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Parametric Study on Thermal-Hydraulic Response Following an ATWS Event

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Abstract

A series of sensitivity calculations for the LOFT L9-3 experiment were performed using RELAP5/MOD3 code to assess parametric effects on thermal-hydraulic response in the event of Anticipated Transient Without Scram (ATWS). The base case calculation was made by the condition which gave a good agreement for the pressure of the reactor coolant system (RCS) with the experimental data. Four parameters of PORV/spray energy loss coefficient, steam generator nodalization and moderator density coefficient (MDC) were selected during the input preparation and investigated by calculating the total discharged energy through relief valves. The energy loss coefficient of the pressurizer spray valve has a significant effect on the behavior of the RCS pressure and the change of the MDC curve within 15 % at the negative region decreased the difference of the coolant temperature between the experiment and the calculation within a range of measurement uncertainty. The finer steam generator nodalization increased the primary to secondary heat transfer rate.

1. Introduction

Anticipated operating transients during which the reactor does not scram as designed can be occurred by multiple failures. The rapid excursion of the RCS pressure and temperature by loss of feedwater and no scram could result in damaging of the reactor core, so the analysis of thermal-hydraulic response in the Anticipated Transient Without Scram (ATWS) event is needed. The major concern of the ATWS analysis is to conform that the ATWS system response is within the maximum pressure of the RCS less than the ASME Service Class C (3200 psia), which has been considered as a bound during the deliberation leading to the final ATWS rule [1].

To resolve this concern, a thermal-hydraulic analysis code to be applied to the response following the ATWS event should be verified for relevant experiment simulating the ATWS event. Experiment L9-3 is one of the anticipated transient performed at the Loss-of-Fluid Test (LOFT) facility and simulated an ATWS induced by a loss of feedwater. The experiment was initiated by turning off the secondary coolant system main feedwater pump. The failures that followed were the absence of both steam generator auxiliary feedwater injection and reactor scram. The steam generator steam control valve was closed manually at 67.3 ± 1.0 s. The pressurizer spray valve opened at 29.5 ± 2.0 s, the pilot operated relief valves (PORV) opened at 73.8 ± 0.2 s, and the safety relief valves (SRV) opened at 96.8 ± 0.2 s. The maximum primary coolant system pressure occurred at 17.4 MPa, and the SRV could prevent the further pressure increase as designed [3].

The analysis of the LOFT L9-3 experiment was conducted by several researchers with the RELAP5 code. It was reported that the code could reasonably predict the RCS thermal-hydraulic response, the reactor power response and the secondary system response during the experiment. However, the SRV was opened more than two times by the over-prediction of the RCS pressure whereas it was opened only one time in the experiment. In addition, it was also noticed that the reactor power and the coolant temperatures at hot and cold legs were over-predicted. So the further sensitivity studies should be needed on the effect of steam generator modeling, the PORV/SRV discharge modeling and the MDC feedback on the system response [10].

The present study is purposed to suggest major modeling scheme for future application to PWR plant analysis, as well as to understand the parametric effect on the thermal-hydraulic response following the ATWS. For those purposes, an effort to improve the predicted system response following the experiment was attempted starting from the previous result [10] as a base line. Assessment calculations were performed varying parameters such as PORV/spray energy loss coefficient, steam generator nodalization and MDC, which were selected from the input model improvement to seek a better agreement with experimental data. And the effect of those parameters was evaluated in terms of the total discharged energy through relief valves, which was believed to provide the insight on the effect in real plant ATWS mitigation.

2. Review of Sensitivity Parameter

As mentioned previously, the study of bang, et al.[10] showed a reasonable prediction on L9-3 transient progression, however, an over-prediction of the reactor power and the system pressure and temperature was identified as a weakness in the RELAP5 assessment. The weakness mainly dues to the input model simulating the test not to the thermal-hydraulic models in the code. Therefore, improvement of the input model from the reference [10] was attempted through extensive sensitivity study on the several parameters. As a result, a base case was selected which gave a good agreement for the RCS pressure, temperature and reactor power with the experimental data within a range of measurement uncertainty.

In the base case calculation, the slope of the MDC curve was varied within 15 % at the negative region and the larger slope variation did not correct the calculation of high primary coolant temperatures after the Main Steam Control Valve (MSCV) closed. The energy loss coefficient of PORV and pressurizer spray valve was set to 10.0, and the steam generator tubes were divided into 12 volumes.

