. The RELAP5 nodalization for the present calculation of L9-1/L3-3 test was shown in Fig.1. It includes the hot legs and cold legs, the pressurizer, the primary coolant pump, the SG primary and secondary sides and the reactor vessel. And the component such as pressurizer PORV, pressurizer spray, MSCV, Condensor Bypass Valves, Main and Auxiliary Feedwater Control Valves, etc., important to the transient were properly modelled, as shown in Fig.1. This allows in-detail modelling of the primary side and secondary side. The detailed modelling scheme was described in reference [9]. To provide the initial conditions throughout the whole system as identical to the experiment, a RELAP5 steady state run was performed. The resulting thermo-

Table 1. Comparison of Initial Conditions Between L9-1/L3-3 Experiment and Calculation

Parameter	Measured	RELAP5/MOD3
Primary Coolant System		
Mass Flow Rate, kg/s	479.1	479.34
Hot Leg Pressure, MPa	14.9	14.89
Cold Leg Temperature,°K	558.9	559.13
Hot Leg Temperature,°K	578.2	578.33
Reactor Power Level, MW	49.6	49.6
Maximum LHGR, KW/m	50.8	50.8
Steam Generator Secondary Side		
Water Level, m	3.01	3.05
Water Temperature,°K	545	542.38
Steam Pressure, MPa	5.67	5.72
Steam Flow Rate, kg/s	27.0	26.7
Broken Loop		
Hot Leg Temperature,°K	563.3	559.14
Cold Leg Temperature,°K	557.6	558.38
Pressurizer		
Steam Volume, m^3	0.43	0.41
Liquid Volume, m^3	0.5	0.52
Water Temperature,°K	614.9	610.4
Pressure, MPa	14.93	14.90
Liquid Level, m	0.92	0.96

