Proceedings of the Korean Nuclear Society Autumn Meeting
Seoul, Korea, October 1998

An Application of RELAP5/MOD3 to the Post-LOCA Long Term Cooling Performance Evaluation

Young Seok Bang, Jae Won Jung, Kwang Won Seul, and Hho Jung Kim
Korea Institute of Nuclear Safety

Abstract

A realistic long-term calculation to be used in the post-LOCA long term cooling (LTC) analysis is described in this study, which was required to resolve the post-LOCA LTC issues including the concern on boric acid precipitation in the reactor core. The analysis scope is defined according to the LTC plan of UCN Units 3/4 and the plant calculation model are developed suitable to the LTC procedure. The LTC sequences following the cold leg small break LOCAs of 0.02 ft^2 to 0.5 ft^2 are calculated by RELAP5/ MOD3.2.2. Based on the calculation results, the establishment of shutdown cooling system entry condition and the behavior of boron transport are evaluated. The effect of model simplification is also investigated.

I. Introduction

Long term cooling (LTC) after a loss-of-coolant-accident (LOCA) is initiated when the core is quenched and terminated when the plant is secured. The objectives of LTC are to maintain the core at a safe temperature level and to avoid the precipitation of boric acid in the core region, which were required in the acceptance criteria on the emergency core cooling system (ECCS) performance in light water reactor (LWR) [1]. In the design of the Korean standard nuclear power plants including the UCN Units 3/4 [2], two kind of LTC strategies to meet those LTC requirements were adopted : for the small break LOCA, an overall cooldown using steam generators (SG) is used to bring the reactor coolant system (RCS) to the shutdown cooling system (SCS) entry condition ; for the large break LOCA, a simultaneous injection to both cold legs and hot legs is activated to avoid the boric acid precipitation, which were based on the LTC plan of the Combustion Engineering (CE) type pressurized water reactor (PWR) [3].

Safety concerns of the LTC were to determine if the boric acid precipitation is avoided using the simultaneous hot leg/cold leg injection and if such a injection can be achieved in the emergency core cooling system (ECCS) design of the plant [4]. The boric acid precipitation may be a treat to the continuous core cooling. To resolve those issues, a long-term calculation following a LOCA and the evaluation of the plant thermal-hydraulic behavior are requested. The Final Safety Analysis Report of UCN Units 3/4 [2], for an example, indicated such a calculation more than nine hours in real time was required. In the calculation, the applicable emergency operation procedures (EOP) including the steam dump operation, the auxiliary feedwater supply, the hot leg injection, etc., as well as break flow should be considered. Boron behavior, also, should be predicted in the calculation. Those considerations increases the computational difficulty. Therefore, a simple method with conservative assumptions has been developed and applied to the current design [3] for the purpose of providing a compliance of the safety criteria. However, it is still questionable if the LTC plan is realistically effective and how much margin in boron precipitation could be available using the simple and conservative LTC method because the simple LTC method was not based on the realistic behavior prediction such as core boil-off process.

The present paper aims to provide a realistic long-term calculation to be used in the LTC behavior analysis. The RELAP5/MOD3.2.2 code [5] was used in the calculation, the code was improved in the computational time step control, etc., which may be effective in this kind of long-term calculation. Two types of plant model (a detailed model and a simplified model) of the UCN Units 3/4 were tested to compare the calculation efficiency. Boron behavior was also discussed based on the calculation results on the small break LOCA.

II. Review on LTC Plan

To define the analysis scope on the post-LOCA LTC performance evaluation, the LTC plan described in the FSAR of UCN Units 3/2 was summarized as in Table 1. Operator actions in the applicable steps in the EOP [6] were compared in this table.