Sensitivity calculations were performed on parameters of PORV/spray energy loss coefficient, steam generator nodalization and MDC. The energy loss coefficient of PORV and pressurizer spray valve was varied to 0.0 instead of 10.0, the steam generator tubes were divided into 34 instead of 12, and the MDC curve was selected as Shape 2 in the Fig. 1.

To assess the comparative effect of selected sensitivity parameters, the total discharged energy through relief valves was calculated as the equation (1). The discharged energy through relief valves is important factor in mitigating the RCS pressure. In Fig. 2, the area means the total discharged energy and it was calculated from the time of the PORV open to 200 seconds. Because the most important thermal-hydraulic phenomenon, i.e., SG secondary side dryout, single- and two-phase coolant discharge and moderator temperature feedback to reactor power were observed before 200 seconds. The differences between calculation and experiment like the equation (2) in each case were compared to identify the most dominant parameter to concern in the analysis of the ATWS at real plant.

$$E_{tot} = \int_{t_{PORV\ OPEN}}^{200} \dot{m}h\,dt \qquad (1)$$

$$dE = E_{Cal} - E_{Exp} \qquad (2)$$

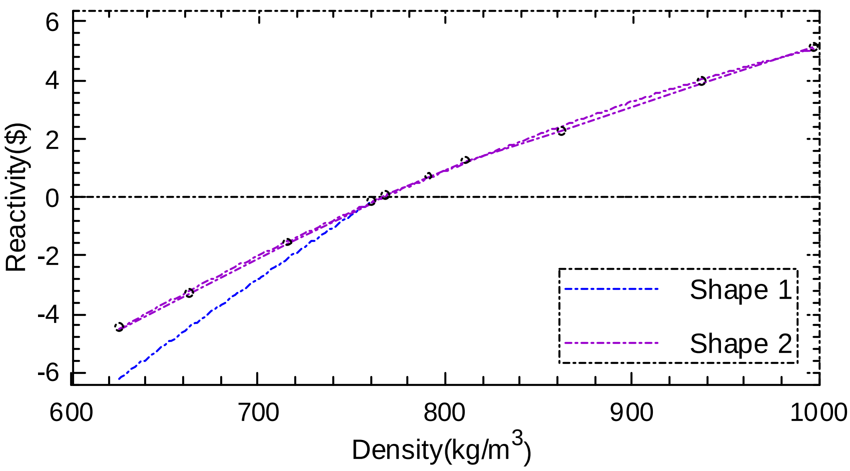

Fig. 1 Moderator Density Coefficient

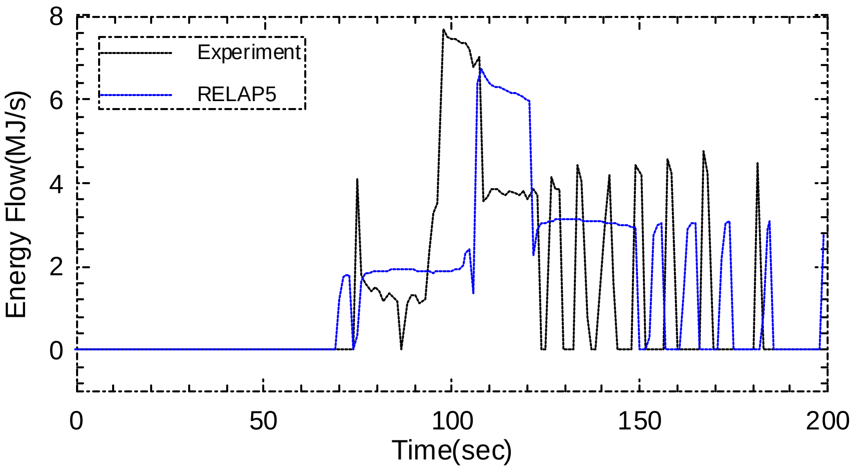

Fig. 2 Energy Flows through Relief Valves

Figure 2 shows the energy flows through the PORV and SRV in cases of experiment and base. In the base case, the PORV was opened earlier than the experiment and closed late. So the total discharged energy is calculated larger than that of the experiment. The total discharged energy of other four cases is calculated like the same method and the value was larger than that of the experiment in the range of 9.49 ~ 71.68 %.