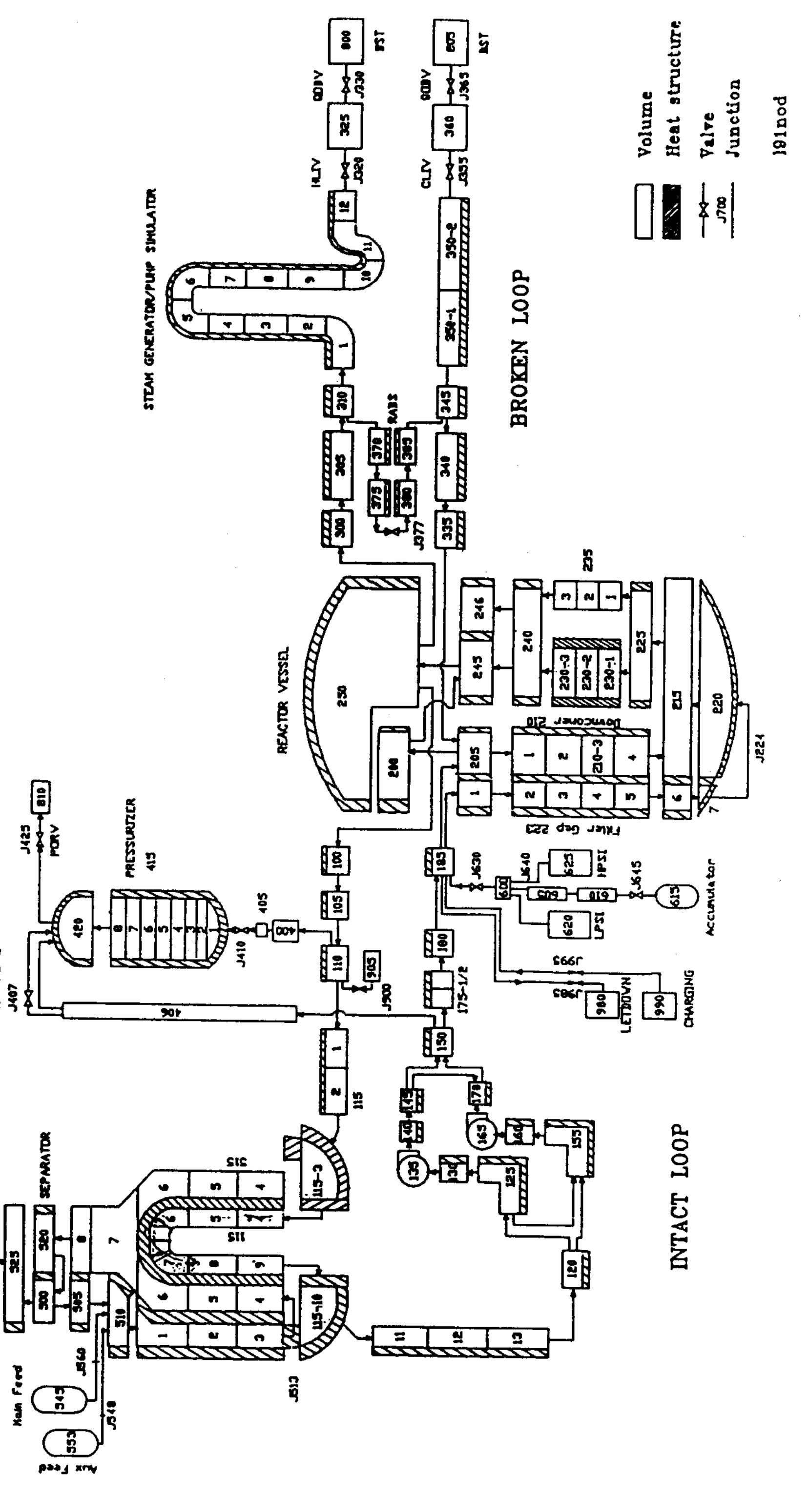

Fig.1. RELAP5 Nodalization for Simulating the LOFT L9-1/L3-3 Experiment

-hydraulic conditions predicted by the code run were well agreed to the experiment as shown in Table 1.

The sequence of event for the L9-1/L3-3 experiment was also described as the calculational boundary condition. This included the pressurizer spray valve control logic, the pressurizer PORV control logic, the high pressure and high temperature reactor trip logic, the main feedwater pump control logic, the auxiliary feedwater control logic, the steam bypass valve control logic, the MSCV control logic, and the PCS pump control logic. Additional boundary condition such as the reactor scram table and the moderator temperature feedback was also specified as accordance with the reference [6].

The calculation was performed up to 8100 seconds following a TLOFW, which corresponded to the SG bleed period.

4. Result and Discussion

4.1. Total Loss of Feedwater

Figure 2 shows a short-term comparison of the primary coolant pressure between the experiment and the RELAP5/MOD3 prediction. According to the experimental result, due to the degradation of the secondary heat sink following a TLOFW, the primary coolant pressure increases, which actuate the pressurizer spray valve to open around 23 seconds. The spray flow from the cold leg slightly lowered the primary coolant pressure, and then the PCS was repressurized by the continuous heat from the reactor core. The PCS repressurization was stopped at the reactor trip time. The 2nd or 3rd spray actuation can be found during this period. After the trip, the cooldown of the PCS was observed and then the PCS reheatup due to the core decay heat was found.

This behavior was overally found in calculation. However, the RELAP5/MOD3 shows an earlier trip time than the experiment and underpredicts the pressure transient after 50 seconds, approximately, i.e. an excessive cooldown.

The reason for those differences in trip time and post-trip cooldown can be considered as an incorrect feedback of the effect of moderator temperature coefficient(MTC). The calculated coolant temperaure was well agreed to the experiment until 50 seconds following the TLOFW, as shown in Fig.3. However, the calculated reactor power was slightly lower than the experiment, as shown in Fig.4, which indicates that the MTC

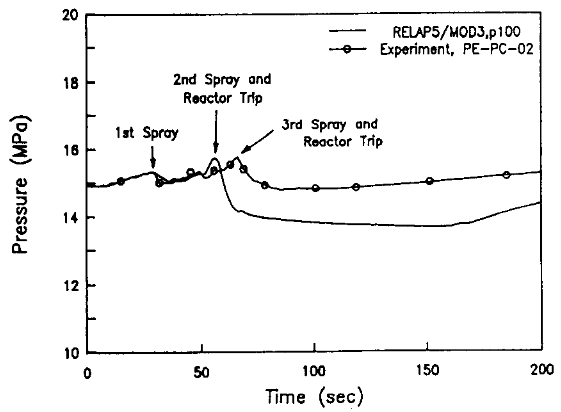

Fig. 2. Comparison of Primary Coolant Pressure (Short Term)