Table 1. Post-LOCA Long Term Cooling Plan

Time (hr)	Events and Actions	EOP (E-2)
	Reactor Trip after LOCA, Core Quenched by Safety Injection, Auxiliary Feedwater (AFW) automatically actuated.	
1>	Plant Cooldown Using Steam Generators Use Steam Bypass System if Offsite Power Available Use Steam Dump System if Offsite Power Not Available	Step-20, 21
1~3	Vent of Isolation of Safety Injection Tanks (SIT)	Step-41
1~4	Plant Cooldown Using Pressurizer Auxiliary Spray	Step-24, 25
2~3	Alignment of High Pressure Safety Injection to Hot Leg and Cold Leg	Step-43
9~10	Check RCS Pressure Greater than 550 psia, and	
	- Maintain Hot/Cold Injection when RCS Pressure Less Than 550 psia	Step-44.2
	- Continue Cooldown to SCS Entry Condition Using SG when RCS Pressure Greater Than 550 psia	Step-20, 21
	Confirm Establishment of SCS Entry Condition Align Cold Leg Injection, Actuate SCS if Confirmed	Step-44 ¡ S-06

Based on the review, the analysis scope was defined as follows:

1) The timing defined in this table may vary depending on the break size. Therefore, at least, two extreme break sizes in small break LOCA (cold leg breaks of 0.02 ft^2 and 0.5 ft^2 break areas) should be investigated to determine the effectiveness of the overall LTC plan.
2) There were no specific conditions to activate SG cooldown and to initiate a simultaneous hot/cold injection in LTC plan, thus, one hour and two hours after LOCA should be selected as timing to SG cooldown initiation and hot leg injection, respectively, regardless of the plant condition.
3) The pressurizer auxiliary spray was not credited in this analysis, since it was not a safety-graded

component. The SIT isolation was also not considered, since there was no criteria to take the action.

4) Entry condition to SCS was defined as hot leg temperature of 400¢ μand RCS pressure of 410 psia in FSAR. For RCS inventory, 15 % of pressurizer water level was proposed in EOP. In this analysis however, the complete refill of hot leg is regarded as a SCS entry condition for phenomenological concern

III. Analysis Method

The RELAP5/MOD3 code has been developed as one of the best estimate system thermal-hydraulic analysis code, its applicability to small break LOCAs and various transients was systematically verified for various experimental data [7]. The code can be applied to the LTC analysis, since the thermal-hydraulic phenomena during LTC period is expected to be similar to those in small break LOCA. The RELAP5/MOD3.2.2, as a recent version of RELAP5/MOD3, has some improved features including Courant time limit based on junction velocity; time step control; flow anormalies reduction; mass error reduction, etc. Those features may be expected to enhance the calculation accuracy. In addition, the current RELAP5 has a boron transport model based on the first-order Gudnov scheme, which was partially verified during the developmental assessment for LOFT L6-6 boron dilution experiment [8]. A RELAP5/MOD3.2 calculation was performed and compared to investigate the improvements.

The once-through calculation from LOCA to LTC was adopted in the present method, which was different from the typical method [3] separating LOCA and LTC. Since this type approach takes the same nodalization for both LOCA and LTC, a huge amount of computational time would be required. Thus, a simplified model was attempted to investigate the increase the computational efficiency. Figure 1 shows a detailed plant model for UCN Units 3/4. The model consisted of 191 hydrodynamic volumes, 218 junctions, and 212 heat structures. The ECCS, SG Auxiliary Feedwater System, Steam Dump System, etc. were modeled. In the simplified model, the number of volumes in reactor vessel and RCS were significantly reduced , e.g., 24 volumes to 12 volumes in core, while SGs and their secondary sides remained unchanged. Table 1 shows a comparison of modeling and the resultant run statistics.

Table 2 Comparison of Modeling and Run Statistics

Model	No. of Vol.	No. of Junction	No. of H.S	Real Time (sec)*	CPU Time (sec)	Time Advance	Mass Error (lbm)**
Detailed	191	218	212	7288.44	40483.1	224250	16.287
Simplified	142	164	157	7250.11	61210.0	511096	-404.75

*Note : * Preliminary calculation on 0.1 ft^2 Break*

*** Total mass of the system = 1040580 lbm*

For the present calculation, the break was opened on 0 second at the broken loop cold leg. The moderator temperature coefficient (MTC) feedback was modeled with a conservative MTC curve at the begin of life (BOL) core of UCN Units 3/4. Reactor trip was assumed to occur at 1555 psia. Loss of offsite power was assumed to occur coincident with break and not to recover throughout the transient. The turbine trip following the reactor trip was assumed with 3 seconds delay. The safety injection was assumed to initiate at 1555 psia with time delay of 50 seconds. During the SG cooldown phase, the atmospheric dump valves (ADV) were modeled to cooldown the RCS to 550¢ µwithin the limit of 100¢ µ/hr according to EOP. The AFW was modeled to maintain the SG inventory at 23.5 % wide range water level as a minimum. The worst single failure, i.e., one diegel generator failure was applied to the ECCS. For the simultaneous injection, the injected water was assumed to distribute evenly to the hot legs and cold legs. The main steam safety valves (MSSV) were also modeled to consider the case with steam pressure increase for the small break.