3. Results and Discussion

The most recent version of RELAP5/MOD3 code, version 3.2.2gamma, was used in the present analysis and the geometric data was based on the LOFT input dataset reference document for RELAP5 validation studies [8]. The initial condition for the transient calculation was compared with the experimental data in Table 1. As shown in the table, all the important parameters were well agreed to those of the measured within a range of measurement uncertainty. The transient calculations started from the steady state run were performed by classifying five cases as shown in Table 2.

Table 1 Initial Condition for Experiment L9-3

Parameter	Measured	Calculated	
		Base Case	Case C
Primary Coolant System			
Mass flow rate(kg/s)	467.6±2.7	467.6	467.6
Hot leg pressure(MPa)	14.98±0.06	14.97	14.97
Core ?T(K)	19.4±2.2	19.36	19.36
Intact loop average temperature (K)	566.7±1.5	567.4	567.01
Cold leg temperature (K)	557.0±1.5	557.72	557.33
Hot leg temperature (K)	576.4±1.6	577.08	576.7
Reactor Vessel			
Power level (MWt)	48.7±1.2	48.7	48.7
Maximum linear heat generation rate (kW/m)	51.6±3.9	51.6	51.6
Pressurizer			
Liquid temperature(K)	615.2±0.3	611.25	611.0
Pressure (MPa)	14.98±0.06	14.98	14.98
Liquid level (m)	1.00±0.03	0.9815	0.9863
Steam Generator Secondary Side			
Liquid level (m)	3.15±0.09	3.149	3.155
Liquid temperature(K)	544.4±0.7	544.07	544.08
Pressure (MPa)	5.61±0.06	5.584	5.584
Mass flow rate(kg/s)	25.7±1.1	25.85	25.85

Table 2 Summary of Parametric Study and Discharged Energy

	PORV Loss Coef.	Spray Loss Coef.	S/G Node #	MDC Curve	E_{Cal} (MJ)	dE (MJ)
Base Case	10.0	10.0	12	Shape 1	277.05	28.288
Case A	0	10.0	12	Shape 1	277.06	28.299
Case B	10.0	0	12	Shape 1	276.35	27.596
Case C	10.0	10.0	34	Shape 1	272.37	23.615
Case D	10.0	10.0	12	Shape 2	427.05	178.294

3.1 Base Case Calculation

The base case calculation was newly selected to improve the defects of previous works after conducting sensitivity studies on PORV/spray energy loss coefficient, steam generator nodalization and MDC.

Figure 3 shows the comparison of RCS pressure between two calculations of the previous work[10] and the present revised result and the experimental data. In the previous work, the SRV was opened four times and the maximum pressure increased to 17.58 MPa. However, in the revised result, the SRV was opened one time, which was the same as the experiment and the maximum pressure was 17.29 MPa which was similar to the experiment of 17.31 MPa. The total discharged energy through the relief valves in the base case was larger by 11.37 % than that of the experiment.

Figure 4 shows the comparison of coolant temperatures at hot and cold legs between the previous work and the revised result. In the previous work, the temperature was over-predicted after the MSCV closed. However, in the revised result, the temperature is well agreed with the experiment within the measurement uncertainty (±4.3K). This is because the negative feedback effect of the MDC as the coolant temperature increase was appropriately modeled.

Such an MDC feedback effect is well described in Fig. 5. The reactor power of the previous work was over-predicted in reference [10] whereas the revised result approaches the experimental curve at the lower values.

3.2 Parametric Effects

The parametric study and discharged energy are summarized in Table 2. The effects of each parameters on the thermal-hydraulic response in ATWS are followings.

PORV Energy Loss Coefficient (Case A)

There were no outstanding effects in the RCS pressure, temperature and reactor power when the energy loss coefficient of the PORV was set to 0.0. Only the PORV cycle was shortened by the reason that the pressure drops sharply after the PORV open and reaches the closing set point fast.