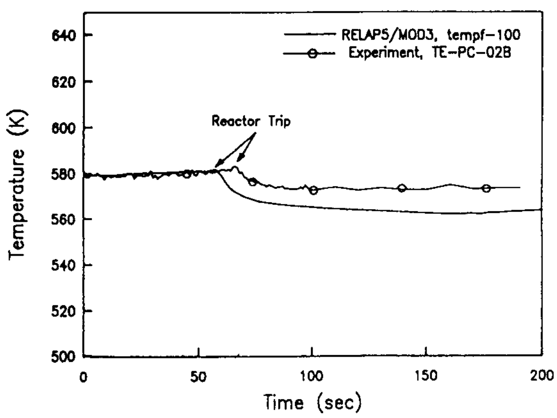

Fig. 3. Comparison of Primary Coolant Temperature (Short Term)

feedback was incorrectly considered in the calculation. Afterwards, the coolant heat up and the resultant power excursion were found in experiment, while those phenomena were not found in the RELAP5/MOD3 calculation. Considering that the temperature increase from initial condition to the trip setpoint was almost identical in both the experiment and calculation, one can conclude that the MTC feedback effect was incorrectly considered over the temperature range of interest in the calculation. Such a power decrease due to MTC feedback was explicitly observed in the experiment, and not in the calculation, as shown in Fig.4, which led the overprediction in power drop from the trip time to the decay heat level, i.e., 46 MW to 2 MW in calculation, 39.4 MW to 2 MW in experiment, approximately. Such an overprediction was one of the reason for the excessive post-trip cooldown. Figure 4 includes only the fission power obtained from the neutron detector, RET-A77, therefore the measured decay power was not available in this figure. The actual decay power can be considered as the same as that in the calculation which uses the ANS-79 model for the decay power.

Figure 5 shows a short-term comparison of the SG steam pressure between the experiment and the RELAP5/MOD3 prediction. As shown in the

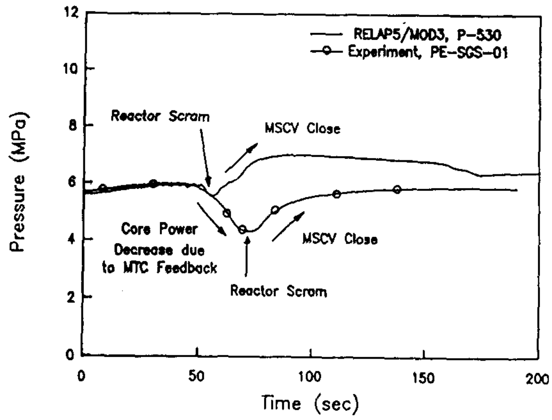

Fig. 5. Comparison of S/G Steam Pressure (Short Term)

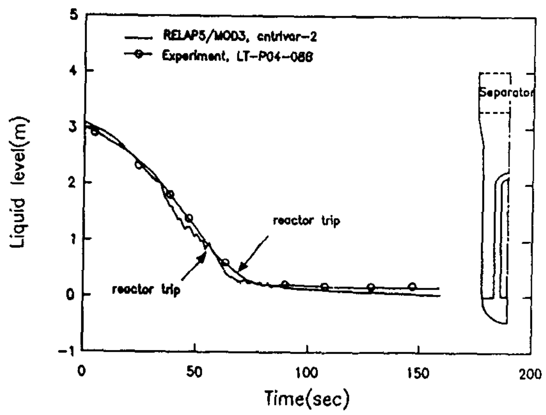

Fig. 6. Comparison of S/G Secondary Side Level (Short Term)

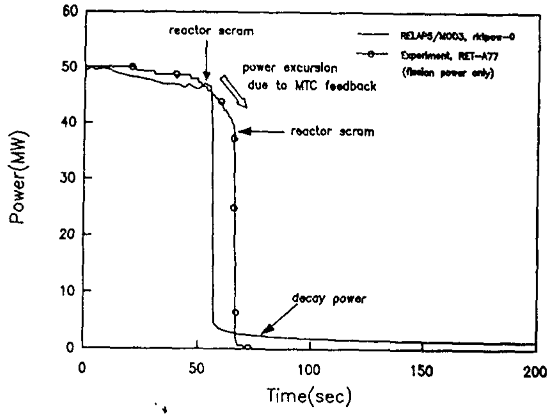

Fig. 4. Comparison of Reactor Power (Short Term)

experiment data, the secondary coolant system (SCS) pressure increased following a TLOFW, and then decreased due to the power decrease as described above. At the reactor trip time, the MSCV started to close and the SCS pressure increased. The discrepancy between the calculation and the experiment was due to the same as described above.

Figure 6 shows a short-term comparison of the SG secondary liquid level between the experiment and the RELAP5/MOD3 prediction. As shown in the figure, the collased liquid level decreased after a TLOFW, and the complete dryout was observed at about 75 seconds. The good agreement between the calculation and the experiment was found.