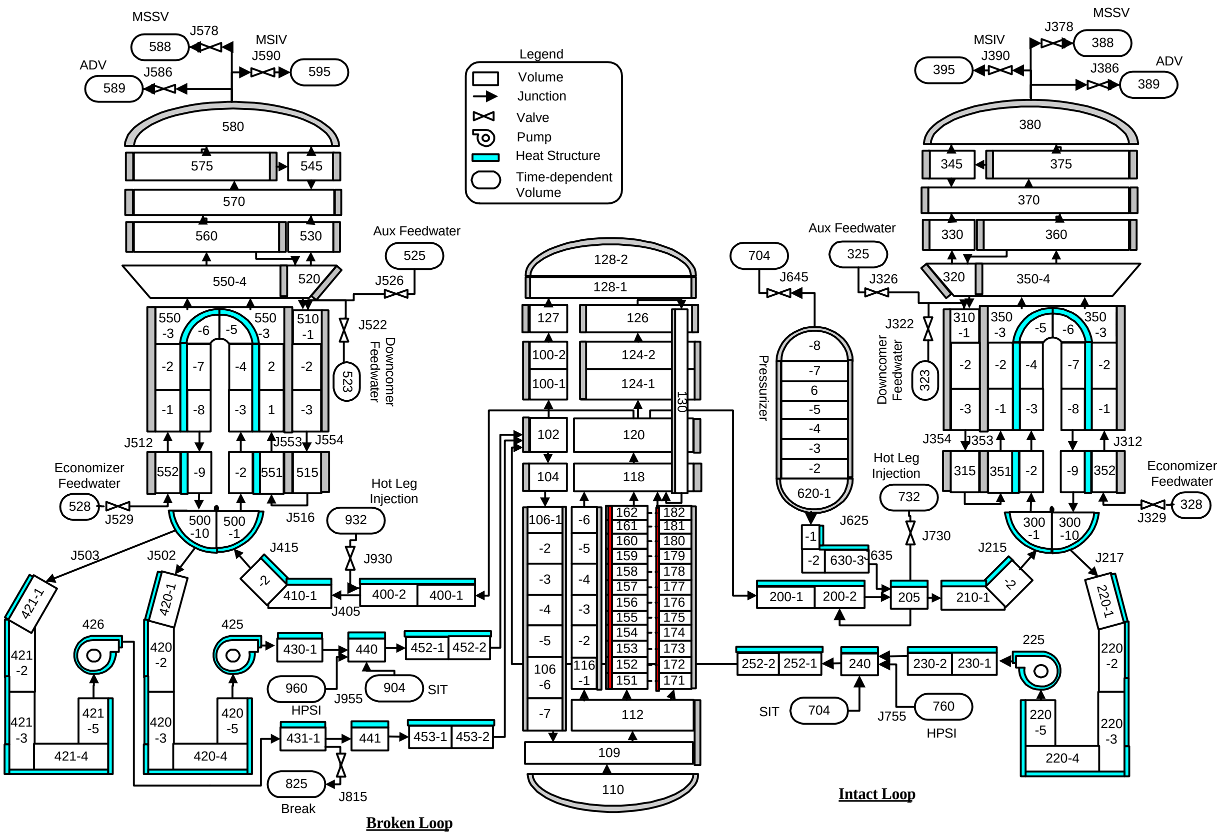

Fig. 1 RELAP5 Model for Post-LOCA Long Term Cooling Analysis of UCN Units 3 & 4

IV. Result and Discussion

Three cases of break size (0.02 ft^2, 0.1 ft^2, and 0.5 ft^2) were calculated with the modeling described above using the RELAP5/MOD3.2.2 code. The smallest break (0.02 ft^2) was calculated to 28000 second (7.7 hours) to confirm the establishment of SCS entry condition. The largest break (0.5 ft^2) was calculated to 11800 seconds to investigate the boric acid behavior. The medium break (0.1 ft^2) was calculated to 7800 seconds to confirm the consistency of RCS cooldown behavior. To determine the LTC calculational efficiency, additional two calculations were performed: 0.1 ft^2 break with the simple model described above; and 0.5 ft^2 break with RELAP5/MOD3.2 code. The result was not compared with the result from the conservative method, since it was not available from the FSAR