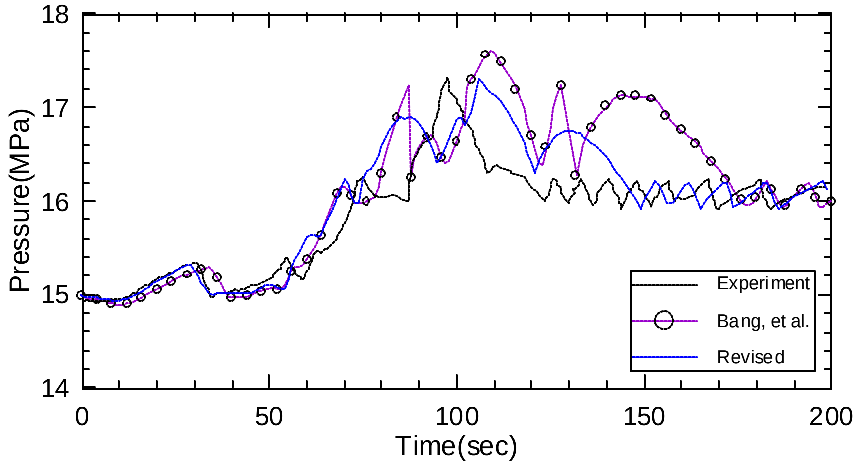

Fig. 3 Comparison of RCS Pressure

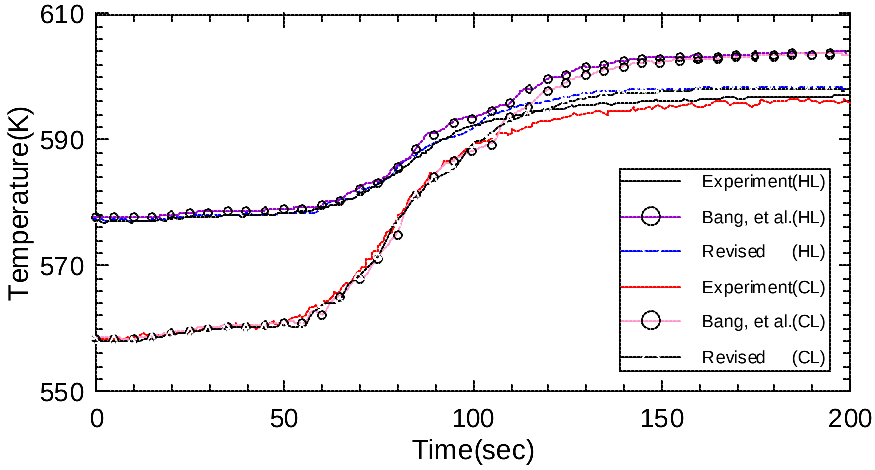

Fig. 4 Comparison of Coolant Temperatures at Hot and Cold Legs

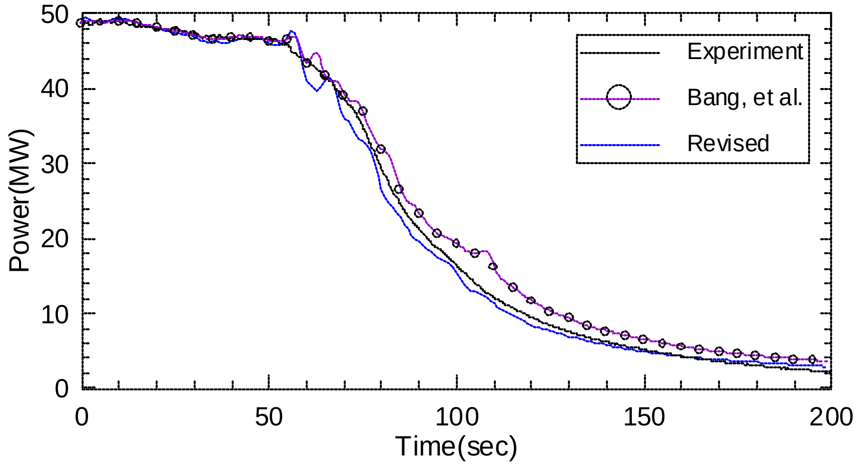

Fig. 5 Comparison of Reactor Power

Spray Energy Loss Coefficient (Case B)

The energy loss coefficient of the pressurizer spray valve has a significant effect on the RCS pressure as shown in Fig. 6. The behavior of the RCS pressure is well agreed with the experiment until the PORV open when the energy loss coefficient is set to 0.0. This is because the RCS pressure was controlled by the large injection of the coolant into the pressurizer. Also, the pressure shows slow increment after the PORV open which is resulted in the decrease of the energy discharge through the PORV. The total discharged energy through the relief valves in the Case B was larger by 11.09 % than that of the experiment.