Figure 7 shows the variation of SG heat removal capability with SG liquid level, which was predicted by RELAP5/MOD3 up to 80 seconds. The calculation shows that the relationship between the SG secondary inventory and the SG heat removal performance during this period is not linear and that the severe degradation of the performance ranges from 1.0 m (40 % height of SG U-tube) to the bottom of U-tube.

Figure 8 shows a mid-term comparison of the PCS pressure between the RELAP5/MOD3 calculation and the experiment. According to the experiment data, the PCS pressure started to increase due to core decay heat and reached the opening setpoint of the pressurizer spray valve, 15.338 MPa at 208.9 seconds. At that time the spray valve was open to inject subcooled water from cold leg into the pressurizer, which condensed the steam in the upper part of the pressurizer with lowering pressure. When PCS pressure moved down to 15.05 MPa, which was a closure setpoint of the spray valve, the spray valve was closed and PCS pressure was increased. Such an valve opening and closure was repeated with showing a saw-tooth behavior in pressure (spray cycling). During the spray cycling period, the pressurizer liquid level was increased due to a thermal expansion of primary coolant. The spray cycling was ended at 1089.7 seconds (18 min.), which corresponds to the time that the pressurizer liquid level reached 1.83 m, i.e, an elevation of spray valve, as shown in Fig.9.

After a completion of spray cycling, PCS pressure was increased and reached the PORV opening setpoint, 16.06 MPa at 1467.9 seconds. From that time the PORV cycling was initiated in a similar way to the spray cycling. The primary coolant also

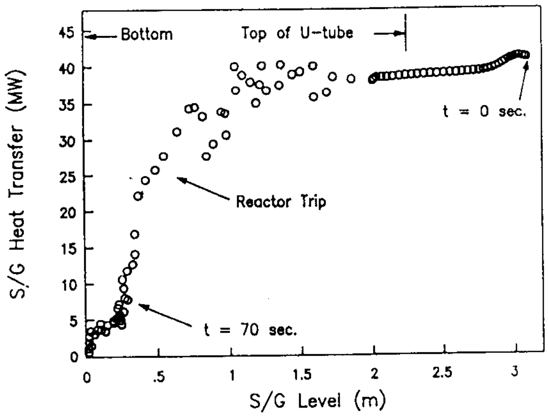

Fig. 7. Variation of S/G Heat Removal Capability

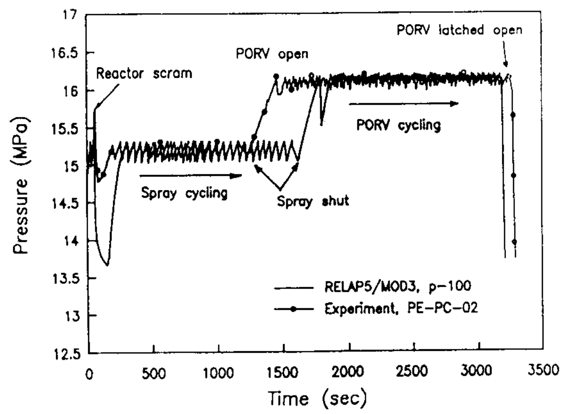

Fig. 8. Comparison of Primary System Pressure (Mid-Term)

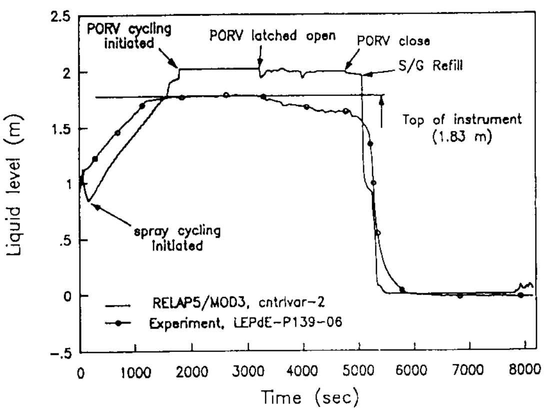

Fig. 9. Comparison of Pressurizer Liquid Level (Long Term)

continued to heatup as shown in Fig.10. The PORV was manually latched open at the time the temperature of PCS reached 598°K (3270 seconds). The durations of the spray cycling and the PORV cycling were found to be 881 seconds (13 min.) and 1800 seconds (30 min.), respectively. The maximum mass release through the PORV amounted to 2.8 kg/sec, which was a major contributing factor for decay heat removal during this period.