IV.1 RCS Cooldown

Figure 2 shows a comparison of the RCS pressure behavior for three cases of break. The RCS was cooled down to a stable level less than 2.7 MPa (410 psia) before 20000 seconds for all three cases. The figure shows that SG secondary cooldown was initiated at 3600 seconds for the 0.02 ft^2 break case, while not activated for the larger breaks since the RCS pressure was sufficiently lowered by the break discharge. The pressure excursion in the behavior of 0.02 ft^2 break case was due to discontinuous water injection from the SITs.

Figure 3 show a comparison of the hot leg temperature behavior for three cases of break. The temperature behavior was almost similar to that of pressure. For 0.02 ft^2 break case, the hot leg coolant temperature decreased to 465 K (337.5¢), which established the SCS entry condition.

Figure 4 show a comparison of the core cladding temperature behavior for three cases of break. From this figure, it can be shown that the LTC initiated after core quenched and it brought the core temperature into a safe level for the three cases of break.

Figure 5 show a comparison of the hot leg liquid fraction behavior for three cases of break. As shown in the figure, the hot leg was refilled before 20000 seconds in the case of 0.02 ft^2 break. For larger breaks, the hot leg was not completely refilled within each calculation time.

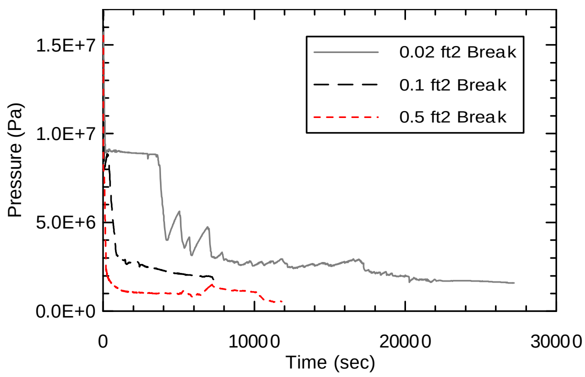

Fig. 2 RCS Pressure

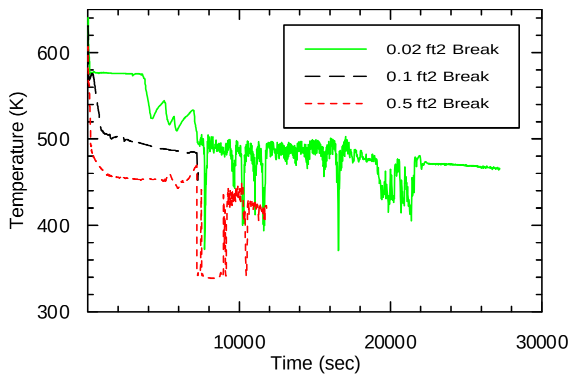

Fig.3 Hot Leg temperature

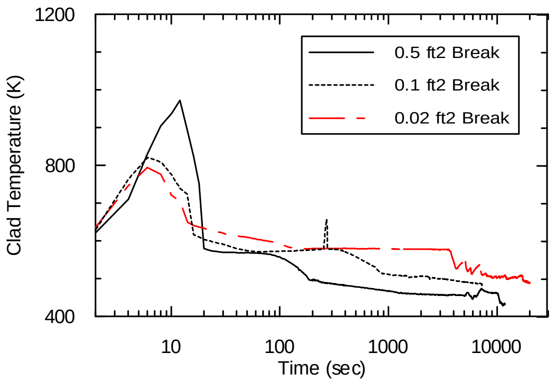

Fig.4 Cladding Temperature

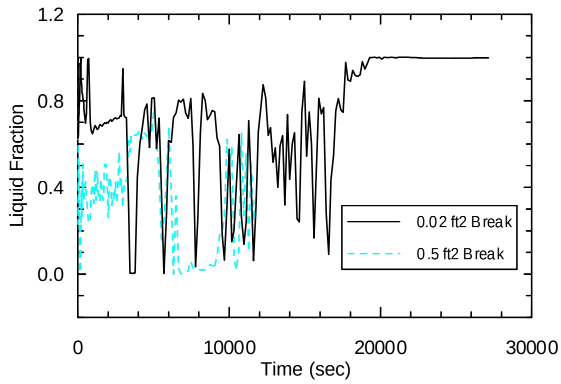

Fig. 5 Hot Leg Liquid Fraction

IV.2 Boron Behavior

Figure 6 shows a comparison of the boron density transient at the core center position for three cases of break. The RCS had an initial boron concentration of 1485 ppm (0.85 wt%). As the cold ECCS water of boric acid (4400 ppm) was injected into the RCS after break, the RCS boron concentration increased. The highest boron concentration might occur at the core center position, since a highly-borated ECCS water (4400 ppm) was transported into the core and steam was discharged out with no boron. The calculated result shows that the boron precipitation limit (29 wt%) was not exceeded for the three cases of break. Especially, for the case of 0.02 ft^2 