Steam Generator Node Number (Case C)

In case of the steam generator tubes were divided into 34 volumes, the coolant temperature increased slowly at the nearest to the experimental curve as shown in Fig. 7. This means that the heat transfer rate increased from the primary to the secondary. The pressure was sharply increased after the closing of the MSCV because the primary to secondary heat transfer rate

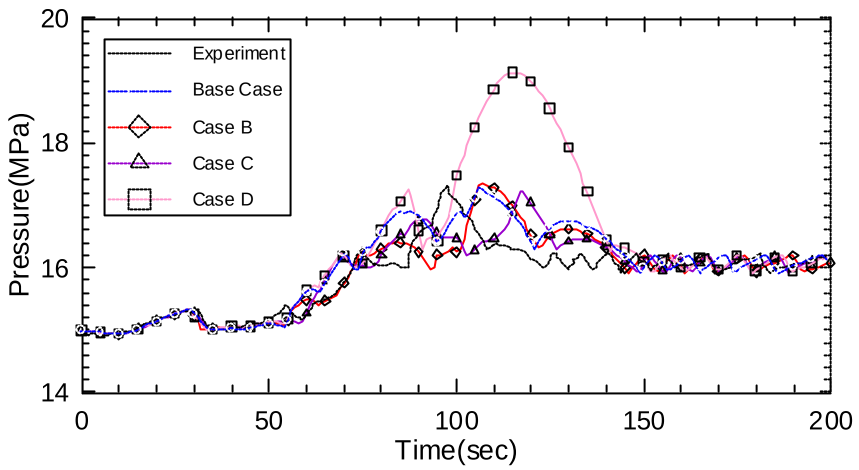

Fig. 6 Parametric Effect on RCS Pressure

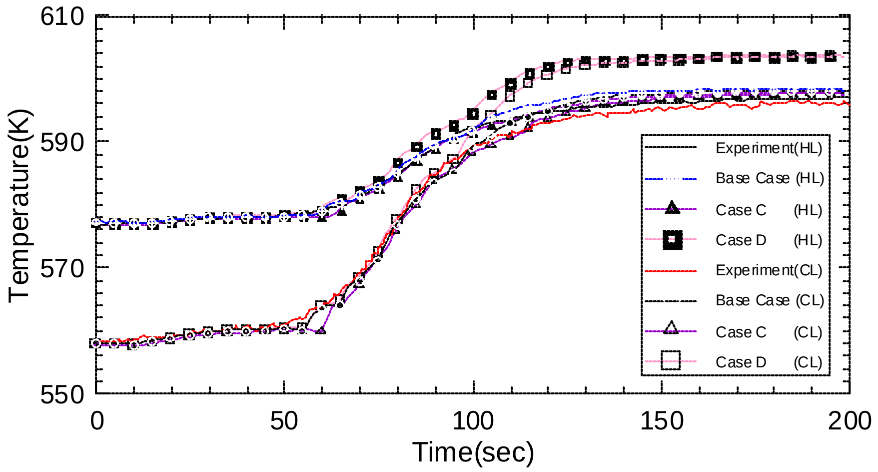

Fig. 7 Parametric Effect on Coolant Temperatures at Hot and Cold Legs

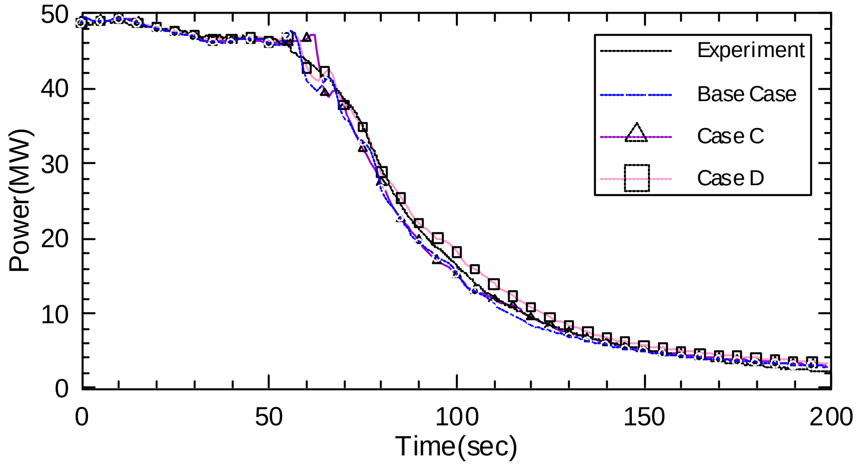

Fig. 8 Parametric Effect on Reactor Power

decreased. The reactor power shows abrupt drop at 60 seconds when the coolant temperature rise rapidly. The total discharged energy through the relief valves in the Case C was larger by 9.49 % than that of the experiment and it is the most approaching case to the experiment among the investigated .

Moderator Density Coefficient (Case D)

The pressure increased unreasonably when the MDC curve was chosen as shape 2 in the Fig. 1 as shown in Fig. 6. The negative feedback effect of MDC was not appropriately modeled though the coolant temperature rising as shown in Fig. 7, so the reactor power was over-predicted as described in Fig. 8. The total discharged energy through the relief valves in the Case D was larger by 71.68 % than that of the experiment. Therefore, the most care should be taken to choosing the accurate input MDC data in the analysis of the ATWS.

4. Concluding Remarks

A parametric study on the thermal-hydraulic response following an ATWS event was performed using RELAP5/MOD3 3.2.2gamma code. The effects of four parameters such as PORV/spray energy loss coefficient, steam generator nodalization and MDC were investigated. Main observations and conclusions are as follows:

1) The energy loss coefficient of the pressurizer spray valve had a significant effect on the behavior of the RCS pressure and the finer steam generator nodalization increased the primary to secondary heat transfer rate. The change of the MDC curve within 15 % at the negative region decreased the difference of the coolant temperature between the experiment and the calculation.