The RELAP5 calculation also predicted these cycling behavior well as shown in Fig.8. The discrepancies between calculation and experiment during this phase are the initiaion/ending time of spray/PORV and the overprediction in heatup rate. The reason for the later starting and later ending of the spray cycling can be considered as an excessive post-scram cooldown as described in previous section. The reason for the overprediction in coolant heatup rate during cycling phase was considered as underestimation of heat transfer to the SG secondary side. The experimental results indicated that a small amount of steam was leaked from the SG through the MSCV, which was not completely closed. It was considered as another heat sink during this period, however, the present RELAP5/MOD3 model did not take this point into consideration.

4.2. Recovery phase

Figure 11 shows a comparison of PCS pressure between the calculation and the experiment after the pressurizer PORV latched open. According to the experiment data, the PCS pressure dropped rapidly upon the PORV open, and the subcooled liquid was discharged out through the PORV. When the PCS pressure moved down to about 12.0 MPa, the voiding occurred in upper plenum of the reactor vessel and top of the SG U-tube and then in hot leg, consequently. This caused the transition in PORV discharge flow from the subcooled to the two-phase and therefore reduced the discharge rate and the depressurization rate significantly. The PORV was held open for 1580 seconds (26.5 min.) and then manually closed. The temperature of primary coolant, as shown in Fig.10, was decreased to 590°K, approximately, due to mass and energy release through the PORV during this 1580 seconds. Since the primary coolant pump was tripped simultaneously with PORV open, the loop mass flow rate was rapidly dropped and the hot leg was stratified.

After 265 seconds (4.3 min.) from the PORV

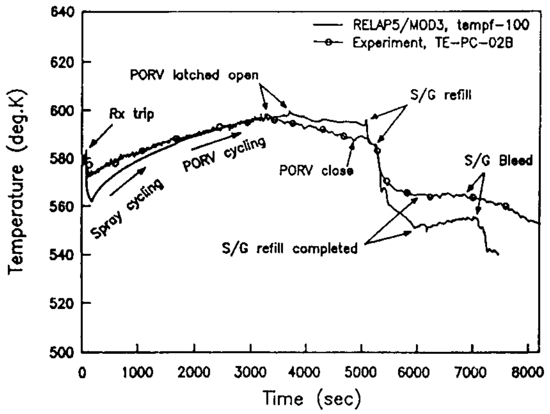

Fig. 10. Comparison of Hot Leg Coolant Temperature

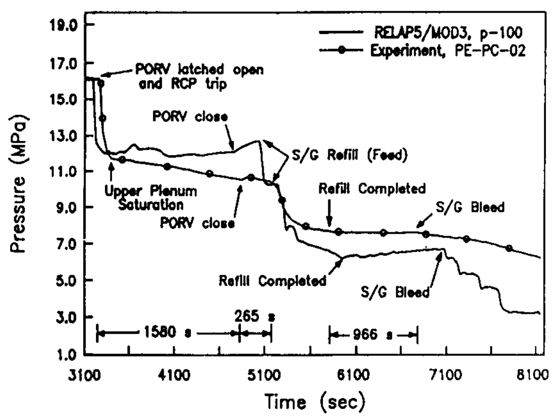

Fig. 11. Comparison of Primary Coolant Pressure (Recovery)

closure, the SG refill was initiated by the auxiliary feedwater and continued until the time when the SG secondary side liquid level reached 2.95 m above the tube sheet. Figure 12 shows the comparison of SG liquid level. The SG refilling initiated the SG heat sink capability, which established a natural circulation flow over the PCS loop, thus the PCS pressure and temperature were decreased to 7.5 MPa and 570 K, approximately due to a natural circulation cooling. The SG refill was ended at 5746 seconds and then the SG steam bleed was initiated by the operation of steam bypass valve after 966 seconds from the completion of the SG refill. This reestablished a natural circulation flow in PCS.

In the RELAP5 calculation, all of the major TH phenomena were fairly predicted, however, some differences were found as follows:

First, the PCS pressure was overpredicted after the transition phase (from 3200 seconds to 5000 seconds). The reason for such a deviation in PCS pressure can be considered as an insufficient energy release through the PORV during the two-phase discharge phase. This behavior can also be found in the coolant temperature in Fig.10. However, considering that the mass discharge flow rate through the PORV was overpredicted during this period as shown in Fig.13, one can conclude that the insufficient energy discharge was due to underprediction of the discharged enthalpy per unit mass, i.e., discharge flow in low quality. Based on the observations above, it is found that the reason for the insufficient energy discharge is a low specific enthalpy of the discharged flow through the PORV, which indicated that the PORV junction quality was underpredicted. This problem may be related to the interfacial drag model at two-phase flow regimes in RELAP5/MOD3 code.