break, the hot leg refilling before 20000 seconds stabilized the core boron concentration at a sufficiently low level. And it is shown that the boron concentration started to decrease with some oscillations when the simultaneous injection both to hot legs cold legs initiated at 7200 seconds. The figure also shows that the peaks in the case of 0.02 ft^2 break were greater than those of 0.5 ft^2 break. It was due to the discontinuous SIT water injection at a pressure around 4 MPa. It may also due to the fact that the boron was transported from the SIT into the core without any diffusion, which was one of the deficiencies of the current RELAP5/MOD3 code. Therefore, the realistic boron concentration of the core could be lower than the calculated values.

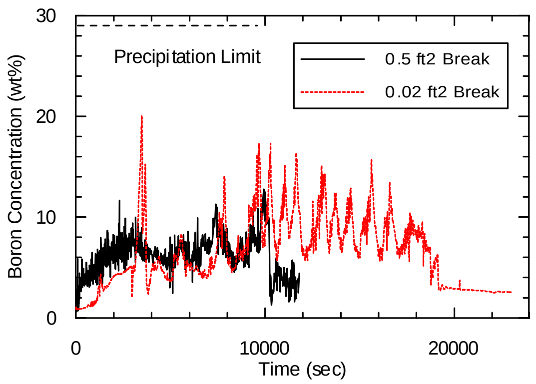

Fig.6 Boron Concentration

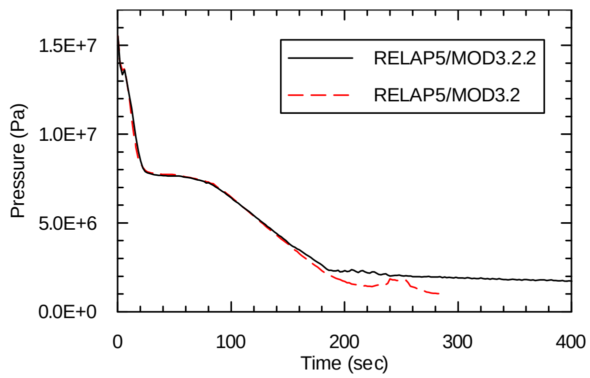

Fig. 7 Comparison Between Codes

IV.3 LTC Calculation Efficiency

Figure 7 shows a comparison of the RCS pressure calculated by RELAP5/MOD3.2.2 and by RELAP5/MOD3.2 calculation for the 0.5 ft^2 break. Two calculation results show an identical behavior up to 120 seconds, after then show a small difference, corresponded to SIT injection. Especially, the RELAP5/MOD3.2 calculation failed before 300 seconds by severe water property variation, while the RELAP5/MOD3.2.2 calculation continued successfully. It, therefore, can be stated that the RELAP5/MOD3.2.2 code can be effectively used in this kind of long-term transient.

Figure 8 shows a comparison of the RCS pressure calculated by the detailed model and one by the simplified model for the 0.1 ft^2 break. Two calculation results show an almost similar behavior although there were some difference in trend. Figure 9 shows comparisons of the CPU times and time steps in both calculations. This figure indicated that the simplified model calculation required more CPU time than the detailed model calculation, which due to a smaller time step size as shown in the Table 2. Small time steps in simple model calculation can be explained by the fact that the Courant time limit was significantly reduced by increasing the volume length. From those comparisons, it can be concluded that the input model simplification may not have an advantage in calculational efficiency within the scope of this study.