2) It is important to use an accurate MDC data in the analysis of an ATWS and this sensitivity study will provide useful information for the analysis of an ATWS in the real plant. Also the energy loss coefficient of PORV/spray should be carefully determined and its effect on the system response should be examined.

References

[1] USNRC, *Final Safety Evaluation Report Related to the Certification of the System 80+ Design*, NUREG-1462, August 1994

[2] A. Taylor, et al., *Best Estimate Prediction for LOFT Nuclear Experiment L9-1/L3-3*, EGG-LOFT-5413, April 1981

[3] P. D. Bayless, et al., *Experiment Data Report for LOFT Anticipated Transient Without Scram Experiment L9-3*, NUREG/CR-2717, May 1982

[4] J. P. Adams, *Experiment Analysis and Summary Report for LOFT ATWS Experiment L9-3 and L9-4*, NUREG/CR-3417 September 1983

[5] Charles L. Nalezny, *Summary of the Nuclear Regulatory Commission's LOFT Program Research Findings*, NUREG/CR-3005, April 1985

[6] J. Eriksson, *Assessment of RELAP5/MOD2 Cycle 36.04 Against LOFT Small Break Experiment L3-5*, NUREG/IA-0037, March 1992

[7] J. C. Birchley, *RELAP5/MOD2 Analysis of LOFT Experiment L9-3*, NUREG/IA-0058, April 1992

[8] J. C. Birchley, *LOFT Input Dataset Reference Document for RELAP5 Validation Studies*, NUREG/IA-0072, April 1992

[9] Young. S. Bang, et. al., *Assessment of RELAP5/MOD3 With the LOFT L9-1/L3-3 Experiment Simulating an Anticipated Transient With Multiple Failures*, NUREG/IA-0114, Feb. 1994

[10] Young. S. Bang, et al., *Prediction of Thermal-hydraulic System Response following ATWS Event*, Proceedings of KNS Meeting , Seoul, Korea, October 1999

[11] Young. S. Bang, et al., *Development of Regulatory Technology for Thermal-Hydraulic Safety Analysis*, KINS/RR-012, April 2000

[12] The Thermal Hydraulics Group, *RELAP5/MOD3 Code Manual*, Formerly NUREG/CR-5535, March 1998

[13] D. Reeder, *LOFT System and Test Description*, NUREG/CR-0247, July 1978