Second, the drops in PCS pressure and temperature during the natural circulation phase were overestimated. The calculated cold leg mass flow rate after the primary pump trip was shown in Fig.14. The loop flow rate was not available in the experiment. According to this figure, a flow reversal was found after the pump trip, which may be induced by PORV open, and then a favor loop flow was reestablished, whose magnitude was 60 kg/sec at maximum. The proportionality in cooling performance of natural circulation to the loop flow rate was well known. Therefore, the overestimation of natural circulation cooling was related to the overprediction of loop mass flow rate, and the reason for such an overprediction can be

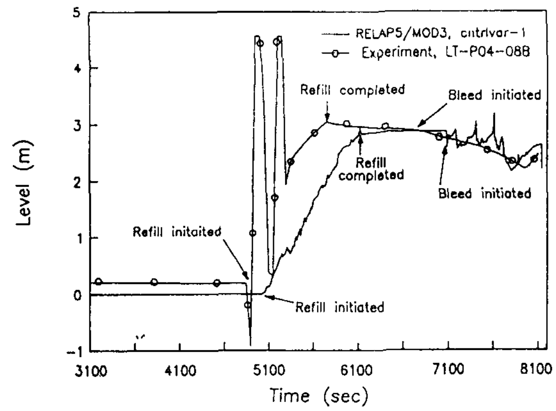

Fig. 12. Comparison of S/G Secondary Liquid Level (Recovery)

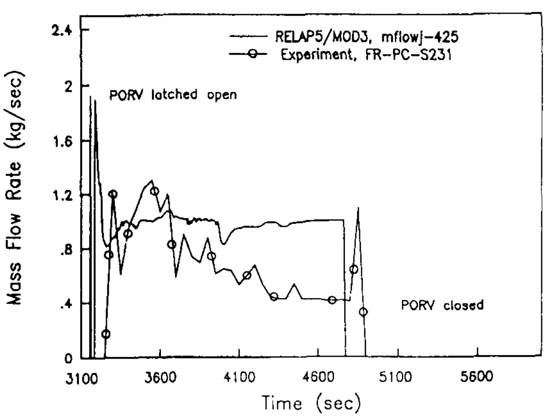

Fig. 13. Comparison of Mass Flow Rate Through RORV

considered as excessive heat transfer to SG secondary side especially at the direct contact condensation mode. In the SG bleed phase, almost similar behavior can be found.

Figure 15 shows the calculated liquid level in the reactor vessel. The measurement of this parameter was not available. From this figure, it is shown that the liquid level did not drop to top of the core, i.e. no core uncovery, despite of the overestimation of cooldown performance which may lead to an excessive voiding in primary coolant.

Based on the discussion above, the phenomena important to TLOFW were identified in both the experiment and the calculation. Table 2 summarizes the major phenomena in the primary system and secondary system for each event following a TLOFW. In summary, the major TH phenomena and their effect on the performance following a TLOFW and recovery procedure can be identified as follows :

1. Reactor power decrease due to MTC feedback over coolant heatup,
2. Significant degradation of heat removal capability due to SG complete dryout,
3. Pressure control using the pressurizer spray actuation up to the top level of the pressurizer,
4. Protection of overpressurization by discharging the coolant through the pressurizer PORV and the depressurization,
5. Recovery of the secondary heat removal capability with the direct-contact condensation heat transfer during SG refill phase,
6. Effective RCS cooldown by the continued natural circulation,
7. Long term reestablishment of secondary heat sink by the SG bleed.