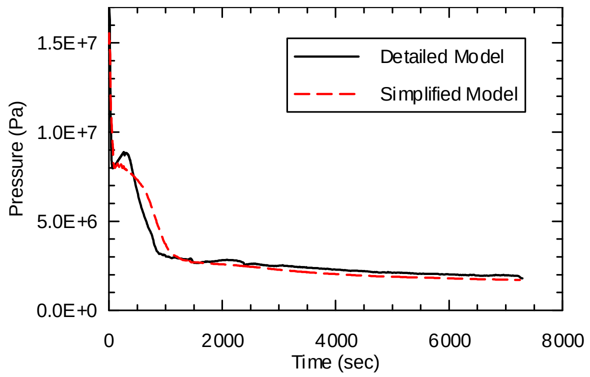

Fig. 8 Comparison Between Modelings

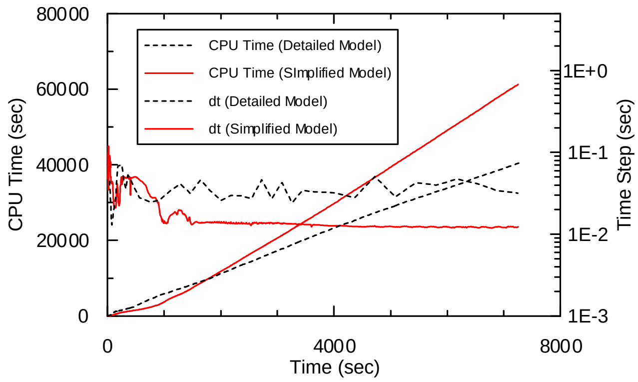

Fig.9 CPU Times and Time Steps

V. Conclusions

A realistic long-term calculation to be used in the post-LOCA LTC analysis was described in this study, which was required to resolve the post-LOCA LTC issues including the concern on boric acid precipitation in the reactor core. The analysis scope was defined according to the LTC plan of UCN Units 3/4 and the plant calculation models were developed suitable to the LTC procedure. The LTC sequences following small break LOCA were calculatied using RELAP5/MOD3.2.2. From the present study, the following conclusions are obtained.

1) The RCS cooldown behavior during LTC sequence was reasonably predicted by the current modeling scheme and the RELAP5 code. The SCS entry condition was established at about 20000 seconds for the 0.02 ft^2 break, which was a little earlier than 7 hours presented at the FSAR
2) The simulatneous injection to hot/cold legs was effective in decreasing the boron concentration in the core to a sufficiently low level. The maximum boron concentration predicted by the calculation was less than the precipitation limit for the range of 0.02 ft^2 to 0.5 ft^2 break.
3) A stable calculation can be achieved by the current RELAP5/MOD3.2.2 when compared to the RELAP5/MOD3.2, which may due to improvements including time step control, etc. And the plant thermal-hydraulic response can be predicted by the input model simplification as similar to that by the detailed input model, however, the calculational efficiency could not be improved in the present LTC calculation.

References

[1] USNRC, Acceptance Criteria for emergency core cooling systems for light water nuclear power reactors, Code of Federal Regulations Part 10 Section 50.46, August 1988.
[2] KEPCO, Final Safety Analysis Report, Ulchin Units 3 & 4, September 1994.
[3] Combustion Engineering Inc., Post-LOCA Long Term Cooling Evaluation Model, CENPD-254-P-A, June 1980 (Proprietary).
[4] USNRC, Generic Letter 77-09-27, Emergency Core Cooling System (Calvert Cliffs Docket), September 1977.
[5] The Thermal Hydraulics Group, RELAP5/MOD3 Code Manual, RELAP5/MOD3.2.2Beta, Formerly NUREG/CR-5535, Scientech Inc., March 1998.
[6] KEPCO, Loss of Coolant Accident, Emergency Operation Procedure, E-01, June 1994.

[7] Richard R. Schultz, International Code Assessments and Applications Program: Summary of Code Assessment Studies Concerning RELAP5/MOD2, RELAP5/MOD3, and TRAC-B, NUREG/IA-0088, September 1992.

[8] Gary W. Johnsen, RELAP5 Development Status, Presented at the Sixth CAMP Meeting, VTT, Finland, May 1995.