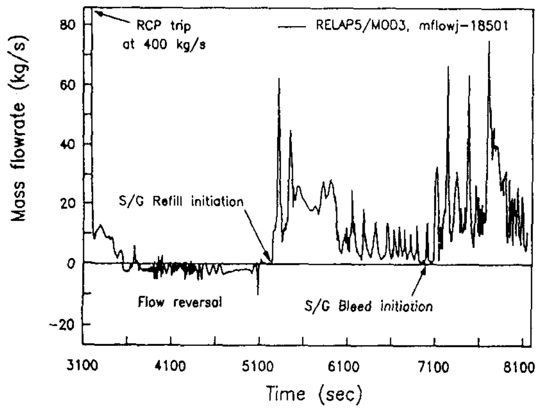

Fig. 14. Mass Flowrate at Cold Leg After RCP Trip

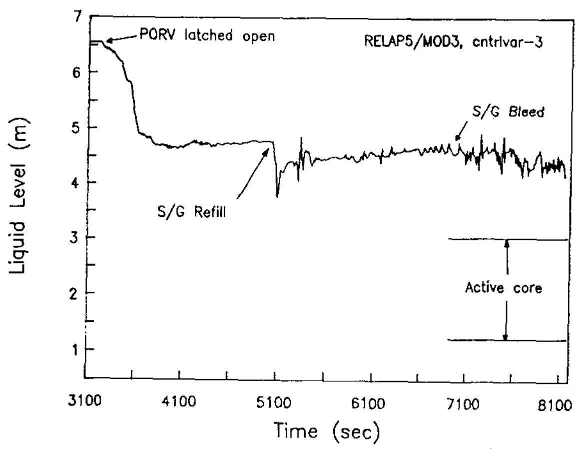

Fig. 15. Collapsed Liquid Level of Core in Reactor Vessel

4.3. Investigation of Plant EOP

The expected emergency procedure in response to the event such as TLOFW in the existing plant can be described as Table 3, according to the Kori Units 3/4 symptom-oriented EOP. Analysis results from the LOFT experiment and calculation were also compared in this table. For the TLOFW in real plant, an operator may take an action according to the E-0 (Reactor Trip) in the EOP, through the step 4.0 (Check SI Required) of E-0, and reached A-0.1 (Reactor Trip and SI). In A-0.1 in the EOP, operator may actuate Auxiliary Feedwater, otherwise may follow H-1 (All Loss of SG Heat Sink) of the EOP.

From the comparison, it is known that an overall sequence expected in the plant EOP is almost identical to the sequence of event in the LOFT L9-1/L3-3 experiment and in the calculation.

Table 2. Major Phenomena Identified in Experiment and Analysis

Event	Primary Phenomena	Secondary Phenomena
Total loss of feedwater	Heatup & spray actuation, Reactor power excursion due to MTC feedback	SG level boiloff
Reactor trip	Post–scram cooling, Pzr Level Shrink, Heatup and pressure buildup without heat sink	Complete dryout, MSCV closure and repressurization, Slow depressurization due to MSCV leakage
Spray cycling actuation	Coolant heatup, Pressure oscillation, Pzr level swell	Slow depressurization due to MSCV leakage
PORV cycling actuation	Cyclic mass & energy release, Pressure oscillation Coolant heatup	Slow depressurization due to MSCV leakage
PORV latched open and RCP trip actuation	Subcooled discharge flow, Rapid depressurization, Coolant voiding, Hot leg stratification, Two–phase discharge flow	Slow depressurization due to MSCV leakage
PORV close	Pressure re–buildup, Flow stabilization	
SG refill initiation	Rapid depressurization and cooldown, Pzr level re–distribution Core level shrink, 2–ϕ Natural circulation	SG level increase and oscillation, Direct–contact condensation
SG refill completed	Breakdown of natural circulation	SG level stabilization
SG bleed initiation	Reestablishment of 2-ϕ natural circulation	SG level decrease with oscillation

Especially, the automatic actuation of the pressurizer spray and PORV as an overpressure protection and the secondary feed–and–bleed as a reestablishment of heat sink were almost identical to the sequence of the present experiment. The only difference between the EOP and LOFT experiment is the AFW operation immediately after reactor trip. In the plant EOP, AFW operation should be confirmed immediately after reactor trip and otherwise operator should actuate AFW manually.

For the case in which an AFW is not available just after reactor trip, the H–1 of the symptom-oriented EOP requires reestablishment of AFW. Consequently, it is necessary to determine the

Table 3. Comparison Between Emergency Operating Procedure and LOFT Experiment Sequence of Event

Symptom Oriented EOP(Kori Units 3/4)	LOFT L9-1/L3-3 Experiment/Analysis Event-oriented Sequence of Event
	– Loss of MFW (MFW pump off)
E-0 Reactor Trip	
	– Pzr Spray Automatically Activated
1.0 Reactor Trip	– Reactor Trip
– MFW Isolated	– MFW Isolated
– Aux. Feedwater Start	– Auxiliary Feedwater Failure
(If not, Manually Actuation)	
2.0 Turbine Trip	
– MSIV Closed	– SG MSCV Closed
3.0 Power onto Emergency AC	
4.0 Check SI Required	
(If not, Go to A-0.1)	
A-0.1 Reactor Trip & SI	
1.0 $Temp_{rcs} > 292$℃	
–SG PORV or Dump Valve :	– RCS Temp. Continues to Rise
Manual Operation	
2.0 $Temp_{rcs} > 296$℃	
– Check Aux. FW 35 l/sec	– Pzr Level Continues to Rise
(if not, Go to H-1)	
3.0 Check Control Rods Full Inserted	
4.0 Check Pzr Liquid Level>22 %	
(If Not, Charging/Letdown)	– Pzr Spray Actuation/Cycling
5.0 Check Pzr Pressure>157 kg/cm^2	– Pzr PORV Actuation/Cycling
– Pzr Heater Off	
– Pzr Spray Auto Actuation	– RCS Temp. Rise to 597°K
– Pzr PORV Operation	– Pzr PORV Latched Open
	– RCP Trip
	(End of L9-1)
	(Beginning of L3-3)
	– Depressurization for 1580 sec.
	– Stabilized Period for 265 sec.
6.0 Establish SG Inventory	– SG Secondary Refill Initiated
(6~50 % in Narrow Range Level)	(1.2 kg/sec by Aux. Feedwater)
– Actuation of Aux. Feedwater	– SG Level Rise up to 2.94 m
(<35 l/sec.)	– SG Refill Completed
	– Stabilized Period for 966 sec.
	– SG Secondary Bleed Initiated
	(SG Turbine Bypass Valves)
8.0 SG Dump to Condenser	
(If Failed, Use SG PORV)	
9.0 Check Natural Circulation Flow	

plant-specific capability to endure an overpressurization until the time AFW was reestablished. Based on the LOFT L9-1/L3-3 experiment and calculation, it is known that the LOFT system can endure the TLOFW accident for about 3000 seconds using the pressurizer spray/PORV without any AFW after reactor trip. Such a capability may be expected to be achieved by the pressurizer spray/PORV in real plant, however, a plant-specific evaluation using the real plant data should be performed to confirm the pressurizer spray/PORV capability and the effectiveness of current EOP.

The recovery procedure using secondary feed-and-bleed was also expected to be effective to cooldown the plant, however, it should be reconfirmed by the plant-specific evaluation.

5. Conclusions

A RELAP5/MOD3 calculation was performed and compared with the LOFT L9-1/L3-3 Experiment to evaluate the sequence of event and the thermohydraulic behavior following a Total Loss of Feedwater Accident and Recovery procedure. And the predictability of the code for major thermohydraulic phenomena during the accident were assessed. As a result, the important phenomena following a TLOFW were identified based on those experiment and calculation. From this study, the following conclusions are obtained:

1. In the present RELAP5/MOD3 code prediction of the major phenomena following a TLOFW, some differences from the LOFT L9-1/L3-3 experiment were identified such as an excursion of reactor power due to MTC feedback, an excessive cooldown after reactor trip, a slightly higher heatup rate of coolant during pressurizer spray/PORV cycling phase, and an excessive natural circulation cooling during SG refill phase. However, it is also found that the code can predict appropriately the overall plant behavior, especially the reactor trip at RCS high pressure, the pressurizer spray/PORV cycling, the complete dryout of SG and the procedure of secondary feed-and-bleed.
2. The major TH phenomena in system behavior following a TLOFW and recovery procedure can be identified as follows : the reactor power decrease due to MTC feedback over coolant heatup, the significant degradation of heat removal capability due to SG complete dryout, the pressure control using the pressurizer spray actuation up to the top level of the pressurizer, the protection of overpressurization by discharging the coolant through the pressurizer PORV and the depressurization. Also, the secondary heat removal capability was recovered with the direct-contact condensation heat transfer during SG refill phase, the RCS cooldown was effectively achieved by the continued natural circulation, and secondary heat sink was reestablished in the long-term by the SG bleed.
3. From the present calculation, it is found that the pressurizer spray and PORV can protect the RCS from overpressurizing for a certain period following a TLOFW. Also the secondary feed-and-bleed as a recovery procedure was effective in cooldown without core uncovery. For the application of those results to the real plant, the plant-specific calculation is necessary with the plant-specific data such as real sizes of plant equipments and specific flow rates of the valves, etc.

In addition, the sequence of event analyzed was compared to the applicable symptom-oriented EOP in Kori Units 3/4. The plant EOP includes the specific procedure to reestablish a heat sink using the secondary feed-and-bleed as a recovery procedure following TLOFW accident. The pressurizer spray/PORV was effective in protecting from overpressurization until actuation of AFW in the LOFT case, however, the plant-specific capability should be evaluated to confirm the effectiveness of the current symptom-oriented EOP.

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